OAK RIDGE NATIONAL LABORATORY
OPERATED BY UNION CARBIDE CORPORATION · FOR THE DEPARTMENT OF ENERGY

POOR ORIGINAL

NUREG/CR-0861
ORNL/NUREG/NSIC-165

Summary and Bibliography of Safety-Related Events at Pressurized-Water Nuclear Power Plants as Reported in 1978

R. L. Scott
R. B. Gallaher

1132 084

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

NUCLEAR SAFETY INFORMATION CENTER

NSIC

AVAILABILITY OF NSIC DOCUMENTS

Recent NSIC reports that may be ordered from the National Technical Information Service, U.S. Department of Commerce, 5285 Port Royal Road, Springfield, Virginia 22161 are listed below.

ORNL/ NSIC	Title	Price*
53	Radiography Incidents and Overexposures, R. L. Scott and R. B. Gallaher, Dec. 1972	$10.50
55 Vol.I	Design Data and Safety Features of Commercial Nuclear Power Plants, Vol. I, Docket No. 50-3 through 50-295, F. A. Heddleson, Dec. 1973	$15.00
55 Vol.II	Design Data and Safety Features of Commercial Nuclear Power Plants, Vol. II, Docket No. 50-296 through 50-395, F. A. Heddleson, Jan. 1972	$15.00
55 Vol.III	Design Data and Safety Features of Commercial Nuclear Power Plants, Vol. III, Docket No. 50-397 through 50-449, F. A. Heddleson, Apr. 1974	$15.00
55 Vol.IV	Design Data and Safety Features of Commercial Nuclear Power Plants, Vol. IV, Docket No. 50-452 through 50-503, F. A. Heddleson, June 1975	$15.00
	See ORNL/NSIC-96 for Vol. V and ORNL/NUREG/NSIC-136 for Vol. VI.	
74	Calculation of Doses Due to Accidentally Released Plutonium from an LMFBR, B. R. Fish, G. W. Keilholtz, W. S. Snyder, and S. D. Swisher, Nov. 1972	$15.00
82	Chemical and Physical Properties of Methyl Iodide and Its Occurrence Under Reactor Accident Conditions — A Summary and Annotated Bibliography, L. F. Parsly, Dec. 1971	$12.50
91	Safety-Related Occurrences in Nuclear Facilities as Reported in 1970, R. L. Scott, Dec. 1971	$10.50
96	Design Data and Safety Features of Commercial Nuclear Power Plants (Fifth Volume of ORNL/NSIC-55), F. A. Heddleson, June 1976	$15.00
97	Indexed Bibliography of Thermal Effects Literature-2, J. G. Morgan and C. C. Coutant, May 1972	$10.50
100	Nuclear Power and Radiation in Perspective, Selections from *Nuclear Safety*, J. R. Buchanan, Mar. 1974	$15.00
101	Indexed Bibliography on Environmental Monitoring for Radioactivity, B. L. Houser, May 1972	$10.50
102	Compilation of National and International Nuclear Standards (Excluding U.S. Activities) 8th Ed., 1972, J. P. Blakely, June 1972	$10.50
103	Abnormal Reactor Operating Experiences, 1969—1971, R. L. Scott and R. B. Gallaher, May 1972	$ 9.00
105	Indexed Bibliography on Nuclear Facility Siting, H. B. Piper, June 1972	$10.50
106	Safety-Related Occurrences in Nuclear Facilities as Reported in 1971, R. L. Scott and R. B. Gallaher, Sept. 1972	$12.50
107	Index to *Nuclear Safety*, A Technical Progress Review by Chronology, Permuted Title, and Author, Vol. 1, No. 1 through Vol. 13, No. 6, J. Paul Blakely and Ann Klein, May 1973	$ 9.00
109	Safety-Related Occurrences in Nuclear Facilities as Reported in 1972, R. L. Scott and R. B. Gallaher, Dec. 1973	$15.00
110	Indexed Bibliography of Thermal Effects Literature-3, J. G. Morgan, July 1973	$12.50
111	Reactor Protection Systems: Philosophies and Instrumentation, Reviews from *Nuclear Safety*, E. W. Hagen, July 1973	$15.00
112	Compilation of Nuclear Standards, 9th Edition, 1972, Part I: United States Activities, J. P. Blakely, Oct. 1973	$12.50
113	A Selected Bibliography on Emergency Core Cooling Systems (ECCS) for Light-Water-Cooled Power Reactors (LWRs), Wm. B. Cottrell, Jan. 1974	$12.50
114	Annotated Bibliography of Safety-Related Occurrences in Nuclear Power Plants as Reported in 1973, R. L. Scott and R. B. Gallaher, Nov. 1974	$15.00
117	Protection of Nuclear Power Plants Against External Disasters, Wm. B. Cottrell, May 1975	$15.00
119	A Selected Bibliography on Pressure Vessels for Light-Water-Cooled Power Reactors (LWRs), Fred A. Heddleson, Jan. 1975	$15.00
120	Annotated Bibliography of Hydrogen Considerations in Light-Water Power Reactors, G. W. Keilholtz, Feb. 1976	$10.00
121	Reactor Operating Experiences, 1972-1974, U.S. Nuclear Regulatory Commission, Dec. 1975	$ 8.00

(Continued on Inside Back Cover)

NUREG/CR-0861
ORNL/NUREG/NSIC-165
Dist. Category AE

Contract No. W-7405-eng-26

Engineering Technology Division

SUMMARY AND BIBLIOGRAPHY OF SAFETY-RELATED EVENTS AT PRESSURIZED-WATER NUCLEAR POWER PLANTS AS REPORTED IN 1978

R. L. Scott
R. B. Gallaher

Manuscript Completed — August 6, 1979
Date Published — September 1979

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

NRC FIN No. B0126

Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
DEPARTMENT OF ENERGY

1132 086

Printed in the United States of America. Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161

CONTENTS

1132 088

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 Research and Technology 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
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. Fissin 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:

J. R. Buchanan, Assistant Director
Nuclear Safety Information Center
P.O. Box Y
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830

Telephone 615-574-0391
FTS number is 624-0391

1132 090

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).

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

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

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

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

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).

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).

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).

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).

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).

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).

12. R. L. Scott and R. B. Gallaher, *Annotated Bibliography of Safety-Related Events in Boiling-Water Nuclear Power Plants as Reported in 1977*, ORNL/NUREG/NSIC-149, (November 1978).

13. R. L. Scott and R. B. Gallaher, *Annotated Bibliography of Safety-Related Events in Pressurized-Water Nuclear Power Plants as Reported in 1977*, ORNL/NUREG/NSIC-150 (November 1978).

SUMMARY AND BIBLIOGRAPHY OF SAFETY-RELATED EVENTS AT PRESSURIZED-WATER NUCLEAR POWER PLANTS AS REPORTED IN 1978

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 1978. The 2084 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. 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). Three of the more interesting events that occurred during the year are reviewed in detail.

INTRODUCTION

This report (along with ORNL/NUREG/NSIC-164) is the eleventh 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 licensees of light-water reactors in the United States during the previous year. In particular, this report contains abstracts of 2084 events reported by licensees of pressurized-water-re... r nuclear power plants in the United States during 1978. The abstracts are presented on microfiche, which are filed in an envelope attached to the back cover of this report. The previous reports in the series[1–13] cover the period 1967 through 1977. In addition, five related NSIC reports[14–18] contain information on reactor operating experiences reported by the NRC (formerly the Atomic Energy Commission) for the period 1966 through 1977.

As a result of the continual growth in the number of facilities and consequently in the number of safety-related events reported each year,

1132 092

it has been necessary since 1975 to compile the abstracts of the events in two separate documents. The 1978 events occurring at boiling-water-reactor nuclear power plants are presented in ORNL/NUREG/NSIC-164, *Summary and Bibliography of Safety-Related Events at Boiling-Water Nuclear Power Plants as Reported in 1978.*

The reports of safety-related events abstracted in the bibliography were submitted by power plant licensees to the 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. 19). 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. 20).

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 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 the bibliography are arranged alphabetically by reactor name 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 occurrences (i.e., during operation, refueling, construction, or testing — both preoperational and surveillance). This 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 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 "Method of Indexing Licensee Event Reports" (p. 23), 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 also indicated, that is, the NRC Public Document Room, 1717 H Street, Washington, D.C. 20555.

SUMMARY OF SAFETY-RELATED EVENT DATA

The 2084 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, the main cooling system and the reactor protection system were the dominant systems involved in the events reported. Combined, these systems accounted for 19% of the total number of 1978 reports. The secondary cooling (steam) system was also involved in a substantial number of reports, followed closely by the electric power system.

Table 2 lists the number of reports concerned with various types of equipment. Valves, pumps, and piping were the equipment items most frequently involved in the events reported, accounting for 43% of the reports. Valves accounted for 21% of the total number of reports, pumps accounted for 13%, and piping accounted for 9%.

Table 3 lists the number of reports concerned with the listed instrumentation. Again, this year, as in every year since 1972, switches

Table 1. Number of reports concerned with the listed systems

System	Percent of total number of reports	Number of reports
Main cooling	11	227
Reactor protection	8	176
Secondary cooling	8	176
Electric power	8	160
Feedwater	7	151
Emergency electric power	6	124
Containment isolation	6	115
Condenser cooling	5	96
Coolant purification	4	85
Radiation monitoring	4	84
Engineered safety features	4	83
Reactor control	4	78
Safety injection	4	75
Service water	3	61
Shutdown cooling	3	61
Fire protection	3	56
Ventilation	3	53
Containment spray	2	52
Containment	2	47
Pneumatic	2	35
Emergency cooling	2	32
Waste disposal	1	28
Containment air cooling	1	27
Containment purge	1	23
Component cooling	1	22
Hydraulic	<1	7
Core reflooding	<1	4
Containment filtering	<1	3

accounted for more reports than any other instrument. In 1978, switches were involved in 285 of the events reported, accounting for 14% of the total number of reports. Lagging far behind were radiation monitors, level sensors, pressure sensors, and relays, 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 failures were involved in 47% of the reports; these are failures for which there was no obvious reason. Examples of these types of events

Table 2. Number of reports concerned with the listed equipment

Equipment	Percent of total number of reports	Number of reports
Valves	21	434
Pumps	13	261
Pipes and pipe fittings	9	194
Seals	7	140
Storage containers	6	121
Diesel generators	6	118
Support structure	5	113
Internal combustion engines	5	107
Cables and connectors	5	98
Valve operators	5	96
Steam generators	4	93
Breakers	4	88
Control rods	3	72
Pressurizer	3	68
Shock absorber	2	48
Fastener	2	45
Control rod drives	2	38
Heat exchangers	2	38
Solenoid	2	37
Blowers	2	36
Motors	2	36
Bearings	1	29
Turbines	1	27
Filters, screen	1	26
Heaters	1	26
Tubing	1	25
Batteries and chargers	1	24
Check valves	1	23
Condensers	1	22
Pressure vessels	1	22
Generators	<1	17
Transformers	<1	15
Accumulators	<1	14
Fuel elements	<1	14
Cooling tower	<1	8
Crane	<1	7
Flanges	<1	7
Nozzle	<1	7
Demineralizer	<1	6
Air driers	<1	4
Recombiners	<1	1

Table 3. Number of reports concerned with the listed instrumentation

Instrumentation	Percent of total number of reports	Number of reports
Switch	14	285
Radiation monitors	5	108
Level sensors	4	90
Pressure sensors	4	76
Relays	4	72
Electronic function units	3	66
Flow sensors	3	60
Position instrument	3	53
Power-range instrument	2	41
Temperature sensors	2	33
Solid state device	1	22
In-core instrument	<1	18
Alarm	<1	17
Indicators	<1	14
Start-up-range instrument	<1	13
Recorders	<1	13
Amplifier	<1	12
Meteorological instrument	<1	12
Intermediate-range instrument	<1	9
Seismic instrument	<1	8
Failed-fuel detection instrument	<1	2

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

Cause	Percent of total number of reports	Number of reports
Inherent failure	47	975
Operator error	11	227
Design error	10	211
Administrative error	9	184
Maintenance error	6	135
Installation error	5	112
Fabrication error	4	86
Weather	1	27
Lightning	<1	3

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 93% of the reports; the remaining 7% 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 be noted that those items discovered during testing could be remedied with little or no effect on reactor operation.

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	1095
Testing	31	644
Construction	9	187
Refueling	8	158

Table 6 is a list of deficiencies considered to be of interest and the number of reports associated with each one. The most frequently reported deficiency is "leak," which includes any type of leak, such as water or steam from pipes, valves, or fittings. A deficiency in communication covers those events involving a misunderstanding between personnel; it also includes misinterpretation 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, 76 nuclear units are represented in this bibliography, which contains 2084 abstracts of reports. For the 38 nuclear units which were operational all year, there are 1613

Table 6. Number of reports concerned with the listed deficiency

Deficiency	Percent of total number of reports	Number of reports
Leak	9	181
Procedures	8	169
Set point drift	7	139
Instrument calibration	6	123
Weld	4	73
Crud	3	68
Communication	3	53
Crack	2	48
Lubrication	2	48
Vibration	2	44
Wear	2	34
Age	1	28
Corrosion	1	26
Records	1	26
Stress	<1	19
Airborne release	<1	14
Fatigue	<1	7
Stress corrosion	<1	6
Erosion	<1	4
Fire	<1	4

reports — an average of 42 reports per unit (compared to 34 in 1976 and 1977). For the 4 units in the power-ascension stage, there were 355 reports — an average of 89 reports per unit. For the 34 units which were under construction, there are 244 reports — an average of 7 reports per unit. It should be pointed out that Table 7 indicates that there are 2212 reports, whereas the bibliography contains abstracts of 2084 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 8 and 9 tabulate the number of reports submitted for the listed units which were commercially operable all year. In Table 8 the tabulation is by age; in Table 9 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 reportable events occurring at a nuclear unit. Both age and power appear to be factors, although it may not be readily apparent from just a cursory inspection of the tables.

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	35	4.4	850
Beaver Valley 1	3	62	2.6	852
Calvert Cliffs 1	3	57	4.0	845
Calvert Cliffs 2	2	48	2.1	845
Cook 1	3	71	3.9	1054
Crystal River 3	5	102	1.9	825
Davis-Besse 1	6	126	1.3	906
Farley 1	5	97	1.4	829
Fort Calhoun	2	46	5.4	457
Ginna	<1	8	9.1	490
Haddam Neck	2	34	11.4	575
Indian Point 2	2	37	5.5	873
Indian Point 3	2	39	2.7	873
Kewaunee	2	40	4.7	535
Maine Yankee	1	28	6.1	790
Millstone 2	1	29	3.1	828
Oconee 1	1	27	5.7	887
Oconee 2	<1	14	5.1	887
Oconee 3	<1	19	4.3	887
Palisades	2	45	7.0	668
Point Beach 1	<1	18	8.2	497
Point Beach 2	<1	11	6.4	497
Prairie Island 1	1	30	5.1	530
Prairie Island 2	1	23	4.0	530
Rancho Seco	<1	19	4.2	913
Robinson 2	2	34	8.3	712
Salem 1	4	77	2.0	1090
San Onofre 1	<1	19	11.5	430
St. Lucie 1	2	49	2.7	802
Surry 1	2	50	6.5	822
Surry 2	2	47	5.8	822
Three Mile Island 1	1	24	4.5	819
Trojan	2	36	3.0	1130
Turkey Point 3	<1	16	6.2	693
Turkey Point 4	<1	14	5.5	693
Yankee Rowe	2	41	18.1	175
Zion 1	5	94	5.5	1040
Zion 2	2	47	5.0	1040

1132 100

Table 7 (continued)

Name	Percent of total number of reports	Number of reports	Age (years)	Design electrical rating [net MW(e)]
	In power ascension part of year			
Arkansas Nuclear 2	2	51		
Cook 2	5	103		
North Anna 1	7	138		
Three Mile Island 2	3	63		
	Under construction all year			
Beaver Valley 2	<1	5		
Bellefonte 1	<1	11		
Bellefonte 2	<1	13		
Braidwood 1	<1	1		
Braidwood 2	<1	1		
Byron 1	<1	3		
Byron 2	<1	3		
Callaway 1	<1	4		
Catawba 1	<1	6		
Catawba 2	<1	6		
Diablo Canyon 1	<1	3		
Diablo Canyon 2	<1	2		
Farley 2	<1	3		
McGuire 1	<1	10		
McGuire 2	<1	10		
Midland 1	<1	10		
Midland 2	<1	10		
North Anna 2	1	28		
North Anna 3	<1	14		
North Anna 4	<1	14		
Salem 2	<1	3		
San Onofre 2	<1	7		
San Onofre 3	<1	6		
Seabrook 1	<1	1		
Seabrook 2	<1	1		
Sequoyah 1	<1	17		
Sequoyah 2	1	21		
South Texas 1	<1	2		
South Texas 2	<1	2		
St. Lucie 2	<1	1		
Tyrone 1	<1	1		
Waterford 3	<1	2		
Watts Bar 1	<1	14		
Watts Bar 2	<1	9		

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

1132 101

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

Name	Age[a] (years)	Percent of total number of reports	Number of reports
Yankee Rowe	18.1	2	41
San Onofre 1	11.5	<1	19
Haddam Neck	11.4	2	34
Ginna	9.1	<1	8
Robinson 2	8.3	2	34
Point Beach 1	8.2	<1	18
Palisades	7.0	2	45
Surry 1	6.5	2	50
Point Beach 2	6.4	<1	11
Turkey Point 3	6.2	<1	16
Main Yankee	6.1	1	28
Surry 2	5.8	2	47
Oconee 1	5.7	1	27
Indian Point 2	5.5	2	37
Turkey Point 4	5.5	<1	14
Zion 1	5.5	5	94
Fort Calhoun	5.4	2	46
Oconee 2	5.1	<1	14
Prairie Island 1	5.1	1	30
Zion 2	5.0	2	47
Kewaunee	4.7	2	40
Three Mile Island 1	4.5	1	24
Arkansas Nuclear 1	4.4	2	35
Oconee 3	4.3	<1	19
Rancho Seco	4.2	<1	19
Calvert Cliffs 1	4.0	3	57
Prairie Island 2	4.0	1	23
Cook 1	3.9	3	71
Millstone 2	3.2	1	29
Trojan	3.0	2	36
Indian Point 3	2.7	2	39
St. Lucie 1	2.7	2	49
Beaver Valley 1	2.6	3	62
Calvert Cliffs 2	2.1	2	48
Salem 1	2.0	4	77
Crystal River 3	1.9	5	102
Farley	1.4	5	97
Davis-Besse 1	1.3	6	126

[a]Average age — 5.4; median age — 5.1.

1132 102

Table 9. 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	2	36
Salem 1	1090	4	77
Cook 1	1054	3	71
Zion 1	1040	5	94
Zion 2	1040	2	47
Rancho Seco	913	<1	19
Davis-Besse 1	906	6	126
Oconee 1	887	1	27
Oconee 2	887	<1	14
Oconee 3	887	<1	19
Indian Point 2	873	2	37
Indian Point 3	873	2	39
Beaver Valley 1	852	3	62
Arkansas Nuclear 1	850	2	35
Calvert Cliffs 1	845	3	57
Calvert Cliffs 2	845	2	48
Farley	829	5	97
Millstone 2	828	1	29
Crystal River 3	825	5	102
Surry 1	822	2	50
Surry 2	822	2	47
Three Mile Island 1	819	1	24
St. Lucie 1	802	2	49
Maine Yankee	790	1	28
Robinson 2	712	2	34
Turkey Point 3	693	<1	16
Turkey Point 4	693	<1	14
Palisades	668	2	45
Haddam Neck	575	2	34
Kewaunee	535	2	40
Prairie Island 1	530	1	30
Prairie Island 2	530	1	23
Point Beach 1	497	<1	18
Point Beach 2	497	<1	11
Ginna	490	<1	8
Fort Calhoun	457	2	46
San Onofre 1	430	<1	19
Yankee Rowe	175	2	41

[a]Average DER — 801; median DER — 824.

The total number of reports for the 19 oldest reactors was 613, whereas the number of reports for the 19 most recently built reactors was 1000 — 63% 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 19 smallest units was 577, whereas the number of reports for the 19 largest units was 1036 — 80% 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 are 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-power, older reactor will probably have fewer problems than a high-power, 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 2084 reports, 67 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. Three events that were considered to be the most interesting are presented here to illustrate the types of experiences that occurred in 1978.

Failure of Burnable-Poison-Rod Assemblies at Crystal River 3

On Dec. 12, 1977, an alarm sounded indicating a possible power tilt at Crystal River Nuclear Plant, Unit 3, a pressurized-water-reactor (PWR) operated by the Florida Power Corporation. Another alarm sounded on

Jan. 1, 1978, followed sporadically during the next few days by other alarms from the loose-parts monitoring system. Finally, on Feb. 17, 1978, the loose-parts system initiated a continuous alarm indicating the presence of loose parts in steam generator B. To minimize potential damage from any loose part in the primary coolant system, the operators reduced reactor power and shut off one of the two reactor coolant pumps in the affected loop.[21,22]

On Mar. 3, 1978, the reactor was shut down for inspection of steam generator B. Several parts of a burnable-poison-rod assembly were found in the steam generator, and the steam generator tubes at the tube-to-tube-sheet welds were damaged. There were indications of a small leak from the primary to the secondary cooling system. Burnable-poison-rod assembly parts were also found in the core-support assembly, in various fuel assemblies, and in the plenum and bottom of the reactor vessel. It was necessary to completely defuel the reactor to facilitate inspection.

The cause of the failure was attributed to wear of the coupling mechanism's positive latch to the fuel assemblies. The burnable-poison-rod assemblies were all removed from the core and will not be replaced, because they are no longer needed to help control the larger reactivity existing during the early stages of core life.

Similar burnable-poison-rod assemblies were also in use at Davis-Besse, Unit 1, and are still in use at Three Mile Island, Unit 2. The burnable-poison-rod assemblies were removed from Davis-Besse, Unit 1, and were not replaced. However, the burnable-poison-rod assemblies were still needed to help control core reactivity at Three Mile Island, Unit 2; therefore, retaining collars were added to the assemblies to keep them in place.

The steam generator at Crystal River, Unit 3, was repaired, and the station was placed back in service but only after a lengthy shutdown of seven months.

Station Blackout at Beaver Valley

On July 28, 1978, the main transformer failed at the Beaver Valley Power Station, a PWR operated by the Duquesne Light Company. As a consequence, the generator, turbine, and reactor tripped, and safety injection occurred. Also, all off-site power was lost. The portions of the plant associated with both the primary and the secondary systems were placed in a hot shutdown, using emergency diesel generator power.[23]

The blackout of the station was caused by the following series of events. The transformer failure caused the following relays to trip: main generator differential, main generator differential auxiliary, main transformer differential, main transformer differential auxiliary, main generator ground overcurrent, and main generator ground backup overcurrent. When these relays tripped, the station service buses successfully transferred to the off-site sources — two 138-kV buses in the nearby Shippingport Power Station switchyard. Fifteen seconds later, as the Beaver Valley turbine generator coasted down while still supplying fault current to the short circuit, the main generator "out-of-step" relays operated — when they should not have — and caused isolation of the Beaver Valley 345-kV switchyard by tripping all 345-kV line breakers and the tie to the 138-kV autotransformer. In addition, three of the five 138-kV breakers in the 138-kV Shippingport switchyard tripped; when this occurred, the Shippingport Power Station was the only power source on the 138-kV bus supplying power to the two remaining 138-kV loads and the Beaver Valley Power Station system transformers. However, this total load exceeded the capacity of the Shippingport Power Station, so the frequency declined to approximately 58 Hz.

The Shippingport Power Station was then manually tripped; this removed the last source of power to Beaver Valley. Immediately, the No. 1 emergency diesel generator came on the line and supplied power to one of the redundant tra[illegible] of safety-related equipment, but the No. 2 emergency diesel failed to [illegible] a field" and, although running at nominal speed, did not provide power to the redundant emergency bus. About 15 min into the blackout, the field was manually "flashed" on the second diesel, and it assumed its load.

No adverse conditions developed in the reactor core as a result of the station blackout; although there was no forced flow in the reactor coolant system during the blackout, heat was removed from the reactor core by thermal circulation and transferred to the steam generators. The reactor coolant system was not adversely affected by the blackout either;

the water remained subcooled and at near-normal level in the pressurizer, and the system was not overpressurized. The main transformer was returned to the vendor's shop for rewinding.

Common-Cause Incident Involving Nonnuclear Instrumentation

On Mar. 20, 1978, a short to ground in nonnuclear instrumentation at Rancho Seco resulted in a reactor trip and a subsequent reactor-cooling-system (RCS) cooldown rate that exceeded the technical specifications limit.[24] This PWR is owned by the Sacramento Municipal Utility District, Sacramento, Calif.

Before the cooldown transient, the plant was operating at a power level of 70%, with all four reactor coolant pumps operating and an average RCS temperature of 306°C. Shortly before 4:25 AM, a control room operator began replacing a burned-out light bulb in a back-lighted push-button switch assembly on one of the control consoles. The dc power for this switch is provided from the "Y" portion of the nonnuclear instrumentation (NNI-Y). During replacement of the light bulb, the assembly was pulled from the panel and flipped down, exposing the bulbs. A bulb was dropped into the open light-assembly cavity, creating a short to ground, whereupon the current-limiting and undervoltage protection for the NNI-Y dc power supplies was actuated, cutting off the ac power to all NNI-Y dc power supplies. Preliminary investigations have shown that approximately two-thirds of the NNI signals (pressure, temperature, level, flow, etc.) were affected. The inevitable erroneous signals gave faulty information to both the control room and the integrated control system (ICS). Attempting to match equipment output with plant requirements, the ICS reduced main feedwater flow to zero, which caused the RCS pressure to increase. A reactor trip from high pressure followed.

After the reactor trip, the operators were still hampered by the lack of instrumentation and by equipment responding to inaccurate signals. For approximately 9 min following the trip, pressure slowly decayed to about 13.8 MPa (2000 psig) in the RCS. It has been postulated that the pressure remained fairly adequate during this period owing to the cooling provided by makeup flow into the RCS and to the lifting of a pressurizer code safety valve below its set point of 17.5 MPa (2500 psig). On the loss of feedwater flow, an auxiliary feedwater pump had started; however, the auxiliary feedwater valves remained closed in response to erroneous level signals from the once-through steam generators — another effect of the NNI-Y dc power failure. The level signal for steam generator A drifted to zero over a 9-min period, whereas that for steam generator B drifted full scale. In reality, both steam generators boiled dry during this period. When the startup level for steam generator A drifted below the low-level set point, the ICS opened the auxiliary feedwater valve, admitting water to the shell side. This influx of water created a heat sink for the RCS, causing a rapid pressure drop. The operators also may have increased the main-feed-pump flow at this time, and this provided another source of water for steam generator A. The RCS pressure dropped rapidly to 11.1 MPa (1600 psig), at which point both auxiliary feedwater bypass valves opened automatically and began filling both steam generators with water.

Until power was restored to NNI-Y approximately 1 hr and 10 min after the reactor trip, the operators continued the injection of auxiliary feedwater that was started automatically. Since it did not appear that the RCS temperature indication was reliable, the operators maintained RCS pressure as well as possible, using the pressurizer level indication and the RCS pressure indication that was available. These two parameters were controlled by adjusting the high-pressure injection flow. Unfortunately, the pressurizer heaters were not available for pressure control because of the NNI power loss. As injection of auxiliary feedwater continued, both steam generators were completely filled, and water began to enter the steam lines. This large heat sink continued to rapidly cool the RCS, unknown to the operators who had no information on the temperature.

When power was finally restored to NNI-Y, the operators realized that the RCS temperature had dropped to $\sim$141°C and that the technical specifications had been violated. Immediate action was taken to return to the permissible operating region: spraying the pressurizer to reduce pressure, keeping three reactor cooling pumps operating (pump combinations were changed) to increase the temperature, shutting off auxiliary feedwater flow, and draining the steam generators.

The short caused by the light bulb drew excessive current through the 24-V dc power supplies that service components in NNI cabinets 5, 6, and 7. The power for these cabinets is designated NNI-Y, while the power for cabinets 1, 2, 3, and 4 is designated as NNI-X. The four power supplies for NNI-Y are operated current-limited with a set point of 7.5 A. The subsequent reduction in voltage caused an undervoltage monitor to operate, opening the two shunt breakers through which ac power from inverter D and inverter J is supplied to the dc power supplies. Loss of these power supplies meant that every component in cabinets 5, 6, and 7 operating on dc power was not functioning properly. An NNI signal could have been affected two ways between its source and the receiving component. The signal could be interrupted completely owing to a contact opening when it was de-energized. Because most of the signals are -10 V to +10 V, this would have resulted in a midscale reading or, in some cases, a reading anywhere between -10 V and + 10 V being transmitted to the indicator or sent to the ICS as an actual plant parameter. If a signal-conditioning component (buffer amplifier, square root extractor) was affected, this would have meant that the desired conditioning would not have been performed on the signal or that the component might not have passed the true signal so that erroneous values would have been sent to the indicator or to the ICS. Since signal paths in the NNI are not restricted to either the X or Y cabinets, about two-thirds of the signals passed through at least one component in cabinets 5, 6, or 7 and therefore would have been invalid.

Practically no permanent records of the plant parameters during the transient were kept. A major source of information was the posttrip transient review, which prints out selected data periodically following a reactor trip. It was not possible to extensively analyze this and the other data available (recorder outputs, hourly log, etc.) during the transient. Over a period of several days following the incident, the plant engineers were able to trace which signals were valid, determine what equipment operated at which times, and then interpolate a temperature trace which indicated that the RCS temperature fell from about 312 to

141°C in slightly more than 1 hr. This cooldown rate of approximately 150°C/hr is well above the permitted rate of 40°C/hr.

To ensure that all components of the plant were in satisfactory condition, Babcock & Wilcox (B&W) personnel evaluated the effects of the transient on the reactor vessel, the reactor coolant piping, the pressurizer, the once-through steam generators, the fuel assemblies, the reactor cooling pumps and seals, and the control-rod-drive mechanisms. The B&W team recommended to the Nuclear Regulatory Commission (NRC) that Rancho Seco be permitted to return to power under certain specified conditions.

The Plant Review Committee reviewed the B&W recommendations and requested that a special test procedure and a casualty procedure be written to ensure compliance. At the committee's request, the following tasks were also completed prior to start-up:

1. Because of possible damage to steam lines from the injection of water, they were checked for any deformations.

2. A 15.5-MPa (2255-psig) leak test was performed on the reactor cooling system.

3. The overvoltage trip set points on the NNI dc power supplies were increased from 27 V to 29 V to prevent spurious trips.

The special test procedure addressed conditions such as reactor maneuvering limits for the first start-up, increased surveillance of the loose-parts monitors for a week, an operability check of on-line and redundant NNI, and daily surveillance of the primary and secondary radiochemistry for a week to check for leaking components. The casualty procedure was written to provide required operator actions for restoration of NNI power following a trip similar to the one experienced. The Plant Review Committee also required that a procedure be written giving operator instructions if NNI power cannot be restored.

On Friday, Mar. 24, the reactor was again taken to criticality, and the initial power ascension began.

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 off-normal incidents at nuclear power plants. In this way, the safety, reliability, and availability of nuclear facilities should be improved.

REFERENCES

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

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

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

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

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

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).

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).

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).

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).

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).

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).

12. R. L. Scott and R. B. Gallaher, *Annotated Bibliography of Safety-Related Events in Boiling-Water Nuclear Power Plants as Reported in 1977*, ORNL/NUREG/NSIC-149 (November 1978).

13. R. L. Scott and R. B. Gallaher, *Annotated Bibliography of Safety-Related Events in Pressurized-Water Nuclear Power Plants as Reported in 1977*, ORNL/NUREG/NSIC-150 (November 1978).

14. Oak Ridge National Laboratory, Nuclear Safety Information Center (Comp.), *Abnormal Reactor Operating Experiences*, ORNL/NSIC-17 (August 1966).

1132 112

15. Oak Ridge National Laboratory, Nuclear Safety Information Center (Comp.), *Abnormal Reactor Operating Experiences 1966–1968*, ORNL/NSIC-64 (October 1969).

16. Oak Ridge National Laboratory, Nuclear Safety Information Center (Comp.), *Abnormal Reactor Operating Experiences 1969–1971*, ORNL/NSIC-103 (May 1972).

17. Oak Ridge National Laboratory, Nuclear Safety Information Center (Comp.), *Reactor Operating Experiences 1972–1974*, ORNL/NSIC-121 (December 1975).

18. Oak Ridge National Laboratory, Nuclear Safety Information Center (Comp.), *Reactor Operating Experiences 1975–1977*, ORNL/NUREG/NSIC-144 (September 1978).

19. U.S. Nuclear Regulatory Commission, *Reporting of Operating Information – Appendix A, Technical Specifications*, NRC Regulatory Guide No. 1.16, Rev. 4, August 1975.

20. 10 CFR 50.55(e), "Conditions of Construction Permits," *Fed. Reg.* 37(62), 6460 (Mar. 30, 1972).

21. T. Oldt, "Troubled Crystal River Nuclear Unit Puts Florida Power in Hot Water," *The Energy Daily*, 6(204), 1–2 (Oct. 23, 1978).

22. Nuclear Regulatory Commission, Office of Management and Program Analysis, *Report to Congress on Abnormal Occurrences, April–June 1978*, NUREG-0090, Vol. 1, No. 2 (September 1978).

23. Letter from J. A. Werling, Duquesne Light Company, Pittsburgh, Pa., to B. H. Grier, Director of Regulation, Nuclear Regulatory Commission, Region 1, King of Prussia, Pa., Aug. 11, 1978; Subject: Beaver Valley Power Station, Unit 1, Docket 50-334, LER 78-43/01T-0.

24. Letter from A. J. Mattimore, Assistant General Manager and Chief Engineer, Sacramento Municipal Utility District, to Director of Regulatory Operations, Nuclear Regulatory Commission Operations Office, Region V, Walnut Creek, Calif., Mar. 31, 1978; Subject: Reportable Occurrence 78-1, Docket 50-312.

METHOD OF INDEXING LICENSEE EVENT REPORTS

DOCUMENT TITLE

CORPORATE AUTHOR

ACCESSION NUMBER – SERIAL NUMBER ASSIGNED TO THE DOCUMENT BY NSIC

DOCUMENT DESCRIPTION, BIBLIOGRAPHIC DATA, AND INFORMATION ON REACTOR DOCKET

138817
HPCI STATIC INVERTER TRIPS AT PEACH BOTTOM 2
PHILADELPHIA ELECTRIC CO., PA
1 PG, LTR W/LER 78-022/03L-0 TO NRC OFFICE OF I&E, REGION I, MAY 23, 1978,
DOCKET 50-277, TYPE--BWR, MFG--GE, AE--BECHTEL

DATE OF EVENT – 042478. POWER LEVEL – 97%. CAUSE – NOT DETERMINED. WHILE AT POWER, THE HPCI SUBSYSTEM WAS DECLARED INOPERABLE FOLLOWING A POWER FAILURE ALARM. INVESTIGATION REVEALED THAT THE HPCI STATIC INVERTER (TOPAZ MODEL 250GWR-125-60-115) HAD TRIPPED. THE INVERTER WAS REPLACED AND HPCI TESTED AND DECLARED OPERABLE WITHIN 7 HOURS. THE INVERTER WAS BENCH TESTED FOR TWO WEEKS AT VARIOUS LOADS. THE TESTING COULD NOT DUPLICATE THE CONDITION THAT CAUSED THE TRIPPING NOR COULD ANY FAULT BE FOUND WITH THE INVERTER.

ABSTRACT OF ORIGINAL DOCUMENT

AVAILABILITY – NRC PUBLIC DOCUMENT ROOM, 1717 H STREET, WASHINGTON, D. C. 20555 (08 CENTS/PAGE –– MINIMUM CHARGE $2.00)

FAILURE + REACTOR, BWR + PEACH BOTTOM 2 (BWR) + HPCI + ELECTRIC POWER + FAILURE, EQUIPMENT + GENERATORS

DOCUMENT AVAILABILITY

KEYWORDS – LISTED ALPHABETICALLY IN THE *KEYWORD INDEX* WITH THE ACCESSION AND PAGE NUMBERS

NUREG/CR-0861
ORNL/NUREG/NSIC-165
Dist. Category AE

Internal Distribution

1. Seymour Baron
2. E. D. Blakeman
3. J. R. Buchanan
4. C. A. Burchsted
5. W. R. Casto
6. R. O. Chester
7. W. B. Cottrell
8. C. E. Davis
9. R. B. Gallaher
10. C. T. Garten
11. E. W. Hagen
12. P. M. Haas
13. F. A. Heddleson
14. M. Heiskell
15. R. D. Hurt
16. W. H. Jordon
17. Milton Levenson
18. G. T. Mays
19. F. R. Mynatt
20. D. S. Queener
21. J. L. Rich
22. R. L. Scott
23. S. D. Swisher
24. H. E. Trammell
25. D. B. Trauger
26. M. L. Winton

27-28. Central Research Library
29. Y-12 Document Reference Section (DRS)
30. Laboratory Records Department
31-49. Nuclear Safety Information Center (NSIC)

External Distribution

50. Office of Assistant Manager, Energy Research and Development, Oak Ridge Operations Office, Oak Ridge, TN 37830

51-52. DOE Technical Information Center, P.O. Box 62, Oak Ridge, TN 37830

53-452. Given distribution as shown in Category AE (NTIS — 10)

1132 115

☆U.S. GOVERNMENT PRINTING OFFICE: 1979-640-079-174

NUCLEAR SAFETY

A BIMONTHLY REVIEW JOURNAL PREPARED BY NSIC

Nuclear Safety covers significant developments in the field of nuclear safety.

The scope is limited to topics relevant to the analysis and control of hazards associated with nuclear reactors, operations involving fissionable materials, and the products of nuclear fission.

Primary emphasis is on safety in reactor design, construction, and operation; however, safety considerations in reactor fuel fabrication, spent-fuel processing, nuclear waste disposal, handling of radioisotopes, and related operations are also treated.

Qualified authors are invited to submit interpretive review articles, which will be reviewed for technical accuracy and pertinency. Authors will be advised as soon as possible of acceptance or suggested changes. Send inquiries or 3 copies of manuscripts (with the draftsman's original line drawing plus 2 copies, and with continuous-tone glossy prints of photographs plus 2 copies) to Wm. B. Cottrell, Oak Ridge National Laboratory, P.O. Box Y, Oak Ridge, Tennessee 37830.

Nuclear Safety is prepared by the Nuclear Safety Information Center at Oak Ridge National Laboratory for the U.S. Nuclear Regulatory Commission and the U.S. Department of Energy. For subscriptions, address Superintendent of Documents, U.S. Government Printing Office, Washington, D.C. 20402. The subscription rate is $14.00 per year. Below is an order blank for your convenience.

U.S. GOVERNMENT PRINTING OFFICE
DIVISION OF PUBLIC DOCUMENTS
WASHINGTON, D.C. 20402

OFFICIAL BUSINESS

RETURN AFTER 5 DAYS

POSTAGE AND FEES PAID
U.S. GOVERNMENT PRINTING OFFICE

Name ____________________
Street ____________________
City __________ State __________ ZIP __________

To Ensure Prompt, Accurate Shipment, Please Print or Type Address on Mailing Label Above

MAIL ORDER FORM TO:

Superintendent of Documents, U.S. Government Printing Office, Washington, D.C., 20402

Enclosed find $ *(check, money order, or Superintendent of Documents coupons).*

Please send me *subscriptions to* Nuclear Safety *at $14.00 per subscription. (Single issues are sold at $2.40 per issue.)*

Please charge this order to my Deposit Account No. __________

Name ____________________
Street ____________________
City __________ State __________ ZIP __________

FOR USE OF SUPT. DOCS.
__Enclosed ______
To be mailed
__later ______
__Subscription ______
Refund ______
Coupon refund ______
Postage ______

POSTAGE AND REMITTANCE: Postpaid within the United States, Canada, Mexico, and all Central and South American countries except Argentina, Brazil, Guyana, French Guiana, Surinam, and British Honduras. For these and all other countries, add $3.50 for each annual subscription; for a single issue, add one-fourth of the single-issue price. Payment should be by check, money order, or document coupons, and MUST accompany order. Remittances from foreign countries should be made by international money order or draft on an American bank payable to the Superintendent of Documents or by UNESCO book coupons.

DOCUMENT/ PAGE PULLED

ANO. 7910100711

NO. OF PAGES 1(microfiche)

REASON:

☒ PAGE ILLEGIBLE:

☐ HARD COPY FILED AT: PDR CF

OTHER ______

☐ BETTER COPY REQUESTED ON ___/___/___

☐ PAGE TOO LARGE TO FILM:

☐ HARD COPY FILED AT: PDR CF

OTHER ______

☐ FILMED ON APERTURE CARD NO. ______

1132 117