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Annotated Bibliography of Safety-Related Events in PRESSURIZED-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
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ANNOTATED BIBLIOGRAPHY OF SAFETY-RELATED EVENTS AT PRESSURIZED-WATER NUCLEAR POWER PLANTS AS REPORTED IN 1977

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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 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
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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)
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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.

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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 PRESSURIZED-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 pressurized-water reactor nuclear power plants in 1977. The 1780 abstracts included in the bibliography describe incidents, failures, and design or construction deficiencies 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 summarizing the information contained in the bibliography are also presented. 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). Four of the more interesting events that occurred during the year are reviewed in detail.

INTRODUCTION

This report (along with ORNL/NUREG/NSIC-149) 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 1780 events reported by licensees of pressurized-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] cover the period 1967 through 1976. In addition, five related NSIC reports[12-16] contain information on reactor operating experiences reported by the NRC (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 boiling-water reactor nuclear power plants are presented in ORNL/NUREG/NSIC-149, *Annotated Bibliography of Safety-Related Occurrences in Boiling-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 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 NSIC computer files in the form of 100-word abstracts of the reports submitted by power 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 file 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, refueling, construction, or testing — both preoperational and surveillance). This is followed by a brief discussion of four 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 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 the "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 1780 abstracts in the bibliography were reviewed and tabulations were 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 previous years, three systems — reactor protection, main

Table 1. Number of reports concerned
with the listed systems

System	Percent of total number of reports	Number of reports
Main cooling	10	176
Secondary cooling	9	152
Reactor protection	8	151
Feedwater	8	149
Electric power	7	119
Coolant purification	6	114
Containment isolation	6	108
Condenser cooling	5	86
Engineered safety features	5	83
Emergency electric power	5	81
Reactor control	3	60
Radiation monitoring	3	53
Service water	3	48
Containment	3	46
Containment spray	3	45
Shutdown cooling	2	43
Ventilation	2	39
Emergency cooling	2	36
Containment air cooling	2	34
Pneumatic	2	32
Component cooling	2	27
Containment filtering	<1	7
Core reflooding	<1	5

cooling, and feedwater — were involved in more events than the other systems. Combined, these systems accounted for 26% of the total number of 1977 reports. The secondary cooling system was also involved in a substantial number of reports — 9% of the total number.

Table 2 lists the number of reports concerned with various pieces of equipment. Pipes, pumps, and valves were the equipment items most frequently involved in the events reported, accounting for 41% of the reports. Valves accounted for 18% of the total number of reports, pumps accounted for 13%, and pipes accounted for 10%.

Table 3 lists the number of reports concerned with the listed instrumentation. Again this year, as in every year since 1972, switches accounted for more reports than any other instrument. In 1977 switches

Table 2. Number of reports concerned with the listed equipment

Equipment	Percent of total number of reports	Number of reports
Valves	18	320
Pumps	13	232
Pipes and pipe fittings	10	177
Steam generators	6	109
Storage containers	6	108
Support structures	5	94
Seals	5	88
Diesel generators	5	86
Cables and connectors	4	78
Pressurizer	3	61
Valve operators	3	55
Control rods	3	51
Breakers	3	49
Control rod drives	2	43
Shock absorbers	2	42
Blowers	2	36
Filters, screen	2	36
Fastener	2	27
Motors	2	27
Tubing	1	24
Bearings	1	23
Check valves	1	22
Demineralizer	1	22
Solenoid	1	20
Turbines	1	20
Condensers	<1	16
Heat exchangers	<1	16
Transformers	<1	16
Batteries and chargers	<1	14
Filters	<1	13
Accumulators	<1	11
Fuel elements	<1	7
Flanges	<1	6

were reported on 222 times, accounting for 12% of the total number of reports. Lagging far behind were radiation monitors, relays, pressure sensors, and level sensors, each of which accounted for 5% or less of the total.

Table 4 lists the identified causes of the safety-related events reported and the number of reports concerned with each cause. Inherent

Table 3. Number of reports concerned with the listed instrumentation

Instrumentation	Percent of total number of reports	Number of reports
Switch	12	222
Radiation monitors	5	84
Relays	4	75
Pressure sensors	4	65
Level sensors	3	62
Flow sensors	3	45
Position instrument	2	41
Temperature sensor	2	33
Power-range instrument	2	27
Amplifiers	<1	15
Recorders	<1	8

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	42	747
Design error	13	223
Maintenance error	11	195
Administrative error	10	172
Operator error	9	162
Installation error	6	106
Fabrication error	4	70
Weather	2	28

failures were involved in 42% of the reports; these are failures for which there was no obvious reason. Examples of these types of events include (1) an excessive number of fish impinged on the intake screens, (2) instrument set-point drift, and (3) spurious trips of instruments or equipment. The causes listed account for 98% of the reports; the remaining 2% of the reports did not give a reason for failure, and most indicated that further investigation was required.

Table 5 lists the time periods in which the various events took place and the associated number of reports. It should b noted that those items discovered during testing could be remedied with little or no effect on reactor operation.

Table 6 is a list of deficiencies considered to be of interest and the number of reports associated with each one. The most frequently

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	52	924
Testing	32	562
Construction	11	194
Refueling	5	100

Table 6. Number of reports concerned with the listed deficiency

Deficiency	Percent of total number of reports	Number of reports
Leak	11	190
Procedures	7	130
Set-point drift	6	108
Instrument calibration	6	107
Welds	4	69
Vibration	3	49
Communication	3	46
Crud	3	45
Lubrication	2	31
Fatigue	1	26
Airborne release	1	25
Corrosion	1	21
Records	<1	12
Stress corrosion	<1	11
Erosion	<1	4
Fire	<1	4

reported deficiency is "leak," which includes any type of leak, such as water or steam from pipes, valves, or fittings. Deficiency 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 reactors which were in commercial operation all year are listed 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. Excluding Indian Point 1, which was shut down all year, 68 nuclear units are represented in this bibliography, which contains 1780 abstracts of reports. For the 32 nuclear units which were operational all year, there are 1101 reports — an average of 34 reports per unit (the same as in 1976). For the 6 units in the power-ascension stage, there were 463 reports — an average of 77 reports per unit. For the 30 units which were under construction, there are 256 reports — an average of 8 reports per unit. It should be pointed out that Table 7 indicates that there are 1820 reports, whereas the bibliography contains abstracts of 1780 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 submitted for the listed units which were commercially operable all year. In Table 8a the tabulation is by age; 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 reported by a nuclear unit. Both age and power appear to be factors, although it may not be readily apparent from just looking at the tables.

The total number of reports for the 16 oldest reactors was 426, whereas the number of reports for the 16 most recently built reactors was 675 — 58% 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.

Table 7. Number of reports involving the alphabetically listed units[a]

Name	Percent of total number of reports	Number of reports	Age (years)	Design electrical rating [net MW(e)]
In commercial operation all year				
Arkansas Nuclear 1	2	29	3.4	850
Calvert Cliffs 1	5	97	3.0	845
Connecticut Yankee	1	19	10.4	575
Cook 1	3	50	2.9	1054
Fort Calhoun 1	3	55	4.4	457
Ginna	<1	10	8.1	490
Indian Point 2	2	32	4.[illegible]	873
Indian Point 3	<1	15	1.7	873
Kewaunee	2	37	3.7	535
Maine Yankee	1	7	5.1	790
Millstone 2	2	34	2.1	828
Oconee 1	3	45	4.7	887
Oconee 2	2	33	4.1	887
Oconne 3	2	33	3.3	887
Palisades	3	54	6.0	668
Point Beach 1	<1	16	7.2	497
Point Beach 2	<1	13	5.4	497
Prairie Island 1	2	44	4.1	530
Prairie Island 2	2	31	3.0	530
Rancho Seco	1	20	3.2	913
Robinson 2	2	29	7.3	712
St. Lucie 1	3	54	1.7	802
San Onofre 1	1	20	10.5	430
Surry 1	1	24	5.5	822
Surry 2	1	20	4.8	822
Three Mile Island 1	1	26	3.5	819
Trojan	3	51	2.0	1130
Turkey Point 3	<1	8	5.2	693
Turkey Point 4	<1	15	4.5	693
Yankee Rowe	2	39	17.1	175
Zion 1	4	75	4.5	1040
Zion 2	4	66	4.0	1040
In power ascension part of year				
Beaver Valley 1	4	63		
Calvert Cliffs 2	4	75		
Cook 2	<1	2		
Crystal River 3	7	133		
Davis-Besse 1	7	130		
Salem 1	4	60		

Table 7. (continued)

Name	Percent of total number of reports	Number of reports
	Under construction all year	
Arkansas Nuclear 2	1	26
Beaver Valley 2	<1	6
Bellefonte 1	<1	6
Bellefonte 2	<1	5
Braidwood 1	<1	2
Braidwood 2	<1	2
Callaway 1	<1	5
Cherokee 1	<1	1
Cherokee 2	<1	1
Cherokee 3	<1	1
Comanche 1	<1	7
Comanche 2	<1	7
Diablo Canyon 1	<1	1
McGuire 1	<1	4
McGuire 2	<1	5
Midland 1	<1	4
Millstone 3	<1	2
North Anna 1	3	52
North Anna 2	3	49
North Anna 3	<1	4
North Anna 4	<1	4
Salem 2	<1	1
San Onofre 2	<1	5
San Onofre 3	<1	5
St. Lucie 2	<1	1
Three Mile Island 2	<1	12
Waterford 3	<1	7
Watts Bar 1	<1	13
Watts Bar 2	<1	14

[a]Three reports not included involved Indian Point 1, which was shut down all year with no decision on its future.

The same type of count was made based on power level. The number of reports for the 16 smallest units was 451, whereas the number of reports for the 16 largest units was 650 — 44% more reports than for the smaller units. This seems to indicate that fewer problems can be expected with smaller units.

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

Name	Age (years)[a]	Percent of total number of reports	Number of reports
Yankee Rowe	17.1	2	39
San Onofre 1	10.5	1	20
Connecticut Yankee	10.4	1	19
Ginna	8.1	<1	10
Robinson 2	7.3	2	29
Point Beach 1	7.2	<1	16
Palisades	6.0	3	54
Surry 1	5.5	1	24
Point Beach 2	5.4	<1	13
Turkey Point 3	5.2	<1	8
Maine Yankee	5.1	<1	7
Surry 2	4.8	1	20
Oconee 1	4.7	3	45
Indian Point 2	4.5	2	32
Turkey Point 4	4.5	<1	15
Zion 1	4.5	4	75
Fort Calhoun	4.4	3	55
Oconee 2	4.1	2	33
Prairie Island 1	4.1	2	44
Zion 2	4.0	4	66
Kewaunee	3.7	2	37
Three Mile Island 1	3.5	1	26
Arkansas Nuclear 1	3.4	2	29
Oconee 3	3.3	2	33
Rancho Seco	3.2	1	20
Calvert Cliffs 1	3.0	5	97
Prairie Island 2	3.0	2	31
Cook 1	2.9	3	50
Millstone 2	2.2	2	34
Trojan	2.0	3	51
Indian Point 3	1.7	<1	15
St. Lucie 1	1.7	3	54

[a]Average age = 5.0; median age = 4.5.

Table 8b. Number of reports for the listed unit which was commercially operable all year (by design electrical rating)[a]

Name	DER [net MW(e)]	Percent of total number of reports	Number of reports
Trojan	1130	3	51
Cook 1	1054	3	50
Zion 1	1040	4	75
Zion 2	1040	4	66
Rancho Seco	913	1	20
Oconee 1	887	3	45
Oconee 2	887	2	33
Oconee 3	887	2	33
Indian Point 2	873	2	32
Indian Point 3	873	<1	15
Arkansas Nuclear 1	850	2	29
Calvert Cliffs	845	5	97
Millstone 2	828	2	34
Surry 1	822	1	24
Surry 2	822	1	20
Three Mile Island 1	819	1	26
St. Lucie 1	802	3	54
Maine Yankee	790	<1	7
Robinson 2	712	2	29
Turkey Point 3	693	<1	8
Turkey Point 4	693	<1	15
Palisades	668	3	54
Connecticut Yankee	575	1	19
Kewaunee	535	2	37
Prairie Island 1	530	2	44
Prairie Island 2	530	2	31
Point Beach 1	497	<1	16
Point Beach 2	497	<1	13
Ginna	490	<1	10
Fort Calhoun	457	3	55
San Onofre 1	430	1	20
Yankee Rowe	175	2	39

[a]Average DER = 784; median DER = ~810.

While it should be recognized that the data presented is not absolute, especially when you consider that the 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 these larger units has been limited. In addition, the newer, larger units are more complicated than the older, smaller units.

The final bit of information gleaned from reviewing the bibliography is that, of the 1780 reports, 83 indicated that a reactor shutdown 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 were of a routine and inconsequential nature; however, a few were significant or unique. Four events that were considered to be the most interesting are presented here to illustrate the types of experiences that occurred in 1977.

Loss of Instrument Air Causes Damage to Reactor Coolant-Pump Seals

The St. Lucie 1 reactor was scrammed on April 15, 1977, when a loss of cooling water for the reactor coolant-pump seals became evident. St. Lucie 1 is owned by the Florida Power & Light Company and is located at Hutchinsons Island, Fla. As usual, one failure led to another. The trouble with the coolant-pump seals started with a seal problem in the containment instrument-air compressor during normal plant operation. The backup air compressor started as designed, but because a check valve on the discharge line of the first air compressor stuck in the open position, pressure could not be maintained. Without compressed air, control of all air-operated valves in the containment was lost, including those on the seals for the reactor coolant pumps. Air pressure was restored within an hour, but the instrumentation records and a visual inspection revealed that the loss of instrument air may have caused damage to the reactor coolant-pump seals. The plant was placed in cold shutdown for further inspection and repairs. The two compressors were completely checked out, and new check valves were installed. For additional backup, the compressed-air system was modified so that compressed air for the turbine-building instrument-air system would automatically be available if the instrument air for the containment was lost.[19]

Rapid Depressurization

Davis-Besse Nuclear Power Station, Unit 1, was partially depressurized in September 1977 while operating at 263 MW(t) but producing no electricity. This new pressurized-water reactor (PWR) is owned and operated by the Toledo Edison Company based in Oak Harbor, Ohio. The event was initiated by an as yet unexplained failure in the steam and feedwater rupture-control system, which closed the feedwater valve to one of the two steam generators. When the steam generator boiled dry, loss of heat transfer caused the pressure in the reactor coolant system to increase. At 15.55 MPa (2255 psig), the pressurizer power relief valve opened nine times and then stuck open. Shortly thereafter, the rupture disk on the quench tank for pressurizer effluents burst and caused the containment pressure to increase to a point greater than 0.6 m of H_2O. Within 6 min the pressure in the reactor coolant system had dropped to the saturation pressure for the temperature of the system, and, as steam formed, water surged into the pressurizer, raising the level to the maximum. Twenty-one minutes after the start of the problem, the operators determined that the power relief valve had stuck open, and they closed its block valve.

The Babcock & Wilcox Company reviewed the transients on the primary system and determined that they were within the design limits. The power relief valve failed because a relay was missing from its control circuit. This relay provides a seal in the circuit that holds the power relief valve open until the pressure decreases to 15.2 MPa (2205 psig). With this control missing, the valve opened and closed as the pressure fluctuated narrowly around 15.55 MPa (2255 psig). After nine cycles, the pilot valve stem failed and the power relief valve remained open. There was no positive determination of failure of the steam and feedwater rupture-control system; therefore, this system will be monitored during the next power escalation to detect any spurious signals.[20]

Two Lightning Strikes Cause Plant Blackout

At Donald C. Cook, Unit 1, lightning caused a plant blackout while the PWR was at 100% power. Indiana and Michigan Electric Company in Bridgeman, Mich., owns and operates this plant. At 6:57 PM on Sept. 1, 1977, a 345-kV transmission breaker failed while a lightning strike on the transmission circuit was being cleared. In order to prevent feeding the failed breaker, the breakers on all lines connected to it were opened. This deenergized the transformers from which Unit 1 receives its normal reserve auxiliary power. The plant was then operating normally, supplying its own power, but without a backup source. Six minutes later, another lightning bolt hit another transmission circuit, causing high-speed opening and reclosing of line breakers and resulting in a voltage dip to 64% of normal. Undervoltage protection relays on the buses for the reactor coolant-pump motors detected this ephemeral voltage dip and tripped the reactor. Then, as designed, both the turbine and the generator tripped, after which the auxiliary plant load was automatically transferred from the normal source to the reserve source, which had been deenergized just minutes before. A station blackout occurred, and the diesels started and assumed the load. Because the plant was scheduled to go down a few hours later for maintenance, it was not restarted immediately. However, it was returned to service on Sept. 5, 1977. No ill effects were indicated from the blackout condition. However, it was subsequently determined that the 64% voltage dip was only of 5-sec duration, whereas a 10-sec delay is acceptable. Accordingly, the undervoltage relays were modified to extend their delay times.[21]

Recurrent Water-Pressure Surges

On Jan. 5, 1977, Beaver Valley 1 experienced its third feedwater-line vibration within a 2-month period. The reactor was at 75% power when a feedwater heater drain pump tripped; this caused the main feed pumps to trip on low suction pressure, resulting in a low feed flow. The turbine load was reduced immediately at a rate of 2%/min. The drain and feed pumps were returned to service, and the plant was operated at 54% power for approximately 3 min, when a loud rumbling noise was heard, followed by a reactor trip initiated by a signal indicating a low water level in the steam-generator coincident with a signal indicating steam flow—feed flow mismatch. The vibration lasted about 15 sec.

It is believed that the pressure surge was caused by dynamic instability of the feedwater regulating valves; the valves became unstable and opened despite the control signal to the valves.

New trims were installed in the three feedwater regulating valves and the feedwater flow-control valves, and the feedwater pipes were extensively instrumented. Preoperational testing, consisting of introducing plant transients while the feedwater control system is in the automatic mode, will demonstrate the degree of valve stability and the effect of this stability on piping movement.[22]

CONCLUSION

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

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