OAK RIDGE NATIONAL LABORATORY

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ANNOTATED BIBLIOGRAPHY OF SAFETY-RELATED EVENTS IN BOILING-WATER NUCLEAR POWER PLANTS AS REPORTED IN 1977

R. L. Scott · R. B. Gallaher

Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Under Interagency Agreements DOE 40-551-75 and 40-552-75

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Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
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FOREWORD

The Nuclear Safety Information Center (NSIC), which was established in March 1963 at Oak Ridge National Laboratory, is principally supported by the U.S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research. Support is also provided by the Division of Reactor Development and Demonstration of the Department of Energy. NSIC is a focal point for the collection, storage, evaluation, and dissemination of safety information to aid those concerned with the analysis, design, and operation of nuclear facilities. The Center has also developed a system of keywords to index the information which it catalogs. The title, author, installation, abstract, and keywords for each document reviewed are recorded at the central computing facility in Oak Ridge. The references are cataloged according to the following categories:

1. General Safety Criteria
2. Siting of Nuclear Facilities
3. Transportation and Handling of Radioactive Materials
4. Aerospace Safety (inactive ~1970)
5. Heat Transfer and Thermal Hydraulics
6. Reactor Transients, Kinetics, and Stability
7. Fission Product Release, Transport, and Removal
8. Sources of Energy Release under Accident Conditions
9. Nuclear Instrumentation, Control, and Safety Systems
10. Electrical Power Systems
11. Containment of Nuclear Facilities
12. Plant Safety Features — Reactor
13. Plant Safety Features — Nonreactor
14. Radionuclide Release, Disposal, Treatment, and Management (inactive September 1973)
15. Environmental Surveys, Monitoring, and Radiation Dose Measurements (inactive September 1973)
16. Meteorological Considerations
17. Operational Safety and Experience
18. Design, Construction and Licensing
19. Internal Exposure Effects on Humans Due to Radioactivity in the Environment (inactive September 1973)

20. Effects of Thermal Modifications on Ecological Systems (inactive September 1973)
21. Radiation Effects on Ecological Systems (inactive September 1973)
22. Safeguards of Nuclear Materials

Computer programs have been developed that enable NSIC to (1) operate a program of selective dissemination of information (SDI) to individuals according to their particular profile of interest, (2) make retrospective searches of the stored references, and (3) produce topical indexed bibliographies. In addition, the Center Staff is available for consultation, and the document literature at NSIC offices is available for examination. NSIC reports may be purchased from the National Technical Information Service. All of the above services are free to NRC and DOE personnel as well as their direct contractors. They are available to all others at a nominal cost as determined by the DOE Cost Recovery Policy. Persons interested in any of the services offered by NSIC should address inquiries to:

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RABIN SA + ATRAZ BG + BADER MB + BUSBOOM HJ + HAZEL VE
EXAMINATION AND EVALUATION OF RUPTURE IN EVESR SUPERHEAT FUEL ROD WITH 0.012-INCH-THICK INCOLOY-800 CLADDING
GENERAL ELECTRIC CO., SUNNYVALE, CALIF. ADVANCED PRODUCTS OPERATION
GEAP-5416 + 72 PAGES, FIGURES, TABLES, REFERENCES, JANUARY 1967

PREMATURE FAILURES WERE OBSERVED IN ONE 0.008-IN.-WALL ROD AND IN ONE 0.012-IN.-WALL ROD OF THE MARK-III EXPERIMENTAL FUEL. CLADDING IS INCOLOY-800. FAILURES WERE IN REGION OF MAXIMUM POWER IN PEAK-POWER ROD. MOST REASONABLE EXPLANATION IS LOW-CYCLE FATIGUE, ACCELERATED BY HIGHER-THAN-DESIGN CLADDING TEMPERATURES. NO SIGNS OF RAPID CORROSION, SUCH AS HAVE BEEN ASSOCIATED WITH TYPE-304 SS, WERE FOUND.

ABSTRACT OF ORIGINAL DOCUMENT

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*FAILURE, FUEL ELEMENT + CLAD + FAILURE, FATIGUE + INCONEL + REACTOR, INTERNAL SUPERHEAT + THERMAL MECHANICAL EFFECT + VESR (ISR)

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PREVIOUS REPORTS IN THIS SERIES

1. W. R. Casto and E. N. Cramer, *Safety-Related Occurrences in Nuclear Facilities as Reported in 1967 and 1968*, ORNL/NSIC-69 (July 1970) (available from NTIS for $5.50).

2. R. L. Scott and W. R. Casto, *Safety-Related Occurrences in Nuclear Facilities as Reported in 1969*, ORNL/NSIC-87 (August 1971) (available from NTIS for $5.50).

3. R. L. Scott, *Safety-Related Occurrences in Nuclear Facilities as Reported in 1970*, ORNL/NSIC-91 (December 1971) (available from NTIS for $10.50).

4. R. L. Scott and R. B. Gallaher, *Safety-Related Occurrences in Nuclear Facilities as Reported in 1971*, ORNL/NSIC-106 (September 1972) (available from NTIS for $12.50).

5. R. L. Scott and R. B. Gallaher, *Safety-Related Occurrences in Nuclear Facilities as Reported in 1972*, ORNL/NSIC-109 (December 1973) (available from NTIS for $15.00).

6. R. L. Scott and R. B. Gallaher, *Annotated Bibliography of Safety-Related Occurrences in Nuclear Power Plants as Reported in 1973*, ORNL/NSIC-114 (November 1974) (availabe from NTIS for $15.00).

7. R. L. Scott and R. B. Gallaher, *Annotated Bibliography of Safety-Related Occurrences in Nuclear Power Plants as Reported in 1974*, ORNL/NSIC-122 (May 1975) (available from NTIS for $15.00).

8. R. L. Scott and R. B. Gallaher, *Annotated Bibliography of Safety-Related Occurrences in Boiling-Water Nuclear Power Plants as Reported in 1975*, ORNL/NUREG/NSIC-126 (July 1976) (available from NTIS for $11.00).

9. R. L. Scott and R. B. Gallaher, *Annotated Bibliography of Safety-Related Occurrences in Pressurized-Water Nuclear Power Plants as Reported in 1975*, ORNL/NUREG/NSIC-127 (July 1976) (available from NTIS for $10.75).

10. R. L. Scott and R. B. Gallaher, *Annotated Bibliography of Safety-Related Occurrences in Boiling-Water Nuclear Power Plants as Reported in 1976*, ORNL/NUREG/NSIC-137 (September 1977) (available from NTIS for $11.75).

11. R. L. Scott and R. B. Gallaher, *Annotated Bibliography of Safety-Related Occurrences in Pressurized-Water Nuclear Power Plants as Reported in 1976*, ORNL/NUREG/NSIC-138 (August 1977) (available from NTIS for $12.00).

ANNOTATED BIBLIOGRAPHY OF SAFETY-RELATED EVENTS AT
BOILING-WATER NUCLEAR POWER PLANTS
AS REPORTED IN 1977

R. L. Scott   R. B. Gallaher

ABSTRACT

This bibliography contains 100-word abstracts of reports submitted to the U.S. Nuclear Regulatory Commission concerning operational events that occurred at boiling-water reactor nuclear power plants in 1977. The 1222 abstracts included in the bibliography describe incidents, failures, and design or construction deficiencies that were experienced at the facilities. They are arranged alphabetically by reactor name and then chronologically for each reactor. Keyword and permuted-title indexes are provided to facilitate location of the abstracts of interest, and tables that summarize the information contained in the bibliography are also provided. The information listed in the tables includes instrument failures, equipment failures, system failures, causes of failures, deficiencies noted, and the time of occurrence (i.e., during refueling, operation, testing, or construction). Three of the more interesting events that occurred during the year are reviewed in detail.

INTRODUCTION

This report (along with ORNL/NUREG/NSIC-150) is the tenth of a series, issued annually by the Nuclear Safety Information Center (NSIC), presenting abstracts of reports of safety-related events submitted to the U.S. Nuclear Regulatory Commission (NRC) by light-water reactor licensees in the United States during the previous year. In particular, this report contains abstracts of 1222 events reported by licensees of boiling-water reactor nuclear power plants in the United States during 1977. The abstracts are presented on microfiche which are filed in an envelope attached to the back cover of the report. The eleven previous reports in the series[1-11] covered the period 1967 through 1976. In addition, five related NSIC reports[12-16] contain information on reactor operating experiences reported by the U.S. Nuclear Regulatory Commission (formerly the Atomic Energy Commission) for the period 1966 through 1977.

Previous reports in this series contained abstracts of reports of safety-related events occurring at both pressurized- and boiling-water reactor facilities; however, due to the continual growth in the number of facilities and consequently in the number of events reported, it has been necessary since 1975 to compile abstracts of the events in two separate documents. The 1977 events occurring at pressurized-water reactor nuclear power plants are presented in ORNL/NUREG/NSIC-150, *Annotated Bibliography of Safety-Related Events in Pressurized-Water Nuclear Power Plants as Reported in 1977.*

The reports of safety-related events abstracted in the bibliography were submitted by power plant licensees to the U.S. Nuclear Regulatory Commission (NRC) in accordance with federal regulations. The reporting requirements for nuclear facility licensees are included in *Title 10, Code of Federal Regulations, Parts 20, 40, 50, 70*, and *73*, and described in detail in NRC Regulatory Guide No. 1.16 (Ref. 17). The requirements for reporting design or construction deficiencies in nuclear facilities that have been granted construction permits are given in *Title 10, Code of Federal Regulations, Part 50, Section 55, Paragraph e* (Ref. 18).

The information for this report was obtained from the computer files of NSIC in the form of 100-word abstracts of the reports submitted by the reactor licensees to the NRC. The abstracts, together with appropriate keywords used for computer storage and retrieval, were prepared by technical specialists at NSIC. Input to the computer is a continuing process; therefore, persons desiring an updating of the information on operating experiences at nuclear power plants may obtain a literature search by contacting the NSIC. (The NSIC computer contains about 7% more abstracts of 1977 events than are contained in the bibliography because the reports of some of the events occurring late in the year were not received in time to be included in the bibliography.)

The NSIC computer also provides a bimonthly printout of those events which resulted in reactor shutdown and their causes; these are published in each issue of the bimonthly journal, *Nuclear Safety*.

The 100-word abstracts in this report are arranged alphabetically according to the name of the reactor and then chronologically for each reactor. In addition, tables are presented that indicate the number of

times a piece of equipment, an instrument, or a system was reported as having been involved in a malfunction. Included in the tables are causes, deficiencies, and time of occurrence (i.e., during operation, testing, refueling, or construction). This summary is followed by a brief discussion of three events that were considered to be the most interesting of those reported during the year.

In addition to the abstracts describing each event, keyword and permuted-title indexes are provided on microfiche for quickly locating the abstracts in which a particular item of interest is discussed. For example, persons interested in the problems experienced with diesel generators can find the relevant abstracts listed under the keyword *generator, diesel*; or using the permuted-title index, they can find the abstracts listed with the word *diesel* or the word *generator*.

Before reviewing the bibliography, it may also be helpful to review "Parts and Method of Indexing Abstracts" (p. vii), which shows a typical abstract with its component parts identified. Note the list of keywords, which gives a quick indication of the contents of the abstract. The availability of the original material is indicated for all abstracts except where it appears in sources such as technical journals, which are available in most technical libraries. In these cases, the name of the journal, issue, date, and page numbers are given above the abstract. Generally, the material related to licensed facilities may be found in the NRC Public Document Room, 1717 H Street, Washington, D.C. 20545, and/or the material may be purchased from the National Technical Information Service, U.S. Department of Commerce, 5285 Port Royal Road, Springfield, Va. 22151.

SUMMARY OF SAFETY-RELATED EVENT DATA

The 1222 abstracts in the bibliography were reviewed and tabulations made of significant items to indicate the total number of reports concerned with those items. These tabulations indicate items that should receive more attention by reactor operators, designers, or other interested parties.

Table 1 lists the number of reports concerned with the various systems. As in 1976, the three systems most frequently reported on were the main cooling system, the containment isolation system, and the reactor protection system. One reason that the containment isolation system was reported on more often than others may be that it encompasses many of the other systems listed, such as high-pressure coolant injection, reactor-core isolation cooling, core spray, etc. This system consists of valves and controls required to isolate the many lines penetrating the containment, and it follows that most of the reports on this system involve malfunctioning valves and controls. It should also be pointed out that the major or critical systems were reported on more often than the less important systems; however, this only reflects the attention given to these major systems by the utilities in the form of surveillance

Table 1. Number of reports concerned with the listed systems

System	Percent of total number of reports	Number of reports
Main cooling	11	134
Containment isolation	9	114
Reactor protection	8	95
High-pressure coolant injection	7	86
Electric power	6	77
Pressure relief	6	74
Emergency power	6	70
Shutdown cooling	5	67
Reactor core isolation cooling	5	58
Radiation monitoring	5	56
Core spray	3	34
Low-pressure coolant injection	3	34
Waste disposal	2	30
Coolant purification	2	27
Ventilation	2	21
Emergency cooling	2	19
Service water	1	18
Pneumatic	1	17
Feedwater	1	14
Secondary shutdown	1	14
Condenser cooling	1	13
Standby gas treatment	1	13

testing, etc., as well as the emphasis which the NRC has placed on them in its reporting requirements.

Table 2 lists the number of reports concerned with various items of equipment. Again this year, as in previous years, valves, piping, and pumps were the equipment items that experienced problems most frequently. These three items accounted for over one-third of the reports. Diesel generators, valve operators, and seals accounted for another 16% of the reports; thus, the first six items in the list accounted for 50% of the reports.

Table 3 is a list of various kinds of instrumentation which presented problems during the year and the number of reports concerned with each item. Since 1971 when similar tables were first prepared, switches

Table 2. Number of reports concerned with the listed component

Equipment	Percent of total number of reports	Number of reports
Valves	20	241
Pipes and fittings	9	106
Pumps	7	80
Diesel generator	6	76
Valve operators	5	58
Seals	5	55
Cables and connectors	5	55
Support structures	4	50
Turbines	4	49
Control rod drives	4	48
Breakers	3	35
Fasteners	3	34
Shock absorber	2	29
Solenoids	2	26
Heat exchangers	2	25
Storage container	2	23
Filters	2	23
Containment vacuum breakers	2	21
Batteries and chargers	1	18
Pressure vessels	1	17
Check valves	1	14
Bearings	1	13
Control rods	1	13
Demineralizers	<1	12

Table 3. Number of reports concerned with the listed instrumentation

Instrumentation	Percent of total number of reports	Number of reports
Switch	25	309
Pressure sensor	10	123
Level sensor	5	60
Radiation monitors	5	56
Relays	4	44
Flow sensor	4	44
Temperature sensor	3	38
Power-range instrument	3	36
Stack monitor	1	16
Recorders	1	16
Position instrument	1	15
Intermediate-range instrument	<1	12

have been at the top of the list, accounting for more malfunctions than any other instruments. Apparently, the reason for this is the large number of switches in safety-related systems as well as their delicacy and sensitivity. Various monitors and sensors account for most of the remaining reports on instrumentation problems.

Table 4 lists the identified causes of the safety-related events reported and the number of reports concerned with each cause. Forty-eight percent of the events involved inherent failures; these are failures for which there was no obvious reason. Examples of items considered to be inherent failures include (1) excessive fish impingement on intake screens (2) instrument set-point drift, and (3) spurious trips of instruments or equipment. Approximately 10% of the reports did not give a cause of failure, and, in most of these cases, an investigation was continuing.

Table 5 lists the various time periods in which the events took place and the associated number of reports. The 651 events which were discovered (or occurred) during testing could be remedied with little or no effect on operation.

Table 4. Number of reports concerned with the listed cause of safety-related events

Cause	Percent of total number of reports	Number of reports
Inherent failure	48	584
Maintenance error	12	145
Administrative error	9	110
Design error	7	89
Operator error	5	67
Installation error	5	63
Fabrication error	3	38
Weather	1	15

Table 5. Number of reports for the listed time of occurrence of off-normal events

Time of occurrence	Percent of total number of reports	Number of reports
Operation	29	350
Testing	53	651
Refueling	13	159
Construction	5	62

Table 6 is a list of deficiencies considered to be of interest and the associated number of reports submitted. As in previous years, instrument calibration and set-point drift were the most frequently reported items followed closely by piping and seal leaks. Procedural deficiencies most frequently involved inadequate procedures, but failure of operators to follow procedures is also included. Deficiencies in communication covers those events involving a misunderstanding between personnel; it also includes misinterpretations of procedures or technical specifications.

Table 7 is an alphabetical listing of the nuclear reactor units from which reports were received and the associated number of reports. Those nuclear units which were in commercial operation all year are listed

Table 6. Number of reports concerned with the listed deficiency

Deficiency	Percent of total number of reports	Number of reports
Instrument calibration	18	223
Set-point drift	16	200
Leak	12	144
Procedures	8	100
Crack	4	52
Crud	3	40
Welds	3	39
Response time	2	28
Age effect	2	21
Communication	2	21
Wear	2	20
Lubrication	1	17
Vibration	1	16
Airborne release	1	13
Corrosion	<1	12

first, followed by those which were in the power-ascension phase part of the year, and then by those which were under construction all year. The total number of nuclear units listed in the table is 42. For the 23 nuclear units which were operable all year, there are 1093 reports — an average of 48 reports per unit. For the 2 units in the power-ascension phase, there are 93 reports — an average of 46 reports per unit. For the 17 units which were under construction during the year, there are 76 reports — an average of 4 reports per year. The total number of reports listed in Table 7 is 1262, whereas the bibliography contains abstracts of 1222 reports. The reason for this discrepancy is that a few of the reports involved more than one unit of a multiple-unit plant, and this is particularly true of those units which were under construction.

Tables 8a and 8b tabulate the number of reports for the listed units which were commercially operable all year. In Table 8a the tabulation is by age, and in Table 8b the tabulation is by power — design electrical rating (DER) in megawatts (electrical) [MW(e)]. These tables were prepared to see if age or power level was a factor in the number of events

Table 7. Number of reports involving the alphabetically listed units

Name	Percent of total number of reports	Number of reports	Age (years)	Design electrical rating [net MW(e)]
In commercial operation all year				
Arnold	8	97	3.6	538
Big Rock Point	5	57	15.1	72
Browns Ferry 1	3	33	4.2	1065
Browns Ferry 2	1	16	3.3	1065
Brunswick 2	7	80	2.7	821
Cooper	5	66	3.6	778
Dresden 1	3	36	17.7	200
Dresden 2	6	76	7.7	809
Dresden 3	5	58	6.4	809
FitzPatrick	4	43	2.9	821
Hatch 1	7	85	3.1	786
Humboldt Bay	<1	5	14.7	63
LaCrosse	<1	8	9.7	50
Millstone 1	2	29	7.1	650
Monticello	3	34	6.8	545
Nine Mile Point 1	3	41	8.2	610
Oyster Creek	2	27	8.3	650
Peach Bottom 2	6	75	3.9	1065
Peach Bottom 3	6	77	3.3	1065
Pilgrim 1	3	36	5.5	655
Quad Cities 1	4	51	5.7	789
Quad Cities 2	3	41	5.6	789
Vermont Yankee	2	22	5.3	514
In power ascension part of year				
Browns Ferry 3	2	26		
Brunswick 1	5	67		
Under construction all year				
Clinton 1	<1	1		
Clinton 2	<1	1		
Grand Gulf 1	1	13		
Grand Gulf 2	<1	10		
Hartsville 1	<1	4		
Hartsville 2	<1	4		
Hartsville 3	<1	4		
Hartsville 4	<1	4		
Hatch 2	<1	8		
Perry 1	<1	3		
Perry 2	<1	3		

Table 7 (continued)

Name	Percent of total number of reports	Number of reports	Age (years)	Design electrical rating [net MW(e)]
Under construction all year (continued)				
Phipps Bend 2	<1	1		
River Bend 1	<1	1		
River Bend 2	<1	1		
Shoreham	1	12		
Susquehanna 1	<1	3		
Susquehanna 2	<1	3		

reported by a nuclear unit. Both appear to be factors, although it may not be readily apparent from just looking at the tables.

The total number of reports for the 11 oldest reactors was 422, whereas the number of reports for the 11 most recently built reactors was 630 — 49% more reports than for the older reactors. This tends to indicate that there will be fewer failures or malfunctions of safety-related equipment as the unit ages and experience is gained in operation.

The same type of count was made based on power level. The number of reports for the 11 smallest units was 392; the number of reports for the 11 largest units was 635 — almost 62% more reports than for the smaller units. This seems to indicate that fewer problems can be expected with smaller units.

While it should be recognized that the data presented is not absolute, especially when you consider that reporting habits throughout the industry may not be uniform, the tables and data do seem to indicate that a low-powered, older reactor will probably have fewer problems than a high-powered, newly built reactor. However, one factor to be considered in this conclusion is that the newly built reactors are the larger units and, to date, the feedback of operating information from the operators to the designers of the larger units has been limited. Also, the newer, larger units are more complicated than the older, smaller units.

Table 8a. Number of reports for the listed units which were commercially operable all year (by age since first electrical generation)[a]

Name	Age (years)[a]	Percent of total number of reports	Number of reports
Dresden 1	17.7	3	36
Big Rock Point	15.1	5	57
Humboldt Bay	14.7	<1	5
LaCrosse	9.7	<1	8
Oyster Creek	8.3	2	27
Nine Mile Point 1	8.2	3	41
Dresden 2	7.7	6	76
Millstone 1	7.1	2	29
Monticello	6.8	3	34
Dresden 3	6.4	5	58
Quad Cities 1	5.7	4	51
Quad Cities 2	5.6	3	41
Pilgrim 1	5.5	3	36
Vermont Yankee	5.3	2	22
Browns Ferry 1	4.2	3	33
Peach Bottom 2	3.9	6	75
Arnold	3.6	8	97
Cooper	3.6	5	66
Browns Ferry 2	3.3	1	16
Peach Bottom 3	3.3	6	77
Hatch 1	3.1	7	85
FitzPatrick	2.9	4	43
Brunswick 2	2.[illegible]	7	80

[a]Average age — 6.5, median age — 5.6.

Tabe 8b. Number of reports for the listed units which were commercially operable all year (by design electrical rating)[a]

Name	DER[a] [net MW(e)]	Percent of total number of reports	Number of reports
Browns Ferry 1	1065	3	33
Browns Ferry 2	1065	1	16
Peach Bottom 2	1065	6	75
Peach Bottom 3	1065	6	77
Brunswick 2	821	7	80
FitzPatrick	821	4	43
Dresden 2	809	6	76
Dresden 3	809	5	58
Quad Cities 1	789	4	51
Quad Cities 2	789	3	41
Hatch 1	786	7	85
Cooper	778	5	66
Pilgrim 1	655	3	36
Millstone 1	650	2	29
Oyster Creek	650	2	27
Nine Mile Point 1	610	3	41
Monticello	545	3	34
Arnold	538	8	97
Vermont Yankee	514	2	22
Dresden 1	200	3	36
Big Rock Point	72	5	57
Humboldt Bay	63	<1	5
LaCrosse	50	<1	8

[a]Average DER — 661; median DER — 778.

The final bit of information gleaned from reviewing the bibliography is that, of the 1222 reports, 44 indicated that a reactor shut-down occurred or was required because of equipment failure or malfunction.

REVIEW OF SELECTED SAFETY-RELATED EVENTS

A review of the reported events indicated that most of them were of a routine and inconsequential nature; however, a few were significant or unique. Three events that were considered to be the most interesting are presented here to illustrate the types of experiences that occurred in 1977.

Unanticipated Short Periods During Shutdown Margin Tests

During two shutdown margin tests on May 4 and 6, unexpected short periods occurred at Unit 1 of the Quad Cities Nuclear Power Station. Commonwealth Edison Company owns this boiling-water reactor (BWR), which is located in Rock Island, Ill. Two nuclear engineers were present during the first test, which involved four rods; there was a written procedure (QTS 1104-5), but it was not rigorously followed in withdrawing control rod H-9 from position 06 to 08, and control rods F-8, H-8, and K-8 were all at position 48 (position 00 is all the way in, and position 48 is all the way out). This action caused a period that was too short to be properly indicated by the period meter. All that could be said with certainty at the time was that the period was possibly shorter than 30 sec. The operator immediately returned control rod H-9 to the 06 position.

During an investigation later the same day, it was decided that the written procedure had not been followed; however, it was also decided that criticality had not occurred. This opinion was based on a close inspection of the recorder chart of the startup-range recorder, which was the only available data. Because of the very short duration of the incident (less than 1 sec for control rod H-9 to move from position 06 to 08 then back to position 06), it was impossible to see anything more than the normal "prompt jump"; there was no [illegible] stable period indicating criticality. Also, just pr[illegible] under discussion,

another test had shown subcriticality with control rod H-8 at position 48 and control rod H-9 at position 08. The only difference between the two tests was that, in the latter test, control rods F-8 and K-8 were at position 48, and each of these two control rods is one control cell away from control rods H-8 and H-9. This difference was not considered significant enough to cause criticality.

Two days later, control rod maneuver sheets were prepared and approved for another shutdown margin test using the same control rods as before, and a step-by-step procedure was written for the control room operator. A nuclear engineer was present during the test. Again, a short period occurred as the operator withdrew control rod H-8 continuously from position 08 to 22, with control rod H-9 at position 08 and control rods F-8 and K-8 at position 48. Once more the operator quickly inserted control rod H-8, this time from position 22 to 14, and the reactor period decreased to 75 sec. The reactor was maintained at a critical level while the desired data for this test were taken, and then the test was terminated.

Because of the experience with the second test, the station personnel reevaluated both tests and concluded that periods of less than 5 sec occurred in both instances. Failure of the nuclear engineers to follow approved procedures caused the first incident. Although failure to follow the approved procedure did not contribute to the second occurrence, the nuclear engineer exercised poor judgment in allowing the operator to continuously withdraw a control rod during the approach to criticality.

A contributing cause of these occurrences was the inadequacy of procedure QTS 1104-5 and its corresponding data sheet QTS 1104-S3. The data sheet calls for readings from the source-range monitoring instrumentation and the control room operator's initials at specific control rod configurations. However, the control rod movements performed in attaining those specific configurations are governed by procedure QTS 1104-5. Using only the data sheet was misleading owing to the required control rod maneuvers between specified rod configurations. A previously successful demonstration of a two-rod face adjacent shutdown margin was performed using procedure QTS 1104-1 and checklist QTS 1104-S4, both of

which specify exact control rod maneuvers. The inconsistency involved in going from QTS 1104-1 to QTS 1104-5 was unnecessarily misleading.

The final contributing factor was a misunderstanding between the nuclear engineers and the control room operators. The control room operators, who were performing shutdown margin demonstrations for the first time, should have been made aware by the nuclear engineers of the potentially high notch worths of the control rod maneuvers they were performing. The uniqueness of shutdown margin criticals should have been discussed and the operator cautioned in each case. Failure to do so, along with the successful previously completed two-rod face adjacent shutdown margin demonstrations, lulled the operators into a false sense of security when they performed the other shutdown margin demonstration tests.

The nuclear group of the technical staff was given additional specific training in the performance of shutdown margin demonstrations, and emphasis was placed on interacting with the control room operator and on technical specification implications when the tests were being performed.

The procedures and checklists used to perform shutdown margin testing were revised to be consistent and to include a complete step-by-step sequence of control rod maneuvers for the control room operator. Furthermore, the procedures were generally reviewed for safety and current applicability.[19]

Two Random Failures Cause Shutdown

The Cooper Nuclear Station in Brownville, Nebr., was shut down when an unexpected loss of feedwater occurred in August 1977. This BWR is owned by the Nebraska Public Power District. Before the shutdown, a contactor on a relay for power switching was known to be out of service, but it was thought to be needed only in the manual mode. However, when a fuse blew on the no-break power panel, it became apparent that the inoperative contactor was needed in transferring this panel to an alternate alternating-current (ac) power source. Consisting of two inverters, an ac line, and solid-state transfer switches, this no-break power panel

supplies 120-V ac power for the control of feedwater flow and various control and indicating instruments. Loss of the power caused the feedwater pumps to decrease to minimum speed, with a corresponding decrease in flow. When the reactor water level dropped to the scram set point, the reactor automatically shut down, and partial isolation of the containment occurred. Since the mainstream isolation valves remained open, the water level continued to drop 1.25 m more, at which point the mainstream isolation valves closed. The pumps for both reactor core isolation cooling and high-pressure coolant injection started as required but went to full speed. The high-pressure coolant injection system had troubles that had previously surfaced, but control of the reactor core isolation cooling system was inhibited by the failure of the no-break power panel. Naturally, the reactor-vessel pressure increased, and, at approximately 7.95 MPa (approximately 1080 psig), one relief valve opened. The minimum level reached by the water in the vessel was 1.15 m above the set point for initiation of the systems for automatic depressurization, residual heat removal, and core spraying. After about 15 min, power was restored manually to all circuits on the no-break power panel. The only corrective actions required were replacement of the blown fuse and repair of the defective contactor. There were no adverse consequences from the standpoint of public health and safety.[20]

Short Periods During Reactor Startups

In April 1977 the NRC circulated information to all licensees of operating BWRs concerning short periods that had been experienced in the startup of two BWRs, Dresden and Monticello. At Dresden 2 on Dec. 28, 1976, during a reactor startup following a scram from unrelated causes about 9 hr earlier, a rod withdrawal of one notch resulted in a rapid power rise within a period of about 1 sec. Commonwealth Edison Company owns this BWR, which is located in Grundy County, Ill. On its most sensitive scale, the intermediate-range monitor scrammed the reactor. At the time of the startup, the moderator was essentially without voids and at a temperature of 170°C (338°F). A similar event occurred at Dresden on Aug. 17, 1972.

Similarly, at the Monticello Nuclear Power Station on Feb. 23, 1977, following a reactor scram about 10 hr earlier, again from unrelated causes, a reactor period of about 1 sec occurred during startup when a single rod was pulled one notch. Northern States Power Company owns this BWR, which is located at Monticello, Minn. In this instance also, the intermediate-range monitor was on its most sensitive scale and terminated the power rise. Few voids existed in the moderator, where the temperature was 249°C (480°F).

The two most recent events were similar in the following respects:

1. Prior to the earlier, unrelated scram, both plants had been operating at or near full power, with axial flux peaking in the bottom portion of the core.

2. The time from the earlier scrams to the subsequent startups maximized the xenon concentrations in the core.

3. High-worth rod locations were similar, and both plants were using the same generic control rod pattern (identified as B1).

4. Prior to the intermediate-range-monitoring scram at both facilities, dramatic indications of high notch worth had been seen, with rod withdrawals resulting in periods ranging from 10 to 30 sec, which were terminated by reinsertion of the rod.

All the systems, including the reactor protection system, functioned as required. The combination of essentially no voids in the moderator and high xenon concentration accounted for the conditions that resulted in the control rod notch acquiring an unusually high differential reactivity worth, which approximated 50% $\Delta k/k$ at Monticello. This excessive worth of rod notch was the result of essentially no voids in the moderator and peak xenon conditions, which necessitated the withdrawal of significantly more control rods than are normally required to reach criticality. The resultant flux distribution at criticality magnified the normal axial peaking at the top of the core because of the heavy xenon concentrations at the bottom. In addition, the radial contribution to flux peaking was enhanced owing to the withdrawal of peripheral rods.

NRC records show that, after the earlier event at Dresden 2 on Aug. 17, 1972, corrective measures were taken for the subsequent startup,

consisting of notchwise withdrawal of the group of rods. This corrective action was taken only for that operating cycle.

An evaluation of these events indicates that essentially trouble-free startups can be accomplished by avoiding peak xenon with no moderator voids or possibly by the use of a rod pattern developed for these particular conditions.

These events indicate a need for all licensees of operating BWRs to review their startup procedures and practices to ensure that their operating staff has adequate information to perform reactor startups avoiding such short periods. Operators should be made aware that extremely high rod notch worths can be encountered under these conditions. The procedures should include requirements for a thorough assessment following the occurrence of a short period before any further rod withdrawals are made. The considerations should be included in the operator training and requalification training programs.[21]

CONCLUSION

We can all profit by the experiences of others as long as there is free communication among the interested parties. This compilation was prepared with this objective in mind and is intended to provide some guidance as to where additional effort can be expended to minimize the occurrence and the recurrence of off-normal incidents. In this way, the safety, reliability, and availability of nuclear facilities should be improved.

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