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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 1, 1993

MEMORANDUM FOR: Commissioner Rogers

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: LERS REPORTING PROBLEMS DISCOVERED DURING DESIGN BASIS REVIEWS

Your November 10, 1992 memorandum requested that the Office for Analysis and Evaluation of Operational Data (AEOD) perform further analysis on the data regarding licensee event reports (LERs) which report problems discovered by licensees during design basis reviews (DBRs). You suggested that AEOD consider the following questions:

(1) Which plants perform design basis reviews each year and how many LERs satisfying the selection criteria come from each of those plants?

(2) Which plants, if any, doing design basis reviews fail to find LERs satisfying the selection criteria?

In an effort to locate sources of data regarding which plants have established design basis review programs, AEOD has reviewed the status of the staff's work on the Design Document Reconstitution Program with cognizant NRR staff. As a result of these discussions, AEOD has concluded that up-to-date, validated data required to adequately answer your questions is not currently available, nor will any useful information be available on a schedule compatible with that for the 1992 AEOD Annual Report. They would need to obtain OMB clearance to ascertain the necessary information. Without such data, the results of any analysis of LERs that address which plants have design basis review programs would heavily rely on informal survey information, and thus would be questionable for answering your request.

Consequently, AEOD does not have reliable information regarding which plants have a formal, structured design basis review process in place and which plants perform a periodic update of these reviews. However, as an alternative, AEOD has addressed what they believe to be the intent of your request - to determine the usefulness/effectiveness of design basis reviews. In order to accomplish this objective, they assumed that the usefulness/effectiveness of DBRs are related to the number of "Significant" problems discovered during DBRs and reported in LERs. The details of AEOD's analysis are provided as an enclosure and summarized below.

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To determine the number of significant DBR LERs, AEOD screened all of the approximately 14,000 LERs submitted during the 5½-year period 1987 through mid-1992. This screening identified about 2,000 candidate LERs which reported design problems. Their review of these 2,000 candidate LERs determined that about 800 involved design problems which were discovered during a DBR. The number of DBR LERs per station ranged from 1 to 33, with at least one DBR LER submitted by each station during the period of interest. The median number of DBR LERs submitted per station was 9.

In order to evaluate the significance of the design problems discovered by DBRs, AEOD grouped the 800 DBR LERs according to the number of DBR LERs submitted per station during the 5½-year period. To reduce the influence of individual reporting practices on the review results, their detailed review concentrated on the 22 stations which were in the two highest reporting categories, "Extensive" (25 or more DBR LERs - 9 stations), and "Above Median" (13-24 DBR LERs - 13 stations).

Each of the LERs reviewed was subjectively characterized into two categories. These categories are intended to represent different levels of significance. AEOD arbitrarily labeled them as "high significance" and "lesser significance." For stations in the "Above Median" reporting category, 28% of the DBR LERs were assessed as having "higher significance." For stations in the "Extensive" reporting category, 36% of the DBR LERs were assessed as having "higher significance." The types of problems identified with the higher significance level DBR LERs can be characterized as follows: (1) containment integrity or containment isolation function adversely affected (20%), (2) code allowables in piping stress, overpressurization, pipe support loading, etc., were exceeded such that modifications were necessary (10%), and (3) the balance of the LERs were fairly uniformly distributed over a variety of causes. A complete characterization of all of the "higher significance" and "lesser significance" level problems reported in DBR LERs is given in Table I of the Enclosure.

The results of the analysis are shown in Table II of the Enclosure and indicate that the percentage of "higher significance" level DBR LERs for the stations in the "Extensive" and the "Above Median" reporting categories ranged from 12% to 48% of the total number of DBR LERs per station, with an average of 32%. In addition, only three plants out of these two groups had less than 20% of the DBR LERs assessed as "higher significance." The results of a detailed review and characterization of 30 DBR LERs from three stations reporting 25 or more DBR LERs are provided in Table III of the Enclosure.

To determine if a correlation existed between plant age and DBR programs, the plants were grouped by plant age into three groupings. The grouping was done in such a manner that each group contained about the same number of plants. The grouping was based on commercial operation dates: Group A - 1974 and earlier, Group B - 1975 through 1983, Group C - 1984 and after. The plants in Group A had 40% of the DBR LERs; Group B, 34% of the DBR LERs; and Group C, 26% of the DBR LERs. The older plant categories also had larger percentages of DBR LERs assessed as "higher significance."

Because of the large percentage of "higher significance" DBR LERs from plants with an assumed DBR program ("Extensive" category or "Above Median" category DBR LER reporting), AEOD has concluded that design basis review programs are useful and effective. This conclusion pertains only to those plants with above median reporting. They were unable to reach any conclusions regarding the effectiveness and usefulness of DBRs for plants with little reporting. They also found that the number of plants with DBR programs is related to the plant age grouping.

A summary of the results will be provided in the AEOD Annual Report.

Further details of AEOD's analysis are discussed in the enclosure. AEOD believes that their response meets the intent of your request. However, if you have any additional questions regarding the analysis, please contact Edward L. Jordan on 492-4848.

James H. Sniezek /for

James M. Taylor
Executive Director
for Operations

Enclosure:
Usefulness and Effectiveness
of Design Basis Reviews

cc: The Chairman
Commissioner Curtiss
Commissioner Remick
Commissioner de Planque
SECY
OGC
OCA
OPA

ENCLOSURE

USEFULNESS AND EFFECTIVENESS OF DESIGN BASIS REVIEWS

James R. Houghton
Division of Safety Programs
Office for Analysis and Evaluation of Operational Data

In response to Commissioner Rogers' November 10, 1992 request for an analysis of LERs reporting problems discovered during design basis reviews (DBRs), and specifically for determining the usefulness and effectiveness of DBRs, the Office for Analysis and Evaluation of Operational Data (AEOD) reviewed more than 2000 candidate abstracts of LERs issued by licensees of operating plants for the period 1987 through mid-1992.

After detailed review of each of the candidate LERs, approximately 800 DBR LERs were identified as a result of a formal or informal design basis review/reconstitution program. The number of DBR LERs by station ranged from 1 to 33. The frequency characterization of these DBR LERs by station is shown in Figure 1. The total number of DBR LERs for the lead/oldest unit was used as the total for the station.

To determine the influence of plant age, Plant Age Groups were established that resulted in approximately one-third of the stations being contained in each of the three groups: Group A, Group B, and Group C. This grouping was based on commercial operation dates: 1974 and earlier, 1975 through 1983, and 1984 and later for Group A, Group B, and Group C, respectively. The groupings were made in this manner only to keep the number of plants in each group constant.

Categories were developed to describe the frequency of reporting of DBR LERs. The median number of DBR LERs occurring per plant was 9. The case where a station had a number of DBR LERs greater than 12 was considered to be indicative of a detailed formal or informal Design Basis Review/Design Basis Reconstitution Program. The DBR LER frequency categories are as follows:

BELOW MEDIAN (BM):	5 or less
MEDIAN RANGE (MR):	6 - 12
ABOVE MEDIAN (AM):	13 - 24
EXTENSIVE (E):	25 or more

Determining Usefulness and Effectiveness of Design Basis Review Programs

Stations were selected from those which were categorized as "Above Median" (AM) and "Extensive" (E) to assure that an adequate base of DBR LERs would be used to evaluate usefulness and effectiveness of the licensees' programs. A detailed review of LER abstracts was performed for all stations within these two categories. Each of the DBR LERs was assumed as having a level of "higher significance" or "lesser significance." In addition, a more detailed characterization of the "higher significance" DBR LERs were provided for two sample stations in Group A and one sample station from Group B Plant Age groups. Each sample station had 25 or more DBR LERs.

Generally, most of the DBR LERs reported one or more of the following: (1) an "Unanalyzed Condition" in accordance with 10 CFR 50.73 (a)(2)(ii), (2) Design Error or Inadequacy, or (3) a situation where a plant was outside its design (including licensing) basis. Therefore, the majority of these DBR LERs have some minimum level of significance. However, to be conservative, "Usefulness" was equated to a "higher significance" level versus those DBR LERs which had a "lesser significance" level. The assumption used in this determination was as follows:

> ***The usefulness and effectiveness of design basis reviews (DBRs) is related to the number of DBR LERs with a "higher significance" level discovered during DBRs and reported in LERs.***

The DBR LERs from the three sample stations were reviewed and the number and percentage of higher significance level DBR LERs determined. For each of these higher significance level DBR LERs the criteria violated and characterization of the significance were documented. DBR LERs for the remainder of the stations in the "AM" and "E" categories were also reviewed for significance level using the same criteria employed in the review for the three sample stations, with the number and percentage of higher significance level DBR LERs tabulated, but further details were not developed to characterize each reported event.

General characteristics of the "higher significance" level DBR LERs and "lesser significance" level DBR LERs used as guidance in the evaluation of all DBR LERs in the "AM" and "E" category stations are listed in **Table I**.

RESULTS

A frequency characterization of the DBR LERs by station is shown in **Figure 1**.

Figure 2 shows the distribution of DBR LERs by Plant Age Group. The older plant age grouping, Group A, had 40% of the total number of DBR LERs; while the newer plant age, Group C, had only 26% of the total number of DBR LERs.

Figure 3 contains multiple charts showing the distribution of the frequency categories of DBR LERs for "All" stations and for each plant age group. The distribution of all frequency categories for the Group A and Group B Plants are similar. For the Group C Plants, the "Below Median" frequency category distribution is much higher (46%), while the "Above Median" and "Extensive" frequency category distributions are significantly lower.

Table II provides the percentage of higher significance level DBR LERs for each of the stations with **"AM"** and **"E"** DBR LER frequency categories. Also included in Table II are the Plant Age Groups, total number of DBR LERs for each station, and the number of higher significance level DBR LERs for each station. For **"AM"** category stations, 28% (average) of the DBR LERs were evaluated as having a higher significance level; while the **"E"** category stations averaged 36%, a somewhat higher distribution percentage. The overall average for the two catigories was 32%.

Table III characterizes the results of the detailed review of DBR LERs with

and Crystal River. The first two are Plant Age Group A stations; while the third is a Plant Age Group B station. Ten higher significance level DBR LERs were identified for each sample station, which is approximately 30% of the total DBR LERs for that station (also listed in Table II).

The "Higher Significance" level DBR LER characteristics listed in Table I were distributed as follows among the 30 LERs listed in Table III:

- Containment integrity or containment isolation function adversely affected (20%).
- Exceeded Code allowables in piping stress, overpressurization, pipe support loading, etc., such that modifications were necessary (10%).
- The balance of the LERs (70%) were distributed in an approximately uniform manner.

CONCLUSION

Because of the large percentage of "higher significance" level DBR LERs from plants with an assumed DBR program ("Extensive" or "Above Median" category DBR LER reporting), we have concluded that design basis review programs are useful and effective. This conclusion pertains only to those plants with above median reporting. We were unable to reach any conclusions regarding the effectiveness and usefulness for plants with little reporting. We also found that the number of plants with DBR programs is related to the plant age grouping.

TABLE I

SIGNIFICANCE LEVEL CHARACTERISTICS
FOR DBR LER'S

Higher Significance Level DBR LER Characteristics

- Containment integrity or containment isolation function adversely affected.
- Deficiencies in high energy line break (HELB) analysis which could result in loss of system function.
- Exceeded Code allowables in piping stress, overpressurization, pipe support loading, etc., such that modifications were necessary.
- Degraded voltage conditions on Safeguards Equipment.
- Electrical faults which can cause common cause failure to redundant trains in safety-related systems.
- Problem could lead to exceeding 10 CFR Part 100 offsite dose limits.
- Containment heat removal or decay heat analyses adversely affected.
- Inadequate isolation of non-safety fluid systems which interface with safety-related fluid systems.
- Exceeding maximum HVAC temperature limits for Emergency Diesel Generator rooms.
- Seismic analysis deficiencies, where plant is located in a higher seismic area and is a more significant factor in plant PRA.
- Inadequate isolation of non-1E circuits from interfacing 1E circuits.
- Common cause or generic concerns affecting multiple trains in one safety-related system or multiple safety-related systems.
- Single failure criteria in safety-related systems, other than HVAC.
- Problem directly affecting FSAR Chapter 15 Accident Analysis.

Lesser Significance Level DBR LER Characteristics

- Seismic analysis deficiencies, where plant is located in a lower seismic zone and is a less significant factor in plant PRA.
- Loss of backup or one train of redundant train system of safety-related systems, including instrument channels.
- Inadequate equipment qualification, but where degradation is over a prolonged period (excluding emergency diesel generator room maximum temperature).
- Problems involving HVAC systems, except where used for containment heat removal functions.
- Inadequate tank storage capacities, but where replenishment may be provided within a reasonable time period (i.e., Fuel Oil Storage Tank, 7 day storage).
- Appendix R/Fire Protection deficiencies.
- Analysis inadequately addresses tornadic winds and wind pressures.
- Excessive piping stresses, but which do not result in deformation or rupture or loss of safety function.
- Loss of normal 1E electrical system, but with emergency diesel generators still available.
- Problems with electrical cable separation (except where containment isolation or both trains of the same safety-related system are affected).
- Problems involving fuse and circuit breaker coordination.

TABLE II
HIGHER SIGNIFICANCE LEVEL DBR LERS
FOR STATIONS WITH DBR LER DISTRIBUTION CATEGORIES "E" AND "AM"

ITEM NO.	STATION NAME	DKT NO.	PLANT AGE GROUP	DBR LER CAT.	NO. DBR LERs	NO./PERC. HIGHER SIGNIF.
1.	San Onofre	50-206	A	E	25	10/40%
2.	Haddam Neck	50-213	A	E	26	12/46%
3.	Millstone	50-245	A	AM	22	6/27%
4.	Turkey Pt.	50-250	A	AM	15	5/33%
5.	Quad Cities	50-254	A	AM	17	2/12%
6.	Palisades	50-255	A	E	27	10/37%
7.	Browns Ferry	50-259	B	AM	18	5/28%
8.	Robinson	50-261	A	AM	21	11/52%
9.	Oconee	50-269	A	E	25	12/48%
10.	Surry	50-280	A	AM	13	4/31%
11.	Ft. Calhoun	50-285	A	E	33	10/30%
12.	Crystal River	50-302	B	E	32	10/31%
13.	ANO	50-313	B	E	28	6/27%
14.	Calvert Cliffs	50-317	B	AM	19	3/16%
15.	Brunswick	50-325	B	AM	13	2/15%
16.	Sequoyah	50-327	B	E	32	13/41%
17.	FitzPatrick	50-333	B	AM	18	5/28%
18.	Trojan	50-344	B	AM	15	3/20%
19.	Limerick	50-352	C	AM	22	7/32%
20.	WNP	50-397	C	E	33	11/33%
21.	Millstone	50-423	C	AM	13	5/38%
22.	Palo Verde	50-528	C	AM	13	3/23%

NOTES:

1. See Table III for more detailed review of Stations 6, 11, and 12.

2. Average percentage of Higher Significance level DBR LERs for "AM" and "E" category stations is as follows:

   "AM" = 61/219 = 28%

   "E" = 94/261 = 36%

TABLE III
DETAILED REVIEW OF DESIGN BASIS REVIEW
LICENSEE EVENT REPORTS EVALUATED WITH
HIGHER SIGNIFICANCE LEVEL FOR SELECTED SAMPLE NUCLEAR POWER STATIONS
WITH EXTENSIVE NUMBER OF DBR LERS

A. FORT CALHOUN - PWR/Plant Age Group A Station

Total No. DBR LERs: 33; No. of Higher Significance level DBR LERs: 10

Higher Significance level DBR LERs are detailed as follows:

1. 285/88-004, Rev. 0, Event Date: 03/11/88

Criteria Violated/LER Reportability: Containment Isolation criteria specified in the Updated Safety Analysis Report (USAR). (Unanalyzed Condition)

Characterization: During a LOCA a potential leakage path through isolation valve PCV-1849 for the instrument air (IA) containment penetration M-73 could exist if IA system pressure is not maintained above Containment pressure. PCV-1849 is designed to close on receipt of low IA system pressure in conjunction with a Containment Isolation Actuation Signal. However, PCV-1849 has an actuator which would allow the valve to open on loss of air. Therefore, containment integrity cannot be assured during a LOCA with concurrent Loss of Offsite Power (LOOP).

2. 285/88-009, Rev. 0, Event Date: 04/06/88

Criteria Violated/LER Reportability: failure of isolation valves to perform their safety function. (Unanalyzed Condition)

Characterization: A self-conducted SSFI identified concerns with the capability of certain isolation valves to perform their design function during a design basis accident with concurrent loss of instrument air. these isolation valves of concern are in the safety injection, reactor coolant, and component cooling water systems.

3. 285/89-007, Rev. 0, Event Date: 03/24/89

Criteria Violated/LER Reportability: Safety-related piping design shall be to piping Code and shall address postulated high energy line breaks. (Unanalyzed Condition)

Characterization: Engineering analysis associated with feedwater system modification MR-81-154 did not adequately address postulated high energy line breaks (HELBs) or resultant pipe stresses. Reevaluation of this 1983 modification (replacing feedwater regulating valves) revealed that maximum allowable ANSI B31.1 Code stresses and HELB thresholds had been exceeded, thereby requiring new breaks to be postulated and analyzed.

TABLE III (Continued)

4. 285/90-003, Rev. 1, Event Date: 02/16/90

Criteria Violated/LER Reportability: Piping shall be designed such that stresses are within those allowed by the design basis codes and standards. (Unanalyzed Condition)

Characterization: Reanalysis of the AFW and SGB systems identified that rigid seismic restraints, installed to control seismic inertia, excessively restricted the thermal movement of the lines and caused stresses to exceed design basis piping code allowables. A revised seismic analysis of main steam and safety injection piping inside the containment, including a review of support loads and capacities, determined that several supports were loaded beyond their design capacity.

5. 285/90-007, Rev. 2, Event Date: 02/28/90

Criteria Violated/LER Reportability: Piping that is designated as Seismic Category I shall be designed such that piping supports would not be overloaded during a seismic event. (Unanalyzed Condition)

Characterization: The non-safety related portion of the feedwater piping (classified as Seismic Class I), and both the non-safety related and safety-related main steam piping were determined to be overloaded during a seismic event. An evaluation was performed to determine the effect that overloading these supports would have on these systems, assuming gross failure of the overloaded supports. The evaluation determined that the design basis support and piping allowables would be exceeded, and stresses in the feedwater piping would result in a previously unanalyzed high energy line break.

6. 285/90-009, Rev. 0, Event Date: 03/16/90

Criteria Violated/LER Reportability: Piping shall be protected from overpressurization that would violate the piping Code and prevent systems from performing their safety function. (Unanalyzed Condition)

Characterization: An analysis of the auxiliary feedwater (AFW) system piping between the containment isolation valves has shown that, in the event of a main steam line break (MSLB) or a LOCA inside the containment, the piping would be overpressurized due to thermal expansion of the process fluid between the closed valves. The failure of piping would result in the inability of the AFW system to provide coolant to the intact steam generator.

TABLE III (Continued)

7. <u>285/91-004, Rev. 0, Event Date: 02/12/91</u>

<u>Criteria Violated/LER Reportability:</u> The safeguards equipment shall be designed for protection from degraded voltage conditions. (Unanalyzed Condition)

<u>Characterization:</u> The offsite power low signal (OPLS) provides degraded voltage protection to safeguards equipment when a degraded voltage condition exists concurrent with a safety injection actuation signal. Engineering analysis revealed that, during a postulated accident, the voltage supplied to some 480 V safeguards loads could degrade to as low as approximately 87.5% of rated voltage without OPLS being actuated. Since the possibility existed for voltage to be lower than the recommended 90% of rated voltage for certain 480 V safeguards loads, the plant was considered to be outside its design basis.

8. <u>285/91-025, Rev. 0, Event Date: 11/14/91</u>

<u>Criteria Violated/LER Reportability:</u> The upset and faulted loadings on safety-related system pipe supports shall not exceed the design capacity of the embedments to which they are attached. (Unanalyzed Condition)

<u>Characterization:</u> The upset and faulted loadings on two safety injection system pipe supports exceeded their design capacity of the embedded unistrut to which they are attached. The primary cause of this condition outside the design basis is attributed to design deficiency.

9. <u>285/92-011, Rev. 0, Event Date: 03/20/92</u>

<u>Criteria Violated/LER Reportability:</u> Containment Isolation criteria specified in the Updated Safety Analysis Report (USAR), in general, require two containment isolation valves. (Unanalyzed Condition)

<u>Characterization:</u> It was determined that the valve arrangement for service air containment penetration M-74 did not meet isolation criteria required for a containment atmosphere-exposed system. Valve HCV-1749 (a normally closed/fail closed air-operated valve) is the only isolation valve for penetration M-74. Because the air compressors are not automatically sequenced to the emergency diesel generators, the condition of "line pressure is greater than Containment design pressure at all normal and postulated accident conditions" is not met.

TABLE III (Continued)

10. 285/92-022, Rev. 0, Event Date: 07/02/92

Criteria Violated/LER Reportability: The electrical system shall be designed such that cables are adequately sized for faulted conditions. (Unanalyzed Condition)

Characterization: During reconstitution of the electrical system design basis, it was discovered that the cables supplying 4160 volt power to three heater drain pump motors were inadequately sized. Engineering analysis determined that a bolted three-phase fault could produce a cable outer jacket temperature of 798 degrees F, exceeding the specified jacket ignition temperature of 700 degrees F, and potentially causing the heater drain cables in both safe shutdown switchgear rooms to exceed their cable ignition temperature. This was determined to be outside the safe shutdown design basis.

B. Palisades - PWR/Plant Age Group A Station

Total No. of DBR LERs: 27; No. of Higher Significance level DBRs: 10

Higher Significance level DBR LERs are detailed as follows:

1. 255/87-001, Rev. 1, Event Date: 01/14/87

Criteria Violated/LER Reportability: Containment isolation valves shall be designed to single failure criteria. (Event which could have prevented fulfillment of a safety function)

Characterization: While reviewing containment isolation circuitry, Plant Engineering personnel discovered that four containment penetrations associated with the hydrogen monitoring system did not employ redundant isolation logic. Although each valve has two solenoid valves, both solenoid valves are operated by the same containment isolation channel.

2. 255/87-007, Rev. 0, Event Date: 02/16/87

Criteria Violated/LER Reportability: The design of the plant systems shall provide assurance that 10 CFR Part 100 offsite dose limits are not exceeded. (Unanalyzed Condition)

Characterization: Engineering identified a discrepancy between the FSAR maximum hypothetical accident (MHA) regarding the containment spray/hydrazine addition system. The design basis enhanced iodine removal is based upon introduction of hydrazine-conditioned spray water being introduced into containment atmosphere within one minute after the initiating event. This introduction time is directly related to maintaining offsite doses below 10 CFR 100 limits. The present design can not introduce hydrazine within the one minute criterion.

TABLE III (Continued)

3. 255/87-039, Rev. 1, Event Date: 10/30/87

Criteria Violated/LER Reportability: Sufficient charging flow shall be provided following a main steam line rupture incident. (Unanalyzed Condition)

Characterization: Through the Palisades system Functional Evaluation (SFE) program, it was determined that charging pump P-55B would not automatically actuate upon a Pressurizer Low Level signal with coincident Safety Injection Signal (SIS), and therefore, outside the design basis. The present design basis calls for 68 gallons per minute flow, equivalent to flow from two charging pumps.

4. 255/88-010, Rev. 0, Event Date: 06/09/88

Criteria Violated/LER Reportability: The design of the plant systems shall provide assurance that 10 CFR Part 100 offsite dose limits are not exceeded. (Unanalyzed Condition)

Characterization: A discrepancy was identified in the radiological consequence analysis of the FSAR "Maximum Hypothetical Accident" (MHA) limiting dose of 100.1 REM to the thyroid at the plant boundary. The analysis did not include the potential contribution from the containment vent path. The vent path is provided from an open clear waste receiver tank to the plant stack.

5. 255/89-006, Rev. 1, Event Date:03/23/89

Criteria Violated/LER Reportability: Plant systems shall be designed to provide for containment heat removal under accident conditions. (Voluntary Report)

Characterization: Through Design Basis Reconstruction efforts, a concern was identified regarding the potential for a complete loss of component cooling water (CCW) system during an event which resulted in a high energy line break (HELB), assuming a LOCA and LOOP. With all of these assumptions, the CCW inventory would be lost to the containment building and the system rendered inoperable. If, in addition, one of the two emergency diesel generators were to fail, all containment heat removal systems would be rendered inoperable.

TABLE III (Continued)

6. 255/89-015, Rev. 0, Event Date: 07/18/89

Criteria Violated/LER Reportability: The plant systems shall be designed to assure containment integrity under postulated accident conditions. (Unanalyzed Condition)

Characterization: An investigation discovered a potential failure that would place the plant outside the current main steam line break (MSLB) containment analysis. A failure of 2400 volt breaker 152-105 (station power to Safeguards bus 1C) to open on a fast transfer signal following a unit trip would prevent the automatic energization of Bus 1C, causing the failure of two of the three containment spray pumps. For MSLB inside containment, this results in a calculated pressure greater than the design limit and a calculated peak containment pressure greater than the current EEQ profile.

7. 255/90-007, Rev. 0, Event Date: 04/18/90

Criteria Violated/LER Reportability: The plant systems shall be designed to assure containment integrity under postulated accident conditions. (Unanalyzed Condition)

Characterization: During a review of MSLB analysis, it was determined that containment pressures could exceed values in the FSAR during MSLB scenarios, where the break size is less than 100% of the steam line cross sectional area.

8. 255/92-007, Rev. 0, Event Date: 02/05/92

Criteria Violated/LER Reportability: The plant systems shall be designed to assure containment integrity under postulated accident conditions. (Unanalyzed Condition)

Characterization: As a result of an on-going equipment classification (Q-List) review program, it was determined that the main steam isolation valve (MSIV) actuator solenoid valves could be rendered inoperable by a MSLB outside containment.

TABLE III (Continued)

9. 255/92-020, Rev. 0, Event Date: 03/03/92

Criteria Violated/LER Reportability: Interfacing non-safety related systems and portions of systems shall isolable or otherwise not cause loss of function to adjacent or interfacing safety-related systems. (Unanalyzed Condition)

Characterization: It was discovered that a design error had caused reliance to be placed on non-safety grade equipment to provide a safety function. A change to the plant design in 1981, made to enhance the high pressure safety injection (HPSI) system, caused each SIS channel to admit flow to all four safety injection headers, but failed to cause all four pressure control valves (PCVs) to be closed by each SIS channel. Therefore, a failure of either SIS channel, concurrent with a failure of a non-safety related PCV control circuit, would cause the PCV to open, providing a flow diversion path for the HPSI flow

10. 255/92-028, Rev. 2, Event Date: 03/31/92

Criteria Violated/LER Reportability: The safety-related equipment shall be capable of performing their safety function throughout design basis conditions. (Unanalyzed Condition)

Characterization: Results of the analysis of the emergency diesel generator rooms cooling requirements and installed safety-related cooling capability revealed that the existing cooling capability was inadequate to maintain the room temperature below the design limit of 104 degrees F, with a design outdoor temperature of 95 degrees F. The analysis results also show that with one cooling fan in operation, the room temperature could be maintained below the design limit (104 degrees F) only if the outdoor temperature does not exceed 75 degrees F.

C. Crystal River - PWR/Plant Age Group B Station

Total No. DBR LERs: 32; No. of Higher Significance level DBR LERs: 10

Higher Significance level DBR LERs are detailed as follows:

1. 302/88-016, Rev. 1, Event Date: 09/07/88

Criteria Violated/LER Reportability: Safety-related systems, components, and structures shall be designed to function under postulated high energy line break (HELB) conditions. (Unanalyzed Condition)

Characterization: It was determined that the plant was operating outside the design basis in that safety-related modifications since 1974 had been performed without the effect of HELB being considered in the design as an on-going design criteria.

TABLE III (Continued)

2. 302/89-006, Rev. 0, Event Date: 02/13/89

Criteria Violated/LER Reportability: Components and systems shall be designed to perform their safety function during normal and accident conditions. (Unanalyzed Condition)

Characterization: It was determined that the plant was operating outside its design basis due to potential failure of circuit breakers which supply power to two decay heat closed cycle cooling pumps. The breaker manufacturer identified that certain breakers did not have rebound springs. Lack of springs may cause slow closure, or may jam in the partially closed position during or following seismic event.

3. 302/89-011, Rev. 2, Event Date: 04/06/89

Criteria Violated/LER Reportability: Components and systems shall be designed to perform their safety function during normal and accident conditions. (Unanalyzed Condition)

Characterization: It was determined that the plant was operating outside the design basis when it was determined that a failure or rupture of a circulating water system expansion joint would flood the turbine building and affect safety-related equipment in the auxiliary room adjacent to the turbine room basement.

4. 302/89-034, Rev. 0, Event Date:09/26/89

Criteria Violated/LER Reportability: Safety-related electrical systems shall be designed as 1E. Non-1E circuits, interfacing with 1E circuits shall be provided with isolation to prevent loss of function to the 1E circuit. (Unanalyzed Condition)

Characterization: Two recently identified conditions were determined to be outside the plant design basis: (1) solenoid control valves for 8 control complex HVAC dampers, 1 HVAC control panel, and 6 containment isolation valves were powered from a non-1E distribution panel and (2) non-safety related testing solenoid valves shared common circuits with safety-related actuation solenoid valves on each of four main steam isolation valves (MSIVs), without proper electrical isolation.

TABLE III (Continued)

5. <u>302/89-035, Rev. 1, Event Date: 09/06/89</u>

<u>Criteria Violated/LER Reportability:</u> Components and systems shall be designed to perform their safety function during normal and accident conditions. (Unanalyzed Condition)

<u>Characterization:</u> On the basis of a recommendation from the B&W Owners Group "Safety and Performance Improvement Program", an investigation of DC powered components was conducted, with resulting discovery of discrepancy between components' rated voltages and actual voltages seen by the components. This was determined to be outside the plant design basis. A total of 270 components were found to be affected, and 32 of these were determined to be inoperable.

6. <u>302/89-037, Rev. 0, Event Date: 10/26/89</u>

<u>Criteria Violated/LER Reportability:</u> Components and systems shall be designed to perform their safety function during normal and accident conditions. (Unanalyzed Condition)

<u>Characterization:</u> During the investigation of a design basis issue associated with the Technical Specification Improvement program, it was determined that the accuracy of the high pressure injection (HPI) flow instrumentation was inadequate. This instrumentation is used to balance flow through the four injection lines during a small break loss of coolant accident (SBLOCA), caused by an HPI line break, to assure adequate HPI flow to the core. Under these conditions, plant operation above 50% power could not be justified and the plant was shutdown.

7. <u>302/90-004, Rev. 0, Event Date: 03/01/90</u>

<u>Criteria Violated/LER Reportability:</u> The plant systems shall be designed such that a single active failure shall not cause loss of safety function. (Unanalyzed Condition)

<u>Characterization:</u> During an Engineered Safeguards (ES) system design review, it was discovered that a failure of a single power supply would defeat the decay heat removal system automatic closure and interlock system (ACTS).

TABLE III (Continued)

8. <u>302/90-006, Rev. 0, Event Date: 04/10/90</u>

<u>Criteria Violated/LER Reportability:</u> Components and systems shall be designed to perform their safety function during normal and accident conditions. (Unanalyzed Condition)

<u>Characterization:</u> It was determined that the spring packs associated with three valve motor operators were undersized, resulting in potential for development of inadequate thrust to open or close the associated valves under maximum postulated differential pressure.

9. <u>302/92-007, Rev. 0, Event Date:05/01/92</u>

<u>Criteria Violated/LER Reportability:</u> Components and systems shall be designed to perform their safety function during normal and accident conditions. (Unanalyzed Condition)

<u>Characterization:</u> During performance of evaluations under the electrical calculation enhancement program (ECEP), it was determined that the calculated input voltage to the motor operators on the two steam admission valves for the steam driven emergency feedwater pump (EFP) may be below the required voltage for operation during a design basis event. The required voltage was derived from differential pressure testing and conservative valve factor assumptions.

10. <u>302/92-010, Rev. 0, Event Date: 06/04/92</u>

<u>Criteria Violated/LER Reportability:</u> Components and systems shall be designed to perform their safety function during normal and accident conditions. (Unanalyzed Condition)

<u>Characterization:</u> While performing reviews of safety-related motor control circuits, it was determined that 28 safety-related components may not function under design basis conditions due to low input voltage at the control devices. This was based on a conservative electrical analysis that assumes a design basis event concurrent with a degraded grid voltage condition.

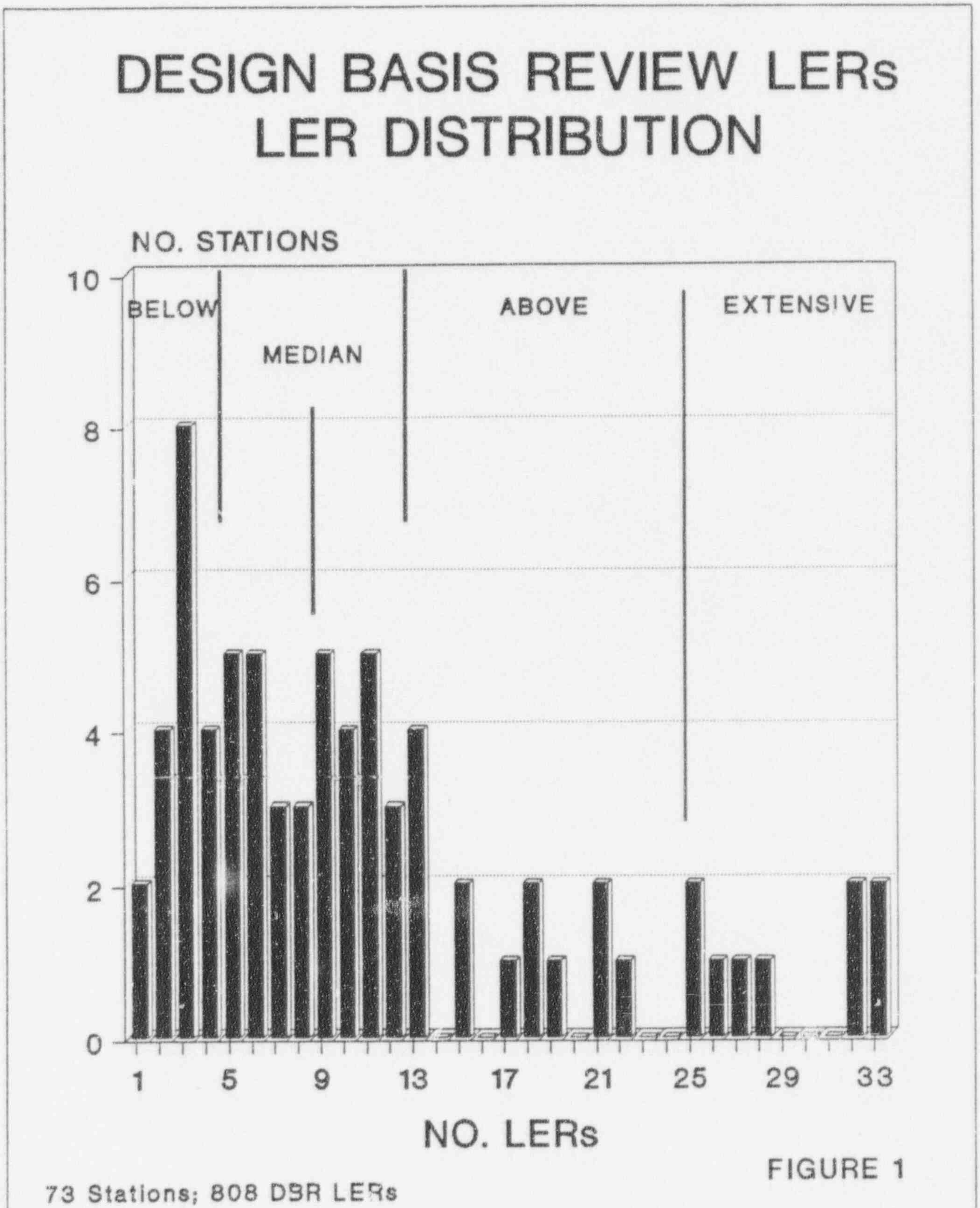
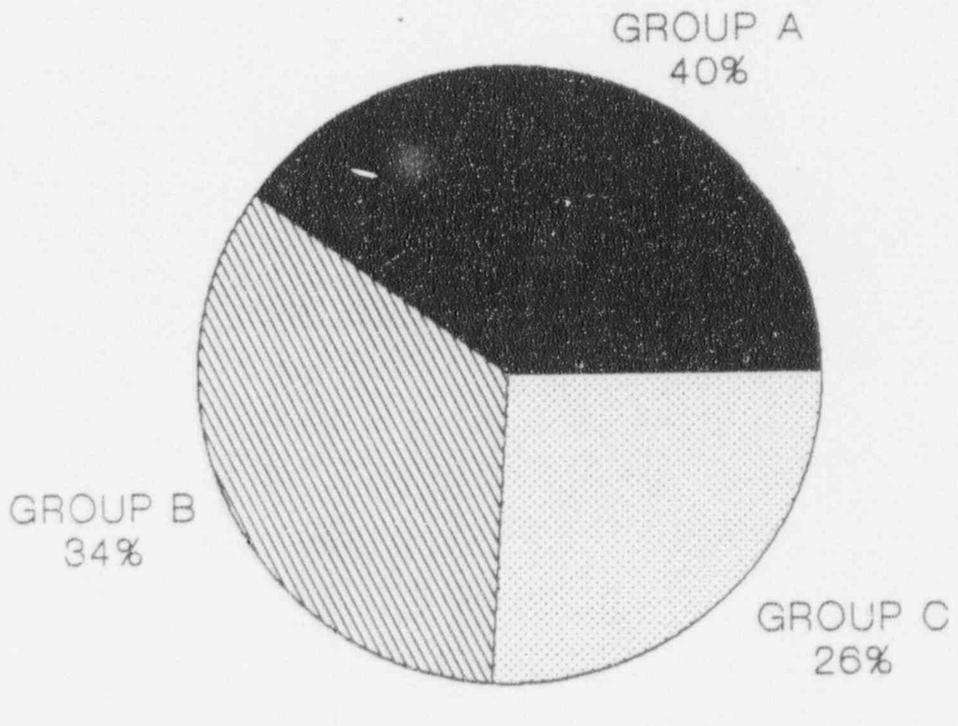

DESIGN BASIS REVIEW LERs
LER DISTRIBUTION
NO. STATIONS
10
8
6
4
2
0
BELOW
MEDIAN
ABOVE
EXTENSIVE
1
5
9
13
17
21
25
29
33
NO. LERs
73 Stations; 808 DBR LERs

FIGURE 1

DESIGN BASIS REVIEW LERs
PLANT AGE GROUPS



LER DISTRIBUTION

GRP A: 1974 or older; GRP B: 1975-1983;
and GRP C: 1984 or newer commercial date

FIGURE 2

DESIGN BASIS REVIEW LERs
DISTRIBUTION OF PLANT CATEGORIES BY AGE GROUP

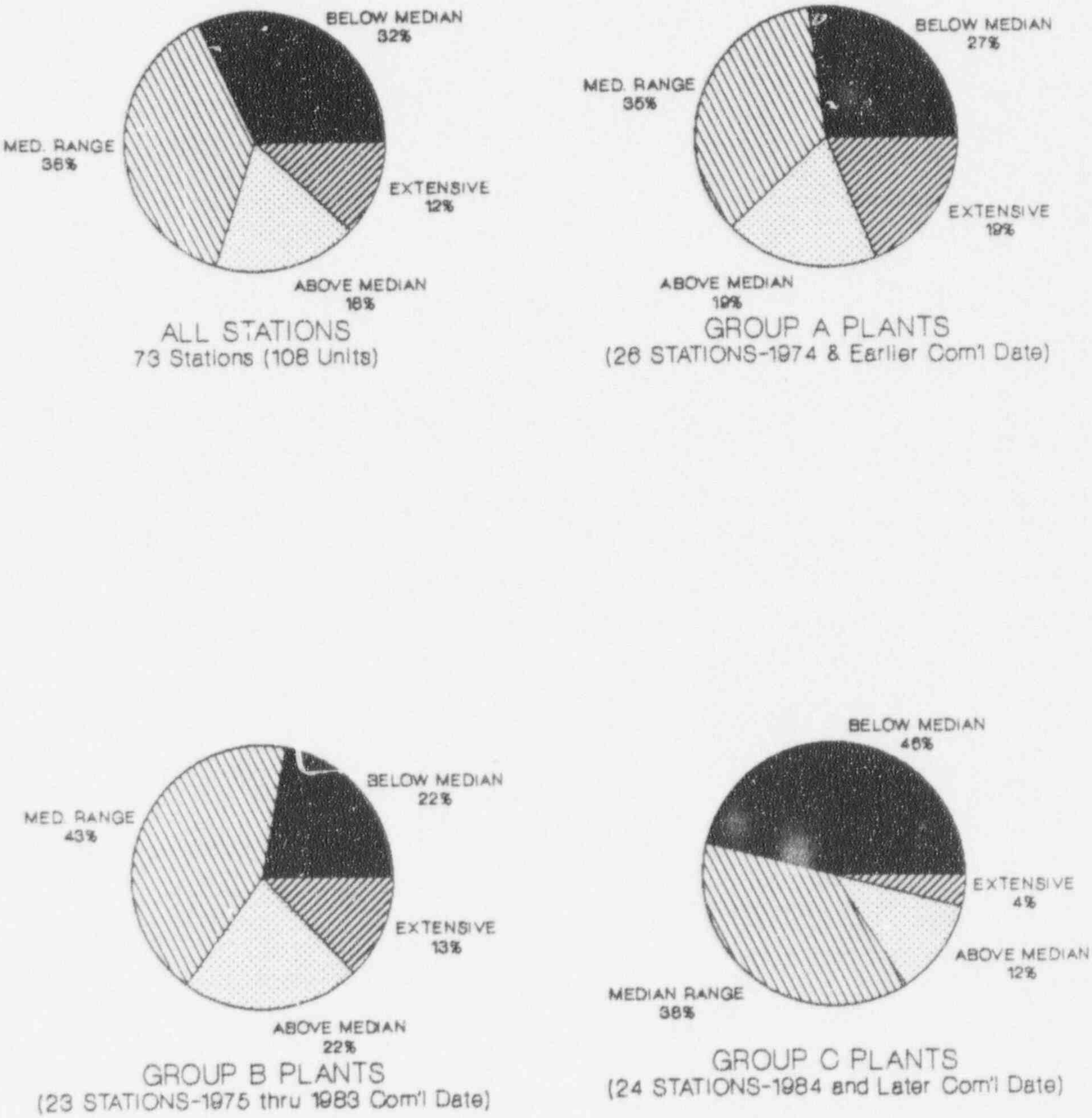


FIGURE 3

Reporting Requirements
Vol. 2
Bx 249
K

Reporting
Requirements
Vol. 2
Bx 249
1