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May 30, 1984

MEMORANDUM FOR: Chairman Palladino
Commissioner Gilinsky
Commissioner Roberts
Commissioner Asselstine
Commissioner Bernthal

FROM: HAEP Herzel H. E. Plaine
General Counsel

SUBJECT: STAFF'S MAY 22, 1984 ORDER IN GRAND GULF

This is in response to Commissioner Bernthal's request that OGC examine the legal bases for the staff's May 22, 1984 immediately effective order in Grand Gulf. The order imposed immediate requirements to disassemble one TDI diesel and to take other measures to compensate for the loss of the TDI diesel and the questionable status of the other TDI diesel, and relaxed a limiting condition for operation (LCO) so that plant shutdown would no longer be required with one TDI diesel out of service and being inspected. We find, based on the current record, that the legal basis is questionable. Our analysis is set forth below.

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Analysis

The text of the order itself suggests two possible grounds for the order. The first is the need "to have increased assurance as to reliable onsite power" at low power operation. The second is "the public interest requires that the questions about the reliability of the Grand Gulf diesel generators be resolved promptly."

A. <u>Increased Low Power Safety</u>

The increased assurance of safety at low power asserted as ground for the order is difficult to reconcile with other staff statements. Staff concluded in a May 24, 1984 memorandum to the Commission that the risk of low power operation with the questionable diesels was "exceptionally small" and that "the risk is not significantly increased by the total loss of the TDI diesel." At the May 24, 1984 Commission meeting, staff similarly advised the Commission that "our analysis shows there was no safety problem with continuing to operate there." Tr. p. 34. <u>See also</u> Tr. pp. 6 ("operation at low power did not pose an undue health and safety risk"), and 14 ("we believed the plant was adequately safe ...."). These statements suggest to us that in staff's view there was little or no safety problem with low power operation. It follows that there was little or no low power operation safety basis for any enforcement order, even an enforcement order limited to the TDI diesel inspection and related compensatory measures, but excluding the LCO change.

The need for increased assurance of safety at low power as a ground for the order becomes even more questionable if one factors in the LCO change, which removed a safety limitation on operation. Staff advised the Commission that the level of plant safety at low power was "at the same level [after the order] as it was before." Tr. p. 45. See also Tr. p. 46 ("Now, I think it probably came out about equal"). If this advice is correct, then the order had no effect on public health and safety at low power, and cannot be justified on that ground.

B. Need to Resolve TDI Issues

Perhaps in recognition of the problems with a safety justification related to low power operation that are discussed above, OELD advised the Commission that "it was primarily public interest" that justified the order. The order itself cites the need to resolve the TDI reliability issue as the relevant "public interest" factor. This, of course, would justify only the TDI inspection portion of the order. However, the public interest in avoiding plant shutdown served as the basis for the LCO relaxation. Tr. pp. 30-31, 43-44.

We believe that a need to resolve the TDI reliability issue on a timely basis, free of the pressures and concerns that inevitably arise when issues remain unresolved up until the last minute before scheduled operation, offers an arguable safety justification for that aspect of the order requiring a TDI inspection. An order demanding information from a TDI inspection, but not amending the low power license, could have been issued under section 161c. and o. of the Atomic Energy Act and 10 CFR

We believe that the LCO relaxation, standing alone, runs into legal difficulties. The so-called "Sholly Amendment," section 189a.(2) of the Atomic Energy Act, provides literally that "the Commission may issue and make immediately effective any amendment to an operating license, upon a determination by the Commission that such amendment involves no significant hazards consideration ...." No such determination was made here. If the Sholly Amendment provides the exclusive means for issuing an immediately effective license amendment, then staff's order cannot stand.

The Sholly Amendment legislative history suggests that section 189a.(2) is ~~not the exclusive means for issuing~~ immediately effective license amendments. The Conference Report recognizes that, apart from Sholly,

> The Commission already has the authority to respond to emergencies involving imminent threats to the public health or safety by issuing immediately effective orders pursuant to the Atomic Energy Act or the Administrative Procedure Act. And the licensee itself has authority to take whatever action is necessary to respond to emergencies involving imminent threat to the public health and safety. H.R. Rep. No. 97-884 at p. 38, 97th Cong., 2d Sess. (September 28, 1982).

The limits of the Commission's authority to take action to respond to "emergencies" to protect health and safety are uncertain. However, clearly that latitude does not include taking action to relax a safety limit on "public interest" grounds. Indeed we are not aware of NRC or AEC ever asserting such a "public interest" authority in their entire regulatory histories.*

However, the Grand Gulf order is complex. The order includes, at least arguably, both measures which enhance safety (timely information on TDI diesels and other compensatory measures), and measures which detract from safety (the LCO relaxation). The question is whether the order can be viewed as a whole, or must be viewed piecemeal. If the order must be justified piecemeal, then the LCO relaxation must fail for the reasons discussed above.

This type of order has been discussed before in a memorandum from the General Counsel, dated January 28, 1980, entitled "Immediately Effective License Amendments" (SECY-80-53). That memorandum concluded that the "viewed as a whole" approach presented litigative risks. The enactment of the Sholly Amendment, with the legislative history cited above, increases those risks.

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*The Administrative Procedure Act's requirement that a licensee be given notice and a chance to bring its activities into compliance before proceedings to suspend or revoke are instituted does not apply "in cases ... in which public health, interest, or safety requires otherwise." 5 U.S.C. § 558(c). However, this provision does not, by its terms, relieve an agency from other procedural requirements in its organic statute (such as the Sholly Amendment). However, some authority to take immediately effective action to protect public health and safety, NRC's paramount concern under the Act, can fairly be inferred from the broad grant of authority in the Atomic Energy Act. Howevever, that broad grant of authority does not include action to further broad "public interest" goals.

On the other hand, the "viewed as a whole" approach has the advantage of enforcement flexibility. It allows NRC to choose the enforcement objective (plant shutdown, plant derating, or additional requirement), and then tailor the enforcement actions to achieve that objective. If one adopts the piecemeal approach, then NRC is faced with the limited choice of shutdown or no action in those situations where other intermediate enforcement actions would violate other license conditions. This limitation could have the unfortunate effect of discouraging enforcement action in difficult cases, to the detriment of public health and safety.

We think that this "viewed as a whole" approach presents considerably more litigative risk than the piecemeal one, but that a court might be convinced by the need for flexibility in an appropriate case.

Conclusion

The LCO relaxation in the Grand Gulf order can be justified only if the order is viewed as a whole. As a general proposition, such an approach presents greater litigative risk than an approach that would require an independent justification for each part of an enforcement order, but has an advantage of enforcement flexibility, and could withstand judicial review in an appropriate case.

However, the Grand Gulf order presents a weak case, even if the order can be viewed as a whole. Even viewed as a whole, the safety advantage of the order is unclear. If, as staff stated, the level of safety with the order is about the same as before without the order, then the order has no safety benefit. All enforcement actions directed at safety must have some overall safety benefit to withstand scrutiny.

The order might be viewed as having the net safety benefit of a timely resolution of the TDI diesel reliability issue. However, it is difficult to construct a strong justification along these lines. This is because the same result could be achieved without <u>any</u> enforcement action by simply seeking Commission concurrence with staff's position that no license above 5% power can be issued without the TDI diesel inspection information. It would be then up to licensee to challenge the Commission's decision that such data is needed, or proceed to obtain the data by disassembling the diesel and requesting an amendment modifying the LCO to permit interim operation. Such an amendment would be subject to <u>Sholly.</u>

We believe that this alternative course was (and still is) the preferable one from the standpoint of litigative risk in this particular case. Licensee could very easily and quickly apply for the necessary LCO license amendment, and staff could proceed to make the appropriate no significant hazards consideration determination. Prior notice and public comment on the no significant hazards consideration finding could be dispensed with under section

189a(2)(C) of the Atomic Energy Act and 10 CFR § 50.91(a)(5) of NRC's regulations. These provisions allow such dispensation in cases where failure to act in a timely way would result in plant shutdown. Efforts should still be made to advise the State prior to issuance of the amendment. See 10 CFR § 50.91(b)(4).

We would note that even if one were to agree with use of the "viewed as a whole" approach here, and not to adopt our alternative approach, a temporary relaxation of the LCO pending satisfaction of the TDI inspection order is the most that is justified.

cc: OPE
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MEMORANDUM FOR: Chairman Palladino
Commissioner Gilinsky
Commissioner Roberts
Commissioner Asselstine
Commissioner Bernthal

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: LER DATA ON PERSONNEL ERRORS

In response to the memorandum, S. J. Chilk to W. J. Dircks, dated March 2, 1984, Subject: "Staff Requirement Briefing on Status of Grand Gulf," I am providing the following report dealing with operating experience at Grand Gulf. This report addresses the following two subjects which were available in preliminary form at the meeting:

1. A count by plant of all LERs for events that occurred in 1983.

2. A count by plant of all LERs reported in 1983 that included at least one personnel error. [Errors that were not attributable to the plant operating staff (i.e., construction errors, design errors, fabrication errors) were not included].

Two previous reports addressing similar aspects of operating experience at Grand Gulf are also enclosed.

The requested data, including comparable counts for 1981 and 1982, are provided in Enclosure 1.

AEOD obtained the data by searching the Sequence Coding and Search System (SCSS) for LERs submitted in 1983 and for LERs that stated or implied that a personnel error was involved in the event.

Because of the extensive amount of information from each LER that is coded in the SCSS, it was not necessary to rely on text searches for particular words (e.g., "personnel error") or to rely on the data coded by the licensee on the LER form. Thus, if the LER text expressly stated that a "personnel error" occurred, or if the LER implied that a personnel error occurred (e.g., "Inadvertently he operated an incorrect valve"), the information was coded into SCSS and was captured by the subsequent search.

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This data is provided in response to the specific reque[illegible] However, the staff is concerned that the data could be easily misinterpret[illegible]. The plants included in Enclosure 1 vary considerably; for example, with respect to the number of years that they have been in operation. Many of these plants have been in operation for several years and have completed the initial "debugging" phase when many equipment problems occur and procedural deficiencies are found. Conversely, some plants (e.g., Grand Gulf) are still in this startup testing phase when large numbers of problems and errors are identified and reported. To place the startup experience at Grand Gulf in better perspective, AEOD searched the SCSS data base for the total number of LERs and the number of LERs that contain at least one personnel error for plants in the startup phase of plant operation. Specifically, AEOD obtained counts of LERs (Table 1) for several plants for the twelve months immediately following issuance of the Low Power Operating License.

TABLE 1

OPERATIONAL EXPERIENCE DURING THE TWELVE MONTHS
IMMEDIATELY FOLLOWING ISSUANCE OF A LOW POWER OPERATING LICENSE

Docket	Facility	Low Power License	Total LERs	Personnel Error LERs*	% Personnel Error LERs
416	Grand Gulf	6/16/82	256	86	34
387	Susquehanna	7/17/82	179	74	41
361	San Onofre 2	2/16/82	186	67	28
373	La Salle	4/17/82	187	67	28
369	McGuire	6/12/81	149	64	43
395	Summer	8/6/82	153	57	37
362	San Onofre 3	11/15/82	93	27	29
328	Sequoyah 2	6/25/81	65	26	40

* The information presented here is based on information available to the staff and has not been verified with the individual licensees.

As indicated in Table 1 and Enclosure 1, Grand Gulf has submitted more LERs and reported more personnel errors than the other units. However, Grand Gulf is the first BWR 6 in the country. As such, there were no personnel previously experienced in preparing procedures or operating this specific model reactor. Even vendor personnel had minimum or no experience with this type reactor. As a result, Grand Gulf may have been more susceptible to personnel errors than, for example, San Onofre 2 and 3 and Sequoyah 2 which are more standard in design. In addition, Sequoyah 2 was the second unit started at that site in a short period. Both San Onofre 3 and Sequoyah 2 had operating personnel with more directly applicable experience. This may have contributed to fewer LERs.

In addition, care should be taken in reaching firm conclusions from this data. Just as there is a risk of focusing too closely on individual events, a number of difficulties are associated with any collective analysis of LER data. For example, when events are reduced to counts they lose their individual identity. This homogenization means all events are treated as if they were

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all reported on the same basis and had the same individual significance -- which often is not the case. Many of the errors reported by Grand Gulf, for example, were missed surveillance requirements that did not directly affect plant operation.

Finally, any variation which is due to factors other than differences in actual safety performance will give a spurious indication of a problem. For example, Region II has indicated that it has a low threshold for requiring licensees to report, and that this was particularly true for Grand Gulf. Such variations are discussed in detail in Appendix E or NUREG-0572, "Review of Licensee Event Reports," which was prepared by the ACRS in 1979. A copy of Appendix E to NUREG-0572 was forwarded to the Commission with my memorandum dated February 24, 1984.

Because of the many factors involved, an apparent trend or pattern in the data does not necessarily imply a real safety problem. Such an apparent trend or pattern requires study to determine the underlying factors and to properly assess the implications and significance of the variations. This in-depth analysis has not been done for the data provided in Table 1 and Enclosure 1.

Region II has performed a review of the LERs issued during the period September 1, 1982 through September 30, 1983, for which personnel error was designated by the licensee as the root cause. Region II found that none of the events had an affect on the health and safety of the public, and the majority did not have the potential for resulting in an event which could have an affect on public health and safety.

In conclusion, while we have provided the requested LER count data, we believe that it is not appropriate and may in fact be misleading to use raw LER counts in isolation as a relative or absolute measure of safety performance. In addition, this practice has the undesired side effect of motivating licensees to minimize the number and content of LERs instead of sharing information for the benefit of all.

(Signed) William J. Dircks

William J. Dircks
Executive Director for Operations

Enclosures:
1. LER Count Data For 1981-1983
2. Grand Gulf Operating Experience
3. Personnel Errors At Selected Operating Plants

bcc: See Page 4
cc w/enclosures:
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OGC
OPE
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Enclosure 1

LER Count Data For 1981-1983

Docket Number	Facility	LERs 1981	LERs 1982	LERs 1983	Personnel Error LERs 1981	Personnel Error LERs 1982	Personnel Error LERs 1983
29	Yankee Rowe	33	42	37	5	6	4
133	Humboldt Bay	5	8	1	1	1	1
155	Big Rock Point	27	35	14	3	6	3
206	San Onofre 1	29	26	5	13	11	3
213	Connecticut Yankee	19	10	18	4	3	1
219	Oyster Creek	72	61	19	22	23	9
220	Nine Mile Point 1	43	18	22	8	3	7
237	Dresden 2	75	53	61	22	12	12
244	Ginna	22	28	28	8	8	9
245	Millstone 1	39	32	28	5	6	4
247	Indian Point 2	33	49	37	4	4	3
249	Dresden 3	33	44	36	4	6	8
250	Turkey Point 3	17	18	17	6	7	8
251	Turkey Point 4	17	14	15	2	6	6
254	Quad Cities 1	24	37	36	4	7	8
255	Palisades	53	49	69	13	11	10
259	Browns Ferry 1	83	91	56	16	21	13
260	Browns Ferry 2	65	35	58	19	7	9
261	Robinson 2	33	18	27	15	6	4
263	Monticello	24	15	8	6	4	3
265	Quad Cities 2	25	21	20	4	2	5
266	Point Beach 1	19	27	7	7	11	4
269	Oconee 1	25	20	18	9	9	9
270	Oconee 2	20	11	7	5	3	2
271	Vermont Yankee	36	26	24	5	4	7
272	Salem 1	118	88	45	24	21	17
275	Diablo Canyon 1	9	12	23	5	6	8
277	Peach Bottom 2	44	42	22	11	11	5
278	Peach Bottom 3	21	26	15	3	4	3
280	Surry 1	83	116	42	22	31	10
281	Surry 2	81	70	37	24	18	5
282	Prairie Island 1	18	13	5	8	6	3
285	Ft. Calhoun 1	11	20	8	1	4	2
286	Indian Point 3	10	4	5	2	0	1
287	Oconee 3	16	14	11	6	5	7
289	Three Mile Island 1	13	16	40	5	5	15
293	Pilgrim 1	58	54	52	13	20	10
295	Zion 1	51	50	45	17	10	12
296	Browns Ferry 3	71	51	50	8	11	6
298	Cooper	25	25	15	12	8	6
301	Point Beach 2	8	11	10	2	5	2
302	Crystal River 3	80	76	45	12	21	16
304	Zion 2	38	29	40	8	7	11
305	Kewaunee	38	35	25	12	5	7
306	Prairie Island 2	11	11	8	3	3	2

Docket Number	Facility	LERs 1981	1982	1983	Personnel Error LERs 1981	1982	1983
309	Maine Yankee	23	39	30	11	11	12
311	Salem 2	123	153	55	25	31	26
312	Rancho Seco	55	31	31	9	12	13
313	Arkansas Nuclear 1	14	30	24	5	9	5
315	Cook 1	64	107	101	18	43	26
316	Cook 2	70	109	104	13	33	29
317	Calvert Cliffs 1	84	82	54	17	24	14
318	Calvert Cliffs 2	57	54	58	9	14	15
320	Three Mile Island 2	32	34	46	13	8	15
321	Hatch 1	140	96	94	33	37	20
324	Brunswick 2	145	136	87	24	38	23
325	Brunswick 1	94	143	46	22	35	20
327	Sequoyah 1	133	77	85	37	16	17
328	Sequoyah 2	27	65	64	14	17	9
331	Arnold	49	81	39	14	18	7
333	Fitzpatrick	78	53	45	21	20	7
334	Beaver Valley 1	102	55	26	21	9	6
335	St. Lucie 1	60	70	26	11	13	7
336	Millstone 2	45	51	25	13	13	4
338	North Anna 1	87	88	70	20	19	13
339	North Anna 2	89	84	67	30	30	21
344	Trojan	31	22	15	13	11	9
346	Davis Besse 1	79	68	55	33	29	16
348	Farley 1	73	62	70	23	11	11
361	San Onofre 2	-	169	124	-	64	41
362	San Onofre 3	-	10	84	-	6	22
364	Farley 2	57	52	37	16	12	10
366	Hatch 2	133	135	117	31	50	29
368	Arkansas Nuclear 2	44	49	45	11	13	15
369	McGuire 1	187	82	100	78	37	32
370	McGuire 2	-	-	64	-	-	28
373	La Salle 1	-	151	116	-	52	36
387	Susquehanna 1	-	80	141	-	38	51
389	St. Lucie 2	-	-	63	-	-	18
395	Summer	-	65	123	-	28	38
409	Lacrosse	15	20	7	5	6	4
416	Grand Gulf 1	-	181	162	-	57	60

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

Enclosure 2

FEB 2 7 1984

MEMORANDUM FOR: Thomas Novak, Assistant Director for Licensing
Division of Licensing

FROM: Frank J. Miraglia, Assistant Director for Safety Assessment
Division of Licensing

SUBJECT: GRAND GULF OPERATING EXPERIENCE

In response to your request (memorandum of October 6, 1983) the Operating Reactors Assessment Branch (ORAB) has reviewed operating experience during the past year at the Grand Gulf facility and prepared the attached report.

The ORAB review included a survey of reported events at Grand Gulf during the past 15 months (i.e. the low power license period) and a comparison of the event reports with reports from two other recently licensed BWRs (LaSalle and Susquehanna) filed during their low power license periods. The sources of event reports included prompt (telephone) notifications filed per 10 CFR 50.72 as well as Licensee Event Reports (LER) required by the Technical Specifications. Operating reactor events briefing summaries were also examined to identify the more significant events. AEOD provided us with substantial support in obtaining event reports.

In general the review revealed that high number of prompt reportable events (10 CFR 50.72) have occurred at Grand Gulf in the past year. The rate of occurrence of these events has been at least three times greater than that of the two other recently licensed BWRs used for comparison. The large number of prompt reports are concerned for the most part with inadvertent actuations of engineered safety features. According to the 50.72 reports, equal numbers of these events have been caused by equipment failure and errors on the part of operators and technicians.

Review of operating reactor event briefing summaries indicates that five "significant" events have been reported for Grand Gulf during the year. They include a low temperature vessel pressurization incident, electrical system malfunction causing inadvertent RPS trips, a diesel generator room fire incident; simultaneous malfunction of both Transamerica DeLaval diesel generators, and an operator error which resulted in 10,000 gallons of water being drained from the reactor vessel to the suppression pool. The number of significant events at Grand Gulf during the low power license period is higher than that for the two other recently licensed BWRs considered in the review. LaSalle had only one event significant enough to be reported at a briefing and Susquehanna had none. It should also be noted that the periods of low power license for LaSalle and Susquehanna were much shorter than Grand Gulf.

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Based on our review we have concluded that operating experience at Grand Gulf during the past year has been atypical. Comparison of Grand Gulf experience with that of other BWRs indicates that the period of operation with the low power license at Grand Gulf has been abnormally long (greater than 12 months versus 4 months for Susquehanna and LaSalle) and that the rate of prompt reportable events has been much greater than expected. Based on discussions with Region II we believe that the high rate of reported events is at least in part related to the large amount of construction and testing activities which have gone on during the past year. This construction and testing activity is the result of design changes being implemented at the plant. The fact that many events which have occurred are related to personnel errors may indicate a lack of experience, on the part of plant personnel.

The rate at which events have occurred at Grand Gulf has not decreased steadily over the long term as the plant has moved closer to commercial operation. However, a sudden sharp decrease in the rate did occur in November 1983 which may be attributed to site inactivity following completion of low power testing in October. On this basis it would be reasonable to expect the incident rate to continue this decreasing trend as the plant moves closer to commercial operation, and testing and construction activities are completed.

We have discussed the results of our review with IE Region II, and they have informed us that our conclusions are consistent with their most recent SALP review. Region II will continue to monitor plant performance and take appropriate actions should problems continue to occur at a high rate.

Frank J. Miraglia

Frank J. Miraglia, Assistant Director
for Safety Assessment
Division of Licensing

Enclosure:
As Stated

OPERATING EXPERIENCE REVIEW

AT GRAND GULF UNIT #1

INTRODUCTION

The staff review of operating experience included a survey of reported events at Grand Gulf during the past 15 months (i.e. the low power license period) and a comparison of the event reports with reports from two other recently licensed BWRs (LaSalle and Susquehanna) filed during their low power license periods. The sources of event reports include prompt (telephone) notifications filed per 10 CFR 50.72 as well as Licensee Event Reports (LER) required by the Technical Specifications. Operating reactor events briefing summaries were also examined to identify the more significant events. These briefings are regularly scheduled meetings among NRC management to discuss recent events at operating reactors.

SURVEY OF EVENT REPORTS

In the period between mid-August 1982 and September 1, 1983 160 incidents requiring prompt notification were reported as required by 10 CFR part 50.72 One hundred and twenty-two of these events involved plant systems. The remaining 38 events involved the plant physical security system. This review has focused on the non-security related events. The security related events were not considered significant and were expected based on the testing and construction occurring at the plant. Thirty-five percent (35%) of the non-security related events have root causes related to operator and technician activities (e.g. testing, troubleshoorting). Equipment problems (mostly electrical) account for thirty-two (32%) of the events. The direct causes of the remaining one-third of the events are unknown or not apparent from the brief 50.72 reports. Most of the events involve inadvertent actuations of safety systems with the plant shutdown (e.g., standby gas treatment system, control room fresh air system, reactor trip, diesel generator start). The average monthly rate at which these events have been reported is approximately 10 events/month. This rate is compared with rates for LaSalle and Susquehanna in Table 1 and appears to be abnormally high. Region II inspectors attribute the high rate to the large amount of testing and construction going on at the plant. A review of the data by month does not reveal any particular trend in the incident rate. Data for the past three months shows a rate of occurrence close to the average in September and October with a sharp decrease in November to 3 events/month. The sharp decrease is attributed to site inactivity following completion of low power tests. A steady reduction in the rate of occurrence is expected as the plant nears commercial operation, since design changes and associated tests are expected to be completed.

In the period beginning August 1, 1982 and ending July 1, 1983 a total of 227 LERs were issued from Grand Gulf. The average monthly rate at which LERs have been issued is shown in table 1 along with comparable rates for LaSalle and Susquehanna. The Grand Gulf rate is similar to the rates for LaSalle and Susquehanna. This is in sharp contrast with the 10 CFR part 50.72 reports discussed above where the Grand Gulf rate was significantly higher than the other two plants. Review of the Grand Gulf LERs indicates that about one-half of the reports relate to problems with fire protection systems. These problems include many instances of smoke detector alarms caused by dust from construction; and, removal of fire barriers for construction purposes. Only nineteen percent (19%) of the 227 reported events involved personnel errors and/or procedural

TABLE 1

RATE OF REPORTED EVENTS AT

THREE BWR PLANTS

DURING LOW POWER LICENSE PERIOD

Facility	Period of Low Power License (months	Rate of Reported Events (Avg. No. reports/month) 50.72	LER
Grand Gulf	12*	10	21
LaSalle 1	4	1	19
Susquehanna 1	4	3	12

* The study period consists of the first 12 months of the low power license period. The actual period of the low power license will be longer than 12 months.

deficiencies. Other causes of reported events include equipment problems and planned entry of technical specification action statements for purposes of testing or construction.

REVIEW OF SIGNIFICANT EVENTS

Significant events which have occurred at Grand Gulf during the past year have been identified through a review of issues raised at the regularly scheduled briefings of NRR management on operating reactor experience. The review consisted of a review of the Operating Reactor Event Briefing meeting minutes. For purposes of comparison a similar review has been performed for LaSalle and Susquehanna for the periods they held low power licenses. Events which are discussed at operating reactor event briefings have been subjected to a screening process in which five or six significant events are selected every two weeks for discussion based on the review of 100 to 150 events reports during the two week period. The purpose of identifying those events here is to provide a measure of the severity and extent of significant operational problems.

During the Grand Gulf low power license period, five significant problems at Grand Gulf were reported. Our review indicates that only one significant event was reported for LaSalle during the period of its low power license. No events were reported for Susquehanna. The Grand Gulf events are summarized below.

Violation of RTNDT Heating Limits During ECCS Injection October 5, 1982

During surveillance testing with the plant in cold shutdown a high DC voltage spike occurred which initiated an ECCS injection. Low pressure core spray injected and caused the reactor vessel to become water solid (extending to the MSIVs). The resulting pressure transient violated the Technical Specification on nil-ductility reference temperature, RTNDT.

Reactor Protection System (RPS) MG-Set Output Breaker Trips, May 19, 1983

Inadvertent tripping of the RPS MG-set output breakers has occurred repetitively resulting in isolation of the instrument air system and a reactor scram signal. The causes of the trips have been identified as thermal overload due to insufficient cabinet ventilation, and low voltage due to voltage swings while the RPS bus is fed from the alternate power supply. To reduce the number of output breaker trips the licensee increased cabinet ventilation, installed voltage regulators to smooth out voltage fluctuations, and installed a new station electrical transmission line from off-site. In addition instrument air system isolation relays have been re-aligned to an interruptable power supply. This problem re-occurred in January 1984. Upward voltage spikes remaining above the setpoint longer than .1 second have caused the protective MG-set output breaker to trip, resulting in de-energization of containment isolation system logic circuits followed by isolation of the RHR system. The licensee has been unable to identify the source of the voltage spikes. To correct the problem, the licensee has increased the output breaker delay time from .1 second to 1.4 seconds. The new delay time is based on measurements of spike duration and consultation with suppliers of the electrical equipment. The modification assures that spikes lasting less than 1.4 seconds will not result in a trip of the protective breaker. Additional corrective actions are also under discussion between the licensee and Region II.

Inadvertent Reactor Vessel Drainage During Shutdown April 3, 1983

On April 3, 1983, approximately 10,000 gallons of water drained from the reactor vessel to the suppression pool through the residual heat removal (RHR) system. This drainage was caused by two RHR valves (F004 and F006) being open simultaneously. At the time of the event, the reactor was at atmospheric pressure with vessel water temperature approximately 100°F (cold shutdown conditions). The vessel water level continued to decrease until the low level isolation signal was received and shutdown cooling isolation valves closed to terminate the leakage.

Diesel Generator Room Fire September 4, 1983

A diesel generator engine fire was caused by a ruptured fuel oil supply line which sprayed oil on the hot exhaust manifold of the diesel. The diesel which caught fire was running at 25 percent load for testing at the time. Two other diesel generators were not affected by the fire. The water deluge system failed to function automatically, but was manually activated to extinguish the fire. The diesel generator governor and turbo chargers were damaged. In addition some electrical equipment in the room suffered water damage.

Inoperability of Delaval Diesel Generators October 28, 1983

On October 28, 1983, a Technical Specification Action Statement was entered when two of the three diesel generators became inoperable. The Division I diesel generator was inoperable due to gasket failure on a lube oil line. The Division II diesel generator became inoperable due to a loose base plate nut on the turbocharger which resulted in a trip of the vibration sensor which tripped the diesel. Corrective action was taken to repair both diesel generators. Both of the diesel generators were manufactured by Transamerica Delaval Inc. (TDI). TDI diesel generators have recently come under close scrutiny by the staff following a crankshaft failure in a TDI diesel generator at the Shoreham plant. Staff review of the Transamerica Delaval diesel generator problem at Grand Gulf is still ongoing.

CONCLUSIONS

Based on our review, we have concluded that operating experience at Grand Gulf during the low power license period has been atypical. Comparison of Grand Gulf experience with that of other BWRs indicates that the period of operation with the low power license at Grand Gulf has been abnormally long (12 months versus 4 months for Susquehanna and LaSalle) and that the rate of prompt reportable events has been much greater than expected. Based on discussions with Region II we believe that the high rate of reported events is related, at least in part, to the large amount of testing and construction activities which have gone on during the past year. This construction and testing activity is the result of design changes being implemented at the plant. The fact that many of the events are related to personnel errors may indicate a lack of experience on the part of plant personnel. The rate at which events have occurred at Grand Gulf has not decreased steadily over the long term as the plant has moved closer to commerical operation. However, a sudden sharp decrease in the rate did occur in November 1983 which may be attributed to site inactivity following completion of the low power testing in October. On this basis, we believe it is reasonable to expect the incident rate to continue this decreasing trend as the plant moves closer to commercial operation, and testing and construction activities cease. Should an abnormally high rate of incidents re-appear, appropriate actions such as initiating a review of personnel training programs and plant procedures should be initiated to identify the root cause of the continuing problem so that necessary corrective measures measures can be taken.

Enclosure 3

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

FEB 24 1984

MEMORANDUM FOR: Chairman Palladino
Commissioner Gilinsky
Commissioner Roberts
Commissioner Asselstine
Commissioner Bernthal

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: PERSONNEL ERRORS AT SELECTED OPERATING PLANTS

The Office of Inspection and Enforcement and the Office for Analysis and Evaluation of Operational Data were requested by Commissioner Gilinsky's staff to provide information on the frequency of personnel errors at selected operating plants (i.e., Grand Gulf, Sequoyah, and Quad Cities). The Commission should understand that the information presented here is strictly a staff effort based on information available to the staff and has not been verified with the individual licensees.

The NRC Operations Center data base contains information on events that are required to be reported under the provisions of 10 CFR 50.72. Many different types of events are reported, including all plant trips and safety system actuations.

The following characteristics of the IE data base should be kept in mind when using the information presented here:

1) The information is called in to the NRC shortly after the event, and at that time an accurate determination of the cause may not be available.

2) Corrections to original reports are not routinely made if later information would indicate a different event cause.

3) Because the search capability of the system relys partially on a text search routine, some events which involve operator error may be missed. This search used "operational failure" and "personnel error." We believe these to be the most frequently used categories for labeling operational errors.

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~~8403080011~~

Table 1 provides a summary of our findings. The tabulated events were reported as operator errors, personnel errors, or procedural errors. Some events were judged to affect the combined units. These are counted separately and not included as Unit 1 or Unit 2 events.

Table 1

Personnel Errors Reported to the
NRC Operations Center 1983*

	Personnel Errors	Site Total
Quad Cities, Unit 1	4	
Quad Cities, Unit 2	1	
Quad Cities (both)	4	9
Grand Gulf, Unit 1	27	27
Sequoyah, Unit 1	6	
Sequoyah, Unit 2	3	
Sequoyah (both)	1	10

* These reports are from calendar year 1983.

In addition, AEOD searched the Sequence Coding and Search System (SCSS) for LERs from Grand Gulf, Quad Cities, and Sequoyah that stated or implied that a personnel action was involved in the event.

Because of the extensive amount of information from each LER that is coded in the SCSS, it was not necessary to rely on text searches for particular words (e.g., "personnel error") or to rely on the data coded by the licensee on the LER form. Thus, if the LER text expressly stated that a "personnel error" occurred, or if the LER implied that a personnel error occurred (e.g., "Inadvertently he operated an incorrect valve"), the information was coded into SCSS and was captured by the subsequent search.

The results of this search were manually reviewed to identify personnel errors that could be attributed to plant personnel (e.g., design errors and fabrication/manufacturing errors were excluded). A rather broad definition of "personnel error" was used which included both errors of commission (e.g., inadvertent operation of the wrong valve) and errors of omission (e.g., missed surveillance requirements).

The results of this analysis are summarized in Table 2.

Table 2

Personnel Errors Reported in LERs

Plant/Unit	Period	Personnel Errors	LERs Received
Quad Cities, Unit 1	1983*	7***	36
Quad Cities, Unit 2	1983*	4	20
Grand Gulf, Unit 1	1983*	58	162
Grand Gulf, Unit 2	1983*	0	0
Sequoyah, Unit 1	1983*	18	85
Sequoyah, Unit 2	1983*	9	64
Sequoyah, Unit 1	June 1, 1982- June 1, 1983**	7	90
Sequoyah, Unit 2	August 1, 1981- August 1, 1982**	18	61

* Some LERs for 1983 have not yet been received and added to the data base. However, the period is essentially the same for all units.

** First year of commercial operation.

*** Many of the personnel errors reported to the Operations Center were also reported in LERs. Therefore, the numbers in Tables 1 and 2 should not be added.

Clearly from Tables 1 and 2, Grand Gulf has reported more personnel errors than the other units analyzed. However, care should be taken in reaching conclusions from this data. As the ACRS discussed in Appendix E to NUREG-0572 (enclosed) there are many reasons for non-randomness (e.g., outliers) in operational data, including differences in reporting requirements, differences in reporting philosophies, etc. It should be noted that many of these differences have been reduced by the recent publication of 10 CFR 50.73, "Licensee Event System", and 10 CFR 50.72, "Immediate Notification Requirement for Operating Nuclear Reactors," which became effective on January 1, 1984. In addition, a review of a count of personnel errors does not consider the severity of the error or its consequences. For example, many of the errors reported by Grand Gulf were missed surveillance requirements that did not directly affect plant operation.

Finally, because of the time available to prepare this analysis and the size of the computer printout, we were not able to make copies of the printout. Consequently, the printouts have been provided (separately) only to Commissioner Gilinsky's office and have not been provided to other interested parties and have not been retained by the staff. If other interested parties want a copy, copies can be made from the enclosed original, or the search strategy can be rerun on the computer and additional printouts produced.

(Signed) William J. Dircks

William J. Dircks
Executive Director for Operations

Enclosures:
As stated

cc w/enclosures:
OGC
OCA
OPE
SECY

APPENDIX E

STATISTICAL ANALYSIS OF LERs: A TRIAL STUDY

Introduction

Approximately 8700 LERs were submitted by the licensees of U.S. commercial nuclear power plants during the years 1976, 1977, and 1978. For several reasons, the number of LERs varies from unit to unit. These variations are important, because, rightly or wrongly, they are often viewed by government agencies and the public as indications of relative safety. While such variations may be indicative of actual differences in safety among nuclear power units, they may have other explanations. It is therefore important to understand all possible explanations and their contributions to variations in the numbers of LERs from unit to unit.

Certain differences in the frequency of submission of LERs from unit to unit will occur as a result of the apparent random nature of the events being reported. Because of this "randomness", it is possible--in fact, probable--that, even among identical nuclear power plant facilities with identical failure probabilities, there will be variations in the reporting rate for LERs. In reality, however, variations beyond those due to "randomness" will frequently be observed. The reasons for such non-random variations include the facts that:

(1) Technical Specifications and license provisions vary among nuclear power plant facilities, because of differences in reactor suppliers, architect/engineers, and constructors, and changes in designs over the years. These variations cause differences in the reporting requirements among facilities.

(2) There may be a tendency at some facilities to report events more readily than at others in cases of marginal reportability. This consideration pertains to events other than obvious "reportable occurrences" (ROs), which all licensees must report[1]*. This tendency can also change with time.

(3) The occurrence of an event may affect the probability of future events. Repair of a facility component or improvement of a deficient procedure may significantly reduce the likelihood of an associated event. On the other hand, ineffective corrective action following an event may result in its repeated occurrence.

(4) The mode of operation (e.g., on-line or shutdown) affects the frequency of various kinds of inspections and the susceptability of systems to random failures. The amount of reactor down-time, for example, may affect the frequency with which LERs are submitted.

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*See reference list following Chapter 4.

(5) Misinterpretations by licensee or NRC personnel involved in the preparation, submission, and processing of LERs can affect relative reporting frequencies among reactor systems.

(6) At some multi-unit power stations, such as Oconee and Browns Ferry, events which involve plant systems or components common to all units, such as swing diesels and electrical switchyards, are filed in the NRC data bank under the docket number of the first unit.

(7) The actual presence of more safety-related deficiencies in a system at an individual facility should result in more frequent submission of LERs. Differences in the number of LERs due to this cause would be a measure of relative safety.

Although the above factors affect the frequency with which LERs are reported, their effects are often relatively small. Frequently the variations produced by these effects are too small to be distinguished from those occurring on a random basis. For example, the Point Beach Nuclear Station in 1976 had 11 reportable occurrence LERs for Unit 1 and 16 for Unit 2. Does this necessarily indicate that one or a combination of the causes listed above produced this difference, or is it possible that a deviation of this magnitude could have been expected if both units had the same average probability of occurrence of reportable events? Statistical analysis indicates that 11 and 16 in one year are both consistent with average occurrence rates in the range of one per 20 days to one per 37 days (10-18 per year). In fact, the pair of numbers, 11 and 16, is the most probable one-year outcome for two units with an average rate of one per 27 days (13.5 per year). In 1978, the Zion Nuclear Station had 85 reportable occurrence LERs for Unit 1 and 39 for Unit 2. In this case, the deviation in the number of LERs between the two units is too large to be attributed solely to random effects. If randomness alone were involved, Unit 1 probably could not have had a reporting rate less than one per 5.2 days (70 per year), while Unit 2 probably could not have had a rate greater than one per 7.2 days (51 per year). In fact, if both Zion units had identical probabilities of reportable events, there is no more than one chance in one million that a deviation this large could occur by chance.

Naturally, there are differences between the Point Beach units. Unit 1 is two years older than Unit 2. During 1976, Unit 2 produced 11% more electrical energy than Unit 1. The results in this example indicate that one should not necessarily conclude that differences in the rates of LER submission between the two units are significant. At Zion, however, one should expect to find that the two units reported at significantly different rates for reasons other than randomness.

Methodology

Methods from probability theory can be used to calculate the impact of randomness on the distribution of the number of LERs among identical nuclear power units. Often, probability tables from reference textbooks are sufficient to perform the analyses. Computer simulations are necessary for the more complicated analyses.

In interpreting the resulting data, it is important to note several basic facts:

(1) The numerical size of expected random variations in event rates increases as the average event rate increases. Deviations of 10 or more are readily expected on a random basis for an average yearly rate of 100, but are unlikely for an average yearly rate of 20. The relative size or percentage variation, however, decreases as the average rate increases.

(2) The chance of large random variations among units increases as the number of units being compared increases. For two units with an assumed average annual LER submission rate of 100, there is only a small chance that one rate will deviate by more than 20 from the average because of randomness. For a comparison among 30 units, however, there is a good chance that at least one will deviate by more than 20 from the annual average rate of 100 because of randomness.

A selected set of LERs was used here to demonstrate the application of this methodology. The sources of the LERs were the 22 BWRs that achieved commercial operation prior to 1976. Records show that this group submitted a total of 27 LERs for 30-day reportable occurrences in auxiliary process systems during 1976, 1977, and 1978. Thus, for this group of units, the average was about one LER of this type for the three-year period. It is first assumed that all units in the group were identical with respect to their chances of generating LERs of this type. Further, it is assumed that if a nuclear power plant experiences a reportable occurrence in an auxiliary process system, the chance of another occurrence is unaffected. Throughout this study a Poisson distribution of events is assumed. Probability theory indicates that, while the average is one, it is very unlikely that each individual unit would experience exactly one. In fact, the probability that all 22 units would each report this number is less than one in ten billion. The most likely result is that about eight units will have no LERs, about eight will have one LER, about four will have two LERs, and about two will have three LERs. Further, it is unlikely (6% chance) that any one unit will have six or more LERs. Comparison to actual LER data shows nine units with no LERs, seven with one LER, two with two LERs, one with three LERs, two with four

LERs, and one with five LERs. The distribution of LERs for the 22 BWRs is consistent with the assumptions stated above.

This example, does not prove, however, that the 22 BWRs are identical to each other with regard to the causes of auxiliary process systems failures. It simply indicates that one should not expect to find significant differences among these units, even though some submitted as few as zero and others as many as five LERs. The value of this analysis is that it provides a methodology through which significantly high deviations can be readily identified among a population of expected random deviations.

Analyses

For purposes of this study, the LERs from 67 nuclear power plants were reviewed. For purposes of analyses, these were divided into PWRs (total = 42) and BWRs (total = 25) and each of these groups was further separated into "older" and "newer" power plants. In this case, "older" was arbitrarily defined as those power plants that went into operation prior to 1976 (see Table E-1). For this group, all LERs submitted during calendar years 1976 through 1978 represent events that occurred during commercial operation.

Data used in these analyses were based on the NRC computer bank and included reportable occurrences only. The ROs were separated into those required to be submitted on a prompt or two-week basis and those submitted on a thirty-day basis. These were analyzed separately since there did not appear to be any correlation in the relative numbers of each type as reported by licensees at the 67 power plants. Lastly, the LERs were further separated according to the system to which they pertained. A listing of these systems is shown in Table E-2.

The primary goal in the analyses was to identify significant deviations or variations in the number of LERs reported from plant to plant and system to system. A deviation was considered to be significant if there was a 5% chance or less that it could have resulted from random variations.

Conclusions

On the basis of these analyses, the following conclusions and/or observations were made:

(1) The frequencies of reportable occurrence LERS among the various nuclear power units were significantly different. There were no identifiable groups of reactor units whose members generated the same average number of reportable occurrence LERs during each of the three years in the study.

(2) Considering the three-year period as a whole, 5 units among the 29 older PWRs deviated significantly from the others in terms of the total number of two-week ROs. The numbers of LERs from Calvert Cliffs-1, Palisades, Rancho Seco, and Three Mile Island-1 were high; Maine Yankee was low. The remaining 24 PWRs reported numbers of LERs consistent with an average of about 20 per unit for the period from 1976 through 1978.

(3) For the same 29 older PWRs, considered year by year, the data showed that the total number of two-week ROs steadily decreased in each successive year. The averages were ten per unit in 1976, six in 1977, and four in 1978. Significant deviations from these occurred at Calvert Cliffs-1 in 1977, Palisades in 1977 and 1978, Point Beach-1 in 1978, Rancho Seco in 1977, and Three Mile Island-1 in 1978. All had higher than normal reporting rates. Maine Yankee had a rate in 1976 significantly lower than normal. These results indicate that the high three-year totals for the four units listed in paragraph 2 above were basically due to high reporting rates in just one of the three years, while the rates for the other two years appear to be normal.

(4) Further analysis of the data showed that the high totals of two-week ROs in four of the older PWRs were attributable to abnormally high numbers of LERs concerning specific systems. Calvert Cliffs-1 had significantly high three-year totals for electric power systems and for reactor systems. Palisades reported high totals for the same two systems, in addition to engineered safety features. Rancho Seco reported a high total for electric power systems. Three Mile Island-1 had high totals for radiation protection systems and for events classed as "systems code not applicable." Many of the electric power system LERs were related to off-site power systems and emergency diesel generators. Reactivity control systems were the source of most of the reactor system LERs from Palisades.

(5) Among the older PWRs with normal yearly totals for two-week ROs, some nevertheless reported significantly higher than normal totals of LERs for specific systems. The number of LERs in reactor systems was higher than normal at Arkansas Nuclear One-1, Oconee-2 and -3, and H.B. Robinson-2. The number for Zion-1 was higher than normal for radiation protection systems. LERs for electric power systems were higher than normal at Fort Calhoun, Oconee-1 and -3, Prairie Island-1, and Turkey Point-3. The systems mentioned here, however, did not contribute significantly to the total number of LERs, since LERs from engineered safety features and reactor coolant systems dominated the two-week ROs from older PWRs. As a result, deviations from normal in the less often reported systems did not have a significant impact on the total number of LERs for these plants.

(6) The data show that newer PWRs, after they achieved commercial operation, had significantly higher LER submission rates for two-week ROs than did older PWRs. The exception was Indian Point-3. As with the older plants, engineered safety features and reactor coolant systems were responsible for a large fraction of the LERs.

(7) With regard to 30-day ROs, there were no identifiable units among the 29 older PWRs that deviated significantly from the average totals for the three-year period. It is possible, however, to identify three separate subgroups among the units in this category. A first subgroup includes seven units with an average reporting rate of about twenty 30-day ROs for the three years. These were Oconee-2, Point Beach-1 and -2, Rancho Seco, San Onofre-1, and Turkey Point-3 and -4. Another group had an average of about forty-five 30-day ROs for the three years. The 10 units in this group were H.B. Robinson-2, Haddam Neck, Indian Point-2, Maine Yankee, Oconee-1 and -3, Prairie Island-1 and -2, R.E. Ginna, and Three Mile Island-1. A third group of 5 units with a normal reporting rate of about 70 for the three-year period included Arkansas Nuclear One-1, Kewanee, Palisades, and Surry-1 and -2. Significant deviations from these groups occurred in 7 units with high reporting rates. These were Calvert Cliffs-1, D.C. Cook-1, Fort Calhoun, Millstone-2, Yankee Rowe, and Zion-1 and -2. It is interesting to note that three of the five operating Combustion Engineering reactors are in this category. These are Calvert Cliffs-1, Fort Calhoun, and Millstone-2. In addition, this category includes all three of the older PWRs having a power level of 1000 MWe or more. These are D.C. Cook-1 and Zion-1 and -2.

(8) The data show that the one-year totals for thirty-day ROs in older PWRs were similar to the three-year totals in that definite subgroups can be identified. In general, a unit that was in a low or higher reporting subgroup in one year remained in the same subgroup in later years. The exceptions were Yankee Rowe, which was in a higher reporting subgroup in 1977, but in lower reporting subgroups in the other two years, and Surry-1 and -2, which were in a lower reporting subgroup during the first two years but in the higher subgroup in 1978. Several significant correlations were found. Those units which tended to remain in the lowest reporting subgroups nevertheless increased their reporting rates for thirty-day ROs from year to year. The sum of their thirty-day and two-week ROs, however, remained essentially constant in time, since the two-week RO total steadily decreased during the three-year period. Large units of 1000 MWe or more reported higher numbers of 30-day ROs, except when the plant factor for the year was low (less than one-third). Later Combustion

Engineering units (not including Maine Yankee) also submitted higher numbers of LERs for thirty-day ROs, except when the plant availability factor was low (less than one-half).

(9) Newer PWRs reported thirty-day ROs at rates consistent with the higher reporting subgroups among older PWRs.

(10) The systems most responsible for the higher LER submission rates for thirty-day ROs in Combustion Engineering units were auxiliary process systems, electric power systems, instrumentation systems, and steam and power conversion systems. These units usually deviated from

the normal reporting rate for these systems. In large units the systems involving a higher than normal number of thirty-day ROs were auxiliary process systems, engineered safety features, instrumentation systems, and radiation protection systems.

(11) With regard to two-week ROs among the 22 older BWRs, eight units deviated from the normal reporting rate during the three-year period. These were Dresden-2, Duane Arnold, E.I. Hatch-1, Fitzpatrick, and Peach Bottom-2 and -3, with higher rates than normal and Dresden-1 and LaCrosse with lower rates than normal. The remaining units reported an average rate of about twenty-four two-week ROs for the three-year period. The rate remained constant at about eight per year.

(12) E.I. Hatch-1 reported two-week ROs at a comparatively high rate for each of the three years. The number of reports pertaining to nearly every system deviated from normal reporting rates for those systems.

(13) Duane Arnold reported two-week ROs at a comparatively high rate in 1976 and 1977. The systems with higher than normal numbers of reports were related to electric power. For Fitzpatrick, the number of two-week ROs for 1976 was high. This unit also had a high number of ROs in instrumentation systems. For Peach Bottom-2 and -3, the number of two-week ROs for 1976 and 1977 was high. Unit 2 had an abnormally high number of ROs for reactor coolant systems and steam and power conversion systems. Unit 3 reported a high number in engineered safety features and for other auxiliary systems. Dresden-3 reported a higher-than-normal number of LERs in 1977. Further, this unit reported an abnormally high number of ROs in electric power systems. Nine Mile Point-1 reported higher-than-normal totals of LERs concerning instrumentation systems. Quad Cities-1 reported a high incidence of two-week ROs in steam and power conversion systems.

(14) Among the three newer BWRs, only Browns Ferry-3 reported abnormally high numbers of two-week ROs in reactor systems after the unit began commercial operation.

(15) Two BWR units, Fitzpatrick and Brunswick-1, reported abnormally high numbers of thirty-day ROs in nearly every system.

As an extension to the above, LERs pertaining to set point drift were analyzed using as a data source the computer bank at the Nuclear Safety Information Center (see Appendix D-III). These analyses showed that there was no significant deviation in the total annual LER submittal rate for setpoint drift among older BWRs or among older PWRs. The average rate for BWRs, however, was approximately five times as large as that for PWRs. Six older PWRs reported rates higher than normal for the three-year period. These were Zion-1 and -2, Fort Calhoun, Millstone-2, Palisades, and Kewanee. It is interesting to note that three of these are Combustion Engineering units. Among newer PWRs, four units reported r high rates in 1978. These were J.M. Farley-1, Indian Point-3, North Anna-1, and Salem. Three older BWRs reported set point drift events at abnormally high rates for the entire three-year period. These were Duane Arnold, Brunswick-2, and Nine Mile Point-1. Six older BWRs reported at abnormally low rates. These were Big Rock Point, Browns Ferry -1, -2, and -3, LaCrosse, and Monticello.

Commentary

This portion of the study has clearly demonstrated the potential usefulness of statistical analyses in the evaluation of LERs submitted by licensees. Such analyses make it possible to distinguish deviations in the numbers of LERs which would be expected on the basis of randomness from those that almost certainly would not. The latter can be used as a means for the identification of areas for possible further investigations. While the deviations noted in this study do not necessarily imply safety-related problems, they should nonetheless be pursued in order to determine the true implications.

It would probably be desirable to computerize these analyses for automatic processing of reports as they are logged into the LER data base. Utilization of the data base in this manner would make it possible to detect significant deviations from normal. Further, an automated system could be programmed to obtain detail beyond the system level, in order to identify reporting rate deviations for relevant subsystems and components.

Table E-1

Number of Reportable Occurrence LERs from Commercial Nuclear Power Plants (1976-1978)

GROUP I: Older PWRs (commercial operation prior to 1976) Total = 29

Nuclear Power Plant	Reportable Occurrences 30 day	Reportable Occurrences 2-week	Nuclear Power Plant	Reportable Occurrences 30-day	Reportable Occurrences 2-week
Arkansas Nuclear One-1	71	17	Point Beach-1	15	30
Calvert Cliffs-1	169	35	Point Beach-2	18	20
D.C. Cook-1	147	20	Prairie Island-1	51	17
Fort Calhoun	109	24	Prairie Island-2	36	18
H.B. Robinson-2	53	26	Rancho Seco	17	40
Haddam Neck	41	19	R.E. Ginna	44	24
Indian Point-2	57	26	San Onofre-1	19	11
Kewanee	75	19	Surry-1	79	19
Maine Yankee	47	6	Surry-2	71	8
Millstone-2	118	21	Three Mile Island-1	44	41
Oconee-1	42	34	Turkey Point-3	24	11
Oconee-2	21	26	Turkey Point-4	20	16
Oconee-3	41	21	Yankee Rowe	99	13
Palisades	64	55	Zion 1	188	25
			Zion 2	122	15
			Average	65.6	22.7

Table E-1 Continued

GROUP II: Newer PWRs (commercial operation after January 1, 1976) Total = 13

Nuclear Power Plant	Reportable Occurrences 30-day	Reportable Occurrences 2-week	Nuclear Power Plant	Reportable Occurrences 30-day	Reportable Occurrences 2-week
Arkansas Nuclear One-2	21	7	Indian Point-3	85	15
Beaver Valley-1	216	27	J.M. Farley-1	138	23
Calvert Cliffs-2	135	25	North Anna-1	98	29
Crystal River-3	154	32	St. Lucie-1	123	22
D.C. Cook-2	96	7	Salem-1	118	68
Davis-Besse-1	220	32	Three Mile Island-2	42	17
			Trojan	63	44
			Average	116.5	26.8

Table E-1 Continued

GROUP III: Older BWRs (commercial operation prior to 1976) Total = 22

Nuclear Power Plant	Reportable Occurrences 30-day	2-week	Nuclear Power Plant	Reportable Occurrences 30-day	2-week
Big Rock Point	105	31	LaCrosse	27	10
Browns Ferry-1	55	26	Millstone-1	80	27
Browns Ferry-2	33	19	Monticello	65	30
Brunswick-2	261	34	Nine Mile Point-1	93	27
Cooper	122	18	Oyster Creek-1	56	35
Dresden-1	70	10	Peach Bottom-2	146	56
Dreaden-2	153	51	Peach Bottom-3	107	56
Dresden-3	109	29	Pilgrim-1	103	25
Duane Arnold	120	88	Quad Cities-1	94	31
E.I. Hatch-1	94	162	Quad Cities-2	75	14
Fitzpatrick	181	41	Vermont Yankee	95	18
			Average	102.0	38.0

Table E-1. Continued

GROUP IV: Newer BWRs (commercial operation after January 1, 1976) Total = 3

Nuclear Power Plant	Reportable Occurrences 30-day	2-week
Browns Ferry-3	58	12
Brunswick-1	211	9
E.I. Hatch-2	65	12
Average	111.3	11.0

Table E-2

System Codes for LERs

System

1. Auxiliary Process Systems
2. Auxiliary Water Systems
3. Electric Power Systems
4. Engineered Safety Features
5. Fuel Storage and Handling Systems
6. Instrumentation and Control Systems
7. Other Auxiliary Systems

System

8. Other Major Systems
9. Radiation Protection Systems
10. Radioactive Waste Management Systems
11. Reactor Systems
12. Reactor Coolant Systems
13. Steam and Power Conversion Systems
14. System Code Not Applicable

F. Hebdon, DTE

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

②

FEB 24 1984

MEMORANDUM FOR: Chairman Palladino
Commissioner Gilinsky
Commissioner Roberts
Commissioner Asselstine
Commissioner Bernthal

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: PERSONNEL ERRORS AT SELECTED OPERATING PLANTS

The Office of Inspection and Enforcement and the Office for Analysis and Evaluation of Operational Data were requested by Commissioner Gilinsky's staff to provide information on the frequency of personnel errors at selected operating plants (i.e., Grand Gulf, Sequoyah, and Quad Cities). The Commission should understand that the information presented here is strictly a staff effort based on information available to the staff and has not been verified with the individual licensees.

The NRC Operations Center data base contains information on events that are required to be reported under the provisions of 10 CFR 50.72. Many different types of events are reported, including all plant trips and safety system actuations.

The following characteristics of the IE data base should be kept in mind when using the information presented here:

1) The information is called in to the NRC shortly after the event, and at that time an accurate determination of the cause may not be available.

2) Corrections to original reports are not routinely made if later information would indicate a different event cause.

3) Because the search capability of the system relys partially on a text search routine, some events which involve operator error may be missed. This search used "operational failure" and "personnel error." We believe these to be the most frequently used categories for labeling operational errors.

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Table 1 provides a summary of our findings. The tabulated events were reported as operator errors, personnel errors, or procedural errors. Some events were judged to affect the combined units. These are counted separately and not included as Unit 1 or Unit 2 events.

Table 1

Personnel Errors Reported to the NRC Operations Center 1983*

	Personnel Errors	Site Total
Quad Cities, Unit 1	4	
Quad Cities, Unit 2	1	
Quad Cities (both)	4	9
Grand Gulf, Unit 1	27	27
Sequoyah, Unit 1	6	
Sequoyah, Unit 2	3	
Sequoyah (both)	1	10

* These reports are from calendar year 1983.

In additic~, AEOD searched the Sequence Coding and Search System (SCSS) for LERs from Grand Gulf, Quad Cities, and Sequoyah that stated or implied that a personnel action was involved in the event.

Because of the extensive amount of information from each LER that is coded in the SCSS, it was not necessary to rely on text searches for particular words (e.g., "personnel error") or to rely on the data coded by the licensee on the LER form. Thus, if the LER text expressly stated that a "personnel error" occurred, or if the LER implied that a personnel error occurred (e.g., "Inadvertently he operated an incorrect valve"), the information was coded into SCSS and was captured by the subsequent search.

The results of this search were manually reviewed to identify personnel errors that could be attributed to plant personnel (e.g., design errors and fabrication/manufacturing errors were excluded). A rather broad definition of "personnel error" was used which included both errors of commission (e.g., inadvertent operation of the wrong valve) and errors of omission (e.g., missed surveillance requirements).

The results of this analysis are summarized in Table 2.

STARTUP PROGRAM NOT PREOPERATIONAL!

	UNIT 1	UNIT 2
1983	95	64
1982	77	65
1981	133	27
1980	197 (RECON)	

Table 2

Personnel Errors Reported in LERs

Plant/Unit	Period	Personnel Errors	LERs Received	
Quad Cities, Unit 1	1983*	7***	36	SCSS HAS NO LERs FOR ≤ 1980.
Quad Cities, Unit 2	1983*	4	20	
Grand Gulf, Unit 1	1983*	58	162	
Grand Gulf, Unit 2	1983*	0	0	
Sequoyah, Unit 1	1983*	18	85	
Sequoyah, Unit 2	1983*	9	64	
Sequoyah, Unit 1	June 1, 1982- June 1, 1983**	7	90	LICENSED SEP 17, 1980
Sequoyah, Unit 2	August 1, 1981- August 1, 1982**	18	61	LICENSED SEP 15, 1981

SEP 15, 1981-1982 => 64 LER

* Some LERs for 1983 have not yet been received and added to the data base. However, the period is essentially the same for all units.

** First year of commercial operation.

*** Many of the personnel errors reported to the Operations Center were also reported in LERs. Therefore, the numbers in Tables 1 and 2 should not be added.

Clearly from Tables 1 and 2, Grand Gulf has reported more personnel errors than the other units analyzed. However, care should be taken in reaching conclusions from this data. As the ACRS discussed in Appendix E to NUREG-0572 (enclosed) there are many reasons for non-randomness (e.g., outliers) in operational data, including differences in reporting requirements, differences in reporting philosophies, etc. It should be noted that many of these differences have been reduced by the recent publication of 10 CFR 50.73, "Licensee Event System"; and 10 CFR 50.72, "Immediate Notification Requirement for Operating Nuclear Reactors," which became effective on January 1, 1984. In addition, a review of a count of personnel errors does not consider the severity of the error or its consequences. For example, many of the errors reported by Grand Gulf were missed surveillance requirements that did not directly affect plant operation.

Finally, because of the time available to prepare this analysis and the size of the computer printout, we were not able to make copies of the printout. Consequently, the printouts have been provided (separately) only to Commissioner Gilinsky's office and have not been provided to other interested parties and have not been

1. INTENSITY OF STARTUP TESTING.

retained by the staff. If other interested parties want a copy, copies can be made from the enclosed original, or the search strategy can be rerun on the computer and additional printouts produced.

(Signed) William J. Dircks

William J. Dircks
Executive Director for Operations

Enclosures:
As stated

cc w/enclosures:
OGC
OCA
OPE
SECY

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DATE	2/23/84	2/23/84	2/ /84	2/23/84	2/ /84	2/23/84	2/ /84

APPENDIX E

STATISTICAL ANALYSIS OF LERs: A TRIAL STUDY

Introduction

Approximately 8700 LERs were submitted by the licensees of U.S. commercial nuclear power plants during the years 1976, 1977, and 1978. For several reasons, the number of LERs varies from unit to unit. These variations are important, because, rightly or wrongly, they are often viewed by government agencies and the public as indications of relative safety. While such variations may be indicative of actual differences in safety among nuclear power units, they may have other explanations. It is therefore important to understand all possible explanations and their contributions to variations in the numbers of LERs from unit to unit.

Certain differences in the frequency of submission of LERs from unit to unit will occur as a result of the apparent random nature of the events being reported. Because of this "randomness", it is possible--in fact, probable--that, even among identical nuclear power plant facilities with identical failure probabilities, there will be variations in the reporting rate for LERs. In reality, however, variations beyond those due to "randomness" will frequently be observed. The reasons for such non-random variations include the facts that:

(1) Technical Specifications and license provisions vary among nuclear power plant facilities, because of differences in reactor suppliers, architect/engineers, and constructors, and changes in designs over the years. These variations cause differences in the reporting requirements among facilities.

(2) There may be a tendency at some facilities to report events more readily than at others in cases of marginal reportability. This consideration pertains to events other than obvious "reportable occurrences" (ROs), which all licensees must report[1]*. This tendency can also change with time.

(3) The occurrence of an event may affect the probability of future events. Repair of a facility component or improvement of a deficient procedure may significantly reduce the likelihood of an associated event. On the other hand, ineffective corrective action following an event may result in its repeated occurrence.

(4) The mode of operation (e.g., on-line or shutdown) affects the frequency of various kinds of inspections and the susceptability of systems to random failures. The amount of reactor down-time, for example, may affect the frequency with which LERs are submitted.

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*See reference list following Chapter 4.

(5) Misinterpretations by licensee or NRC personnel involved in the preparation, submission, and processing of LERs can affect relative reporting frequencies among reactor systems.

(6) At some multi-unit power stations, such as Oconee and Browns Ferry, events which involve plant systems or components common to all units, such as swing diesels and electrical switchyards, are filed in the NRC data bank under the docket number of the first unit.

(7) The actual presence of more safety-related deficiencies in a system at an individual facility should result in more frequent submission of LERs. Differences in the number of LERs due to this cause would be a measure of relative safety.

Although the above factors affect the frequency with which LERs are reported, their effects are often relatively small. Frequently the variations produced by these effects are too small to be distinguished from those occurring on a random basis. For example, the Point Beach Nuclear Station in 1976 had 11 reportable occurrence LERs for Unit 1 and 16 for Unit 2. Does this necessarily indicate that one or a combination of the causes listed above produced this difference, or is it possible that a deviation of this magnitude could have been expected if both units had the same average probability of occurrence of reportable events? Statistical analysis indicates that 11 and 16 in one year are both consistent with average occurrence rates in the range of one per 20 days to one per 37 days (10-18 per year). In fact, the pair of numbers, 11 and 16, is the most probable one-year outcome for two units with an average rate of one per 27 days (13.5 per year). In 1978, the Zion Nuclear Station had 85 reportable occurrence LERs for Unit 1 and 39 for Unit 2. In this case, the deviation in the number of LERs between the two units is too large to be attributed solely to random effects. If randomness alone were involved, Unit 1 probably could not have had a reporting rate less than one per 5.2 days (70 per year), while Unit 2 probably could not have had a rate greater than one per 7.2 days (51 per year). In fact, if both Zion units had identical probabilities of reportable events, there is no more than one chance in one million that a deviation this large could occur by chance.

Naturally, there are differences between the Point Beach units. Unit 1 is two years older than Unit 2. During 1976, Unit 2 produced 11% more electrical energy than Unit 1. The results in this example indicate that one should not necessarily conclude that differences in the rates of LER submission between the two units are significant. At Zion, however, one should expect to find that the two units reported at significantly different rates for reasons other than randomness.

Methodology

Methods from probability theory can be used to calculate the impact of randomness on the distribution of the number of LERs among identical nuclear power units. Often, probability tables from reference textbooks are sufficient to perform the analyses. Computer simulations are necessary for the more complicated analyses.

In interpreting the resulting data, it is important to note several basic facts:

(1) The numerical size of expected random variations in event rates increases as the average event rate increases. Deviations of 10 or more are readily expected on a random basis for an average yearly rate of 100, but are unlikely for an average yearly rate of 20. The relative size or percentage variation, however, decreases as the average rate increases.

(2) The chance of large random variations among units increases as the number of units being compared increases. For two units with an assumed average annual LER submission rate of 100, there is only a small chance that one rate will deviate by more than 20 from the average because of randomness. For a comparison among 30 units, however, there is a good chance that at least one will deviate by more than 20 from the annual average rate of 100 because of randomness.

A selected set of LERs was used here to demonstrate the application of this methodology. The sources of the LERs were the 22 BWRs that achieved commercial operation prior to 1976. Records show that this group submitted a total of 27 LERs for 30-day reportable occurrences in auxiliary process systems during 1976, 1977, and 1978. Thus, for this group of units, the average was about one LER of this type for the three-year period. It is first assumed that all units in the group were identical with respect to their chances of generating LERs of this type. Further, it is assumed that if a nuclear power plant experiences a reportable occurrence in an auxiliary process system, the chance of another occurrence is unaffected. Throughout this study a Poisson distribution of events is assumed. Probability theory indicates that, while the average is one, it is very unlikely that each individual unit would experience exactly one. In fact, the probability that all 22 units would each report this number is less than one in ten billion. The most likely result is that about eight units will have no LERs, about eight will have one LER, about four will have two LERs, and about two will have three LERs. Further, it is unlikely (8% chance) that any one unit will have six or more LERs. Comparison to actual LER data shows nine units with no LERs, seven with one LER, two with two LERs, one with three LERs, two with four

LERs, and one with five LERs. The distribution of LERs for the 22 BWRs is consistent with the assumptions stated above.

This example, does not prove, however, that the 22 BWRs are identical to each other with regard to the causes of auxiliary process systems failures. It simply indicates that one should not expect to find significant differences among these units, even though some submitted as few as zero and others as many as five LERs. The value of this analysis is that it provides a methodology through which significantly high deviations can be readily identified among a population of expected random deviations.

Analyses

For purposes of this study, the LERs from 67 nuclear power plants were reviewed. For purposes of analyses, these were divided into PWRs (total = 42) and BWRs (total = 25) and each of these groups was further separated into "older" and "newer" power plants. In this case, "older" was arbitrarily defined as those power plants that went into operation prior to 1976 (see Table E-1). For this group, all LERs submitted during calendar years 1976 through 1978 represent events that occurred during commercial operation.

Data used in these analyses were based on the NRC computer bank and included reportable occurrences only. The ROs were separated into those required to be submitted on a prompt or two-week basis and those submitted on a thirty-day basis. These were analyzed separately since there did not appear to be any correlation in the relative numbers of each type as reported by licensees at the 67 power plants. Lastly, the LERs were further separated according to the system to which they pertained. A listing of these systems is shown in Table E-2.

The primary goal in the analyses was to identify significant deviations or variations in the number of LERs reported from plant to plant and system to system. A deviation was considered to be significant if there was a 5% chance or less that it could have resulted from random variations.

Conclusions

On the basis of these analyses, the following conclusions and/or observations were made:

(1) The frequencies of reportable occurrence LERS among the various nuclear power units were significantly different. There were no identifiable groups of reactor units whose members generated the same average number of reportable occurrence LERs during each of the three years in the study.

(2) Considering the three-year period as a whole, 5 units among the 29 older PWRs deviated significantly from the others in terms of the total number of two-week ROs. The numbers of LERs from Calvert Cliffs-1, Palisades, Rancho Seco, and Three Mile Island-1 were high; Maine Yankee was low. The remaining 24 PWRs reported numbers of LERs consistent with an average of about 20 per unit for the period from 1976 through 1978.

(3) For the same 29 older PWRs, considered year by year, the data showed that the total number of two-week ROs steadily decreased in each successive year. The averages were ten per unit in 1976, six in 1977, and four in 1978. Significant deviations from these occurred at Calvert Cliffs-1 in 1977, Palisades in 1977 and 1978, Point Beach-1 in 1978, Rancho Seco in 1977, and Three Mile Island-1 in 1978. All had higher than normal reporting rates. Maine Yankee had a rate in 1976 significantly lower than normal. These results indicate that the high three-year totals for the four units listed in paragraph 2 above were basically due to high reporting rates in just one of the three years, while the rates for the other two years appear to be normal.

(4) Further analysis of the data showed that the high totals of two-week ROs in four of the older PWRs were attributable to abnormally high numbers of LERs concerning specific systems. Calvert Cliffs-1 had significantly high three-year totals for electric power systems and for reactor systems. Palisades reported high totals for the same two systems, in addition to engineered safety features. Rancho Seco reported a high total for electric power systems. Three Mile Island-1 had high totals for radiation protection systems and for events classed as "systems code not applicable." Many of the electric power system LERs were related to off-site power systems and emergency diesel generators. Reactivity control systems were the source of most of the reactor system LERs from Palisades.

(5) Among the older PWRs with normal yearly totals for two-week ROs, some nevertheless reported significantly higher than normal totals of LERs for specific systems. The number of LERs in reactor systems was higher than normal at Arkansas Nuclear One-1, Oconee-2 and -3, and H.B. Robinson-2. The number for Zion-1 was higher than normal for radiation protection systems. LERs for electric power systems were higher than normal at Fort Calhoun, Oconee-1 and -3, Prairie Island-1, and Turkey Point-3. The systems mentioned here, however, did not contribute significantly to the total number of LERs, since LERs from engineered safety features and reactor coolant systems dominated the two-week ROs from older PWRs. As a result, deviations from normal in the less often reported systems did not have a significant impact on the total number of LERs for these plants.

(6) The data show that newer PWRs, after they achieved commercial operation, had significantly higher LER submission rates for two-week ROs than did older PWRs. The exception was Indian Point-3. As with the older plants, engineered safety features and reactor coolant systems were responsible for a large fraction of the LERs.

(7) With regard to 30-day ROs, there were no identifiable units among the 29 older PWRs that deviated significantly from the average totals for the three-year period. It is possible, however, to identify three separate subgroups among the units in this category. A first subgroup includes seven units with an average reporting rate of about twenty 30-day ROs for the three years. These were Oconee-2, Point Beach-1 and -2, Rancho Seco, San Onofre-1, and Turkey Point-3 and -4. Another group had an average of about forty-five 30-day ROs for the three years. The 10 units in this group were H.B. Robinson-2, Haddam Neck, Indian Point-2, Maine Yankee, Oconee-1 and -3, Prairie Island-1 and -2, R.E. Ginna, and Three Mile Island-1. A third group of 5 units with a normal reporting rate of about 70 for the three-year period included Arkansas Nuclear One-1, Kewanee, Palisades, and Surry-1 and -2. Significant deviations from these groups occurred in 7 units with high reporting rates. These were Calvert Cliffs-1, D.C. Cook-1, Fort Calhoun, Millstone-2, Yankee Rowe, and Zion-1 and -2. It is interesting to note that three of the five operating Combustion Engineering reactors are in this category. These are Calvert Cliffs-1, Fort Calhoun, and Millstone-2. In addition, this category includes all three of the older PWRs having a power level of 1000 MWe or more. These are D.C. Cook-1 and Zion-1 and -2.

(8) The data show that the one-year totals for thirty-day ROs in older PWRs were similar to the three-year totals in that definite subgroups can be identified. In general, a unit that was in a low or higher reporting subgroup in one year remained in the same subgroup in later years. The exceptions were Yankee Rowe, which was in a higher reporting subgroup in 1977, but in lower reporting subgroups in the other two years, and Surry-1 and -2, which were in a lower reporting subgroup during the first two years but in the higher subgroup in 1978. Several significant correlations were found. Those units which tended to remain in the lowest reporting subgroups nevertheless increased their reporting rates for thirty-day ROs from year to year. The sum of their thirty-day and two-week ROs, however, remained essentially constant in time, since the two-week RO total steadily decreased during the three-year period. Large units of 1000 MWe or more reported higher numbers of 30-day ROs, except when the plant factor for the year was low (less than one-third). Later Combustion

Engineering units (not including Maine Yankee) also submitted higher numbers of LERs for thirty-day ROs, except when the plant availability factor was low (less than one-half).

(9) Newer PWRs reported thirty-day ROs at rates consistent with the higher reporting subgroups among older PWRs.

(10) The systems most responsible for the higher LER submission rates for thirty-day ROs in Combustion Engineering units were auxiliary process systems, electric power systems, instrumentation systems, and steam and power conversion systems. These units usually deviated from

the normal reporting rate for these systems. In large units the systems involving a higher than normal number of thirty-day ROs were auxiliary process systems, engineered safety features, instrumentation systems, and radiation protection systems.

(11) With regard to two-week ROs among the 22 older BWRs, eight units deviated from the normal reporting rate during the three-year period. These were Dresden-2, Duane Arnold, E.I. Hatch-1, Fitzpatrick, and Peach Bottom-2 and -3, with higher rates than normal and Dresden-1 and LaCrosse with lower rates than normal. The remaining units reported an average rate of about twenty-four two-week ROs for the three-year period. The rate remained constant at about eight per year.

(12) E.I. Hatch-1 reported two-week ROs at a comparatively high rate for each of the three years. The number of reports pertaining to nearly every system deviated from normal reporting rates for those systems.

(13) Duane Arnold reported two-week ROs at a comparatively high rate in 1976 and 1977. The systems with higher than normal numbers of reports were related to electric power. For Fitzpatrick, the number of two-week ROs for 1976 was high. This unit also had a high number of ROs in instrumentation systems. For Peach Bottom-2 and -3, the number of two-week ROs for 1976 and 1977 was high. Unit 2 had an abnormally high number of ROs for reactor coolant systems and steam and power conversion systems. Unit 3 reported a high number in engineered safety features and for other auxiliary systems. Dresden-3 reported a higher-than-normal number of LERs in 1977. Further, this unit reported an abnormally high number of ROs in electric power systems. Nine Mile Point-1 reported higher-than-normal totals of LERs concerning instrumentation systems. Quad Cities-1 reported a high incidence of two-week ROs in steam and power conversion systems.

(14) Among the three newer BWRs, only Browns Ferry-3 reported abnormally high numbers of two-week ROs in reactor systems after the unit began commercial operation.

(15) Two BWR units, Fitzpatrick and Brunswick-1, reported abnormally high numbers of thirty-day ROs in nearly every system.

As an extension to the above, LERs pertaining to set point drift were analyzed using as a data source the computer bank at the Nuclear Safety Information Center (see Appendix D-III). These analyses showed that there was no significant deviation in the total annual LER submittal rate for setpoint drift among older BWRs or among older PWRs. The average rate for BWRs, however, was approximately five times as large as that for PWRs. Six older PWRs reported rates higher than normal for the three-year period. These were Zion-1 and -2, Fort Calhoun, Millstone-2, Palisades, and Kewanee. It is interesting to note that three of these are Combustion Engineering units. Among newer PWRs, four units reported at high rates in 1978. These were J.M. Farley-1, Indian Point-3, North Anna-1, and Salem. Three older BWRs reported set point drift events at abnormally high rates for the entire three-year period. These were Duane Arnold, Brunswick-2, and Nine Mile Point-1. Six older BWRs reported at abnormally low rates. These were Big Rock Point, Browns Ferry -1, -2, and -3, LaCrosse, and Monticello.

Commentary

This portion of the study has clearly demonstrated the potential usefulness of statistical analyses in the evaluation of LERs submitted by licensees. Such analyses make it possible to distinguish deviations in the numbers of LERs which would be expected on the basis of randomness from those that almost certainly would not. The latter can be used as a means for the identification of areas for possible further investigations. While the deviations noted in this study do not necessarily imply safety-related problems, they should nonetheless be pursued in order to determine the true implications.

It would probably be desirable to computerize these analyses for automatic processing of reports as they are logged into the LER data base. Utilization of the data base in this manner would make it possible to detect significant deviations from normal. Further, an automated system could be programmed to obtain detail beyond the system level, in order to identify reporting rate deviations for relevant subsystems and components.

Table E-1

Number of Reportable Occurrence LERs from Commercial Nuclear Power Plants (1976-1978)

GROUP I: Older PWRs (commercial operation prior to 1976) Total = 29

Nuclear Power Plant	Reportable Occurrences 30 day	Reportable Occurrences 2-week	Nuclear Power Plant	Reportable Occurrences 30-day	Reportable Occurrences 2-week
Arkansas Nuclear One-1	71	17	Point Beach-1	15	30
Calvert Cliffs-1	169	35	Point Beach-2	18	20
D.C. Cook-1	147	20	Prairie Island-1	51	17
Fort Calhoun	109	24	Prairie Island-2	36	18
H.B. Robinson-2	53	26	Rancho Seco	17	40
Haddam Neck	41	19	R.E. Ginna	44	24
Indian Point-2	57	26	San Onofre-1	19	11
Kewanee	75	19	Surry-1	79	19
Maine Yankee	47	6	Surry-2	71	8
Millstone-2	118	21	Three Mile Island-1	44	41
Oconee-1	42	34	Turkey Point-3	24	11
Oconee-2	21	26	Turkey Point-4	20	16
Oconee-3	41	21	Yankee Rowe	99	13
Palisades	64	55	Zion 1	188	25
			Zion 2	122	15
			Average	65.6	22.7

Table E-I Continued

GROUP II: Newer PWRs (commercial operation after January 1, 1976) Total = 13

Nuclear Power Plant	Reportable Occurrences 30-day	2-week	Nuclear Power Plant	Reportable Occurrences 30-day	2-week
Arkansas Nuclear One-2	21	7	Indian Point-3	85	15
Beaver Valley-1	216	27	J.M. Farley-1	138	23
Calvert Cliffs-2	135	25	North Anna-1	98	29
Crystal River-3	154	32	St. Lucie-1	123	22
D.C. Cook-2	96	7	Salem-1	118	68
Davis-Besse-1	220	32	Three Mile Island-2	42	17
			Trojan	68	44
			Average	116.5	26.8

Table E-1 Continued

GROUP III: Older BWRs (commercial operation prior to 1976) Total = 22

Nuclear Power Plant	Reportable Occurrences 30-day	Reportable Occurrences 2-week	Nuclear Power Plant	Reportable Occurrences 30-day	Reportable Occurrences 2-week
Big Rock Point	105	31	LaCrosse	27	10
Browns Ferry-1	55	26	Millstone-1	80	27
Browns Ferry-2	33	18	Monticello	65	30
Brunswick-2	261	34	Nine Mile Point-1	93	27
Cooper	122	18	Oyster Creek-1	56	35
Dresden-1	70	10	Peach Bottom-2	146	56
Dreaden-2	153	51	Peach Bottom-3	107	56
Dresden-3	109	29	Pilgrim-1	103	25
Duane Arnold	120	88	Quad Cities-1	94	31
E.I. Hatch-1	94	162	Quad Cities-2	75	14
Fitzpatrick	181	41	Vermont Yankee	96	18
			Average	102.0	38.0

Table E-1 Continued

GROUP IV: Newer BWRs (commercial operation after January 1, 1976) Total = 3

Nuclear Power Plant	Reportable Occurrences 30-day	Reportable Occurrences 2-week
Browns Ferry-3	58	12
Brunswick-1	211	9
E.I. Hatch-2	65	12
Average	111.3	11.0

Table E-2

System Codes for LERs

System	System
1. Auxiliary Process Systems	8. Other Major Systems
2. Auxiliary Water Systems	9. Radiation Protection Systems
3. Electric Power Systems	10. Radioactive Waste Management Systems
4. Engineered Safety Features	11. Reactor Systems
5. Fuel Storage and Handling Systems	12. Reactor Coolant Systems
6. Instrumentation and Control Systems	13. Steam and Power Conversion Systems
7. Other Auxiliary Systems	14. System Code Not Applicable