# DRAFT

## Causes and Significance of Design-Basis Issues at U.S. Nuclear Power Plants

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**ABSTRACT**

This report documents the results of a systematic and comprehensive study of design-basis issue trends and patterns following a limited review that began in early 1997 by the former Office for Analysis and Evaluation of Operational Data*. The study provides insights from reported design-basis issues with respect to: (1) their causes, significant patterns within both the power reactor industry and power reactor systems, frequency trends, safety consequences, and risk significance; (2) the lessons that may be useful in assessing regulatory effectiveness of NRC's evolving inspection and plant performance assessment processes and the definition of plant design basis and; (3) regulatory burden implications related to NRC licensee event reporting requirements for design-basis issues. It is intended that the insights from this study assist NRC and industry ongoing efforts to make NRC's regulatory framework and oversight process more risk informed and performance based and to reduce unnecessary regulatory burden.

* Effective March 28, 1999, the Office for Analysis and Evaluation of Operational Data (AEOD) was disbanded. The work described in this report which was initiated by AEOD is being completed by the Regulatory Effectiveness Assessment and Human Factors Branch of the NRC's Office of Nuclear Regulatory Research.

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## ABBREVIATIONS

| | |
|---|---|
| AEOD | Analysis and Evaluation of Operational Data, Office for (NRC) |
| ASP | Accident Sequence Precursor |
| BWR | boiling water reactor |
| CCDP | conditional core damage probability |
| CFR | *Code of Federal Regulations* |
| DBI | design-basis issue |
| ECCS | emergency core cooling system |
| EDSFI | Electrical Distribution System Functional Inspection |
| GL | generic letter |
| IN | information notice |
| LER | licensee event report |
| LOCA | loss-of-coolant accident |
| NRC | Nuclear Regulatory Commission, U.S. |
| NUMARC | Nuclear Management and Resources Council (now the Nuclear Energy Institute) |
| NUREG | NRC technical report designation (Nuclear Regulatory Commission) |
| ORNL | Oak Ridge National Laboratory |
| PIM | Plant Issues Matrix |
| PRA | probability risk assessment |
| PWR | pressurized water reactor |
| RHR | residual heat removal |
| RIM | risk information matrix |
| SCSS | Sequence Coding and Search System |
| SDP | Significance Determination Process |
| SSC | structures, systems, and components |
| SSEI | Safety System Engineering Inspection |
| SWSOPI | service water system operational performance inspection |
| TS | technical specification |

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## EXECUTIVE SUMMARY

For a number of years the Nuclear Regulatory Commission has been concerned about the continuing discovery and reporting of design-basis issues (DBIs) at nuclear power plants. These concerns were heightened in 1995 when design issues emerged at Millstone and at other nuclear power plant facilities, raising concerns regarding the ability of licensees to operate their facilities within their design basis. In January 1997, the Office for Analysis and Evaluation of Operational Data (AEOD)* was requested to assess and periodically report on the trends and patterns of DBIs identified by nuclear power plant licensees in event notifications and licensee event reports (LERs).

This report documents the results of a systematic and comprehensive study of DBI trends and patterns following a limited-scope AEOD review that began in early 1997. The goal of the study was to develop and document insights from reported DBIs with respect to: (1) their causes, significant patterns within both the power reactor industry and power reactor systems, frequency trends, safety consequences, and risk significance; (2) the regulatory effectiveness of NRC inspection and plant performance assessment processes and the definition of plant design basis applicable at the time the DBIs were reported in LERs and; (3) regulatory burden implications related to current NRC licensee event reporting requirements for DBIs. It is intended that the insights from this report assist NRC and industry ongoing efforts to make NRC's regulatory framework and oversight process more risk-informed and performance-based and to reduce unnecessary regulatory burden.

For this study, a DBI was generally defined and captured in accordance with the licensee event reporting requirement in 10 CFR 50.73(a)(2)(ii). This requirement states that licensees shall report an actual or a potential event (condition) that resulted in the nuclear power plant, including its principal safety barriers, being seriously degraded, in an unanalyzed condition, in a condition outside the plant's design basis, or in a condition not covered by the plant's operating or emergency procedures. The initial data for the study came from a special database of DBIs reported in LERs for 1997 and compiled by AEOD. All of the LERs for 1997 were reviewed manually to determine if they involved a DBI as documented by the licensee. Although there was agreement with the licensees' determinations in most cases, there were some differences. Automated searches and sorts of the Sequence Coding and Search System (SCSS) LER database, maintained by the Oak Ridge National Laboratory (ORNL), were performed to determine the trends and patterns of DBIs over the period 1985–1997. The Accident Sequence Precursor (ASP) database was used to obtain risk perspectives in terms of conditional core damage probability (CCDP) of the DBI events.

The study found that U.S. nuclear power plants reported over 3100 LERs with DBIs during the period 1985–1997, or on average, about 240 per year. For the period 1985–1987 the number of DBIs ranged from 155–184 per year. For 1988 and 1989, the number significantly increased to 254 and 251, respectively. This increase coincided with the broad implementation, beginning in 1987, of NRC safety systems functional inspections and safety systems outage modification

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inspections. There was another significant increase in the number of DBIs reported in LERs in 1996 (from 194 to 377) and again in 1997 (from 377 to 512). These increases appeared to coincide with certain NRC initiatives including: NRC team inspections with a significant design element, NRC surveys of licensees on DBIs, licensee review in response to elevated NRC focus on DBIs, and NRC generic communications.

For 1997, the study identified 1975 LERs (when considering multi-plant applicability) that were submitted for the 110 nuclear power plants in the United States. As indicated above, of these, 512 involved DBIs, or about 26 percent of the total. For 1997, the most common causes of DBIs were original design error, procedure deficiency and human error. Licensees often cited multiple causes of DBIs. The most frequent contributing causes included design errors dating back to the time of original plant licensing (72 percent), procedure deficiencies (27.5 percent), human error (22 percent), poor work control practices (15 percent), and plant modifications (13 percent).

The study found a significant variation among plants in the number of reported DBIs. For 1997, the average number of DBIs reported in LERs for the 110 operating plants was 4.7. However, 6 PWRs accounted for about 30 percent of the reported DBIs: Crystal River 3 (37 DBIs), Point Beach 1 (27 DBIs), Point Beach 2 (24 DBIs), Millstone 3 (24 DBIs), D.C. Cook 2 (21 DBIs), and D.C. Cook 1 (18 DBIs). Additionally, during the period 1990–1997, 11 plants (9 pressurized water reactors [PWRs] and 2 boiling water reactors [BWRs]) accounted for about 29 percent of the reported DBIs.

Only a few safety-related systems accounted for about half of the DBIs. For 1997, 6 of the 26 safety-related plant system categories used for the study accounted for about half of the 512 DBIs reported in LERs. These systems were: emergency core cooling (13 percent), emergency ac/dc power (11 percent), containment and containment isolation (10 percent), primary reactor (7 percent), auxiliary/emergency feedwater (6 percent), and essential service water (6 percent).

Older plants (those licensed before 1975) generally reported more DBIs than newer plants (licensed after 1984) reported. For 1997, newer plants reported an average of about 3.1 DBIs while older plants reported an average of about 5.6 DBIs.

Of the 512 DBIs reported in LERs for 1997, 410 were screened, characterized and ranked by potential risk significance. The remaining 102 LERs with DBIs which were not risk ranked involved either seismic or fire protection deficiencies. These were excluded because significant uncertainties exist in the current risk assessment methods for these kinds of design issues. The risk category for each DBI was assessed on the basis of the Phase 1 process step documented in SECY-99-007A, “Recommendations for Reactor Oversight Process Improvements,” Appendix A, “Process for Characterizing the Risk Significance of Inspection Findings.” The specific guidance used for characterizing DBI risk significance was the generic risk information matrices (RIMs) tables for PWRs and BWRs documented in “Development of Risk-Informed Baseline Inspection Program,” dated February 10, 1999. DBIs were categorized as either “potentially risk significant,” “minimal risk,” or “no risk significance.” A DBI was categorized as potentially risk significant if it involved structures, systems, and components (SSCs) which were in the RIMs table <u>and</u> it was relevant to one or more of the sequences which placed the SSC in the RIMs table. A DBI was categorized as involving minimal risk if it involved an SSC that was either not in the RIM table or it involved an adverse effect on an SSC which was not relevant to any of the reasons the SSC was in the RIMs table. A DBI was categorized

as having no risk significance if the DBI was only programmatic. That is, it involved inadequate design basis analysis documentation and where the remedial actions only involved correcting or completing the design basis analysis.

Of the 410 DBIs reported in LERs in 1997 that were screened for risk significance, a small fraction (23 percent) were identified as potentially risk significant. The majority (77 percent) were determined to only involve either minimal risk or no risk significance. A sorting of these DBIs by system found that 3 of the 26 safety-related systems accounted for 64 percent of the of the potentially risk significant DBIs. These systems were: emergency core cooling (34 percent) emergency ac power (18 percent) and containment and containment isolation (12 percent).

In general, “older” plants (operating license between 1964 and 1974) reported more potentially risk significant DBIs than “newer’ plants (operating license between 1985 and 1997). About 57 percent of the “older” plants had at least one DBI categorized as potentially risk significant, whereas, about 19 percent of the “newer” plants had at least one DBI categorized as potentially risk significant. This tendency was also more pronounced at multi-unit sites than at single-unit sites. Consequently, “older” multi-unit sites had a higher percentage of potentially risk significant DBIs than did “older” single-unit sites. The apparent reasons for the difference included: generally lower quality, level of completeness and accessibility of plant design basis information at older plants. Shared systems was believed to be the major reason for the higher percentage of potentially risk significant DBIs for multi-unit sites. The percentage of plants associated with multi-unit sites that had at least one potentially risk significant DBI was 63, 50, and 21 for “older,” “medium” (operating license between 1975 and 1984) and “newer” licensed plants respectively, whereas, for single-unit sites that had at least one potentially risk significant DBI, the percent was 50, 33, and 17 for “older,” “medium,” and “newer” licensed plants respectively.

Potentially risk significant DBIs also varied by NRC region. For 1997, plants in Regions I and III reported the largest number (32 and 35 respectively) of potentially risk significant DBIs, while plants in Regions II and IV reported the fewest number of potentially risk significant DBIs (22 and 6 respectively). Region III plants also had the highest percentage of plants with at least one potentially risk significant DBI (59 percent), followed by Region I (52 percent), Region II (36 percent) and Region IV (19 percent). The lower incidence of potentially risk significant DBIs in Regions II and IV may have been due in part to the generally fewer engineering inspection hours and the higher percentage of newer plants (i.e., better design basis documentation) in these regions.

During the period from 1991 to 1997, the percent of DBIs reported in LERs that were ASP events steadily decreased, while the number of DBIs reported in LERs increased. In 1991, about 8.3 percent of DBIs reported in LERs were determined to be ASP events (i.e., CCDP $\geq 10^{-6}$). However, by 1997, only about 0.6 percent of the LERs with DBIs were classified as ASP events. However, 3 of the 5 ASP events in 1997 involved DBIs indicating that DBIs were an important contributor to the relatively few risk significant operating events which occurred. The study also found that during 1992–1997 there were 14 “important” ASP events (CCDP $\geq 10^{-4}$). Of these, 12 occurred at PWRs and 2 occurred at BWRs. However, three of the important ASP events during this period involved DBIs, and all of these occurred at PWRs.

The study also examined apparent correlations of the number of DBIs (total and potentially risk significant) reported in LERs with other NRC program areas and initiatives. These efforts were

intended to explore insights and potential lessons which might be associated with NRC regulatory effectiveness and regulatory burden.

The study found that increases in the number of reported DBIs coincided with NRC initiatives. The number varied from a low of 155 in 1985 to a high of 512 in 1997. Significant increases in the number of reported DBIs from the previous years were observed in 1988 and 1989, and again in 1996 and 1997. The increases appeared to coincide with certain NRC initiatives including: NRC team inspections with a significant design element, NRC surveys of licensees on DBIs, licensee reviews in response to elevated NRC focus on DBIs, and NRC generic communications.

For the period from 1995 to 1997, the number of reported DBIs appeared to correlate with NRC engineering inspection effort. The study found that during this period, as NRC engineering inspection hours at a plant increased, the number of DBIs reported by the plant generally increased. The increase was considered to be the result of both NRC inspection teams finding DBIs at the plant and licensees increasing their efforts to identify DBIs in connection with these NRC inspections. NRC generic communications on DBIs during the period was also considered a factor. Conversely, fourteen of the 21 plants that reported no DBIs during 1997, received less than the median number of engineering inspection hours.

As noted above, if a plant had a thorough engineering inspection for design compliance, it often reported more DBIs. The study also found that this often resulted in the plant receiving a lower plant engineering rating (under the former Systematic Assessment of Licensee Performance (SALP) program) in the subsequent assessment period. In some instances, the lower assessment rating led to increased regional or agency oversight. The correlation between engineering inspection effort, the number of reported DBIs and subsequent performance ratings was evident for the SALP program, and may also be relevant to the NRC's revised reactor oversight program which features decision criteria leading to additional inspection efforts for selected plants that meet an established performance threshold.

Also as noted above, the majority of the DBIs reported in LERs involved minimal or no safety significance. In this regard, it would appear that the staff's ongoing efforts to make 10 CFR 50.73(a)(2)(ii) more risk-informed should have a significant impact on reducing unnecessary regulatory burden.

NRC engineering inspection teams and design inspection teams have been particularly successful in identifying DBIs at nuclear power facilities. When DBIs are identified as part of an NRC inspection they frequently resulted in the licensee submitting an LER for the DBI and the staff documenting the finding in an inspection report. However, based on the ASP program insights, most DBIs, by themselves, have been of relatively low safety significance. It is anticipated that the inspection reports which conforms to the revised reactor oversight program will screen out DBI-related inspection findings that are of low risk significance and include only those that are significant in a risk-informed, performance-based context.

NRC and industry awareness and recognition of significant and potentially generic DBIs have emerged over time from the coalescing of insights that are drawn from operating experience, performance information, safety analyses, and system analyses and reviews. The safety importance and applicability of DBIs documented and fed back to industry in NRC generic communications occasionally has taken several years for some licensees to fully recognize and

address. However, with NRC's generic communications program and reactor inspection program becoming more risk-informed, the timeliness and reliability of licensee corrective actions for applicable risk-significant DBIs in NRC generic communications should be expected to improve.

As a final observation, as evidenced by the ASP program, over the period from 1990 to 1997, there has been a steady decline in the ratio of the number of ASP events with DBIs to the total number DBIs reported in LERs. By 1997, less than 1 percent of all DBIs reported in LERs were ASP events. However, 60 percent (3 of 5) of the ASP events for 1997 involved DBIs. Thus, although the percentage of DBIs that are risk significant is very small, it may be expected that, to the extent that ASP events occur (and risk significant NRC inspection findings are identified), DBIs may continue to be an important contributor.

# 1 INTRODUCTION

## 1.1 Background

In Title 10 of the *Code of Federal Regulations* (CFR) Section 50.2 (10 CFR 50.2), design bases is defined as follows:

> ... information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted "state of the art" practices for achieving functional goals or (2) requirements derived from analysis (based on calculation and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals.

The licensing process for the operating U.S. nuclear power plants required each applicant to submit a preliminary safety analysis report, including the principal design criteria for the facility (10 CFR 50.34). The principal design criteria establish the necessary design, testing, and performance requirements for structures, systems, and components (SSCs) important to safety; that is, the SSCs that give reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. The general design criteria in 10 CFR Part 50, Appendix A, establish minimum requirements for the principal design criteria in general terms. Each plant's design bases are usually derived from these criteria.

The preliminary safety analysis report contains a preliminary analysis and evaluation of the design and performance of SSCs of the facility important to safety. The objective of the analysis and evaluation was to assess the risk to public health and safety resulting from the operation of the facility. This assessment included a determination of (1) the margin of safety during normal operations and transient conditions anticipated during the life of the facility and (2) the adequacy of SSCs provided for the prevention of accidents and the mitigation of the consequences of postulated accidents (10 CFR 50.34). The final analysis and evaluation were included in the final safety analysis report. These analyses and evaluations were usually performed using the deterministic approach and established that the facility could be operated safely within its design bases.

The deterministic method used for the safety analysis could be supplemented by a probabilistic method, to estimate the probability and risk of an accident. However, the probabilistic method alone cannot provide all analyses required by the current regulations.

For the purposes of this study, a "design-basis issue" (DBI) is defined by the licensee event reporting requirements contained in 10 CFR 50.73(a)(2)(ii), which states that licensees shall report:

> "Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or that resulted in the nuclear power plant being:(A) In an unanalyzed condition that significantly compromised plant safety; (B) In a condition that was outside the design basis of the plant; or (C) In a condition not covered by the plant's operating and emergency procedures."

Section 2.1 of this report provides additional discussion of the scope of DBIs considered in this review.

For a number of years the Nuclear Regulatory Commission has been concerned about the continuing identification and reporting of DBIs at nuclear power plants. In the latter years of the 1980s, NRC inspection teams found that some power reactor licensees were not properly maintaining design-basis information for their facilities and were making modifications without having a complete or accurate understanding of their facility's design bases. The NRC staff viewed this situation as having the potential to be a significant contributor to DBIs being either unintentionally introduced or not recognized by nuclear power plant licensees. The NRC took several actions to resolve these concerns. In 1989, the NRC staff surveyed nuclear power plant design control practices and design reconstitution efforts at six nuclear utilities and one nuclear steam system supply vendor (NUREG-1397, "An Assessment of Design Control Practices and Reconstitution Programs in the Nuclear Industry," January 1991 (Ref. 1)). Following this assessment the Commission issued a policy statement, "Availability and Adequacy of Design Bases Information at Nuclear Power Plants," August 1992 (Ref. 2). The NRC staff also published for comment a draft generic letter (GL), "Availability and Adequacy of Design Bases Information" (Ref. 3). In addition, because the NRC's findings heightened the nuclear industry's awareness of the potential safety issues associated with the adequacy and availability of design documentation, the Nuclear Management and Resources Council (NUMARC), now the Nuclear Energy Institute, issued guidelines on the subject in NUMARC 90-12, "Design Basis Program Guidelines," October 1990 (Ref. 4). This document provided guidance to licensees that had participated in the NUMARC voluntary initiative to develop a program for collecting and organizing design-basis data and supporting design information.

In 1995, design issues emerged at Millstone and at other nuclear power plant facilities and raised concerns regarding the licensees' ability to operate and maintain their facilities in accordance with the facility's design basis. Because of the potential scope of these DBIs and the potential generic applicability of these or other design issues, the staff asked licensees to describe their programs and processes for ensuring that they were operating their facilities in accordance with their facility's design bases. With Commission approval, in October 1996 the NRC issued a letter in accordance with 10 CFR 50.54(f) to each nuclear power plant licensee, asking licensees to describe the programs and processes established to control and maintain operations within the facility's design basis. Additionally, licensees were asked to describe the effectiveness of these programs and processes, and any initiatives they had implemented for design-bases documentation. The NRC staff's review of the licensees' responses is described in SECY-97-160, "Staff Review of Licensee Responses to the 10 CFR 50.54(f) Request Regarding the Adequacy and Availability of Design Bases Information," July 24, 1997 (Ref. 5).

On the basis of its review, the staff concluded that although licensees had established programs and processes to maintain their facilities' design basis and that no further generic action was required, the NRC needed to perform some plant-specific followup because of instances in which: (1) a licensee's regulatory performance brought into question the effectiveness of its design control program and processes or (2) the staff determined that there was a need to validate the effectiveness of a particular element of a licensee's program and processes.

In January 1997, the Office for Analysis and Evaluation of Operational Data (AEOD)* was assigned to periodically report on the trends and patterns of DBIs that were being reported by reactor licensees in event notifications in accordance with 10 CFR 50.72, "Immediate notification requirements for operating nuclear power reactors" (Ref. 6), and subsequently in licensee event reports (LERs) for 1997 in accordance with 10 CFR 50.73, "Licensee event report system" (Ref. 7). This formal report is an extension and closure of that effort.

The purpose of this report is to document the results of a systematic and comprehensive analysis and evaluation of the DBI trends and patterns review that began in early 1997. In particular, the goal of this expanded study is to develop and document insights from reported DBIs with respect to: (1) their causes, significant patterns within both the power reactor industry and power reactor systems, frequency trends and their causes, safety consequences and risk significance; (2) the lessons that may be useful in assessing regulatory effectiveness of NRC's evolving inspection and plant performance assessment processes, in the period associated with the data, and the definition of plant design basis and; (2) regulatory burden implications related to NRC licensee event reporting requirements for DBIs. It is intended that the insights from this report would be useful to NRC and industry ongoing efforts to make NRC's regulations framework and oversight process more risk-informed and performance-based and to reduced unnecessary regulatory burden. As of March 2000, the current requirements to report "a condition outside the design basis of the plant," which have been in effect since 1983, are not clearly understood and in some cases result in reporting of events with little or no significance. A rulemaking effort to address these issues has been underway since early 1998.

## 1.2 Scope

For this study, LERs for 1997 were individually reviewed and those that had DBIs identified were entered into a special DBI database. The special database included fields to capture information on events and plant design. LERs with DBIs received a second review to verify the identified DBIs. During the second review, events which involved the failure of equipment, missed surveillance testing, or work control practices that did not involve an actual DBI were eliminated. The LERs for 1997 were analyzed to identify the DBI initiators, bases for trends, and the safety consequences of the DBIs.

Automated searches were also performed using the Sequence Coding and Search System (SCSS) database (Ref. 8) to find the same 1997 DBIs entered in the special DBI database. This search indicated a very good agreement between the two databases. Based on this result, the SCSS database was used to find DBIs for the period 1985–1997. The LERs for this period were used to identify DBI trends and the bases for the trends.

The Accident Sequence Precursor (ASP) database (Ref. 9) was used to obtain risk perspectives in terms of the conditional core damage probability (CCDP) of the DBI events. The NRC's events tracking system database (Ref. 10) was also reviewed to identify DBIs that were reported during 1997. Event data were screened to determine the reportability reasons,

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* Effective March 28, 1999, the Office for Analysis and Evaluation of Operational Data (AEOD) was disbanded. This study was completed by the Regulatory Effectiveness Assessment and Human Factors Branch of the NRC's Office of Nuclear Regulatory Research.

the extent of corrective action0s, whether a forced outage was declared, and the emergency classification, if any.

## 2 DISCUSSION

### 2.1 Event Reporting Requirements and Guidance for Design-Basis Issues

10 CFR 50.73(a)(2)(ii) states that licensees shall report the following:

> Any event or condition that resulted in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded; or that resulted in the nuclear power plant being:
>
> (A) In an unanalyzed condition that significantly compromised plant safety;
>
> (B) In a condition that was outside the design basis of the plant; or
>
> (C) In a condition not covered by the plant's operating and emergency procedures.

These conditions apply whether or not the plant was operating at the time the condition was discovered.

Licensee event report review experience has shown some inconsistency among licensees in interpreting event reporting requirements and definitions, and some inconsistency in reporting thresholds for DBIs (i.e., when a plant is in a condition outside its design basis). In NUREG-1022, Revision 1, "Event Reporting Guidelines 10 CFR 50.72 and 50.73," January 1998 (Ref. 11), examples of DBIs are provided that the NRC staff considers reportable errors in actual design. Examples include the emergency core cooling system design not meeting the single-failure criterion contained in Appendix K to 10 CFR 50; hardware problems such as high-energy line break restraints not being installed; fire protection commitments for safe shutdown not being met; and one train of a required two-train safety system not being capable of performing its design function for an extended period during plant operation. Such deficiencies indicate that the plant may be outside its design basis.

### 2.2 Review of Design-Basis Issues

#### 2.2.1 Design-Basis Issue Determination

All LERs for 1997 were reviewed to determine if they involved a DBI as documented by the licensee. Although there was agreement with the licensees' determinations in most cases, there were some differences.

Although technically they could be considered to place a plant outside its design bases, the following kinds of DBIs in LERs were not included in this DBI study: (1) components that had never been tested or verified to determine that they would, or could, perform in a manner that would support the design bases of the plant, but upon subsequent testing were found to be

acceptable, (2) temporary plant conditions caused by operator error, such as mispositioning of controls or improper procedural requirements, (3) failure to perform required testing within specified time requirements, (4) expiration of environmental qualification of applicable components, (5) certain minor deviations from Appendix R requirements that appeared to have marginal safety significance, (6) certain improper setpoint calculations, and (7) failure to control heavy loads.

### 2.2.2 Licensee Event Reports Classified as Actual or Potential Events

Design-basis issues were classified as either an “actual event” or a “potential event.” An “actual event” is an actual operational occurrence (e.g., an actual failure of a structure, system, or component to operate within design requirements) which takes place while the plant is operating at power or shutdown. A “potential event” exists when it is found that a structure, system, or component might not operate as designed given some set of postulated conditions.

### 2.2.3 Special Design-Basis Issue Database of Licensee Event Reports for 1997

The initial data for this study were derived from a special DBI database established by AEOD. The database contained the following data for each LER having an event date in 1997: (1) whether an LER involved a DBI, (2) whether it involved an “actual event” or a “potential event,” (3) the NRC region responsible for regulatory oversight of the plant, (4) the event date, (5) the docket number(s) of affected plant(s), (6) the nuclear steam supply system vendor, (7) the architect-engineer, (8) the operating license date, (9) the event initiator (activity that resulted in the DBI being reported), (10) the NRC generic communication number (if issued), (11) the plant system(s) involved, and (12) the CCDP if the event was analyzed by the ASP program and was greater than or equal to $1 \times 10^{-6}$.

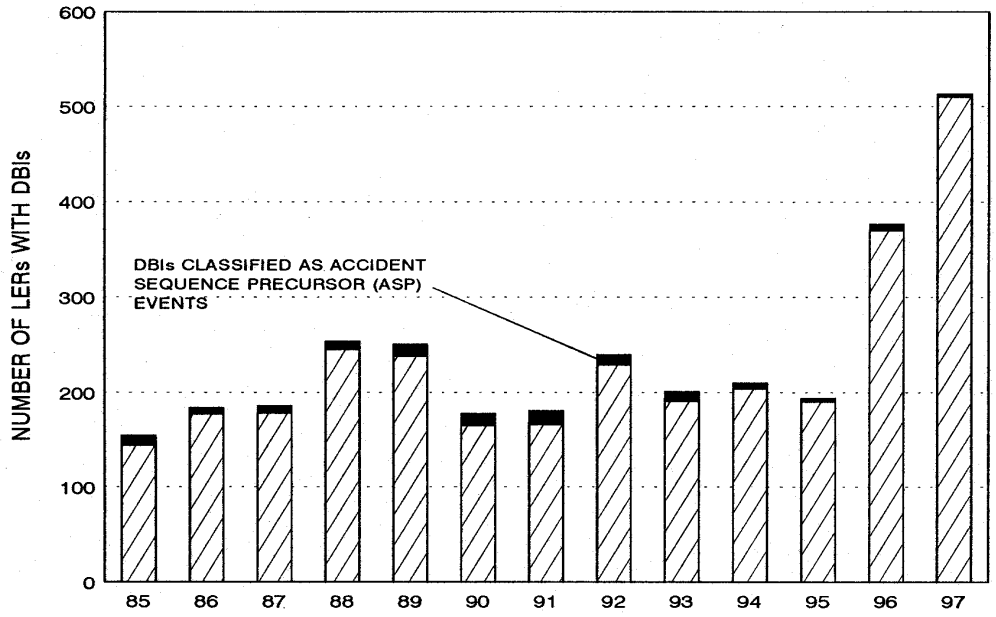
### 2.2.4 NRC Sequence Coding and Search System Database

The SCSS database was searched to identify the LERs that met the “reportability” criterion of 10 CFR 50.73(a)(2)(ii), “unanalyzed condition.” Although LERs identifying DBIs were reported by licensees under several other criteria, the majority of LERs were reported under the above criterion based on comparing the actual LERs with DBIs during 1997 and the corresponding LERs with DBIs found with the above SCSS search criteria. The difference appears due to plant applicability coding. The SCSS database counts an event applicable to multiple units of the same design at the same site, as a single LER with a DBI. To be compatible with this report, the SCSS database sort results were revised to account for multi-plant applicability of the reported DBI. The adjusted results (i.e., DBI count) of these SCSS searches were used to make general observations on the long-term trend of plants with DBIs for the period 1985–1997.

# 3 ANALYSIS OF DESIGN-BASIS ISSUES REPORTED IN LICENSEE EVENT REPORTS

## 3.1 Trends of Plants with Design-Basis Issues for the Period 1985–1997

U.S. operating plants reported over 3100 LERs with DBIs during the period 1985–1997, or about 240 per year. The number of DBIs on a year-to-year basis for 1985–1997 is shown in Figure 1. The portion of DBIs that met the ASP CCDP of at least $1 \times 10^{-6}$ is shown in black for each year. From 1985–1997, LERs have identified increasing numbers of DBIs involving both




trains and systems, however the majority of these DBIs have been of low safety significance from an ASP perspective.

**Figure 1 Trend of design basis issues reported for 1985–1997**

For 1985–1987 the number of DBIs ranged from 155–184 per year. For 1988 and 1989, the number significantly increased to 254 and 251, respectively. This increase coincided with the broad implementation, beginning in 1987, of NRC safety systems functional inspections and safety systems outage modification inspections. For 1985–1995, the number of DBIs varied between 155 and 194. There was a significant increase in DBIs in 1996 (from 194 to 377) and again in 1997 (from 377 to 512). This increase appeared to coincide with certain NRC initiatives including: NRC team inspections with a significant design element, NRC surveys of licensees on DBIs, licensee review in response to elevated NRC focus on DBIs, and NRC generic communications.

During the early 1990s, based on voluntary industry activities, the NRC reduced its general emphasis on design-basis and licensing-basis reviews and design-basis reconstitution activities. However, from 1990 through 1993, the NRC performed 75 electrical distribution system functional inspections (EDSFIs), which led to the identification of 620 electrical system DBIs (Office of Nuclear Reactor Regulation EDSFI database). Similarly, from 1993 to 1995, NRC performed 18 full scope service water system operational performance inspections (SWSOPIs) and 2 reduced scope SWSOPI inspections, and there were 27 reduced scope licensee SWSOPI self-assessments (a total of 47). Based on a comparison of the EDSFI and SWSOPI databases with the LERs reported for this period, there appeared to be significant under-reporting of the EDSFI and SWSOPI DBI findings. Also, the Maine Yankee Independent Safety Assessment Team report (1996) stated that 18 SWSOPI findings were still unresolved, noting that an effective program was not yet in place to ensure that heat exchanger fouling did not exceed the bounding values assumed in calculations. These DBI findings were not reflected in LERs.

The NRC and industry refocused their design-basis and licensing-basis efforts in response to DBIs for 1995–1997 identified in response to GLs and inspection findings at Millstone, Haddam Neck, Maine Yankee, Crystal River, Clinton, D.C. Cook, and other plants.

## 3.2 Analysis of Design-Basis Issues Reported in 1997

For 1997, there were 1975 LERs (when considering multi-plant applicability) for the 110 operating nuclear power plants in the United States. Approximately 26 percent (i.e., 512) involved DBIs. This section provides data and observations on all 512 reported DBIs and the effects of plant age, plant location, system affected, reactor type, discovery mode, reportability reasons, causes, and the extent of corrective actions. Section 6 provides data and observations on a subset (95 DBIs) that were determined to be potentially risk significant. Similar observations can be made about both sets of data (i.e., the total population of DBIs and the subset population that was determined to be potentially risk significant).

### 3.2.1 Design-Basis Issues by Plant Age

To analyze the effects of plant age on the number of DBIs reported by licensees, the 110 operating nuclear power plants in the U.S. were divided into three age groups. Group A represented the “older” plants, group B represented the “medium” age plants, and group C represented the relatively “newest” plants. Group A plants obtained their operating licenses between 1964 and 1974; group B between 1975 and 1984; and group C between 1985 and 1997. Each group spanned about one-third of the 34-year plant operating license period (1964–97). There are 44 plants in group A, 35 plants in group B, and 31 plants in group C.

Figure 2 shows the average number of DBIs per plant for 1997 for the three age groups. Group A, B, and C plants had 5.6, 4.7, and 3.1 DBIs per plant, respectively. Older plants (group A) had significantly more DBIs per plant than the newer plants (groups B and C). The industry average was found to be 4.7 DBIs per plant.

Analysis identified several apparent reasons for the higher number of DBIs for older plants. Apparent reasons included less rigorous design-basis information, missing design information, and differences in both computational methodologies and applicable design code criteria.

### 3.2.2 Design-Basis Issues by Plant

The number of DBIs for 1997 varied widely among plants. Figures 3 through 6 show the number of DBIs for 1997 for plants located in each of the four NRC regions. As noted earlier, in 1997 the industry average for all 110 operating plants was 4.7 DBIs per plant. In addition, the industry median was 3 DBIs per plant as shown in Figures 3 through 6. Other observations include:

- Six pressurized water reactors (PWRs) accounted for 30 percent of the total number of DBIs; Crystal River 3 (37 DBIs), Point Beach 1 (27 DBIs), Point Beach 2 (24 DBIs), Millstone 3 (24 DBIs), D.C. Cook 2 (21 DBIs), and D.C. Cook 1 (18 DBIs).

- Twenty-one plants reported no DBIs during 1997

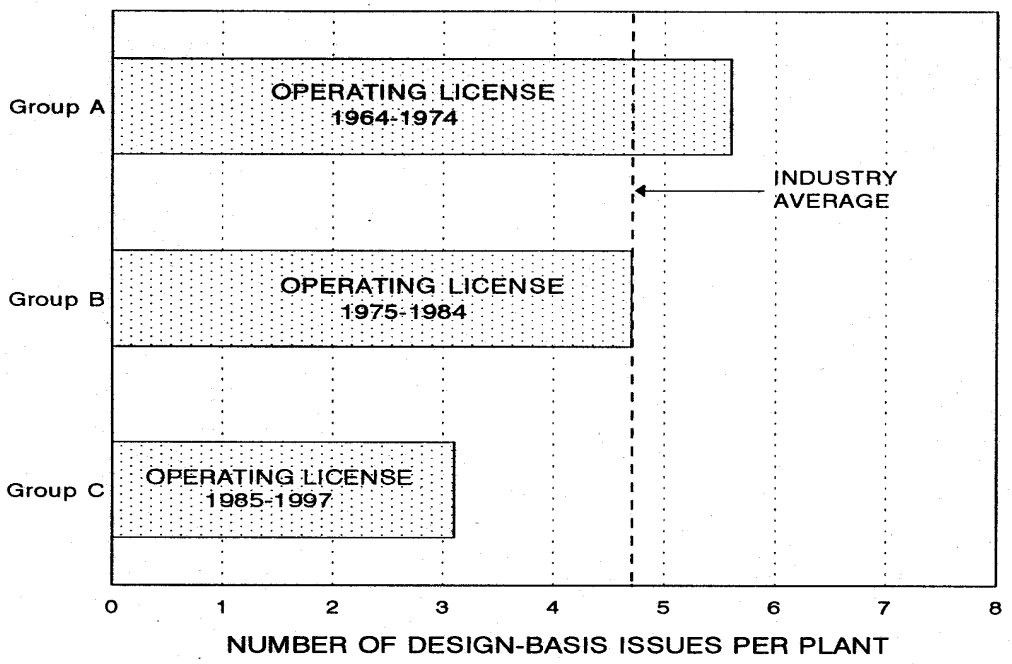



**Figure 2 Plant age and design-basis issues for 1997**.

The number of DBIs for each plant could be influenced by the following factors: (1) the quality, completeness and accessibility of the plant's design-basis documents (2) regulatory actions that cause licensees and the NRC to look for DBIs, (3) the capability of the licensee's engineering programs and organization to avoid introduction of DBIs and to find pre-existing DBIs, and (4) the rigor of the plant's reporting of the identified DBIs in LERs to the NRC.

As noted in Section 1.1, in 1997 the NRC staff reviewed all licensee responses to the NRC's 10 CFR 50.54(f) letters on licensees' design-basis document programs. The staff noted that most licensees had initiated a design-basis document program; however, the depth and scope varied significantly. The staff concluded that no further generic NRC action was required on this issue, but that plant-specific followup actions might be warranted to verify certain features of licensee programs. As a part of these followup actions, the NRC conducted design team inspections and other design-type inspections at selected sites.

In 1997, comprehensive design inspections were performed at 4 of the 21 plants that did not report DBIs for 1997—Ginna (Ref. 12), Arkansas Nuclear One 1 (Ref. 13), Perry (Ref. 14), and Washington Nuclear 2 (Ref. 15). A review of the team inspection reports indicated that some potential DBIs had been identified by the teams and had been left as unresolved items at the end of inspections pending further review by the NRC.

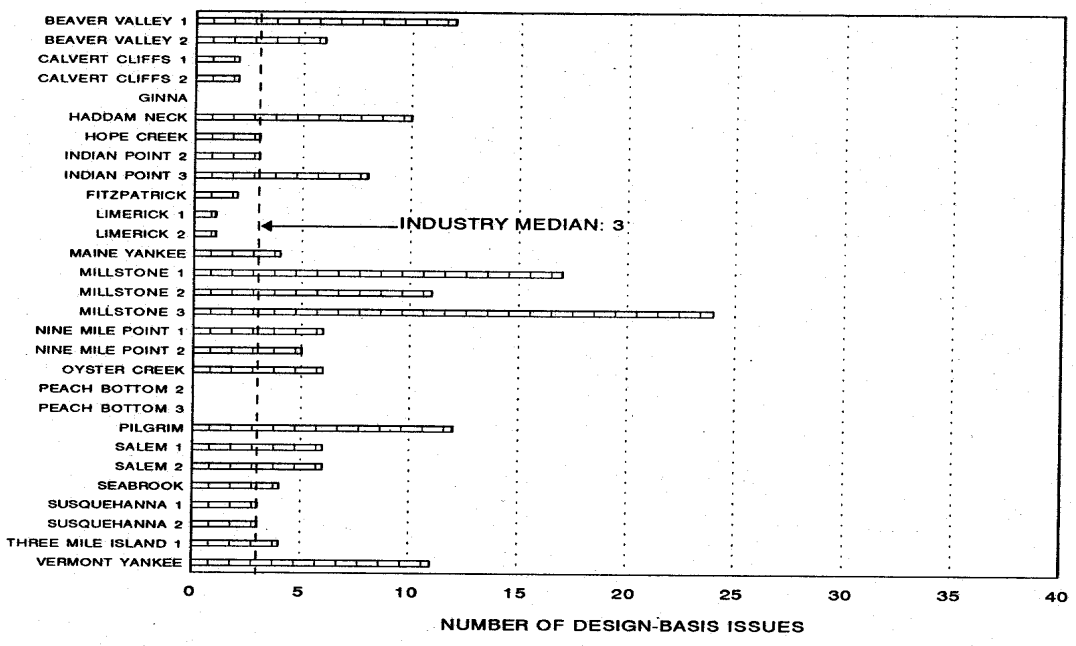



**Figure 3 Region I plant distribution of design-basis issues for 1997**

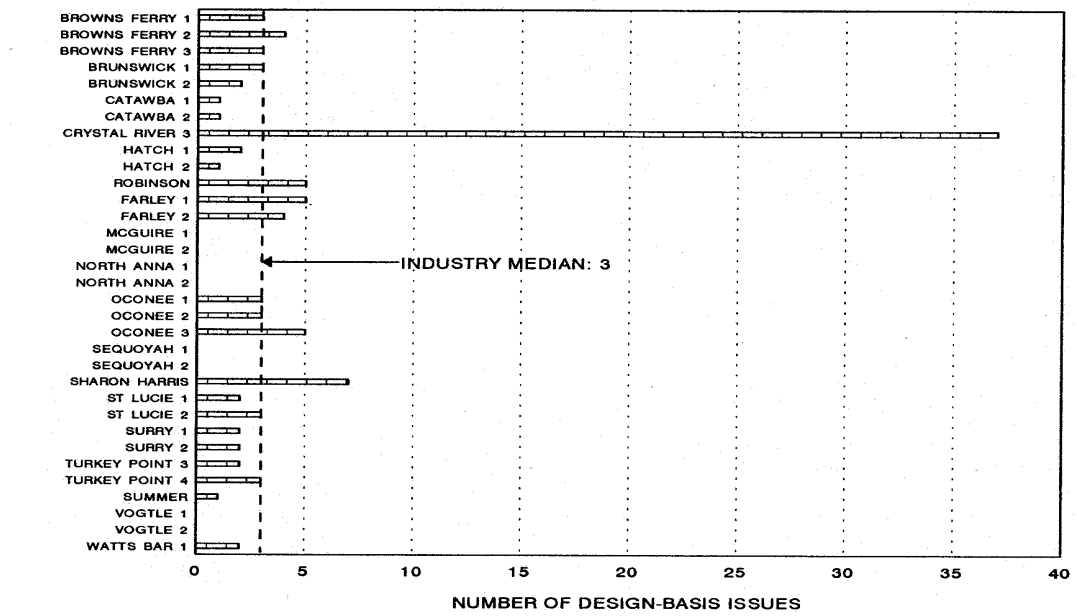
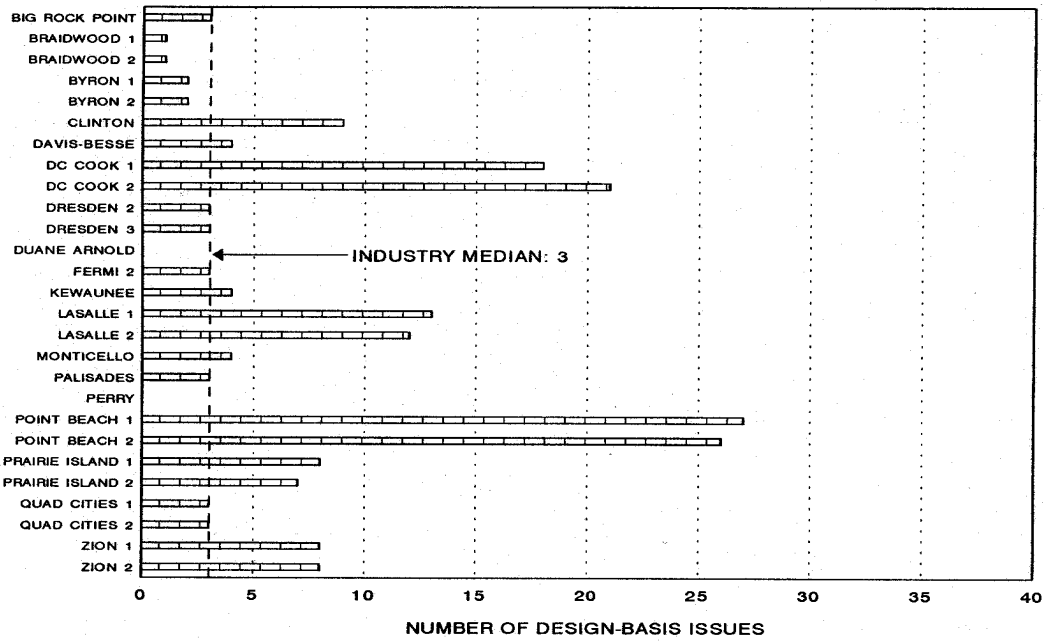



**Figure 4 Region II plant distribution of design-basis issues for 1997**



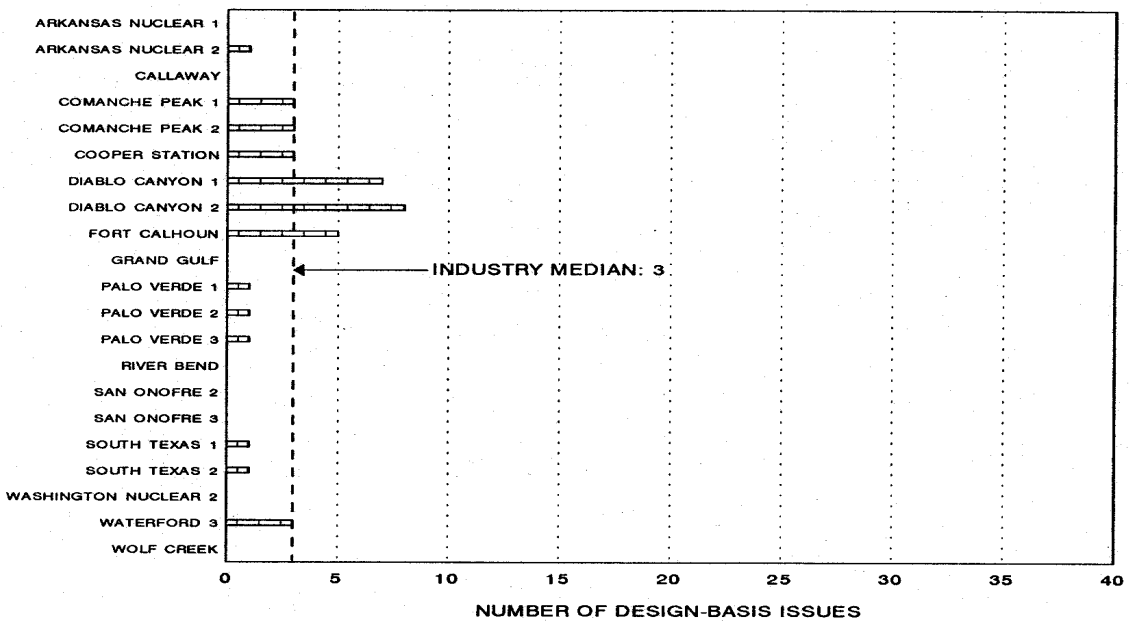




**Figure 5 Region III plant distribution of design basis issues for 1997**

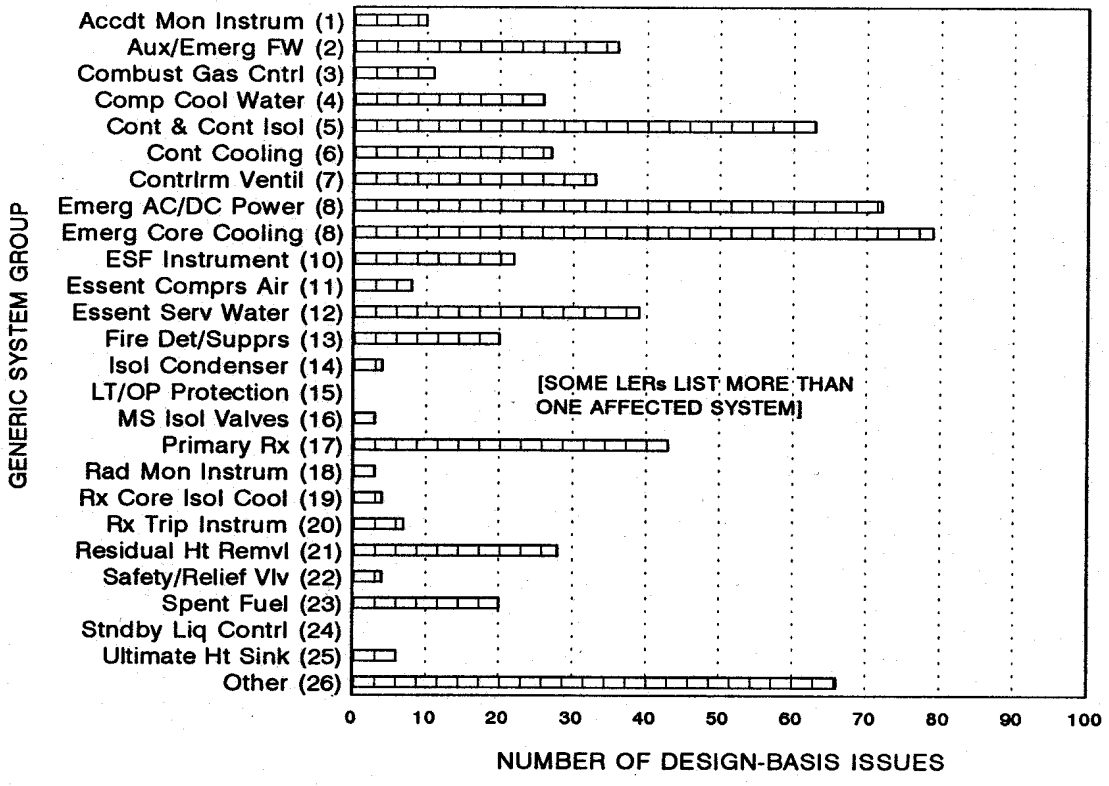
**Figure 6  Region IV plant distribution of design-basis issues for 1997**

### 3.2.3 Design-Basis Issues by Plant Systems

The DBIs reported in LERs for 1997 involved plant systems that have deterministic importance with respect to ensuring plant safety. A generic list of 26 plant system groups was used to categorize the system(s) reported in LERs. For a complete listing of systems that are listed under each group, see Appendix E. Some of the LERs reported more than one DBI, resulting in a total of 634 DBI-system counts for the 26 plant systems. Figure 7 shows the DBI count for each plant system. Analysis results (based on the 634 DBI-system counts) included the following observations:

- Six plant systems accounted for approximately one-half of the 512 LERs with DBIs. These systems were the emergency core cooling (13 percent), emergency ac/dc power (11 percent), containment and containment isolation (10 percent), primary reactor (7 percent), auxiliary/emergency feedwater (6 percent), and essential service water (6 percent).

- Approximately 13 percent of the LERs with DBIs did not mention the specific system or component that failed to meet design requirements (i.e., classified as “other”). Most of these DBIs involved multiple nonspecific systems, fire, or external events such as




floods.

**Figure 7 Distribution of design-basis issues by plant system in 1997**

### 3.2.4 Design-Basis Issues for Boiling Water Reactors and Pressurized Water Reactors

For 1997 there were slightly fewer DBIs per plant for boiling water reactors (BWRs) than there were for PWRs. There were 149 DBIs for BWRs and 364 for PWRs. On the basis of the number of operating BWRs (37 plants) and PWRs (73 plants), this represents an average of 4.0 DBIs per BWR plant and 5.0 per PWR plant. As discussed later in Section 5.1, of the 14 most risk significant events (as determined by ASP analysis) for the period 1992–1997, 12 occurred at PWRs and 3 involved DBIs. All three of the important ASP events that involved DBIs occurred at PWRs.

### 3.2.5 Mode of Discovery and Reporting of Design-Basis Issues

A review of the 512 DBIs for 1997 identified the following six predominant initiator categories (identifying activities) as shown in Figure 8:

- Unspecified (206 DBIs). If the licensee did not specifically mention the initiating activity that identified the DBI, it was categorized as unspecified.

- Self-revealing (103 DBIs). This initiator consisted of fortuitous discoveries of DBIs through activities not specifically performed to look for or find the DBI. These resulted from operational problems at the plant or a routine plant activity.

- Design-basis review (85 DBIs). This category was chosen if the licensee indicated that the DBI was discovered during an ongoing design basis or reconstitution program efforts.

- Generic communications (66 DBIs). This category was chosen if the licensee indicated that the DBI was discovered as a result of followup to an NRC generic communication (GLs, bulletins, or information notices [INs]). Generic Letter 96-06, “Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions,” (Ref. 16) was the predominant contributor.

- NRC identified (38 DBIs). This category was chosen if the licensee mentioned in its LER that the NRC discovered the DBI and reported it to the licensee.

- Industry initiatives (14 DBIs). This category was chosen if the licensee indicated that the DBI was discovered as a result of followup activities prompted by nuclear industry identified design issues.

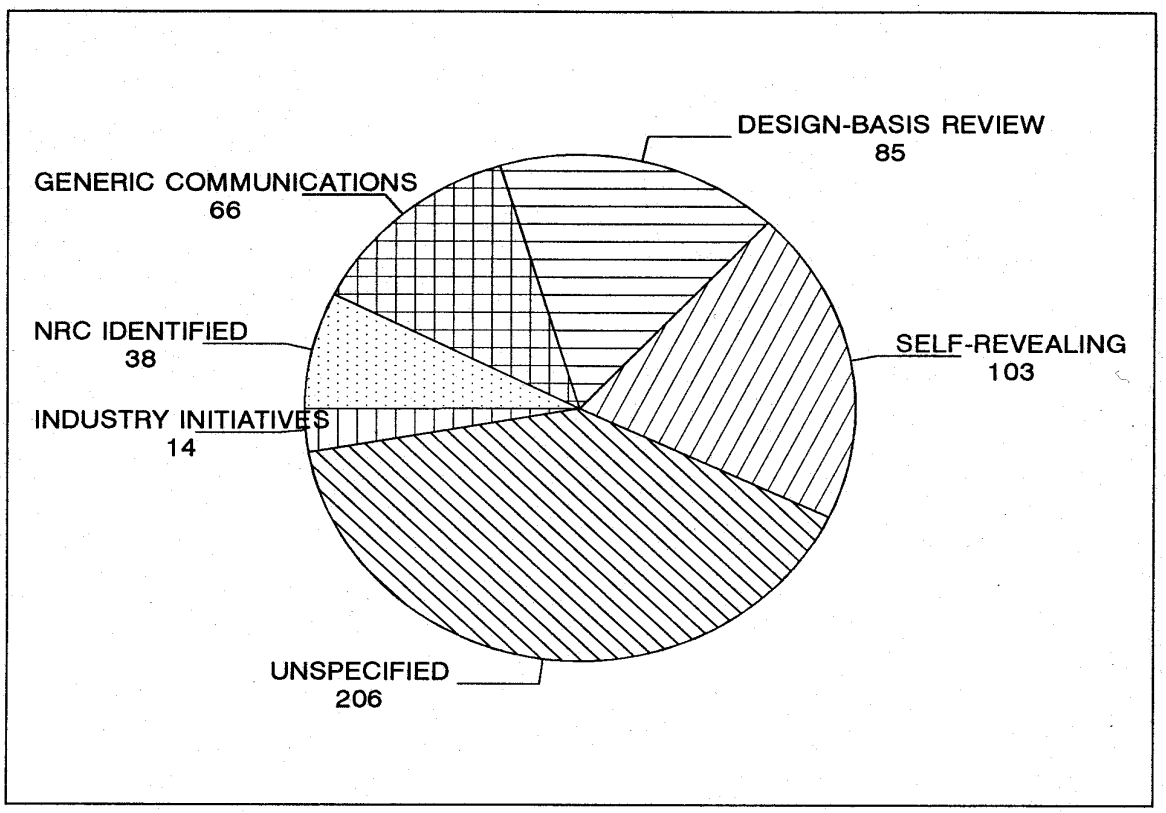



**Figure 8 Activity that initiated licensee event reports with design-basis issues for 1997**

3.2.6 Reportability Reasons of Design-Basis Issues

Licensee event reports containing DBIs for 1997 were analyzed using the SCSS database to determine (1) the reportability reasons and (2) plant status at the time of discovery (i.e., operating at power or shutdown). When LERs are coded in the SCSS database by Oak Ridge National Laboratory (ORNL), some may receive inputs in more than one category. Consequently, some of the LERs with DBIs discussed in this section, and in Section 3.2.7, may have more than one "reportability reason," "primary cause," or "corrective action."

As shown in Figure 9, the most frequent reportability reason (83 percent of DBIs) was "Unanalyzed Conditions," as required by 10 CFR 50.73(a)(2)(ii). Other significant reportability reasons were "Could Affect Safety Function" (14 percent of DBIs) [10 CFR 50.73(a)(2)(v)], and "Technical Specification Violations or Plant Shutdown by Technical Specification Requirements" (12 percent of DBIs) [10 CFR 50.73(a)(2)(v)].

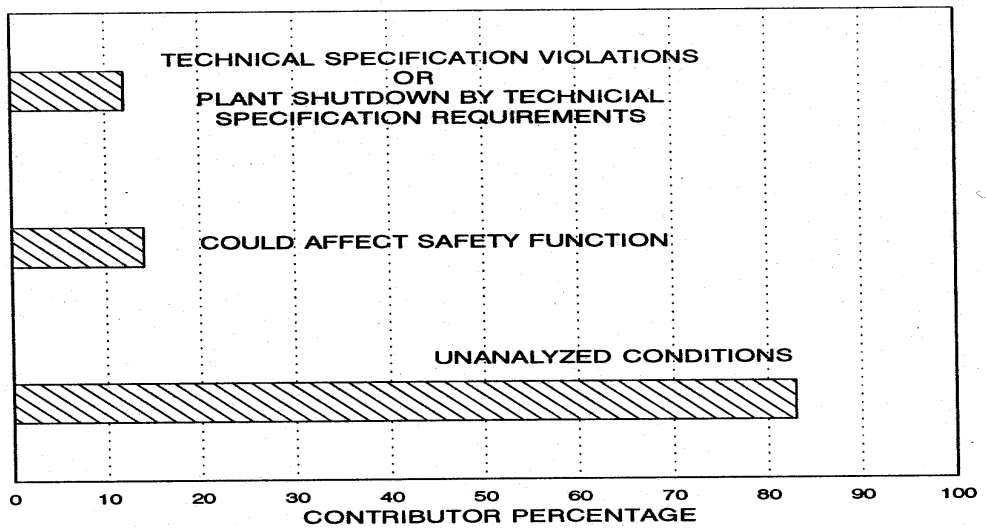



**Figure 9 Reportability reasons for design-basis issues for 1997**

3.2.7 Causes of Design-Basis Issues

Licensees often cited multiple causes for DBIs as shown in Figure 10. As such, individual LERs with DBIs can contribute to more than one cause category. Of the 512 LERs with DBIs reported for 1997, the most frequent contributing causes included design errors dating back to the time of original licensing (72 percent), procedural deficiencies (27.5 percent), human error (22 percent), poor work control practices (15 percent), and plant modifications (13 percent). Because most DBIs were reported while the plants were shut down, licensees typically stated that the risk associated with the DBI was minimal, even though in some instances, multiple trains or systems had been outside their respective design basis over several operating cycles.

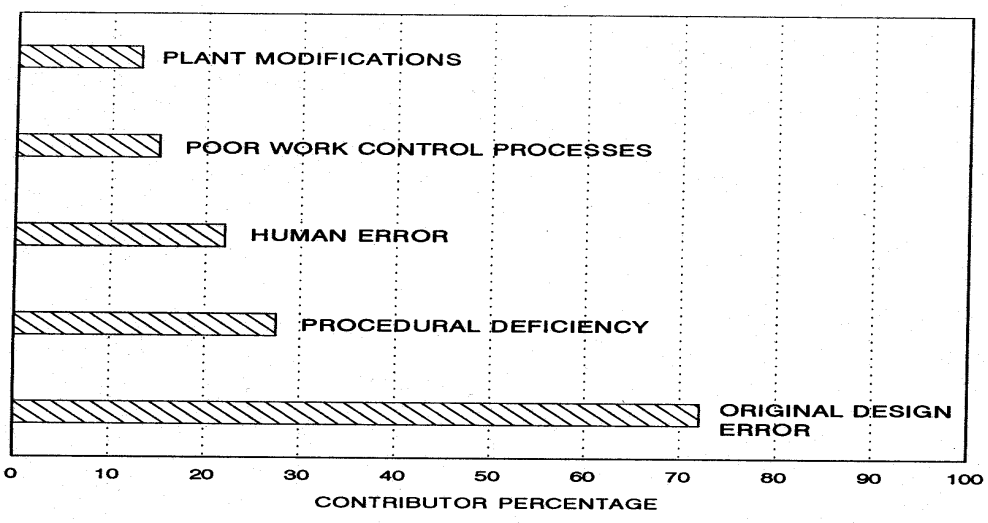


**Figure 10 Causes of design-basis issues for 1997**

# 4 NRC ENGINEERING AND DESIGN INSPECTION EFFORT AND DESIGN-BASIS ISSUE IDENTIFICATION

NRC inspections, especially those which focus on engineering and design activities, contribute to the identification of DBIs that are subsequently documented by licensees in LERs. Engineering and design inspections typically have included the areas of onsite engineering, engineering modifications, safety evaluations, configuration control, design control, evaluations of licensee engineering assessment activities, followup of engineering activities, and testing activities to verify design adequacy. To assess the relationship, if any, between NRC engineering and design inspection effort (including contractor support) and the identification of DBIs during the period 1994-1997, several factors and analyses were considered:

- The average number of engineering inspection hours at each plant during 1994–1997.
- The number of engineering inspection hours at each plant during 1997.
- The number of engineering inspection hours at each plant during 1997 divided by the number of DBIs for 1997 at the plant. (A low quotient value would indicate a high relative incidence of DBIs, while a high value would indicate a low relative incidence of DBIs).
- The relationship between the total number of reported DBIs for the period 1995–1997 and the average number of NRC engineering inspection hours for the same period.

## 4.1 Trends in Engineering Inspection Effort by Region and Industry

In each year during the period 1994–1997, plants located in Regions I and III had more engineering inspection hours than plants in Regions II and IV. During this period, Regions I and III averaged approximately 730 hours of engineering inspections per plant per year, whereas plants in Regions II and IV averaged approximately 535 hours of engineering inspections per plant per year. During the same period, plants in Regions I and III also received more total inspection hours for all inspection areas (see Figure 11). These results may have been influenced by the higher percentage of older plants in Regions I and III (i.e., older plants tend to report more DBIs, see Section 3.2.1), and the greater number of plants that were the focus of increased regulatory attention both at the regional level and the agency level during the period.

The total number of engineering inspection hours increased from 1994 through 1996 and then decreased in 1997, largely as a result of the changes in inspection effort by Regions I and III during this period (see Figure12). For example, Table 1 provides a listing of plants that averaged more than 1000 hours of engineering inspection effort annually for the period 1994 through 1997.

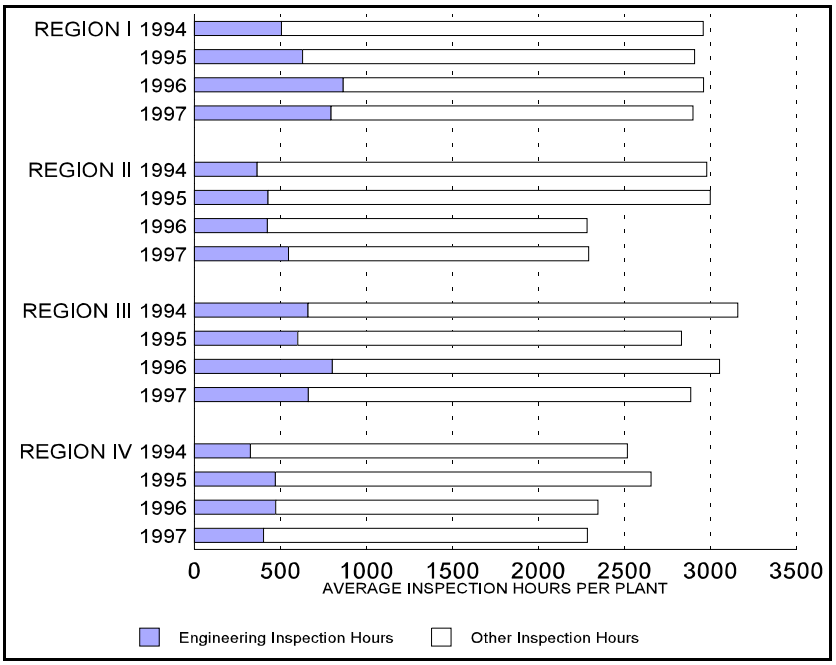



**Figure 11 Engineering inspection hours vs. total inspection hours**

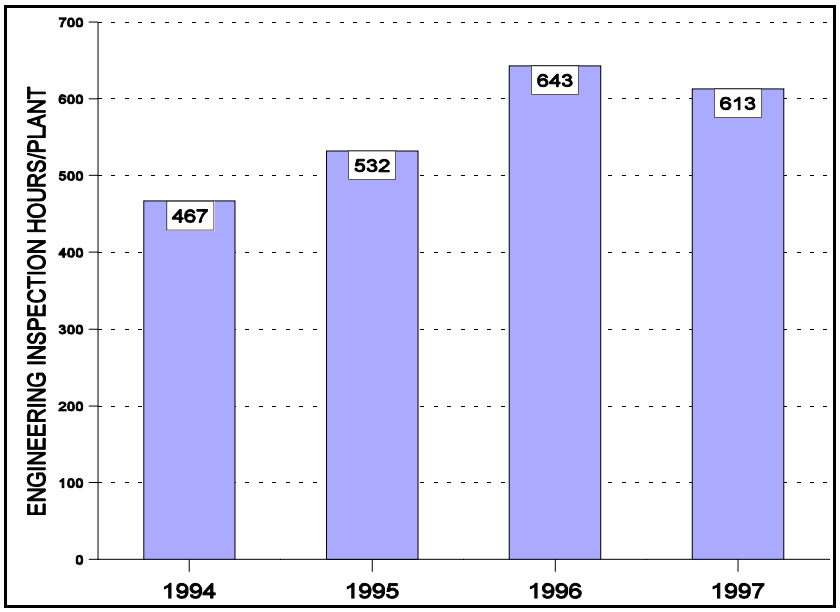


**Figure 12  Trend in NRC engineering inspection hours per plant**

**Table 1 Plants having more than 1000 engineering inspection hours/year (1994–1997)**

| Year | Region I | Region II | Region III | Region IV |
|---|---|---|---|---|
| 1997 | Ginna<br>Millstone 2<br>Millstone 3<br>Pilgrim<br>Salem 1<br>Salem 2<br>TMI-1<br>Vermont Yankee | Crystal River 3<br>Robinson<br>St. Lucie 1 | Clinton<br>Davis-Besse<br>Perry | ANO-1<br>WNP-2<br>Waterford 3 |
| 1996 | R.E. Ginna<br>Haddam Neck<br>Maine Yankee<br>Millstone 1<br>Millstone 2<br>Millstone 3<br>Salem 1<br>Salem 2 | Crystal River 3 | Dresden 2<br>Dresden 3<br>Fermi 2<br>Monticello<br>Palisades<br>Point Beach 2 | Cooper<br>WNP-2<br>Waterford |
| 1995 | Millstone 1<br>Millstone 2<br>Salem 1 | Browns Ferry 3<br>Watts Bar 1 | Fermi 2<br>Palisades<br>Perry | |
| 1994 | Indian Point 2 | Watts Bar 1 | Dresden 3<br>Fermi 2<br>Palisades<br>Perry | Cooper |

## 4.2 Variations in Engineering Inspection Effort by Plant During 1997

Figures 13–16 show the engineering inspection hours for each plant in Regions I–IV for 1997, respectfully. The 1997 industry median and average for NRC engineering inspection hours per plant were 457 hours and 613 hours, respectively. Browns Ferry 1 (Region II) had a low of approximately 90 engineering inspection hours, while Millstone 3 (Region I) had a high of almost 3700 engineering inspection hours. As shown in Figure 2, older plants reported significantly more DBIs than new plants during 1997. Since plants in Regions I and III include a disproportionately higher percentage of older plants, plant age appears to be a factor for the larger number of DBIs in these regions. However, another possible reason was the higher level of engineering inspection effort in these regions. The hypothesis that NRC engineering inspection effort had a direct and indirect effect on the number of DBIs reported by licensees is evaluated in Section 4.4.

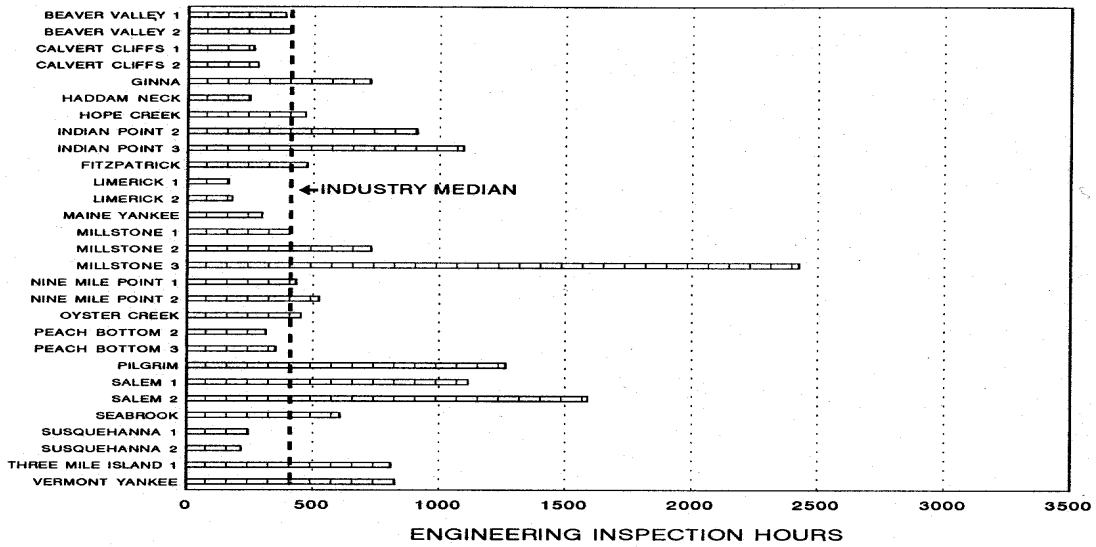



**Figure 13 Region I plant engineering inspection hours for 1997**

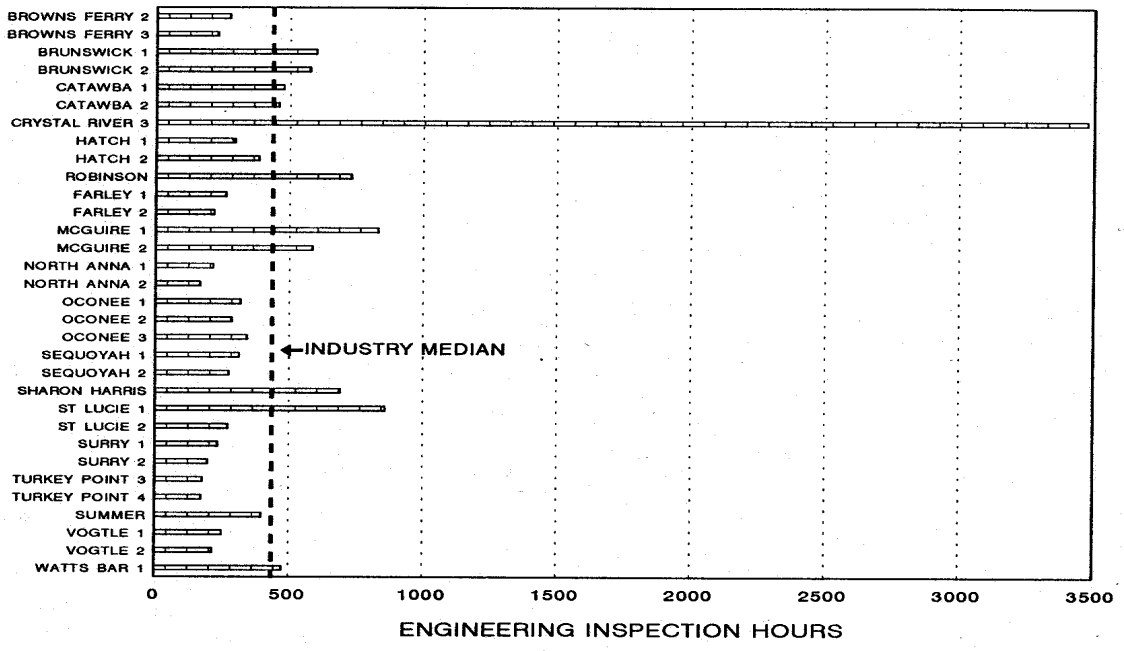



**Figure 14 Region II plant engineering inspection hours for 1997**

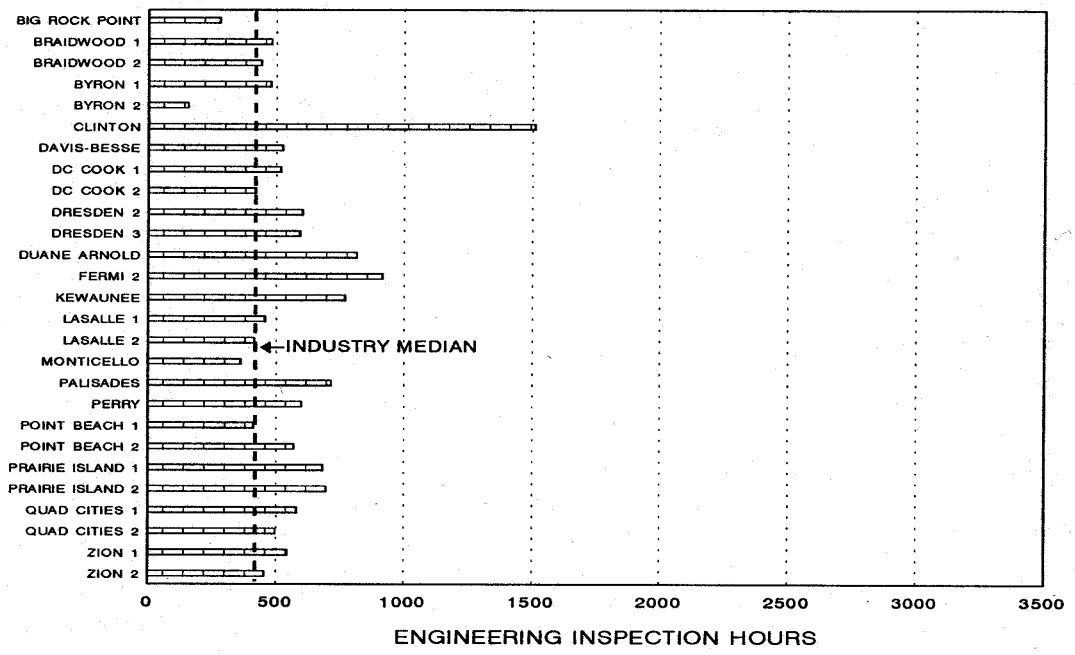



**Figure 15 Region III plant engineering inspection hours for 1997**

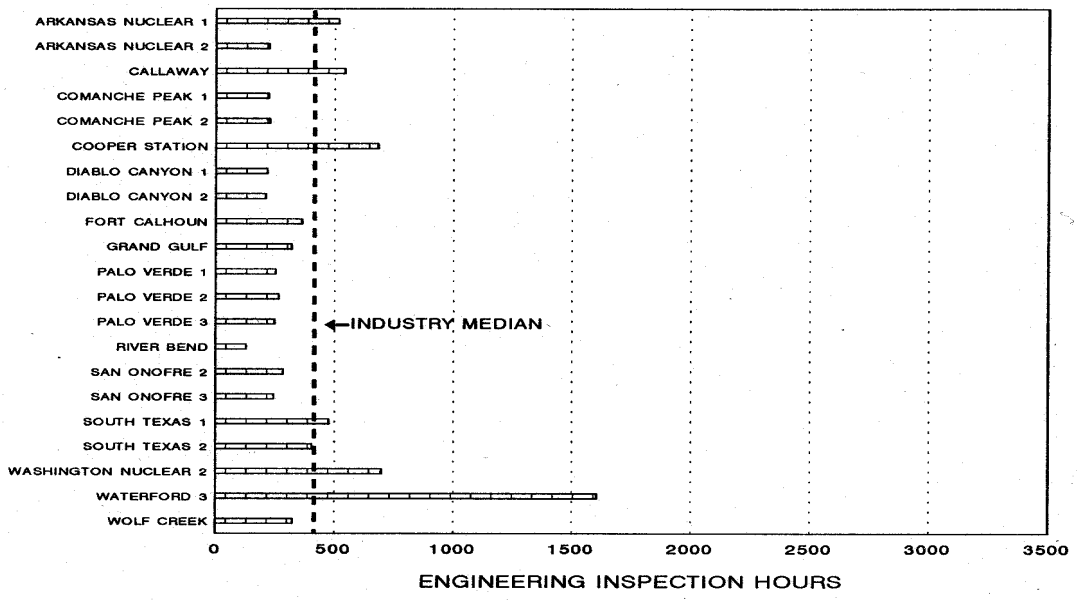
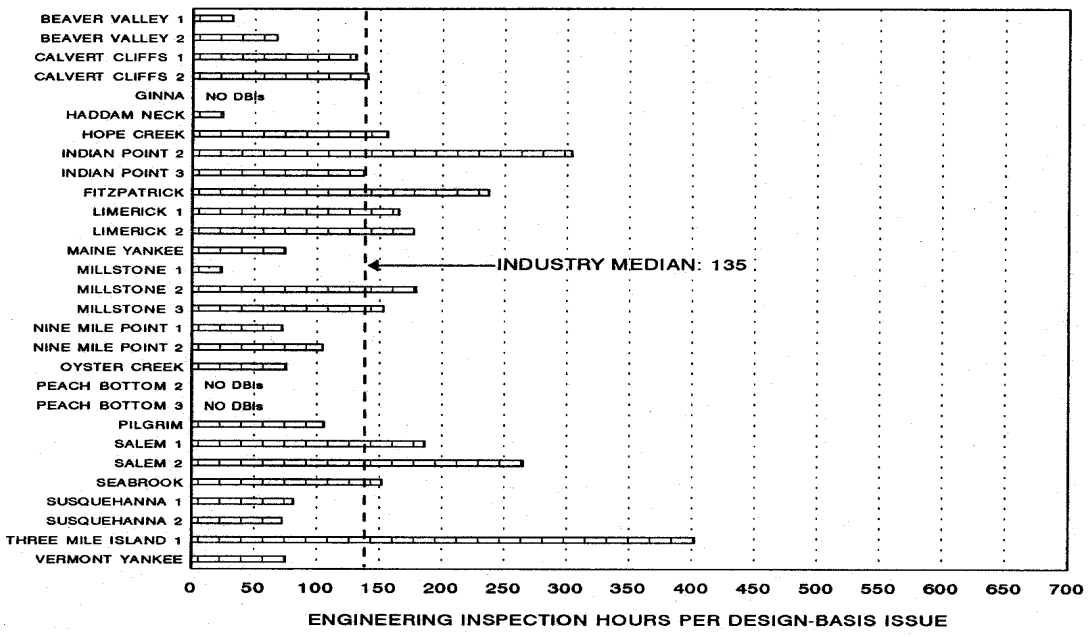



**Figure 16 Region IV plant engineering inspection hours for 1997**

## 4.3 Variations in the Number of Engineering Inspection Hours per Reported Design-Basis Issue

Figures 17–20 show the number of engineering inspection hours per DBI for each plant during 1997 for Regions I–IV, respectively. These figures were produced using the information shown in Figures 13–16 which shows the plant-by-plant distribution of NRC engineering inspection effort (in hours) for 1997, and Figures 3–6, which shows the plant-by-plant distribution of DBIs for 1997. Dividing the inspection hours for each plant by the total number of DBIs (found from all initiators), provides the number of engineering inspection hours per DBI for each plant. The 1997 industry median for engineering inspection hours per DBI was 135.

The results for the 21 plants with no DBIs during 1997 (which would result in an infinite ratio) are not plotted, but are annotated as “no DBIs.” Of the 21 that had no DBIs during 1997, 16 came from either Region II or IV. In addition, 14 of the 21 plants that reported no DBIs during 1997, received less than the median number of engineering inspection hours. In many cases, the plants that did not submit LERs with DBIs for 1997 had similar designs to plants that did submit LERs with DBIs. Additionally, DBIs documented in 1997 generic feedback correspondence was typically applicable to a broad range of nuclear power plant facilities. Similarly, the 10 CFR 50.54(f) letter issued in 1996 requesting reviews of design-basis control programs and processes contributed to the identification of DBIs during 1997. Had comprehensive NRC engineering team inspections been conducted at more facilities reporting no DBIs for 1997, it would be expected that additional DBIs may have been discovered and reported.



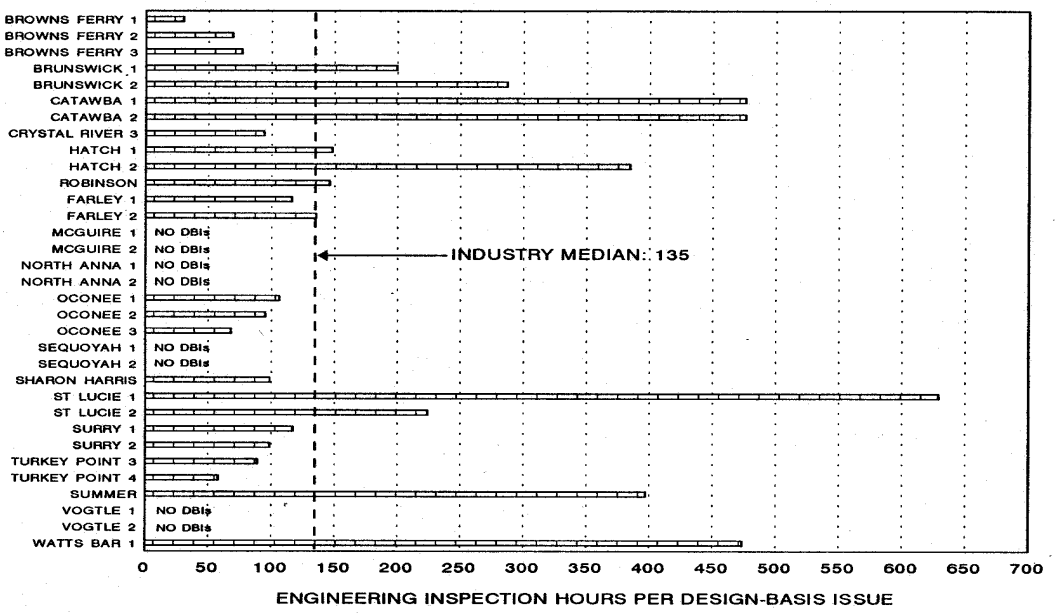



**Figure 17 Region I plant engineering inspection hours per design-basis issue for 1997**

**Figure 18 Region II plant engineering inspection hours per design-basis issue for 1997**

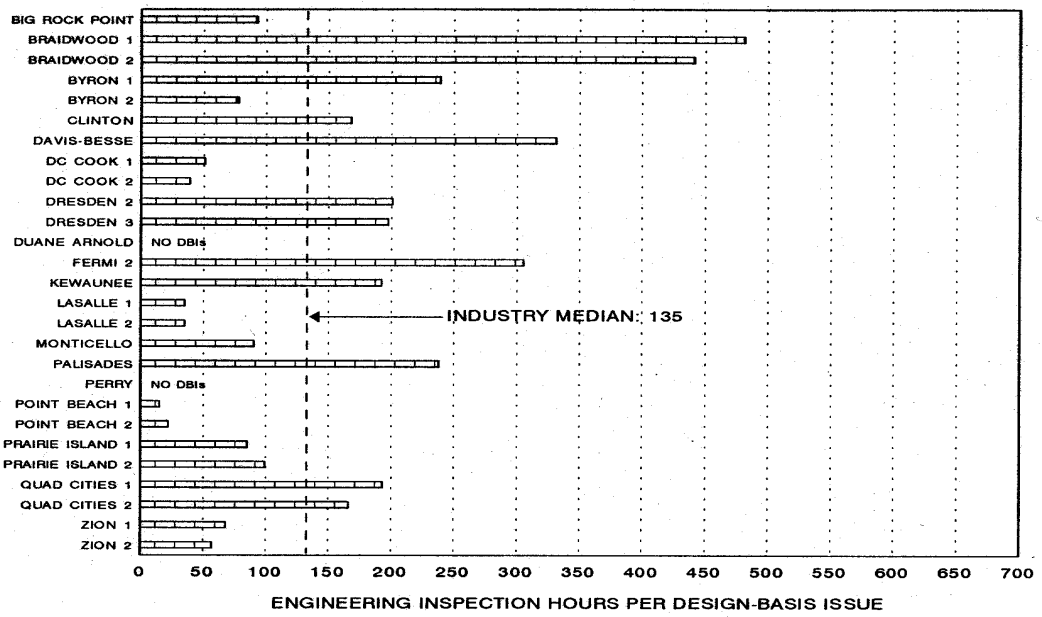


**Figure 19  Region III plant engineering inspection hours per design-basis issue for 1997**

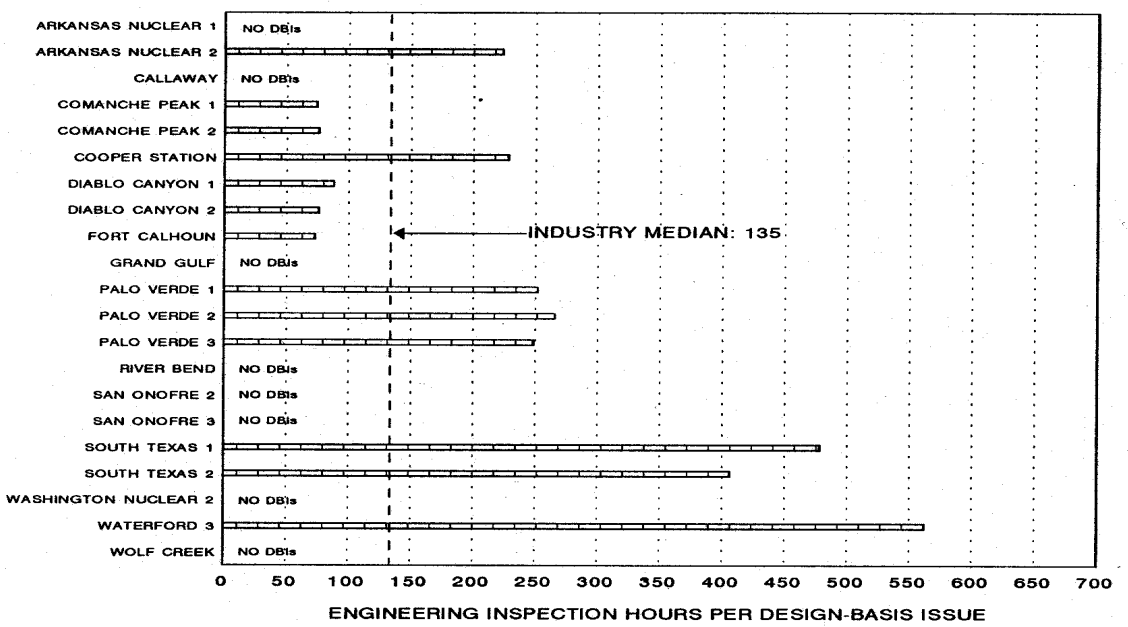



**Figure 20 Region IV plant engineering inspection hours per design-basis issue for 1997**

## 4.4 Correlation Between Design-Basis Issues and NRC Engineering Inspection Effort

NRC inspection programs are routinely supplemented to address inspection or operating experience issues. As the number or significance of DBIs increases, a licensee (and the NRC) may increase its effort to determine the extent, root causes and corrective actions necessary to resolve the findings. With respect to DBIs, an escalation of engineering inspection effort by both the industry and the NRC appears to have resulted in additional DBI findings.

During 1995–1997, the NRC average annual number of engineering inspection hours for a plant correlated with the total number of DBIs for the plant as shown in Figure 21. Consequently, an increase in the number of engineering inspection hours generally was accompanied by an increase in the total number of reported DBIs. The total number of DBIs reported by the industry was a direct result of NRC inspection findings; or the indirect result of licensee actions in response to NRC inspection findings or generic communications related to DBIs, and self-revealing design problems.

A linear regression analysis of the average number of annual engineering inspection hours and corresponding total number of reported DBIs was performed. For this analysis, the first variable was the three year (1995–1997) total number of DBIs from all initiators for a given facility, and the second variable was the average annual number of engineering inspection hours (1995–1997) for that facility. For example, a dual unit facility having identical units counted as one data point for the analysis. Similarly, if the dual unit facility had different reactor type designs, it was counted as two separate data points. In addition, the three year total number of

DBIs and the average number of annual engineering inspection hours used in the linear model was the value reported by the lead unit only. The data set was also truncated to remove the facilities that reported five or fewer DBIs over the three year period. The result of the regression analysis is shown in Figure 21.

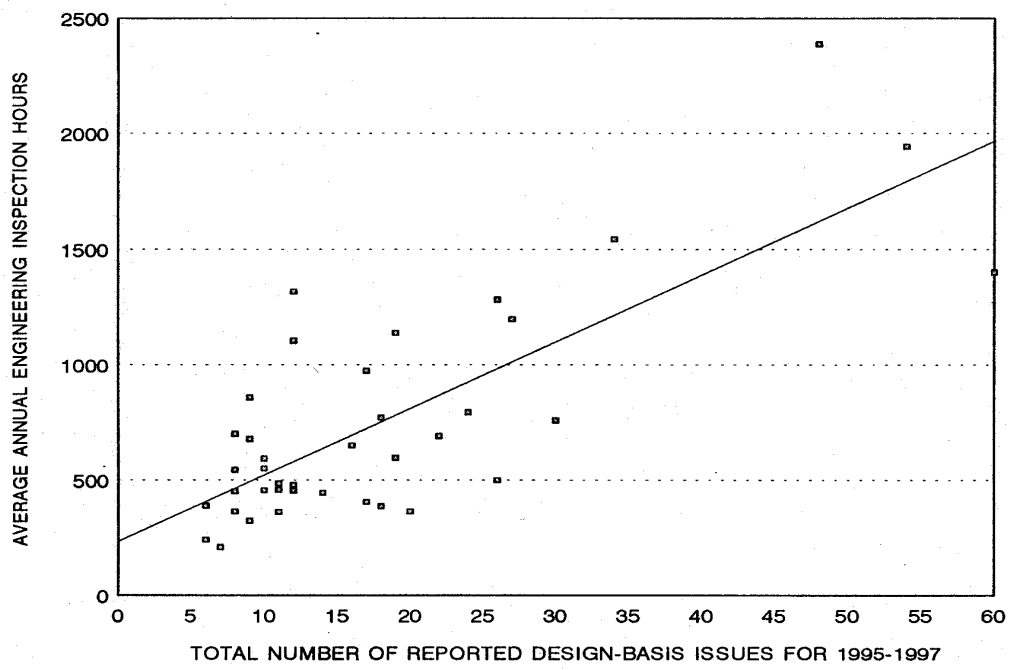



**Figure 21 Engineering inspection hours and design-basis issue relationship (1995–1997)**

The statistical analysis results indicated that the linear relationship between the two variables had a correlation coefficient of 0.77, indicating a moderately strong relationship between the variables. The “P” value was less than 0.01, indicating a statistically significant relationship between the two variables at the 99 percent confidence level. The “R-Squared” statistic indicates that the linear model as fitted explains approximately 60 percent of the variability in the dependent variable (engineering hours expended). A “Lack-of-Fit” test was performed to determine whether the linear regression model was adequate to describe the observed data. This test indicated that the linear regression model was adequate for the observed data.

As shown in Figure 8, there are several initiators for DBIs, the largest category being “Unspecified” (206 of 512). A review of the LERs which specified an initiator resulted in the following observation; 62 percent (189) of the LERs with DBIs during 1997 were in the categories “NRC identified,” “generic communications,” and “design-basis review.” Each of

these initiator categories appears to be the result of NRC engineering and design focus, initiatives, or NRC inspection efforts.

## 4.5 Design-Basis Issues Reported in Safety System Engineering Inspections Performed in 1997

In response to DBIs discovered at Millstone and at other facilities during the mid-1990s, the NRC developed the Safety System Engineering Inspection (SSEI) program. Implementation of SSEIs began in early 1997. Plant selection criteria at the time included NRC ratings of the licensee's engineering area, NRC watch list status, region insights from other inspections, plant age, and the status of design-basis reconstitution. Plant systems selected for review were based on individual plant examination results, and previous NRC inspection experience with the system.

NRC IN 98-22, "Deficiencies Identified During NRC Design Inspections," lists the most significant technical and programmatic issues, including: (1) modifications or evaluations that resulted in operation of the plant outside the licensing bases, (2) errors in analyses of the pump suction swap over from the refueling water storage tank/borated water storage tank to the reactor sump during a loss-of-coolant accident (LOCA), (3) inadequate testing of safety-related components, (4) issues concerning implementation of computer evaluation models used for analyzing emergency core cooling system (ECCS) response to design-basis accidents, (5) system operation at a temperature in excess of the design basis, (6) errors made in evaluating post-accident temperatures for safety related pump rooms, and (7) lack of controls or specified outage times for limiting system lineups that could challenge design-basis considerations.

The IN (98-22) states that most issues have resulted from errors in the original-design or in design modifications, calculational errors, inadequate corrective action, inadequate testing, and documentation discrepancies. Many of the original design, design modifications, and calculational errors can be attributed to the inadequate specification and control of system and discipline interfaces, inadequately verified calculational assumptions, or the use of superseded calculations. Some licensees did not fully evaluate the impact of calculational revisions on other calculations, or on operating and test procedures. Changes to operating and test procedures were not always reviewed against the design calculations to ensure that the assumptions in the calculations were still bounding. Also, the lack of a controlled, easily retrievable design-basis documentation had, in some instances, hindered the ability of licensee engineers to identify all design-basis safety functions of a system or component. Appendix D, "Significant 1997 Safety System Engineering Inspection Findings," lists the 14 SSEIs performed during 1997 and their major findings.

## 4.6 The Effect of Engineering Inspection on Facility Assessments

The analyses in this section relate to NRC inspection program activities which have been substantially superseded by the NRC's revised reactor inspection program in support of the revised plant assessment process. However, these analyses were conducted to provide insights on relationships between the level of inspection effort and plant performance assessment which may be of value in identifying issues and assessing the effectiveness of the revised reactor inspection program.

<u>Plants with a High Number of DBIs for 1997</u>

The routine engineering inspection program of the mid-1990s (e.g., other than engineering team inspections) resulted in more favorable engineering assessments of licensees. Most plants with a large number of DBIs for 1997 had previously received favorable engineering

assessment ratings. Sometimes the more favorable assessments in the engineering area appeared to have been the result of more limited inspection effort. Of the 21 plants that reported no DBIs in LERs during 1997 (see Figures 3–6), 14 received less than the median number of engineering inspection hours.

A review was conducted of the trends in regional plant engineering assessments for those plants issuing a high number of DBIs for 1997. The review found that plants often reported more DBIs following a more thorough engineering inspection for design compliance using contractor support. The increase in the number of DBIs, however, generally lowered the engineering assessment in subsequent assessment periods. In some instances, this lower assessment rating led to increased regional or agency oversight.

Based on the correlation between engineering inspection hours and reported DBIs discussed in Section 4.4 (i.e., an increase in the number of engineering inspection hours results in an increase in the number of DBIs) it could be concluded that plants with a higher number of DBIs and subsequent lower engineering assessments, were caused, in part, by the increase in engineering inspection hours.

Plants with a High Number of DBIs Between 1990–1997

As shown in Table 2, during the period 1990–1997, 11 plants were responsible for 29 percent of the DBIs. Each of these plants had an NRC inspection of its design basis. These data suggest that the number of DBIs reported in LERs by a plant was elevated by an NRC design-basis review, beyond those that resulted from NRC generic documents, such as GL 96-06 (which also influenced the increase of DBIs in 1996 and 1997). For example, this hypothesis appears to be supported by Pilgrim, which had 21 DBIs for 1990–1997. Seventeen of these DBIs were for 1997 and many were the direct result of an in-depth Region I design inspection (50-293/97-05) conducted after the plant completed a design-basis review in response to the 1996 10 CFR 50.54 (f) letter.

**Table 2 Plants with the largest total number of design-basis issues (1990–1997)**

| Plant Name | Number of DBIs |
|---|---|
| Crystal River 3 | 93 |
| Millstone 1 | 85 |
| Indian Point 3 | 59 |
| Millstone 3 | 55 |
| Palisades | 55 |
| Fort Calhoun | 45 |
| Millstone 2 | 43 |
| Maine Yankee | 41 |
| Dresden 2 | 41 |
| Haddam Neck | 36 |

# 5 SAFETY SIGNIFICANCE OF LICENSEE EVENT REPORTS WITH DESIGN-BASIS ISSUES BASED ON ACCIDENT SEQUENCE PRECURSOR DATA

A review of ASP data for selected events occurring between 1990 and 1997 was performed to assess the potential risk importance of DBIs at nuclear power plants,. ASP Program information and findings related to DBIs are contained in Sections 5.1 through 5.3.

## 5.1 Accident Sequence Precursor Program Information

The ASP program reviews and evaluates operational events (primarily reported in LERs) at U.S. operating reactors. The ASP program identifies and categorizes precursors to potential severe core damage accident sequences. Accident sequences are those that, if additional failures occurred, would have resulted in inadequate core cooling, causing severe core damage. For example, in a postulated LOCA with a failure of a high-pressure injection system, the precursor would be the high-pressure injection system failure. The ASP program analyzes potential precursors and calculates their CCDP (Ref. 17). The CCDP is the probability that the event or condition could have progressed to core damage given the existence of the failed or degraded protective or mitigating features or initiating event. To be classified as an ASP event, the event must have a CCDP of at least $1.0 \times 10^{-6}$.

Initially, a computerized search of the SCSS data base at the Nuclear Operations Analysis Center of the ORNL is conducted to identify LERs that meet minimum selection criteria for precursors. Other events may also be selected for review and analyses. Events that are typically analyzed have the following characteristics:

- unexpected core damage initiators (loss of off-site power, steam generator tube rupture, and small-break LOCA);
- all events in which a reactor trip was demanded and a safety-related component failed;
- all support system failures, including failures in cooling water systems, instrument air, instrumentation and control, and electric power systems;
- any event where two or more failures occurred;
- any event or operating condition that was not predicted and proceeded differently from the plant design basis; and
- any event that, based on the reviewer’s experience, could have resulted in or significantly affected a chain of events leading to potential severe core damage.

Many events are not analyzed within the ASP program if they involved only one of the following:

- a component failure with no loss of redundancy,
- a short-term loss of redundancy in only one system,

- an event that occurred prior to initial criticality,
- design or qualification error that is small relative to what is predicted (e.g., an error of a few percent in an actuation setpoint),
- an error bounded by a reactor trip or a loss of feedwater,
- an event with no appreciable impact on safety systems, or
- an event involving only post-core-damage impacts.

Accident Sequence Precursor Program Limitations

- The ASP program does not evaluate potentially significant events that are impractical to analyze because of the lack of information or inability to reasonably model within a probabilistic risk assessment framework, considering the level of detail typically available in probabilistic risk assessment models and the resources available to the ASP program. Such events are thought to be capable of impacting core damage sequences. The events usually involve component degradations in which the extent of the degradation cannot be determined or the impact of the degradation on plant response can not be ascertained.
- The present ASP program does not model containment-related events, such as unavailability of containment function, containment isolation, containment cooling, containment spray, or post-accident hydrogen control. Models are currently under development.
- In addition, the frequencies and failure probabilities used in the calculations are derived in part from data obtained across the population of light water reactors. An attempt has been made to make the frequencies and failure probabilities plant-specific. However, this effort is not complete. Because of this, the conditional probabilities determined for each precursor cannot be rigorously associated with the probability of severe core damage resulting from the actual event at the specific reactor plant where it occurred.

Other Limitations

- Uses a plus or minus 15 day window to consider related (“windowed”) LERs in determining risk.
- Limits long-term equipment issues to a 1-year period. Thus multiple equipment degradations or failures caused by multiple DBIs that all existed since plant startup but were unidentified for many years would be analyzed separately rather than collectively. Thus the overall plant risk is not evaluated collectively.
- Limits the period for reviewing revisions to a specific LER to about June of the year following the discovery date of the initial LER. Thus the latest information in an LER revision received after about June would not be included in the analysis.
- Typically uses a similar previous ASP analysis if one was performed. This analysis will be used as a generic model and a plant-specific analysis is not performed.

- May not have the data in the LERs necessary for a meaningful analysis.
- Does not question licensee statements in LERs or request clarification.
- Uses licensee individual plant examinations.
- Are conducted regardless of the lack of information in LERs, if the LER passes a threshold. This necessitates the frequent use of engineering judgment.

## 5.2 Accident Sequence Precursor Observations

Accident Sequence Precursor Event Trend Results (1990–1997)

A review of trends of the number of LERs that were classified as ASP events from all causes (human performance, maintenance, operations, design, or others) indicates a significant reduction since 1990. An 82 percent reduction (see Table 3) in the number of ASP events occurred from 1990 through 1997. The reduction in the number of ASP events is an indicator that plant performance has improved, which is in agreement with many other plant performance indicators used by the industry and NRC. The reduction in the number of ASP events is also due in part to ASP selection criteria and model changes.

**Table 3 Accident sequence precursor event trends (1990–1997)**

| Year | Number of events |
|---|---|
| 1990 | 28 |
| 1991 | 27 |
| 1992 | 27 |
| 1993 | 16 |
| 1994 | 9 |
| 1995 | 10 |
| 1996 | 14 |
| 1997 | 5 |

Trends of Design-Basis Issues Classified as Accident Sequence Precursor Events (1990–1997)

A similar reduction in the number of ASP events can be seen when considering only LERs with DBIs. From 1992 to1997 there was an increasing trend in the number DBIs reported in LERs, but the number which were ASP events (CCDP of at least $1 \times 10^{-6}$) generally decreased during the period. In 1991, DBIs that were classified as ASP events peaked at approximately 8 percent of reported DBIs, by 1997, this percentage had dropped to approximately 0.6 percent (See Figure 22). The reduction in the percentage of DBIs that are classified as ASP events

may indicate that many of the most significant DBIs have been discovered through many years of design reviews and inspections. Caution should be used in extrapolating the line shown in Figure 22, since it is expected that DBIs will continue to be reported by licensees, and that some of them will be potentially safety significant.

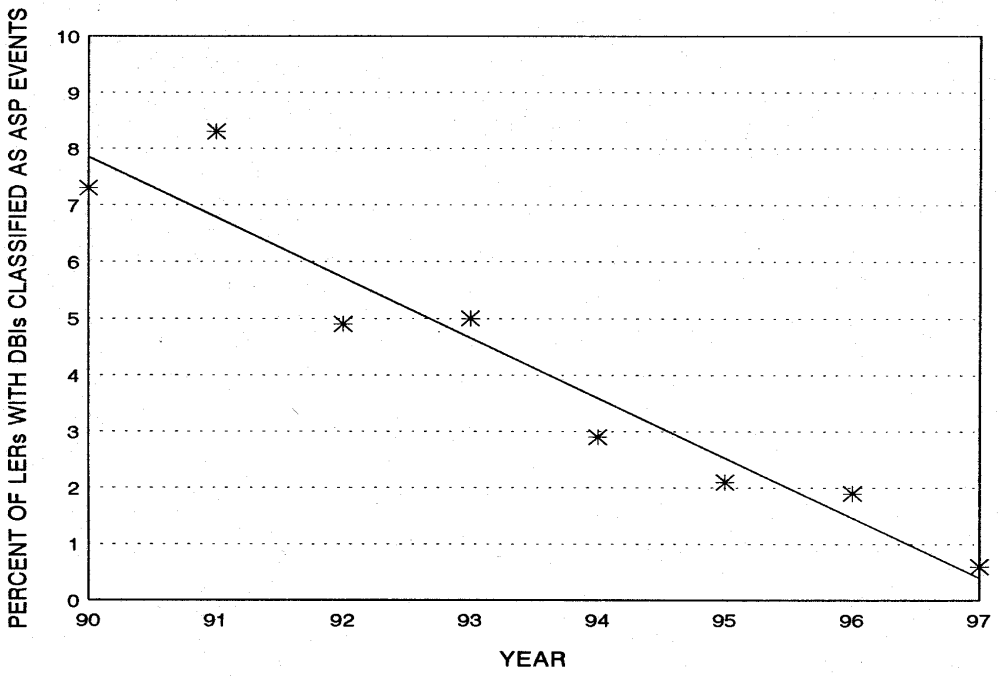



**Figure 22 Percent of licensee event reports with design-basis issues classified as accident sequence precursor events**

Accident Sequence Precursor Events Results (1997)

A search using the ORNL ASP database, showed that three (approximately 0.6 percent) of the 512 DBIs reported in 1997 had a CCDP of at least $1.0 \times 10^{-6}$. For all the LERs for 1997, the ASP computer search algorithm selected 797 for engineering review as potential precursors. Of these, 48 LERs (55 including revisions) were determined to be potentially significant. Of these 48, 31 LERs were rejected after detailed analysis, 2 LERs were determined to be impractical to analyze, and 8 LERs (5 events) were documented as “interesting” events. Review and analysis of the events described in the remaining seven LERs led to the identification of five (0.25 percent) of the 1975 total number of LERs as ASP events as shown in Table 4. All five ASP events involved older PWRs located in either Regions I or II.

**Table 4 Accident sequence precursor events (1997)**

| Plant | Event Date | LER Number | Description | Involved DBI | CCDP |
|---|---|---|---|---|---|
| Maine Yankee | 01/22/97 | 309/97-004 | Reactor coolant system loop fill header motor-operated valve overpressure | Yes | $1.3 \times 10^{-5}$ |
| Oconee 2 | 04/21/97 | 270/97-001 | Unisolable reactor coolant leak due to inadequate surveillance program | No | $2.2 \times 10^{-5}$ |
| Oconee 3 | 05/03/97 | 287/97-003 | High-pressure injection system inoperable due to design deficiency and improper work practices | Yes | $5.4 \times 10^{-6}$ |
| TMI 1 | 06/21/97 | 289/97-007 | Generator output breaker failure resulting in a loss of offsite power and reactor trip | No | $9.6 \times 10^{-6}$ |
| St. Lucie 1 | 11/02/97 | 335/97-011 | Nonconservative recirculation actuation signal set point | Yes | $3.4 \times 10^{-5}$ |

## 5.3 Trends and Patterns of Important Accident Sequence Precursor Events (1992–1997)

Between 1992 and 1997, PWRs had more “important” ASP events than BWRs. In the ASP program, an ASP event with CCDP greater than or equal to $1.0 \times 10^{-4}$ have traditionally been referred to as “important” ASP events. Table 5 lists the important ASP events occurring between 1992 and 1997. Observations include:

- PWR plants were more likely to have important ASP events than BWR plants. Of the 14 important ASP events shown in Table 5, 12 occurred at PWRs and 2 occurred at BWRs. All three of the important ASP involving DBIs occurred at PWRs.

- There has been a downward trend in the number of important ASP events during 1992–1997, with no important ASP events occurring during 1997.

- Between 1992-1997, a total of 1735 DBIs were reported, of which 4 (0.23 percent) were determined to be important ASP events.

**Table 5 Important accident sequence precursor events (1992-1997)**

| Plant | Event Date | Event | Description | Involved DBI | CCDP |
|---|---|---|---|---|---|
| Ft. Calhoun | 07/03/92 | 285/92-023 | Reactor Trip Due to Invertor Malfunction and Subsequent Pressurizer Safety Valve Leak | No | $2.5 \times 10^{-4}$ |
| Robinson 2 | 08/22/92 | 261/92-17 | Unusual Event Due to Loss of Off-Site Power and Reactor Trip | No | $2.1 \times 10^{-4}$ |
| Turkey Pt. 3, 4 | 08/24/92 | 250/92-S01* | Loss of Offsite Power Due to Hurricane Andrew | No | $1.6 \times 10^{-4}$ |
| Oconee 2 | 10/19/92 | 270/92-004 | Loss of Off-site Power and Unit Trip Due to Management Deficiencies, Less than Adequate Corrective Action Program | No | $2.1 \times 10^{-4}$ |
| Sequoyah 1, 2 | 12/31/92 | 327/92-27 | Reactor Trip as a Result of a Switchyard Power Circuit Breaker Fault and a Unit 2 Entry Into Limiting Condition for Operation [LCO] 3.0.3 when Both Centrifugal Charging Pumps were Removed from Service | No | $1.8 \times 10^{-4}$ |
| Catawba 1, 2 | 02/25/93 | 413/93-002 | Technical Specification 3.0.3 Entered Due to Inoperable Pump Discharge Valves | Yes | $1.5 \times 10^{-4}$ |
| Perry | 04/19/93 | 440/93-011 | Excessive Strainer Differential Pressure Across the residual heat removal (RHR) Suction Strainer Could Have Compromised Long Term Cooling During Post-LOCA Operation | No | $1.2 \times 10^{-4}$ |
| LaSalle 1 | 09/14/93 | 373/93-015 | Unit 1 Scram and Loss of Off-Site Power Due to Bus Duct Water Intrusion | No | $1.3 \times 10^{-4}$ |
| Haddam Neck | 02/16/94 | 213/94-004 | Automatic 480 Volt Bus Transfer Failure Due to Circuit Breaker Malfunction | No | $1.4 \times 10^{-4}$ |
| Wolf Creek | 09/17/94 | 482/94-018* | Reactor Coolant System Blows Down to Refueling Water Storage Tank During Hot Shutdown | No | $3.0 \times 10^{-3}$ |
| St. Lucie 1 | 08/02/95 | 335/95 -004, -005, -006 | Failed PORVs, Reactor Coolant Pump, Seal Failure, Relief Valve and Subsequent Shutdown Cooling System Unavailability, Plus Other Problems | Yes | $1.1 \times 10^{-4}$ |
| Wolf Creek | 01/30/96 | 482/96-001 | Loss of Circulating Water Due to Icing on Traveling Screens Causes Reactor Trip | No | $2.1 \times 10^{-4}$ |
| Catawba 2 | 02/06/96 | 414/96-001 | Loss of Off-Site Power Due to Electrical Component Failures | No | $2.1 \times 10^{-3}$ |
| Haddam Neck | 08/01/96 | 213/96-016 | Potential for Inadequate RHR Pump Net Positive Suction Head During Sump Recirculation | Yes | $1.1 \times 10^{-4}$ |

*Not an LER number

# 6 DESIGN BASIS ISSUE SAFETY SIGNIFICANCE ANALYSIS

Of the 512 DBIs reported in LERs for 1997, 410 were screened, characterized and ranked by potential risk significance. See Appendix A for a complete listing of all 512 LERs with DBIs in 1997. The remaining 102 LERs with DBIs for 1997 that were not included in the analysis consisted of either DBIs involving seismic or fire protection deficiencies. These LERs were excluded because significant uncertainties exist in the calculated risks using current risk assessment methods for these kinds of design issues. As discussed in the following section, the framework for screening and characterizing safety significance of the other 410 DBIs involved both risk insights and traditional engineering (i.e.,deterministic) insights.

## 6.1 Risk Characterization and Deterministic Significance Classification Framework

The risk category for each DBI was assessed on the basis of the “Phase 1” step process documented in SECY-99-007A, “Recommendations for Reactor Oversight Process Improvements,” Appendix A, “Process for Characterizing the Risk Significance of Inspection Findings” (Ref. 18). The description and effect of the DBI based on the narrative information documented in the LER was used. The LER abstracts and full narrative texts provided the information needed to identify the SSC equipment impacted by the DBI. Direct impacts and consequential impacts (at the level of SSC operability and functionality) were based on the information documented in the LER. (Licensees were not contacted to obtain additional information.)

The specific guidance for characterizing DBI risk significance was the generic risk information matrices (RIMs) tables for PWRs and BWRs documented in BNL letter report JCN W6234, “Development of Risk-Informed Baseline Inspection Program,” dated February 10, 1999 (Ref. 19) The RIMs tables document the systems, initiators and human actions that are important to the risk of BWRs and PWRs and the reasons for their importance. The generic risk insights used to develop the RIMs are based on the IPE insights reports (NUREG-1560), the IPE database (NUREG-1603), lessons learned from maintenance rule implementation and, insights from other sources such as shutdown PRAs.

Overall, the process is expected to identify more DBIs as potentially risk significant than would be characterized as such by a more detailed, plant-specific process. The conservatism results from the fact that the generic RIMs tables contain those systems and human actions that are important to most of the plants in the population, plus the items that are potentially important to only some of the plants in the population. Thus, applying all of the items in the generic RIMs tables to all the plants would be conservative for individual plants.

Finally, the phase I step of the significance determination process (SDP) provides a first step screening of inspection findings. Subsequent steps in the SDP result in the elimination of many potentially risk significant findings based on further more detailed risk assessment. These additional steps were beyond the scope of this study. It is expected that findings from the phase I step of the SDP which are subsequently determined to be risk significant will be a small fraction of the total.

For this study, DBI risk characterizations were blended with traditional deterministic significance classifications that are based on the concepts of defense-in-depth against certain design-basis events.

### 6.1.1 Risk Categories

The safety-significance characterization framework for screening and ranking the 410 DBIs involved three broad risk significance categories:

- "Potentially Risk Significant." Those DBIs which involved SSCs which met one or more of the reasons for importance documented in the RIM table.

- "Minimal Risk." Those DBIs which either involved an SSC which was not on the RIMs table, or involved an SSC on the RIMs table, but the effect of the DBI was not relevant to the reasons why the SSC was important.

- "No risk significance." Those DBIs which were programmatic (i.e., involved inadequate design basis analysis documentation and where the remedial actions only involved correcting or completing the design basis analysis).

Design-basis issues characterized as either potential or minimal were further classified and ranked by deterministic significance described in Section 6.1.2.

### 6.1.2 Deterministic Significance Classifications

The DBI deterministic significance classification scheme involved a taxonomy based on the effects of the DBI. The classification scheme considered three factors:

(1) The presence or absence of a safety demand for the SSC that was adversely effected by the DBI.

Higher classification (safety demand)

The higher classification of deterministic significance involved an occurrence of a safety demand that required proper functioning of the SSC. The DBI would be evaluated whether or not an initiating event occurred that caused an actual safety demand for the SSC that was adversely effected by the DBIs. The higher classification involved an occurrence of a safety demand for the SSC while the DBIs adversely impacted the ability of the SSC to successfully perform in response to the demand (e.g., a demand for high pressure injection at a time when the DBIs caused the high pressure injection pumps to be degraded or failed).

Lower classification (no safety demand)

The lower classification of deterministic significance involved no occurrence of a safety demand that required proper functioning of the SSC.

(2) Whether the DBI resulted in an actual or a potential failure of the adversely affected SSC.

The second element of the classification evaluated the type of effect of the DBI in terms of whether the DBI resulted in an actual or potential failure or degradation of the affected SSC.

Higher classification

The higher classification involved an actual failure or degradation in performance of the SSC due to the DBI. To be classified as an “actual” failure would require an actual operating performance deficiency to be present.

Lower classification

The lower classification involved a potential failure or degradation of the affected SSC for a given set of conditions.

(3) The extent to which the DBI failed or degraded the affected SSC.

The third element of the classification evaluated the extent to which the DBI affected the SSC in terms of whether the DBI involved a system failure, a system degradation or a train failure (or degradation). The highest classification of deterministic significance involved a failed system, the second highest classification involved a degraded system, and the lowest classification involved a degraded or failed train. For purposes of these classifications a degraded condition was defined as the SSC being capable of operating but judged as marginally failing to meet minimum design requirements. A failed condition was defined as the SSC having a gross or complete loss of design function.

### 6.1.3 Risk-Informed, Deterministic Significance Framework

Table 6 shows the risk category and deterministic significance classification combinations in decreasing safety significance order used for the study. Potentially risk significant DBIs were sorted by decreasing deterministic significance in the upper half of the table. DBIs of minimal risk or no risk significance were sorted by decreasing deterministic significance in the lower half of the table. Overall, the detailed framework involves 19 significance sorting “bins.” The resulting framework provides a qualitative relative ordering of DBI safety significance. As shown in Table 6, each of the 19 safety significance categories in the framework is assigned a unique alpha-numeric designation (e.g., 1a, 1b, 1c, 2a, 2b). These categories were used to construct a safety significance histogram for the 410 DBIs analyzed. Imbedded in Table 6 is a simplified framework which drops the “effect extent” element in the deterministic significance classification and was used for most of the DBI distribution analyses. The simplified framework reduces the number of safety significance categories from 19 to 7. Finally, DBIs that were categorized as potentially risk significant (i.e., safety significance categories 1, 2 and 3) were grouped together and designated as Group I, while DBIs that were determined to be either minimal risk or no risk significance (i.e., safety significance categories 4, 5, 6 and 7) were designated as Group II.

**Table 6 DBI Risk-informed, Deterministic Significance Framework**

| GROUP | DBI SAFETY SIGNIFICANCE CATEGORY | | DBI RISK CATEGORY | | | DBI DETERMINISTIC SIGNIFICANCE CLASSIFICATION | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | | Safety Demand | | Effect Type | | Effect Extent | | |
| | | | Potential | Minimal | None | Yes | No | Actual | Potential | Failed System | Degraded System | Degraded or Failed Train |
| I | 1 | a | x | | | x | | x | | x | | |
| | | b | x | | | x | | x | | | x | |
| | | c | x | | | x | | x | | | | x |
| | 2 | a | x | | | | x | x | | x | | |
| | | b | x | | | | x | x | | | x | |
| | | c | x | | | | x | x | | | | x |
| | 3 | a | x | | | | x | | x | x | | |
| | | b | x | | | | x | | x | | x | |
| | | c | x | | | | x | | x | | | x |
| II | 4 | a | | x | | x | | x | | x | | |
| | | b | | x | | x | | x | | | x | |
| | | c | | x | | x | | x | | | | x |
| | 5 | a | | x | | | x | x | | x | | |
| | | b | | x | | | x | x | | | x | |
| | | c | | x | | | x | x | | | | x |
| | 6 | a | | x | | | x | | x | x | | |
| | | b | | x | | | x | | x | | x | |
| | | c | | x | | | x | | x | | | x |
| | 7 | - | | | x | | | | | | | |

## 6.2 Overall Design-Basis Issue Safety Significance Analysis Results

The overall results of the detailed sorting of the 410 DBIs into the 19 DBI safety significance categories is shown in Figure 23. As seen in Figure 23, during 1997 there were no Group I DBIs involving a safety demand (Category 1) for an SSC whose function was adversely impacted by the DBIs. Consequently, there were no DBIs in the category of highest safety significance. As shown in the figure, the DBIs which were ASP events in 1997 were in either Category 2 or 3. Category 2 had 13 DBIs and Category 3 had 82 DBIs. Most of the DBIs (77percent) were in Group II (Categories 4, 5, 6, and 7) where their risk significance was classified as either minimal or none.

Group I (Potentially Risk Significant)

There were 95 Group I DBIs. That is, as noted above, about 23 percent (95 out of 410) of the DBIs in 1997 were judged to have a potential to be risk significant. Of the 95 Group I DBIs, about 14 percent (13 out of 95) were in Category 2 (i.e., actual failures or degradations). However, most of these (69 percent) involved degraded systems as opposed to failed systems or degraded or failed trains. Of the 95 Group I DBIs, about 86 percent (82 out of 95), were in Category 3 (i.e., potential failures or degradations). The DBIs in this category were dominated by the combination of system-level failures (50 percent) and system-level degradations (34 percent).

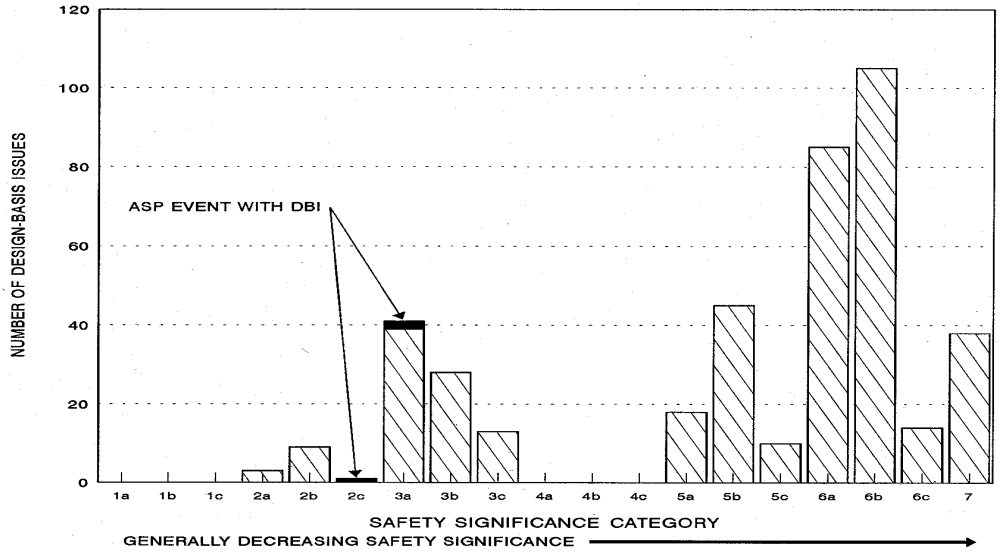



**Figure 23 Distribution of design-basis issues by safety significant category**

The distribution of DBIs across the safety significance categories of the simplified framework (i.e., Categories 2 and 3 in Group I; Categories 5 and 6/7 in Group II) are shown in Figure 24.

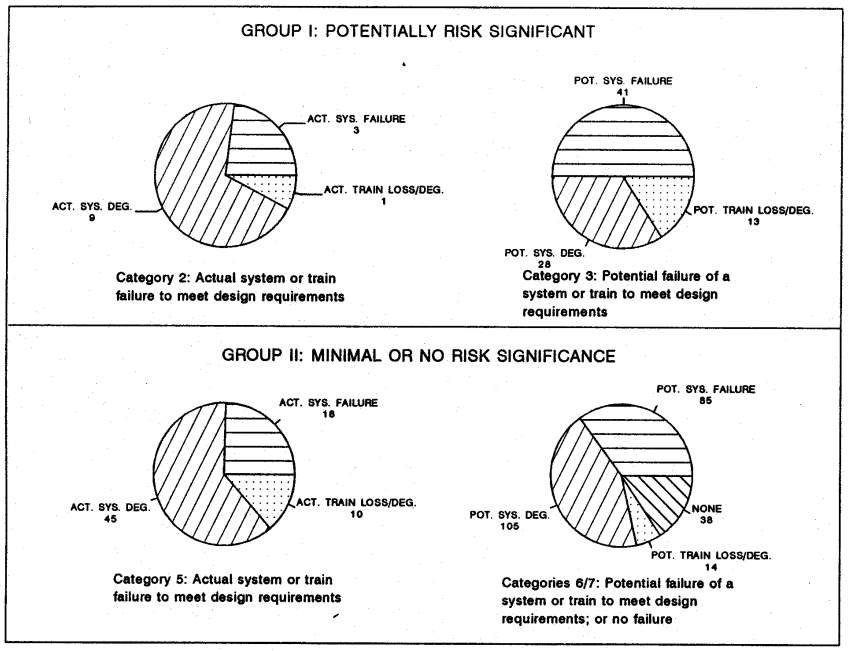



**Figure 24 Distribution of Group I and Group II design basis issues**

6.3 Distribution of Group I Design-Basis Issues

As noted in Section 6.2, there were 95 Group I DBIs. These DBIs were reported at 47 of the 110 licensed operating nuclear power plants. The distribution of number of plants vs the number of Group I DBIs for 1997 is shown in Figure 25. About 57 percent of the plants (63) had no Group I DBIs while about 26 percent had only one Group I DBI (29 plants). Crystal River 3 had the greatest number of Group I DBIs at 10.

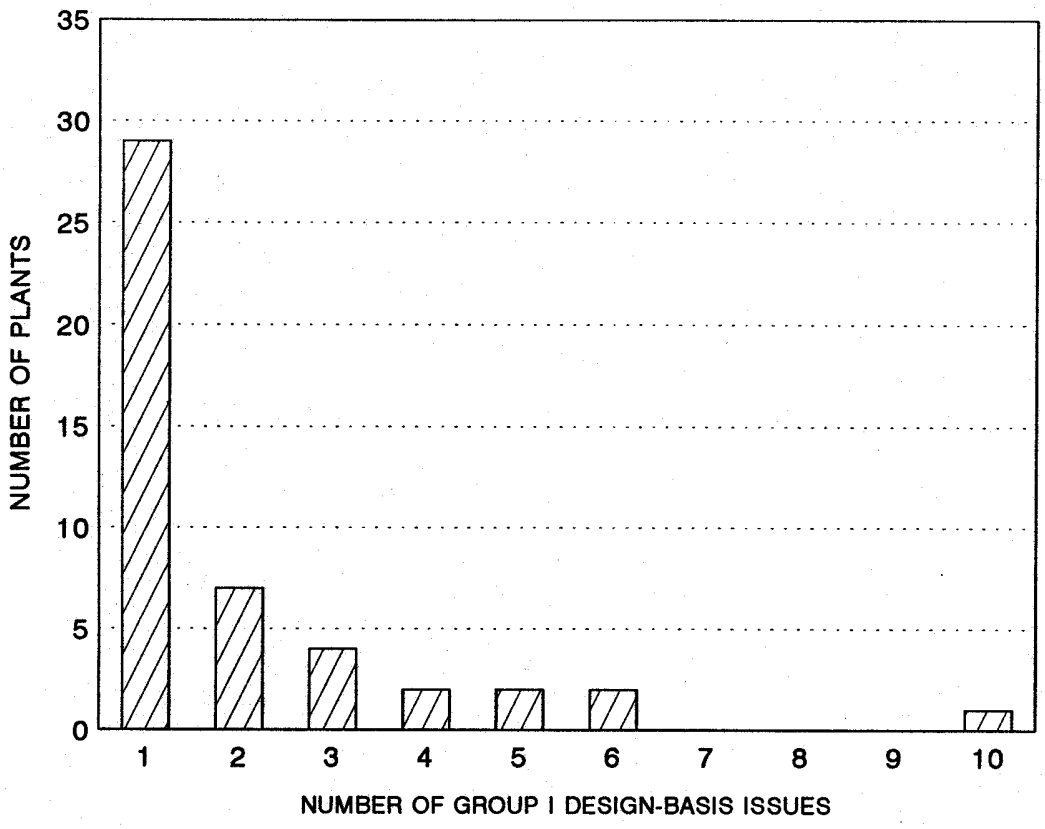
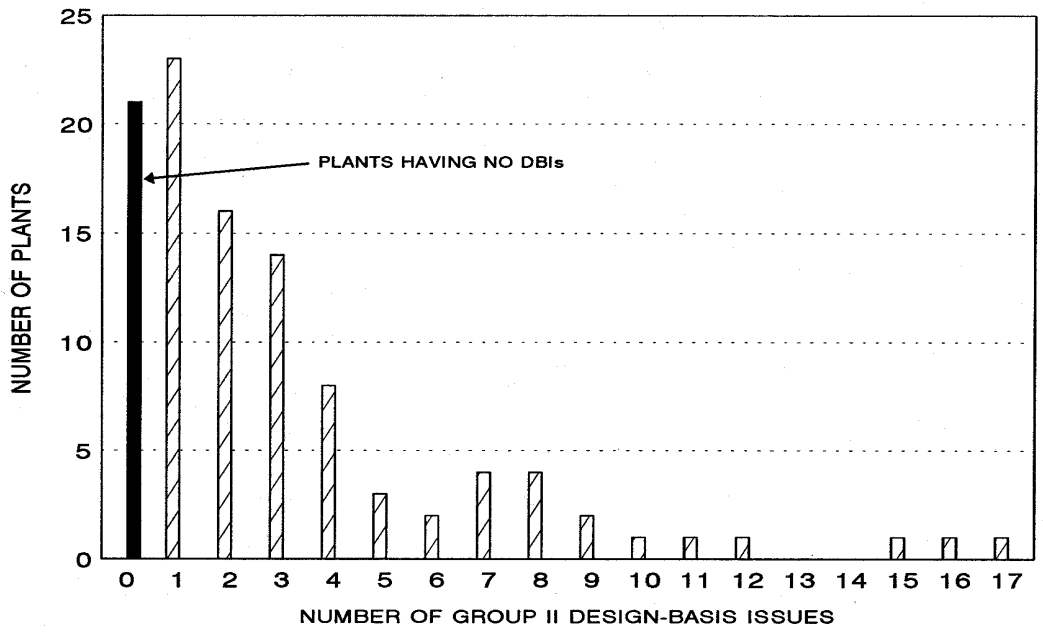



**Figure 25 Plant distribution of Group I design-basis issues**

6.4 Distribution of Group II Design-Basis Issues

Figure 26 shows the distribution of the number of plants with varying numbers of Group II DBIs. For 1997, out of the 410 DBIs analyzed, 315 DBIs were of minimal risk or no risk significance (Group II). These 315 Group II DBIs involved 82 different plants. A total of 21 plants had no DBIs during 1997. The most frequent number of Group II DBIs submitted by a plant was one, which occurred at 23 plants. However, in 1997, 16 plants submitted from 7 to 17 Group II DBIs involving minimal risk or no risk significance - an average of about 10 LERs with minimal or no risk significant DBIs per plant. As shown in Figure 26, a significant number of plants reported a relatively large number of LERs with DBIs having minimal or no risk significance, indicating that




the LER rule in effect at the time was not effective in discriminating (screening out) the reporting f DBIs having minimal or no risk significance.

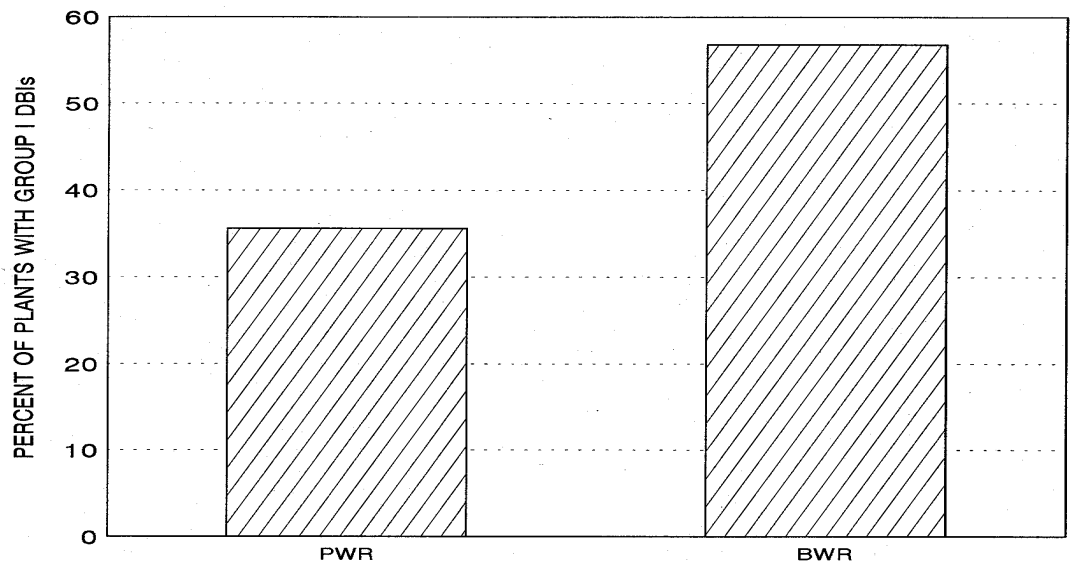
**Figure 26 Plant distribution Group II design-basis issues**

6.5 Distribution of Group I Design-Basis Issues by Plant Age

For 1997, as shown in Figure 27, “older” plants generally had a higher percentage of Group I DBIs than plants of “medium” age or the “newer” plants. This result is similar to the relative distribution shown in Figure 2 which shows that plants in Group C (operating license between 1985 and 1997) had much fewer DBIs per plant than plants in Group A (operating license between 1964 and 1974). For example, for 1997, the approximate percentage of plants with Group I DBIs for plants in plant age Groups A, B, and C was 57 percent, 46, percent and 19 percent. The apparent reasons for the difference included: generally lower quality, completeness and accessibility of plant design basis information at older plants. The impact of NRC engineering inspection in Regions I and III (which have more older plants) compared to Region II and IV plants was also considered a factor.

6.6 Group I Design-Basis Issues for Single and Multi-Unit Sites

As shown in Figure 28, for 1997, plants associated with multi-unit sites had a higher percentage of Group I DBIs than did single-unit sites. Shared systems was believed to be the major reason for the higher percentage of Group I DBIs for these sites. In addition, plant age was also a factor for both single and multi-unit sites, in that older plants, whether single or multi-unit had a greater percentage of Group I DBIs than did newer plants. The percentage of plants associated with single-unit sites that had at least one Group I DBI was 50, 33, and 17 for plant




age groups A, B, and C respectively, whereas, for multi-unit sites that had at least one Group I DBI, the percentage was 63, 50, and 21 for plant age groups A, B, and C respectively. Plant age group A represents older licensing dates (plant licensed before 1974), and group C represents the newer licensing dates (plants licensed after 1984). See Section 3.2.1 for plant age group definition.

**Figure 27 Group I design-basis issues by plant age**

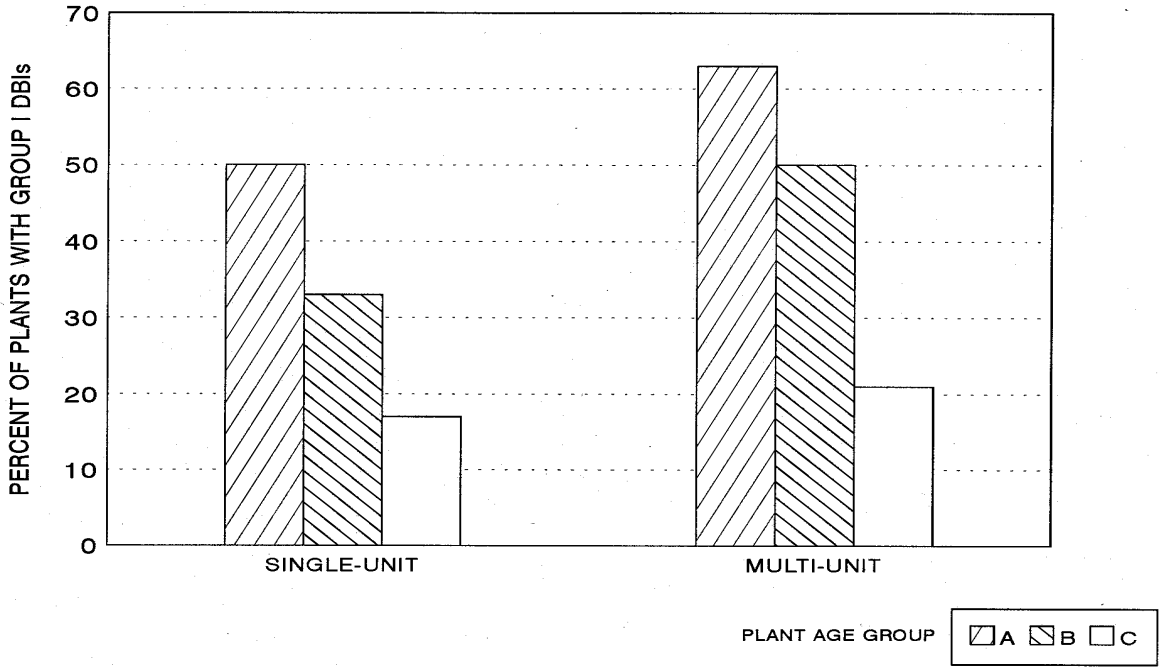



**Figure 28 Percentage of single-unit and multi-unit plants with Group I design-basis issues**

## 6.7 Group I Design-Basis Issues for Boiling Water Reactors and Pressurized Water Reactors

Boiling water reactors had a higher percentage of Group I DBI than PWRs as shown in Figure 29. For BWRs, about 57 percent had at least one Group I DBI, while for PWRs, about 36 percent had at least one Group I DBI. Although for 1997, BWRs had a higher percentage of Group I DBIs than PWRs, from an ASP perspective, all of the ASP events in 1997 involving DBIs occurred at PWRs.

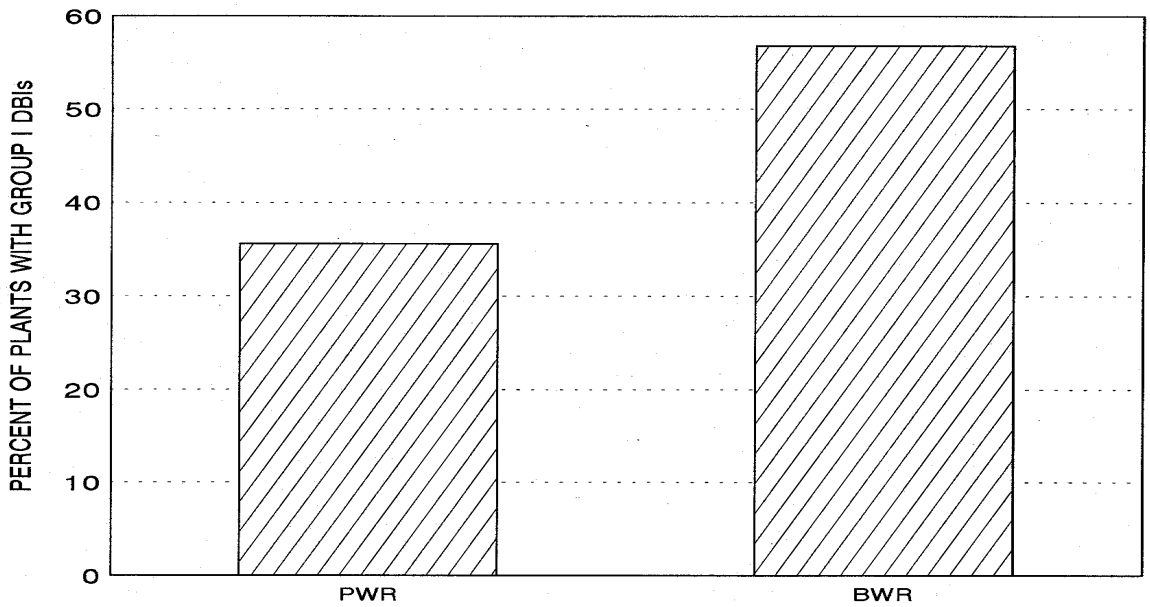



**Figure 29 Group I design-basis issues for boiling-water reactors and pressurized water reactors**

6.8 Group I and Group II Design-Basis Issues by Plant Systems

For 1997, each Group I DBI (95 total) and Group II DBI (315 total) was sorted into the generic plant systems listed in Appendix E*. For each Group I DBI, only the system that was determined to be important to risk as discussed in the LER was included, resulting in 95 DBI-systems affected. For Group II DBIs, up to two systems may have been associated with each DBIs, which resulted in 382 DBI-systems. The analysis results shown in Figure 30 indicate that relatively few systems disproportionally accounted for the Group I DBIs. Three plant systems accounted for approximately 64 percent of the Group I DBIs - ECCS (34 percent), and emergency ac/dc power (18 percent), and containment and containment isolation (12 percent). The auxiliary/emergency feedwater system, which is generally considered a very risk significant system, had a total of 21DBIs, however only three were classified as Group I. This may indicate that many of the risk significant DBIs for this system have been previously discovered and corrected through intensive licensee and NRC design reviews and inspections.

*Does not include fire protection related and seismic related DBIs (102 total) as discussed in Section 6.0.

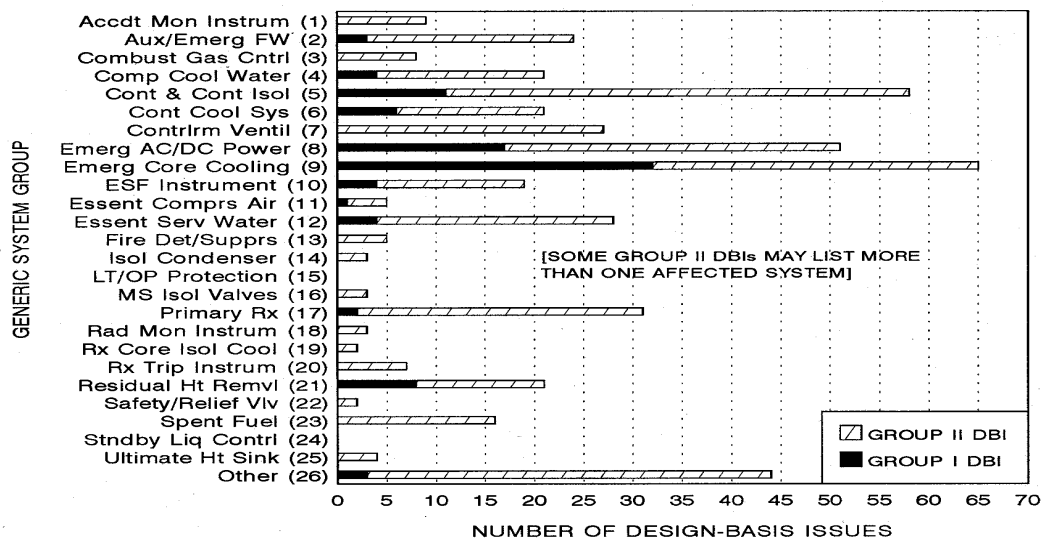



**Figure 30 Group I and Group II design-basis issues by plant system**

## 6.9 Group I Design-Basis Issues by Region

For 1997, plants in Regions I and III had reported the largest number of Group I DBIs (32 and 35 respectively), while plants in Regions II and IV reported the fewest number of Group I DBIs (22 and 6 respectively). Figure 31 shows the percentage of plants in each region that had at least one Group I DBI. As shown in the figure, Region III had the highest percentage of plants with at least one Group I DBIs (59 percent), followed by Region I (52 percent), Region II (36 percent) and Region IV (19 percent). Possible reasons for the lower incidence of Group I DBIs in Regions II and IV may involve the generally fewer engineering inspection hours, decreased regulatory attention, and a higher percentage of newer plants having better design documentation (discussed in Section 4). Plants having four or more Group I DBIs during 1997 are shown in Table 7.

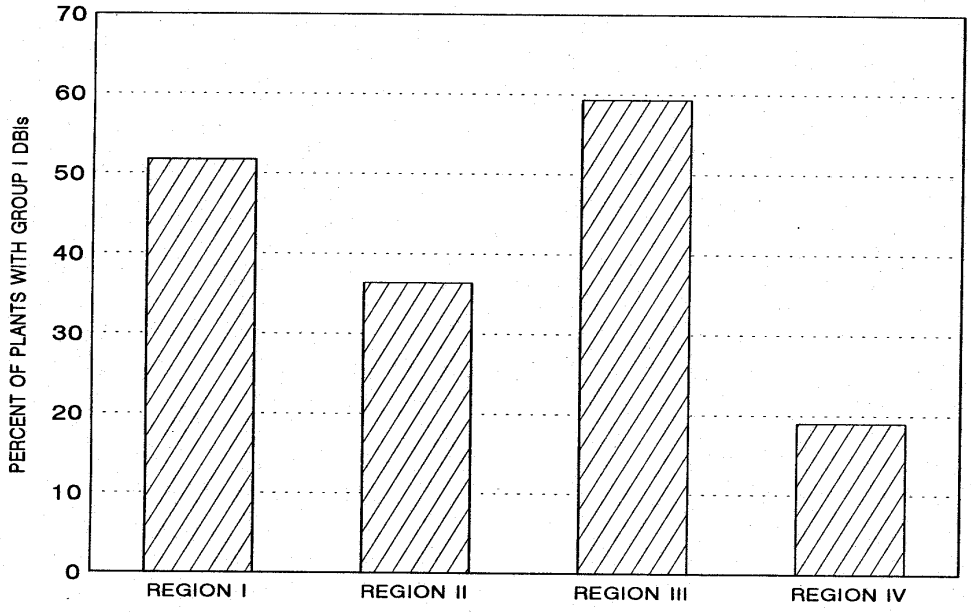



**Figure 31 Regional distribution of Group I design-basis issues**

**Table 7 Plants having four or more Group I design-basis issues (1997)**

| PLANT NAME | NUMBER OF GROUP I DBIs |
|---|---|
| Millstone 1 | 4 |
| Pilgrim | 4 |
| Crystal River 3 | 10 |
| Point Beach 1 | 5 |
| Point Beach 2 | 6 |
| D.C. Cook 1 | 5 |
| D.C. Cook 2 | 5 |

# 7 DESIGN-BASIS SCOPE, FUNCTION OPERABILITY, AND LICENSEE EVENT REPORTING REQUIREMENTS

There are a number of views within the NRC and the nuclear industry on how best to assess DBIs, including their associated risk, compliance with license requirements, and the timeliness and scope of needed corrective actions. Differences in plant designs, license requirements, code requirements, and licensee commitments add to the difficulty of assessing DBIs. Additionally, since some DBIs involved license requirements, in some cases licensees have not been in compliance with their operating license. However, very few of these DBIs have been risk significant from an ASP perspective.

Regardless of low risk significance (from an ASP perspective), the current regulatory principle is to ensure adherence to the design basis and that appropriate corrective actions are taken. For example, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-specific Changes to the Licensing Basis" (Ref. 20), provides guidance on the use of probabilistic risk assessment findings and risk insights in support of licensee requests for changes to a plant's licensing basis. Section 2.2.1 of the regulatory guide (Evaluation of Defense-in-Depth Attributes and Safety Margins) states

> One aspect of the engineering evaluations is to show that the fundamental safety principles on which the plant design was based are not compromised. Design-basis accidents play a central role in nuclear power plant design. Design-basis accidents are a combination of postulated challenges and failure events against which plants are designed to ensure adequate and safe plant response. During the design process, plant response and associated safety margins are evaluated using assumptions that are intended to be conservative. National standards and other consideration such as defense-in-depth attributes and the single failure criterion constitute additional engineering considerations that influence plant design and operation.

Current NRC guidance in defining operability and functional capability, resolving degraded or nonconforming conditions, and using risk assessment techniques in assessing DBIs are provided in Appendix C.

# 8 REGULATORY EFFECTIVENESS AND REGULATORY BURDEN INSIGHTS

On the basis of the observations discussed in the previous sections, a number of insights and potential lessons are evident with respect to NRC regulatory effectiveness and regulatory burden. These insights and lessons, involve (1) reporting of DBIs in LERs, (2) identification of DBIs by NRC inspection activities, (3) impact of identified DBIs on past NRC plant performance assessments, and (4) low risk significance of the identified DBIs which typically bear on the assumptions in the current design-basis scope. These regulatory effectiveness and regulatory burden insights and lessons are provided in the following paragraphs.

<u>Licensee Event Reporting Thresholds and DBI Risk Significance</u>

As discussed in Section 2.1, event reporting requirements and guidance for DBIs in paragraph 50.73(a)(2)(ii)(B) of 10 CFR requires that nuclear power plant licensees report any event or

condition that resulted in the condition of the nuclear power plant, including its principal safety barriers that is outside of the plant's design-basis.

However, as discussed in Section 6, for 1997, the majority of LERs with DBIs that were ranked for risk significance (77 percent) were found to be of minimal risk or no risk significance. The study found that 16 plants reported from 7 to 17 LERs with DBIs which were of minimal or no risk significance. Additionally, by 1997 approximately 0.6 percent of the LERs with DBIs reached the threshold of an ASP event. These results indicate that opportunities exist for reducing unnecessary regulatory burden with respect to reporting of DBIs.

The staff is continuing its efforts to examine 10 CFR 50.73 for the purpose of making the LER rule and supporting regulatory guidance more risk-informed. These efforts are being pursued in order to reduce the number of LERs that are submitted by licensees that are of low safety significance and thereby reduce unnecessary regulatory burden. In this regard, it would appear that there may be opportunities for reducing unnecessary regulatory burden by making 10 CFR 50.73(a)(2)(ii) more risk-informed.

NRC Inspection Findings Involving Design-Basis Issues

As discussed in Section 4, NRC engineering inspection teams and design inspection teams have been successful in identifying DBIs at nuclear power facilities (based largely on lack of compliance with license requirements that were initially established using deterministic analyses). When DBIs are identified as part of an NRC inspection they frequently result in the licensee submitting an LER for the DBI and the staff documenting the finding in an inspection report. Based on the risk categorization of LERs discussed in Section 6 and the ASP program insights, most DBIs discovered through the inspection process have been of relatively low risk significance. The staff is currently in the process of implementing a revised reactor oversight program. The new program incorporates a risk-informed inspection program component. For each inspection finding, the new inspection program involves the screening and characterization of the potential risk impact of specific inspection findings using the SDP. Accordingly, inspection reports in support of the revised reactor oversight program will document the new screening and characterization process results for DBI findings in a risk-informed, performance-based context.

The Effect of Increased Inspection Effort on Licensee Performance Assessment

As discussed in Section 4.6, a review of plant engineering assessments for plants issuing a high number of DBIs in 1997 found that plants often reported many additional DBIs in LERs after a more thorough engineering inspection for design compliance using contractor support. The increased number of DBIs often had the effect of lowering the NRC's assessment of licensee performance in subsequent years. The NRC's new reactor oversight process will also involve a baseline inspection for all plants, supplemented by additional inspection follow up for selected plants having performance assessment results which meet the threshold criteria. The additional findings from these supplemental inspections will also be documented in the Plant Issues Matrix (PIM). Although the inspection findings will be screened from a risk perspective, this additional inspection effort may result in additional risk important issues documented in the PIM when compared to the PIMs of plants that did not receive a supplemental inspection. Consequently, The additional findings might affect the assessment of plant performance and decisions to increased NRC oversight compared to comparable plants that did not receive

additional inspection effort. Thus, in connection with the staff's bench marking efforts for the new reactor oversight process, the staff may consider evaluating the general effect of supplemental inspection on (increasing) the total number of risk significant inspection findings in the PIM, the effect of this on plant evaluations, and its eventual effect on regulatory oversight decisions.

Design-Basis Scope

The staff has been evaluating the regulatory interpretation of the scope and content of plant design-basis information including the functionality assumptions related to SSCs. The staff has been developing guidance that provides a clearer understanding of what constitutes design-basis information, and involves a clarification of the terms design-basis functions and design-bases values as used in 10 CFR 50.2. Reducing the scope of the design basis would reduce unnecessary regulatory burden associated with the regulations framework and broader regulatory framework which references the plant design basis. Most DBIs involved a structure, system, or component that has traditionally been considered important to safety. However, as discussed in Section 6, the majority of identified DBIs reported in LERs are of minimal risk or no risk significance. That is, the majority of DBIs involve either SSCs which are not risk significant or design issues which do not impact the risk important accident sequences of risk significant SSCs. Thus, the results of this study would appear to provide useful risk insights and information for any future staff efforts aimed at re-evaluating the scope and content of a plant's design basis scope based on risk significance.

# 9 STATUS OF LICENSEE PROGRAMS AND PROCESSES TO CONTROL AND MAINTAIN DESIGN-BASIS INFORMATION

Based on the review of licensee responses to the 10 CFR 50.54(f) letter, the staff concluded in SECY-97-160 (Ref. 5) that while licensees had established programs and processes to maintain their facility's design bases, there was a need to implement plant-specific inspection followup activities. This determination was based upon the staff having identified: (1) instances in which licensees failed to reconcile regulatory performance with their assertions that their programs and processes were effective in maintaining their design bases, or (2) that there was a need to gain a better understanding or to validate a particular aspect of a licensee's programs and processes.

As part of this study, a sampling of the licensee 10 CFR 50.54(f) responses for 27 sites representing 42 of the operating plants were reviewed with the following results:

- Responses typically concerned design control methodologies.
- Certain responses indicated that design-basis reviews were for document retrievability only.
- One response indicated that the plant was in compliance with its design basis, but a subsequent 1997 regional inspection identified design-basis problems that resulted in a Severity Level III enforcement action.

- One licensee response included a list of 25 safety-significant systems (out of an identified 29 systems) for which design-basis verification actions were not complete, including areas such as design-basis accidents and events, primary containment, environmental qualification, separation and single failure, electrical design considerations, and equipment seismic requirements. Immediately before this plant's response, an NRC regional inspection report stated that it had raised concerns about an apparent gradual erosion of the plant's design and licensing bases.

- None of the licensees' responses to the 10 CFR 50.54(f) letters that were reviewed indicated that all plant safety systems had been reviewed, that there were no design problems, or that their design-basis reviews of safety-significant systems would be completed in the near future.

- NRC design inspections had subsequently found design-basis problems affecting plant safety after licensees stated they had reviewed their design basis.

- Certain licensees stated that they were collecting original design-basis data, but did not state that this data would be re-verified for completeness and accuracy.

The staff concluded (Ref. 5) that no further generic NRC action was required on this issue (the adequacy and availability of design-basis information), but that plant-specific followup actions might be warranted to verify certain features of licensee programs. As a part of these followup actions, the NRC conducted (or planned) design team inspections and other design-type inspections at selected sites. In this regard the staff planned to implement a change to the reactor inspection program to evaluate licensee design control programs and processes. At that time, the SSEI process was to provide an alternative method to assess a licensee's engineering effectiveness.

However, as noted earlier, the core inspection program is being replaced with a new risk-informed baseline inspection program. The baseline inspection program has a strong design element and assesses the risk significance of design issues. Additionally, the revised inspection program also will include a safety systems functional inspection capability. The new baseline inspection program will also emphasize licensee problem identification and resolution, including the resolution of identified DBIs.

## 10 DESIGN-BASIS ISSUES IN NRC GENERIC COMMUNICATIONS

Some risk-significant DBIs have taken many years to be identified in NRC generic communications. This is because the significant DBIs described in generic communications often are not self-evident but have periodically emerged and been fed back in generic correspondence over time with insights from operating experience, performance information, safety analyses and system analyses and reviews. The emergence of these insights enable industry and NRC personnel to recognize a significant new DBI (e.g., inadequate RHR pump NPSH during sump recirculation). Therefore, significant new DBIs continue to be identified despite several hundred reactor-years of operating experience. Additionally, significant and potentially generic DBIs identified in NRC generic communications, in some cases, take several years to be fully recognized by licensees as applicable to their plants and resolved on a plant-specific basis. For example, DBIs identified by NRC inspectors and licensees in the mid-1990s

at Millstone, Haddam Neck, Crystal River, and Maine Yankee typically had been cited previously in NRC generic communications. However, as the NRC's generic communications program and reactor inspection program become more risk-informed, it is expected that the timeliness and reliability of licensee corrective actions for risk-significant DBIs in NRC generic communications will improve. NRC generic communications related to DBIs covering the period 1987–1997 are listed in Appendix B.

Like the DBIs reviewed as part of this study, most DBIs described in generic communication documents such as INs, bulletins, and GLs existed since initial plant startup. For example, in May 1988, IN 88-28, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," was issued to address problems with both PWR containment sump strainer and BWR ECCS strainer clogging. NRC generic communication history on this general issue (excluding supplements and revisions) are shown in Table 8. This issue is still pending final resolution for PWRs, for which a 3 year study has recently been funded by the Office of Nuclear Regulatory Research as Generic Safety Issue 191. Actions are being taken by the staff to improve the generic communication processes (e.g., INs, bulletins, and GLs) and the generic issue program to be more timely, more risk-informed, and to reduce regulatory burden for areas of limited risk significance. A new pilot program to address generic issues began in 1999.

**Table 8 Generic communications on pressurized-water reactor containment sump strainer and boiling-water reactor emergency core cooling system strainer clogging**

| Date Issued | Information Notice/ Bulletin Number | Title |
|---|---|---|
| 05/88 | IN 88-28 | Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage |
| 11/89 | IN 89-77 | Debris in Containment Emergency Sumps and Incorrect Screen Configurations |
| 01/90 | IN 90-07 | New Information Regarding Insulation Materials Performance and Debris Blockage of PWR Containment Sumps |
| 09/92 | IN 92-71 | Partial Plugging of Suppression Pool Strainers at a Foreign BWR |
| 04/93 | IN 93-34 | Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment |
| 05/93 | IEB 93-02 | Debris Plugging of Emergency Core Cooling Suction Strainers |
| 10/95 | IEB 95-02 | Unexpected Clogging of a RHR Pump Strainer While Operating in Suppression Pool Cooling Mode |
| 10/95 | IN 95-47 | Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage |
| 05/96 | IEB 96-03 | Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors |
| 10/96 | IN 96-059 | Potential Degradation of Post Loss-of-Coolant Recirculation Capability as a Result of Debris |

| Date Issued | Information Notice/ Bulletin Number | Title |
|---|---|---|
| 05/97 | IN 97-027 | Effect of Incorrect Strainer Pressure Drop on Available Net Positive Suction Head |

## 11 OBSERVATIONS

### 11.1 The Most Common Causes of DBIs Were Original Design Error, Procedure Deficiency and Human Error

For 1997, the most common causes of DBIs were original design error, procedure deficiency, and human error, Licensees often cited multiple causes of DBIs. Of the 512 LERs with DBIs reported for 1997, the most frequent contributing causes included design errors dating back to the time of original plant licensing (72 percent), procedure deficiencies (27.5 percent), human error (22 percent), poor work control practices (15 percent), and plant modifications (13 percent).

### 11.2 There Was a Significant Variation Among Plants in the Number of Reported DBIs

For 1997, the average number of DBIs reported in LERs for the 110 nuclear plants was 4.7. However, 6 PWRs accounted for 30 percent of the reported DBIs: Crystal River 3 (37 DBIs), Point Beach 1 (27 DBIs), Point Beach 2 (24 DBIs), Millstone 3 (24 DBIs), D.C. Cook 2 (21 DBIs), and D.C. Cook 1 (18 DBIs). Additionally, during the period 1990–1997, 11 plants (9 PWRs and 2 BWRs) accounted for about 29 percent of the reported DBIs.

### 11.3 A Few Safety-related Systems Accounted for about Half of the DBIs

For 1997, 6 of the 26 safety-related plant system categories used for the study accounted for about half of the 512 reported DBIs. These systems were: emergency core cooling (13 percent), emergency ac/dc power (11 percent), containment and containment isolation (10 percent), primary reactor (7 percent), auxiliary/emergency feedwater (6 percent), and essential service water (6 percent).

### 11.4 A Small Fraction of the DBIs Were Classified as Group I (Potentially Risk Significant)

For 1997, of the 512 DBIs that were reported in LERs, 410 were categorized for risk. Of these, 95 DBIs (23 percent) were categorized as Group I. The majority (77 percent) were determined to involve either minimal risk or no risk significance.

### 11.5 Three Safety-Related Systems Accounted for a Majority of the Group I DBIs

For 1997, 3 of the 26 safety-related plant system categories used for the study accounted for about 64 percent of the 95 Group I DBIs. These systems were: emergency core cooling (34 percent) emergency ac/dc power (18 percent) and containment and containment isolation (12 percent).

## 11.6 Older Plants Generally Reported More DBIs than Newer Plants Reported

For 1997, newer plants (licensed after 1984) reported an average of about 3.1 DBIs while older plants (licensed before 1974) reported an average of about 5.6 DBIs. Additionally, about 19 percent of the newer plants had at least one DBIs categorized as Group I, while about 57 percent of the older plants had at least one DBIs categorized as Group I. The apparent reasons for the difference included: generally lower quality, level of completeness and accessibility of plant design basis information at older plants. The impact of NRC engineering inspection in Regions I and III compared to Region II and IV plants was also considered a factor.

## 11.7 Group I DBIs Were More Likely at Multi-Unit Sites than Single-Unit Sites

Plants associated with multi-unit sites had a higher percentage of Group I DBIs than did single-unit sites. Shared systems was believed to be the major reason for the higher percentage of Group I DBIs for these sites. In addition, plant age was also a factor for both single and multi-unit sites, in that older plants, whether single or multi-unit, had a greater percentage of Group I DBIs than did newer plants. The percentage of plants associated with single-unit sites that had at least one Group I DBI was 50, 33, and 17 for plant age groups A, B, and C respectively, whereas, for multi-unit sites that had at least one Group I DBI, the percent was 63, 50, and 21 for plant age groups A, B, and C respectively. Plant age group A represents older licensing dates (plants licensed before 1974), and group C represents the newer licensing dates (plants licensed after 1984).

## 11.8 During 1990–1997, the Percent of LERs with DBIs That Were ASP Events Steadily Decreased, While the Number of DBIs Increased

In 1991, approximately 8.3 percent of DBIs reported in LERs were accident sequence precursor (ASP) events (i.e., CCDP $\geq 10^{-6}$). From 1992 to 1997, there was an increasing trend in the number of DBIs reported in LERs but the number which were ASP events generally decreased during the period. By 1997, approximately 0.6 percent of the LERs with DBIs were classified as ASP events.

## 11.9 Between 1992–1997, All of the “Important” ASP Events Involving DBIs Occurred at PWRs

During this period, there were 14 important (CCDP $\geq 10^{-4}$) ASP events. Of the 14 important ASP events, 12 occurred at PWRs and 2 occurred at BWRs. Three of the 14 important ASP events involved DBIs, and all three occurred at PWRs.

## 11.10 Increases in the Number of Reported DBIs Coincided with NRC Initiatives

U.S. nuclear plants reported over 3100 LERs with DBIs during the period 1985–1997, or about 240 per year. The number varied from a low of 155 in 1985 to a high of 512 in 1997. Increases from previous years were observed in 1988 and 1989, and again in 1996 and 1997. The increases appeared to coincide with certain NRC initiatives including: NRC team inspections with a significant design element, NRC surveys of licensees on DBIs, licensee reviews in response to elevated NRC focus on DBIs, and NRC generic communications.

### 11.11 Group I DBIs Varied by NRC Region

For 1997, plants in Regions I and III reported the largest number (32 and 35 respectively) of Group I DBIs, while plants in Regions II and IV reported the fewest number of Group I DBIs (22 and 6 respectively). Region III plants also had the highest percentage of plants with at least one Group I DBI (59 percent), followed by Region I (52 percent), Region II (36 percent) and Region IV (19 percent). Possible reasons for the lower incidence of Group I DBIs in Regions II and IV may involve the generally fewer engineering inspection hours and the higher percentage of newer plants (i.e., better design basis documentation).

### 11.12 For 1995–1997, DBIs Appeared to Correlate with NRC Engineering Inspection Effort

For 1995–1997, as NRC engineering inspection hours increased, the number of reported DBIs generally increased. The increase was considered to be the result of NRC DBI inspection findings and increased licensee efforts to identify DBIs as a result of NRC inspections and NRC generic communications on DBIs. Fourteen of the 21 plants that reported no DBIs during 1997, received less than the median number of engineering inspection hours.

### 11.13 Licensee Engineering Performance Ratings Often Declined with Increased Engineering Inspection Effort

For 1997, a correlation between engineering inspection hours and the number of DBIs was identified. If a plant had a thorough engineering inspection for design compliance, it often reported more DBIs. This often resulted in the plant receiving a lower plant engineering rating in the subsequent assessment period. In some instances, the lower assessment rating led to increased regional or agency oversight.

### 11.14 The Importance and Applicability of DBIs Discussed in NRC Generic Communications Occasionally Takes Several Years for Licensees to Recognize and Address

Some risk-significant DBIs have taken many years to identify in NRC generic communications. NRC and industry awareness of significant and potentially generic DBIs have emerged from insights drawn from operating experience, performance information, safety analyses, and system analyses and reviews. Most DBIs described in generic communication documents such as NRC information notices, bulletins, and generic letters existed since initial plant startup.

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