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REVIEW OF THE OPERATING EXPERIENCE HISTORY THROUGH 1984 OF MILLSTONE 1 FOR THE NUCLEAR REGULATORY COMMISSION'S INTEGRATED SAFETY ASSESSMENT PROGRAM

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REVIEW OF THE OPERATING HISTORY OF
MILLSTONE UNIT 1 THROUGH 1984

EXECUTIVE SUMMARY

The Systematic Evaluation Program Branch of the Nuclear Regulatory Commission (NRC) is conducting a pilot program for the Integrated Safety Assessment Program (ISAP), which will evaluate all pending licensing actions and safety issues for an operating reactor. Under it a new approach has been evolved to address a growing need to provide order and efficiency to the implementation and resolution of licensing requirements for operating nuclear power plants.

The new ISAP approach provides a structure for the regulatory management of licensing requirements on a plant-specific basis, assuring that the greatest safety measures will be accomplished in the near-term, and that efficient use of both NRC staff and licensee resources can be made. To accomplish this objective, ISAP will: (1) evaluate all applicable issues related to plant safety in accordance with a pre-established scope; (2) identify cost-effective corrective actions, where necessary, to enhance safety on a plant-specific basis; (3) establish a technical basis to judge implementation schedules, and (4) document the results of the evaluation so that the implementation schedule can be periodically updated, as necessary, to incorporate corrective actions for issues that may arise in the future.

Tools which will be utilized in the ISAP evaluation process include: (1) deterministic review of all pending licensing actions and safety issues; (2) a plant-specific Probabilistic Safety Analysis (PSA); and (3) an evaluation of plant operating experience and reliability data, including licensee performance (e.g., from existing SALP evaluations).

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Two plants volunteered to be included in the pilot program. These are Millstone 1 and Haddam Neck. As part of ISAP, the NRC contracted with the Oak Ridge National Laboratory's Nuclear Operations Analysis Center (NOAC) to perform operating history reviews for both the Millstone 1 and Haddam Neck Plants. These reviews will be used as an integral part of the ISAP evaluation process. Each review includes collection and evaluation of data on availability and capacity factors, forced shutdowns, forced power reductions, reportable events, environmental events, and radiological release events. This data is analyzed and evaluated to determine any trends and symptoms that can be discerned which will be important in the resolution of regulatory actions to be applied to the plant. Observations and conclusions are provided which focus on the key findings of the review.

This review for the Millstone plant incorporates a report previously generated under the Systematic Evaluation Program (SEP), NUREG-0824, "Integrated Plant Safety Assessment, Systematic Evaluation Program - Millstone Nuclear Power Station, Unit 1." The review presents the results of an updated assessment of the operating experience. Millstone Unit 1 Nuclear Power Plant is a General Electric-designed boiling-water reactor, owned and operated by Northeast Nuclear Energy Company. The plant is located in Waterford, Connecticut. The reactor has a licensed thermal power of 2011 MWt and a design electric rating of 660 MWe. Millstone 1 achieved initial criticality on October 26, 1970 and began commercial operation in December 1970.

The operating history review focused on data evaluation which was divided into two segments: (1) evaluation of forced shutdowns and power reductions and (2) evaluation of reportable events. Design basis events (DBEs), which are defined in the NRC's <u>Standard Review Plan</u>, are failures

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that initiate system transients and challenge engineered safety features. In the forced shutdown and power reduction segment, the review identified DBEs and recurring events that might indicate a potential operating concern. In the reportable event segment which included environmental events and radiological release events, the review identified significant events and recurring events that might indicate a potential operating concern. Significant events were either DBEs or events with a loss of engineered safety function.

Availability and Capacity Factors

From 1971 through 1984, the reactor availability factor at Millstone averaged 78.1%, and the unit capacity factor averaged 73.6%. The cumulative values were 75.2% and 74.6%, respectively, both of which are above average for commercial nuclear power plants. The reactor availability factor fell below 70% in only two years, 1973 and 1981. The major unit shutdowns in 1973 were for refueling and for feedwater sparger replacement. These two shutdowns combined for over five months of downtime. In 1981, two shutdowns, for refueling and for balancing of the turbine, again combined for over five months of downtime.

Of particular note, while not reflected in the data or the analysis, Millstone 1 accomplished 367 consecutive days of operation as of August 6, 1985. This is a world record, and reflects on the quality of operations of the plant.

Forced Shutdowns and Power Reductions

Of the 201 forced shutdowns and power reductions between 1971 and 1984 at Millstone 1, 55 were DBEs of 1 of the 11 following types:

1. turbine trip (34),

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2. steam pressure regulatory failure resulting in increased steam flow (3),
3. steam pressure regulator failure resulting in decreased steam flow (3),
4. loss of normal feedwater (3),
5. inadvertent opening of a safety or relief valve (3),
6. increased feedwater flow (3),
7. loss of external electric load (2),
8. inadvertent closure of main steam isolation valve (MSIV) (2),
9. decreased feedwater temperature (1),
10. loss of condenser vacuum (1), and
11. reactor recirculation pump trip (1).

Forty-nine of the fifty-six DBEs were the result of equipment failure. Human error caused the remaining seven events. In all DBEs, the engineered safety features operated properly to mitigate the transient.

DBEs averaged about 5 occurrences per year over the operating history at Millstone 1. The largest number of events in a single year (24) occurred in 1971. For the last three years 1982-1984, the total number of occurrences of DBE events has been only 2. The frequency of occurrence of each type of DBE is consistent with the experience of other plants except for turbine trips. Problems with moisture separator drain tank level control during power changes was the primary cause of turbine trips (21 of 34 events). The level control problem occurred less frequently over time with 14 events in 1971 and 1 event in 1981.

An increase in the rate of power reductions since 1980 resulted mainly from the need to locate and repair leaking main condenser tubes. Such leakage results in increased volumes of radioactive waste (radwaste)

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products which are generated. Thus when it became more difficult to dispose of radwaste products it was very important to reduce condenser tube leaks.

Reportable Events

A total of 404 reportable events included in the operating history of Millstone 1 were reviewed. The number of event reports submitted by Millstone 1 is generally stable with peak years of 1977, 1979, 1981, and 1983 with 38, 36, 44, and 38 events, respectively. These reportable events have primarily been caused by inherent equipment failures, which contributed to 56% of the total. Human error (including administrative, design, fabrication, installation, maintenance, and operator error) was identified in 43% of the reported events. Other causes, such as adverse environmental conditions, were reported for the remaining 1%. There is no apparent trend in the causes of reported events.

Significant Events

Of the 404 reported events, 13 are considered significant:

- loss of the isolation condenser (1),
- provisions for emergency core cooling during a loss of normal power lost (2),
- ECCS failures (2),
- loss of offsite power with partial loss of emergency power (1),
- complete and potential loss of emergency power (2),
- inadvertent criticality (1),
- all control rod drive accumulators require replacement (1),
- recirculation pumps trip with no alarm given (1),
- loss of pressure control followed by a blowdown (1), and
- hydrogen explosion in the off-gas system (1).

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Human error was reported to be the cause of 11 of the 13 events. The remaining 2 events were caused by equipment failures in the diesel generator and gas turbine generator. There have been no significant events since 1981.

Recurring failures of the emergency power system were reported. A gas turbine generator is one of two emergency power supplies at Millstone 1, the other being a conventional diesel generator. In the case of a loss of offsite power, the feedwater coolant injection system (FWCI), one loop of the low-pressure coolant injection and core spray systems are fed from the gas turbine generator. On two occasions in 1976 the gas turbine generator failed in coincidence with an inoperable isolation condenser. During a second loss of offsite power in 1976, the gas turbine again failed to run, and the diesel generator was then the only source of ac power. On December 1, 1977, both emergency power sources failed simultaneously. During design reviews in 1979 and 1981, two potential emergency power system failures were discovered in both cases. The possibility existed for loss of emergency power to emergency cooling systems. One case involved failure to sense a power loss, and the other involved a single relay failure which could disable both the gas turbine and the diesel generator.

Recurring Events

The following eight types of recurring events were noted during the two segments of the operating history review:

1. partial loss of emergency power,
2. excessive cooldowns,
3. pipe cracks,

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4. isolation condenser valve failures,
5. MSIV failures,
6. safety/relief valve setpoint drift,
7. stack gas monitor failures, and
8. radioactive levels in marine life.

The emergency power system at Millstone 1 consists of one diesel generator and one gas turbine generator. If normal power to the plant is lost, the gas turbine is the sole power source for the feedwater coolant injection (FWCI) system. The gas turbine generator failed to start or run for its entire mission 37 times. As discussed earlier, many of these failures occurred when redundant power systems or systems redundant to the FWCI system were not operable.

Millstone 1 experienced five excessive thermal transients in eight blowdowns due to safety and relief valve failures. The cooldown rates during the transients ranged from 105°F/h to 450°F/h. The first of these events occurred in 1971. Between 1975 and 1981 the transients occurred at a rate of about one every two years. No excessive cooldowns occurred during 1982, 1983, or 1984.

Millstone 1 reported ten instances of pipe cracks. Cracks appeared in feedwater spargers, head spray piping, main steam line supports, and condenser nozzles. Pipe cracking found at Millstone is typical of the generic problems found in many BWRs.

A variety of problems caused thirteen isolation condenser failures between 1970 and 1984. In seven of the thirteen events, a supply valve opened too wide, or failed to open, and caused an isolation condenser system failure. On one occasion, a valve transferred open and initiated the iso-

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lation condenser system. Another event occurred because a return valve failed to close. One event involved a valve motor failure. The final three isolation valve failures occurred due to limit switch problems.

There were eight failures of the main steam isolation valves (MSIVs). The predominate cause for MSIV failures involved poor quality control air to the pilot valves. This failure mechanism has the potential to affect more than one MSIV at a time. The last MSIV related failure occurred in 1980.

Millstone 1 experienced two events involving set point drift in safety/relief valves. A total of six valves were affected. This is a generic problem for BWR's.

On two occasions in 1982, Millstone 1 experienced a loss of the stack gas monitor due to operator error. A design change was made to include a timer and an alarm to the monitor to prevent recurrences of this problem. No events involving the stack gas monitor were reported after 1982.

Millstone 1 reported nine incidents of higher than allowed levels of radioactive silver and cobalt in marine life in the period from 1981 through 1984. The majority of occurrences were in 1982 and in 1983.

Plant Visit

As part of the review effort the ISAP operations assessment team visited the plant. Valuable insights were gained from discussions with plant personnel and a tour of the Millstone plant facilities. Plant personnel openly answered questions arising out of the analysis for which there was no documented information in the NOAC files. Visual inspections of the general plant conditions, the state of operating equipment and the areas where problems in operations had been experienced provided additional perspective to the team for their evaluations.

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Conclusions

This review identified no major challenges to plant safety. As evidenced by the above average availability and capacity factors and the relatively small number of significant reportable events, Millstone was judged to have a better than average operational record.

However, a number of problems were identified and eight areas of significant recurring problems were established. For these eight areas, the data showed that two were no longer recurring. Five of the problem areas reflect concerns raised in regulatory topics. The topics correlated to these problem areas are:

- USI A-44 Station Blackout
- USI A-42 Pipe Cracks in BWR's
- 10 CFR 50.49 Environment Qualification of Equipment
- SEP Topic VI-4 Containment Isolation System, and
- SEP Topic VIII-2 Onsite Power Systems

The majority of the operating experience problems was confined to random events, reflecting many problems experienced by other operating BWRs. No other significant problems were identified in the trends and patterns analysis. Thus no direct correlation to regulatory topics was attempted for individual events.

Operational review findings provide real indicators of importance which can be used in the evaluation and prioritization of regulatory topics which are to be applied to a plant. Such reviews can provide direct correlations between the operating experience and any regulatory topics which have bearing on it. Thus the importance of a given regulatory topic to a specific plant can be established from a review of that

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plants operating experience. Further, this correlation can be used to establish just how the regulatory topic should be resolved for that plant. If the topic is resolved upon giving full recognition to the operational experience findings, then there should be a direct consequential and positive result in the operational performance. Thus operational performance would provide a direct measure of the effectiveness by regulatory actions.

Accordingly, the results and conclusions from this operational experience review can be included as a basis for the prioritization of the five regulatory topics identified for Millstone 1. These five topics are justified through the operating experience as deserving a higher priority than do other regulatory topics. All other regulatory topics do not appear to qualify for Such a level of prioritization from this analysis.

ABSTRACT

A review of the operating experience of Millstone 1 nuclear power plant through 1984 was performed by the staff of the Nuclear Operations Analysis Center for the Nuclear Regulatory Commission's Integrated Safety Assessment Program (ISAP). ISAP is a new program. Under it a new approach has been evolved to address a growing need to provide order and efficiency to the implementation and resolution of licensing requirements for operating nuclear power plants. A significant element of the ISAP is a review and evaluation of the operating experience of Millstone 1. This review will provide significant insights into the strengths and weaknesses of the plants operations.

The review included collection and evaluation of data on availability and capacity factors, forced shutdowns, power reductions, reportable events (reportable occurrence, licensee event reports, etc.), and environmental considerations. The review methodology used in the review and evaluation are discussed. Data and information collected for forced shutdowns, power reductions, and reportable events are presented in appendices. A set of overall observations and conclusions are presented.

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1. INTRODUCTION

In the early 1980's the Nuclear Regulatory Commission (NRC) implemented a program of reviews of older operating commercial nuclear power plants. This program, called the Systematic Evaluation Program (SEP), grew out of a concern that some older plants were licensed under regulations that may have been less stringent than those in existence today. Specifically, SEP had five objectives:

1. The program established documentation that shows how the criteria for each operating plant reviewed compare with current criteria on significant safety issues, and provided a rationale for acceptable departures from these criteria.
2. The program provided the capability to make integrated and balanced decisions with respect to any required backfitting.
3. The program was structured for early identification and resolution of any significant deficiencies.
4. The program assessed the safety adequacy of the design and operation of currently licensed nuclear power plants.
5. The program used available resources efficiently and minimize requirements for additional resources by NRC or industry.

Through SEP, the NRC recognized a need to provide order and efficiency to the implementation and resolution of regulatory requirements for operating nuclear plants. The experience gained from SEP and the National Reliability Evaluation Program (NREP) enabled the NRC to develop a new and integrated approach called the Integrated Safety Assessment Program (ISAP). In early 1985, the NRC implemented a pilot program

for the ISAP which will examine two volunteer plants — Millstone 1 and Haddam Neck.

The features of ISAP lie in the integration of many different programs currently applied to the management and regulation of operating nuclear plants. Through an integrated evaluation of these programs, ISAP will enable a balanced and cost effective approach to the management of regulatory requirements on a specific basis for individual operating plants. Evaluation tools for ISAP will typically include: (1) the deterministic reviews of all pending licensing actions and safety issues; (2) a plant specific Probabilistic Safety Analysis (PSA); (3) an evaluation of plant operating experience; and (4) a review of the plant betterment programs developed by the utility.

In SEP it was found that plant Operating Experience Reviews contributed significant value to the program. These reviews compiled and evaluated data on availability/capacity factors, forced shutdowns and reportable occurrences. They provided additional perspective for the integrated assessment of the plant, and it was recognized that their contributions were equally as significant as those made by the safety topic evaluations and the risk analyses.

Thus, operating experience reviews are included as an integral part of the ISAP evaluation process and will be used to (1) confirm the adequacy of the data used in the plant specific PSA, and (2) highlight strengths and weaknesses in plant operation and maintenance which could be considered as factors for judging the adequacy of any proposed corrective actions to resolve an issue.

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The NRC contracted with the Nuclear Operations Analysis Center, part of the Oak Ridge National Laboratory, to perform the operating experience review portion of ISAP. This report presents the results of the operating experience review of Millstone 1. The review includes collection and evaluation of availability and capacity factors, forced shutdowns, forced power reductions, reportable events, environmental events, and radiological release events.

Section 2 of this report contains a brief description of the methodology employed in the review.

Section 3 discusses the results of the operating experience review. Section 3.1 describes the review of operational performance (availability and capacity factors), and environmental problems. Sections 3.2 and 3.3 describe the forced shutdowns (including power reductions) and reportable events, respectively. These two sections give yearly summaries as well as summaries by system, cause, and significance.

Section 4 presents based on observable trends or patterns in the operating experience of Millstone 1. It also discusses the relationship of the review topics with other topics which may be considered in the overall ISAP review.

Appendices A and B describe the coding of both shutdown and reportable events. They contain more detailed information about each event in tabular form.

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2. TECHNICAL APPROACH

This report presents the findings of a review of the operating experience at Millstone 1. It updates an earlier operating experience review conducted for the Systematic Evaluation Program (SEP). As a major difference from the SEP review, a goal of this report is to provide specific conclusions about the plants' actual operational performance which can be used in the overall assessment of the importance of regulatory issues applied to Millstone 1.

The objective of the review is to provide insights into the actual strengths and weaknesses of the design, operation and maintenance of the Millstone 1 plant. The results of the review will be used in an integrated approach with other ISAP tasks to establishing a manageable regulatory baseline from which can be generated a plant specific "living schedule" of plant modifications.

The evaluation of the operational history consisted of a four-step process: (1) compilation of information on plant operating events, including forced shutdowns and reportable occurrences, (2) screening of the events to determine their significance, using selected criteria and guidelines (3) evaluation and categorization of the events to facilitate an analytical search for trends in operating characteristics, and (4) making recommendations based on any trends or patterns identified by analysis, with special emphasis on operational performance.

Data was compiled on the following aspects of operation: availability and capacity factors, events of environmental importance including radioactivity releases, forced shutdown and power reduction events and reportable events.

The main focus in this evaluation was on forced shutdowns and power reductions, and on reportable events. Availability and capacity factors, and information about environmental events were used in establishing an overall perspective on plant operations. Procedures for the screening and categorizing of the information about these events that assured the consistency of the review were applied. After the screening and categorizing, a safety significance assessment of the events was made and the existence of any determinable trends or relationships was determined.

2.1 Availability and Capacity Factors

Both reactor and unit availability factors were compiled for all years. Starting with 1974, the unit capacity factors using the design electrical rating (DER) in net megawatts (electric) and the maximum dependable capacity (MDC) in net megawatts (electric) were compiled as well. Data for the capacity factors were not available from earlier years.

The two availability and two capacity factors are defined as follows:

1. Reactor Availability =

$$\frac{\text{hours reactor critical + reactor reserve shutdown hours}}{\text{period hours}} \times 100$$

2. Unit Availability =

$$\frac{\text{hours generator on line + unit reserve shutdown hours}}{\text{period hours}} \times 100$$

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3. $\text{DER Unit Capacity} = \frac{\text{net electrical energy generated}}{\text{period hours} \times \text{DER net}} \times 100$

4. $\text{MDC Unit Capacity} = \frac{\text{net electrical energy generated}}{\text{period hours} \times \text{MDC net}} \times 100$

The reserve shutdown hours are the amounts of time the reactor is not critical (or the unit is shutdown for administrative or other similar reasons) when operation could have been continued. The period hours are the total number of hours in the period under consideration.

2.2 Environmental Events

Any significant or recurring environmental problems were summarized based on the review of forced shutdowns, power reductions, reportable events (environmental LERs), and other operating reports. Routine radioactivity releases were tabulated and those releases where established limits were exceeded were reviewed.

2.3 Forced Shutdowns and Power Reductions

Data on forced shutdown and power reduction events were collected and reviewed for each incident. Forced shutdowns generally result from equipment failures or human errors that present an abnormal challenge to the unit's operation. Power reductions in general are caused by some need for maintenance or operations upgrade which does not require a full shutdown. The power reductions and forced shutdowns are included in chronological sequence in Appendix A.

Each shutdown or power reduction was placed in one of two sets of significance categories. The shutdown and power reductions were first evaluated against criteria for DBEs delineated in Chap. 15 of the *Standard Review Plan*.[2] If the shutdown or power reduction could not be

categorized as a DBE initiator, then it was placed in one of the Nuclear Operations Analysis Center (NOAC) categories. The method of assigning significance to and the coding of events is described in detail in Appendix A.

2.4 Review of Reportable Events

Information on operating events reported in licensee event reports (LERs) and LER predecessors [e.g., abnormal occurrence reports (AORs), unusual event reports, reportable occurrences (ROs)] was reviewed. This information was retrieved from the NOAC operational data files. Any documents that contained LER-type information (such as equipment failures or abnormal events) were coded or indexed so that they could be reviewed in the same manner as an LER. Primarily, this involved various types of operating reports and general correspondence for the early 1970s. Other sources of information such as *Nuclear Safety Journal* and reports from the NRC Office for the Analysis and Evaluation of Operational Data were also used.

Two sets of criteria were used in determining the significance of reportable events. The first set addressed those events whose results include challenges to the safety protection features of the plant; these events are termed "safety significant". The second set addressed events that have the potential to challenge the safety protection features of the plant. These events, which might require additional information or evaluation to determine their full implication, were termed "conditionally significant". The methods for assigning significance and for the coding of events are described in detail in Appendix B.

2.5 Evaluation of Operating Experience

The operating history of the plant was evaluated based on a review that involved screening, compiling, and categorizing data. Judgments and conclusions were made regarding safety problems, operations, trends (recurring problems), or potential safety concerns. Events were analyzed to determine their safety significance from the information provided through the various operating reports and the review process. The Final Safety Analysis Report (FSAR) and conversations with plant personnel provided specific plant and equipment details when necessary.

3. MILLSTONE 1 OPERATING EXPERIENCE

The review and assessment of the Millstone 1 operating experience for ISAP addressed the time from initial criticality through 1984. (The period from initial criticality through 1981 was included in the SEP review.) This report incorporated additional data for the period 1982 through 1984 and makes fresh observations, conclusions and recommendations.

Data was compiled on the following aspects of operation: availability and capacity factors, events of environmental importance including radioactivity releases, forced shutdown and power reduction events and reportable events.

Millstone Unit 1 is a General Electric Boiling Water Reactor (BWR) of 652 MWe net capacity. The plant is owned by Northeast Nuclear Energy Company and is located in Waterford, Connecticut. Ebasco Services Incorporated was the Architect/Engineer and the Constructor. The condenser cooling method is once-through, and Long Island Sound is the source of condenser cooling water (Fig. 3.1). The Plant is subject to license DPR-21, issued October 7, 1970, pursuant to Docket No. 50-245. The date of initial reactor criticality was October 26, 1970, and commercial power generation began in March 1971.

3.1 Operational Performance and Environmental Impact

3.1.1 Availability and capacity factors

Millstone 1 availability and capacity factor data is summarized in Table 3.1. Figure 3.2 shows that the reactor availability from 1971 through 1984 stayed above 70% with the exception of two years, 1973 and

1981, when major outages were necessary for repairs. In 1973, the feedwater spargers had to be replaced which necessitated an outage lasting almost 90 days. In 1981, after a seven month refueling outage, turbine balancing problems forced a 57 day repair outage. During the 14 full years of operation, 1971 through 1984, the plant had an average reactor availability of 78.1% and an average plant availability of 73.6%. Capacity factors were not available prior to 1973. The MDC and DER capacity factors from 1973 through 1984 averaged 75.2% and 74.6%, respectively.

3.1.2 Events of environmental importance

3.1.2.1 Radioactivity release events. Table 3.2 summarizes eleven events at Millstone 1 that resulted in radioactive releases or personnel overexposures. Six of the 11 reported events were radioactivity releases caused by mishandling in or failures of the waste disposal system. The remaining five events involved inadvertent exposures of maintenance workers. Human error was the cause of 10 of the 11 events. The eleventh event was a consequence of design features inherent in all BWRs. On December 13, 1977, two hydrogen explosions occurred in the off-gas system; the second explosion caused a small, uncontrolled release of radiation.

The first of the human error events occurred on March 25, 1974. Two workers were overexposed to Co-58 and Co-60 due to poor ventilation in a maintenance area. Badge readings taken during the fall 1974 refueling outage showed that three men had received doses in excess of their allowable limits. An overflow of a surge tank onto the boiler room floor resulted in the contamination of the shoes of a worker on

March 27, 1975. In September, 1975, one worker ingested small amounts (<1500 nanocuries) of Co-60 and Mn-54. In October 1975, five workers received radiation doses in excess of allowable limits while performing maintenance on the feedwater spargers.

Table 3.3 lists the annual radioactive releases from the plant. Through the fifteen years of operating experience that was covered by this review, there were 17 reports of radioactivity in excess of regulatory limits occurring in plants and animals near the Millstone site. Most of these involved high levels of activity in the oysters in Niantic Bay. One report, however, revealed a high iodine activity in cow's milk samples taken from the area. This was not related to the plant but rather to fallout from the Chinese atomic bomb tests.

Beginning in 1981, Millstone began reporting high levels of Ag-110 m, Co-58, and Co-60 in aquatic organisms. Ten reports between May 24, 1981 and May 16, 1984, detailed radioactivity levels that were higher than the technical specification reporting level. However, the activity was no higher than that found in previous years. The utility calculated the dose consequences and concluded that the doses were insignificant and that the reporting requirements are overly conservative.

3.1.2.2 Nonradiological events. The nonradiological environmental events included abnormal fish impingement on intake screens, pump fouling by aquatic organisms, and salt water induced equipment failures. Excessive fish impingement was reported on 18 occasions: 16 times in 1976, and 2 times in 1977. The unusually high numbers were attributed to an abnormal increase in fish population in 1976 and 1977. An emergency service water pump failed twice in 1982 as a result of marine life fouling the impellers.

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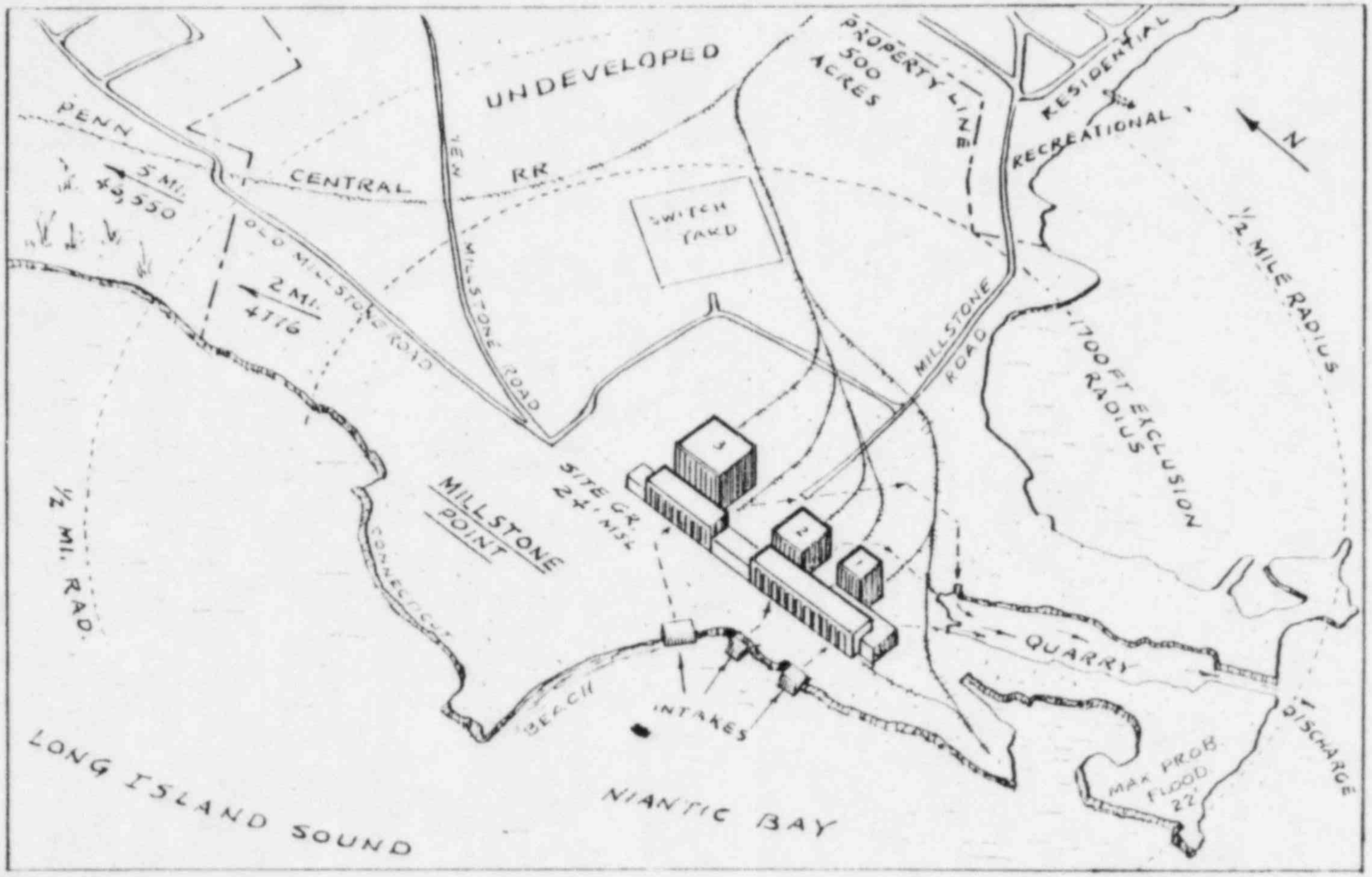

Figure 3.1 Millstone Nuclear Power Station Site

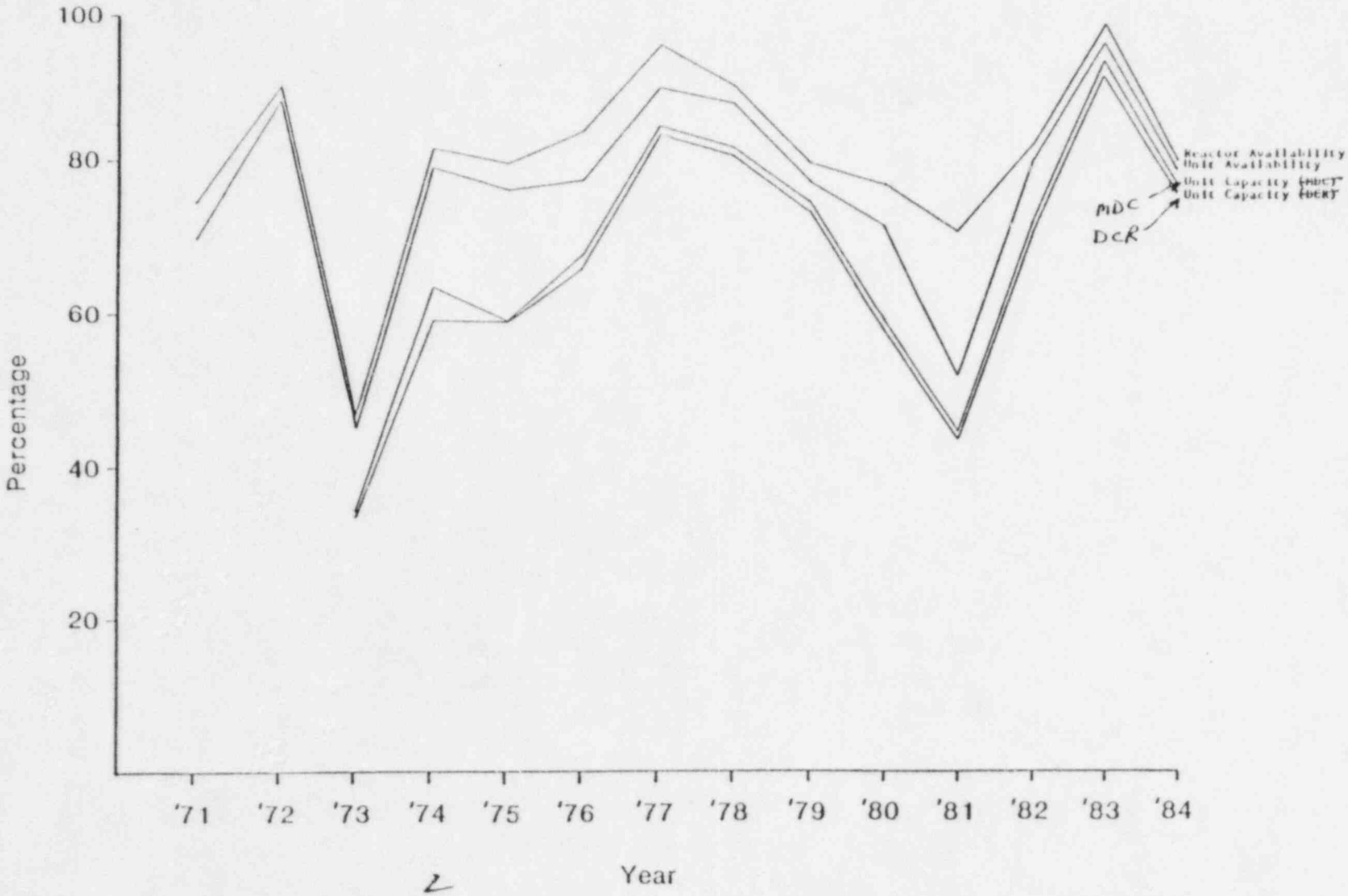

Fig 3.~~1~~ 2 Availability and capacity factors for Millstone 1

Salt water corrosion has caused recurring problems in electrical systems at Millstone. Corrosion in the air start system for the gas turbine caused the turbine to fail four times between July 14, 1981, and February 7, 1983. The carbon steel piping in the air start system was replaced by stainless steel piping in 1983. On December 9, 1983, excessive salt deposits built up on the bushings of the reserve station service transformer (RSST). The RSST was declared inoperable until the salt deposits were removed.

3.2 Forced Reactor Shutdowns and Power Reductions

3.2.1 Review of reactor shutdowns and power reductions

A total of 152 forced reactor shutdowns and 49 power reductions occurred at Millstone 1 during the operating period from plant startup (in 1970) through 1984. These events were categorized and analyzed in the review.

Table A.1 "Forced shutdowns and power reductions for Millstone 1," in Appendix A provides a comprehensive summary of the forced shutdown and power reduction events at Millstone 1. For those forced shutdowns and power reductions which were reportable events, more detailed descriptions are included in Sect. 3.3.

Tables 3.4 and 3.5 summarize the forced shutdowns and power reductions respectively. Causes (Item I.3 in Table 3.4 and Item I.2 in Table 3.5) are dominated at Millstone 1 by equipment failures. Only six shutdowns attributable to operator errors were reported, and no power reductions were attributed to operator error.

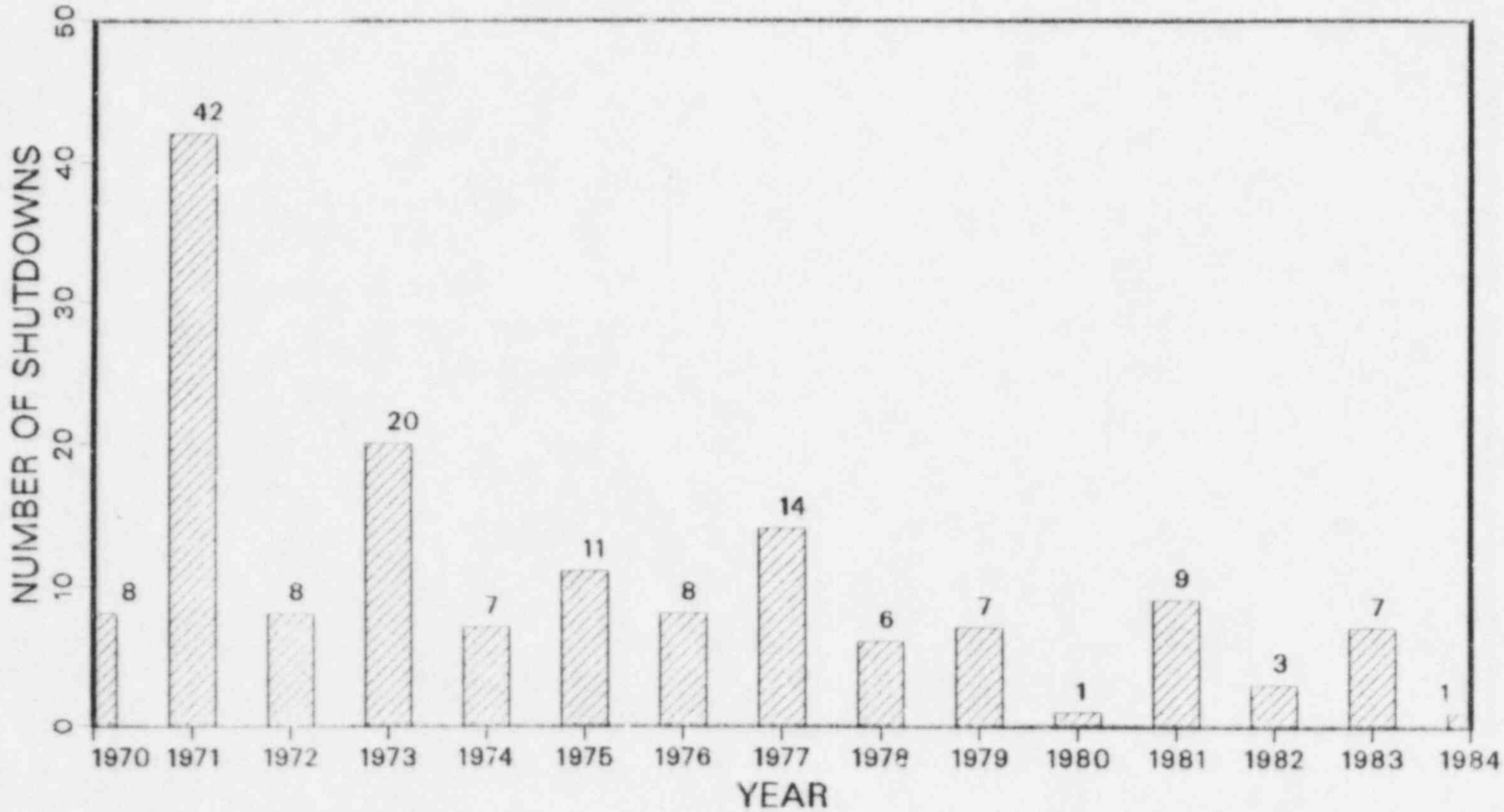

Figure 3.3 Number of shutdowns per year at Millstone 1

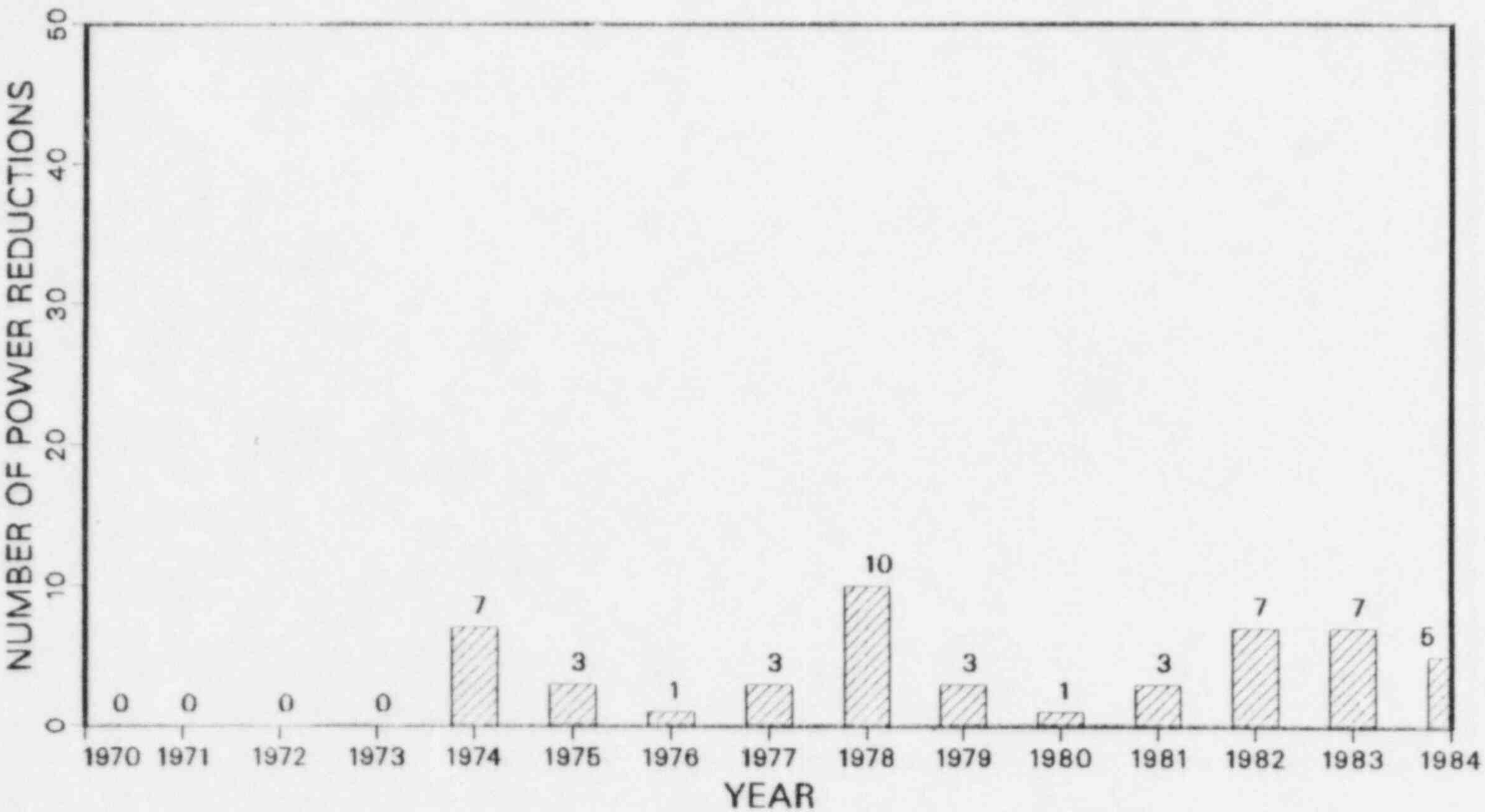

Figure 3.4 Number of power reductions per year at Millstone 1

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3.2. Yearly summaries of Forced Shutdowns and Power Reductions

Millstone 1. Figures 3.3 and 3.4 show the annual rate of forced shutdowns and power reductions for the years 1970 through 1984. A brief discussion of the events occurring in each of those years follows:

1970

The Millstone 1 BWR went critical on October 26, 1970. Eight forced outages occurred during thgis year seven of which were during startup testing. These seven were split about evenly between instrumentation malfunctions and faulty equipment installations. Maintenance and testing were the causes for all seven forced outages. The reactor coolant system was involved three times, and the steam and power systems were involved three times.

One shutdown was due to a cracked seal weld on the main condenser. Oscillation of a pressure control torque tube tore a bypass valve linkage away from its support, necessitating a shutdown.

A problem, recurring in later periods, first surfaced during this report period. The problem involved a moisture separator drain tank level indicator which indicated high, subsequently tripping the turbine and resulting in a reactor shutdown.

1971

During 1971 the reactor experienced 42 forced shutdowns, the most occurrences in a single year during the 15 years of operation covered by this review (Fig. 3.3). These shutdowns were attributed primarily to equipment failures (37 times). Three of the forced shutdowns were caused by maintenance and testing; 2 were for operator error.

The longest forced shutdown occurred on October 10, when the unit was down for 10 days to repair a turbine control valve. The next longest forced shutdown occurred on August 30, and lasted for 7 days. The traveling screens of the circulating water system became clogged with sea-weed causing the loss of the main condenser vacuum and a reactor scram.

At the beginning of 1971, only two months after the initial criticality, a high level indication in the moisture separator drain tank caused a steam turbine trip. This event recurred 11 more times during the year. These high level indications were attributed to broken baffle plate welds and level control instrumentation malfunctions.

Another recurring event during this year first occurred on March 23. A steam turbine control valve malfunctioned necessitating a 2 day shutdown for valve testing. Six additional forced shutdowns occurred due to the malfunctioning of this valve.

There were six shutdowns due to miscellaneous instrumentation malfunctions. During two shutdowns, Millstone 1 repaired leaks in the main steam line.

1972

The number of forced outages dropped to 8 in 1972 (Fig. 3.2), with no reported power reductions. The total forced outage time was only 585 h, the fourth lowest in Millstone 1's history.

Two forced shutdowns in February lasted for a total of 12 days. These were due to improper responses from the main steam line venturi differential pressure transmitters. The sensing tubes were replaced. On March 12, a reactor scram occurred during plant heatup when a reactor

feedwater pump was started when the water level in the reactor vessel reached the 10-in. level indication. Collapse of voids from the cold feedwater caused the level to decrease further and a low level scram resulted.

During the refueling outage which commenced September 1, the leaking tubes in the main condenser were plugged and all incore power range detectors were replaced.

1973

This year saw the second most forced shutdowns in Millstone 1's operating history: 20 forced shutdowns for 2892 h total (Table 3.4). Sixteen of these shutdowns were attributed to equipment failures. The longest forced shutdown occurred on April 18, and lasted for 89 days. At that time, all of the feedwater spargers were replaced because of cracks.

There was one forced shutdown of 10 days at the direction of the Atomic Energy Commission (AEC) to examine for possible inverted control rod internals. The General Electric Company had provided notice that some of the boron carbide poison pins were inverted during the blade fabrication process of some control rods. Some of these defective rods might have been installed in the Millstone core. In this inverted configuration, it was conceivable that axial downward shifting of the boron carbide powder could occur, and that this shifting could result in a change in the reactor core shutdown margin. A series of shutdown tests subsequently indicated nothing remiss.

Four of the shutdowns were due to instrumentation malfunctions, and four were due to lube oil pressure alarms on the reactor recirculating pump motor.

1974

This year saw a significant reduction in forced downtime in Millstone 1's operation: seven forced shutdowns for 91 h total (Table 3.4). The seven power reductions were attributed to equipment failures. Sea water leakage in the main condenser was detected necessitating two power reductions to plug tubes.

Recurrence of the problem of turbine trips, attributed to high level indications in the moisture separator drain tank, accounted for three more power reductions.

A refueling outage commenced on September 1, during which time the feedwater spargers were replaced due to excessive vibration at reactor power levels greater than 80%.

1975

This year Millstone 1 suffered the third worst level of forced downtime in its history. Of a total of 976 h forced out of service (11 forced shutdowns), 56% (547 hours) were caused by a transformer failure. On September 12, a combustible gas mixture was detected in the transformer, and a shutdown was made to change-out the transformer. Refueling was completed during this outage.

On March 3, a blown valve stuffing box on the low pressure cooling injection system occurred, resulting in water blowing from the identified leakage sump into the unidentified leakage sump.

1976

During this year there were eight forced shutdowns. On August 10, high winds deposited salt spray on the main transformer insulators causing arching between insulators. This caused the generator to trip.

During the resulting outage, the gas turbine speed control became inoperable necessitating the replacement of the electronic governor.

On July 16, a shutdown was necessary to repair the motor operator of the isolation condenser isolation valve. On December 17, this same valve malfunctioned resulting in a shutdown to clean it.

1977

In 1977, there were 14 forced shutdowns due to equipment failure for a total of 738 h. On June 14, a mechanical pressure regulator malfunctioned, tripping the steam turbine and causing a 5-day shutdown. In August, main condenser tube failures occurred again at Millstone 1. Several power reductions were made to plug the leaking tubes. On December 13, 1977, the first of two hydrogen explosions occurred in the off-gas system. Since the explosion was confined within a massive underground pipe, damage was minor and the reactor was able to continue operating while the damage was being repaired. However, since the second explosion was not confined, considerable damage in a two-level room at the base of the stack and to the plant stack itself occurred. The reactor was manually tripped.

1978

This year there were only six forced shutdowns for a total of 197 h. Starting in June, main condenser tubes began leaking and troubled Millstone 1 for the rest of the year. Ten power reductions took place in order to plug leaking tubes. Again, level control malfunctions occurred in the moisture separator drain tank.

1979

Seven forced shutdowns occurred this year for a total of 424 h. The most significant event occurred on January 6 and lasted for 11 days. Stress corrosion cracking in the clean-up return line necessitated replacement of this piping. Again, the plugging of leaking main condenser tubes caused three power reductions. On July 2, a shutdown, due to low water level from the feedwater regulator valve lockup, resulted from the loss of both plant air compressors.

1980

Only one forced shutdown was reported in 1980. A water hammer was experienced in the isolation condenser piping on December 19, 1979. The isolation condenser was placed out of service on January 5. Power was reduced and restricted to 40% for 27 d during the modifications made to the isolation condenser piping supports.

1981

At the beginning of 1981, Millstone 1 was continuing a refueling/ maintenance outage which had begun October 4, 1980. The outage continued 2598 h from January 1 until the middle of April. There were nine forced shutdowns totaling 1641 h of downtime and three forced power reductions.

On April 21, the turbine experienced high vibration and was manually tripped. The reactor scrammed due to high main condenser conductivity. Balancing problems caused the turbine outage to last 1372 h.

On July 12, a feedwater regulator valve failed to close. On August 8, circulation pump "A" tripped while recirculation pump "B" was offline. The unit was manually scrammed.

On August 10, a scram occurred during a surveillance test when an operator failed to reset a scrammed channel before testing the other channel. On September 14, a power surge to the ATWS system caused the scram air header to depressurize.

1982

In 1982, there were three forced shutdowns, which resulted in a total of 100.9 h of outage. On February 12 the "A" train feedwater regulator valve cycled open and then closed and this generated an automatic scram. On April 13 a reactor recirculation pump trip was automatically initiated by Division I of the ATWS system. The trip resulted when a 125 volt DC circuit breaker, supplying the Division I panel, was switched off to enable the varying of the battery charger output voltage, without affecting the associated ATWS channel, in an attempt to locate a ground in the 125 volt DC power system. On July 31, the generator out-of-step relay, located in the switchyard, malfunctioned and tripped open the switchyard breakers. This caused a full-load rejection followed by an ATWS Division I scram.

Seven power reductions were required for maintenance activities, five of which involved leaking main condenser tubes.

1983

There were seven forced shutdowns in 1983 accounting for 383.2 h of outage. In addition, seven power reductions were required for maintenance.

On January 24, an automatic scram occurred when a steam tunnel high-temperature alarm, which was caused by moisture intrusion into the switch, resulted in a ground short in the switch. On March 24, while

effecting an orderly shutdown to identify and repair drywell leakage, an addition of feedwater at too rapid a rate caused a level transient and the reactor was scrammed. In June the gas turbine became inoperable and the reactor was shut down. In August when the instrument air compressor tripped, a scram occurred on low air pressure. The reactor was shut down in November to repair the "A" recirculation pump seal because of leakage.

In March, a high moisture separator level and in June, a high reactor water level both caused reactor trips.

1984

During 1984 there was only one forced outage. On August 8, the plant was shutdown to allow entry into the drywell to find and repair a leak in the recirculation system riser instrument valve. There were five power reductions, two of which were required to find and fix leaking main condenser tubes. On December 5, power was reduced to repair a sticking "B" train feedwater regulator valve.

3.2.3 Systems involved in forced shutdowns and power reductions

Thirty-one different systems were involved in forced shutdown and power reduction events; the turbine generator and control system and the main steam system control system together accounted for 45% of these events. Excluding these two systems, the average number of forced shutdowns per system was three during the operating period of the plant through 1984.

There were 36 forced shutdowns involving the turbine-generator and its associated controls with 20 of these occurring during 1971. In 1971, 14 events involved the high moisture separator drain tank level

indicators, 4 involved turbine control valves, and 2 were caused by the loss of an offsite power line due to lightning. In 1979, a loss of main generator excitation was experienced.

There were 27 forced shutdowns involving the main steam system and its controls. Almost one-third of these occurred in 1971. On 11/11/70, 11/21/70, 11/4/74, and 7/12/77 inadvertent closure of MSIVs occurred. On 3/2/71, 10/21/71, 9/20/73, 5/20/77, 11/29/77, and 2/26/79, pressure relief or safety valves either failed to close or opened prematurely.

There were seven forced shutdowns involving condensate and feedwater systems. On 5/25/71 and 12/16/74, feedwater control valve closures prevented the flow of feedwater. On 3/12/72, a void collapse occurred from an increase in cold feedwater flow. On 3/6/73, a condensate booster pump was not started in time, resulting in a low water level. On 4/18/73, the feedwater sparger was replaced requiring a shutdown lasting 90 days. On 8/10/73, high reactor water level was experienced due to the starting of a feedwater pump.

Of the 49 power reductions, 65% involved the main condenser system. Three of the power reductions were classified as initiators.

3.2.4 Causes of forced reactor shutdowns and forced power reductions

Of the 152 forced shutdowns, 80% were caused by equipment failures for a total of 8533 h. Maintenance and testing accounted for 10% of the shutdowns, for a total 844 h. There were only seven events due to operational errors for a total of 140 h.

Of the 49 power reductions, 47% were caused by equipment failures and 53% were caused by maintenance and testing errors.

3.2.5 Non-DBE shutdowns

Table 3.6 summarizes the NOAC categories assigned to non-DBE shutdowns. Only the major NOAC categories are listed in this table. Equipment failures accounted for 67% of the events with no apparent decline during the first nine years of operation. Instrumentation and control problems accounted for 20% of the events, and these recurred throughout the period addressed by the operational review.

3.2.6 DBE initiating events

Of the 201 forced shutdowns and power reductions accumulated at Millstone 1, 56 were classified as DBE initiating events as shown in Table 3.7. However, none of these events actually initiated a sequence leading to any significant economic loss or safety hazard to the plant or to the environs. The trend in the total number of DBEs per year showed no correlation with any other trends, such as plant performance as measured by total number of shutdowns per year, or total downtime per item (Table 3.4). Each specific type of DBE is discussed below.

3.2.6.1 DBE Sect. 1 events — increases in heat removal. Six events (12%) were categorized into this section. Four of the six were due to instrumentation malfunctions. One was attributed to operator error. Those six events were subcategorized as described below:

D1.1 — Feedwater System Malfunctions Resulting in a Decrease in Feedwater Flow. On May 25, 1971, a low water level reactor scram occurred when the valve positioner on one of the feedwater regulation valves failed. This caused the partial closure of a feedwater control valve resulting in insufficient flow and a reactor scram on a low reactor water level.

D1.2 — Feedwater System Malfunctions that Result in an Increase in Feedwater Flow. On March 12, 1972, reactor scram occurred during plant heatup when a reactor feedwater pump was started at the 10-in. level indication in the reactor vessel. Void collapse from injection of cold feedwater into the reactor vessel resulted in a level decrease and low level scram.

On August 10, 1973, a malfunction of a reactor water level transmitter resulted in a false low level signal to the feedwater control valve circuit. This resulted in increased flow following by reactor high water level resulting in a scram. Later the same day, a scram occurred due to a high water level which was the result of an operator starting a feedwater pump with the feedwater regulating valves in the full open position.

On June 7, 1983, the reactor scrammed on a high reactor water level trip.

D1.3 — Steam Pressure Regulator Malfunction or Failure that Results in an Increasing Steam Flow. There were three events of this kind, all attributed to instrumentation malfunctions. On November 19, 1970, installation of scrubbers near the main steam line high flow ΔP switches was in progress. While wiring in an isolation switch, a momentary high flow signal was initiated, and the main steam lines isolated generating the reactor scram.

On December 5, 1970, while shifting to Electric Pressure Regulator (EPR) pressure control, EPR oscillated giving bypass valve swings and pressure and reactor vessel swings. The result was a low level signal yielding a reactor scram. On September 21, 1973, a fault in the EPR

controls opened the turbine bypass valves, causing reactor pressure to drop and the plant to scram.

3.2.6.2 DBE Sect. 2 Events — Decrease in Heat Removal. Forty-five events were categorized into this section. Thirty-four of them (75%) were caused by turbine trips. Three of these were associated with steam pressure regulator malfunctions, two with losses of external electric loads, two with inadvertent closures of main steam isolation valves, three with failures in the feedwater system, and one with loss of condenser vacuum. The events in this section were subcategorized as discussed below:

D2.1 — Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow. On November 21, 1970, vibration of the reactor mode switch resulted in a main steam isolation. On January 19, 1971, the turbine control valve closed, causing the reactor to shut down. On June 25, 1980, an electric pressure regulator malfunction induced an APRM scram.

D2.2 — Loss of External Load. On June 24, 1971, and again on June 25, 1971, there were turbine full load rejects due to lightning strikes which caused loss of the 345 kv line.

D2.3 — Turbine Trip (Stop Valve Closure). Twenty-two of the 34 events were attributable to malfunctions of the moisture separator drain tank level control. This type of event first occurred on December 30, 1970, and continued to recur irregularly 14 times in 1971, 3 times in 1974, 3 times in 1978, and one time in 1983.

On five occasions turbine control valve malfunctions occurred: 5/27/71, 9/29/71, 10/3/71, 10/10/71, 2/4/72. Each time this caused a turbine trip.

The remaining five DBE's were isolated failures.

D.2.4 — Inadvertent Closure of Main Steam Isolation Valves. The two events in this category, occurring on 11/4/74 and 4/7/77, were attributed to mechanical failures in main steam valve actuators.

D.2.5 — Loss of Condenser Vacuum. While this classification addresses the complete loss of condenser vacuum, an event which occurred on August 30, 1971 involving a low condenser vacuum was included in this subcategory.

D2.7 — Loss of Normal Feedwater Flow. Of the three events in this category, one, occurring on March 6, 1973, was attributed to operator error for failing to start a condensate booster pump in time. The other event, on December 16, 1974, occurred as a result of a broken stem on the feedwater control valve.

3.2.6.3 DBE Sect. 3 Events — Decrease in Reactor Recirculation Flow Rate. The sole shutdown which resulted from a decrease in reactor recirculation flow occurred in 1981. On August 8, 1981, the "A" reactor recirculation pump tripped on generator overload. The "B" reactor recirculation pump was off-line at the time and the unit was manually scrammed.

3.2.6.4 DBE Sect. 6 — Decrease in Reactor Coolant Inventory. Three shutdowns were due to a decrease in reactor coolant inventory. On March 2, 1971, a main steam safety valve started leaking and blowing steam. On November 29, 1977, the automatic pressure relief valve lifted prematurely. On February 26, 1979, a safety relief valve lifted prematurely and failed to reseat.

3.2.7 Trends and Safety Implications of Shutdowns and Power Reductions

One recurring problem associated with forced outage and power reductions was inadvertent or premature opening of pressure relief or safety valves or their failure to close or seat. On 10/22/71, a relief valve failed to close completely. On 9/20/73, a relief valve was leaking. On 5/20/75, a relief valve failed to close completely. On 11/29/77, a relief valve opened prematurely. On 2/26/79, a relief valve lifted prematurely and failed to seat. On 3/21/71, a relief valve was found blowing steam.

3.3 Reportable Events

This study reviewed 404 reportable events from Millstone 1. Information that was reviewed included telegrams, letters, abnormal occurrences (AOs), reportable occurrences (ROs), and licensee event reports (LERs) that were filed by the utility when a technical specification was violated. The information contained in the reportable events was coded as discussed in Sect. 2.2.4. Data tables, arranged by year, are presented in Appendix B.

3.3.1 Review of Reportable Events from 1970 through 1984

Figure 3.5 illustrates the number of reportable events filed per eyear at Millstone 1. There is no discernible trend in the number of reports submitted on a yearly basis. The following sections present a summary for each year of operating experience at Millstone 1.

1970

The plant went critical on October 26, 1970. Ten reportable events were experienced during the year. Although none of the events were deemed significant to safety, two events are conditionally significant.

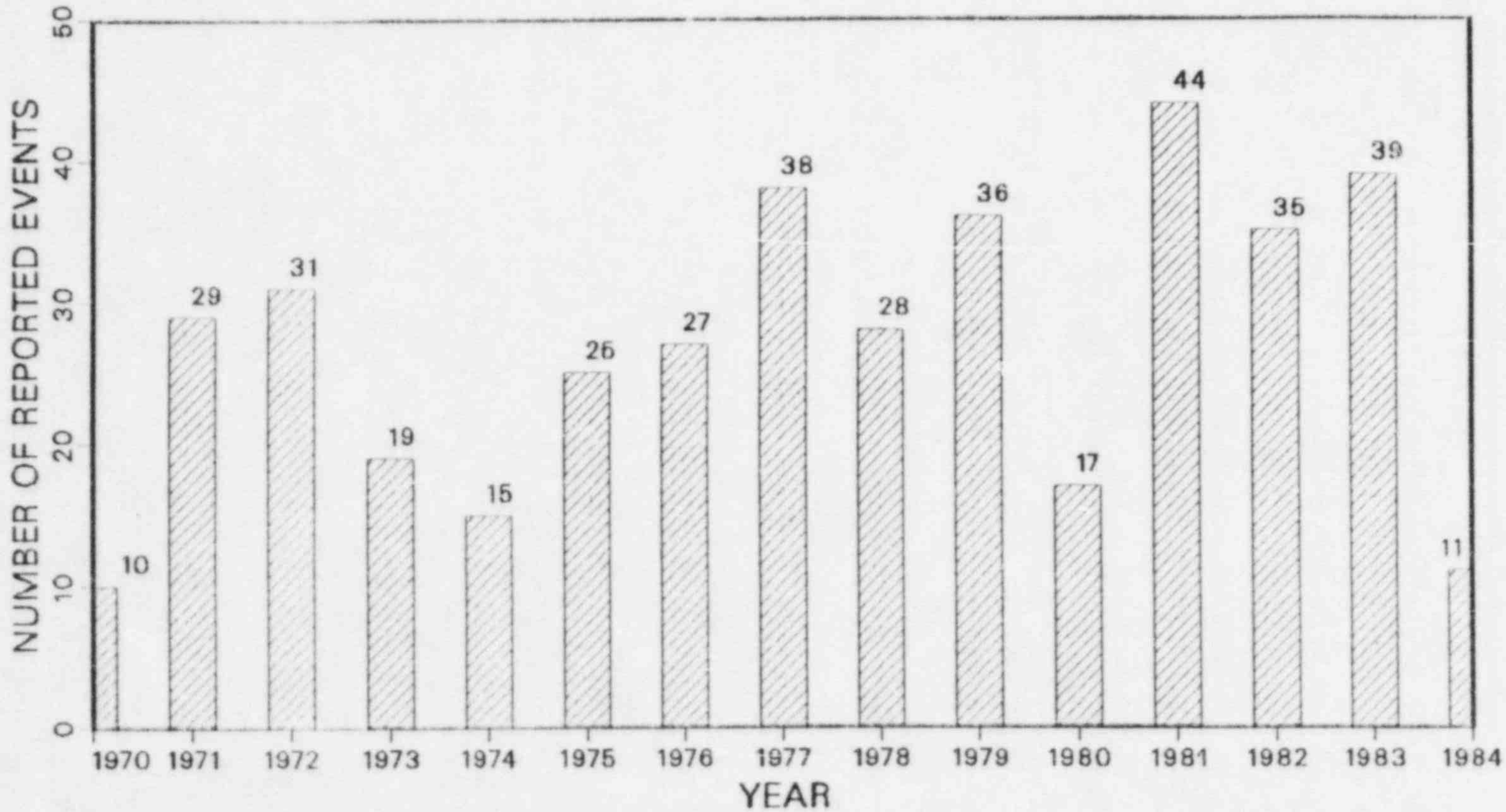

Figure 3.5 Number of reportable events per year at Millstone 1

Eight of the reportable events that occurred during the first two months of operation in 1970 resulted from human error.

Human errors consist of design errors (4 events), installation errors (3 events), and operator and maintenance errors (1 event). Three of the design errors consisted of inadequate pipe support of the main steam lines (RO-70-6), inadequate lubrication oil pump speed (RO-70-4), and an inappropriate wiring change that resulted in a loss of full rod control (RO-70-7). The fourth error caused the isolation condenser to isolate (RO-70-5) when a high-pressure signal initiated the operation of the isolation condenser. The resulting high condensate flow caused the isolation condenser to isolate. A restart of the system was unsuccessful and the pressure transient was brought under control through the use of relief valves.

Three reportable events resulted from installation errors. Installation errors included faulty welds in the main condenser (AO-70-8), failure of an MSIV due to missing parts (RO-70-6), and a vent line on the lube oil pump discharge line not being reinstalled (AO-70-10). Both events that are identified as RO-70-6 are conditionally significant to safety.

<u>1971</u>

Millstone 1 began commercial power generation in March 1971 and experienced 29 reportable events during the year. Two of these events are significant to safety and three are conditionally significant to safety.

The first significant event occurred on January 19, 1971, when the entire low pressure ECCS was rendered inoperable during a test (AO-71-1). The core spray injection valve failed to close and two LPCI valves

failed to open. A turbine control valve unexpectedly closed causing the other three control valves to open further. Severe pressure oscillations caused reactor power oscillations of 8%. As soon as pressure control was changed from the mechanical pressure regulator to the electrical pressure regulator, the oscillations stopped.

The second significant event occurred on October 10, and involved a failure of the steam turbine bypass valve. This event resulted in a reactor blowdown. Replacing the defective valve eliminated the problem.

Two of the three conditionally significant events involved the gas-turbine generator (AO-71-12, AO-71-25). The remaining event occurred when a service water heat exchanger leak caused flooding of the DC motor control center (AO-71-09).

1972

The number of reported events increased to 31 in 1972. Human errors again contributed to a large number of the events that were reported. Maintenance errors (7 events), design errors (3 events), and installation errors (2 events) all contributed. The systems involved in the most events were the Main Steam System (6 events) and the Reactor Core Isolation Cooling system (5 events). No significant events occurred during the year, however four conditionally significant events occurred.

Two of the conditionally significant events involved failures of multiple components (AO-72-17 and AO-72-18). In each case, the components involved were found to be out-of-calibration and were subsequently recalibrated.

The third conditionally significant event involved salt water intrusion in the main condenser (AO-72-22). Loss of the main condenser is not critical for plant safety, but the presence of chlorine in the system manifested itself in damaging other components in the reactor coolant system. Some of the resulting failures were not discovered until 1976 when 116 of the 120 local power range monitors (in-core), failed due to stress corrosion brought about by the excessive chloride content in the reactor water. Problems also arose from cracking of feedwater spargers in 1972 (AO-72-26).

The fourth conditionally significant event occurred on August 25. The plant experienced a loss of power to the shutdown transformers when a plane crashed into the 27.6 kv power line. The line was only 500 ft from the switchyard. The plant was operating at 82% power and did not reduce power after the crash.

1973

Nineteen reportable events occurred in 1975. Most of the events involved the feedwater control system (4 events) or the reactivity control system (4 events). One event is significant to safety and one event is conditionally significant.

Several control rod system problems surfaced in 1973. The event that is significant to safety involved degradation of the control rod drive accumulators due to flaking of plating into the hydraulic system. Six of the 143 accumulators had sufficient flaking to impair the operation of the control rod drive mechanism. These six were replaced and the hydraulic system was flushed. A program for monitoring accumulator conditions was initiated to prevent recurrence of the incident.

On April 5, pressure adjustments were being made as reactor power increased (AO-73-4). The pressure momentarily reached 1042 psig initiating the isolation condenser. The position switches were incorrectly set causing the inboard condensate return valve to open too wide. The resulting high flow condition automatically isolated the condenser. Due to the frequency of the isolation condenser valve failures, this event was classified as conditionally significant.

1974

The third smallest number of reportable events occurred in 1974 (15 events). None of the reportable events is considered to be significant or conditionally significant to safety. Most of the reportable events involved the reactor coolant system (8 events).

Over one-half of the events involved a human error (9 events). Design errors and maintenance errors each accounted for four events. Two of the maintenance errors caused Main Steam Isolation Valve failures. The failures appear to be due to the poor quality of air provided to the pilot valves since the air slide valves stuck on both occasions. This failure mode is important in that it can cause more than one MSIV to fail at the same time. This would lead to plant conditions not considered in the plant's safety analysis.

1975

Twenty-five reportable events occurred in 1975. Although no significant events occurred, two events are conditionally significant. The human errors occurring during the year were due to maintenance errors (5 events), design errors (3 events), administrative errors (1 event), and installation errors (1 event).

The first conditionally significant event, which occurred in January, involved a hydrogen explosion in the condensate demineralizer regeneration system acid day tank (Ltr. 1/24/75). No one was injured. The hydrogen was formed by moisture leaking into a tank of concentrated sulfuric acid which was ignited by a spark caused by welders working in the area. The most noteworthy (and conditionally significant) human error was an administrative error, that occurred on March 30, when radioactive liquids were discharged twice from an unmonitored sump. Previous samples from the sump showed no activity and none was anticipated. Procedures were modified to prevent a recurrence.

More valve problems occurred in 1975. In particular, two events involved degradation of safety/relief valves. In one incident (AO-75-9) a safety/relief valve failed to reseat due to a malfunction of two pilot valves. The reactor pressure dropped to 160 psig. During the blowdown, the reactor cooldown rate reached 155°F/h, well in excess of the limit of 100°F/h. No damage to reactor components occurred. Both pilot valves were replaced. In a second incident (AO-75-17) the bellows integrity of two safety relief valves could not be verified. Instrument air lines had become fouled and had to be cleaned.

1976

Five of the 27 reportable events that occurred at Millstone 1 in 1976 are significant. Three of the significant events involve the gas-turbine generator (RO-76-10, RO-76-12, and RO-76-29). One failure of the generator occurred during a total loss of offsite power. Millstone 1 experienced several problems with the isolation condenser, beginning in February (RO-76-4) and continuing throughout the year with various valve and control troubles.

On November 17, 1976, an inadvertent criticality during a shutdown margin test occurred when an operator selected the wrong control rods for the test (RO-76-34).

It should also be noted that the majority (16 events) of the environmental violations at Millstone 1 occurred in 1976. These events are attributed to an unexpected increase in the abundance of marine life around the plant during the year.

A single conditionally significant event occurred on April 23, 1976 when an excessive stack gas release occurred due to fuel clad perforations.

1977

The third largest number of reportable events were recorded in 1977 (38). Two of these failures involved safety/relief valves at Millstone 1. Four failures involved the emergency electrical power systems. Three of these failures involved the diesel generator. On two occasions, the diesel generator failed as a result of a fuel oil leak. The third event resulted in the diesel generator being declared inoperable. The cause was unknown. There was only one gas turbine generator failure in 1977. A spurious noise signal in the control system caused the gas turbine generator's startup test to be stopped.

Both of the significant events that occurred in 1977 took place in December. The first significant event resulted in a total loss of emergency power when the diesel generator was declared inoperable at a time when the gas-turbine generator was unavailable. The second significant event to occur in 1977 involved the off-gas system. The stack release rate increased after an explosion in the system. Four hours later,

another explosion occurred. The second detonation occurred at the base of the stack. Two workers were injured but the safety of the plant was not challenged.

The three conditionally significant events that occurred in 1977 involved various systems. In the first event, both stack gas monitors were inoperable due to a blown fuse in the power supply. On November 18 the reactor cooldown rate exceeded the allowable limit when a safety/relief valve lifted at a low pressure. The final event occurred when a maintenance foreman inadvertently short circuited a valve operator on a LPCI valve.

1978

Although there was a large number of reportable events in 1978 (28 events), none of them are significant. The majority of the events involved setpoint drifts (10 events) or single failures in redundant systems (9 events).

Of the 12 human errors that occurred in 1978, eight were due to maintenance. The most notable error, and the only conditionally significant event, is an administrative error. On July 25, the containment was purged with a high radiation signal to the containment purge valves bypassed. The high radiation override also bypasses the containment isolation actuation signal to the purge valves. On 12 occasions, the purge interval was greater than four hours. The operability requirements for these valves was not discussed in the procedures.

1979

In 1979, the fourth highest number of reportable events (36 events) over the operating experience of the plant were reported. On September

14, 1979, it was discovered that a design error allowed loss of power to the ECCS to go undetected under a certain electrical distribution arrangement (LER 79-26). The logic was changed to eliminate this possibility. This event is classified as significant to safety.

Three conditionally significant events occurred during 1979. On February 26, a pressure relief valve lifted prematurely and then failed to reseat. An uncontrolled blowdown and excessive cooldown rate resulted. The cooldown rate limit of 100°/h was exceeded by 5°. The overall effect on reactor pressure vessel structural integrity was considered to be small.

Two other conditionally significant events occurred due to human errors. On February 1, an operator operated the plant in a degraded mode of operation. One control rod had an inoperable accumulator and another control rod in the same array was electrically disarmed. The disarming of the second control rod was an oversight by the operator.

Defective procedures led to the sodium pentaborate concentration in the standby liquid control tanks being less than the allowable limit on July 13, 1979. The concentration was increased immediately upon discovery and procedural changes were made to preclude a recurrence.

1980

Seventeen reportable events were recorded at Millstone 1 in 1980. Problems involving weld cracks in two main steam lines and in the condenser nozzle were reported. The cracks were discovered during the fall refueling outage. No significant or conditionally significant events occurred during 1980.

1981

In 1981, 44 events were reported, which constitutes the largest number of reportable events submitted by Millstone during a single year of operation. Instrumentation and control systems were the most reported systems (14 events). Half of these events were due to set point drifts. The emergency generator system accounted for six events. All six of the events involved a diesel generator, a gas turbine generator, or both. The event that involved both generators was considered significant (LER 81-02). On April 3, personnel discovered that the single relay failure in the loss of normal power circuits conditionally could prevent the diesel generator and the gas turbine generator from energizing the emergency buses. The cause was a design oversight.

The second significant event occurred on September 15 (LER 81-25). Voltage fluctuations in the 125 V DC electrical system resulted in an isolation of the ATWS system. This caused both recirculation pumps to trip. This condition was not annunciated in the control room and consequently, control room personnel were not alerted to the incident.

Eight conditionally significant events occurred in 1981. Three events involved radioactive liquid waste releases. Gas-turbine generator failures accounted for two events. The remaining three events involved a cracked pipe weld, two closed containment isolation valves, and a high cooldown rate during a manual blow down.

1982

Although there was a large number of reportable events in 1982 (35 events), none of them are significant to safety. Nineteen events resulted from inherent error. Nine of these events involve set point

drift. Thirteen events resulted from human errors. Human errors include operation errors (4 events), installation errors (3 events), maintenance errors (3 events), and fabrication errors (3 events).

Of the 35 reportable events that occurred in 1982, thirteen are conditionally significant to safety. The gas-turbine generator was involved in four events. Operator failures involving the stack gas monitoring system accounted for two events. Two events involved radiation levels that were above Technical Specifications levels in marine life. The remaining events involved a stress crack in a sparger weld, set point drift in 5 safety relief valves, melted relays in the reactor protection system, a fire in a switcher, and multiple breaker failures due to water damage.

1983

The second largest number of reportable events occurred in 1983 (39 events). Instrumentation and control systems were the most reported systems (13 events). Over half (8 events) of these events were due to set point drifts. The emergency generator system accounted for six events. Five of these events involved the gas-turbine generator. No significant events occurred during 1983.

A total of seven conditionally significant events occurred during 1983. The emergency generator system was involved in four of the events. Improper calibration of time delay relays for the turbine-generator accounted for two events. The remaining event involved a switch failure in one channel of the average power range monitor.

1984

In 1984 the smallest number of reportable events (11 events) for any full year of operation occurred. Half of the reportable events were caused by maintenance problems. For the third consecutive year no significant events occurred.

The isolation condenser was involved in two of the five conditionally significant events that occurred in 1984. Another conditionally significant event involved three safety relief valves that failed to open during a bench test due to set point drift. This is a recurring problem for BWRs. The remaining events include the failure of a fire detector string due to low supervisory air pressure and welds that have intergranular stress corrosion cracking.

3.3.1.1 Systems involved in reportable events. A compilation of all reportable events by system and year is presented in Table 3.8. Some systems which were not involved in the reports were omitted. There are no discernible time-dependent trends among the systems identified. Most of the reports involved the following systems: reactor coolant (33.4%), instrumentation and controls (21.3%), engineered safety features (18.1%), and electrical power (12.1%). Each of these systems is discussed in the following subsections.

3.3.1.1.1 Reactor Coolant System. The designation of reactor coolant system encompasses a broad range of heat transfer-related equipment in the reactor. For Millstone 1, this system includes all steam line monitors and valves, especially safety/relief valves; the isolation condenser; main steam isolation valves; pressure regulator; feedwater system and controls; and recirculation system. One-third

(33.4%) of the reportable events involved the reactor coolant system (135 events). A large majority of reports involving the reactor coolant system concerned valve failures during tests, failures of the electrical and mechanical pressure regulators, and failures of various coolant parameter monitoring components. Excessive pipe movement was reported several times but no apparent damage resulted.

Eleven weld-related failures were reported for Millstone 1. Extensive work was performed on the feedwater spargers, and all spargers have been replaced at least once. Cracks in BWRs are a common problem, and Millstone 1 has been no exception. Cracking at Millstone 1 is analyzed in greater detail in Sect. 3.3.3.1.

The isolation condenser provided several problems at Millstone 1. The isolation condenser isolation valves failed during testing fourteen times. These failures are discussed in more detail in Sect. 3.3.3.5. The condenser itself had to be completely retubed in 1976 (RO 76-04). This incident is discussed in Sect. 3.3.2.1.

The electrical or mechanical pressure regulator failed four times from 1970 to 1972. Three of the failures involved only the electrical pressure regulator (AOs 71-12, 71-27, 72-2) and were inconsequential because the mechanical pressure regulator served as a backup. A fourth failure (AO-71-20) resulted in a reactor blowdown.

Along with resulting in a reactor blowdown, the above event was one of five that produced an excessive cooldown rate (AO 71-20, AO 75-09, LER 77-33, LER 79-05, LER 81-04). The October 10, 1971, blowdown cooled down the reactor vessel 135° in 18 min (450°F/h). The other excessive cooldown rates ranged from 105°F/h to 210°F/h. A rapid cooldown rate is

of continuing concern due to the added stress placed on the reactor vessel.

On March 10, 1978, two uncontrolled blowdowns occurred. A review of the temperature charts for both blowdowns revealed that the average rate of reactor coolant temperature change, over 1 hour, was less than the technical specifications limit (100°F/h). The first blowdown resulted when a safety/relief valve failed to close. The reactor was manually scrammed. Reactor pressure was allowed to increase due to other activities in the plant. The same safety/relief valve lifted prematurely resulting in another blowdown.

3.3.1.1.2 Instrumentation and control. This system is comprised of all reactor safety and trip instrumentation as well as all control functions for normal operation. Its frequent involvement (21.3% or 86 events) in reportable events can be attributed to instrument set-point drift, mis-calibration, and spurious trips. No safety functions were compromised as a result of these problems.

On February 16, 1982, the closure of a main steam isolation valve failed to generate a reactor protection system scram signal (LER 82-15). When the MSIV was closed, the relay did not deenergize. This prevented the reactor protection relay from deenergizing which in turn prevented an automatic scram. The armature on the limit switch was out of adjustment. Two events of this nature occurred previously on June 18, 1981 (LER 81-16) and on August 18, 1981 (LER 81-22).

3.3.1.1.3 Engineered safety features. The engineered safety features system was involved in 71 of the reportable events (18.1%). Most failures occurred during testing of the ECCS and the isolation

condenser. An unusually high number of reports appeared in 1971 when problems surfaced during tests of the core spray valves. On January 19, 1971, the core spray valves were tested at 1000 psig instead of the core spray operating pressure of 300 psig (AO 71-01). The valves all failed and the ECCS was declared inoperable. The torque sitting on the switches was too low and had to be reset. On March 31, 1971, the motor operator on an LPCI valve failed due to a short in its windings. The motor was replaced. On September 18, 1971, another LPCI valve motor operator burned out and was replaced by a larger motor.

Overall, 30 of the reportable events in this system resulted from human errors. Five of the events that resulted from human errors are of interest. On April 19, 1981, maintenance personnel left two containment isolation valves in the closed position (LER 81-03). The closure of these valves isolated the high drywell pressure switch associated with the ECCS and RPS initiation. While investigating problems with a LPCI motor-operated valve, a maintenance foreman opened the wrong breaker (RO 77-38). The open breaker caused the LPCI injection valve to become inoperable. Defective procedures allowed the boron concentration in the standby liquid control tanks to be less than the technical specifications limit (LER 79-18). On September 14, 1979, personnel discovered a design error in the power distribution to the ECCS buses (LER 79-26). A loss of power could occur to the supply for the ECCS electrical buses without the loss of normal power initiation logic being able to sense the loss. The fifth event was the previously mentioned event on January 19, 1971 (AO 71-01). Four core spray valves failed when the system was tested at 1000 psig rather than its operating pressure of 300 psig.

3.3.1.1.4 Electrical power system. Thirty-seven of the 49 reportable events concerning the electrical power systems were attributed to failures of the gas turbine generator. These failures in the electrical power system comprised 12.1% of all reported events for Millstone 1.

The emergency power system consists of a gas turbine generator and a diesel generator. Upon loss of offsite power, the gas turbine generator is required to supply power to the Feedwater Coolant Injection (FWCI) System. Therefore, when the gas turbine generator fails, the FWCI system is unavailable. The FWCI system or the isolation condenser are provisions for emergency core cooling during a loss of normal auxiliary power. On two occasions (RO 76-10, RO 76-12) the gas turbine generator was declared inoperable while the isolation condenser was out of service. The plant was immediately shut down on both occasions. Further details are given in Sect. 3.3.2.

On December 10, 1977, the diesel generator was declared inoperable while the gas turbine generator was out of service (RO 77-39). Consequently, all emergency power systems were unavailable. Further investigation of the emergency power system on April 3, 1981, revealed that a single failure mode existed (LER 81-02). A single relay failure in the loss of normal power circuits would inhibit the diesel generator and the gas turbine generator from loading the emergency buses.

Millstone also experienced a loss of offsite power when salt built up on the 345 kV lines and insulators as hurricane Belle passed (RO 76-29). The diesel generator and gas turbine generator were being run without loads as a precaution. When normal power was lost, the gas turbine tripped and the diesel generator loaded the emergency buses. The

gas turbine tripped twice. The first trip was due to a loss of the AC auxiliaries due to the loss of offsite power. The second trip was attributable to a loss of DC control power caused by the gas turbine running on DC auxiliaries which it is not designed to do.

3.3.1.2 Causes of reportable events. Table 3.9 presents a summary of causes of reportable events at Millstone 1. Over half of all reportable events (231) at the plant were attributed to inherent failure. Inherent failure includes set point drifts, wear out, and many of the failures for which no cause could be identified.

Over the operating experience reviewed, human error was responsible for almost half of all reportable events (173 events). Administrative, design, fabrication, installation, maintenance, and operator errors are all considered human errors. Human errors played an important role in events categorized as significant at Millstone 1. These human errors involved design, administrative, and maintenance errors. An examination of causes of significant events revealed that 11 of the 13 significant events were attributed to human errors. The last 2 events were inherent failures.

3.3.2 Review of significant events

The analysis of the operating history of Millstone 1 examined reported events to find those occurrences which represented significant threats to continued safe operation or to systems designed to mitigate transient conditions. Reportable events were therefore significant if they met one of these criteria:

1. an event in which the failure or failures initiated a design basis event (DBE) as listed in Appendix A, or

2. an event in which the failure or failures compromised a function of the engineered safety features

Thirteen events at Millstone 1 met the above significance criteria. Table 3.10 summarizes the significant categories assigned to these events decription of the significance categories is found in Appendix B. Table 3.11 summarizes the significant events which occurred at Millstone 1. The total in the table, 18, is greater than the actual number of significant events, 13, because 5 events required multiple significance categories. The events designated as significant were:

- o loss of the isolation condenser (1),
- o loss of provisions for emergency core cooling during a loss of normal power (2),
- o ECCS failures (2),
- o loss of offsite power with partial loss of emergency power (1),
- o complete and potential loss of emergency power (2),
- o inadvertent criticality (1),
- o all control rod drive accumulators require replacement (1),
- o recirculation pumps trip with no alarm given (1),
- o loss of pressure control followed by a blowdown (1), and
- o hydrogen explosion in the off-gas system (1).

3.3.2.1 Loss of the isolation condenser On February 12, 1976, the plant shut down due to arcing of the main transformer during a storm. Normal post-shutdown pressure transients caused the main steam isolation valves to close. As a result of this, pressure increased momentarily in the isolation condenser, causing a tube in the condenser to fail (RO 76-4).

Steam leaked into the shell side of the condenser and was vented into the atmosphere. The operators did not recognize the cause of the steam release until 1 h and 16 min after the shutdown, when the control room received a high radiation alarm from the steam vent line. The isolation condenser was then isolated and the steam releases halted.

An area of approximately one acre, all inside the fenced area, was contaminated. No reportable personnel exposures resulted from the release of contaminants. Eight minutes after the reactor trip, the main condenser was valved in to act as a primary heat sink. Recovery from the transient proceeded smoothly.

Examination of the condenser revealed a tube with a 1-in. by 2-in. hole. The failure resulted from stress corrosion cracking. The corrosion was attributed to the intrusion of chlorine from a previous failure of the main condenser. Other tubes in the isolation condenser were retubed with Inconel 600 tube material.

The isolation condenser at Millstone serves as the equivalent to a reactor core isolation cooling (RCIC) system and, thus, is an engineered safety feature.

3.3.2.2 Loss of provisions for emergency core cooling during a loss of normal auxiliary power. During a loss of offsite power, two methods of heat removal are available. Either the isolation condenser or the feedwater coolant injection system (FWCI) in conjunction with the pressure relief valves can remove decay heat. The isolation condenser system operates by natural circulation without the need for driving power other than the DC electrical system used to place the condenser system in operation. The operation of the FWCI system requires an AC

power source. Given the loss of offsite power, the gas turbine generator provides the driving force to the FWCI system. The loss of the gas turbine generator would mean the loss of the FWCI system, one loop of the LPCI system, and one loop of the core spray system.

On two occasions within a one-week period, the gas turbine generator and the isolation condenser were unavailable simultaneously (RO 76-10, RO 76-12). Consequently, the FWCI and isolation condenser were simultaneously unavailable. The isolation condenser system was inoperable due to retubing activities required after the pressure transient on February 12, 1976. On March 8, the gas turbine generator failed to start due to a maladjusted governor. The governor being out of adjustment was attributed to the higher test frequency of the gas turbine while the isolation condenser was operable. The governor was readjusted and the gas turbine declared operable after a successful start.

On March 15, the gas turbine again failed to start due to problems with the governor. The electric governor-magnetic board which provides signal conditioning of various input signals and an output signal to the gas turbine speed control system failed. The failure was due to the higher test frequency of the gas turbine generator.

3.3.2.3 ECCS failures. Two events challenged the integrity of the ECCS. On January 19, 1971, operability tests of the core spray and LPCI systems revealed several component failures. Two core spray injection valves, one on each core spray loop, failed. Testing of the LPCI system revealed two other valves that would not operate properly. After discovering these failed valves, a turbine control valve unexpectedly closed with the reactor at full power. The other three turbine control

valves opened wider to compensate. Pressure oscillations of 25% caused reactor power fluctuations of 8%. As soon as control was switched from the mechanical pressure regulator to the electrical pressure regulator, the oscillations stopped. The four valves in the ECCS failed because their torque switches were set too low.

The LPCI system and both core spray loops are all capable of supplying emergency cooling during a large line break with or without normal offsite power. If offsite power is lost, either the standby diesel generator or the gas turbine generator must be available. Hence, given a large pipe rupture, one of the three aforementioned systems must be available. On January 19, 1971, the integrity of each system was suspect.

On September 14, 1979, a situation arose that was not specifically considered in the safety analysis report. A review of the control circuitry and logic associated with the 4160 V circuit breakers revealed that a loss of power could occur to the supply for the ECCS electrical buses. The Loss of Normal Power (LNP) initiation logic senses the loss of power and immediately starts the emergency power sources. With the design error in the logic system, the loss of power would not have been sensed, and the emergency power system would not automatically start. A modification to the LNP initiation logic corrected the design error.

3.3.2.4 Loss of offsite power with partial loss of emergency power. On August 10, 1976, the Millstone 1 plant lost all offsite power due to salt buildup on the 354 kV lines and insulators. Prior to the loss of power, the gas turbine generator and the diesel generator were running without a load, as a precaution during storm conditions. The

plant was operating at 45% power and tripped during the loss of offsite power. The gas turbine generator failed on the loss of normal power; however, the diesel generator accepted load successfully. The gas turbine generator restarted and ran for ~8 min before tripping again (RO 76-29).

Investigation revealed that operator error caused both failures of the gas turbine generator. The generator has both AC and DC auxiliaries. The gas turbine generator is the primary source of AC power for its own auxiliaries once it is up to rated speed and voltage. The design also Provides for an alternate source of AC power during testing. Transfer between the two sources of power is accomplished by a manually operated throwover switch.

During precautionary operation of the gas turbine generator, the operators performed a bimonthly surveillance test using the alternate source of AC power. At the conclusion of the test, the throwover switch was not returned to its normal (primary source) position. Upon loss of offsite power and unit trip, the alternate source of AC power was lost, causing the gas turbine generator to trip.

In its normal position, the throwover switch also enables automatic transfer from the DC auxiliaries (only used to start the unit) to the AC auxiliaries. The unit restarted after the initial trip. Transfer from DC auxiliaries to AC was not accomplished. The unit ran for 8 min and then tripped a second time when the DC batteries failed.

Recurrence of the event was mitigated by the following corrective actions:

1. The procedure for testing the gas turbine was rewritten so that the AC auxiliaries are energized from the primary source during surveillance testing.
2. The throwover switch was l ked into its normal position.
3. A breaker position monitoring circuit was installed to alert the operators of an incorrect alignment of the power sources for the AC auxiliaries.

The gas turbine generator serves as the emergency power source of 4.16 kV power for the feedwater coolant injection (FWCI) pumps and control, which are part of the ECCS. This event seriously compromised the safety of the plant. For over 3 h the lone diesel generator was the only source of power to Millstone 1. The isolation condenser, which requires only DC power to place it in service, was used to cool the core. The isolation condenser had experienced severe problems only 6 months prior to this incident (see Sect. 3.3.2.1). Had it failed during the loss of normal power, the diesel generator would have carried the entire burden of supplying power for cooling the core via the low pressure ECCS. In general, diesel generators have had historically high failure rates; however, the lone diesel generator at Millstone 1 has a very good performance record.

3.3.2.5 <u>Complete and potential loss of emergency power</u>. Millstone 1 experienced one complete loss of emergency power and later discovered that a single failure mode existed in the emergency power system. On December 10, 1977, a test of the FWCI system revealed a fault in the gas turbine generator's governor (RO 77-39). Since the emergency power system consists of the gas turbine generator and a diesel generator, the

diesel generator was tested for operability. The diesel generator failed the operability test. The cause was unknown.

The single failure point in the emergency power system was discovered through a fault tree analysis of the electrical control circuitry (LER 81-02).

The analysis identified that the potential existed for a single relay failure in either of the two LNP circuits to prevent both the gas turbine generator and the diesel generator from energizing the 4160 V emergency buses. The relay failure would occur if the contacts of the time delay relay in either LNP circuit failed to reopen following a loss of normal power initiation signal. This would result in a continuous trip signal to all 4160 V and certain 480 V circuit breakers. In this situation, the buses could be energized by removing the control circuit fuses and manually operating the circuit breakers. The installation of a second time delay relay corrected the design error.

3.3.2.6 Inadvertent criticality. While performing a shutdown margin test on November 12, 1976, an inadvertent criticality and reactor trip occurred (RO 76-34). An operator error in selecting rods for the test caused the unplanned criticality.

The test is performed by positioning the highest worth rod (46-23) to notch position 10, the diagonal control rod (42-19) is then withdrawn to notch position 10, followed by full withdrawal of the maximum worth rod. The licensed reactor operator incorrectly notched out adjacent control rod 46-19 (instead of 42-19, the correct and designated rod). Without recognizing his selection error, he then withdrew the highest worth rod. The reactor tripped on high flux.

At the time of the incident, the rod worth minimizer had been by-passed and the operator was performing the test by himself — a violation of test procedures. The circumstances of the trip were reported to the supervisor who dismissed the condition as "spurious noise." Per NRC, normal procedures should be that the operator believe all instrument indications as true, unless proved otherwise.

A second test was performed contrary to procedural requirements concerning evaluation of instrumentation. Again, the operator erroneously withdrew rod 46-19. Subsequent withdrawal of rod 46-23 resulted in a flux increase, and the high worth rod reinserted to avert a second trip. Following the recognition of the previous rod selection errors, a third shutdown margin test was successfully performed. Procedural requirements were again violated as the third test was performed without assessing the potential for radiation exposure or fuel damage caused by the two criticalities. The incident was not reported to the appropriate management personnel until their arrival on the next work day, another violation of procedures.

Refueling and fuel movement were suspended for three days by the NRC. No personnel exposures occurred. The licenses of the two operators involved were suspended.

This incident represents the only major operator error committed over the operating experience of Millstone 1. The multiple procedural errors resulted in a $15,000 fine.

3.3.2.7 <u>All control rod drive accumulators require replacement.</u> In January 1973, two accumulators gave indications of leaking water into the instrumentation block (Ltr. 1/26/73). Investigation revealed that

the nickel and chromium plating had flaked from areas of the inner walls of the two accumulators. The discovery prompted the inspection of the other 143 accumulators. All had at least minor plating defects, most of which were blisters or pits. Only six of the accumulators had amounts of blistering or flaking that were sufficient to possibly impair the operation of the control rod drive mechanism. These six were replaced with new ones. Flushing the entire control rod drive hydraulic system cleaned the system of foreign particles. Millstone 1 implemented a program for close monitoring of the instrumentation and promptly investigating any abnormalities in the system. No events of this type have occurred since the new program was implemented.

3.3.2.8 Recirculation pumps trip with no alarm given. On September 15, 1981, voltage fluctuations in the 125 V DC electrical system resulted in spurious isolation of the Anticipated Transient Without Scram (ATWS) system (LER 81-25). Reenergizing the ATWS system caused an automatic trip of both recirculation pumps. Since the pumps were operating at minimum speed (reactor power was at 1%), pump coastdown and natural circulation maintained recirculation flow. With no significant change in the differential pressure across the recirculation pumps and no annunciated alarm for recirculation pump trip, control room personnel were not immediately aware of the event.

The forced recirculation flow is required to provide mixing of feedwater entering the reactor vessel to mitigate the consequences of rapid changes in reactor moderator density. The flow also eliminates the potential for high local reactor vessel stresses due to thermal stratification of reactor coolant. Although natural circulation flow

provides some mixing, forced circulation is required to maintain the parameters within acceptable limits at high power levels.

The effects of no forced circulation were analyzed with respect to fuel and reactor vessel thermal effects for this event. The effects were negligible due to the low reactor power level and the short amount of time at which there was no forced circulation flow before the reactor scram occurred. An annunciator was installed off the recirculation pump field breaker to alert control room personnel of a recirculation pump trip. Additionally, after the ATWS system isolates, operator action is required to reset the system.

3.3.2.9 <u>Loss of pressure control followed by a blowdown</u>. On October 10, 1971, the Millstone 1 plant experienced a pressure transient followed by a blowdown of 75,000 gal of water to the torus (AO 71-20). With the reactor at 100% power, the electric pressure regulator caused the pressure to rise to 1040 psig. The operator placed the mechanical pressure regulator into service, but this did not mitigate the transient. The reactor scrammed on a high flux reading on the average power range monitor (APRM). The turbine tripped and the turbine bypass valves opened. As the pressure of the vessel began to drop, the number one turbine valve failed to close. The main steam isolation valves closed, but pressure continued to decrease in the vessel. It was then discovered that a relief valve that opened at the time of the scram failed to reseat.

The stuck-open valve seated when pressure dropped to 263 psig. The isolation condenser was then put into service and a normal cooldown proceeded.

During the transient the reactor pressure dropped from 1040 psig to 263 psig, and the moderator temperature dropped from 525°F to 390°F in 18 min. This represents a significant stress on the vessel even though no technical specifications were violated. General Electric determined that the vessel blowdown conditions would not affect the integrity of the vessel.

Failure of the electric pressure relief valve was attributed to a loose dashpot connection to the pressure regulator torque tube.

The relief valve itself experienced two failures. First, the valve opened at 1040 psig, instead of 1095 psig. The set point change was caused by relaxation of the set pressure-adjustment spring due to its exposure to temperatures near 550°F. The valves had been insulated with asbestos blankets, and the reduction in heat transfer ability caused the valve internals to be exposed to elevated temperatures. This insulation was partially removed. Second, the relief valve failed to reseat after opening. This was attributed to leaking of the pilot valve (caused by lower setpoint) and the erosion of the pilot-valve disk.

3.3.2.10 Hydrogen explosion in off-gas system. On December 13, 1977, two hydrogen explosions occurred at Millstone 1. The first explosion occurred at 9:30 a.m. and was mostly confined to the off-gas system. Damage was minor and the plant reduced power while repairs commenced. A second explosion occurred at 1:00 p.m., outside the off-gas system, causing considerably more damage and injuring personnel.

The second explosion occurred when the Millstone personnel were unsuccessful in restoring water to the loop seals after the first explosion. Without these seals, the gas accumulated and was ignited by a

spark from the liquid level switch in the stack base sump. The explosion propelled the door of the room into a warehouse 200 ft away, breached the reinforced concrete ceiling beams, damaged supports of a radiation monitor for the stack, and cracked the stack. The control room was alerted to the second detonation of hydrogen in the core. The second explosion injured one man and resulted in a small, uncontrolled release of radiation.

This is a generic problem in BWRs, but most expolsions are confined inside the off-gas system, which is designed to mitigate the effects of a detonation. However, explosions outside the system have occurred, resulting in far more damage to equipment and structures.

Immediately after the event, the NRC required that all BWR licensees take steps to correct five identified deficiencies in the off-gas system, with particular attention being paid to the loop seals.

1. Review the operations and maintenance procedures of the off-gas system to assure operation in accordance with all design parameters.
2. Review the adequacy of the ventilation of spaces and areas where there is piping containing explosive gases.
3. For those spaces identified, describe what action has been taken to ensure that explosive mixtures cannot accumulate, that monitoring equipment would warn of such an accumulation should it occur.
4. Describe the design features that minimize and detect the loss of liquid from loop seals, and describe operating procedures that ensure prompt detection and resealing of blown seals.
5. Review operating and emergency procedures to ensure that the operating staff has adequate guidance to respond properly to off-gas system explosions.

3.3.3 Analysis of Reportable Events

Using the data gathered on reportable events listed in Appendix B, specific problem areas in safety-related functions were identified: (1) Pipe Cracking, (2) Safety/Relief Valves failures, (3) Stack Gas Monitor failures, (4) Losses of Emergency Power, (5) Isolation Condenser Valve failures, (6) High Radioactive Levels in Marine Life, (7) Excessive Reactor Cooldown Rates, and (8) Main Steam Isolation Valve failures.

3.3.3.1 Pipe cracking. Millstone 1 reported ten incidents of pipe cracking. Pipe cracks continue to be a generic problem in BWRs. The most significant cracking events occurred in 1972, 1976, 1980, and 1982. However, no safety related incident occurred at the plant as a result of pipe cracking.

In 1972, cracks were detected in the feedwater spargers. Subsequently, all spargers were replaced, a task which resulted in excessive down time for the plant. Two pipe cracking events occurred in 1976. The first event involved a nozzle-to-steam supply weld that cracked due to stress corrosion and was replaced. The two cracks in the main steam line supports were attributed to inadequate welds during installation. The supports were reinstalled. In 1982 a stress crack was found in a weld on a core spray system sparger and the weld was repaired.

The final pipe cracking event occurred in 1984 and was identified during the refueling outage. Intergranular stress corrosion cracking was found in three welds in the core spray, cleanup, and reactor recirculation systems. The affected welds and pipe sections were repaired or replaced.

3.3.3.2 Safety/relief valve failures. Three events involving safety/ relief valves were identified for Millstone 1. In 1977 a safety/relief valve inadvertently opened. The cause for this event is not known. Both of the remaining events involve setpoint drift that was discovered during bench tests. A total of six valves (three during 1982 and three during 1984) failed to open at the set point due to high friction at the labyrinth seal. This is a generic problem for BWRs. The BWRs Owner's Group subcommittee for safety/relief valve setpoint drift is currently evaluating this problem.

3.3.3.3 Stack gas monitor failures. Millstone 1 has experienced operational problems concerning the stack gas monitor. On two occasions during 1982 an operator failed to return the monitor to the normal operating condition following the daily test. During the ISAP plant visit it was noted that a timer and an alarm has been added to the monitor to prevent recurrences of this problem. No events involving the stack gas monitor were reported after 1982. This operational problem appears to be solved.

3.3.3.4 Loss of emergency power. Millstone 1 has experienced 47 failures of the emergency generator systems and controls. Gas turbine generator failures were the dominant contributor to the degradation of the emergency power system. The gas turbine generator failed to start or run its mission 37 times. Five of these were failures on demand, the remainder occurred under test conditions. Table 3.12 provides a description of the 37 gas turbine generator failures, along with corrective actions taken to restore the unit to service.

An analysis of gas turbine generator failures reveals that 7 of the 47 failures were attributed to a faulty speed switch. This switch was totally replaced four times during the 15-year history of the plant. The most recent replacement was required in February 1979. Five of the failures of the unit were attributed to operator or procedural errors. A procedural error caused the most significant failure of the gas turbine generator (RO 76-29) which is described in more detail in Sect. 3.3.2.4.

The emergency power system at Millstone 1 consists of one diesel generator and one gas turbine generator. If normal power to the plant is lost, the FWCI can be powered only by the gas turbine generator. Failure of the gas turbine, therefore, eliminates the cooling capacity of the FWCI. The plant does have the use of an isolation condenser at all times, even during a loss of all AC power. Unit 1 currently has use of the diesel generators at unit 2 (through manual switching) but technical specifications do not take credit for these as a source of emergency power.

The plant is located on a point and all power lines must share the same right of way for several miles. This increases the chance of losing all offsite power due to common mode failures. Despite the potential for loss of emergency power, the Millstone 1 plant has experienced remarkably few failures of its diesel generator and only one complete loss of offsite power (AO 76-29).

3.3.3.5 Isolation condenser valve failures. Millstone 1 has experienced 13 failures of the isolation condenser valves over the time period from 1970 to 1984. The condenser isolated twice as it was placed

in operation (AOs 70-5, 73-4). Both times the inboard condensate return valve was opened too wide and caused excessively high flow through the condenser. This high flow condition automatically isolates the condenser. After the first occurrence, the valve opening was restricted to reduce flow to the isolation condenser. The second failure occurred when a maintenance worker failed to properly set the valve opening restrictor after working on the valve.

Five failures have occurred with the inboard steam supply valve (1-IC-1). On December 17, 1976, the torque switch actuator setting was found to be incorrect, actuating the torque switch prematurely and not allowing the valve to move. On October 31, 1977, the breaker for the valve malfunctioned, causing the valve to be inoperable. On February 14, 1979, the valve operator gear casing was fractured, causing inoperability of the valve. On December 17, 1976, a faulty microswitch on the closing torque switch caused the valve to fail to close. This identical incident recurred on September 4, 1979. All of these failures occurred during surveillance testing, and the plant was immediately shut down after each failure.

On October 19, 1978, the condensate return isolation valve spuriously opened causing an inadvertent initiation of the isolation condenser. The initiation was then secured by an operator who closed the valve. The spurious opening signal resulted from a set point drift of two switches in separate logic channels. The set points were subsequently readjusted. On December 18, 1976, the isolation condenser inboard condensate return valve failed to close during a test. The torque switch setting was incorrect and was readjusted. On December 3, 1981, the isolation condenser valve had a valve motor fail.

Three failures have occurred with the isolation valves for the isolation condenser. In each event, limit switch problems caused the isolation valves to fail. One event (LER 82-30) involved a failure-to-close and two events (LER 84-14 and LER 84-18) involved a failure-to-open. In each case, the limit switches were readjusted and the valves placed back in service.

3.3.3.6 High radioactive levels in marine life. Millstone 1 reported nine incidents of higher than allowed levels of radioactive silver and cobalt in marine life during the operating history of the plant. All of those events occurred in the period from 1981 through 1984. The majority of the events (7) occurred during 1982 and 1983. In each event the levels of radioactivity measured were extremely small but did exceed the control station average activity by greater than a factor of ten.

3.3.3.7 Excessive reactor cooldown rates. Millstone 1 experienced five incidents of excessive cooldown rates throughout its operating history. Any large cooldown rate is of concern since a thermal stress is placed on the reactor vessel and the resulting fatigue is a cumulative effect. The first and most significant cooldown occurred in 1971 (AO 71-20). The cooldown rate was equivalent to 450°F/h and 75,000 gal. of water was blown out of the system (see Sect. 3.3.2). The other four blowdowns that resulted in excessive cooldown rates occurred in 1975, 1977, 1979, and 1981 (RO 75-09, RO 77-33, LER 79-05, and LER 81-04, respectively). The cooldown rates ranged from 105 to 210°F/h. Three additional blowdowns occurred during the operating history but the cooldown rate for each occurrence was below the Technical Specification limit of 100°F/h (RO 77-17 and RO 78-04 with two blowdowns).

Four of the excessive cooldown rate events occurred due to safety/relief valve failures (as well as the other three blowdown events). Overall, twelve events occurred due to safety/relief valve failures. Half of the events produced no deleterious effects to the operation of the plant or the environment. Failures included fouled instrument air lines, a clogged filter, wiring short, valve failing to open, or setpoint drift. The other six events resulted in blowdowns with four of these producing excessive cooldown rates. On March 10, 1978, two blowdowns occurred but a review of the temperature charts revealed that the cooldown rate was not exceeded. Therefore, the six events represent seven blowdowns and excessive cooldowns. The failure modes of the safety-relief valves fell into two categories. The valve either lifted prematurely (4 occasions), or failed to close (3 occasions).

No excessive cooldown rate events occurred during 1982, 1983, or 1984. This fact is attributed to fewer failures of the safety-relief valves during the three-year period.

3.3.3.8 Main steam isolation valve failures. Eight of the reportable events occurred due to MSIV failures. Based upon the data available in the LER data files, recent MSIV failures are primarily related to the following causes: (1) poor quality control air to the pilot valves, and (2) binding of MSIV valve stems with the valve stem packing. These two failure modes are significant in that: (1) they identify mechanisms by which more than one MSIV may fail to close at the same time, and (2) they continue to occur even though corrective actions indicate that the technology is available to prevent such failures. At

Millstone 1, four of the MSIV failures were failure to close with three of these due to sticking air slide valves. The fourth failure to close was due to a parted venting slide valve. Three failures (RO 75-29, LER 79-11, LER 80-14) were MSIV related failures but not valve failures. In 1975, a valve position switch was in the wrong position due to a relay that failed to close. A relay failed to de-energize during 3 events in 1979, 1982, and 1983 due to a maladjustment of the relay limit switch. Two failures (LER 82-6 and LER 83-10) involve slow closing of the MSIVs during tests. one failure was due to an out-of-position hydraulic cylinder and the other was due to tight valve stem packing. The valves were reworked and returned to service. The last MSIV related failure was the failure rate test performed on two MSIVs (LER 80-14).

Table 3.1. Availability and capacity factors for Millstone 1

	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	Cumulative
Reactor availability	ND[a]	74.1	89.2	47.3	80.9	79.0	84.0	96.0	89.0	79.1	76.0	69.0	80.4	96.7	79.6	78.1
Unit availability	ND	69.3	88.1	45.5	79.1	75.6	76.1	89.6	87.6	77.3	69.0	51.6	79.9	95.6	78.8	73.6
Unit capacity (MDC)[b]	ND	ND	ND	33.8	63.1	68.4	66.1	84.1	81.3	73.7	59.0	44.0	71.2	93.5	75.2	66.0
Unit capacity (DER)[c]	ND	ND	ND	33.2	59.6	68.4	65.6	83.4	80.5	73.0	58.5	43.6	70.5	92.6	74.6	65.4

[a]ND = No data.

[b]MDC = maximum dependable capacity.

[c]DER = design electrical rating.

Table 3.2. Events of radioactivity releases or personnel exposures at Millstone 1

Number	Event date	Cause	Description
—	3/25/74	A	Three workers were overexposed due to poor ventilation in area
—	1974	A	Badge readings showed three men exceeded their dose limits
AO 75-5	3/27/75	E	Wiring error caused flow of contaminate into boiler system
AO 75-6	3/30/75	A	Inadvertent discharge of radioactive liquid to environment
—	9/75	H	Worker exposed to airborne activity. Did not have work permit
—	10/75	H	Two workers exposed to airborne activity. Exhaust trunk was not operating
—	11/76	B	Two unmonitored liquid release paths discovered
RO 76-17	4/23/76	B	Noble gas release rate exceeded limits
RO 77-40	12/13/77	D	Two hydrogen explosions caused excessive release out the stack
LER 81-02E	6/22/81	H	Unmonitored radioactive liquid waste released
LER 81-03E	8/13/81	H	Unmonitored release of liquid effluent

Table 3.2. Summary of radioactivity released from Millstone 1

Release (curies)	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984
Airborne:															
Total noble gases	4.1E+2	2.76E+5	7.26E+5	7.90E+4	9.12E+5	2.97E+6	5.07E+5	6.20E+5	5.66E+5	2.06E+4	1.19E+4	1.43E+4	8.33E+3	6.34E+3	2.81E+3
Total I-31	3.2E-10		1.23E+0	1.54E-1	3.18E+0	9.77E+0	2.19E+0	4.66E+0	3.19E+0	4.01E-1	2.14E-1	7.52E-2	9.92E-2	2.92E-2	2.05E-2
Total halogens	NA	3.94E+0	1.23E+0	1.54E-1	3.18E+0	6.29E+1	3.65E+1	6.10E+1	3.19E+0	4.01E-1	3.16E+0	1.05E+0	1.41E+0	5.69E-1	5.91E-1
Total particulates	NA	5.87E-2	8.75E-6	4.10E-2	8.77E-2	1.88E-1	1.49E-1	2.01E-1	1.36E+0	1.89E-1	1.13E-1	6.86E-2	1.10E-1	3.33E-2	4.10E-2
Total tritium	NA	3.21E+0	4.21E+0	1.69E+0	7.85E+0	1.72E+1	2.87E+1	6.52E+1	3.36E+1	5.30E+1	9.55E+1	9.47E+1	5.38E+1	7.59E+1	7.71E+1
Liquid:															
Total mixed products	NA	1.97E+1	5.15E+1	3.34E+1	1.98E+2	1.99E+2	9.65E+0	5.27E-1	1.74E-1	2.09E-1	7.16E-1	3.96E-1	1.15E+0	8.08E-1	3.78E-2
Total tritium	NA	1.27E+1	2.09E+1	3.70E+0	2.41E+1	8.03E+1	2.01E+1	4.41E+0	2.22E+0	7.92E+0	2.73E+1	2.62E+0	6.21E+0	8.38E+0	8.58E+0
Total noble gases	NA	NA	0	0	0	1.11E+1	3.56E-1	3.67E-1	7.65E-1	7.00E-1	4.92E-1	4.00E-2	4.12E-1	2.43E-1	1.94E-2
Solid:															
Total	1.1E+0	2.61E+2	1.64E+3	1.51E+3	2.57E+3	2.58E+3	1.70E+3	3.03E+3	8.15E+4	1.16E+3	4.66E+3	1.83E+3	1.08E+3	6.82E+2	1.97E+3

Table 3.4. Forced shutdown summary for Millstone 1

	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	Totals
I. Forced shutdowns																
1. Total number	8	42	8	20	7	11	8	14	6	7	1	9	3	7	1	152
2. Total hours down	102	1336	585	2692	91	976	624	738	197	424	13	[illegible]	99	395	42	10155
3. Cause																
A. Equipment failure		32 (1207)	8 (585)	16 (2594)	7 (91)	8 (502)	7 (319)	14 (738)	5 (97)	7 (424)	1 (913)	7 (1534)	1 (30)	4 (199)		122 (8533)
B. Maintenance or testing	8 (102)	3 (121)				1 (45)			1 (100)			1 (55)			1 (42)	15 (844)
D. Regulatory restriction				1 (236)										1 (139)		2 (375)
E. Operator training/license exam																
F. Administrative																
G. Operational error		2 (8)		1 (62)								1 (52)	1 (18)			7 (140)
H. Other													1 (53)	2 (45)		3 (98)
4. Shutdown method																
A. Manual	3	11	4	12	2	7	5	2	2	2		1		1	1	55
B. Manual scram						1		4				2	1			9
C. Automatic scram	5	31	4	8	5	3	2	8	3	3	1	4	2	4		83
D. Other							1		1	1		2				5
II. Total number of DBE related shutdowns (these are included in totals of Part I)	4 (53)	25 (858)	4 (218)	3 (151)	6 (23)	0 (0)	0 (0)	2 (65)	3 (39)	3 (85)	1 (13)	4 (1436)		1 (12)		55 (2053)
1. Reactor vessels (CA)				6												6
2. Coolant recirculation system (CB)		2		4	1			1				1				9
3. Main steam systems and controls (CC)	1	8	1	3	2	3		1	2	1	1			1	1	29
4. Main steam isolation systems (CD)	1			1				1		1						4
5. Reactor core isolation cooling system (CE)	1	1		1			2									5
6. Reactor coolant cleanup systems (CG)								1		1						2
7. Feedwater system (CH)		1	1	1	2			1		1		2	1			12
8. Offsite power systems (EA)	1															1
9. AC onsite power systems (EB)						1	2									5
10. Onsite power systems (ED)									1							1
11. Emergency generator (EE)														1		1
12. Emergency lighting systems (EF)				1												1
13. Turbine generator and controls (HA)	2	26	1		1		1	1	1	1		3	1	2		36
14. Main steam supply system (HB)								5	1							6
15. Main condenser systems (HC)	1	2										1	1		1	6
16. Turbine bypass systems (HE)	1															1
17. Circulating water systems (HF)		2														2
18. Condensate and feedwater systems (HH)		1														1
19. Steam generator blowdown systems (HI)		1														1
20. Reactor trip systems (IA)		1						1	1			1				6
21. Instrumentation not required for safety (IE)														1		1
22. Instrumentation (IF)														1		1
23. Gaseous radwaste management system (MB)								1								1
24. Compressed air systems (PA)								1		1						2
25. Reactivity controls systems (RB)				2												2
26. Reactor core (RC)						1										1
27. Reactor containment systems (SA)		1	1		1									1		4
28. Emergency core cooling system (SF)										1						1
29. Low pressure safety injection system (SF-B)						1										1
30. Core spray system (SF-D)		2														2
31. Station service water systems (WA)						1										1

Number of hours associated with cause of shutdown is in parentheses.

Table 3.5. Power reduction summary for Millstone 1

	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	Totals
1. Power reductions																
1. Total number					7	3	1	3	10	3	1	3	7	7	5	49
2. Cause																
A. Equipment failure					7	2		1	10		1	2				23
B. Maintenance or testing						1		2		3		1	7	7	5	26
H. Other							1									1
3. System involved																
A. Turbine-generator (HA)					3			1								4
B. Main condenser (HC)					2			2	10	3		1	5	7	2	32
C. Reactor (RB)							1									1
D. Reactor (RC)													1		1	3
E. Electric power (EB)					1	1										2
F. Reactor coolant (CC)					1	1							1			3
G. Reactor coolant (CD)											1					1
H. Other instruments required for safety (IE)												1				1
I. Station service water (WA)												1				1
4. Total number of DBE's related power reductions (included in totals of Part 1)					3											3

Table 3.6. Non-DBE shutdowns and power reductions

	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	Total
Equipment failure N1.0		1														1
Failure on demand N1.1	1	1					1						1		1	5
End of design life N1.1.4	2	10	3	11	6	12	7	9	11	6		2	4	9	4	89
Instrumentation and controls N2.0																
Hardware failure N2.1		1		3		1		2					2	1		10
Spurious signal N2.4		1		2				4	1	1						9
Non-DBE coolant loss (lea^s) N3.0																
Primary system N3.1		2	1									1	1	2	1	8
Maintenance error N5.0																
Failure to repair N5.1						1										1
Operator error N6.0																
Incorrect action N6.1		2										1				3
Inadvertent action N6.3				1									1			2
Regulatory restriction N8.0																
Backfit/reanalysis N8.3				1												1
External events N9.0																
Environment induced N9.2							1									1
Environmental operating N10.0				1									1			2
	3	18	4	19	6	14	9	15	13	7	0	8	9	12	6	136

Table 3.7. DBE initiating events at Millstone 1

	DBE category	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	Totals
Feedwater system malfunctions that result in a decrease in feed water temperature	D-1.1		1														3
Feedwater system malfunctions that result an increase in feedwater flow	D-1.2			1	1										1		3
Steam pressure regulator malfunction or failure that results in increasing steam flow	D-1.3	2			1												3
Steam pressure regulator malfunction or failure that results in decreasing steam flow	D-2.1	1	1									1					3
Loss of external electric load	D-2.2		2														2
Turbine trip (stop valve closure)	D-2.3	1	18	3		4				3	2	1	2		1		34
Inadvertent closure of main steam isolation valves	D-2.4					1			1								2
Loss of condenser vacuum	D-2.5		1														1
Loss of normal feedwater flow	D-2.7				1	1							1				3
Single or multiple reactor recirculation pump trips	D-3.1												1				1
Inadvertent opening of a pressurizer safety or relief valve in a PWR or a safety or relief valve in a BWR	D-6.1		1						1		1						3
TOTAL		4	24	4	3	6	0	0	2	3	3	2	4	0	2	0	58

Table 3.8. Summary of systems involved in reportable events at Millstone 1

System	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	Totals
Reactor	1	1	1	4			4	1	1	5	1	1	1	1		22
Reactor coolant	4	8	14	8	8	9	10	14	7	11	7	7	9	11	8	135
Engineered safety features		12	4	3	4	1	4	7	4	8	2	9	6	4	5	73
Instrumentation and controls		4	7	2		6	3	7	7	10	6	14	5	13	2	86
Electrical power	2	7	4	2		3	5	4	4	2		6	8	1	1	49
Fuel handling									1							1
Auxiliary water		4			1	3			1			1	5	3		18
Steam and power	3	3	1		1	1	1	1	3			1	1	7		23
Radiation protection								4		2	1	1		1	1	10
Radioactive waste management						2						2	2			6
No system applicable			1	1	2	1	1	1	2		1			1		11
Other Auxiliary systems										1		2		2	1	6
TOTAL	10	39	32	20	16	26	28	39	30	39	18	44	37	44	18	440

Table 3.9. Causes of Reportable events for Millstone 1

Cause	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	Totals
Administrative		1				2	2	1	1	2		2		1		12
Design	4	8	3	2	4	3	4	4	2	2	2	3				41
Fabrication											1		3	0		4
Inherent failure	2	8	19	10	7	15	13	24	16	24	9	33	28	28	2	231
Installation	3	5	2	1	1	2	2	2	1	3	2		3	1	1	29
Lightning		1														1
Maintenance	1	6	7	6	4	5	7	4	8	2	3	3	2	7	5	70
Operator	1	2		1			1	1		2		3	4	2	0	17
Weather								1								1
TOTAL	11	31	31	20	16	27	29	37	28	35	17	44	33	39	8	406

Table 3.10. Summary of significant events at Millstone 1

Significance category	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	1982	1983	1984	Totals
S1							2	1								3
S2				1			1									2
S3																0
S4												1				1
S5																0
S6																0
S7		2		1			5			1		1				11
S8							2	1								2
S9																0
TOTAL	0	2	0	2	0	0	10	2	0	1	0	2	0	0	0	19

Table 3.11. Tabulation of significant events at Millstone 1

Report section	Report number	Accession number	Significance	Event description
3.3.2.3	AO 71-01	62296	S7	Four simultaneous valve failures rendered ECCS inoperable
3.3.2.9	AO 71-20	66996	S7	Failure of steam turbine bypass valve caused a blowdown
3.3.2.7		77956	S7, S2	All control rod drive accumulators require replacement
3.3.2.1	RO 76-04	111647	S7	Tube failures in Isolation Condenser resulted in radiation release
3.3.2.2	RO 76-10	112309	S7, S1	Gas turbine generator inoperable while Isolation Conderser inoperable
3.3.2.2	RO 76-12	112310	S7, S1	Gas turbine generator inoperable while Isolation Condenser inoperable
3.3.2.4	RO 76-29	116780	S7, S8	Gas turbine genertor tripped on incorrect feed during a LOOP
3.3.2.6	RO 76-34	120436	S8	Unplanned criticality achieved when wrong control rods selected
3.3.2.5	RO 77-39	144187	S7, S1	Diesel generator and gas turbine generator inoperable simultaneously
3.3.2.10	RO 77-40	144186	S9	Two hydrogen explosions in the off-gas system occurred
3.3.2.3	LER 79-26	151912	S7	Potential existed for loss of power to ECCS to go undectected
3.3.2.5 single	LER 81-02	165884	S7	Potential existed for a failure mode in the emergency power system
3.3.2.8	LER 81-25	169185	S4	Both recirculation pumps tripped, ATWS isolated, no alarm

Table 3.12. Gas turbine generator failures at Millstone 1

Report No.	Event date	Event description and problem solution
RS 70-4	11-8-70	Gas turbine generator (GTG) fails to start due to low pressure in the lube oil pump. Start-up governing system adjusted.
RS 70-4	12-4-70 (reported)	GTG fails to start due to low pressure in lube oil pump. Two additional immersion heaters installed, set points readjusted.
RS 70-4	1-8-71 (reported)	GTG fails to start within 48 s due to installation error of lube oil discharge line. Line reinstalled.
AO 71-5	2-21-71	GTG fails to start after main turbine trip due to blown fuse and faulty relay. Fuse and relay replaced.
AO 71-8	4-22-71	GTG inoperative due to procedural errors. An operator left a switch in the wrong position. Operators instructed as to proper procedure.
AO 71-12	5-27-71	GTG failed to reach startup speed due to a short circuit in speed switch. Switch replaced.
AO 71-24	11-2-71	GTG failed to ignite due to loose solder connections on a transistor speed switch. Transistor replaced.
AO 71-25	11-30-71	Procedural error caused a loss of heating of the lube oil for the GTG. Operators instructed as to proper operation.
AO 72-3	2-4-72	GTG failed to start after plant trip due to wiring errors in vibration monitor package. Errors fixed.
AO 72-11	3-9-72	GTG failed to start after plant trip due to faulty transistor in speed switch. All transistors replaced.
AO 73-5	4-5-73	Operator disabled GTG by turning wrong controller. Cover placed over controller.
AO 75-4	1-29-75	GTG removed from service to replace faulty relay.
AO 75-8	5-20-75	High generator lube oil temperature due to incorrect valving caused trip of GTG. Valves locked into correct position.

Table 3.12. (Continued)

Report No.	Event date	Event description and problem solution
AO 76-8	2-29-75	GTG did not start due to improper governor setting. Governor readjusted.
AO 76-10	3-8-76	During daily testing of GTG, unit failed to start due to improper governor setting. Governor readjusted.
RO 76-12	3-15-76	GTG declared inoperable due to governor failure. Switches replaced.
AO 76-29	8-10-76	GTG became inoperable when it could not accept plant load on reactor trip. Cause was incorrect AC feed to GTG auxiliaries. AC feed restructured.
AO 76-30	8-31-76	GTG inoperable on overspeed condition due to faulty speed switch. Switch replaced.
LER 77-27	9-9-77	Spurious noise causes GTG to fail to complete startup sequence. No repair reported.
LER 78-12	5-19-78	GTG failed to start due to incorrect fuel scheduling. No repair reported.
LER 78-14	6-13-78	GTG trips on overspeed due to defective speed switch channel. Speed switch assembly replaced.
LER 78-21	9-14-78	GTG tripped due to faulty speed switch. No repair reported.
LER 78-29	11-22-78	GTG inoperable due to opening of lube oil pump circuit breaker. Breaker indicator bulb replaced.
LER 79-7	2-14-79	GTG fails to start due to faulty speed switch. Switch replaced.
LER 81-20	7-14-81	GTG failed to start due to rust in the air motor start valve. The valve was cleaned.
LER 81-28	8-11-81	GTG output breaker failed to close due to corrosion on the automatic voltage regulator. The contacting surfaces were cleaned.
LER 81-31	9-10-81	GTG output breaker failed to close due to a wire wound ceramic resistor which failed open. The resistor was replaced.
LER 81-41	12-8-81	GTG governor failed due to contaminated oil. The oil system was flushed with clean oil.

Table 3.12. (Continued)

Report No.	Event date	Event description and problem solution
LER 82-11	5-8-82	GTG declared inoperable due to blown fuse for AC lube oil pump. The undervoltage relay and fuse were replaced.
LER 82-13	6-15-82	Air pressure regulating valve for GTG fails due to rust in starting air system. Valve was replaced.
LER 82-17	8-17-82	GTG fails 2 start tests due to rust in air pressure regulating fail. Valve was replaced.
LER 82-32	12-17-82	GTG fails to start due to generator overspeed. Setpoints for the stator vane controller, fuel valve servo-limiter, and the governor were readjusted.
LER 83-14	3-31-83	GTG fails to start due to failure of both ignitors. one ignitor was replaced.
LER 83-18	5-17-83	GTG fails to reach ready to load status due to worn ignitors. One ignitor was replaced.
LER 83-24	6-12-83	GTG declared inoperable due to failed EGR oil pump motor. The motor was replaced.
LER 83-25	7-12-83	GTG declared inoperable due to blown fuse during calibration. The fuse was replaced.
LER 83-26	8-16-83	GTG shut down due to an oil impregnated cable. The cable was replaced. Cables are now inspected during each refueling outage.

4. OBSERVATIONS AND CONCLUSIONS

4.1 Overall Plant Operating Experience

In general the record of the overall operating experience for Millstone 1 was judged to be better than the average for all operating BWRs in the United States, as evidenced by the above average availability and capacity factors and the relatively small number of significant reportable events. It appeared from the data that the operating performance is improving — the number of forced shutdowns experienced during the first half of the period under review was twice the number for occurrences in the second half, the number of reportable events was about the same in both the first and second halves of the period under review, and there were no significant events at all from 1981 through 1984. As of August 6, 1985, the plant had accomplished 367 days of continuous operation without interruption, which establishes a world record.

As part of the review effort, the ISAP operations assessment team visited the plant. Valuable insights were gained from discussions with plant personnel and a tour of Millstone plant facilities. On the basis of the team assessments from the plant visit it was concluded that the plant appears to be operated in an orderly and disciplined manner, and that effective mechanisms are in place to assure levels of maintenance and housekeeping which should prevent problems from occurring.

The results of the review evaluations also showed that there are some problems identifiable through trends and symptoms found in the

data. Correlations were made between the trends and symptoms and certain regulatory topics and issues. Using these correlations the operating experience review can thus be utilized to facilitate appropriate prioritization of regulatory topics and issues as they may be applied to Millstone 1.

4.2 Trends and Identified Symptoms

The analysis of the operational experience addressed recurring events and symptoms and the significance of these. Observations about general trends derived from the analyses of Availability and Capacity Factors, Forced Shutdowns, and Power Reductions are discussed below. Eight other specific trends and their relationship to regulatory issues are also discussed.

4.2.1 Trends in Plant Availability and Capacity Factors

The cumulative availability over the operating life of BWR's presently in operation (26 plants) ranges from 55.5% to 79.9%, with the average being 68.9%. Millstone 1 compares favorably to these other BWR's, in eighth place out of the twenty-six plants, with a cumulative availability of 73.6%. Most recently the plant has shown its ability to sustain prolonged periods of operation. This reduces challenges to safety systems.

4.2.2 Trends in Forced Shutdowns

In general, the plant is experiencing a lowering of the number of forced shutdowns and in the consequential number of hours of outage time.

4.2.3 Trends in Power Reductions

Other than those reductions in power necessitated for locating and repairing main condenser tube leaks, no determinable trends were discerned from the power reduction event data. The increased number of occurrences for repairing main condenser tube leaks in the last 3 years is the result of the plants' efforts to minimize generation of radwaste products needing disposal. All other causes of power reductions constituted only 35% of the total, and the causes appeared random.

4.2.4 Power Reductions for Finding and Fixing Main Condenser Tube Leaks

Over the last 2 years, through 1984, there has been a significant increase in the number of power reductions which were needed to find and fix tube leaks in the main condenser. This observation was reviewed with the utility. The utility responded that when main condenser tube leaks occur there is a consequential increase in the volume of radwaste products. The tube leaks themselves do not contribute any significant effect on the safe operation of the plant. However, the utility experienced tightening constraints on the volume of radwaste products that it can ship offsite for disposal, and accordingly, the condenser tube leaks, as contributors to these volumes, became high priority items for maintenance. Further, the utility explained that it is examining possible change out of the main condenser tubes with titanium tubes with the objective of virtually eliminating the problem.

4.2.5 Gas Turbine Generator

Problems with the Gas Turbine Generator continued to occur at Millstone 1 at a steady rate. No single component failure or operational problem dominated the undependability of the Gas Turbine. Most problems

were identified during the monthly tests Because few actual demands (either on a loss of offsite power or actuation of an ECCS) occurred.

The consequence of the failure of the gas turbine to operate on demand is a degradation of the ECCS, because the FWCI can be powered only by the gas turbine. Thus, there is a consequential reduction in the loss of capability to withstand degraded A/C power conditions, Such as will occur with USI-44 "Station Blackout." SEP Issue VIII-2 is also applicable.

4.2.6 Pipe Cracks

As with all BWRs, pipe cracks continued to be a problem at Millstone 1. Efforts to resolve this problem included replacing all feedwater spargers in 1972 and the reworking of cracked welds. In spite of these efforts, pipe cracking events continued to be reported in 1982 and 1984. USI-42 addresses Pipe Cracks in BWR's.

4.2.7 Safety/Relief Valves

During both the 1982 and 1984 refueling outages set point drift in safety/relief valves was identified as a recurring problem. This is a generic issue for BWR's that is currently being evaluated by the BWR Owners Group Subcommittee.

4.2.8 Isolation Condenser Valves

Improper functioning of isolation condenser valves was a recurring problem. Most of the problems involved steam supply valves that failed to close on demand. This type of failure had little effect on the operation of the isolation condenser but did affect the ability to isolate the containment. This concern was addressed in SEP VI-4.

4.2.9 Main Steam Isolation Valves

Main steam isolation valve failures were another generic BWR problem that occurred at Millstone 1. Problems with MSIV's that occurred during the early years of plant operation involved sticking air slide valves. Changes were implemented that improved the quality of the air supply to the valves. Also a program of increased testing and maintenance was implemented. These measures appear effective in resolving the problem.

Problems with MSIV's in recent years primarily involved failures of limit switches. The utility stated that all MSIV limit switches have now been environmentally qualified for service, and that this should resolve the problem.

Environmental qualification of equipment is addressed through Reg. Guide 1.86 and 10CFR50.49.

4.2.10 Radioactivity Levels in Marine Life

It was noted that the number of reported incidents of higher-than-allowed levels of radioactive material in marine life increased. This increase could be due to a change in reporting requirements, for this event, or due to a slow buildup of radiation levels in marine life near the plant. Sufficient information was not available during this study to make a conclusion as to the exact cause of the increase. Due to the extremely small levels of radioactive material discovered in the marine life, a buildup is not likely. The utility has submitted and requested approval for a revision to its technical specifications which would raise the allowable limits and thus eliminate the need to submit reports of extremely low levels.

4.2.11 Other Observations

Two other problem areas that were originally identified in SEP now appear to be resolved. The first involved excessive reactor cooldown rates. There were no recurrences of excessive reactor cooldown rates during the years 1980-1984 and it is concluded that this problem has been resolved. The second involved an operational problem with the stack gas monitor. An alarm was added to the monitor in 1982 and subsequently no further events have been reported. This operational problem appears to be resolved.

4.3 Conclusions

This review identified no major challenges to plant safety. As evidenced by the above average availability and capacity factors and the relatively small number of significant reportable events, Millstone was judged to have a better than average operational record.

However, a number of problems were identified and eight areas of significant recurring problems were established. For these eight areas, the data showed that two were no longer recurring. Five of the problem areas reflect concerns raised in regulatory topics. The topics correlated to these problem areas are:

- USI A-44 Station Blackout
- USI A-42 Pipe Cracks in BWR's
- 10 CFR 50.49 Environment Qualification of Equipment
- SEP Topic VI-4 Containment Isolation System, and
- SEP Topic VIII-2 Onsite Power Systems

The majority of the operating experience problems was confined to random events, reflecting many problems experienced by other operating BWRs. No other significant problems were identified in the trends and patterns analysis. Thus no direct correlation to regulatory topics was attempted for individual events.

Operational review findings provide real indicators of importance which can be used in the evaluation and prioritization of regulatory topics which are to be applied to a plant. Such reviews can provide direct correlations between the operating experience and any regulatory topics which have bearing on it. Thus the importance of a given regulatory topic to a specific plant can be established from a review of that plants operating experience. Further, this correlation can be used to establish just how the regulatory topic should be resolved for that plant. If the topic is resolved upon giving full recognition to the operational experience findings, then there should be a direct consequential and positive result in the operational performance. Thus operational performance would provide a direct measure of the effectiveness by regulatory actions.

Accordingly, the results and conclusions from this operational experience review can be included as a basis for the prioritization of the five regulatory topics identified for Millstone 1. These five topics are justified through the operating experience as deserving a higher priority than do other regulatory topics. All other regulatory topics do not appear to qualify for Such a level of prioritization from this analysis.

Appendix A

REVIEW OF FORCED SHUTDOWNS AND POWER REDUCTIONS

This appendix presents the data on shutdowns and power reductions in tabular form and lists the information sources. It describes how the information was encoded and how significance screening criteria were applied.

A.1 Scope

Data collected in this review include information about each forced shutdown and power reduction that occurred between 1970 and 1984. Forced shutdowns result from equipment failures that present an abnormal challenge to the unit's operation. Scheduled shutdowns for refueling and maintenance were not included. However, if the utility scheduled a refueling or maintenance outage to coincide with a shutdown that resulted from an abnormal event, that shutdown was included even though the utility reported it as scheduled. That portion of the outage time caused by the abnormal event was attributed to the shutdown and included in the compilations.

Power reductions provide information and details that add detail to a previous or subsequent shutdown, or indicate a safety significant trend. The power reductions were included in the proper chronological sequence with the shutdowns in the data tables for the forced shutdowns and power reductions (Table A.1).

The table includes the following data for each event listed by year:

1. date of occurrence,
2. duration (hours),
3. power level (percent),
4. notation of whether the shutdowns were also reportable events [e.g., a license event report (LER) or abnormal occurrence report (AO)],
5. summary description of events,
6. cause of shutdown (Table A.16),
7. method of shutdown (Table A.16),
8. system that was directly involved with the shutdown or power reduction (Table A.17),
9. component directly involved with the shutdown or power reduction (Table A.18), and
10. categorization of the shutdown or power reduction.

A.2 Data Sources

The review of the forced shutdowns and power reductions included data from the following sources.

1. *Nuclear Power Plant Operating Experience for 19XX*, for the years 1973—1984.
2. NUREG-0020 series (Gray Books).
3. Annual or semiannual reports from the time of startup through 1977. For 1977 through 1984, monthly operating reports were used because the utility was no longer required to file annual reports. The review of power reductions involved primarily the annuals, semiannuals, and monthly reports.

When LERs describing shutdowns and power reductions were available, their event descriptions gave additional information and helped to support significance screening.

A.3 Significance Screening

Shutdowns and power reductions were evaluated against design basis events (DBEs) found in Chap. 15 of the *Standard Review Plan* (Table A.19). DBEs are those postulated disturbances in process variables or failures and malfunctions of equipment that plants are designed to withstand. Licensees issue the results of their analyses of these events in safety analysis reports.

Generic design-basis initiating events such as "Increase in Heat Removal by the Secondary System" or "Decrease in Reactor Coolant System Flow Rate," were used as primary flags for reviewing the forced shutdowns (and power reductions). Once the generic type of event was identified, the particular initiating event was determined from the details associated with the shutdown. For example, if the reactor shuts down as a result of an increase in heat removal because a feedwater regulator valve failed open, the event falls into the category of generic Type 1 DBEs. Based on the specific initiating event (valve failed open), the event is classified as a 1.2 DBE — "Feedwater System Malfunction that Results in an Increase in Feedwater Flow." Some shutdowns were readily identifiable as specific DBEs, such as the tripping of a main coolant pump, classified as a 3.1 DBE. Once categorized as a DBE, the shutdown was considered significant regardless of the resulting effect on the plant (because a DBE had been initiated).

Loss of flow from one feedwater loop was considered sufficient to qualify as a 2.7 DBE — "loss of normal feedwater flow." The closure of a main steam isolation valve in one loop was considered sufficient to qualify as a 2.4 DBE — "inadvertent closure of main steam isolation valves."

Those shutdowns that were not DBEs were assigned NOAC categories (Table A.20) to provide more information on the failure or error associated with the shutdown. With these categories, more specific types of errors and failures could be examined through tabular summaries to focus the reviewer's attention on problem areas (safety related or not) that were not revealed by the DBE categories.

TABLE A.1 1970 FORCED SHUTDOWNS AND POWER REDUCTIONS FOR MIL-STG. 1

DATE (1970)	DURATION (HRS)	POWER (%)	REPORTABLE EVENT	DESCRIPTION	CAUSE	SHUTDOWN METHOD	SYSTEM INVOLVED	COMPONENT INVOLVED	DER(D)/ NSIC(N) EVENT CATEGORY
11/19/1970	16	NA	RO 70-40	MOMENTARY MAIN STEAM LINE HIGH FLOW SIGNAL	B	3	C	INSTRU	D1.3
11/21/1970	11	NA		REACTOR MODE SWITCH WIGGLING RESULTED IN MAIN STEAM ISOLATION	B	3	C	INSTRU	D2.1
11/22/1970	2	15		PACKING LEAK ON ISOLATION CONDENSER STEAM SUPPLY VALVE	B	1	C	VALVEX	N1.1.4
12/05/1970	2	0		ELECTRICAL PRESSURE REG CONTROL OSCILLATION CAUSED SPURIOUS LO LEVEL INDICATION	B	3	H	INSTRU	D1.3
12/23/1970	24	0	AO 70-08	CRACKED WELD ON MAIN CONDENSER	B	1	H	HTEXCH	N1.1.4
12/25/1970	23	0	AO 70-09	PRESSURE CONTROL BYPASS VALVE LINKAGE TORN FROM ITS SUPPORTS	B	1	H	VALVEX	N1.1.
12/30/1970	1	13		TURBINE TRIP DUE TO HIGH LEVEL IN MOISTURE SEPARATOR DRAIN TANK	B	3	H	TURBIN	D2.3

TABLE A.1 1971 FORCED SHUTDOWNS AND POWER REDUCTIONS FOR MILLSTONE 1

DATE (1971)	DURATION (HRS)	POWER (%)	REPORTABLE EVENT	DESCRIPTION	CAUSE	SHUTDOWN METHOD	SYSTEM INVOLVED	COMPONENT INVOLVED	PRE(D)/NSIC(N) EVENT CATEGORY
1/02/1971	1	40		TURBINE TRIP DUE TO HIGH LEVEL IN MOISTURE SEPARATOR DRAIN TANK	A	3	H	TURBIN	D2.3
1/02/1971	7	11.5		TURBINE TRIP - HIGH LEVEL IN MOISTURE SEPARATOR DRAIN TANK	A	3	H	TURBIN	D2.3
1/14/1971	16	64		TURBINE MANUALLY TRIPPED TO FIX CONDENSER LEAK	A	1	H	HTEXCH	N3.1
1/19/1971	109	0	AO 71-02	REACTOR SHUTDOWN - TURBINE CONTROL VALVE CLOSED	A	1	S	VALVEX	D2.1
1/25/1971	33	0		REACTOR SHUTDOWN TO REPAIR CORE SPRAY INJECTION VALVES	B	1	S	VALVEX	N1.0
1/27/1971	85	0		TRAVELING SCREEN DAMAGED CIRCULATING WATER PUMP-DAMAGED SHAFT	B	1	H	PUMPXX	N1.1.4
2/12/1971	24	0	RS 71-17	SPURIOUS INDICATION OF LOW REACTOR WATER LEVEL TRIP OF ECCS	A	3	C	INSTRU	N1.1.4
2/21/1971	21	75		TURBINE TRIP - HIGH MOISTURE SEPARATOR LEVEL	A	3	H	TURBIN	D2.3
3/02/1971	65	16		MAIN STEAM LINE SAFETY VALVE BLOWING STEAM	A	1	H	VALVEX	D6.1
3/12/1971	5	100		TURBINE TRIP - HIGH MOISTURE SEPARATOR DRAIN TANK LEVEL	A	3	H	TURBIN	D2.3
4/14/1971	3	0		REPAIR COMMUTATOR RINGS ON M-G SETS	B	1	C	GENERA	N1.1.4
4/19/1971	6	90		MAIN STEAM LINE LOW PRESSURE TRIP SENSING LINE BROKE	A	3	C	INSTRU	N2.1
4/21/1971	8	45		TURBINE TRIP - HIGH MOISTURE SEPARATOR DRAIN TANK LEVEL	A	3	H	TURBIN	D2.3
4/22/1971	1	100		TURBINE TRIP HMSDTL	A	3	H	TURBIN	D2.3
4/22/1971	1	100		TURBINE TRIP HMSDTL	A	3	H	TURBIN	D2.3
4/22/1971		100		.BINE TRIP HMSDTL	A	3	H	TURBIN	D2.3

TABLE A.1 1971 FORCED SHUTDOWN AND POWER REDUCTIONS FOR MILL[illegible] 1

DATE (1971)	DURATION (HRS)	POWER (%)	REPORTABLE EVENT	DESCRIPTION	CAUSE	SHUTDOWN METHOD	SYSTEM INVOLVED	COMPONENT INVOLVED	DPR(D)/ NSIC(N) EVENT CATEGORY
4/22/1971	1	100		TURBINE TRIP HMSDTL	A	3	H	TURBIN	D2.3
4/22/1971	1	100		TURBINE TRIP HMSDTL	A	3	H	TURBIN	D2.3
5/01/1971	39	60		REPAIR CONDENSATE TEST LINE	A	1	H	PIPEXX	N1.1.4
5/12/1971	17	70		REPAIR AIR LEAK IN DRYWELL	A	1	S	VESSEL	N1.1.4
5/25/1971	7	70	AO 71-11	FEEDWATER CONTROL VALVE CLOSED	A	3	C	INSTRU	D1.1
5/27/1971	57	80	AO 71-12	TURBINE CONTROL VALVES FAILED SHUT	A	3	H	VALVEX	D2.3
5/30/1971	12	20		STEAM LEAK IN MAIN STEAM LINE	A	1	C	PIPEXX	N3.1
6/11/1971	2	100		TURBINE TRIP HMSDTL	A	3	H	TURBIN	D2.3
6/11/1971	1	100		TURBINE TRIP HMSDTL	A	3	H	TURBIN	D2.3
6/24/1971	2	100	LTR 7/21/71	TURBINE FULL LOAD REJECT DUE TO LIGHTNING CAUSING LOSS OF 383 LINE	A	3	H	TURBIN	D2.2
6/25/1971	2	100		TURBINE TRIP DUE TO LIGHTNING CAUSING LOSS OF 383 LINE	A	1	H	TURBIN	D2.2
6/26/1971	3	0		SPURIOUS IRM TRIP	A	3	I	INSTRU	N2.4
8/12/1971	4	45		TURBINE TRIP HMSDTL	A	3	H	TURBIN	D2.3
8/29/1971	7	45		TRAVELING SCREEN FAILURE CAUSED LOSS OF CIRC WATER PUMPS & TURBINE TRIP	A	3	H	PUMPXX	N1.1
8/30/1971	168	0		MAIN CONDENSER LOW VACUUM TRIP	A	3	H	HTEXCH	D2.5
9/23/1971	134	80		850 PSI LOW PRESSURE TRIP	A	3	C	INSTRU	N1.1.4
9/29/1971	83	0	AO 71-19	TURBINE CONTROL VALVE MALFUNCTION	A	3	H	VALVEX	D2.3
10/03/1971	60	0		TURBINE CONTROL VALVE MALFUNCTION	A	3	H	VALVEX	D2.3
10/10/1971	248	0	AO 71-20	TURBINE CONTROL VALVE MALFUNCTION	A	3	H	VALVEX	D2.3

TABLE A.1 1971 FORCED SHUTDOWNS AND POWER REDUCTIONS FOR MILLSTONE 1

DATE (1971)	DURATION (HRS)	POWER (%)	REPORTABLE EVENT	DESCRIPTION	CAUSE	SHUTDOWN METHOD	SYSTEM INVOLVED	COMPONENT INVOLVED	OBE(D)/ NSIC(N) EVENT CATEGORY
10/22/1971	20	0		FAILURE OF AUTO PRESSURE RELIEF VALVE TO SEAT DUE TO SCORED PILOT VALVE DISC	A	3	C	VALVEX	N1.1.4
10/24/1971	10	0		APR BELLOWS LEAK	A	3	C	VALVEX	N1.1.4
12/11/1971	4	100		INADVERTENT MAKE-UP OF REACTOR PRESSURE SWITCH	G	3	C	INSTRU	N6.1
12/11/1971	4	100		INSTRUMENT ERROR ON MAIN STEAM LINE FLOW DETECTORS	A	3	C	INSTRU	N1.1.4
12/12/1971	4	100	AO 71-27	REACTOR PRESSURE SWITCH INADVERTENTLY TRIPPED	G	3	C	INSTRU	N6.1
12/15/1971	3	30		TURBINE TRIP - HIGH MOISTURE SEPARATOR LEVEL	A	3	H	TURBIN	D2.3
12/20/1971	57	100		FAULTY ISOLATION CONDENSER RETURN VALVE	A	1	C	VALVEX	N1.1.4

TABLE A.1 1972 FORCED SHUTDOWNS [illegible]D POWER REDUCTIONS FOR MILL[illegible] 1

DATE (1972)	DURATION (HRS)	POWER (%)	REPORTABLE EVENT	DESCRIPTION	CAUSE	SHUTDOWN METHOD	SYSTEM INVOLVED	COMPONENT INVOLVED	OPE(O)/ NSIC(N) EVENT CATEGORY
2/04/1972	149	70		TURBINE STOP VALVE TESTING INDUCED PRESS OSCILLATIONS RESULTING IN TURBINE TRIP	A	3	H	VALVEX	D2.3
2/11/1972	43	0		LEAKY MAIN STEAM LINE GASKETS	A	1	C	PIPEXX	N3.1
2/14/1972	129	30		IMPROPER RESPONSE FROM MAIN STEAM LINE VENTURI DIFFERENTIAL PRESS CELLS	A	1	C	INSTRU	N1.1.4
2/23/1972	168	100	AO 72-08	DEGRADED MAIN STEAM LINE VENTURI	A	1	C	INSTRU	N1.1.4
3/09/1972	62	20		FAULTY OPERATION OF THRUST BEARING WEAR DETECTOR INDUCED TURBINE TRIP	A	3	H	TURBIN	D2.3
3/12/1972	2	30		VOID COLLAPSE FROM COLD FEEDWATER INCREASE IN FLOW	A	3	C	PUMPXX	D1.2
8/18/1972	5	100		TESTING OF THRUST BEARING WEAR DETECTOR TRIPPED TURBINE	A	3	H	INSTRU	D2.3
8/29/1972	27	100		DRYWELL FLOOR DRAIN SUMP LEAKAGE	A	1	S	VESSEL	N1.1.4

TABLE A.1 1973 FORCED SHUTDOWNS AND POWER REDUCTIONS FOR MILLSTONE 1

DATE (1973)	DURATION (HRS)	POWER (%)	REPORTABLE EVENT	DESCRIPTION	CAUSE	SHUTDOWN METHOD	SYSTEM INVOLVED	COMPONENT INVOLVED	DOE(9)/NSIC(F) EVENT CATEGORY
3/03/1973	7	50		POWER REDUCTION - LEAK IN MAIN CONDENSER	A	5	H	HTEXCH	N1.1.4
3/06/1973	13	50	AO 74-01	EXCESSIVE DRYWELL LEAKAGE FROM STEM PACKING OF REACTOR RECIRC EQUALIZER VALVE	A	1	S	VALVEX	N10
3/06/1973	53	0		CONDENSATE BOOSTER PUMP NOT STARTED IN TIME-LOW WATER LEVEL	G	3	C	PUMPXX	D2.7
3/11/1973	31	40		REPAIRED MAIN STEAM ISOLATION POSITION INDICATION SWITCH	A	1	C	VALVEX	N2.4
3/13/1973	6	75		LOW LUBE-OIL PRESSURE ALARM ON RECIRC PUMP MOTOR	A	1	C	MOTORX	N1.1.4
3/14/1973	83	80	AO 73-02	BLOWN FUSE ON CRD SCRAM SOLENOID SCRAMMED GROUP II CONTROL RODS	A	3	R	CKTBRK	N1.1.4
3/19/1973	40	58		LOW LUBE-OIL PRESS ALARM REACTOR PUMP	A	1	C	PUMPXX	N1.1.4
4/18/1973	2152	18		REPLACED FEEDWATER SPARGER	A	1	C	PIPEXX	N1.1.4
7/14/1973	64	0		LEAK ON VENT LINE FOR RECIRCULATION PUMP DISCHARGE	A	1	C	PIPEXX	N1.1.4
7/16/1973	236	0		EXAMINE FOR INVERTED CONTROL ROD INTERNALS	D	1	R	CONROD	N8.3
7/30/1973	12	60		REACTOR VESSEL WATER LEVEL TRANSMITTER FAILURE	A	3	C	INSTRU	N2.1
8/10/1973	14	76		REACTOR VESSEL WATER LEVEL TRANSMITTER LEVEL MALFUNCTION	A	3	C	INSTRU	N2.1
8/10/1973	4	0		HIGH REACTOR WATER LEVEL DUE TO STARTING FEEDWATER PUMP	G	3	C	PUMPXX	D1.2
9/15/1973	36	0		LEAKING INSTRUMENT TAP IN FEEDWATER LINE	A	1	C	PIPEXX	N1.1.4
9/20/1973	25	0		LEAKING AUTOMATIC PRESSURE RELIEF VALVE	A	1	C	VALVEX	N1.1.4

TABLE A.1 1973 FORCED SHUTDOWNS AND POWER REDUCTIONS FOR MILLSTONE 1

DATE (1973)	DURATION (HRS)	POWER (%)	REPORTABLE EVENT	DESCRIPTION	CAUSE	SHUTDOWN METHOD	SYSTEM INVOLVED	COMPONENT INVOLVED	DOE(D)/NSIC(N) EVENT CATEGORY
9/21/1973	11	50		FAULT IN EPR CONTROLS OPENED TURBINE BYPASS VALVES DROPPING REACTOR PRESSURE	A	3	C	INSTRU	D1.3
10/06/1973	16	67		RECIRC PUMP MOTOR FAILED	A	1	C	MOTORX	N1.1.4
10/13/1973	76			FAULT IN MODE SWITCH CAUSED A SCRAM	A	3	C	INSTRU	N2.1
10/14/1973	20	68		TESTED RECIRC PUMP MOTOR	A	1	C	MOTORX	N1.1.4
10/28/1973	42	67		OPENED AND TESTED RECIRC LOOP CROSS-TIE VALVES	A	1	C	VALVEX	N2.4
12/07/1973	5	80		PERSON BUMPED LEVEL INSTRUMENT RACK	G	3	C	INSTRU	N6.3
12/30/1973	16	0	AO 73-42	ISOLATION CONDENSER FLANGE LEAK	A	1	C	PIPEXX	N1.1.4

TABLE A.1 1974 FORCED SHUTDOWNS AND POWER REDUCTIONS FOR MILLSTONE 1

DATE (1974)	DURATION (HRS)	POWER (%)	REPORTABLE EVENT	DESCRIPTION	CAUSE	SHUTDOWN METHOD	SYSTEM INVOLVED	COMPONENT INVOLVED	DRE(D)/ NSIC(N) EVENT CATEGORY
3/02/1974	4	50		POWER REDUCTION - SUSPECTED LEAK IN MAIN CONDENSER	A	5	H	HTEXCH	N1.1.4
6/11/1974	7	65		RECIRC PUMP SPEED CONTROL FAULT	A	3	C	INSTRU	N1.1.4
11/03/1974	15	35		TURBINE TRIP CAUSED BY MAINTENANCE ON FEEDWATER TRANSMITTER	A	3	H	INSTRU	D2.3
11/04/1974	5	40		APRM HIGH FLUX DUE TO PRESSURE OSCILLATIONS	A	3	C	INSTRU	N1.1.4
11/04/1974	9	35	AO 74-09	MSIV MALFUNCTION DUE TO MOISTURE IN THE SLIDE VALVE THAT CONTROLS VALVE ACTION	A	3	C	VALVEX	D2.4
11/15/1974	8	0	AO 74-10	POWER REDUCTION - MSIV FAILED TO CLOSE	A	5	C	VALVEX	N1.1.4
11/18/1974	15	0		POWER REDUCTION - MAIN GENERATOR EXCITER GROUND FAULT	A	5	E	GENERA	N1.1.4
12/16/1974	8	97		BROKEN STEM - FEEDWATER CONTROL VALVE PREVENTED FLOW OF FEEDWATER	A	3	C	VALVEX	D2.7
12/27/1974	1	15		POWER REDUCTION - TURBINE TRIP DUE TO HIGH LEVEL IN MOISTURE SEPARATOR	A	5	H	TURBIN	D2.3
12/27/1974	34	55		REPAIR MAIN FEEDWATER CONTROL VALVE	A	1	C	VALVEX	N1.1.4
12/29/1974	3	10		POWER REDUCTION -TURBINE TRIP DUE TO HIGH LEVEL IN MOISTURE SEPARATOR	A	5	H	TURBIN	D2.3
12/29/1974	2	10		POWER REDUCTION - TURBINE TRIP DUE TO HIGH LEVEL IN MOISTURE SEPARATOR	A	5	H	TURBIN	D2.3

TABLE A.1 1975 FORCED SHUTDOWNS AND POWER REDUCTIONS FOR MILLSTONE 1

DATE (1975)	DURATION (HRS)	POWER (%)	REPORTABLE EVENT	DESCRIPTION	CAUSE	SHUTDOWN METHOD	SYSTEM INVOLVED	COMPONENT INVOLVED	OFF(O)/ NSIC(N) EVENT CATEGORY
2/15/1975	9	20		POWER REDUCTION - DRYWELL ENTRY MADE TO PERFORM MAINTENANCE ON TRIP INDEX	A	5	R	INSTRU	N2.1
3/11/1975	87	60		BLOWN VALVE STUFFING BOX - LPCI SYSTEM	A	1	S	VALVEX	N1.1.4
5/20/1975	55	100	AO 75-09	PRESSURE RELIEF VALVE FAILED TO SEAT	A	2	C	VALVEX	N1.1.4
6/20/1975	35	15		SERVICE WATER PUMP REPAIRS	A	1	W	PUMPXX	N1.1.4
8/13/1975	16	80		POWER REDUCTION - ARCING ON B-PHASE DISCONNECT	A	5	E	CKTBRK	N1.1.4
8/18/1975	5	96		POWER REDUCTION - APR'S FAILED TO MEET TECH SPECS	B	5	C	VALVEX	N1.1.4
8/30/1975	8	0		PRESSURE REG TRANSIENT CAUSED APRM SCRAM DUE TO PLUGGED MCOG VALVE FILTER	H	3	C	FILTER	N1.1.4
9/12/1975	451	94		TRANSFORMER INSULATION BREAKDOWN	B	1	E	TRANSF	N1.1.4
10/25/1975	15	2		INSTALLING INSULATING BOLTS ON MAIN TRANSFORMER	H	1	E	TRANSF	N1.1.4
10/27/1975	19	60		MAINTENANCE ON TRANSVERSING INCORE PROBE SYSTEM	A	1	R	INSTRU	N1.1.4
11/13/1975	223	90		CRACK IN JET PUMP BREAK DETECTION SENSING LINE	A&B	-	C	INSTRU	N1.1.4
11/23/1975	9	0		OSCILLATIONS IN PRESSURE REGULATOR DUE TO DIRT IN SENSING LINES	A	3	C	INSTRU	N1.1.4
11/26/1975	8	60		PRESSURE SPIKE OCCURRED WHILE SWITCHING TO MECH PRESSURE REGULATOR	A	3	C	INSTRU	N1.1.4
12/11/1975	66	90		INSTALLED MISSING PART ON MAIN TRANSFORMER	A&B	1	E	TRANSF	N5.1

TABLE A.1 1976 FORCED SHUTDOWNS AND POWER REDUCTIONS FOR MILLSTONE 1

DATE (1976)	DURATION (HRS)	POWER (%)	REPORTABLE EVENT	DESCRIPTION	CAUSE	SHUTDOWN METHOD	SYSTEM INVOLVED	COMPONENT INVOLVED	DBE(D)/ NSIC(N) EVENT CATEGORY
2/12/1976	42	100		ARCING ACROSS HIGH VOLTAGE BUSING ON MAIN TRANSFORMER-TRIPPED GENERATOR	A	3	E	TRANSF	NI.1.4
3/08/1976	85	100	RO 76-10	INOPERABILITY OF GAS TURBINE GENERATOR DUE TO GOVERNOR OUT OF ADJUSTMENT	A&B	1	E	TURBIN	NI.1.4
3/15/1976	132	20		INOPERABILITY OF GAS TURBINE-REPLACED ELECTRONIC GOVERNOR	A&B	1	E	TURBIN	NI.1.4
7/16/1976	64	90		REPAIRED MOTOR OPERATOR OF ISOLATION CONDENSER ISOLATION VALVE	A	1	C	VALVOP	NI.1.4
8/10/1976	105	95		HIGH WINDS DEPOSITED SALT ON MAIN TRANSFORMER INSULATORS - ARCING	H	3	E	TRANSF	NS.2
8/10/1976	100	95	RO 76-29	PROBLEMS WITH SPEED CONTROL OF GAS TURBINE GENERATOR - OUTAGE EXTENSION	A	1	E	TURBIN	NI.1.4
8/22/1976	0	65		POWER REDUCTION - ADJUST CONTROL ROD PATTERN	H	5	R	CONROD	NI.1
12/01/1976	45	100		PROBLEMS WITH MAIN TURBINE GENERATOR PRESSURE REGULATOR CAUSING IT TO OPEN	A	1	H	INSTRU	NI.1.4
12/17/1976	51	100	RO 76-42	MALFUNCTION OF ISOLATION CONDENSER ISO VALVE, CLEANUP ISOLATION VALVE	A	1	C	VALVOP	NI.1.4

TABLE A.1 1977 FORCED SHUTDOWNS AND POWER REDUCTIONS FOR MILLSTONE 1

DATE (1977)	DURATION (HRS)	POWER (%)	REPORTABLE EVENT	DESCRIPTION	CAUSE	SHUTDOWN METHOD	SYSTEM INVOLVED	COMPONENT INVOLVED	OPE(O)/ NSTC(N) EVENT CATEGORY
1/03/1977	49	100		MALFUNCTION OF MAIN GENERATOR ELECTRIC PRESSURE REG DUE TO PLUGGED LINE	A	3	H	INSTRU	N2.1
1/26/1977	13	100	RO 77-05	INCREASING COOLANT CONDUCTIVITY DUE TO DEMINERALIZER MALFUNCTION	A	2	C	DEMINX	N1.1.4
3/06/1977	36	100		REPAIR OF SMALL STEAM LEAK ON MAIN LINE DRAIN LINE TO MAIN CONDENSER	A	1	H	PIPEXX	N1.1.4
4/07/1977	9	100		WHILE DOING REACTOR WATER LEVEL SURVEILLANCE,INADVERTENT REACTOR SCRAM OCCURRED	A	3	I	VALVEX	N2.4
4/23/1977	0	100		POWER REDUCTION - REPAIR STEAM LEAK ON EXTRACTION NON-RETURN VALVE	A	5	H	VALVEX	N1.1.4
5/14/1977	11	100		STEAM LEAK ON AN EXTRACTION NON-RETURN VALVE	A	3	H	VALVEX	N1.1.4
6/13/1977	8	0		FAULTY TEST SOLENOID PREVENTED MS VALVE FROM RETURNING TO OPEN POSITION	A	3	H	INSTRU	N2.1
6/14/1977	132	0		MECHANICAL PRESSURE REGULATOR SWING TRIPPED TURBINE	A	2	H	CKTBRK	N2.4
6/19/1977	6	100		MECHANICAL PRESSURE REGULATOR DID NOT TAKE CONTROL-BYPASS VALVES FAILED OPEN	A	3	H	CKTBRK	N2.4
7/12/1977	10	100		BENT ACTUATOR ON MAIN STEAM ISOLATION VALVE ALLOWED VALVE TO CLOSE	A	3	C	CKTBRK	O2.4
7/22/1977	37	100		FALSE SIGNAL TRIPPED BREAKER ON LUBE OIL PUMP FOR RECIRC MG SET	A	3	C	INSTRU	N2.4
8/06/1977	37	100	RO 77-24	LOSS OF PLANT AIR DUE TO LOSS OF COOLING WATER TO AIR COMPRESSORS	A	3	P	BLOWER	N1.1.4
8/27/1977	0	100		POWER REDUCTION - PLUG TUBES IN MAIN CONDENSERS	B	5	H	HTEXCH	N1.1.4

TABLE A.1 1977 FORCED SHUTDOWNS AND POWER REDUCTIONS FOR MILLSTONE 1

DATE (1977)	DURATION (HRS)	POWER (%)	REPORTABLE EVENT	DESCRIPTION	CAUSE	SHUTDOWN METHOD	SYSTEM INVOLVED	COMPONENT INVOLVED	DOR(D)/ NSIC(N) EVENT CATEGORY
5/21/1977	0	90		POWER REDUCTION - CONDENSER TUBE MAINTENANCE	B	5	H	HTEXCH	N1.1.4
11/22/1977	66	100		LEAK IN FEEDWATER HEATER	A&B	1	C	HTEXCH	N1.1.4
11/25/1977	55	50		AUTOMATIC PRESSURE RELIEF VALVE LIFTED PREMATURELY	A&B	2	C	VALVEX	D6.1
12/13/1977	269	80	RO 77-40	MANUAL SCRAM IN RESPONSE TO A SECOND HYDROGEN EXPLOSION IN OFFGAS STACK	A	2	H	XXXXXX	N1.1.4

TABLE A.1 1978 FORCED SHUTDOWNS AND POWER REDUCTIONS FOR MILLSTONE 1

DATE (1978)	DURATION (HRS)	POWER (%)	REPORTABLE EVENT	DESCRIPTION	CAUSE	SHUTDOWN METHOD	SYSTEM INVOLVED	COMPONENT INVOLVED	DOE(S)/NSIC(N) EVENT CATEGORY
1/25/1978	0	75		REDUCED POWER FOR MAIN CONDENSER MAINTENANCE	A	5	H	HTEXCH	N1.1.4
1/31/1978	14	90		STEAM LEAK ON 2 INCH STEAM LINE	A	1	C	PIPEXX	N1.1.4
5/19/1978	22	100		MALFUNCTION OF LEVEL CONTROL SYSTEM FOR MOISTURE SEPARATOR TRIPPED TURBINE	A	3	C	INSTRU	D2.3
5/29/1978	16	100		BROKEN AIR SUPPLY TUBE TO MOISTURE SEPARATOR LEVEL CONTROLLER TRIPPED TURBINE	A	3	H	PIPEXX	D2.3
6/10/1978	0	100		POWER REDUCTION - PLUG MAIN CONDENSER TUBES	A	5	H	HTEXCH	N1.1.4
7/03/1978	0	93		REDUCED POWER TO PLUG MAIN CONDENSER TUBING	A	5	H	HTEXCH	N1.1.4
7/14/1978	100	100		REPLACED CONTMT ISO VALVE CABLE SPLICES WITH ONES ENVIRONMENTALLY QUALIFIED	B	1	E	ELECON	N10
7/20/1978	1	100		MALFUNCTION OF MOISTURE SEPARATOR DRAIN TANK LEVEL CONTROLLERS TRIPPED TURBINE	A	4	H	INSTRU	D2.3
7/25/1978	0	80		REDUCED POWER TO PLUG MAIN CONDENSER TUBING	A	5	H	HTEXCH	N1.1.4
7/26/1978	0	80		REDUCED POWER TO PLUG MAIN CONDENSER TUBING	A	5	H	HTEXCH	N1.1.4
7/27/1978	0	80		REDUCED POWER TO PLUG MAIN CONDENSER TUBING	A	5	H	HTEXCH	N1.1.4
7/29/1978	0	97		REDUCED POWER TO PLUG MAIN CONDENSER TUBING	A	5	H	HTEXCH	N1.1.4
8/18/1978	0	95		REDUCED POWER TO PLUG MAIN CONDENSER TUBING	A	5	H	HTEXCH	N1.1.4
8/28/1978	0	40		REDUCED POWER TO PLUG MAIN CONDENSER TUBING	A	5	H	HTEXCH	N1.1.4
9/04/1978	0	70		REDUCED POWER FOR MAIN CONDENSER MAINTENANCE	A	5	H	HTEXCH	N1.1.4

TABLE A.1 1978 FORCED SHUTDOWNS AND POWER REDUCTIONS FOR MILLSTONE 1

DATE (1978)	DURATION (HRS)	POWER (%)	REPORTABLE EVENT	DESCRIPTION	CAUSE	SHUTDOWN METHOD	SYSTEM INVOLVED	COMPONENT INVOLVED	DPE(D)/ NSIC(N) EVENT CATEGORY
12/12/1978	44	100		MAIN STEAM LINE FLOW INSTRUMENTATION CHECK OUT	A	3	I	INSTRU	N2.4

TABLE A.1 1979 FORCED SHUTDOWNS AND POWER REDUCTIONS FOR MILLSTONE 1

DATE (1979)	DURATION (HRS)	POWER (%)	REPORTABLE EVENT	DESCRIPTION	CAUSE	SHUTDOWN METHOD	SYSTEM INVOLVED	COMPONENT INVOLVED	DBE(D)/ NSIC(N) EVENT CATEGORY
1/06/1979	269	100	LER 79-01	STRESS CORROSION CRACKING OF CLEAN-UP RETURN LINE	A	1	C	PIPEXX	N1.1.4
1/15/1979	30	70		CHANGE AUTOMATIC PRESSURE RELIEF VALVE TOPWORKS	A	1	S	VALVOP	N1.1.4
1/22/1979	0	97		REDUCED POWER FOR MAIN CONDENSER MAINTENANCE	B	5	H	HTEXCH	N1.1.4
2/26/1979	33	100	LER 79-05	SAFETY RELIEF VALVE LIFTED PREMATURELY AND FAILED TO RESEAT	A	2	C	VALVEX	D6.1
3/10/1979	0	97		REDUCED POWER FOR MAIN CONDENSER MAINTENANCE	B	5	H	HTEXCH	N1.1.4
3/17/1979	10	100		MSIV POSITION INDICATING PROBLEMS	A	9	C	INSTRU	N2.4
6/28/1979	0	15		TURBINE TRIP DUE TO FEEDWATER REGULATING VALVE LOCK-UP	A	3	C	VALVEX	D2.3
7/02/1979	30	90		LOSS OF BOTH PLANT AIR COMPRESSORS	A	3	P	BLOWER	N1.1.4
8/15/1979	0	70		REDUCED POWER FOR MAIN CONDENSER MAINTENANCE	B	5	H	HTEXCH	N1.1.4
12/15/1979	52	100		MAIN GENERATOR LOSS OF EXCITATION	A	3	H	GENERA	D2.3

TABLE A.1 1980 FORCED SHUTDOWNS AND POWER REDUCTIONS FOR MILLSTONE 1

DATE (1980)	DURATION (HRS)	POWER (%)	REPORTABLE EVENT	DESCRIPTION	CAUSE	SHUTDOWN METHOD	SYSTEM INVOLVED	COMPONENT INVOLVED	DOE(D)/ NSIC(N) EVENT CATEGORY
1/05/1980	648	100		REDUCED POWER TO 40% DUE TO ISOLATION CONDENSER OUT OF SERVICE	A	3	C	HTEXCH	N1.1.3
6/25/1980	13	20		ELECTRIC PRESSURE REGULATOR MALFUNCTION INDUCED APRM SCRAM	A	3	C	INSTRU	D2.1

TABLE A.1 1981 FORCED SHUTDOWNS AND POWER REDUCTIONS FOR MILLSTONE 1

DATE (1981)	DURATION (HRS)	POWER (%)	REPORTABLE EVENT	DESCRIPTION	CAUSE	SHUTDOWN METHOD	SYSTEM INVOLVED	COMPONENT INVOLVED	DEE(D)/ NSIC(N) EVENT CATEGORY
4/15/1981	3	0		LOW PWR ASCENSION TURBINE TRIP ON MOISTURE SEPARATOR HI LEVEL-NO CHANGE ON POWER	A	9	H	TURBIN	D2.3
4/21/1981	1372		LER 81-004	MANUAL TRIP ON HI VIBRATE-REACTOR SCRAM ON HI MAIN CONDENSER CONDUCTIVITY	A	2	H	TURBIN	D2.3
6/15/1981	13	0		REACTOR MODE SWITCH FAILURE	A	9	I	INSTRU	N2.0
7/06/1981				POWER REDUCTION - PROCESS COMPUTER FAILURE	A	5	I	INSTRU	N2.0
7/12/1981	26			FEEDWATER REGULATION VALVE CLOSURE	A	3	C	VALVEX	D2.7
7/18/1981				PWR REDUCTION-CONDENSATE BOOSTER PMP REPAIR/SERVICE WATER CROSS-OVER VLV REPAIR	B	5	W	VALVEX	N1.1.4
8/08/1981	35			REACTOR RECIRC PUMP 'A' TRIPPED ON GENERATOR OVERLOAD WITH 'B' PUMP OFF-LINE	A	2	C	GENERA	D3.1
8/10/1981	52		LER 81-023	UNIT SCRAM ON HI RISK TEST-SCRAM'D CHANL NOT RESET PRIOR TO TESTING ALT CHANLS	G	3	I	INSTRU	N6.1
9/05/1981	55			SHUTDOWN TO REBALANCE TURBINE	B	1	H	TURBIN	N1.1.4
9/14/1981	42			SPURIOUS ATWS SYS ISO PWR SURGE TO ATWS SYS-SCRAM AIR HEADER TO DEPRESSURIZER	A	3	I	INSTRU	N2.0
12/03/1981	0	100		POWER REDUCTION - FIND AN REPAIR MAIN CONDENSER LEAKS	A	5	H	HTEXCH	N3.1
12/28/1981	43			COND PMP TRIP DUE TO DISCREPANCY BET HOWELL LEVEL TRANSMITTER & ACTUAL LEVEL	A	3	C	INSTRU	N2.0

TABLE A.1 1982 FORCED SHUTDOWNS & POWER REDUCTIONS FOR MILLSTONE 1

DATE (1982)	DURATION (HRS)	POWER (%)	REPORTABLE EVENT	DESCRIPTION	CAUSE	SHUTDOWN METHOD	SYSTEM INVOLVED	COMPONENT INVOLVED	OBE(O)/ NSIC(N) EVENT CATEGORY
1/28/1982	0.0	91		STEAM LEAK IN STEAM TUNNEL	B	5	C	PIPEXX	N3.1
2/12/1982	30.0	91		"A" FDWTR REG VALVE CYCLED AND CAUSED SCRAM	A	3	C	VALVOP	N2.1
4/03/1982	0	91		REPAIR LEAKING MAIN CONDENSER TUBES & CHANGE CR PATTERN	B	5	H	HTEXCH	N1.1.4
4/10/1982	0	91		LEAKING MAIN CONDENSER TUBES	B	5	H	HTEXCH	N1.1.4
4/13/1982	17.8			RECIRC PUMP TRIP FROM ATWSS WHILE TRACING DC GROUND	G	2	H	HTEXCH	N6.3
4/15/1982	0			LEAKING MAIN CONDENSER TUBES	B	5	H	HTEXCH	N1.1.4
4/25/1982	0			LEAKING MAIN CONDENSER TUBES	B	5	H	HTEXCH	N1.1.4
6/26/1982	0			MG SET REBRUSHING & CR PATTERN ADJUST	B	5	R	CONROD	N1.1
7/27/1982	0			FIND AND PLUG LEAKING MAIN CONDENSER TUBES	B	5	H	HTEXCH	N1.1.4
7/31/1982	53.1			GENERATOR OUT-OF-STEP RELAY CAUSED TRIP & LOAD REJECT PLUS ATWSS SCRAM	H	3	H	GENERA	N2.1

TABLE A.1 1983 FORCED SHUTDOWNS & POWER REDUCTIONS FOR MILLSTONE 1

DATE (1983)	DURATION (HRS)	POWER (%)	REPORTABLE EVENT	DESCRIPTION	CAUSE	SHUTDOWN METHOD	SYSTEM INVOLVED	COMPONENT INVOLVED	OPER(O)/ NSIC(N) EVENT CATEGORY
1/24/1983	29.9			MOISTURE CAUSED STEAM TUNNEL HIGH TEMP ALARM SWITCH TO GROUND	A	3	I	INSTRU	N2.1
2/24/1983	138.8			IDENTIFY & REPAIR DRYWELL LEAKAGE. F/W TRANSIENT LATER CAUSED SCRAM	D	1	S	PIPEXX	N3.1
3/31/1983	12.2			HI MOISTURE SEP LEVEL PLUS TURB TRIP CAUSED RX SCRAM	H	3	H	TURBIN	O2.3
4/05/1983	0			FIND & REPAIR LEAKING MAIN CONDENSER TUBES	B	5	H	HTEXCH	N1.1.4
4/29/1983	0			FIND & REPAIR LEAKING MAIN CONDENSER TUBES	B	5	H	HTEXCH	N1.1.4
5/19/1983	0			FIND & REPAIR LEAKING MAIN CONDENSER TUBES	B	5	H	HTEXCH	N1.1.4
6/07/1983	33.2			TURBINE TRIP ON HI RX WATER LEVEL	H	3	H	TURBIN	O1.2
6/18/1983	90.7		83-24	INOP GAS TURBINE	A	1	E	TURBIN	N1.1.4
6/25/1983	16.4			SCRAM ON LOW AIR PRESS. WHEN INST. AIR COMPRESSOR TRIPPED	A	3	I	BLOWER	N1.1.4
9/04/1983	0			FIND & REPAIR LEAKING MAIN CONDENSER TUBES	B	5	H	HTEXCH	N1.1.4
10/05/1983	0			FIND & REPAIR LEAKING MAIN CONDENSER TUBES	B	5	H	HTEXCH	N1.1.4
10/24/1983	0			FIND & REPAIR LEAKING MAIN CONDENSER TUBES	B	5	H	HTEXCH	N1.1.4
11/27/1983	62.0			REPAIR LEAKING "A" RECIRC PUMP SEAL	A	1	C	PUMPXX	N3.1
12/21/1983	0			FIND & REPAIR LEAKING MAIN CONDENSER TUBES	B	5	H	HTEXCH	N1.1.4

TABLE A.1 1984 FORCED SHUTDOWNS & POWER REDUCTIONS FOR MILLSTONE 1

DATE (1984)	DURATION (HRS)	POWER (%)	REPORTABLE EVENT	DESCRIPTION	CAUSE	SHUTDOWN METHOD	SYSTEM INVOLVED	COMPONENT INVOLVED	ORE(O)/ NSIC(N) EVENT CATEGORY
1/16/1984	0			FIND & REPAIR LEAKING MAIN CONDENSER TUBES	B	5	H	HTEXCH	N1.1.4
2/21/1984	0			ADJUST CONTROL ROD PATTERN	H	5	R	CONROD	N1.1
8/02/1984	41.5			DRYWELL ENTRY TO FIND & FIX LEAK IN RECIRC SYS RISER INST VALVE	B	1	C	VALVEX	N3.1
9/17/1984	0			REBRUSH "A" RECIRC MG SET	B	5	C	MOTORX	N1.1.4
10/20/1984	0			FIND & REPAIR LEAKING MAIN CONDENSER TUBES	B	5	H	HTEXCH	N1.1.4
12/05/1984	0			REPAIR STICKING "B" FEEDWATER REG VALVE	B	5	C	VALVEX	N1.1.4

Table A.16. Codes for causes of forced shutdowns or power reductions and methods of shutdown

	Causes
A	Equipment failure
B	Maintenance or testing
C	Refueling
D	Regulatory restriction
E	Operator training and license exams
F	Administrative
G	Operational error
H	Other
	Methods
1	Manual
2	Manual scram
3	Automatic scram
4	Continuation
5	Load reduction
9	Other

Table A.17. Systems involved with forced shutdowns and power reductions

System	Code
Reactor	RX
Reactor vessel internals	RA
Reactivity control systems	RB
Reactor core	RC
Reactor coolant and connected systems	CX
Reactor vessels and appurtenances	CA
Coolant recirculation systems and controls	CB
Main steam systems and controls	CC
Main steam isolation systems and controls	CD
Reactor core isolation cooling systems and controls	CE
Residual heat removal systems and controls	CF
Reactor coolant cleanup systems and controls	CG
Feedwater systems and controls	CH
Reactor coolant pressure boundary leakage detection systems	CI
Other coolant subsystems and their controls	CJ
Engineered safety features	SX
Reactor containment systems	SA
Containment heat removal systems and controls	SB
Containment air purification and cleanup systems and controls	SC
Containment isolation systems and controls	SD
Containment combustible control systems and controls	SE
Emergency core cooling systems and controls	SF
Core reflooding	SF-A
Low-pressure safety injection system and controls	SF-B
High-pressure safety injection system and controls	SF-C
Core spray system and controls	SF-D
Control room habitability systems and controls	SG
Other engineered safety feature systems and their controls	SH
Containment purge system and controls	SH-A
Containment spray system and controls	SH-B
Auxiliary feedwater system and controls	SH-C
Standby gas treatment systems and controls	SH-D
Instrumentation and controls	IX
Reactor trip systems	IA
Engineered safety feature instrument systems	IB
Systems required for safe shutdown	IC
Safety-related display instrumentation	ID
Other instrument systems required for safety	IE
Other instrument systems not required for safety	IF

Table A.17 (continued)

System	Code
Electric power systems	EX
Offsite power systems and controls	EA
AC onsite power systems and controls	EB
DC onsite power systems and controls	EC
Onsite power systems and controls (composite ac and dc)	ED
Emergency generator systems and controls	EE
Emergency lighting systems and controls	EF
Other electric power systems and controls	EG
Fuel storage handling systems	FX
New fuel storage facilities	FA
Spent-fuel storage facilities	FB
Spent-fuel pool cooling and cleanup systems and controls	FC
Fuel handling systems	FD
Auxiliary water systems	WX
Station service water systems and controls	WA
Cooling systems for reactor auxiliaries and controls	WB
Demineralized water makeup systems and controls	WC
Potable and sanitary water systems and controls	WD
Ultimate heat sink facilities	WE
Condensate storage facilities	WF
Other auxiliary water systems and controls	WG
Auxiliary process systems	PX
Compressed air systems and controls	PA
Process sampling systems	PB
Chemical, volume control, and liquid poison systems and controls	PC
Failed-fuel detection systems	PD
Other auxiliary process systems and controls	PE
Other auxiliary systems	AX
Air conditioning, heating, cooling, and ventilation systems and controls	AA
Fire protection systems and controls	AB
Communication systems	AC
Other auxiliary systems and controls	AD
Steam and power conversion systems	HX
Turbine-generators and controls	HA
Main steam supply systems and controls (other than CC)	HB
Main condenser systems and controls	HC
Turbine gland sealing systems and controls	HD

Table A.17 (continued)

System	Code
Turbine bypass systems and controls	HE
Circulating water systems and controls	HF
Condensate cleanup systems and controls	HG
Condensate and feedwater systems and controls (other than CH)	HH
Steam generator blowdown systems and controls	HI
Other features of steam and power conversion systems (not included elsewhere)	HJ
Radioactive waste management systems	MX
Liquid radioactive waste management systems	MA
Gaseous radioactive waste management systems	MB
Process and effluent radiological monitoring systems	MC
Solid radioactive waste management systems	MD
Radiation protection systems	BX
Area monitoring systems	BA
Airborne radioactivity monitoring systems	BB
Other	XX
Not applicable	ZZ

Table A.18. Components involved with forced shutdowns and power reductions

Component type	Including
Accumulators	Scram accumulators Safety injection tanks
Air dryers	
Annunciator modules	Alarms Bells Buzzers Claxons Horns Gongs Sirens
Batteries and chargers	Chargers Dry cells Wet cells Storage cells
Blowers	Compressors Gas circulators Fans Ventilators
Circuit closers/interruptors	Circuit breakers Contactors Controllers Starters Switches (other than sensors) Switchgear
Control rods	Poison curtains
Control rod drive mechanisms	
Demineralizers	Ion exchangers
Electrical conductors	Bus Cable Wire
Engines, internal combustion	Butane engines Diesel engines Gasoline engines Natural gas engines Propane engines
Filters	Strainers Screens
Fuel elements	

Table A.18 (continued)

Component type	Including
Generators	Inverters
Heaters, electric	
Heat exchangers	Condensers Coolers Evaporators Regenerative heat exchangers Steam generators Fan coil units
Instrumentation and controls	
Mechanical function units	Mechanical controllers Governors Gear boxes Varidrives Couplings
Motors	Electric motors Hydraulic motors Pneumatic (air) motors Servo motors
Penetrations, primary containment air locks	
Pipes, fittings	
Pumps	
Recombiners	
Relays	
Shock suppressors and supports	
Transformers	
Turbines	Steam turbines Gas turbines Hydro turbines
Valves	Valves Dampers
Valve operators	
Vessels, pressure	Containment vessels Dry wells Pressure suppression Pressurizers Reactor vessels

Table A.19. Initiating event descriptions for DBEs as listed in Chap. 15, *Standard Review Plan* (Revision 3)

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1. Increase in heat removal by the secondary system
   - 1.1 Feedwater system malfunction that results in a decrease in feedwater temperature
   - 1.2 Feedwater system malfunction that results in an increase in feedwater flow
   - 1.3 Steam pressure regulator malfunction or failure that results in increasing steam flow
   - 1.4 Inadvertent opening of a steam generator relief or safety valve
   - 1.5 Spectrum of steam system piping failures inside and outside of containment in a pressurized-water reactor (PWR)
   - 1.6 Startup of idle recirculation pump[a]
   - 1.7 Inadvertent opening of bypass resulting in increase in steam flow[a]
2. Decrease in heat removal by the secondary system
   - 2.1 Steam pressure regulator malfunction or failure that results in decreasing steam flow
   - 2.2 Loss of external electric load
   - 2.3 Turbine trip (stop valve closure)
   - 2.4 Inadvertent closure of main steam isolation valves
   - 2.5 Loss of condenser vacuum
   - 2.6 Coincident loss of onsite and external (offsite) ac power to the station
   - 2.7 Loss of normal feedwater flow
   - 2.8 Feedwater piping break
   - 2.9 Feedwater system malfunctions that result in an increase in feedwater temperature[a]
3. Decrease in reactor coolant system flow rate
   - 3.1 Single and multiple reactor coolant pump trips
   - 3.2 Boiling-water reactor (BWR) recirculation loop controller malfunction that results in decreasing flow rate
   - 3.3 Reactor coolant pump shaft seizure
   - 3.4 Reactor coolant pump shaft break
4. Reactivity and power distribution anomalies
   - 4.1 Uncontrolled control rod assembly withdrawal from a subcritical or low-power start-up condition (assuming the most unfavorable reactivity conditions of the core and reactor coolant system), including control rod or temporary control device removal error during refueling

A.19 (continued)

4.2 Uncontrolled control rod assembly withdrawal at the particular power level (assuming the most unfavorable reactivity conditions of the core and reactor coolant system) that yields the most severe results (low power to full power)
4.3 Control rod maloperation (system malfunction or operator error), including maloperation of part length control rods
4.4 Start-up of an inactive reactor coolant loop or recirculating loop at an incorrect temperature
4.5 A malfunction or failure of the flow controller in a BWR loop that results in an increased reactor coolant flow rate
4.6 Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant of a PWR
4.7 Inadvertent loading and operation of a fuel assembly in an improper position
4.8 Spectrum of rod ejection accidents in a PWR
4.9 Spectrum of rod drop accidents in a BWR

5. Increase in reactor coolant inventory

5.1 Inadvertent operation of emergency core cooling system during power operation.
5.2 Chemical and volume control system malfunction (or operator error) that increases reactor coolant inventory
5.3 A number of BWR transients, including items 1.2 and 2.1—2.6

6. Decrease in reactor coolant inventory

6.1 Inadvertent opening of a pressurizer safety valve in either a PWR or a BWR
6.2 Break in instrument line or other lines from reactor coolant pressure boundary that penetrate containment
6.3 Steam generator tube failure
6.4 Spectrum of BWR steam system piping failures outside of containment
6.5 Loss-of-coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary, including steam line breaks inside of containment in a BWR
6.6 A number of BWR transients, including items 1.3, 2.7, and 2.8

A.19 (continued)

7. Radioactive release from a subsystem or component
   - 7.1 Radioactive gas waste system leak or failure
   - 7.2 Radioactive liquid waste system leak or failure
   - 7.3 Postulated radioactive releases due to liquid tank failures
   - 7.4 Design basis fuel handling accidents in the containment and spent fuel storage buildings
   - 7.5 Spent fuel cask drop accidents
8. Anticipated transients without scram
   - 8.1 Inadvertent control rod withdrawal
   - 8.2 Loss of feedwater
   - 8.3 Loss of ac power
   - 8.4 Loss of electrical load
   - 8.5 Loss of condenser vacuum
   - 8.6 Turbine trip
   - 8.7 Closure of main steam line isolation valves

[a]These initiating events were added for BWRs to be more specific than DBE events 5.3 and 6.6.

A.20. NOAC event categories for non-DBE shutdowns

- N 1.0 Equipment failure
  - N 1.1 Failure on demand under operating conditions
    - N 1.1.1 Design error
    - N 1.1.2 Fabrication error
    - N 1.1.3 Installation error
    - N 1.1.4 End of design life/inherent failure/random failure
  - N 1.2 Failure on demand under test conditions
    - N 1.2.1 Design error
    - N 1.2.2 Fabrication error
    - N 1.2.3 Installation error
    - N 1.2.4 End of design life/inherent failure/random failure
- N 2.0 Instrumentation and control anomalies
  - N 2.1 Hardware failure
  - N 2.2 Power supply problem
  - N 2.3 Setpoint drift
  - N 2.4 Spurious signal
  - N 2.5 Design inadequacy (system required to function outside design specifications
- N 3.0 Non-DBE reductions in coolant inventory (leaks)
  - N 3.1 In primary system
  - N 3.2 In secondary system and auxiliaries
- N 4.0 Fuel/cladding failure (densification, swelling, failed fuel elements as indicated by elevated coolant activity)
- N 5.0 Maintenance error
  - N 5.1 Failure to repair component/equipment/system
  - N 5.2 Calibration error
- N 6.0 Operator error
  - N 6.1 Incorrect action (based on correct understanding on the part of the operator and proper procedures, the operator turned the wrong switch or valve — incorrect action)
  - N 6.2 Action on misunderstanding (based on proper procedures and improper understanding or misinterpretation on the operator's part of what was to be done — incorrect action)
  - N 6.3 Inadvertent action (purpose and action not related, for example, bumping against a switch or instrument cabinet)
- N 7.0 Procedural/administrative error (incorrect operating or testing procedures, incorrect analysis of an event — failure to consider certain conditions in analysis)

Table A.20 (continued)

N 8.0	Regulatory restriction
	N 8.1 Notice of generic event
	N 8.2 Notice of violation
	N 8.3 Backfit/reanalysis
N 9.0	External events
	N 9.1 Human induced (sabotage, plane crashes into transformer)
	N 9.2 Environment induced (tornado, severe weather, floods, earthquake)
N 10.0	Environmental operating constraint as set forth in Technical Specifications

Appendix B

REVIEW OF REPORTABLE EVENTS

This appendix presents the data an reportable events in tabular forma nd lists the information sources. It describes how the information was encoded and how significance screening criteria were applied.

B.1 Scope

This study reviewed operating events reported in LERs and LER predessors [e.g., AORs, unusual events reports, reportable occurrences (ROs)]. Data on these types of events was retrieved from the NOAC data files. Any documents that contained LER-type information (such as equipment failures or abnormal events) were included to obtain a total picture of the unit's equipment failure history. Primarily, this involved various types of operating reports and general correspondence.

The reportable event table (Table B.1) contains the following information for each reportable event reviewed:

1. LER number or other means of identification of report type,
2. NSIC accession number (a unique identification number assigned to each document entered into the computer file),
3. date of the event,
4. date of the report or letter transmitting the event description,
5. status of the plant at the time of the occurrence (Table B.16),
6. system involved with the reportable event (Table B.17),
7. type of equipment involved with the reportable event (Table B.18),
8. type of instrument involved with the reportable event (Table B.18),

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9. status of the component (equipment) at the time of the occurrence (Table B.16),
10. abnormal condition associated with the reportable event (e.g., corrosion, vibration, leak) (Table B.19),
11. cause of the reportable event (Table B.16),
12. significance of the reportable event, and
13. comments and/or details on the event.

B.2 Data Sources

The NOAC files of LERs (including the *Sequence Coding and Search System*) were the primary source of information for the review of reportable events. When additional information on the event was needed, the original LER (or equivalent) was consulted by examining (1) those full-sized copies on file at NOAC (for the years 1976—1984); (2) the microfiche file of docket material at NOAC; or (3) the appropriate operating report (semiannual, annual, or monthly).

Printouts obtained from the computer files also covered other types of "docket material" besides reportable events where the licensee may have been in correspondence with NRC [or the Atomic Energy Commission (AEC)] concerning a particular event. Licensees are often requested to submit additional information or perform further analysis. Before the LERs came into existence in the mid-1970s, it was not unusual for licensees to submit on their own or at the request of NRC or AEC more than one letter transmitting information on a particular event. Thus, these printouts provided additional sources of information on reportable events.

Several special publications were reviewed to provide details on events of significance. After further analyses and examination of the following publications, details, evaluations, or assessments could be found other than those provided in the appropriate NRC-requested transmission.

1. *Reports to Congress on Abnormal Occurrences*, NUREG-0090 series;
2. "Power Reactor Event Series" (formerly Current Event Series) published bimonthly by NRC;
3. "Operating Experiences," a section of each issue of the *Nuclear Safety* journal; and
4. the publications of NRC's Office of Inspection and Enforcement (IE), such as operating experience bulletins, IE bulletins, IE circulars, and IE information notices.

B.3 Significance Screening

Two sets of criteria were used in determining the significance of reportable events. The first set of criteria (Table B.20) address those events whose results include challenges to the safety protection features of the plant; these events are termed "safety significant". The second set of criteria (Table B.21) address those events that have the potential to challenge the safety protection features of the plant. These events, which might require additional information or evaluation to determine their full implication, were termed "conditionally significant".

The reportable events were all reviewed, applying the two sets of criteria for significance rather liberally. A number of significant

events and conditionally significant events were noted. The events initially identified as significant or conditionally significant were analyzed and evaluated further based on (1) engineering judgment; (2) the systems, equipment, or components involved; or (3) whether the safety of the plant was compromised. The conditionally significant events were subsequently "upgraded" to significant or "downgraded" to nonsignificant as necessary. The final evaluation for significance considered whether a DBE was initiated or a safety function was compromised such that the system as designed could not mitigate the progression of events. Thus, the number of events finally categorized as significant was reduced considerably by these steps in the review process.

The reportable events not identified as significant or conditionally significant were categorized as not significant (with an 'N' in the significance column of the coding sheets in the tables). These events and the events rejected during the additional review step were further reviewed by compiling a tabular summary of the systems to detect trends and recurring problems (Table 1.4 provides a listing of the systems).

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1970

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
RS-70-4	58005	11/08/1970	11/17/1970	B	EE	CD.NN	M	C	BK	B	N	GAS TURBINE GENERATOR FAILED DURING TEST DUE TO LOW PRESSURE IN LUBE OIL PUMP
AO-70-5	60225	12/04/1970	12/14/1970	B	CE	H.OO	M	C	BJ	B	N	ISOLATION CONDENSER ISOLATED DURING TESTS DUE TO DESIGN ERROR
RS-70-6	58011 57474	11/19/1970	11/25/1970	B	CD	OO		C	OJ	G,H	N	STEAM ISOLATION VALVE FAILURE - IMPROPER MAINTENANCE PROCEDURE AND OPERATOR ERROR COMBINED (REACTOR SHUTDOWN)
RS-70-6	59927	12/09/1970	12/19/1970	B	CC	Z		B	AH	B	C4	EXCESSIVE PIPE MOVEMENT DURING TRANSIENT DUE TO DESIGN ERROR
RS-70-06	59927	12/09/1970	12/19/1970	B	HE	OO		B	AC	D	N	STEAM BYPASS VALVE CAUSED MALFUNCTION OF PRESSURE CONTROL

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1970

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
RS-70-06	59927	12/09/1970	12/19/1970	B	CD	DD		B	AL	E	CE	MSIV CLOSED DUE TO MISSING SPRING IN SOLENOID VALVE
AO-70-7	59595	12/23/1970	1/02/1971	B	RB	J	K	C	AM	B	N	APPROVED WIRING CHANGE NOT TESTED RESULTING IN LOSS OF FULL ROD CONTROL
AO-70-8	59595	12/23/1970	1/02/1971	B	HC	H,Z		C	AO	E	N	WELDING ERROR CAUSED DECREASE IN CONDENSER VACUUM (REACTOR SHUTDOWN)
AO-70-9	59594	12/29/1970	1/06/1971	B	HE	CG		B	AI	D	N	MAIN STEAMLINE BYPASS VALVE FAILURE CAUSED BY BROKEN LINKAGE ON VALVE OPERATOR (REACTOR SHUTDOWN)
AO-70-10	59593	12/31/1970	1/08/1971	B	EE	T,DD		C	HK	E	N	SLOW TURBINE START DUE TO REINSTALLATION ERROR IN LUBE OIL PUMP DISC. LINK

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1971

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
AC-71-1	62296 63204	1/19/1971	2/01/1971	B	SF-D, SF-B	OO	T	C	EI	E	S7	FOUR SIMULTANEOUS VALVE FAILURES RENDER ECCS INOPERABLE (REACTOR SHUTDOWN) TORQUE SWITCHES SET TOO LOW
AC-71-2	61450	1/19/1971	1/25/1971	B	SF-B	OO		C	AK	D	N	GALLED THREADS ON VALVE CAUSE TURBINE CONTROL VALVE TO CLOSE
RS-71-17	62298	2/12/1971	3/09/1971	B	SF,IA	OO	I	A	OJ	H	N	ERROR DURING MAINTENANCE ON LEVEL INDICATOR
AO-71-5	62279	2/21/1971	3/03/1971	B	EE	S,T,NN	T,P	B	BD	D	N	GAS TURBINE GENERATOR FAILS TO START
AC-71-6	62300	3/25/1971	4/02/1971	B	CC	OO,NN		C	BI	B	N	LOSS OF PRESSURE TO VALVE DURING TEST REACTOR SHUTDOWN
AC-71-7	63140	3/31/1971	4/08/1971	B	SF-B	X,OO		C	ED	D	N	MOTOR FAILED ON LPCI VALVE DUE TO SHORT IN WINDINGS

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1971

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
AC-71-8	63139	4/22/1971	4/27/1971	B	EE	NN,OO	T	A	OK	A,G	N	GAS TURBINE GENERATOR INOPERABLE DUE TO A SWITCH BEING LEFT IN THE WRONG POSITION
AC-71-9 AO-71-10	63125	5/01/1971 5/02/1971	5/12/1971	B	CG,WA EB	D,U X,OO		B	AU,AT	D	CB	THREE SIMULTANEOUS FAILURES IN MOTOR CONTROL CENTER DUE TO MOISTURE
AC-71-11	64435	5/25/1971	6/04/1971	B	CH	OO	K	B	HD	B	N	FEEDWATER CONTROL VALVE PROBLEMS
AC-71-12	64436	5/27/1971	6/04/1971	B	CJ,EE	OO,NN	T	B	EE,ED	E	CB	FAILURE OF MECHANICAL AND ELECTRICAL VALVE REGULATOR & SHORT CIRCUIT IN SPEED SWITCH ON GAS TURBINE
	64810	6/24/1971	7/12/1971	B	EA		P	B	EG,OF	F	N	OFFSITE RELAY TRIPPED REACTOR INADVERTENTLY DURING THUNDERSTORMS (REACTOR SHUTDOWN)

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1971

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
AO-71-13	65552	8/13/1971	8/19/1971	B	SA,IA	OO	T	C	AL	E,G	N	DRYWELL HIGH PRESSURE SWITCH HAD LOOSE HOLD DOWN SCREWS
AO-71-14	66469	8/30/1971	9/09/1971	B	CD	OO		B	AG	D	N	PLUNGER IN AIR SLIDE VALVE RESISTED MOVEMENT
	67211	9/22/1971	9/22/1971	B	WA,SF	CD		C	OA	D	N	INSUFFICIENT HEAD SERVICE WATER PUMPS
AO-71-15	67459 87460	9/17/1971	9/28/1971	B	SF-B	X		C	AH	B	N	TWO MOTORS BURNED OUT FOR SAME TEST
AO-71-16	67459	9/18/1971	9/28/1971	B	SF-B	CC		C	AC	D	N	LPCI OUTBOARD CONTAINMENT SPRAY VALVE FAILS DUE TO PIES IN WRONG POSITION
AO-71-17	68264	9/21/1971	9/30/1971	B	SF,WA	CD		C	OA	B	N	SERVICE WATER PUMPS PROVIDE LTA HEAD
AO-71-17	67461	9/27/1971	9/30/1971	B	SF,WA	CD		C	OA	D	N	EMERGENCY SERVICE WATER PUMPS PROVIDE LTA HEAD

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1971

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
AO-71-18	67993	9/23/1971	10/04/1971	B	HE	NN,00		B	AB	B	N	TURBINE BYPASS VALVE FAILS-BRITTLE FRACTURE OF STUDS ON LINKAGE. REACTOR SCRAMMED
AO-71-19	67992	9/29/1971	10/08/1971	B	HE	NN,00	M	B		G	N	FAILURE OF STEAM TURBINE BYPASS VALVE. REACTOR SHUTDOWN
AO-71-20	66996	10/10/1971	10/22/1971	B	CC,HE	NN,00	M	B	AP,BL	B	57	FAILURE OF STEAM TURBINE BYPASS VALVE REACTOR BLOWDOWN
AO-71-21	68303	10/12/1971	10/20/1971	D	IE	DD	N	B	AA,BL	E	N	TWO STACK GAS SAMPLE PUMPS FAIL OFF. THIRD TAGGED OUT
AO-71-22	68304	10/31/1971	10/20/1971	D	SF	DD		C	AP	B	N	REGID AIR-LINES SHOULD HAVE BEEN FLEXIBLE
AO-71-23	68305	10/16/1971	10/20/1971	D	CF	DD		C	AP	G	N	VALVE ACTUATOR FAILED DUE TO LOOSE PARTS FROM VIBRATION

TABLE A.22 DATA TABLES FOR REPCRTABLE EVENTS FOR MILLSTONE 1-1971

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
AO-71-24	68788	11/02/1971	11/11/1971	B	EE	KK	T	C	AL	E,G	N	GAS TURBINE GENERATOR FAILS TO START DUE TO LOCSE CONNECTION
	68307		11/16/1971	B	RB	DD		C	AT	E	N	PUMP IN STANDBY LIQUID CONTROL SYSTEM LEAKED
AC-71-25	68308	11/30/1971	12/09/1971	B	EE	KK,T	U	C	HB,OJ	H	CB	COLD LUBE CIL CALSED GAS TRLBINE TRIP CFF. CPERATOR DID NUT TURN ON CIL HEATERS
AC-71-26	69199	12/10/1971	12/20/1971	B	CE	H		C	AN	B	N	STRIPPED THREADS ON YOKE SLEEVE CALSED ISOL. COND. VALVE TO FAIL CLCSED
AO-71-27	69318		12/27/71	B	IB		M,P	B	AL	G	N	ELECTRIC PRESSURE REGULATOR FAILS (REACTOR SHUTDOWN)

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1972

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
AO-72-1		2/03/1972	2/15/1972	B	CC,IB		M	C	OA	B	N	MAIN STEAM LINE DIFFERENTIAL PRESSURE SENSORS DO NOT MEET MANUFACTURES SPECS
AO-72-2	39368	2/04/1972	2/15/1972	B	IB	QQ,P	M	B	AQ	G	N	FILTER CLOGGED IN ELECTRIC PRESSURE REGULATOR PILOT VALVE (REACTOR SHUTDOWN)
AO-72-3	39314	2/04/1972	2/15/1972	B	EE	NN		B	EF	E	N	GAS TURBINE VIBRATION MONITOR FAILS
AO-72-4	39195	2/08/1972	2/29/1972	B	CC	QQ	M	C	EI	G	N	SET POINT DRIFT IN PRESSURE RELIEF VALVE CONTROLLER
AO-72-5	39317	2/11/1972	2/15/1972	B	SF-B	QQ	M	B	EG	D	N	FAILURE OF LPCI LOW PRESSURE SWITCH
AO-72-6	39177	2/18/1972	2/28/1972	D	CC	H	M	C	EH	D	N	SET POINT DRIFT CAUSES PRESSURE SENSORS IN CONDENSER TO TRIP TOO HIGH

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1972

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
AO-72-7	39177	2/19/1972	2/28/1972	D	CC		M	C	EH	D	N	SET POINT DRIFT IN MAIN STEAM LINE PRESSURE SENSORS
AO-72-8	39218	2/23/1972	3/02/1972	B	CC		M	C	AI	B	N	DELTA P SENSOR FAILED, REACTOR SHUTDOWN
AO-72-9	55348	3/03/1972	3/13/1972	B	SF-B	OO	T	C		D	N	CLEANUP AUXILIARY PUMP BYPASS VALVE INOPERABLE
AO-72-10	55347	3/06/1972	3/13/1972	B	CE	QQ,DD	T	B	EI	D	N	LOSS OF MONITORING CAPABILITY OF CONTAINMENT ISOLATION VALVES
AO-72-11	55353	3/09/1972	3/20/1972	B	EE	NN	T	B	EG,ED	G	N	GAS TURBINE GENERATOR FAILS TO START-FAULTY SPEED SWITCH
AO-72-12	70045	3/11/1972	4/20/1972	B	IA		T	B	EH	D	N	SET POINT DRIFT OF LEVEL SENSORS
AO-72-13	700[illegible]	4/12/1972	4/20/1972	B	IA		T	B	EH	D	N	

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1972

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
AO-72-14	70045	4/18/1972	4/20/1972	B	CC		M,T	C	EH	D	N	SET POINT DRIFT OF PRESSURE SENSORS IN MAIN STEAM LINE
AO-72-15	71716	6/08/1972	6/08/1972	B	CE	QQ,H,OO		C		G	N	IMPROPER ASSEMBLY OF VALVE
AO-72-16	72424	6/24/1972	7/05/1972	B	IA		T	C	EH	D	N	SET POINT DRIFT OF HIGH FLOW SWITCH
AO-72-17	72561	7/13/1972	7/14/1972	B	IA		M	C	EI	D	C8	THREE PRESSURE SENSORS TRIP ABOVE TOLERANCE LIMIT
	72562	6/12/1972	7/13/1972	B	ZZ	KK		B	AE	D	N	PIPE HANGERS SHIFTED
AO-72-18	72774	7/19/1972	7/20/1972	B	SF		P	C	EI	G	C8	FOUR OF FIVE TIME DELAY RELAYS DO NOT MEET SPECS
	73637		8/09/1972	B	EE	N	M,T	C	EH	D	N	SET POINT DRIFT CAUSES DG TO FAIL TO START

TABLE A.2? DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1972

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
AO-72-19	73638	8/10/1972	8/11/1972	B	RA		I	C	EH	D	N	SET POINT DRIFT OF REACTOR VESSEL LOW LEVEL SWITCH
AO-72-20	75492	8/15/1972	8/23/1972	B	CE	H	T	C	EH	D	N	ISOLATION CONDENSER FLOW SWITCHES SET POINT DRIFT
AO-72-21	75398	8/31/1972	8/31/1972	B	CB	DD,FF		B	AW	D	N	EXCESSIVE COOLANT RATE LEAK DUE TO BAD PUMP SEALS
AO-72-22	75078 75399 75755 77519	9/01/1972	9/11/1972	B	HC	H		B	AR,AU,OF	B	C4	MAIN CONDENSER TUBES LEAK SALT WATER INTO COOLANT SEALS
AO-72-23	75400	9/11/1972	9/13/1972	C	IA		T,I	C	EI	G	N	LOW-LOW WATER LEVEL DETECTOR OUT OF CALIBRATION
AO-72-24	75345	9/25/1972	9/06/1972	B	SC	OO		C	AO,AV	E	N	CRACK IN HEADER OF ATMOSPHERIC CONTROL
AO-72-25	75795	10/12/1972	10/25/1972	C	CE	H	T,M	C	EH	D	N	ISOLATION CONDENSER PRESSURE SWITCH SET POINT DRIFT ON 3 OUT OF 4 SENSORS
AO-72-26	75910		10/21/1972	C	CH	Z		C	AV	D	N	TWO FEEDWATER SPARGER LEAKS
AO-72-27	75902		10/28/1972	C	CA	OO		C	OA	G	N	PENETRATION LEAKS AND ISOLATION VALVE LEAKS DURING TESTING
	76011	8/25/1972	9/23/1972	B	EA	LL		B		D	C5	PLANE CRASHES INTO POWER TRANSFORMER
	76071		10/13/1972	C	CE	H	T	C	EH	D	N	SET POINT DRIFT OF 3 ISOLATION CONDENSER SWITCHES

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1973

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
	77956		1/26/1973	B	RB	I,A		B	AT,AR	B,G	S7,S2	ACCUMULATORS FOR CONTROL ROD DRIVE FOUND TO BE BAD, ALL REPLACED
AO-73-1	80136	3/07/1973	3/08/1973	B	RB	I		C	BI	D	N	EXCESSIVE SCRAM TIME
AO-73-2	79560	3/14/1973	3/21/1973	B	RB,EE	S,T,H,NN	P	B	EI,ED	G	N	SCRAM ON BLOWN CONTROL ROD DRIVE FUSE; GAS-TURBINE FAILS TO START
AO-73-3	80118	3/21/1973	3/27/1973	B	CH	DD,D		B	AB	G	N	INBOARD BEARING FAILURE
AO-73-4	80274	4/05/1973	4/12/1973	B	CE	H,QQ	T	B	BC,HA,OK	G	CB	ISOLATION CONDENSER ISOLATES ON HIGH FLOW DUE TO INCORRECT MAINTENANCE
AO-73-5	80275	4/05/1973	4/13/1973	B	EE	T,NN		C	OI,OA	H	N	GAS TURBINE GENERATOR FAILS TO START AFTER ISOLATION CONDENSER FAILURE-OPERA TOR MADE WRONG ADJUSTMENT DURING STARTUP

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1973

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
AO-73-6	80728	4/18/1973	5/18/1973	D	CF	QQ		B	AD,AI,AM	B	N	VALVE OPERATOR BROKE ON SHUTDOWN COOLING SYSTEM ISOLATION VALVE-IMPROPER SIZED MOTOR
	80710		5/21/1973	D	CH	Z		C	AV	D	N	REPORT ON SPARGERS CRACK IN FEEDWATER SYSTEM
	84795	7/18/1973	7/18/1973	C	ZZ	GG		C	AT	G	N	SHOCK SUPPORESSORS LEAKED HYDRAULIC FLUID, DETERIORATION OF SEALS
	82972		7/13/1973	C	CB	OO		B	BA	D	N	RECIRCULATION VENT VALVE LINE LEAKS
AO-73-8	82957	8/02/1973	8/02/1973	C	SF-B		M,T	C	FB	D	N	SET POINT DRIFT ON DP SENSORS OF LPCI
	82966		8/08/1973	C	RB	J		C	CA	D	N	CONTROL ROD UNCOUPLED
	82966		8/06/1973	C	CH	Z	M	C	AU	D	N	LEAK IN PRESSURE SENSING LINE

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1973

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
AO-73-9	83225	8/13/1973	8/20/1973	C	IA		I	B	AM	E	N	LEVEL SENSORS READ WRONG DUE TO MISMATCH IN VENTILATION AROUND SENSOR
	84017		8/30/1973	C	SF-D	DD,QQ		C	BA,EC	D	N	FAILURE OF CORE SPRAY VALVE TO OPEN
AO-73-10	84491	9/21/1973	10/01/1973	B	IA	OO	I	B	AT	D	N	MISMATCH IN LEVEL INDICATORS DUE TO VALVE LEAK
AO-73-11	84497	9/22/1973	10/01/1973	B	CH	D		A	BL	G	N	CONDENSATE BOOSTER PUMP BEARING OVER HEATS-REWORK BEARINGS
	84551		11/04/1973	B	SH-B		T,I	C		D	N	LEVEL SWITCH FAILED TO TRIP
AO-73-12	88079	12/30/1973	1/07/1974	B	CE	Q		B	AU	D	N	STEAM LEAK IN ISOLATION CONDENSER FLANGE (REACTOR SHUTDOWN)

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
	53695	1/06/1974	2/01/1974	B	CA	OO		B	AB	D	N	PRESSURE RELIEF VALVE LIFTS DUE TO WORN PRELOAD ASSEMBLY
	89258		3/08/1974	B	ZZ	GG,FF		B	AT	B	N	SNUBBERS LEAK
AO-74-1	89340	3/06/1974	3/15/1974	B	CB	BB,OO,FF		B	AU	D	N	DRYWELL LEAK DUE TO SEAL PACKING FAILURE (REACTOR SHUTDOWN)
	89663		4/01/1974	B	CB	DD,QQ,OO	P	B	BB,OK	G	N	PUMP SUCTION VALVE ON RECIRC. PUMP FAILS TO CLOSE
AO-74-2	92184	5/17/1974	5/28/1974	B	CB	DD,D		B	AB	B	N	INBOARD BEARING FWCI CONDENSATE PUMP - SECOND FAILURE
AO-74-4	95140	8/28/1974	9/05/1974	B	SB-D	FF		C	AW	D	N	TEAR IN SEAL AT REACTOR BUILDING TURBINE BUILDING INTERFACE
AO-74-6	95426	9/18/1974	9/25/1974	C	CB	Z		C	AO	B	N	WELD CRACKS IN RECIRCULATION LOOP DISCHARGE BYPASS LINE
AO-74-8	97077	10/31/1974	11/06/1974	C	WA, SF-B	OO		C	BA	B	N	EMERGENCY SERVICE WATER VALVE FAILS DUE TO MANUAL OVERTORQUEING
AO-74-9	97076	11/04/1974	11/12/1974	B	CD	QQ,P		C	HC	G	N	FOREIGN MATERIAL IN AIR SLIDE VALVE DISABLES VALVE OPERATOR (REACTOR [illegible]

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1974

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
AO-74-10	97506	11/15/1974	11/18/1974	B	CD	QQ,P		C	HC	G	N	AIR SLIDE VALVE CLOGS SECOND TIME IN ONE MONTH (REACTOR SHUTDOWN)
AO-74-11	98161	12/13/1974	12/19/1974	B	CG	QQ		A	AB,BC	D	N	MISALIGNMENT OF GEAR TRAIN ON VALVE MOTOR OPERATOR
AO-74-12	98582	12/20/1974	12/30/1974	B	CE	H	I	C	EH	D	N	SET POINT DRIFT IN LEVEL SENSOR IN ISOLATION CONDENSER-SET POINT LOCK INSTALLED
AO-74-13	98581	12/20/1974	12/30/1974	B	BC	H	M	C	EH	D	N	SET POINT DRIFT ON CONDENSER VACUUM SWITCHES
AO-74-14	98705	12/23/1974	1/02/1975	B	SF-B	MM	I	B	HC,BT	G	N	LEVEL SENSOR TUBING BLOCKED - BLOWOUT TUBE INSTALLED
AO-74-15	98706	12/28/1974	1/02/75	B	ZZ	GG		C	AT	E	N	2 SNUBBERS HAVE NO FLUID - IMPROPER INSTALLATION

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1975

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	I[illegible]TRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
	99684		1/24/1974	B	WC	M		B	OF	D	C8	HYDROGEN EXPLOSION IN ACID DAY TANK DUE TO MOISTURE
AO-75-2	93569	1/25/1975	1/30/1975	B	CC	Z	E	B	AD	D	N	FLOW INDICATOR READING DECREASES DUE TO WELD CRACK IN SENSING TUBE
AO-75-4	93515	1/29/1975	1/31/1975	B	EE	NN,T	P	B	EF	D	N	SPURIOUS OPERATION OF RELAY CAUSES TURBINE STARTUP SEQUENCE TO BEGIN
AO-75-5	101408	3/27/1975	4/03/1975	B	WF,MA	V		B	AW,OD	E	N	WIRING ERROR CAUSES FLOW OF CONTAMINATE INTO BOILER SYSTEM
AO-75-6	101701	3/30/1975	4/08/1975	B	MA			B	OD,OK	A	C3	INADVERTENT DISCHARGE OF RADIOACTIVE LIQUID DUE TO PROCEDURE ERROR
AO-75-7	102297	4/18/1975	4/25/1975	B	EE	T,C	P	B	EB	D	N	OSCILLATOR BOARD TRIPS INVERTER OPERATION
AO-75-8	103148	5/20/1975	5/29/1975	B	EE	NN,T, OO	T,B	C	EI,BL	G	N	INCORRECT VALVE POSITION PLUS TEMPERATURE SENSOR ERROR CAUSES GAS TURBINE TRIP
AO-75-9	104066	5/20/1975	7/11/1975	B	CC	OO		B	BP,BB	D	N	SAFETY RELIEF VALVE FAILS TO RESET - NO CAUSE FOUND (REACTOR SHUTDOWN)
AO-75-10	103074	5/21/1974	5/28/1975	B	?	GG,FF		A	BT	B,E	N	NO FLUID IN TWO SNUBBERS DUE TO IMPROPER ASSEMBLY

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1975

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
AO-75-24	108888	12/19/1975	12/23/1975	B	CD		P,T	C	BC	G	N	VALVE POSITION SWITCH OUT OF POSITION CAUSES MSIV CLOSE RELAY TO FAIL
AO-75-25	109191	12/17/1975	12/23/1975	B	IB		P	C	EH	D	N	SET POINT DRIFT IN RELAY
AO-75-26	109454	12/30/1975	1/07/1975	B	WA	P		C	AQ	G,A	N	GRIT IN STRAINER CAUSES EMERGENCY SERVICE WATER PUMP TRIP
AO-75-27	109459	12/23/1975	1/07/1975	B	SF-B		H,T	C	EH	D	N	SET POINT DRIFT IN LPCI PRESSURE SWITCH
AO-75-28	110932		12/28/1976	B	CG	JJ		B	AR	D	N	HOLE IN CLEANUP FILTER SLUDGE TANK

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1976

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
RO-76-2	111648	1/26/1976	2/25/1976	B	SA	F		C	AA	D	N	VACUUM BREAKER FAILS DUE TO TEFLON BUSHING OUT OF ROUND
RO-76-3	111658	2/10/1976	3/09/1976	B	HC	H	M,T	C	EH	D	N	SET POINT DRIFT IN CONDENSER LOW VACUUM SWITCH
RO-76-4	111647 120080	2/12/1976	3/05/1976 9/28/1976	B	CE	H		B	AR	A,B	S7	ISOLATION CONDENSER TUBE FAILURE DUE TO CORROSION CRACKING
RO-76-5	112163	2/12/1976	3/12/1976	D	CH	QQ	M	B	AI	D	N	FEEDWATER REGULATION VALVES LOCK DUE TO DIAPHRAGM FAILURE IN REGULATOR
RO-76-7	112308	2/19/1976	3/15/1976	B	CC	NN	M,T	C	EH	D	N	INSTRUMENT DRIFT CAUSES TURBINE RELAYS TO TRIP
RO-76-10	112309	3/08/1976	3/22/1976	B	EE	NN,T		C	BC	G	S7,S1	GAS TURBINE GENERATOR GOVERNOR OUT OF ADJUSTMENT (REACTOR SHUTDOWN)

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1976

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
RO-76-11	112162	3/09/1976	3/23/1976	B	ZZ	GG		C	BT,AU	G	N	SHOCK SUPPRESSOR LOW ON FLUID
RO-76-12	112310	3/15/1976	3/29/1976	B	EE	KN,T		C	BC	D	N	GAS TURBINE GENERATOR INOPERABLE DUE TO GOVERNOR FAILURE WHILE ISOLATION CONDENSER UNAVAILABLE
RO-76-16	113541	4/22/1976	5/04/1976	B	CD	CC		C	AG	D	C	PRIMARY CONTAINMENT ISOLATION VALVES FAILT O CLOSE DUE TO INTERNAL BINDING
RO-76-17	113540	4/23/1976	4/30/1976	B	RB	R		B	DG	B	C3	EXCESS STACK GAS RELEASE DUE TO FUEL CLAD PERFORATIONS
RO-76-23	114643	5/28/1976	6/11/1976	B	CD	H		B	BU	E	N	INCREASE IN CHLORIDE ION CONCENTRATION IN CONDENSER RETURN LEG
RO-76-27	115725	6/23/1976	7/09/1976	B	IA		M,T	B	EH	D	N	SET POINT DRIFT IN PRESSURE SWITCH

TABLE A.22 DATA TABLES FOR RE' TABLE EVENTS FOR MILLSTONE 1-1976

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
RO-76-28	116269	7/14/1976	7/30/1976	D	RC	QQ		C	CA	E	N	ISOLATION CONDENSER INBOARD ISOLATION VALVE INOPERABLE DUE TO PREVIOUS FIX FOR LEAK
RO-76-29	116780	8/10/1976	8/24/1976	B	EE	NN,T		C	EA,BF	B	S8,S7	GAS TURBINE GENERATOR TRIPS OUT ON INCORRECT AC FEED
RO-76-30	117678	8/31/1976	9/8/1976	B	EE	NN,T		C	BF	D	N	GAS TURBINE GENERATOR TRIPS ON OVERSPEED - SPEED SWITCH FAULTY
RO-76-31	117679	8/13/1976	9/9/1976	D	IA		L	C	CB	D	N	SET POINT DRIFT IN INTERMEDIATE RANGE MONITORS
RO-76-32	118795	9/25/1976	10/08/1976	B	SF-B	PP		B	AW	G	N	PACING LEAK IN LPCI TESTABLE CHECK VALVE
RO-76-33	120533	10/28/1976	11/23/1976	C	RB	R		C	AD	D	N	NEUTRON SOURCE ROD BROKEN

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1976

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
RO-76-34	120436	11/12/1976	11/24/1976	C	RB	I		C	OJ,OK	A,H	S8	INADVERTENT CRITICALITY DUE TO OPERATOR ERROR DURING SHUTDOWN MARGIN TEST
RO-76-35	120272	11/9/1976	12/3/1976	C	CC		B,T	C	EH	D	N	SET POINT DRIFT OF STEAM TUNNEL TEMPERATURE SWITCH
RO-76-36	120228	11/22/1976	11/27/1976	C	SF-D	Z		C	AO,AV	B	N	WELD JOINT LEAK IN HEAD SPRAY SYSTEM DUE TO STRESS CORROSION
RO-76-37	120670	12/07/1976	12/30/1976	B	IA		H,T	C	EH	D	N	SET POINT DRIFT IN DRYWELL PRESSURE SWITCHES
RO-76-38	120669	12/18/1976	12/30/1976	B	SH-B		H,T	C	EH	D	N	SET POINT DRIFT OF CONTAINMENT SPRAY PRESSURE SWITCHES

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1976

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
RO-76-42	121034	12/17/1976	1/14/1977	B	CE	QQ	T	C	BB,EI	G	N	ISOLATION CONDENSER STEAM SUPPLY
RO-76-43	121035	12/18/1976	1/14/1977	B	CE	QQ	T	C	BB,EI	G	N	ISOLATION CONDENSER CONDENSATE RETURN VALVE TORQUE SWITCH MISCALIBRATED
RO-76-44	121036	12/18/1976	1/14/1977	B	CG	QQ,X		C	AH,BA,DK	G	N	PROCEDURE ERROR CAUSES OVERLOAD OF MOTOR ON VALVE

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1977

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
RO-77-2	122206	1/08/1977	2/07/1977	B	BB	AA,S	N	B	EA	B	C4	BOTH STACK GAS MONITORS INOPERABLE DUE TO BLOWN FUSE IN POWER SUPPLY
RO-77-3	122174	1/12/1977	2/10/1977	B	IA		M,T	C	EH	D	N	SET POINT DRIFT IN DRYWELL PRESSURE SWITCH
RO-77-4	122136	1/25/1977	2/23/1977	B	SA	QQ,MM		B	AQ,BB	G	N	DRYWELL VENT BYPASS VALVE FAILS TO CLOSE DUE TO GRIT IN AIR OPERATOR LINE
RO-77-5	122137	1/26/1977	2/24/1977	B	CB	W		B	BU	G	N	HIGH COOLANT CONDUCTIVITY DUE TO IMPROPER RINSING OF RESIN IN DEMINERALIZER (REACTOR SHUTDOWN)
RO-77-6	122138	1/28/1977	2/28/1977	B	CE	H		A	OF	I	N	IMPENDING STORM CONDITIONS HALTS MAINTENANCE OF ISOLATION CONDENSER

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1977

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
RO-77-7	122429	2/01/1977	3/01/1977	B	EE	Y,T,N		C	OA,AT,AV	D	N	DIESEL GENERATOR INOPERABLE DUE TO FULE OIL LEAK - CRACKED NIPPLE
RO-77-8	122139	2/01/1977	2/28/1977	B	CE	H		A	BU	B	N	HIGH CHLORIDE ION CONCENTRATION IN ISOLATION CONDENSER
RO-77-9	123022	2/11/1977	3/11/1977	B	IA		I,T	C	EH	D	N	SET POINT DRIFT IN REACTOR WATER LEVEL SWITCH
RO-77-10	123023	2/14/1977	3/11/1977	B	IA	BB	M,T	C	EH	D	N	SET POINT DRIFT IN DRYWELL PRESSURE SWITCH
RO-77-11	129462	2/14/1977	3/14/1977	B	CC		M,T	C	EH	D	N	SET POINT DRIFT IN MAIN STEAM LINE PRESSURE SWITCH
RO-77-12	123024	2/15/1977	3/14/1977	B	HC	H	M,T	C	EH	D	N	SET POINT DRIFT IN CONDENSER VACUUM SWITCH

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1977

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
RO-77-13	124872	3/18/1977	3/30/1977	B	SF-B			C	BV	A	N	LOW CONCENTRATION IN STANDBY LIQUID CONTROL SYSTEM DUE TO LOW TEMP IN STORAGE TANK
RO-77-14	125214	5/0[illegible]/1977	[illegible]/17/1977	B	IA		M,T	C	EH	D	N	SET POINT DRIFT IN REACTOR LOW PRESSURE START PERMISSIVE SWITCH
RO-77-15	125177	5/14/1977	6/07/1977	B	SH-D	S		B	EH,OA	D	N	STANDBY GAS TREATMENT SYSTEM CIRCUIT HAS BLOWS FUSE
RO-77-16	125594	6/14/1977	6/27/1977	B	CB	Z		A	AW	D	N	LEAK IN RECIRCULATION LOOP DRAIN - CAUSE UNKNOWN
RO-77-17	130698	6/17/1977	6/17/1977	B	CC	OC		C	AY	D	N	SAFETY/RELIEF VALVE OPENS -NO CAUSE KNOWN

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1977

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
RO-77-18	126008	6/18/1977	7/01/1977	B	CC	CC,P		B	AT	D	N	SAFETY/RELIEF VALVE SEAT LEAKAGE DUE TO FILTER FAILURE
RO-77-19	126489	6/15/1977	7/15/1977	B	CE	H	M,T	C	EH	D	N	SET POINT DRIFT IN ISOLATION CONDENSER ACTUATION PRESS SWITCH
RO-77-20	143504	7/14/1977	7/26/1977	B	BB	DD		C	AC	D	N	COOLING FAN DETERIORATION CAUSES LOSS OF STACK SAMPLER
RO-77-21	143468	7/13/1977	8/05/1977	B	CH	CO		B	AC	D	N	DEGRADATION OF VALVE DISPHRAGM CAUSES LOSS OF FULL FWCI CAPABILITIES
LER 77-22	143507	7/21/1977	8/19/1977	B	CC	CC,G	A	B	ED	D	N	SAFETY/RELIEF VALVE BELLOWS ALARM RECEIVED DUE TO WIRING SHORT CIRCUIT
RO-77-23	143469	8/05/1977	8/31/1977	B	IC	F		B	AB	B	N	3 VACUUM BREAKERS FAIL TO CLOSE DUE TO FRICTION
RO-77-24	143470	8/06/1977	8/30/1977	B	IX	CO		B	AC	B	N	REACTOR SCRAM ON LOSS OF INSTRUMENT AIR CAUSED BY GASKET FAILURE (REACTOR SHUTDOWN)

TABLE A.22 DATA TABLES FOR REPO BLE EVENTS FOR MILLSTONE 1-1977

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER 77-25	143509	8/07/1977	8/30/1977	B	CC	CC	K,T	C	AC	D	N	PRESSURE SWITCH ON BELLOWS MONITOR FAILS
LER 77-27	143511	9/09/1977	10/07/1977	B	EE	NN		C	EG	D	N	SPURIOUS NOISE HALTS GAS TURBINE STARTUP TESTS
LER 77-28	143512	9/12/1977	10/11/1977	B	CE	H	M,T	B	EH	D	N	SET POINT DRIFT IN ISOLATION CONDENSER PRESSURE SWITCHES
RO-77-29	143561	9/27/1977	10/14/1977	B	EE	N,MM		B	AT	D	N	DIESEL GENERATOR INOPERABLE DUE TO FUEL OIL LEAK
RO-77-30	143471	10/12/1977	11/02/1977	B	SH-D	FF		C	AV	D	N	STANDBY GAS TREATMENT SYSTEM INOPERABLE DUE TO LEAK IN SEAL
LER 77-32	143555	11/01/1977	11/15/1977	B	BB	DD		B	AA	D	N	STACK SAMPLE PUMP TRIPS DUE TO NORMAL PUMP
LER 77-33	143543	11/18/1977	12/02/1977	B	RX,CC	CC	M	B	BP	D	C6	REACTOR COOLDOWN RATE EXCEEDED LIMIT WHEN A SAFETY/RELIEF VALVE LIFTED AT LOW PRESSURE
LER 77-34	143546	10/28/1977	11/23/1977	B	CC	DD	A	B	ED	E	N	BELLOWS LEAKAGE ALARM SOUNDS DUE TO WIRING SHORT CIRCUIT

NUMBER	NRC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
RC-77-35	143472	10/31/1977	11/29/1977	B	CE	CQ		B	OA	D	N	ISOLATION CONDENSER STEAM SUPPLY VALVE INOPERABLE DUE TO UNKNOWN CAUSE
LER 77-36	143547	11/07/1977	12/02/1977	B	IA		I,T	C	AG,AQ	G	N	LOW-LOW LEVEL SWITCH FAILS TO TRIP DUE TO GRIT
RO 77-37	144121	11/30/1977	3/03/1978	C	ZZ	GG		C	OA	E	N	TWO SNUBBERS ARE DECLARED INOPERABLE
RO-77-38	143485	11/10/1977	12/09/1977	B	SF-B	QQ		C	ED	G	C8	MAINTENANCE FOREMAN INADVERTENTLY SHORT CIRCUITS VALVE OPERATOR ON LPCI VALVE
RO-77-39	144187	12/10/1977	12/12/1977	B	EE	K,KK		C	OA	D	S1,S7	DIESEL GENERATOR IS DECLARED INOPERABLE. GAS TURBINE UNAVAILABLE
RO 77-40	144186	12/13/1977	12/14/1977	B	SC					G	S9	2 HYDROGEN EXPLOSIONS IN OFF-GAS SYSTEM (REACTOR SHUTDOWN)
RO 77-43	133684	12/19/1977	1/18/1978	D	BB	KK	N	A	BC	H	N	STACK GAS SAMPLER AND STRUCTURE DAMAGED DUE TO HYDROGEN EXPLOSION

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1978

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
RO-78-1	134979	1/16/1978	2/14/1978	B	CC		M,T	C	EH	D	N	SET POINT DRIFT IN MAIN STEAM LINE LOW PRESSURE SWITCHES
RO-78-2	134961	1/17/1978	2/14/1978	B	SH-D	P		C	AA	D	N	PRESSURE DROP LIMIT EXCEEDED DUE TO NORMAL WEAR OF FILTER
RO-78-3	136464	2/14/1978	2/28/1978	B	HA	QQ	P	C	EH	D	N	SET POINT DRIFT IN TURBINE CONTROL TIME DELAY RELAY
RO-78-4	142572	3/10/1978	3/24/1978	C	IB	CC		B	BB	D	N	SAFETY/RELIEF VALVE FAILS TO RESEAT - CAUSE UNKNOWN
RO-78-5	137251	3/19/1978	4/03/1978	C	CB	CC,Z		A	AO,AI,AT	B	N	FATIGUE CAUSES WELD LEAK IN RECIRCULATION DISCHARGE VALVE VENT LINE
RO-78-6	137252	3/20/1978	4/03/1978	C	CD	CC		A	AT	D	N	CONTAINMENT ISOLATION VALVE EXCESS LEAKAGE
RO-78-7	137504	3/11/1978	4/10/1978	C	CE	H,Z		A	BS	G	N	PIPE MOVEMENT IN ISOLATION CONDENSER DUE TO WATER IN STEAM LINES

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1978

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
RO-78-8	137504	4/17/1978	4/20/1978	B	SF-B SF-D	KK		B	AH	B	N	DEGRADATION OF CORE SPRAY AND LPCI PIPING SUPPORT STRUCTURE DUE TO POOR DESIGN
RO-78-9	138250	4/03/1978	4/25/1978	B	CC		M,T	C	EH	D	N	SET POINT DRIFT IN STEAM TUNNEL TEMPERATURE SWITCHES
RO-78-10	138838	4/24/1978	5/23/1978	B	RB	J,00		C	BI	E	N	SCRAM TIME TOO SLOW FOR CONTROL RODS DUE TO TIGHT PACKING ON VALVE
RO-78-11	139384	5/08/1978	6/06/1978	B	IA		M,T	C	EH	D	N	SET POINT DRIFT IN CONTAINMENT PRESSURE SWITCH
RO-78-12	139640	5/19/1978	6/16/1978	D	EE	KN,T		C	BI,EI	D	N	GAS TURBINE FAILS TO COMPLETE STARTUP SEQUENCE DUE TO INCORRECT GOVERNOR SETTING

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1978

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
RO-78-13	139817	5/29/1978	6/20/1978	D	CE	M		B	BL	G	N	STEAM TRAP BLOWBY CAUSES ISOLATION CONDENSER TO BE REMOVED FROM SYSTEM
RO-78-14	139876	6/13/1978	7/12/1978	B	EE	KN,T	T	B	BW	D	N	GAS TURBINE TRIPS OUT DUE TO OVERSPEED SWITCH FAILURE
RO-78-16	140261	7/05/1978	8/04/1978	B	HC	M	M,T	B	EH	D	N	SET POINT DRIFT IN CONDENSER LOW VACUUM SWITCH
LER 78-16	140386	7/25/1978	8/09/1978	B	SH-A		A	B	OK	G	C8	PROCEDURAL ERROR ALLOWED CONTAINMENT TO BE PURGED WITHOUT HIGH RADIATION MONITOR
RO-78-17	140032	7/10/1978	8/04/1978	B	FB		N	C	EI	G	N	CALIBRATION ERRROR IN SPENT FUEL STORAGE AIR MONITORS
RO-78-18	141762	9/5/1978	10/04/1978	B	IA		I	C	AK	G	N	LACK OF LUBRICATION CAUSES LOW-LOW WATER LEVEL SENSORS TO FAIL
RO-78-19	141763	9/12/1978	10/10/1978	B	IA,HA	KN	P	C	EH	D	N	SET POINT DRIFT IN TURBINE CONTROL TIME DELAY RELAY

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1978

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
RO-78-20	148110	10/10/1978	10/10/1978	B	WC	W		B	DA	D	N	CHLORINE CAPACITY OF DEMINERALIZER LESS THAN TECHNICAL SPECIFICATION
RO-78-21	141139	9/14/1978	10/12/1978	B	EE	NN	T	C	BF	D	N	FAULTY SPEED SWITCH TRIPS GAS TURBINE
RO-78-22	141149	9/14/1978	10/13/1978	B	ZZ	QQ,BB		B	AQ	G	N	DRYWELL VENT BYPASS VALVE FAILS TO CLOSE DUE TO DIRT IN AIR OPERATOR
RO-78-23	141726	10/11/1978	11/6/1978	B	IA	BB	M,T	C	EB,AK	D	N	SET POINT DRIFT IN DRYWELL HIGH PRESSURE SWITCH
RO-78-24	141729	10/19/1978	11/16/1978	B	CE	QQ,H	T	B	BF	G	N	SPURIOUS OPENING OF ISOLATION CONDENSER ISOLATION VALVE DUE TO SET POINT DRIFT
RO-78-26	142286	11/06/1978	11/30/1978	B	IA	BB	M,T	C	EH	D	N	SET POINT DRIFT IN DRYWELL HIGH PRESSURE SWITCH

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1978

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
RO-78-27	142285	11/06/1978	12/01/1978	B	IA		I	C	EH	D	N	SET POINT DRIFT IN REACTOR LOW LEVEL SWITCH
RO-78-29	142849	11/22/1978	12/22/1978	B	EE	NN,T	G	C	OA,AA	G	N	GAS TURBINE INOPERABLE WHEN MAINTENANCE CREW HAD TO REPAIR DAMAGED INDICATOR LIGHT SOCKET
LER-78-30	146505	12/18/1978	1/17/1979	B	ZZ	BB			OK,BU	A	N	EXCESS OXYGEN IN DRYWELL DUE TO PROCEDURAL ERROR

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1979

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-79-1	146724	1/06/1979	1/18/1979	D	CG	Z		A	AO.AV	D	N	STRESS CORROSION CRACKING OF CLEANUP SYSTEM RETURN LINE WELD (REACTOR SHUTDOWN)
LER-79-2	146643	1/07/1979	2/06/1979	D	SH-D		P	B	BC.BD	E	N	STANDBY GAS TREATMENT SYSTEM 'A' FAILS TO START DUE TO MISALIGNMENT OF STARTUP RELAY
LER-79-3	146644	1/08/1979	2/07/1979	D	CC.ID		M.T	C	EH	D	N	SET POINT DRIFT IN MAIN STEAM LINE PRESSURE SWITCH
LER-79-4	147428	1/25/1979	2/23/1979	B	RB	I		C	AG	D	N	CONTROL ROD STICKS - CAUSE UNKNOWN
LER-79-5	147414	2/26/1979	2/26/1979	B	CC	OO		D	AB	D	C8	PRESSURE RELIEF VALVE LIFTS PREMATURELY AND FAILS TO SEAT DUE TO STEAM CUTTING OF DISC (REACTOR SHUTDOWN)
LER-79-6	147295	2/01/1979	3/02/1979	B	RB	I		B	OJ	H	C6	2 CONTROL RODS INOPERABLE DUE TO OPERATOR ERROR

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1979

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-79-7	147413	2/14/1979	3/09/1979	B	EE	NN,T	T	C	BD	D	N	GAS TURBINE GENERATOR FAILS TO START DUE TO FAULTY SPEED SWITCH
LER-79-8	147415	2/14/1979	3/14/1979	B	CE,ID	QQ	K	A	AD	D	N	RCIC VALVE LOSES POSITION INDICATOR DUE TO FAILURE OF VALVE OPERATOR CASING
LER-79-9	148229	2/22/1979	3/22/1979	D	CC	CC	G	C	OA,AV	D	N	SAFETY/RELIEF VALVE BELLOWS MONITOR INOPERABLE DUE TO CRACK IN AIR TUBE FITTING
LER-79-10	149268	4/03/1979	4/18/1979	B	RC			B	DJ	H	N	CORE THERMAL POWER EXCEEDS TECHNICAL SPECIFICATIONS DUE TO OPERATOR ERROR
LER-79-11	149267	3/17/1979	4/17/1979	B	CD	CO	P	C	BB,BC	D	N	MSIV RELAY FAILS TO DEENERGIZE DUE TO MALADJUSTMENT OF RELAY LIMIT SWITCH (REACTOR SHUTDOWN)
RO-79-11	152303	5/14/1979	5/18/1979	C	SA	Z		A	AA	D	N	PENETRATION LEAK RATE EXCEEDED

TABLE A.22 DATA TABLES FOR REP ABLE EVENTS FOR MILLSTONE 1-1979

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-79-12	149544	4/23/1979	5/21/1979	B	BA		N	C	EH	D	N	SET POINT DRIFT IN REACTOR BUILDING RADIATION MONITOR
LER-79-15	150110	5/28/1979	6/22/1979	C	RB	J	K	C	AH	D	N	CONTROL ROD OUT BLOCK FUNCTION FAILS DUE TO WORN LIMIT SWITCH
LER-79-16	150109	5/30/1979	6/29/1979	C	IA		M,T	C	EH	D	N	SET POINT DRIFT IN REACTOR LOW PRESSURE SWITCH
LER-79-17	150108	6/02/1979	6/28/1979	C	IA		U,T	C	EH	D	N	SET POINT DRIFT IN STREAM TUNNEL TEMPERATURE SWITCH
LER-79-18	150748	7/13/1979	7/26/1979	B	SF-B			A	OK,BU	A	C8	CONCENTRATION IN STANDBY LIQUID CONTROL SYSTEM LTA DUE TO PROCEDURAL ERROR
LER-79-19	150782	6/28/1979	8/03/1979	D	CE	CC		C	AA,AU	D	N	CONTAINMENT ISOLATION VALVE LEAKAGE DUE TO NORMAL WEAR
	150990	2/16/1979	3/19/1979	B	IA		G	B	BU,CK	A	N	ABNORMAL OXYGEN LEVELS IN CONTAINMENT DUE TO CANCELLATION OF SHUTDOWN

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1979

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-79-20	151245	7/11/1979	8/09/1979	B	IA		I,T	C	EH	D	N	SET POINT DRIFT IN REACTOR LOW-LOW LEVEL SWITCH
LER-79-21	151244	7/12/1979	8/10/1979	B	IA		I,T	C	AQ	G	N	LOW-LOW WATER LEVEL SWITCH FAILS TO TRIP DUE TO GRIT ON SHAFT ASSEMBLY
LER-79-22	151211	7/23/1979	8/22/1979	B	SC	QQ		B	BB	D	N	CONTAINMENT VENT BYPASS VALVE FAILS TO CLOSE - CAUSE UNKNOWN
LER-79-23	151207	8/08/1979	8/24/1979	B	SC	QQ		B	BB	D	N	CONTAINMENT VENT BYPASS VALVE FAILS TO CLOSE - CAUSE UNKNOWN
LER-79-24	151425	7/31/1979	8/30/1979	B	BA		N	C	EH	D	N	SET POINT DRIFT IN REFUELING FLOOR RADIATION MONITOR
LER-79-25	151913	9/13/1979	9/25/1979	B	RB			B	OK	B	N	TOTAL PEAKING FACTOR IN ERROR DUE TO DESIGN OVERSIGHT
LER-79-26	151912	9/14/1979	9/27/1979	B	SF			B	OK	B	S7	POTENTIAL EXISTS FOR LOSS OF POWER TO ECCS TO GO UNDETECTED - DESIGN ERROR

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1979

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-79-27	151758	8/27/1979	9/24/1979	B	AB			B	BC	G	N	SYSTEM NOT TESTED ACCORDING TO SCHEDULE
LER-79-28	151911	8/28/1979	9/25/1979	B	CC	QQ		C	BB,AR	D	N	PRESSURE SUPPRESSION CHAMBER VENT BYPASS VALVE FAILS TO CLOSE DUE TO RUST BUILDUP
LER-79-29	151930	9/04/1979	10/04/1979	B	CE	QQ,H	I	C	BB	D	N	ISOLATION CONDENSER ISOLATION VALVE FAILS TO CLOSE DUE TO FAULTY MICROSWITCH
LER-79-30	152940	10/09/1979	11/08/1979	B	EE	KK,N		B	AL	E	N	SERVICE WATER SYSTEM PIPES TO DIESEL GENERATOR NOT RESTRAINED DUE TO INSTALLATION ERROR
LER-79-31	153647	10/16/1979	11/15/1979	B	SF-C	KK		B	DA,AL	E	N	FEEDWATER COOLANT INJECTION SYSTEM DECLARED INOPERABLE DUE TO MISSING SUPPORT STRUCTURE
LER-79-32	153362	11/06/1979	12/03/1979	B	IA		M,T	C	EH	D	N	SET POINT DRIFT IN DRYWELL PRESSURE SWITCHES

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1979

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-79-33	153360	11/13/1979	12/04/1979	B	IA,CC		M,T	C	EH	D	N	SET POINT DRIFT IN MAIN STEAM LINE DELTA P SWITCH
LER-79-34	153749	11/15/1979	12/14/1979	B	SF-B	GQ,X		C	OA,ED	D	N	LPCI VALVE INOPERABLE DUE TO ELECTRICAL FAULT IN MOTOR CONTROLLER
LER-79-35	153906	12/04/1979	12/20/1979	B	IA		M,T	C	EH	D	N	SET POINT DRIFT IN MAIN STEAM LOW PRESSURE SWITCH
RO-79-36	153942	12/19/1979	1/18/1980	D	CE	Z,H		B	HH	D	N	WATER HAMMER IN ISOLATION CONDENSER PIPING

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1980

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-80-01	154444	1/03/1980	2/01/1980	B	CB	DD			BW	D	N	PUMP SPEED MISMATCH DURING RECIRCULATION PUMP RUNBACK
LER-80-001	155196	1/31/1980	2/14/1980	B						B	N	MAPLHGR WRONG
LER-80-3	154614	1/04/1980	2/04/1980	B	CB	KK		B	AH	B	N	STRUCTURAL DEFICIENCY IN ISOLATION CONDENSER SYSTEM SUPPLY LINE (REACTOR SHUTDOWN)
LER-80-4	155195	1/21/1980	2/14/1980	B	IA,CC		E,T	C	EH	D	N	SET POINT DRIFT IN MAIN STEAM LINE HIGH FLOW SWITCH
LER-80-5	156005	2/20/1980	3/17/1980	B	SF-D	KK		C	AC,HH	D	N	CORE SPRAY SUPPORTS DAMAGED BY WATER HAMMER
LER-80-6	156157	4/07/1980	4/21/1980	B	IA		M,T	C	OK	E	N	PRESSURE SENSOR ISOLATES DUE TO INSTALLATION
LER-80-8	158283	6/05/1980	7/02/1980	B	SA		M,T	C	EH	D	N	SET POINT DRIFT IN TWO REACTOR LOW-PRESSURE SWITCHES

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1980

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-80-9	158276	6/05/1980	7/02/1980	B	IA		M,T	C	EM	D	N	SET POINT DRIFT IN REACTOR PROTECTION LOW PRESSURE SWITCHES
LER-80-10	159128	6/13/1980	7/11/1980	B	RA	KK		B	AC	E	N	BOLTS ON PENETRATION BASE PLATE FAULTY DUE TO INSTALLATION ERROR
LER-80-11	160233	8/03/1980	8/15/1980	B	BB	DD	N	C	BC,DA	G	N	OFF GAS RADIATION MONITORING SYSTEM INOPERABLE DUE TO VALVE MISALIGNMENT
LER-80-12	159289	7/25/1980	8/22/1980	B	IC		N	C	EI	G	N	APRM READS LOW DUE TO CALIBRATION ERROR
LER-80-13	160459	9/08/1980	10/03/1980	B	IA		M,T	C	EM	D	N	SET POINT DRIFT IN HIGH PRESSURE SWITCH
LER-80-14	160060	10/05/1980	10/16/1980	C	CD	CC		C		D	N	LEAK RATE TEST FAILURE OF TWO MSIV'S

TABLE A.22 DATA TABLES FOR RE TABLE EVENTS FOR MILLSTONE 1-1980

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-80-15	160923	10/07/1980	10/24/1980	C	IA		I,T	C	AQ	G	N	SET POINT DRIFT IN LOW WATER LEVEL SCRAM SWITCH - GRIT
LER-80-16	161870	10/23/1980	11/06/1980	C	CB	DD,KK		C	AH	C	N	CRACKS FOUND IN JET PUMP SUPPORT BEAMS
LER-80-18	161470	11/05/1980	11/19/1980	C	CC	Z		C	AO	D	N	WELD FAILURE IN TWO MAIN STEAM LINES
LER-80-19	161471	11/06/1980	11/20/1980	C	CC	Y		C	AO	D	N	CONDENSER NOZZLE WELD CRACKS DUE TO STRESS CORROSION

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1981

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-81-01	164391	2/09/1981	2/23/1981	C	SF-B	Z		C	AV	D	C8	A LPCI SYSTEM PIPING WELD IS CRACKED
LER-81-02	165884	4/03/1981	4/20/1981	C	EE	N,NN	P		OA	B	S7	POTENTIAL SINGLE FAILURE IN EMERGENCY POWER SYSTEM
LER-81-03	166144	4/19/1981	5/01/1981	B	SD	OO		B	BC	G	C8	TWO CONTAINMENT INSTRUMENT ISOLATION VALVES WERE CLOSED
LER-81-04		4/21/1981	5/05/1981	D	RF	BB		B	BP	B	C4	A HIGH COOLDOWN RATE OCCURRED WHEN THE REACTOR HAD TO BE BLOWN DOWN MANUALLY
LER-81-005	166575	4/01/1981	6/17/1981	C	IA		S	B	EJ	D	N	HIGH COUNT RATE ON TWO STARTUP RANGE MONITORS
LER-81-05	165924	4/07/1981	4/30/1981	C	IB		U,T	C	EH	D	N	SET POINT DRIFT IN TWO STEAM TUNNEL SWITCHES
LER-81-06	165944	4/17/1981	5/06/1981	C	MC		N	C	EH	D	N	SET POINT DRIFT IN REACTOR BUILDING RADIATION MONITOR
LER-81-07	166243	4/18/1981	5/18/1981	B	CC	OO		C	BA,AQ	D	N	ONE OF SIX RELIEF VALVES FAILS TO OPEN

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1981

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-81-08	168485	4/27/1981	6/08/1981	D	EE	N		C	BF,AT	D	N	DIESEL GENERATOR TRIPS ON HIGH CRANKCASE PRESSURE DUE TO VACUUM BREAK
LER-81-09	166632	5/11/1981	6/08/1981	D	IA		N	C	EH	D	N	SET POINT DRIFT IN MAIN STEAM LINE RADIATION MONITOR
LER-81-10	166630	5/15/1981	6/12/1981	D	SF		M,T	C	EH	D	N	SET POINT DRIFT IN THREE BREAK DETECTION PRESSURE SWITCHES
LER-81-011	167616	5/24/1981	7/02/1981	D	WE			B	BU	D	C3	HIGH COBALT AND SILVER ACTIVITY IN OYSTERS
LER-81-12	166839	6/03/1981	7/03/1981	D	CE		P	C	EH	D	N	SET POINT DRIFT IN ISOLATION CONDENSER TIME DELAY RELAY

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1981

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-81-13	167165	6/12/1981	7/10/1981	B	SE	P		B	AQ,HB	D	N	LOW STANDBY GAS TREATMENT FLOW RATE DUE TO FOULED FILTER
LER-81-15	167182	6/17/1981	7/17/1981	B	SA	FF		B	DC	K	N	CONTAINMENT AIR LOCK LEAK RATE TEST MISSED
LER-81-16	167166	6/18/1981	7/17/1981	B	IA	DD	P	C	EH	D	N	MSIV CLOSURE FAILS TO GENERATE REACTOR PROTECTION SIGNAL DUE TO RELAY DRIFT
LER-81-17	167517	6/20/1981	7/17/1981	B	SA				BU,DK	A	N	TORUS OXYGEN CONCENTRATION EXCEEDED LIMIT
LER-81-02E	167613	6/22/1981	7/02/1981	B	HE	CC		B	AX	D	C3	UNMONITORED RADIOACTIVE LIQUID WASTE RELEASE
LER-81-18	167924	6/29/1981	7/27/1981	B	AB		U	C	AC	G	N	DIESEL DAY TANK FIRE DETECTOR SYSTEM FAILS

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1981

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-81-19	167923	7/01/1981	7/27/1981	B	IA		N,I	C	EH	D	N	SET POINT DRIFT IN REACTOR VESSEL HIGH PRESSURE SWITCH
LER-81-20	168108	7/14/1981	8/06/1981	B	EE	HB		C	DB,DA	D	C7	GAS TURBINE GENERATOR FAILS TO START DUE TO STUCK VALVE
LER-81-03S	168780		9/03/1981	B	CE	Z		B	OC,OK	A	N	ISOLATION CONDENSER SUPPLY LINE WELD NOT INSPECTED
LER-81-21	168111	7/16/1981	8/11/1981	O	IA		I	C	EH	D	N	SET POINT DRIFT IN CONDENSER LOW VACUUM SWITCH
LER-81-22	168555	8/10/1981	9/09/1981	O	IA	UE	F	C	EH	D	N	MSIV CLOSURE FAILS TO GENERATE REACTOR PROTECTION SIGNAL SIGNAL DUE TO RELAY DRIFT
LER-81-23	169053	8/10/1981	9/09/1981	O	IA		N	C	EH	D	N	MAIN STEAM LINE HI RADIATION CHANNEL TRIPS ABOVE LIMIT
LER-81-03L	168043	8/13/1981	8/24/1981	B	MC		N	O	OC,OJ	H	C3	UNMONITORED RELEASE OF LIQUID EFFLUENT
LER-81-24	168935	8/17/1981	9/17/1981	O	HB		N	C	EH	D	N	SET POINT DRIFT IN REACTOR BUILDING EXHAUST DUCT RADIATION MONITOR

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1981

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-81-25	169185	9/15/1981	9/30/1981	B	CB	DD		B	BF	B	S4	BOTH REACTOR RECIRCULATION PUMPS TRIP. ATWS WAS ISOLATED, NO ALARM
LER-81-26	169235	9/05/1981	10/05/1981	B	CE		M,T	C	EH	D	N	SET POINT DRIFT IN ISOLATION CONDENSER PRESSURE SWITCH
LER-81-27	169213	9/06/1981	10/06/1981	B	SD	DD	T	C	BI,EH	D	N	CONTAINMENT ISOLATION VALVE CLOSING TIME EXCEEDS LIMIT
LER-81-28	169347	8/11/1981	10/07/1981	D	EE	F,NN		C	BB,BD	D	C7	BREAKER FAILS TO CLOSE CAUSING NO OUTPUT FROM GAS TURBINE
LER-81-29	169346 170103	9/08/1981	11/19/1981	B	IB		I,T	C	AG	D	N	LO-LO REACTOR WATER LEVEL SWITCH STICKS
LER-81-30	169998	9/14/1981	10/13/1981	B	IA		P	C	EH	D	N	SET POINT DRIFT IN TURBINE CONTROL VALVE CLOSURE RELAYS
LER-81-31	169337	9/10/1981	10/09/1981	B	EE	F,NN		C	BB,BD	D	N	GAS TURBINE GENERATOR OUTPUT BREAKER FAILS TO CLOSE

TABLE 4.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1 (1981)

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-81-32	165207	9/25/1981	10/21/1981	B	AB	DD	P	C	BB	D	N	DIESEL FIRE PUMP FAILS TO START
LER-81-33	169605	10/06/1981	11/05/1981	B	SB		L,T	C	EH	D	N	SET POINT DRIFT IN REACTOR VESSEL LO WATER LEVEL ISOLATION SWITCH
LER-81-34		10/13/1981	11/12/1981	B	IA		P	C	EH	D	N	SET POINT DRIFT IN TURBINE CONTROL VALVE BYPASS RELAY
LER-81-35	171268	11/10/1981	12/09/1981	B	IA	G		B	AP	G	N	MAIN STEAM LINE RADIATION MONITOR CHANNEL FAILS WHEN WORKERS BUMPED CABLES
LER-81-36	171790	11/12/1981	12/11/1981	B	IA		H,T	C	EH	D	N	SET POINT DRIFT IN DRYWELL PRESSURE SWITCH
LER-81-38	171933	11/12/1981	12/17/1981	B	IA		H,T	C	BC,DJ	H	N	TURBINE CONTROL PRESSURE SWITCH LEFT ISOLATED
LER-81-39	171599	11/24/1981	12/29/1981	B	SD		P	B	EE	D	N	RELAY FAILURE DISABLES MAIN STEAM HI FLOW ALARM
LER-81-40	171601	12/03/1981	1/03/1982	B	CE	QC,X		C	BA,AQ	D	N	CONTAINMENT ISOLATION VALVE MOTOR FAILS
LER-81-41	172262	12/08/1981	1/12/1982	B	EE	EA		C	AQ	D	N	GAS TURBINE GOVERNOR FAILS
LER-81-42	172064	12/15/1981	1/15/1982	B	CF	CC		B	AG	D	N	ISOLATION VALVE IN SHUTDOWN COOLING LINE INOPERABLE

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1982

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
ETS-82-03		5/04/1982	6/03/1982	B	WE			B	BU	D	N	HIGH IODINE AND SILVER ACTIVITY IN OYSTERS
ETS-82-05		8/04/1982	9/13/1982	B	WE			B	BU	D	C3	HIGH COBALT AND SILVER ACTIVITY IN OYSTERS
ETS-82-06		11/08/1982	12/16/1982	B	WE			B	BU	D	N	HIGH COBALT AND SILVER ACTIVITY IN OYSTERS
LER-82-1	172925	1/15/1982	2/22/1982	B	CC		E,T	C	EH	D	N	SET POINT DRIFT ON 2 FLOW SWITCHES IN MAIN STEAM LINE
LER-82-2	172926	1/15/1982	2/22/1982	B	WA	DD		C	BC,DA	G	N	EMERGENCY SERVICE WATER PUMP IS DECLARED INOPERABLE DUE TO MARINE FOULING
LER-82-3	172726	2/17/1982	3/11/1982	B	EB	LL	T	B	AB,FD,BY	D	C8	CONTACT IN 345KV CIRCUIT SWITCHER STICKS OPEN RESULTING IN FIRE IN SWITCHER
LER-82-4	172457	2/12/1982	3/23/1982	C	CE	H,OO		C	BA,OJ	B	N	CONDENSER ISOLATION VALVE STICKS CLOSED DUE TO OPERATOR ERROR
LER-82-5	172502	2/24/1982	3/23/1982	B	IA		F	B	AE,AN	C	C4	CONTINUOUSLY ENERGIZED RELAYS IN REACTOR PROTECTION SYSTEM ARE FOUND MELTED
LER-82-6	172945	2/12/1982	3/25/1982	C	CD	OO,QQ		C	BC,BI	G	N	MAIN STEAM ISOLATION VALVE CLOSED SLOWLY DUE TO OUT-OF-ADJUST-MENT HYDRAULIC CYLINDER
LER-82-7	172836	3/15/1982	4/09/1982	B	CE		M,T	C	EB	D	N	SET POINT DRIFT CAUSES CONDENSER PRESSURE SWITCH TO TRIP [illegible]

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1982

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-82-8	173024	3/18/1982	4/26/1982	B	CC,SB,B, F,OO,EB			B	AT,AU	D	C4	FAILURE OF 3 BREAKERS IN AN MOTOR CONTROL CENTER DUE TO WATER DAMAGE
LER-82-9	173644	4/12/1982	5/24/1982	B	SA,IA		T,M	C	EB	D	N	SET POINT DRIFT CAUSES DRYWELL PRESSURE SWITCH TO FAIL HIGH
LER-82-10	173617	4/13/1982	5/25/1982	B	WA,SB	OO,QQ		C	BA,BC	D	N	VALVE IN EMERGENCY SERVICE WATER SYSTEM FAILS TO OPEN - POTENTIOMETER OUT OF ADJUSTMENT
LER-82-11	175146	5/08/1982	6/22/1982	B	EB,EE	NN,DD,S	P	B	AT,BF, OA	D	C7	UNDERVOLTAGE RELAY FOR AC LUBE PUMP SHORTED/BLEW FUSE - GAS TURBINE DECLARED INOPERABLE
LER-82-12	175230	7/02/1982	7/27/1982	B	SF-B	OO,G	P	B	OK	E	N	LPCI VALVE TIMER WIRED INCORRECTLY TIMER PREVENTS THROTTLING IN 1ST 5 MINUTES
LER-82-13	175233	6/15/1982	7/27/1982	B	ED,FA	OO,NN, Y		C	AR,BA, BB,OA	D	C7	AIR PRESSURE REGULATING VALVE FOR GAS TURBINE FAILS DUE TO RUST IN STARTING AIR SYSTEM
LER-82-14	175235	6/14/1982	7/27/1982	B	WA,SB	DD		C	AQ,BB,OA	D	N	ONE EMERGENCY WATER SUPPLY PUMP HAS LOW PRESSURE DUE TO MARINE GROWTH AND MUSSELS
LER-82-15	175979	7/16/1982	9/09/1982	B	CC,IA	OO	F,T	C	BA,BC	D	N	NO SCRAM SIGNAL ON CLOSING OF MSIV DUE TO OUT-OF-ADJUST MENT LIMIT SWITCH
LER-82-16	175980	3/10/1982	9/09/1982	B	MC		N,T	B	OJ	H	C7	OPERATOR FAILS TO PLACE STACK GAS MONITORING SYSTEM IN NORMAL POSITION AFTER DAILY TEST

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1982

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-82-17	176743	8/17/1982	9/24/1982	B	EE	NB,T, OO		C	AR,BO,OA	D	C7	GAS TURBINE FAILS 2 FAST START TESTS DUE TO RUST IN AIR PRESSURE REGULATING VALVE
LER-82-18	177793	9/25/1982	10/21/1982	C	RA	Y,Z,KK		C	AO,AV	C	C4	STRESS CRACK IN WELD ON CORE SPRAY SPARGERS
LER-82-19	177794	9/28/1982	10/22/1982	C	CC	OO		C	BA,EH	D	C7	SET POINT DRIFT IN 5 SAFETY RELIEF VALVES
LER-82-20	180374	9/30/1982	1/14/1983	C	ZZ	GG		C	AT	E	N	2 HYDRAULIC SNUBBERS FAIL FUNCTIONAL TEST DUE TO LACK OF HYDRAULIC FLUID
LER-82-21	177796	9/13/1982	10/22/1982	C	EE	LL,G, N		C	AP,BF,AE	E	N	DIESEL GENERATOR TRIPS DUE TO BROKEN WIRE LUG AT CURRENT TRANSFORMER
LER-82-22	178034	9/28/1982	11/06/1982	C	MC		O	B	OJ	H	C7	OPERATOR INADVERTENTLY SECURED THE STACK-GAS RECORDER
LER-82-23	184680	10/05/1982	7/06/1983	C	SB	OO		C	AU,OA	D	N	LEAK RATE FOR 6 ATMOSPHERE CONTROL VALVES IS GREATER THAN TECHNICAL SPECIFICATION LIMITS
LER-82-24	179433	10/19/1982	11/29/1982	C	CD,IB		E,T	C	EH,OA	D	N	SET POINT DRIFT OF FLOW SWITCH IN MAIN STEAM LINE
LER-82-25	179487	10/26/1982	12/03/1982	C	IA		M,T	C	EH	D	N	SET POINT DRIFT CAUSES PRESSURE SWITCH IN REACTOR TRIP SYSTEM TO TRIP TOO HIGH
LER-82-26	179542	11/04/1982	12/06/1982	C	CC		U,T	C	EH	D	N	SET POINT DRIFT CAUSES 6 STEAM TUNNEL TEMPERATURE SWITCHES TO TRIP TOO HIGH

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1982

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-82-27	181506	11/11/1982	2/03/1983	C	HI		F	C	EH	D	N	SET POINT DRIFT CAUSES PRESSURE RELIEF TIME DELAY RELAYS IN BLOWDOWN LOGIC TO FAIL LONG
LER-82-28	180403	11/29/1982	1/04/1983	B	AB	G		C	AL	C	N	MISSING FIRE BARRIER IN THE CABLE VAULT AREA
LER-82-29	186233	12/07/1982	10/18/1983	B	AB	QQ,JJ		B	AG,OA	D	N	LOW WATER LEVEL IN 2 FIRE WATER STORAGE TANKS DUE TO FAILURE OF 2 VALVE OPERATORS
LER-82-30	183922	12/13/1982	4/27/1983	B	CE,SD	OO,H, QQ	T	C	AK,BB,OA	G	N	ISOLATION CONDENSER ISOLATION VALVE FAILS TO CLOSE DUE TO LUBRICANT ON LIMIT SWITCH
LER-82-31	181451	12/14/1982	1/20/1983	B	CC		M,T	C	EB	D	N	SET POINT DRIFT CAUSES PRESSURE SWITCH IN MAIN STEAM 'INE TO TRIP 'OO HIGH
LER-82-32	181452	12/17/1982	1/25/1983	B	EE	NN,T	S	C	BD,OJ	H	C7	GAS TURBINE FAILS TO START DUE TO GENERATOR OVERSPEED

TABLE A.22 DATA TABLE FOR REPORTABLE EVENTS FOR MILLSTONE 1-1983

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
ETS-83-02		2/23/1983	5/20/1983	B	WE			B	BU	D	N	HIGH COBALT ACTIVITY IN MUSSELS
ETS-83-04		8/11/1983	10/03/1983	B	WE			B	BU	D	N	HIGH COBALT AND SILVER ACTIVITY IN OYSTERS
ETS-83-05		10/12/1983	11/10/1983	B	WE			B	BU	D	N	HIGH COBALT ACTIVITY IN AQUATIC FLORA
ETS-83-06		11/15/1983	12/22/1983	B	WE			B	BU	D	N	HIGH COBALT ACTIVITY IN OYSTERS
LER-83-1	181197	1/03/1983	2/03/1983	B	CC		M,T	C	EB	D	N	SET POINT DRIFT CAUSES PRESSURE SWITCH IN MAIN STEAM LINE TO TRIP TOO LOW
LER-83-2	181224	1/04/1983	2/10/1983	B	IA	BB	M,T	C	EI	G	N	PRESSURE SWITCH FOR DRYWELL PRESSURE IS OUT OF CALIBRATION
LER-83-3	182110	2/03/1983	3/14/1983	B	IA		M,T	C	EB	D	N	SET POINT DRIFT CAUSES A REACTOR VESSEL PRESSURE SWITCH TO FAIL TO TRIP
LER-83-4	182111	2/07/1983	3/14/1983	B	EE	BB,N		E	AO,BA, ED,DA	D	C/	DIESEL GENERATOR FAILS TO START DUE TO STUCK AIR START SOLENOIDS
LER-83-5	181862	2/07/1983	3/08/1983	B	CC		M,T	C	EB	D	N	SET POINT DRIFT CAUSES MAIN STEAM LINE PRESSURE SWITCH TO TRIP TOO LOW
LER-83-6	181863	1/25/1983	3/08/1983	D	CC		U,T	C	EB	D	N	SET POINT DRIFT CAUSES A STEAM TUNNEL TEMPERATURE SWITCH TO TRIP TOO HIGH
LER-83-7	183578	2/16/1983	4/27/1983	B	CC,IA	OO	P,T	C	BA,BC	D	N	NO SCRAM SIGNAL ON CLOSING OF MSIV DUE TO AN OUT-OF-ADJUSTMENT LIMIT SWITCH
LER-83-8	[illegible]2225	2/15/1983	3/24/1983	B	CC		M,T	C	EB	D	N	SET POINT DRIFT CAUSES ISOLATION CONDENSER PRESSURE SWITCH TO TRIP TOO HIGH
LER-83-9	182398	3/07/1983	4/16/1983	B	IA		I,T	C	EB	D	N	SET POINT DRIFT CAUSES REACTOR WATER LEVEL SWITCH TO TRIP TOO HIGH

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1983

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-83-10	182597	3/24/1983	4/27/1983	B	CD	OO		C	BI	G	N	MSIV CLOSES TOO SLOWLY DUE TO TIGHT PACKING AROUND THE VALVE STEM
LER-83-11	185473	3/15/1983	4/22/1983	B	HA,IA	NN,OO	P	C	BI,EH	D	N	SET POINT DRIFT CAUSES TIME DELAY RELAY THAT BYPASSES TURBINE SCRAM TO FAIL LONG
LER-83-12	188837	3/26/1983	1/30/1984	D	SF-B, SF-D	OO	T	B	OJ	H	N	VALVE PERMISSIVE INTERLOCK SWITCH FOR LPCI/OS NOT RETURNED TO SERVICE AFTER TEST
LER-83-13	183025	3/24/1983	4/28/1983	B	CH	DD,OO		B	BF,OK	H	N	REACTOR FEED AND BOOSTER PUMPS TRIP DUE TO LOW SUCTION FLOW
LER-83-14	185711	3/31/1983	8/16/1983	D	EE	NN		C	AB,BD	G	C7	THE GAS TURBINE GENERATOR FAILS TO START DUE TO FAILURE OF BOTH IGNITORS
LER-83-15	182861	4/11/1983	5/23/1983	B	HA,IA	NN,OO	P	C	BI,EH	D	N	SET POINT DRIFT CAUSES TIME DELAY RELAY THAT BYPASSES TURBINE TRIP TO FAIL LONG
LER-83-16	183101	5/02/1983	6/03/1983	B	IA		M,T	C	EH	D	N	SET POINT DRIFT CAUSES A REACTOR VESSEL HIGH PRESSURE SWITCH TO TRIP TOO HIGH
LER-83-17	183102	5/02/1983	6/03/1983	B	IA		M,T	C	EH	D	N	SET POINT DRIFT CAUSES REACTOR LOW PRESSURE SWITCH TO TRIP TOO HIGH
LER-83-18	183935	5/17/1983	7/01/1983	B	EE	T,NN		C	AA,BI	D	C7	GAS TURBINE FAILS TO REACH READY TO LOAD STATUS DUE TO WORN IGNITOR
LER-83-19	183149	5/03/1983	6/17/1983	B	IA		M,T	C	EH	D	N	SET POINT DRIFT CAUSES REACTOR LOW PRESSURE SWITCH TO [illegible]

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1983

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-83-20	183936	6/07/1983	7/01/1983	D	HG	OO,Z, Q		B	AI,AH,AO	D	N	CRACKED WELD ON WELDOLET THAT SUPPORTS A RELIEF VALVE IN THE CLEANUP SYSTEM
LER-83-21	183937	5/17/1983	6/27/1983	B	HA,IA	NN,OO	F	C	BI,EI,OE	G	C7	2 TIME DELAY RELAYS FAIL LONG DUE TO IMPROPER CALIBRATION - 5 EVENTS SINCE 1978
LER-83-22	184080	6/03/1983	7/11/1983	B	HA,IA	NN,OO	F	C	BI,EI,OE	G	C7	TIME DELAY RELAY FAILS LONG DUE TO IMPROPER CALIBRATION
LER-83-23	184570	6/16/1983	7/19/1983	B	CC	OO		B	OC	A	N	2 ACOUSTICAL VALVE MONITOR TESTS NOT PERFORMED DUE TO RECORDS ERROR
LER-83-24	184571	6/12/1983	7/19/1983	B	EE	I,X,OO,NN		B	AA,OA	D	N	GAS TURBINE GENERATOR WAS DECLARED INOPERABLE DUE TO A FAILED E&R OIL PUMP MOTOR
LER-83-25	185304	7/12/1983	8/25/1983	B	EE	S,I,NN		C	BD, G,OA, OI	G	N	GAS TURBINE GENERATOR IS DECLARED INOPERABLE DUE TO BLOWN FUSE DURING CALIBRATION
LER-83-26	185712	8/16/1983	9/23/1983	B	EE	C,I,NN		C	AC,BE,BI	D	C7	GAS TURBINE GENERATOR SHUT DOWN DUE TO AN OIL IMPREGNATED CABLE
LER-83-27	186134	8/29/1983	10/25/1983	B	CE	H	F	C	IH	D	N	SET POINT DRIFT CAUSES TIME DELAY RELAY FOR ISOLATION CONDENSER TO FAIL LONG
LER-83-28	186690	9/30/1983	11/03/1983	B	BB		N,C	B	BE,ED	D	N	1 CHANNEL OF OFF-GAS RADIATION MONITOR IS DECLARED INOPERABLE DUE TO SHORTED RESISTOR
LER-83-29	187376	11/08/1983	12/01/1983	B	IA,SA		M,T	B	FH	D	N	SET POINT DRIFT CAUSES 2 DRYWELL HIGH-PRESSURE SWITCHES TO TRIP TOO HIGH

TABLE A.22 DATA TABLES FOR REPORTABLE EVENTS FOR MILLSTONE 1-1983

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-83-30	187718	8/16/1983	12/13/1983	B	SA,CC	OO		B	AU,OA	D	N	MAIN STEAM DRAIN VALVE IS DECLARED INOPERABLE DUE TO STEM PACKING LEAK
LER-83-31	187719	11/04/1983	12/13/1983	B	RB,ID		N,T	B	AG,OA	D	CB	1 CHANNEL OF APRM IS DECLARED INOPERABLE DUE TO A FAILED SWITCH
LER-83-32	187930	11/25/1983	1/04/1984	B	CB,SA	DD,FF		B	AA,AU	D	N	REACTOR COOLANT LEAK RATE EXCEEDS LIMIT DUE TO A FAILED SEAL ON RECIRCULATION PUMP
LER-83-33	187931	11/26/1983	1/04/1984	D	SH-D	V	P	B	AL,BD	E	N	1 TRAIN OF STANDBY GAS TREATMENT IS DECLARED INOPERABLE DUE TO LOOSE SCREW
LER-83-34	188117	12/05/1983	1/10/1984	B	EB	O,LL		B	AT,AL,OA	G	N	RESERVE STATION SERVICE TRANSFORMER HAS OIL LEAK DUE TO LOOSE BOLTS
LER-83-35	188118	12/09/1983	1/10/1984	B	EB	D,LL		B	AQ,OA	D	N	RESERVE STATION SERVICE TRANSFORMER HAS SALT CONTAMINATION OF BUSHINGS

TABLE A.22 DATA TABLES FOR [illegible]RTABLE EVENTS FOR MILLSTONE 1-1984

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-84-3	190282	2/13/1984	3/16/1984	B	BA		N	B	OD		N	GAMMA SCAN OF 2 OYSTERS SHOWS HIGH LEVELS OF AG-110M
LER-84-5	189594	4/14/1984	5/08/1984	C	CD	OO		C	AT	D	N	COMBINED LEAK RATE OF MSIV'S IS HIGHER THAN MAXIMUM ALLOWED FOR SINGLE MSIV
LER-84-6	189650	4/18/1984	5/22/1984	C	IB		C	B	EF	G	N	REACTOR SCRAM DUE TO ELECTRONIC NOISE SPIKE CAUSED BY MAINTENANCE ACTIVITIES
LER-84-7	191540	4/09/1984	5/16/1984	B	BA		N	B	OD		N	ROUTINE GAMMA SCAN SHOWS HIGH LEVEL OF CO-60 IN AQUATIC FLORA
LER-84-8	190244	4/27/1984	5/30/1984	C	SF-D, CB,CG	Z		C	AO	E	C4	3 WELDS HAVE INTERGRANULAR STRESS CORROSION CRACKING
LER-84-9	190245	4/25/1984	5/30/1984	C	ZZ	GG		C		D	N	HYDRAULIC SNUBBER FAILS BLEED RATE TENSION AND COMPRESSION TEST

TABLE A.22 DATA TABLES FOR [illegible]RTABLE EVENTS FOR MILLSTONE 1-1984

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENTS
LER-84-11	190466	5/27/1984	7/03/1984	D	AB.PA		T	C	BD.BK OK		C8	2 FIRE DETECTOR STRINGS IN CONDENSER BAY FAIL DUE TO LOW SUPERVISORY AIR PRESSURE
LER-84-12	190646	6/14/1984	7/20/1984	D	CC	FF.00		C	AG.BA OH	G	C7	SET POINT DRIFT CAUSES 3 SAFETY RELIEF VALVES TO FAIL TO OPEN
LER-84-14	190978	7/09/1984	8/07/1984	B	CE	H.X.00	K.T	C	BA.BC	G	C7	ISOLATION VALVE FOR ISOLATION CONDENSER FAILS CLOSED DUE TO LIMIT SWITCH
LER-84-17	191288	8/02/1984	8/29/1984	B	CB.IA	CG		B	AU	G	N	REACTOR SHUT DOWN DUE TO LEAK IN STOP VALVE ON RECIRCULATION RISER
LER-84-18	191140	8/03/1984	9/10/1984	D	CE	F.H.X OC	K.T	C	BA.BC BY	G	C7	ISOLATION VALVE FOR ISOLATION CONDENSER FAILS CLOSED - SECOND FAILURE IN ONE MONTH

Table B.16. Plant status, component status, and cause of reportable events

Code	Plant status	Component status	Cause of reportable event
A	Construction	Maintenance and repair	Administrative error
B	Operation	Operation	Design error
C	Refueling	Testing	Fabrication error
D	Shutdown		Inherent error
E			Installation error
F			Lightning
G			Maintenance error
H			Operation error
I			Weather

Table B.17. Systems involved with reportable event

System	Code
Reactor	RX
Reactor vessel internals	RA
Reactivity control systems	RB
Reactor core	RC
Reactor coolant and connected systems	CX
Reactor vessels and appurtenances	CA
Coolant recirculation systems and controls	CB
Main steam systems and controls	CC
Main steam isolation systems and controls	CD
Reactor core isolation cooling systems and controls	CE
Residual heat removal systems and controls	CF
Reactor coolant cleanup systems and controls	CG
Feedwater systems and controls	CH
Reactor coolant pressure boundary leakage detection systems	CI
Other coolant subsystems and their controls	CJ
Engineered safety features	SX
Reactor containment systems	SA
Containment heat removal systems and controls	SB
Containment air purification and cleanup systems and controls	SC
Containment isolation systems and controls	SD
Containment combustible control systems and controls	SE
Emergency core cooling systems and controls	SF
Core reflooding	SF-A
Low-pressure safety injection system and controls	SF-B
High-pressure safety injection system and controls	SF-C
Core spray system and controls	SF-D
Control room habitability systems and controls	SG
Other engineered safety feature systems and their controls	SH
Containment purge system and controls	SH-A
Containment spray system and controls	SH-B
Auxiliary feedwater system and controls	SH-C
Standby gas treatment systems and controls	SH-D
Instrumentation and controls	IX
Reactor trip systems	IA
Engineered safety feature instrument systems	IB
Systems required for safe shutdown	IC
Safety-related display instrumentation	ID
Other instrument systems required for safety	IE
Other instrument systems not required for safety	IF

Table B.18. Equipment and instruments involved in reportable events

Code		Code	
		Equipment	
A	Accumulator	W	Internal combustion engine
B	Air drier	X	Motor
C	Battery and charger	Y	Nozzle
D	Bearing	Z	Pipe and pipe fitting
E	Blower and dampers	AA	Power supply
F	Breaker	BB	Pressure vessel
G	Cables and connectors	CC	Pressurizer
H	Condenser	DD	Pump
I	Control rod	EE	Recombiner
J	Control rod drive	FF	Seal
K	Cooling tower	GG	Shock absorber
L	Crane	HH	Solenoid
M	Demineralizer	II	Steam generator
N	Diesel generator	JJ	Storage container
O	Fastener	KK	Support structure
P	Filter/screen	LL	Transformer
Q	Flange	MM	Tubing
R	Fuel element	NN	Turbine
S	Fuse	OO	Valve
T	Generator	PP	Valve, check
U	Heat exchanger	QQ	Valve operator
V	Heater	QQ	Valve operator
		Instrumentation	
A	Alarm	L	Power range instrument
B	Amplifier	M	Pressure sensor
C	Electronic function unit	N	Radiation monitor
D	Failed fuel detection instrument	O	Recorder
E	Flow sensor	P	Relay
F	In-core instrument	Q	Seismic instrument
G	Indicator	R	Solid state device
H	Intermediate range instrument	S	Start-up range instrument
I	Level sensor	T	Switch
J	Meteorological instrument	U	Temperature sensor
K	Position instrument		

Table B.19. Abnormal conditions of reportable events

	Mechanical
AA	Normal wear/aging/end of life: expected effect of normal usage
AB	Excessive wear/clearance: component (especially a moving component) experiences excessive wear or too much clearance or gap exists because of overuse, lack of lubrication
AC	Deterioration/damage: component is no longer at an acceptable level of quality (e.g., high temperature causes rubber seals to chemically break down or deteriorate, insulation breaks down)
AD	Break/shear: structural component physically breaks apart (not when something "breaks down")
AE	Warp/bend/deformation: shape of component is physically distorted
AF	Collapse: tank or compartment has an external pressure exerted that results in deformation
AG	Seize/bind/jam: component has inhibited movement caused by crud, foreign material, mechanical bonding, another component
AH	Excessive mechanical loads: mechanical load exceeds design limits
AI	Mechanical fatigue: failure due to repeated stress
AJ	Impact: the result of the force of one object striking another
AK	Improper lubrication: insufficient or incorrect lubrication
AL	Missing/loose: component is missing from its proper place or is loose or has undesired free movement
AM	Wrong part: incorrect component installed in a piece of equipment
AN	Wrong material: incorrect material used during fabrication or installation
AO	Weld-related failure: failure caused by defective weld or located in the heat-affected zone
AP	Vibration other than flow induced: vibration from any cause other than fluid flow
AQ	Crud buildup: buildup of foreign material such as dust, sticks, trash (not corrosion or boron precipitation)
AR	Corrosion/oxidation: unanticipated attack
AS	Dropped: component is dropped (includes control rod that is "dropped" into core)
AT	Leak, internal, within system: leak from one part of a system to another part of the same system
AU	Leak, internal, between systems: leak from one system to a different system
AV	Crack: defect in a component does not result in a leak through the wall
AW	Leak, external: defect in a component results in a leak from the system that is contained in an onsite building
AX	Leak to environment: leak not resulting from a cracked or broken component
AY	Was opened/transfers open: component is/was opened by error or spuriously opens
AZ	Was closed/transferred closed: component is/was wrongly closed by error or spuriously closes

Table B.19 (continued)

BT	Low level/volume: lower than normal or desired level or volume exists in a component (not for instrument misindication)
BU	Abnormal concentration/pH: an abnormal (either high or low) concentration of a chemical or reagent exists in a fluid system or an abnormal pH exists (does not include abnormal boron concentration)
BV	Abnormal boron concentration: process system control rod has an abnormal boron concentration from burnup, dilution, or overaddition
BW	Overspeed: speed in ecess of design limits
BX	Cladding failure: cladding of a component fails (e.g., the cladding of a fuel pellet is breached, and radioactive fuel leaks out)
BY	Burning/smoking: component is on fire or smoking
BZ	Engaged: component engages or meshes (this is not to be used when a component binds or becomes stuck or jammed)
CA	Disengaged/uncoupled: component disengages, loses required friction, or is no longer meshed (as in gears), for example, the clutch on the motor disengages from the shaft (this should not be used for dropped control rods)
	Electric/instruments
EA	Excessive electrical loads: electrical loads exceed design rating
EB	Overvoltage/undercurrent: component failure produces an overvoltage/undercurrent condition other than open circuits
EC	Undervoltage/overcurrent: component failure produces an undervoltage/overcurrent condition other than shorts
ED	Short circuit/arcing/low impedance: electrical component shorts or arcs in the circuit or has a low impedance including shorts to ground
EE	Open circuit/high impedance/bad electrical contact: electrical component has a structural break, or electrical contacts fail to contact and fail to pass the desired current
EF	Erratic operation: component (especially electrical or instrument) behaves erratically or inconsistently (if an instrument produces a bad but constant signal, use "EG", if an instrument produces an inconsistent signal use "EF")
EG	Erroneous/no signal: electrical component or instrument produces an erroneous signal or gives no signal at all (not for out-of-calibration error)
EH	Drift: a change in a setting caused by aging or change of physical characteristics (does not include personnel errors or a physical shift of a component)
EI	Out of calibration: component (particularly instruments) become out of adjustment or calibration (does not include drift)
EJ	Electromagnetic interference: abnormal indication or action resulting from unanticipated electromagnetic field
EK	Instrument snubbing: dampening of pulsating signals to an instrument

Table B.19 (continued)

	<u>Hydraulic</u>
HA	High flow: higher than normal or desired flow exists in a component/system (does not include instrument misindication) (see code EG)
HB	Low flow: lower than normal or desired flow exists in a component/system (does not include instrument misindication)
HC	No flow or impulse: fluid flowing through a pipe, filter, orifice, or trench or the fluid in an impulse line (e.g., instrument sensing line) is blocked completely or decreased due to some foreign material, crud, closed (either partially or completely) valve or damper, or insufficient flow area
HD	Flow induced vibration
HE	Cavitation
HF	Erosion
HG	Vortex formation
HH	Water hammer
HI	Pressure pulse/surge
HJ	Air/steam binding
HK	Loss of pump section
HL	Boron precipitation
	<u>Other</u>
OA	Declared inoperable: component or system is declared inoperable as required by Technical Specifications but may be capable of partially or completely performing its desired duties when requested (a component/system that is <u>completely</u> failed should not use this code)
OB	Flux anomaly: flux characteristics of the reactor core are not as required or desired (e.g., flux spike due to xenon burnout)
OC	Test not performed: operator or test personnel fails to perform a required test within the required period
OD	Radioactivity contamination: component, system, or area becomes more radioactive than desired or expected
OE	Temporary modification: an installation intended for short term use (usually this is for maintenance or modification of installed equipment)
OF	Environmental anomaly
OG	Airborne release
OH	Waterborne release
OI	Operator communication
OJ	Operator incorrect action
OK	Procedure or record error

Table B.19 (continued)

BA	Fails to open: component is in the closed state and fails to open on demand (e.g., the circuit breaker "fails to open" when an overcurrent occurs)
BB	Fails to close: component is in the open state and fails to close on demand
BC	Malposition or maladjustment: component is out of desired position (e.g., normally open valve is closed) or adjusted improperly (not for instrument drift or out of calibration)
BD	Failure to start/turn on: component fails to start on demand
BE	Stopped/failed to continue to run: component fails to continue running when it has previously started
BF	Tripped: component automatically trips on or off (desired or undesired) (e.g., the turbine tripped because of overspeed, the circuit breaker tripped because of overspeed, or the circuit breaker tripped because of overload)
BG	Deenergized/power removed: component on system loses its driving potential but not necessarily electrical power [e.g., (1) a fuse blows and there is no power to a sensor, and the sensor is deenergized; (2) a valve closes off the steam supply to a turbine, and the turbine has no driving power]
BH	Energized/power applied: component or system gains its driving potential but not necessarily electrical power (e.g., valve is opened allowing steam to turn a turbine)
BI	Unacceptable response time: component does not respond to a demand within a desired time frame but does not otherwise fail (e.g., a diesel generator fails to come to full speed within the time constraint)
BJ	High pressure: higher than normal or desired pressure exists in a component or system (does not include instrument misindications)
BK	Low pressure: lower than normal or desired pressure exists in a component or system (does not include instrument misindication)
BL	High temperature: component experiences a higher than normal or desired temperature
BM	Low temperature: component (or system) experiences a lower than normal or desired temperature
BN	Freezing: fluid medium (e.g., water) freezes in or on a component
BO	Excessive thermal cycling: frequent changes in temperature that could result in metal fatigue or cracking
BP	Unacceptable heatup/cooldown rate: heatup or cooldown rate exceeds limits
BQ	Thermal transient: system experiences an undesired or unstable thermal transient or thermal change
BR	Excessive number of pressure cycles: system experiences an undesired number of significant pressure changes (e.g., pressure pulses as from a positive displacement pump)
BS	High level/volume: higher than normal or desired level or volume exists (actual or potential) in a component, such as tank or sump, or area, such as auxiliary building (not for instrument misindication)

B.20. Reportable event criteria - significants

Category of significance	Event description
S1	Two or more failures occur in redundant systems during the same event
S2	Two or more failures due to a common cause occur during the same event
S3	Three or more failuresoccur during the same event
S4	Component failures occur that would have easily escaped detection by testing or examination
S5	An event proceeds in a way significantly different from what would be expected
S6	An event or operating condition occurs that is not enveloped by the plant design bases
S7	An event occurs that could have been a greater threat to plant safety with (1) different plant conditions, (2) the advent of another credible occurrence, or (3) a different progression of occurrences
S8	Administrative, procedural, or operational errors are committed that resulted from a fundamental misunderstanding of plant performance or safety requirements
S9	Other (explain)

Table B.21. Reportable event criteria - conditionally significant

Category of conditional significance	Event description
C1	A single failure occurs in a nonredundant system
C2	Two apparently unrelated failures occur during the same event
C3	A problem results in an offsite radiation release or exposure to personnel
C4	A design or manufacturing deficiency is identified as the cause of a failure or potential failure
C5	An problem results in a long outage or major equipment damage
C6	An engineering safety feature actuation occurs during an event
C7	A particular occurrence is recognized as having a significant recurrence rate
C8	Other (explain)