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Nuclear Safety Information Center

Engineering Technology Division

# REVIEW OF THE OPERATING EXPERIENCE HISTORY OF MILLSTONE 1 THROUGH 1980 FOR THE NUCLEAR REGULATORY COMMISSION'S SYSTEMATIC EVALUATION PROGRAM

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# FOREWORD

The Systematic Evaluation Program Branch (SEPB) of the Nuclear Regulatory Commission (NRC) is responsible for the conduct of the Systematic Evaluation Program (SEP) whose purpose is to determine the safety margins of the design and operation of the 11 oldest operating commercial nuclear power plants in the United States. These 11 plants are being reevaluated in terms of present NRC licensing requirements and regulations. In addition, SEP must:

1. establish documentation that shows how these operating plants compare with current acceptance criteria and guidelines on significant safety issues and provide a technical rationale for acceptable departures from these criteria and guidelines,
2. provide the capability for making integrated and balanced decisions with respect to any required backfitting, and
3. provide for the early identification and resolution of any potential safety deficiency.

The SEP is evaluating specific safety topics (called the Topic List) based on an integrated review of the overall ability of a plant to respond to certain design-basis events (DBEs), including normal operation, transients, and postulated accidents. The evaluation will result in a reassessment of the overall safety margins for each facility and documentation of the reassessment on the basis of current criteria.

The review approach with respect to operational events (forced shutdowns and reportable occurrences) consists primarily of a three-step process: (1) compilation of information on the events, (2) screening of events for significance using selected criteria and guidelines, and (3) evaluation of significance and importance of the events from a safety standpoint. Trends in equipment failures and events where systems failed to perform their intended function are identified. Other types of operating information as noted in Sect. 1 are compiled to provide an overall view of the operating histories of the plants.

In this report, the operating experience of the Millstone 1 nuclear power plant is reviewed for the purpose of compiling and interpreting data on plant operational occurrences and events for application and input to the SEP. The results of this report will be used by SEPB in performing the integrated assessment of overall plant safety for Oyster Creek.

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## ABSTRACT

A review of the operating experience of the Millstone 1 nuclear power plant from initial criticality through 1980 was performed by the staff of the Nuclear Safety Information Center for the Nuclear Regulatory Commission's Systematic Evaluation Program (SEP). Under the SEP, the safety margins of the design and operation of the 11 oldest operating commercial nuclear power plants in the United States are being reevaluated.

The review of the operating experience for Millstone 1 included data collection and evaluation of availability and capacity factors, forced shutdowns, power reductions, reportable events (reportable occurrence, licensee event reports, etc.), and environmental considerations. As well, the review methodology and procedures as used in the review and evaluation are discussed. Data and information collected for forced shutdowns, power reductions, and reportable events are presented in Appendixes.

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# 1. SCOPE OF REVIEW

The assessment of the operating experience review for Millstone 1 covered the time from initial criticality through 1980. The review included the following aspects of operation: availability and capacity factors, forced shutdowns and power reductions, reportable events, events of environmental importance and radioactivity releases, and evaluation of the operating experience in total. Tables in Chap. 1 show the codes assigned to operational aspects of forced shutdowns, power reductions, and reportable events. These codes are used in the reporting of data collected during the review of operating experience.

## 1.1 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 =

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

2. unit availability =

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

3. $\text{unit capacity (DER} = \dfrac{\text{net electrical energy generated}}{\text{period hours x DER net}} \times 100$ ,

4. $\text{unit capacity (MDC)} = \dfrac{\text{net electrical energy generated}}{\text{period hours x MDC net}} \times 100$ .

### 1.2 Review of Forced Shutdowns and Power Reductions

Forced shutdowns and power reductions were reviewed, and data were collected on each incident. Scheduled shutdowns for refueling and maintenance were not included in the review. However, if a utility had a refueling outage scheduled, the plant experienced a shutdown as a result of an abnormal event prior to the scheduled refueling, the utility reported that the refueling was being rescheduled to coincide with the current shutdown, and the utility reported the cause of the shutdown as refueling, then this shutdown was considered as forced. Only that portion of the outage time concerned with the abnormal event, not the refueling time, was included in the compilations.

The power reductions were included to provide information and details that may have been associated with a previous or subsequent shutdown. The power reductions are included in the proper chronological sequence with the shutdowns in the data tables for the forced shutdowns and power reductions (see Appendixes).

The following data were compiled annually for the forced shutdowns and power reductions:

Table 1.1. Codes and causes of forced shutdown or power reduction 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 1.2. Codes and systems involved with the forced shutdown, power reduction, or 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 system | 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 |
| 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 |

Table 1.2 (continued)

| System | Code |
|---|---|
| Fuel storage and 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 |
| 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 |

Table 1.2 (continued)

| System | Code |
|---|---|
| Radiation protection systems | BX |
| Area monitoring systems | BA |
| Airborne radioactivity monitoring systems | BB |

Table 1.3. Components involved with the forced shutdown or power reduction

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

Table 1.3 (continued)

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

1. date of occurrence,
2. duration (hours),
3. power level (percent),
4. notation of whether the shutdowns were also reportable events [e.g., a licensee event report (LER) or abnormal occurrence report (AOR)],
5. summary description of events associated with the forced shutdown or power reduction,
6. cause of shutdown (Table 1.1),
7. method of shutdown (Table 1.1),
8. system taken from NUREG-0161 (Ref. 1) that was directly involved with the shutdown or power reduction (Table 1.2),
9. component directly involved with the shutdown or power reduction (Table 1.3), and
10. categorization of the shutdown or power reduction.

Each shutdown or power reduction was placed in one of two sets of significance categories. The shutdowns and power reductions were first evaluated against criteria for DBEs as described in Chap. 15 of the *Standard Review Plan*.[2] If the shutdown or power reduction could not be categorized as a design-basis initiating event, then it was placed in one of a series of Nuclear Safety Information Center (NSIC) categories. For further discussions of the two sets of significance categories, use of the categories, and a listing of them, see Sect. 3.1.

The listings for the cause, shutdown method, system involved, and component involved along with their respective codes are those used in the NUREG-0020 series ("Gray Books") on shutdowns. Note that the information listed under the "System involved" column in the data tables in the appendixes indicates (1) a general classification of systems (fully written out) and (2) a specific system, which is coded with two letters, within the general classification.

## 1.3 Review of Reportable Events

The operating events as reported in LERs and LER predecessors [e.g., AORs, unusual events reports, reportable occurrences (ROs)] were reviewed. These types of reportable events were retrieved from the NSIC computer file. Approximately five years ago, operating experience information for operating nuclear power plants in the NSIC file for the period of time before LERs was reviewed. Any documents that contained LER-type information (such as equipment failures or abnormal events) were coded or indexed so that they could be retrieved in the same manner as an LER. Primarily, this involved various types of operating reports and general correspondence for the late 1960s and early 1970s.

The following information was recorded 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 NSIC 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 1.4),
6. system involved with the reportable event (Table 1.2),

7. type of equipment involved with the reportable event (Table 1.5),
8. type of instrument involved with the reportable event (Table 1.5),
9. status of the component (equipment) at the time of the occurrence (Table 1.4),
10. abnormal condition associated with the reportable event (e.g., corrosion, vibration, leak) (Table 1.6),
11. cause of the reportable event (Table 1.4), and
12. significance of the reportable event.

As a step in the evaluation process, each reportable event was screened using the criteria further discussed in Sect. 3.2.

Note that in the tables of reportable events in Appendix A for Oyster Creek, comments and/or details on the events were included.

## 1.4 Events of Environmental Importance and Releases of Radioactivity

Any significant or recurring environmental problems were summarized based on the review of forced shutdowns, power reductions, reportable events (environmental LERs), and operating reports. Routine radioactivity releases were tabulated as well, and releases where limits were exceeded were reviewed and are discussed in Sect. 4.6.

## 1.5 Evaluation of Operating Experience

The operating history of the plant was evaluated based on a review that involved screening, categorizing, and compiling data. Judgments and conclusions were made regarding safety problems, operations, trends (recurring problems), or potential safety concerns.

Table 1.4. Codes for data collected on plant status, component status, and cause of reportable events

| Code | Plant status | Component status | Cause of reportable event |
|---|---|---|---|
| A | Construction | Maintenance and repair | Adminsistrative 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 1.5. Codes for 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 | | |
| | **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 1.6. Codes used for reportable events abnormal conditions

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 |

Table 1.6 (continued)

| | |
|---|---|
| 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 |
| BA | Fails to open: component is in the closed state <u>and</u> 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 <u>and</u> 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 <u>automatically</u> 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 (<u>does</u> <u>not</u> include instrument misindications) |

Table 1.6 (continued)

| | |
|---|---|
| BK | Low pressure: lower than normal or desired pressure exists in a component or system (<u>does</u> <u>not</u> 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 builidng (not for instrument misindication) |
| 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 concentrations) |
| BV | Abnormal boron concentration: process system control rod has an abnormal boron concentration from burnup, dilution, or overaddition |
| BW | Overspeed: speed in excess 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) |

Table 1.6 (continued)

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

Hydraulic

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

Table 1.6 (continued)

| | |
|---|---|
| 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 |
| | Other |
| 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 completely 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 |

From the information provided through the various operating reports and the review process, events were analyzed to determine their safety significance, using the final safety analysis reports to provide specific plant and equipment details when necessary.

# 2. SOURCES OF INFORMATION USED IN THE REVIEW

Several sources of information including periodic (annual, quarterly, and monthly) NRC publications were used in the review. Some sources contained information relative to more than one area within the scope of the review.

## 2.1 Availability and Capacity Factors

The availability and capacity factors were either extracted or calculated from data given in the Gray Books[3] from 1974 through 1980 (the first Gray Book was issued in May 1974). Prior to 1974, annual or semiannual reports were used to compile availability factors only.

## 2.2 Forced Reactor Shutdowns and Power Reductions

Review of the forced power reductions involved checking the following sources for accuracy and completeness of details.

1. *Nuclear Power Plant Operating Experience for 19XX*, for the years 1973-1979 (Refs. 4-10). The report for 1980 has not been published. However, because work on the section on outages in these reports has been performed by NSIC since 1973, the draft copy of this report for 1980 was available.
2. NUREG-0020 series[3] (Gray Books).
3. Annual or semiannual reports of individual plants from the time of startup through 1977. For 1977 through 1980, monthly operating reports were used because the utilities were no longer required to file annual reports. The review of power reductions involved primarily the annuals, semiannuals, and monthly reports.

## 2.3 Reportable Events

The NSIC computer file of LERs was the primary source of information in reviewing reportable events. Material on the NSIC computer file consists of the appropriate bibliographic material, title, 100-word abstract, and keywords. 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 NSIC (for the years 1976–1980); (2) the microfiche file of docket material at NSIC; or (3) the appropriate operating report (semiannual, annual, or monthly).

## 2.4 Environmental Events and Releases of Radioactivity

Events of environmental importance were obtained as a result of conducting the overall review of the plant's operating history, and the sources of information involve all types of documents listed thus far.

The data for radioactivity releases were compiled primarily from *Radioactive Materials Released from Nuclear Power Plants — Annual Report 1977* (Ref. 11). This report presents year-by-year comparisons for plants in a number of different categories (such as solid, gas, liquid, noble gas, and tritium). Data for 1978 were taken from *Radioactive Materials Released from Nuclear Power Plants — Annual Report 1978* (Ref. 12). Data for 1979 and 1980 were compiled from the annual environmental reports submitted by the licensees.

## 2.5 Use of Computer Files on RECON and Special Publications

Two computer files on RECON (a computer retrieval system containing ~40 data bases operated at ORNL) were used extensively for another purpose in addition to those indicated thus far. Printouts were obtained from the files for Millstone 1 to provide coverage on 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[13];
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 bulletions, IE circulars, and IE information notices.

# 3. CRITERIA AND CATEGORIZATION FOR EVALUATIONS OF OPERATING HISTORY

Forced shutdowns (and power reductions) and reportable events were the two areas focused on in the review of the operating history of the plants of interest. Given the large number of both forced shutdowns and reportable events, it was necessary to develop consistent review procedures that involved screening and categorizing of both occurrences. After the events were screened and categorized, the study then assessed the safety significance of the events and analyzed the categories of events for various trends and recurring problems.

Shutdowns were evaluated against the DBEs found in Chap. 15 of the *Standard Review Plan*.[2] The DBEs are those postulated disturbances in process variables or postulated malfunctions or failures of equipment that the plants are designed to withstand and that licensees are expected to analyze and include in safety analysis reports (SARs). The SAR provides the opportunity for the effects of anticipated process disturbances and postulated component failures to be examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures and situations (or to identify the limitations of expected performance).

The intent is to organize the transients and accidents considered by the licensee and presented in the SAR in a manner that will:

1. ensure that a sufficiently broad spectrum of initiating events has been considered,
2. categorize the initiating events by type and expected frequency of occurrence so that only the limiting cases in each group need to be quantitatively analyzed, and

3. permit the consistent application of specific acceptance criteria for each postulated initiating event.

Each postulated initiating event is to be assigned to one of the following categories:

1. increase in heat removal by the secondary system (turbine plant),
2. decrease in heat removal by the secondary system (turbine plant),
3. decrease in reactor coolant system flow rate,
4. anomalies in reactivity and power distribution,
5. increase in reactor coolant inventory,
6. decrease in reactor coolant inventory,
7. radioactive release from a subsystem or component, or
8. anticipated transients without scram.

Typical initiating events that are representative of those to be considered by the licensee in the SAR are presented in Table 3.1.

Those shutdowns identified as design-basis initiating events were categorized as such. If the shutdown was not a DBE, then it was assigned a category from a list developed by NSIC to indicate the nature and type of error or failure. The NSIC categories for shutdowns not caused by DBEs were examined as part of a trends analysis.

Reportable events were screened using the criteria presented in Sect. 3.2 and were categorized according to their significance. The information collected on the reportable events (as outlined in Tables 1.2 and 1.4–1.6) was used to analyze trends for all reportable events, both significant and not significant.

The review approach with respect to operational events (forced shutdowns and reportable occurrences) consisted primarily of a three-step process: (1) compilation of information on the events, (2) screening of the events for significance using selected criteria and guidelines, and (3) evaluation of the significance and importance of the events from a safety standpoint. The evaluations were to determine those areas where safety problems existed in terms of systems, equipment, procedures, and human error.

## 3.1 Significant Shutdowns and Power Reductions

For the purposes of compiling information and evaluation, power reductions were treated in the same manner as forced shutdowns.

### 3.1.1 Criteria for significant shutdowns and power reductions

As indicated previously, the occurrences identified as DBEs were used as criteria to categorize and note significant shutdowns. These events are listed in Table 3.1 as they are found in Chap. 15 of the *Standard Review Plan*.[2]

### 3.1.2 Use of criteria for determining significant shutdowns and power reductions

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

Table 3.1. Initiating event descriptions for DBEs as listed in Chap. 15, *Standard Review Plan* (Revision 3)

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
   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

Table 3.1 (continued)

- 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 or relief 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

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.

with the shutdown. For example, if the reactor shuts down because of an increase in heat removal because a feedwater regulator valve failed open, the shutdown is a generic type 1 DBE. Specifically, based on the initiating event (valve failed open), it is 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 tripping of a main coolant pump, 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."

### 3.1.3 Non-DBE shutdown and power reduction categorization

Those shutdowns that were not DBEs were assigned NSIC categories (Table 3.2) 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.

The causes (Table 1.1) for non-DBE shutdowns taken from the Gray Books are limited and very general, while NSIC cause categories are more specific. Thus, as an example, the number of Gray Book causes noted as equipment failure should not be expected to equal those identified as equipment failures with the NSIC categories. Other NSIC categories, such

Table 3.2. NSIC 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)
- N 8.0 Regulatory restriction
  - N 8.1 Notice of generic event
  - N 8.2 Notice of violation
  - N 8.3 Backfit/reanalysis

Table 3.2 (continued)

| | |
|---|---|
| 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 |

as component failure, could be classified as an equipment failure if the only available designations for cause were those listed in the Gray Books.

## 3.2 Significant Reportable Events

### 3.2.1 Criteria for significant reportable events

Two groups of criteria were used in determining significant reportable events. The first set of criteria (Table 3.3) indicates those events that are definitely significant in terms of safety; they are termed significant. The second set of criteria (Table 3.4) indicates events that may be of potential concern. These events, which might require additional information or evaluation to determine their full implication, were noted as conditionally significant.

### 3.2.2 Use of criteria for determining significant reportable events

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.

Table 3.3. Reportable event criteria – significant

| 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 failures occur 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 3.4. 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 | A 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) |

Thus, no events in the tables in the Appendix appear as having been categorized as conditionally significant. The final evaluation for significance considered whether a DBE was initiated or whether a safety function was compromised so 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.

### 3.2.3 Reportable events that were not significant

Those 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 appendixes). 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).

# 4. OPERATING EXPERIENCE REVIEW OF MILLSTONE-1

## 4.1 Summary of Operational Events of Safety Importance

The operational history of Millstone-1 has been reviewed to indicate those areas of plant operation that compromised plant safety. The review included a detailed examination of plant shutdowns, power reductions, reportable events, and events of special environmental importance. The criteria used to show degradations in plant safety were (1) events that initiated a DBE and (2) events that compromised safety functions designed to mitigate the propagation of the initiating events.

Shutdowns and power reductions indicated the number and types of DBE's entered. The reportable events and special environmental events indicated the number of times each engineered safety function was compromised. The results of the analyses identified 53 DBE's entered. Additionally, two events were identified in which a loss of safety system function occurred in some engineered safety features.

## 4.2 General Plant Description

The Millstone Nuclear Power Station Unit 1 is a General Electric boiling water reactor (BWR) of 652 MWe net maximum dependable capacity, owned by Northeast Nuclear Energy Company and located in Waterford, Connecticut. The Architect/Engineer was Ebasco Services Incorporated, and the constructor was the General Electric Company. The condenser cooling method is once-through, and Long Island Sound is the condenser cooling water source. 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 generation of power began in March 1971.

The nearest city is New London, Connecticut, 3.2 miles away. The population within 6 miles is about 67,000, increasing in the summer to 83,000. The population within 20 miles is estimated at 330,000.

## 4.3 Availability and Capacity Factors

Table 4.1 contains the Millstone-1 availability and capacity factors. The reactor availability from 1971 through 1980 stayed above 70% except for one year, 1973, when major outages were necessary for repairs (see Sect. 4.4). The ten full years of operation, 1971 to 1980, averaged 79.5% reactor availability and 75.7% plant availability. Capacity factors were not available prior to 1973. The MDC and DER capacity factors from 1973 through 1980 averaged 66.2 and 65.3%, respectively.

## 4.4 Forced Reactor Shutdown and Power Reductions

### 4.4.1 Review of reactor shutdowns and power reductions

Table A1.1 through A1.11 in the Appendix provides a comprehensive summary of information concerning forced shutdowns and power reductions at Millstone-1. More complete information was provided when events generated reportable events; in such instances, more detailed descriptions are in Sect. 4.5.

Tables 4.2 and 4.3 of forced shutdowns and power reductions summarize Table A.1. Causes of forced shutdowns, item I.3 in Table 4.2 and item I.2 in Table 4.3, are dominated at Millstone-1 by equipment failures. Shutdowns reported to be caused by operator errors amount to only five of the total, and no power reductions were attributed to operator error.

4.4.1.1 Yearly summaries for Millstone-1 . A discussion of shutdowns and power reductions for each year, 1970 through 1980 follows.

Table 4.1. Availability and capacity factors for Millstone-1

| | 1970 | 1971 | 1972 | 1973 | 1974 | 1975 | 1976 | 1977 | 1978 | 1979 | 1980 |
|---|---|---|---|---|---|---|---|---|---|---|---|
| Reactor availability | ND | 74.1 | 89.2 | 47.3 | 80.9 | 79.0 | 84.0 | 96.0 | 89.0 | 79.1 | 76.0 |
| Unit availability | ND | 69.3 | 88.1 | 45.5 | 79.1 | 75.6 | 76.1 | 89.6 | 87.6 | 77.3 | 69.0 |
| Unit capacity (MDC)[a] | ND | ND | ND | 33.8 | 63.1 | 68.4 | 66.1 | 84.1 | 81.3 | 73.7 | 59.0 |
| Unit capacity (DER)[c] | ND | ND | ND | 33.2 | 59.6 | 68.4 | 65.6 | 83.4 | 80.5 | 73.0 | 58.5 |

[a]MDC = maximum dependable capacity.

[b]ND = No data.

[c]DER = design electrical rating.

Table 4.2. Forced shutdown summary for Millstone-1

| | 1970 | 1971 | 1972 | 1973 | 1974 | 1975 | 1976 | 1977 | 1978 | 1979 | 1980 | Total |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| I. Forced shutdowns | | | | | | | | | | | | |
| 1. Total number | 8 | 42 | 8 | 20 | 7 | 11 | 8 | 14 | 6 | 7 | 1 | 132 |
| 2. Total hours down | 102 | 1336 | 585 | 2892 | 91 | 976 | 624 | 738 | 197 | 424 | 13 | 7978 |
| 3. Cause[a] | | | | | | | | | | | | |
| A. Equipment failure | | 37 (1207) | 8 (585) | 16 (2594) | 7 (91) | 8 (502) | 7 (519) | 14 (738) | 5 (97) | 7 (424) | 1 (13) | 110 (67 |
| B. Maintenance or testing | 8 (102) | 3 (121) | | | | 1 (451) | | | 1 (100) | | | 13 (77 |
| D. Regulatory restriction | | | | 1 (236) | | | | | | | | 1 (23 |
| E. Operator training/license exam | | | | | | | | | | | | |
| F. Administrative | | | | | | | | | | | | |
| G. Operational error | | 2 (8) | | 3 (62) | | | | | | | | 5 (70 |
| H. Other | | | | | | 2(23) | 1(105) | | | | | |
| 4. Shutdown method | | | | | | | | | | | | |
| A. Manual | 3 | 11 | 4 | 12 | 2 | 7 | 5 | 2 | 2 | 2 | | 50 |
| B. Manual scram | | | | | | 1 | | 4 | | 1 | | 6 |
| C. Automatic scram | 5 | 31 | 4 | 8 | 5 | 3 | 2 | 8 | 3 | 3 | 1 | 73 |
| D. Other | | | | | | | 1 | | 1 | 1 | | 3 |
| II. Total number of DBE related shutdowns (these are included in totals of Part I) | 5 (53) | 25 (882) | 4 (218) | 4 (151) | 3 (23) | 0 (0) | 0 (0) | 2 (65) | 3 (39) | 3 (85) | 1 (13) | 1529 (50 |
| 1. Reactor vessels (CA) | | | | 6 | | | | | | | | 6 |
| 2. Coolant recirculation system (CB) | | 2 | | 4 | 1 | | | 1 | | | | 8 |
| 3. Main steam systems and controls (CC) | 1 | 8 | 3 | 3 | 2 | 5 | | 1 | 2 | 1 | 1 | 27 |
| 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 | 3 | 2 | | | 1 | | 1 | | 9 |
| 8. Offsite power systems (EA) | 1 | | | | | | | | | | | 1 |
| 9. AC onsite power systems (EB) | | | | | | 3 | 2 | | | | | 5 |
| 10. Onsite power systems (ED) | | | | | | | | | 1 | | | 1 |
| 11. Emergency lighting systems (EF) | | | | | | 1 | | | | | | 1 |
| 12. Turbine generator and controls (HA) | 2 | 20 | 3 | | 1 | | 1 | 1 | 1 | 1 | | 30 |
| 13. Main steam supply system (HB) | | | | | | | | 5 | 1 | | | 6 |
| 14. Main condenser systems (HC) | 1 | 2 | | | | | | | | | | 3 |
| 15. Turbine bypass systems (HE) | 1 | | | | | | | | | | | 1 |
| 16. Circulating water systems (HF) | | 2 | | | | | | | | | | 2 |
| 17. Condensate and feedwater systems (HH) | | 1 | | | | | | | | | | 1 |
| 18. Steam generator blowdown systems (HI) | | 1 | | | | | | | | | | 1 |
| 19. Reactor trip systems (IA) | | 1 | | | | | | 1 | 1 | | | 3 |
| 20. Gaseous radwaste management system (MB) | | | | | | | | 1 | | | | 1 |
| 21. Compressed air systems (PA) | | | | | | | | 1 | | 1 | | 2 |
| 22. Reactivity controls systems (RB) | | | | 2 | | | | | | | | 2 |
| 23. Reactor core (RC) | | | | | | 1 | | | | | | 1 |
| 24. Reactor containment systems (SA) | | 1 | 1 | | 1 | | | | | | | 3 |
| 25. Emergency core cooling system (SF) | | | | | | | | | | 1 | | 1 |
| 26. Low pressure safety injection system (SF-B) | | | | | | 1 | | | | | | 1 |
| 27. Core spray system (SF-D) | | 2 | | | | | | | | | | 2 |
| 28. Station service water systems (WA) | | | | | | 1 | | | | | | 1 |

[a]Number of hours associated with cause of shutdown is in parentheses.

Table 4.3. Power reduction summary for Millstone-1

| | 1970 | 1971 | 1972 | 1973 | 1974 | 1975 | 1976 | 1977 | 1978 | 1979 | 1980 | Total |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| I. Power reductions | | | | | | | | | | | | |
| 1. Total number | | | | | 7 | 3 | 1 | 3 | 10 | 3 | 1 | 28 |
| 2. Cause | | | | | | | | | | | | |
| A. Equipment failure | | | | | 7 | 2 | | 1 | 10 | | 1 | 21 |
| B. Maintenance or testing | | | | | | 1 | | 2 | | 3 | | 6 |
| H. Other | | | | | | | 1 | | | | | 1 |
| 3. System involved | | | | | | | | | | | | |
| A. Turbine-generator (HA) | | | | | 3 | | | 1 | | | | 4 |
| B. Main condenser (HC) | | | | | 2 | | | 2 | 10 | 3 | | 17 |
| C. Reactor (RB) | | | | | | | 1 | | | | | 1 |
| D. Reactor (RC) | | | | | | 1 | | | | | | 1 |
| E. Electric power (EB) | | | | | 1 | 1 | | | | | | 2 |
| F. Reactor coolant (CC) | | | | | 1 | 1 | | | | | | 2 |
| G. Reactor coolant (CD) | | | | | | | | | | | 1 | 1 |
| 4. Total number of DBEs related power reductions (included in totals of Part 1) | | | | | 3 | | | | | | | 3 |

1970

The Millstone-1 BWR went critical on October 26, 1970. Maintenance and testing were the causes for all forced outages. The reactor coolant system was involved three times, and the steam and power system was involved three times.

Seven shutdowns took place during startup testing. These were split about evenly between instrumentation malfunctions and faulty equipment installations.

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 involved a momentary moisture separator drain tank high level indication which subsequently trips the turbine resulting in a reactor shutdown.

1971

In 1971, the reactor and control system proved its capacity to operate under the severe transient conditions: 20 automatic scrams from 100% power (652 MWe) occurred, and the plant responded as designed each time.

During this year the reactor experienced 42 forced shutdowns, the most in the 10 years of operation at Millstone-1 (Table 4.2). The shutdowns are 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 d to repair a turbine control valve. The next longest forced shutdown occurred on August 30, and lasted for 7 d. 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.

One incident at the beginning of 1971, only two months after the initial criticality, exemplified the problems in maintenance and testing that plagued Millstone-1 during its first full year of operation. 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. The malfunction of the steam turbine control valve necessitated a 2-d shutdown for valve testing. Six additional forced shutdowns occurred due to malfunctioning of this valve.

There were six shutdowns due to miscellaneous isntrumentation malfunctions, and two to repair leaks in the main steam line.

1972

The number of forced outages dropped to 8 in 1972 (Table A.1.3), with no reported power reductions. The total forced down time was only 585 h, fourth lowest in Millstone-1's history.

Two forced shutdowns in February lasted for 12 d. These were due to improper responses from the main steam line venturi differential pressure transmitters, and the sensing tubes were replaced. On March 12, a reactor scram occurred during plant heatup when a reactor feed pump was started at 10 in. level indication in the reactor vessel. Collapse of voids from the cold feedwater resulted in a level decrease and the resultant low level scram.

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

1973

This year saw the second most forced shutdowns in Millstone-1's history: 20 forced shutdowns for 2892 h total (Table A.1.4). Sixteen of these shutdowns were attributed to equipment failures. The longest forced shutdown occurred on April 18, and lasted for 89 d. At this time, all of the feed-water spargers were replaced, due to cracks.

There was once forced shutdown of 10 d 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, and some of these 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 the second least forced downtime in Millstone-1's history: seven forced shutdowns for 91 h total (Table A.1.4). A total of seven power reductions were made which were attributed to equipment failure. Sea water leakage in the main condenser was detected necessitating 2 power reductions to plug tubes.

A recurring problem of turbine trips accounted for 3 more power reductions. These were again attributed to high level indication in the moisture separator drain tank.

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 most forced downtime in its history. Of a total of 976 h forced out of service, 56% were caused by a transformer failure that caused a 44 d shutdown.

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.

Five instrumentation malfunctions were the next largest contributor to the total downtime.

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

This year there were 8 forced shutdowns: with two of these due to high winds depositing salt spray on the main transformer insulators, causing arcing and tripping the generator. At the same time the gas turbine speed control became inoperable, necessitating the replacement of the electronic governor.

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

<u>1977</u>

There were 14 forced shutdowns due to equipment failure for a total of 738 h.

In August, main condenser tube failures started to plague Millstone-1 again. Several power reductions were made to plug leaking tubes. On June 14, a mechanical pressure regulator malfunctioned, tripping the steam turbine and causing a five day shutdown. On December 13, the second of two hydrogen explosions which occurred that day in the off-gas system caused an 11 d shutdown.

<u>1978</u>

This year there were only 6 forced shutdowns for a total of 197 h. Starting in June, main condenser leakage plagued 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.

<u>1979</u>

Seven forced shutdowns occurred this year for a total of 424 h. The most significant one occurred on January 6 and lasted for 11 d. 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, resulting 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, necessitated putting the isolation condenser 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.

4.4.1.2 Systems involved. There were 30 forced shutdowns involving turbine-generator and controls, with 67% of these occurring in 1971. Twenty-seven forced shutdowns involved the main steam systems and controls. The next most involved systems were the feedwater systems and the coolant recirculation systems, with 9 and 8 forced shutdowns, respectively.

Of the 28 power reductions, 61% involved the main condenser system. Three of the power reductions were considered DBE events.

4.4.1.3 Causes of forced reactor shutdowns and forced power reductions. Of the 132 forced shutdowns, 83% were caused by equipment failures for a total of 6770 h. Maintenance and testing accounted for 10% of the shutdowns, for a total of 774 h. There were only 5 due to operational error for a total of 70 h.

Of the 28 power reductions 75% were caused by equipment failures. Maintenance and testing accounted for 25%.

4.4.1.4 Non-DBE shutdowns. Table 4.4 summarizes the NSIC categories assigned to non-DBE shutdowns. Only the major NSIC categories are listed in Table 4.5. 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 occurred throughout the life of the operations.

Table 4.4. NSIC primary category summary for non-DBE shutdowns for Millstone 1

| | 1970 | 1971 | 1972 | 1973 | 1974 | 1975 | 1976 | 1977 | 1978 | 1979 | 1980 | Total |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1. Equipment failures | 3 | 11 | 3 | 9 | 3 | 9 | 7 | 6 | 1 | 3 | 0 | 55 |
| 2. Instrumentation and controls anomalies | | 2 | | 5 | | 1 | | 6 | 1 | 1 | | 16 |
| 3. Non-DBE reductions in coolant inventory (leaks) | | 2 | 1 | | | | | | | | | 3 |
| 4. Fuel/cladding failure | | | | | | | | | | | | |
| 5. Maintenance error | | | | | | 1 | | | | | | 1 |
| 6. Operator error | | 2 | | 1 | | | | | | | | 3 |
| 7. Procedural/administrative error | | | | | | | | | | | | |
| 8. Regulatory restriction | | | | 1 | | | | | | | | 1 |
| 9. External events | | | | | | | 1 | | | | | 1 |
| 10. Environmental operating constraints Tech specs | | | | | 1 | | | | 1 | | | 2 |
| TOTAL | 3 | 17 | 4 | 16 | 4 | 11 | 8 | 12 | 3 | 4 | 0 | 82 |

4.4.2 DBE initiating events

Of the 160 total forced shutdowns and power reductions accumulated at Millstone-1, 53 fell into DBE initiating event categories as shown in Tables 4.2 and 4.3. None of these events initiated any sequence that led to any significant economic loss or safety hazard to the plant or the environs.

Note that the trend of total number of DBE's per year bears no correlation with other trends, such as plant performance as measured by total number number of shutdowns per year or total downtime per time (Table 4.5).

4.4.2.1 DBE Sect. 1 events - increased in heat removal. Six events (12%) are categorized into this section. Four of the six were due to instrumentation malfunctions. One was attributed to operator error.

4.4.2.1.1 D1.1 feedwater system malfunctions resulting in a decrease in feedwater flow. On May 25, 1971, an instrumentation malfunction caused the closing of the feedwater control valve.

4.4.2.1.2 D1.2 - feedwater system malfunctions that result in an increase in feedwater flow. On March 12, 1972 an IRM trip occurred due to cold water addition resulting in a collapse of voids. On August 10, 1973, a high reactor water level trip was attributed to operator error in starting a feedwater pump.

4.4.2.1.3 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, a scram was caused by a momentary main steam line high flow signal. On December 5, 1970, an electrical pressure regulator pressure control oscillation caused a spurious low level indication. On September 21,

Table 4.5. DBE initiating events at Millstone 1

| | DBE category | 1970 | 1971 | 1972 | 1973 | 1974 | 1975 | 1976 | 1977 | 1978 | 1979 | 1980 | Total |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Feedwater system malfunctions that result in a decrease in feedwater temperature | D-1.1 | | 1 | | | | | | | | | | 1 |
| Feedwater system malfunctions that result in an increase in feedwater flow | D-1.2 | | | 1 | 1 | | | | | | | | 2 |
| 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 | 2 | 18 | 3 | | 4 | | | | 3 | 2 | | 32 |
| 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 | | | | | | | 2 |
| BWR recirculation loop controller malfunctions that result in decreasing flow rate | D-3.2 | | 1 | | | | | | | | | | 1 |
| Control rod maloperation (system malfunction or operator error), including maloperation of part length control rods | D-4.3 | | | | 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 | | 5 | 25 | 4 | 4 | 6 | 0 | 0 | 2 | 3 | 3 | 1 | 53 |

1973, a fault in the EPR controls opened the turbine bypass valves, dropping the reactor pressure.

4.4.2.2 DBE Sect. 2 events - decrease in heat removal. Thirty-nine events are categorized into this section. Twenty-nine of these 39 (74%) were caused by turbine trips. Three of these were associated with steam pressure regulator malfunction, two with loss of external electric load, two with inadvertent closures of main steam isolation valves, two with feedwater system, and one with loss of condenser vacuum.

4.4.2.2.1 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 main steam isolation. On January 19, 1971, the turbine control valve closed, causing the reactor shutdown. On June 25, 1980, an electric pressure regulator malfunction induced an APRM scram.

4.4.2.2.2 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 causing loss of 383 kv line.

4.4.2.2.3 D2.3 - turbine trip (stop valve closure). Twenty-one of the thirty-two 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 for the next eight years - 14 occurring in 1971, 3 in 1974, and 3 in 1978.

Five turbine control valve malfunctions (5-27-71, 9-29-71, 10-3-71, 10-10-71, 2-4-72) caused turbine trips.

4.4.2.2.4 D2.4 - Inadvertent closure of main steam isolation valves. The two events (11-4-74, 4-7-77) in this category were attributed to mechanical failures in valve actuators.

4.4.2.2.5 D2.5 - Loss of condenser vacuum. While this classification of D2.5 deals with the complete loss of condenser vacuum, it was felt that the only event (8-30-71) which dealt with low condenser vacuum should be included in this category.

4.4.2.2.6 D2.7 - Loss of normal feedwater flow. Of the two events in this category, one (3-6-73) was attributed to operator error in failing to start a condensate booster pump in time. The other event (12-16-74) was due to a broken stem on the feedwater control valve.

4.4.2.3 DBE Sect. 4 events - reactivity and power distribution anomalies. On March 14, 1973, a blown fuse on the control rod scram solenoid scrammed the Group II control rods a 4.3 DBE.

4.4.2.4 DBE Sect. 6 - decrease in reactor coolant inventory. All three of the following shutdowns were 6.1 DBEs. On March 2, 1971, a main steam safety valve started leaking, blowing steam. On November 29, 1977, the automatic pressure relief valve lifted prematurely. On February 26, 1979, the safety relief valve lifted prematurely and failed to reseat.

### 4.4.3 Trends and safety implications of shutdowns and power reductions

Over the more than 10 years of operation at Millstone-1, forced downtime averaged 34 d/year. Annual downtown was dominated by single large events such as the 90-d shutdown in 1973 to replace the feedwater spargers. Total annual hours of forced downtime followed no trend. Years 1971, 1973, 1975, and 1977 each recorded more than 700 h of forced outage time; the other 6 years each recorded less than 700 h of forced outage time.

Four out of every five forced shutdowns were caused by equipment failures. There were two reported operator errors in 1970 that caused shutdowns and three in 1973. There were none reported thereafter.

All of the 82 forced shutdowns that did not fall into DBE categories were put in NSIC categories (see Sect. 3.1.1). In NSIC event categories, 55 shutdowns were due to equipment failures, such as broken piping welds, valve stem packing failures, and condenser leaks. The next major type of NSIC event was instrumentation and control anomalies, with 16 leading to shutdowns. There were no fuel/cladding failures requiring shutdowns, only 2 shutdowns were due to environmental operating constraints.

Forced reductions in reactor power occurred 28 times. Twenty-one of these were due to equipment failures, and 6 were to maintenance and testing. Forced power reductions were not reported until 1974, when 7 occurred. The causes were divided about equally between condenser leakage and instrumentation malfunction. After 1974, reductions were reported about 3 times a year. They usually lasted only a few hours, and power was seldom reduced more than 50%. In 1978, 11 reductions were reported due to main condenser leakage.

## 4.5 Reportable Events

This study reviewed 310 reportable events from Millstone-1. The events include telegrams, letters, abnormal occurrences (AO's), reportable occurrences (RO's) and licensee event reports (LER's) that were filed by the utility when a technical specification was violated.

The information contained in the reportable events has been synthesized and coded as discussed in Sect. 1.3. These tables, arranged by year, are presented in Appendix A, part 2.

4.5.1 Review of reportable events from 1970 through 1980

Figure 4.1 illustrates the number of reportable events filed per year at Millstone-1. There are no yearly trends on the numbers of reports. The following sections presented a summary for each year of operating experience at Millstone-1, omitting environmental reports which are discussed in 4.5.1.4.

4.5.1.1 Yearly summaries of reportable events.

1970

The plant went critical in October of 1970. As expected, several design and installation errors surfaced during the first few months of operation.

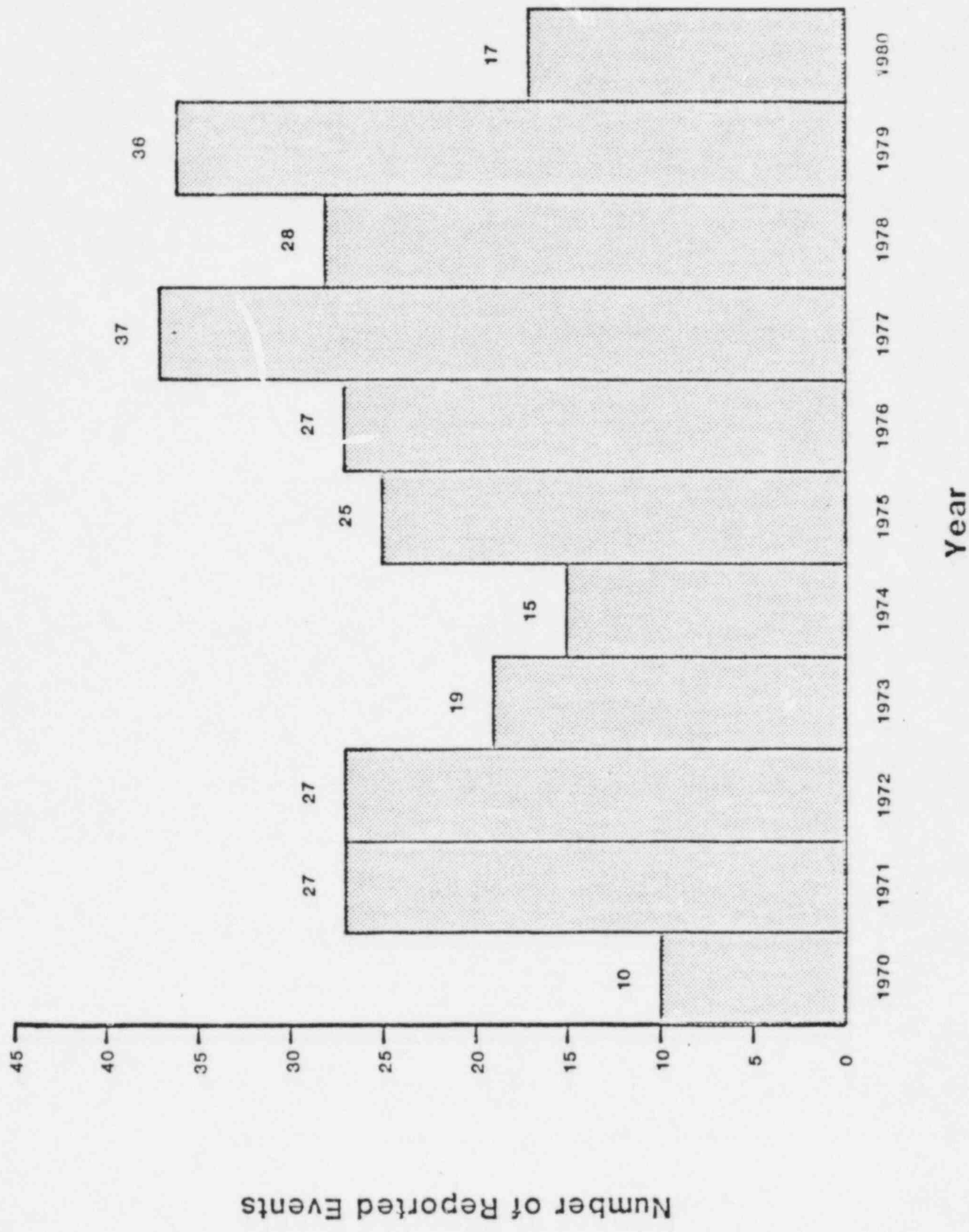
On December 4, 1970, problems were encountered when the isolation condenser was placed into service during a startup test.[14] This incident is discussed in Sect. 4.5.3.3.

Other design errors consisted of inadequate pipe support of the main steam lines (RS-70-6), inadequate pump speed (RS-70-4), and an inappropriate wiring change (RO-70-7). All of these problems have been resolved by design changes.

Installation errors included faulty welds in the main condenser (AO-70-8) and failure of the MSIV due to missing parts (RS-70-6).

1971

Millstone-1 encountered several valve problems in 1971, most of them occurring under test conditions. On January 19, 1971, the entire low pressure ECCS was rendered inoperable during a test. This event (AO-71-1) is described in more detail in Sect. 4.5.1.2.3.


Number of Reported Events
0
5
10
15
20
25
30
35
40
45
10
27
27
19
15
25
27
37
28
36
17
1970
1971
1972
1973
1974
1975
1976
1977
1978
1979
1980
Year

Problems with the gas turbine generator were first experienced in 1971. The gas turbine generator problems surfaced in later years as well and are discussed in Sect. 4.5.3.1.

The plant also experienced several problems with the turbine electrical and mechanical pressure regulators in 1971. These problems were eliminated by replacing defective parts and only one serious transient resulted from these malfunctions. This incident is discussed in detail in Sect. 4.5.2.4.

<u>1972</u>

The 1972 operating experience was marred by 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 damage to other components in the reactor coolant system, some of which were not discovered until 1976. Problems also arose from cracking of feedwater spargers in 1972. All defective spargers were replaced.

<u>1973</u>

Several control rod system problems surfaced in 1973, none producing serious problems. The problems varied in nature, the most notable incident involved degradation of the control rod drive accumulators due to flaking of plating into the hydraulic system.[15] 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. Other problems involved valve and line leaks, none of which were serious and were easily repaired.

1974

The fewest number of reportable events occurred in 1974. Most reported events involved the reactor coolant system. In particular the 1-MS-1D inboard isolation valve failed twice (AO-74-9,10) in one month. The cause of failure each time involved foreign matter on the slide valve for the air operator. The slide valve was cleaned and the problem did not return.

1975

Twenty-five reportable events occurred in 1975. In January, a hydrogen explosion occurred in the condensate demineralizer regeneration system acid day tank. No one was injured. The hydrogen was formed by moisture leaking into a tank of concentrated sulfuric acid and was ignited by a spark from welders.

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.[16] The reactor pressure dropped to 160 psig. During the blowdown, reactor cooldown rate reached 155 F/h, well above 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 were fouled and had to be cleaned.

1976

Three events of a total of 27 reportable events which occurred at Millstone-1 in 1976 are considered very significant. Millstone-1 experienced several problems with the isolation condenser, beginning with a total failure of the condenser in February (see Sect. 4.5.2.1) and continuing throughout the year with various valve and control troubles.

Gas turbine generator problems also plagued the plant. One failure of the generator occurred during a total loss of offsite power (see Sect. 4.5.2.2). Gas turbine generator failures are discussed in detail in Sect. 4.5.3.1.

On November 17, 1976, a Millstone-1 operator caused an inadvertent criticality during a shutdown margin test. This event is discussed in Sect. 4.5.2.3.

It should also be noted that almost all (16/18) of the environmental violations at Millstone-1 occurred in 1976. These events were attributed to an unexpected increase in the abundance of marine life around the plant during the year.

<u>1977</u>

The greatest number of reportable events were recorded in 1977. Several of these involved failures of valves in the main steam lines. Two of these failures affected safety/relief valves at Millstone-1. The first failure occurred on June 16 (AO-77-17) when the safety/relief valve B opened at 555 psig and did not reseat until pressure dropped to 180 psig. No cause was reported. The second failure (AO-77-18) involved safety relief valve F which leaked due to a collapsed filter. The filter was replaced. Four failures involveed the emergency electrical power systems. Three of these failures involvd the gas turbine generator. The fourth event involved failure of diesel generator (AO-77-7) due to a fuel oil leak. The problem was resolved by replacing a cracked nipple.

There were two explosions in the off-gas system on December 13, 1977, in which two workers were injured but the safety of the plant was not challenged. This event is discussed in more detail in Sect. 4.5.2.5.

<u>1978</u>

Although there was a large number of reportable events in 1978 (30), very few of them were significant. On March 10, 1978, the plant experienced problems with a safety/relief valve failing to seat after manual operation, and then opening prematurely during an automatic test (AO-78-4).

On May 29, 1978 the isolated condenser was removed from the system due to failure of a steam trap (AO-78-13). Recurring problems with the isolation condenser are discussed in Sect. 4.5.3.3.

On September 5, 1978, one of four low-low water level sensors failed due to lack of lubrication (AO-78-18). The sensor is one of four used for ECCS initiation. The sensor was lubricated and then satisfactorily tested.

<u>1979</u>

In 1979, the second highest number of reportable events (36) 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. The logic was changed to eliminate this possibility.

On December 19, 1979, a water hammer occurred in the isolation condenser piping. The operators had been instructed to maximize reactor vessel water level in order to minimize thermal stress to the feedwater nozzles. The water hammer was caused by the introduction of the excess water into the steam supply lines of the isolation condenser. Operating procedures have been revised to avoid recurrence of this incident.

Pressure/relief valve problems continued with the valve lifting prematurely and then failing to seat (February 26, 1979). The remaining reports involved minor incidents which violated technical specifications and were inconsequential with respect to plant safety.

1980

Seventeen reportable events were recorded at Millstone-1 in 1980. Problems involving weld cracks in two main steam line and in the condenser nozzle were reported. The cracks were discovered during the fall refueling outage. Pipe cracks at Millstone-1 are discussed in Sect. 4.5.3.2.

4.5.1.2 Systems involved in reportable events. A compilation of all reportable events by system and year is presented in Table 4.6. Some systems which had no reports filed are omitted. There are no discernible time-dependent trends among the systems identified. Most of the reports involved the following systems: reactor coolant, instrumentation and controls, engineered safety features, and electrical power. Each of these systems is discussed in the following subsections.

4.5.1.2.1 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. Over one-third of the reportable events (35%) involved the reactor coolant system. A large majority of reports involving the reactor coolant system concerned valve failures during tests, failures of the electrical and mechanical pressure regulators, and

Table 4.6. Summary of systems involved in reportable events

| System | 1970 | 1971 | 1972 | 1973 | 1974 | 1975 | 1976 | 1977 | 1978 | 1979 | 1980 | Total |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Reactor | 1 | 1 | 1 | 4 | | | 4 | | | 5 | 1 | 17 |
| Reactor coolant | 5 | 8 | 10 | 8 | 8 | 9 | 10 | 15 | 6 | 11 | 7 | 97 |
| Engineered safety features | | 11 | 4 | 3 | 4 | 1 | 4 | 6 | 4 | 7 | 2 | 46 |
| Instrumentation and controls | | 4 | 7 | 2 | | 6 | 5 | 7 | 5 | 10 | 6 | 52 |
| Electrical power | 2 | 7 | 4 | 2 | | 3 | 2 | 4 | 4 | 2 | | 30 |
| Fuel handling | | | | | | | | | 1 | | | 1 |
| Auxiliary water | | 4 | | | 1 | 3 | | | 1 | | | 9 |
| Steam and power | | 2 | 2 | | 1 | 1 | 1 | 1 | 3 | | | 11 |
| Radiation protection | | | | | | | | 4 | | 2 | 1 | 7 |
| Radioactive waste management | | | | | | 2 | | | | | | 2 |
| No system applicable | | | 1 | 1 | 2 | 1 | 1 | 1 | 2 | | | 9 |
| TOTAL | 8 | 37 | 29 | 20 | 16 | 26 | 27 | 38 | 26 | 37 | 17 | 282 |

failures of various coolant parameter monitoring components. Excessive pipe movement was reported several times but no apparent damage resulted.

The isolation condenser provided several problems at Millstone-1. The isolation condenser isolation valves failed during testing nine times. These failures are discussed in more detail in Sect. 4.5.3.3. The condenser itself had to be completely retubed in 1976. This incident is discussed in Sect. 4.5.2.1.

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. This event is discussed in more detail in Sect. 4.5.2.4.

Eight 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. 4.5.3.2.

4.5.1.2.2 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 appearance (18%) as the system involved in reportable events can be attributed to instrumentation set point drift, miscalibration, and spurious trips. No safety functions were compromised as a result of these problems.

4.5.1.2.3 Engineered safety features. Most failures of the engineered safety features (ESF) system 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. In one incident (AO-71-1), the valves were tested at 1000 psig instead of the core spray operating pressure of 300 psig. 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.

4.5.1.2.4 Electrical power system. Almost all reportable events concerning the electrical power systems were attributed to failures of the gas turbine generator. These failures in the electrical power system comprised 10% of all reported events for Millstone-1. A detailed discussion of these failures is presented in Sect. 4.5.3.1.

4.5.1.3 Causes of reportable events. Table 4.7 presents a summary of causes of reportable events at Millstone-1. Almost half of all reportable events 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 found.

Over the operating experience reviewed, human error was responsible for half of all reportable 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 reveals that only 24% of those events were attributed to inherent failures.

Table 4.7. Causes of Reported events for Millstone-1

| Cause | 1970 | 1971 | 1972 | 1973 | 1974 | 1975 | 1976 | 1977 | 1978 | 1979 | 1980 | Total |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Administrative | | 1 | | | | 2 | 2 | 1 | 1 | 2 | | 9 |
| Design | 4 | 7 | 3 | 2 | 4 | 3 | 6 | 2 | 2 | 2 | 2 | 37 |
| Fabrication | | | | | | | | | | | | 0 |
| Inherent failure | 2 | 8 | 15 | 10 | 7 | 15 | 18 | 15 | 14 | 16 | 9 | 129 |
| Installation | 3 | 5 | 2 | 1 | 1 | 2 | 2 | 2 | 1 | 3 | 2 | 24 |
| Lightning | | 1 | | | | | | | | | | 1 |
| Maintenance | 1 | 6 | 7 | 6 | 4 | 5 | 9 | 2 | 8 | 2 | 3 | 53 |
| Operator | 1 | 2 | | 1 | | | 1 | 1 | | 2 | | 8 |
| Weather | | | | | | | 1 | | | | | 1 |
| TOTAL | 11 | 30 | 27 | 23 | 16 | 27 | 39 | 23 | 26 | 27 | 16 | 262 |

4.5.1.4 Events of environmental importance.

4.5.1.4.1 Radioactivity release events. Eight events at Millstone-1 resulted in radioactive releases. Table 4.8 summarizes the amount of radiation released annually from the plant. Three of the eight reported release events were caused by mishandling or failures of the waste disposal system. The remaining five events involved inadvertent exposures of maintenance workers. The quarterly release rate was exceeded four times during the period of 1976 to 1979.

On March 25, 1974, two workers were overexposed to Co-58 and Co-60 due to poor ventilation in a maintenance area. Also, badge readings during the fall 1974 refueling outage showed that three men had exceeded their dose limits. An overflow of a surge tank onto the boiler room floor contaminated 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 excess doses of radiation while performing maintenance on the feedwater spargers.

Throughout the ten years of operating experience there were eight reports of excess radioactivity 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 attributed to fallout from the Chinese atomic bomb tests.

4.5.1.4.2 Nonradiological events. The only nonradiological environmental events consisted of the impingement of species of fish on intake screen above prescribed limits. There were 18 reports of excess fish impingement, 16 in 1976, two in 1977. The unusually high numbers were attributed to an abnormal increase in fish population in 1976 and 1977.

Table 4.8. Summary of radioactivity released from Millstone 1

| Release (curies) | 1970 | 1971 | 1972 | 1973 | 1974 | 1975 | 1976 | 1977 | 1978 | 1979 | 1980 |
|---|---|---|---|---|---|---|---|---|---|---|---|
| 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 | |
| Total I-131 | 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 | |
| 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 | |
| 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 | |
| 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 | |
| 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 | |
| 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 | |
| Total noble gases | NA | NA | 0 | 0 | 0 | 1.11E+1 | 3.56E-1 | 3.67E-1 | 7.65E-1 | 7.00E-1 | |
| 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 | |

4.5.2 Review of significant events

Each reportable event was screened using criteria as a step in the evaluation process (see Sect. 3.2). A compilation of the number of significant events appears in Table 4.9. Events with serious safety implications are described in detail in subsequent sections. These events, which degraded a safety function or initiated a DBE are: loss of isolation condenser (4.5.2.1), loss of offsite power with partial loss of emergency power (4.5.2.2), inadvertent criticality (4.5.2.3), loss of pressure control followed by a blowdown (4.5.2.4), and hydrogen explosion in the off-gas system (4.5.2.5).

4.5.2.1 Isolation condenser failure. On February 12, 1976, the plant shut down due to arcing of the main transformer during a storm.[17,18] 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.

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

Table 4.9. Summary of significant events at Millstone-1

| Significance category | 1970 | 1971 | 1972 | 1973 | 1974 | 1975 | 1976 | 1977 | 1978 | 1979 | 1980 | Total times assigned |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| S1 | | 1 | | | | | | 1 | | | | 2 |
| S2 | | 1 | | | | | | | | | | 1 |
| S3 | 1 | | 2 | | | | | | | | | 3 |
| S4 | | | | | | | | | | | | 0 |
| S5 | | | | | | | | | | | | 0 |
| S6 | | | | | | 1 | | | | | | 1 |
| S7 | | 2 | 2 | 2 | | | 2 | | | 3 | | 11 |
| S8 | | 1 | | | | 1 | 2 | 2 | 1 | 1 | | 8 |
| S9 | | | | | | | | 1 | | | | 1 |
| TOTAL | 1 | 5 | 4 | 2 | 0 | 2 | 4 | 4 | 1 | 4 | 0 | 27 |

was attributed to the intrusion of chlorine from a previous failure of the main condenser. Other tubes in the isolation condenser was retubed with Inconel 600 tube material.

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

4.5.2.2 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.[19] 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.

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 then tripped a second time when the DC batteries failed.

Recurrence of the event was mitigated by the following corrective actions:[20]

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 locked 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. 4.5.1.1). 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 love diesel one at Millstone-1 has a very good performance record.

4.5.2.3 Inadvertent criticality. While performing a shutdown margin test on November 12, 1976, an inadvertent criticality and reactor trip occurred. An operator error in selecting rods for the test caused the unplanned criticality.[21]

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 bypassed 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.[22]

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 was 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 to the plant management. This violation had the potential for causing or contributing to an occurrence related to health and safety.

4.5.2.4 Pressure transient and blowdown. On October 19, 1971 the Millstone-1 plant experienced a pressure transient followed by a blow down of 75,000 gal of water.[23] With the reactor at 100% power, the electric pressure regulator caused pressure to rise to 1040 psig. The operator placed the mechanical pressure regulator into service, but this did not resolve the transient. The reactor scrammed on a high trip of the average power range monitor (APRM). The turbine tripped and the turbine bypass valve 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 a service and 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 pilot-valve disk.

4.5.2.5 Hydrogen explosions in off-gas system. On December 13, 1977, two hydrogen explosions occurred at Millstone-1.[24] 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 60 m away, breached the reinforced-concrete ceiling of the room at the base of the stack, extensively damaged the ceiling beams, damaged supports of a radiation monitor for the stack, and cracked the stack. The control room was alerted to the second detonation and the operators manually trip the plant, terminating the generation 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 BWR's, but most explosions 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.[25]

Immediately after the event, the NRC required that all BWR licensees to 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.

### 4.5.3 Trends and safety implications of reportable events

Using the systems involved in reportable events listed in Table 4.2, specific trends and problem areas were identified for safety-related functions: (1) partial loss of emergency power, (2) pipe cracks, (3) isolation condenser valving problems.

4.5.3.1 Partial loss of emergency power. Millstone-1 has experienced 31 failures of the emergency generator systems and controls. Gas turbine generator failure was the dominant contributor to degradation of the emergency power system. The gas turbine generator failed to start or run its mission 24 times. Five of these were failures on demand, the remainder occurred under test conditions. Appendix B provides a description of the 24 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 24 failures were attributed to a faulty speed switch. This switch was totally replaced four times during the ten 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 error. Procedural error caused the most significant failure of the gas turbine generator (AO 76-29) described in more detail in Sect. 4.5.1.2.

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, 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. In light of the numerous failures of the gas turbine generator and the high potential for common mode loss of offsite power, the performance of the emergency power system should be examined in greater detail. 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).

4.5.3.2 Pipe cracks. Millstone-1 reported eight incidents of pipe cracking. Pipe cracks have always been a generic problem in BWRs.[26] The most significant cracking events occurred in 1972, 1976, and 1980.

In 1972 the cracks were detected in the feedwater spargers. Subsequently, all spargers were replaced, a task which resulted in excessive down time for the utility. In 1976 a weld leak was found in the head spray system. The failure was attributed to stress corrosion cracking and cracked components were replaced. In 1980 cracks were found in two main steam line supports and in a condenser nozzle. The condenser nozzle-to-steam supply weld 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.

It is beyond the scope of this study to analyze the potential effects of the cracking at Millstone-1; however, no safety related incident has occurred at the plant as a result of pipe cracking.

4.5.3.3 Isolation condenser valve failures. Millstone-1 has experienced nine failures of the isolation condenser valves over the time period from 1970 to 1980. The condenser isolated twice as it was placed in operation (AO's 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 failed to close during a test. The torque switch setting was incorrect and was readjusted.

### 4.6 Evaluation of operating Experience

The overall review of Millstone-1's operating experience is based on information contained in two generic sources: reports of forced shutdowns and power reductions, and a compilation of reportable occurrences. Each category was reviewed by a separate reviewer with periodic discussions of observations between reviewers.

Equipment failure was the dominant cause of forced shutdowns at Millstone-1. This includes both component and instrumentation failures. Relatively few shutdowns were initiated by human errors. During the first few years of operation, problems with the level controller on the moisture separator between the high an low side of the main turbine forced the plant to shut down often. The plant was also plagued with corrosion failures of tube bundles, beginning with chlorine intrusion of the main condenser in 1972. The chlorine was never completely purged from the coolant system, causing corrosion problems in various other system components, some of them appearing years later. General Electric installed completely redesigned equipment, built with a more prudent choice of metals, and the problem seems to have disappeared.

Roughly half of all reportable events were caused by human errors. No discernable time trends appeared in the number of reportable events. A majority of the significant events were caused by human-related error involving maintenance and design errors. Maintenance errors disabled the gas turbine generator, an engineered safety feature several times. Operator error caused an inadvertent criticality in 1976; however, no major transients resulted from human error.

Mechanical problems of greatest significance stem from failures of the gas turbine generator speed switch, weld cracks, corrosion in portions of the emergency heat removal system and a variety of valve problems with the isolation condenser. Pipe cracking in BWR's is a generic problem and corrective actions implemented at the plant should control the problem. The gas turbine generator speed switch has been replaced several times, as recently as February, 1979. No trouble has been reported since through 1980, but two years is insufficient to indicate that the problem has been solved.

Further examination of the isolation condenser valves is warranted in view the number of failuress of the various valves and the function of the isolation condenser to provide removal of afterheat from the core.

Millstone-1 experienced a total loss of offsite power and was forced to operate for 3 h on one diesel generator on August 19, 1976. No major problems resulted during this event; however it has been identified as a potential precursor to a more serious accident.[27] The performance of the gas turbine generator should be examined in greater detail, based on its poor reliability characteristics.

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Appendix A: Millstone 1

Part 1. Forced Shutdown and Power Reduction Tables

Table A1.1 1970 Forced Shutdowns and Power Reductions for Millstone 1

| No. | Date (1970) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 11/19 | 16 | NA | | Momentary main steam line high flow signal. | B | 3 | Reactor Coolant (CC) | Instrumentation & Controls | D1.3 |
| 2 | 11/21 | 11 | NA | | Reactor mode switch wiggling resulted in main steam isolation. | B | 3 | Reactor Coolant (CD) | Instrumentation & Controls | D2.1 |
| 3 | 12/5 | 2 | 0 | | Electrical pressure regulator pressure control oscillation caused spurious low level indication. | B | 3 | Steam & Power (HA) | Instrumentation & Controls | D1.3 |
| 4 | 12/22 | 2 | 15 | C4 | Packing leak on isolation condenser steam supply valve. | B | 1 | Reactor Coolant (CE) | Valves | N1.1.4 |
| 5 | 12/23 | 24 | 0 | C4 | Cracked weld on main condenser. | B | 1 | Steam & Power (HC) | Heat Exchangers (Main Condenser) | N1.1.4 |
| 6 | 12/29 | 23 | 0 | | Pressure control bypass valve linkage torn from its support. | B | 1 | Steam & Power (HE) | Valves | N1.1 |
| 7 | 12/30 | 1 | 13 | | Turbine trip due to high level in moisture separator drain tank. | B | 3 | Steam & Power (HA) | Turbines | D2.3 |

Table A1.2 1971 Forced Shutdowns and Power Reductions for Millstone 1

| No. | Date (1971) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 1/2 | 1 | 40 | | Turbine trip due to high level in moisture separator drain tank. | A | 3 | Steam & Power (HA) | Turbines | D2.3 |
| 2 | 1/2 | 7 | 11.5 | | Turbine trip - high level in moisture separator drain tank. | A | 3 | Steam & Power (HA) | Turbines | D2.3 |
| 3 | 1/14 | 16 | 64 | | Turbine manually tripped to fix condenser leak. | A | 1 | Steam & Power (HC) | Heat Exchangers (Main Condenser) | N3.1 |
| 4 | 1/19 | 109 | 0 | C1 | Reactor shutdown - turbine control valve closed. | A | 1 | Engineered Safety Features (SF-D) | Valves | D2.1 |
| 5 | 1/25 | 33 | 0 | C4 | Reactor shutdown to repair core spray injection valves. | B | 1 | Engineered Safety Features (SF-D) | Valves | N1.0 |
| 6 | 1/27 | 85 | 0 | | Traveling screen damaged circulating water pump - damaged shaft. | B | 1 | Steam & Power (HF) | Pumps | N1.1.4 |
| 7 | 2/12 | 24 | 0 | | Spurious indication of low reactor water level trip of ECCS. | A | 3 | Reactor Coolant (CB) | Instrumentation & Controls | D3.2 |
| 8 | 2/15 | 3 | 30 | | Turbine trip - high moisture separator level. | A | 3 | Steam & Power (HA) | Turbines | D2.3 |
| 9 | 2/21 | 21 | 75 | | Turbine trip - high moisture separator level. | A | 3 | Steam & Power (HA) | Steam Turbine | D2.3 |
| 10 | 3/2 | 65 | 16 | | Main steam line safety valve blowing steam. | A | 1 | Steam & Power (HI) | Valves | D6.1 |

Table A1.2 (Continued)

| No. | Date (1971) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 11 | 3/12 | 5 | 100 | | Turbine trip - high moisture separator drain tank level. | A | 3 | Steam & Power (HA) | Turbines | D2.3 |
| 12 | 4/14 | 3 | 0 | | Repair commutator rings on M-G sets. | B | 1 | Reactor Coolant (CB) | Generators | N1.1.4 |
| 13 | 4/19 | 6 | 90 | | Main steam line low pressure trip sensing line broke. | A | 3 | Reactor Coolant (CC) | Instrumentation & Controls | N2.1 |
| 14 | 4/21 | 8 | 45 | | Turbine trip - high moisture separator drain tank level. | A | 3 | Steam & Power (HA) | Turbines | D2.3 |
| 15 | 4/22 | 1 | 100 | | Turbine trip HMSDTL. | A | 3 | Steam & Power (HA) | Turbines | D2.3 |
| 16 | 4/22 | - | 100 | | Turbine trip HMSDTL. | A | 3 | Steam & Power (HA) | Turbines | D2.3 |
| 17 | 4/22 | 1 | 100 | | Turbine trip HMSDTL. | A | 3 | Steam & Power (HA) | Turbines | D2.3 |
| 18 | 4/22 | 1 | 100 | | Turbine trip HMSDTL. | A | 3 | Steam & Power (HA) | Turbines | D2.3 |
| 19 | 4/22 | 1 | 100 | | Turbine trip HMSDTL. | A | 3 | Steam & Power (HA) | Turbines | D2.3 |
| 20 | 5/1 | 39 | 60 | | Repair condensate test line. | A | 1 | Steam & Power (HH) | Pipes, Fittings | N1.1.4 |

Table A1.2 (Continued)

| No. | Date (1971) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 21 | 5/12 | 17 | 70 | | Repair air leak in drywell. | A | 1 | Engineered Safety Features (SA) | Vessels, Pressure | N1.1.4 |
| 22 | 5/25 | 7 | 70 | | Feedwater control valve closed. | A | 3 | Reactor Coolant (CH) | Instrumentation & Controls | D1.1 |
| 23 | 5/27 | 57 | 80 | | Turbine control valves failed shut. | A | 3 | Steam & Power (HA) | Valves | D2.3 |
| 24 | 5/30 | 12 | 20 | | Steam leak in main steam line. | A | 1 | Reactor Coolant (CC) | Pipes, Fittings | N3.1 |
| 25 | 6/11 | 1 | 100 | | Turbine trip HMSDTL. | A | 3 | Steam & Power (HA) | Turbines | D2.3 |
| 26 | 6/11 | 2 | 100 | | Turbine trip HMSDTL. | A | 3 | Steam & Power (HA) | Turbines | D2.3 |
| 27 | 6/24 | 2 | 100 | | Turbine full load reject due to lightning causing loss of 383 line. | A | 3 | Steam & Power (HA) | Turbines | D2.2 |
| 28 | 6/25 | 3 | 100 | | Turbine trip due to lightning causing loss of 383 line. | A | 1 | Steam & Power (HA) | Steam Turbine | D2.2 |
| 29 | 6/26 | 3 | 0 | | Spurious IRM trip. | A | 3 | Instrumentation & Controls (IA) | Instrumentation & Controls | N2.4 |
| 30 | 8/12 | 4 | 45 | | Turbine trip HMSDTL. | A | 3 | Steam & Power (HA) | Turbines | D2.3 |

Table A1.2 (Continued)

| No. | Date (1971) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 31 | 8/29 | 7 | 45 | | Traveling screen failure caused loss of circulating water pumps and turbine trip. | A | 3 | Steam & Power (HF) | Pumps | N1.1 |
| 32 | 8/30 | 168 | 0 | | Main condenser low vacuum trip. | A | 3 | Steam & Power (HC) | Heat Exchangers (Main Condenser) | D2.5 |
| 33 | 9/23 | 134 | 80 | | 850 PSI low pressure trip. | A | 3 | Reactor Coolant (CC) | Instrumentation & Controls | N1.1.4 |
| 34 | 9/29 | 83 | 0 | | Turbine control valve malfunction. | A | 3 | Steam & Power (HA) | Valves | D2.3 |
| 35 | 10/3 | 60 | 0 | | Turbine control valve malfunction. | A | 3 | Steam & Power (HA) | Valves | D2.3 |
| 36 | 10/10 | 248 | 77 | | Turbine control valve malfunction | A | 3 | Steam & Power (HA) | Valves | D2.3 |
| 37 | 10/22 | 20 | 0 | | Failure of automatic pressure relief valve to seat due to scored pilot valve disc. | A | 3 | Reactor Coolant (CC) | Valves | N1.1.4 |
| 38 | 10/24 | 10 | 0 | | APR bellows leak. | A | 3 | Reactor Coolant (CC) | Valves | N1.1.4 |
| 39 | 12/11 | 4 | 100 | | Instrument error on main steam line flow detectors. | A | 3 | Reactor Coolant (CC) | Instrumentation & Controls | N1.1.4 |
| 40 | 12/12 | 4 | 100 | | Reactor pressure switch inadvertently tripped. | G | 3 | Reactor Coolant (CC) | Instrumentation & Controls | N6.1 |

Table A1.2 (Continued)

| No. | Date (1971) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 41 | 12/11 | 4 | 100 | | Inadvertent make-up of reactor pressure switch. | G | 3 | Reactor Coolant (CC) | Instrumentation & Controls | N6.1 |
| 42 | 12/20 | 57 | 100 | | Faulty isolation condenser return valve. | A | 1 | Reactor Coolant (CE) | Valves | N1.1.4 |

Table A1.3 1972 Forced Shutdowns and Power Reductions for Millstone 1

| No. | Date (1972) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 2/4 | 149 | 70 | | Turbine stop valve testing induced pressure oscillations resulting in turbine trip. | A | 3 | Steam & Power (HA) | Valves | D2.[illegible] |
| 2 | 2/11 | 43 | 0 | | Leaky main steam line gaskets. | A | 1 | Reactor Coolant (CC) | Pipes, Fittings | N3.1 |
| 3 | 2/14 | 129 | 30 | | Improper response from main steam line venturi differential pressure cells. | A | 1 | Reactor Coolant (CC) | Instrumentation & Controls | N1.1.4 |
| 4 | 2/23 | 168 | 100 | | Degraded main steam line venturi. | A | 1 | Reactor Coolant (CC) | Instrumentation & Controls | N1.1.4 |
| 5 | 3/9 | 62 | 20 | | Faulty operation of thrust bearing wear detector induced turbine trip. | A | 3 | Steam & Power (HA) | Turbines | D2.3 |
| 6 | 3/12 | 2 | 30 | | Void collapse from cold feedwater increase in flow. | A | 3 | Reactor Coolant (CH) | Pumps | D1.2 |
| 7 | 8/18 | 5 | 100 | | Testing of thrust bearing wear detector tripped turbine. | A | 3 | Steam & Power (HA) | Instrumentation & Controls | D2.3 |
| 8 | 8/29 | 27 | 100 | C4 | Drywell floor drain sump leakage. | A | 1 | Engineered Safety Features (SA) | Vessels, Pressure | N1.1.4 |

Table A1.4 1973 Forced Shutdown and Power Reduction for Millstone 1

| No. | Date (1973) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D) NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 3/6 | 53 | 0 | | Condensate booster pump not started in time - low water level. | G | 3 | Reactor Coolant (CC) | Pumps | D2.7 |
| 2 | 3/14 | 83 | 80 | | Blown fuse on CRD scram solenoid scrammed Group II control rods. | A | 3 | Reactor (RB) | Circut Closers/ Interrupters | D4.3 |
| 3 | 7/30 | 12 | 60 | | Reactor vessel water level transmitter failure. | A | 3 | Reactor Coolant (CB) | Instrumentation & Controls | N2.1 |
| 4 | 8/10 | 14 | 76 | | Reactor vessel water level transmitter level malfunction. | A | 3 | Reactor Coolant (CB) | Instrumentation & Controls | N2.1 |
| 5 | 8/10 | 4 | 0 | | High reactor water level due to starting feedwater pump. | G | 3 | Reactor Coolant (CB) | Pumps | D1.2 |
| 6 | 8/13 | 26 | 76 | | Fault in mode switch caused a scram. | A | 3 | Reactor Coolant (CB) | Instrumentation & Controls | N2.1 |
| 7 | 9/21 | 11 | 50 | | Fault in EPR controls opened turbine bypasss valves dropping reactor pressure. | A | 3 | Reactor Coolant (CC) | Instrumentation & Controls | D1.3 |
| 8 | 12/7 | 5 | 80 | | Person bumped level instrument rack. | G | 3 | Reactor Coolant (CB) | Instrumentation & Controls | N6.3 |
| 9 | 3/11 | 31 | 40 | | Repaired main steam isolation position indication switch. | A | 1 | Reactor Coolant (CD) | Valves | N2.4 |
| 10 | 3/13 | 6 | 75 | | Low lube-oil pressure alarm on recirc. pump motor. | A | 1 | Reactor Coolant (CA) | Motors | N1.1.4 |

Table A1.4 (Continued)

| No. | Date (1973) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 11 | 3/19 | 40 | 58 | | Low lube oil press. alarm reactor recirc. pump. | A | 1 | Reactor Coolant (CA) | Pumps | N1.1.4 |
| 12 | 4/18 | 2152 | 18 | | Replaced feedwater sparger. | A | 1 | Reactor Coolant (CH) | Pipes, Fittings | N1.1.4 |
| 13 | 7/14 | 64 | 0 | | Leak on vent line for recirculation pump discharge. | A | 1 | Reactor Coolant (CA) | Pipes, Fittings | N1.1.4 |
| 14 | 7/16 | 236 | 0 | | Examine for inverted control rod internals. | D | 1 | Reactor (RB) | Control Rods | N8.3 |
| 15 | 9/19 | 36 | 0 | | Leaking instrument tap in feedwater line. | A | 1 | Reactor Coolant (CH) | Pipes, Fittings | N1.1.4 |
| 16 | 9/20 | 25 | 0 | | Leaking automatic pressure relief valve. | A | 1 | Reactor Coolant (CC) | Valves | N1.1.4 |
| 17 | 10/6 | 16 | 67 | | Recirc pump motor failed. | A | 1 | Reactor Coolant (CA) | Motors | N1.1.4 |
| 18 | 10/14 | 20 | 68 | | Tested recirc. pump motor. | A | 1 | Reactor Coolant (CA) | Motors | N1.1.4 |
| 19 | 10/28 | 42 | 67 | | Opened and tested recirc. loop cross-tie valves. | A | 1 | Reactor Coolant (CA) | Valves | N2.4 |
| 20 | 12/30 | 16 | 0 | | Isolation condenser flange leak. | A | 1 | Reactor Coolant (CE) | Pipes, Fittings | N1.1.4 |

Table A1.5 1974 Forced Shutdowns and Power Reductions for Millstone 1

| No. | Date (1974) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 3/2 | 4 | 50 | | Power reduction - Suspected leak in main condenser. | A | 5 | Steam & Power (HC) | Heat Exchanger (Main Condenser) | N1.1.4 |
| 2 | 3/3 | 7 | 50 | | Power reduction - Leak in main condenser. | A | 5 | Steam & Power (HC) | Heat Exchanger (Main Condenser) | N1.1.4 |
| 3 | 3/6 | 13 | 50 | | Excessive drywell leakage from stem packing of reactor recirculation equalizer valve. | A | 1 | Engineered Safety Features (SA) | Valves | N10 |
| 4 | 6/11 | 7 | 65 | | Recirc. pump speed control fault. | A | 3 | Reactor Coolant (CB) | Instrumentation & Controls | N1.1.4 |
| 5 | 11/3 | 15 | 35 | | Turbine trip caused by maintenance on feedwater transmitter. | A | 3 | Steam & Power (HA) | Instrumentation & Controls | D2.3 |
| 6 | 11/4 | 9 | 35 | | MSIV malfunction due to moisture in the slide valve that controls valve action. | A | 3 | Reactor Coolant (CC) | Valves | D2.4 |
| 7 | 11/5 | 5 | 40 | | APRM high flux due to pressure oscillations. | A | 3 | Reactor Coolant (CC) | Instrumentation & Controls | N1.1.4 |
| 8 | 11/15 | 8 | 0 | | Power reduction - MSIV failed to close. | A | 5 | Reactor Coolant (CC) | Valves | N1.1.4 |
| 9 | 11/18 | 15 | 0 | | Power reduction - Main generator exciter ground fault. | A | 5 | Electric Power (EB) | Generators | N1.1.4 |

Table A1.5 (Continued)

| No. | Date (1974) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 10 | 12/16 | 8 | 97 | | Broken stem - feedwater control valve - prevented flow of feedwater. | A | 3 | Reactor Coolant (CH) | Valves | D2.7 |
| 11 | 12/27 | 34 | 95 | | Repair main feedwater control valve. | A | 1 | Reactor Coolant (CH) | Valves | N1.1.4 |
| 12 | 12/27 | 1 | 15 | | Power reduction - turbine trip due to high level in moisture separator. | A | 5 | Steam & Power (HA) | Turbine | D2.3 |
| 13 | 12/29 | 2 | 10 | | Power reduction - turbine trip due to high level in moisture separator. | A | 5 | Steam & Power (HA) | Turbine | D2.3 |
| 14 | 12/29 | 3 | 10 | | Power reduction - turbine trip due to high level in moisture separator. | A | 5 | Steam & Power (HA) | Turbine | D2.3 |

Table A1.6 1975 Forced Shutdowns and Power Reductions for Millstone 1

| No. | Date (1975) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DOE(D)/ NSIC(M) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 2/15 | 9 | 20 | | Power reduction - drywell entry made to perform maintenance on TIP index. | A | 5 | Reactor (RC) | Instrumentation & Controls | N2.1 |
| 2 | 3/11 | 87 | 60 | | Blown valve stuffing box - LPCI system. | A | 1 | Engineered Safety Features (SF-B) | Valves | N1.1.4 |
| 3 | 5/20 | 55 | 100 | C-4 | Pressure relief valve failed to seat. | A | 2 | Reactor Coolant (CC) | Valve | N1.1.4 |
| 4 | 6/20 | 35 | 15 | | Service water pump repairs. | A | 1 | Auxiliary Water (WA) | Pumps | N1.1.4 |
| 5 | 8/13 | 16 | 80 | | Power reduction - arcing on B-phase disconnect. | A | 5 | Electric Power (EB) | Circuit Closers/ Interrupters | N1.1.4 |
| 6 | 8/18 | 5 | 96 | C-4 | Power reduction - APRs failed to meet Tech Specs. | B | 5 | Reactor Coolant (CC) | Valves | N1.1.4 |
| 7 | 8/30 | 8 | 0 | | Pressure regulator transient caused APRM scram due to plugged moog valve filter. | H | 3 | Reactor Coolant (CC) | Filters | N1.1.4 |
| 8 | 9/12 | 451 | 94 | | Transformer insulation breakdown. | B | 1 | Electric Power (EB) | Transformer | N1.1.4 |
| 9 | 10/25 | 15 | 2 | | Installing insulating bolts on main transformer. | H | 1 | Electric Power (EB) | Transformer | N1.1.4 |
| 10 | 10/27 | 19 | 60 | | Maintenance on transversing incore probe system. | A | 1 | Reactor (RC) | Instrumentation & Controls | N1.1.4 |

Table A1.6 (Continued)

| No. | Date (1975) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 11 | 11/13 | 223 | 90 | C-4 | Crack in jet pump break detection sensing line. | A&B | 1 | Reactor Coolant (CC) | Instrumentation & Controls | N1.1.4 |
| 12 | 11/23 | 9 | 0 | | Oscillations in pressure regulator due to dirt in sensing lines. | A | 3 | Reactor Coolant (CC) | Instrumentation & Controls | N1.1.4 |
| 13 | 11/26 | 8 | 60 | | Pressure spike occurred while switching to mech. pressure regulator. | A | 3 | Reactor Coolant (CC) | Instrumentation & Controls | N1.1.4 |
| 14 | 12/11 | 66 | 90 | | Installed missing part on main transformer. | A&B | 1 | Electric Power (EB) | Transformer | N5.1 |

Table A1.7 1976 Forced Shutdowns and Power Reductions for Millstone 1

| No. | Date (1976) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 2/12 | 42 | 100 | | Arcing across high voltage bushing on main transformer. Tripped generator. | A | 3 | Electric Power (EB) | Transformers | N1.1.4 |
| 2 | 3/8 | 85 | 100 | | Inoperability of gas turbine generator due to governor out of adjustment. | A&B | 1 | Electric Power (EF) | Turbines | N1.1.4 |
| 3 | 3/15 | 132 | 20 | | Inoperability of gas turbine. Replaced electronic governor. | A&B | 1 | Electric Power (EF) | Turbines | N1.1.4 |
| 4 | 7/16 | 64 | 90 | | Repaired motor operator of isolation condenser isolation valve. | A | 1 | Reactor Coolant (CE) | Valve Operators | N1.1.4 |
| 5 | 8/10 | 105 | 95 | | High winds deposited salt on main transformer insulators. Arcing. | H | 3 | Electric Power (EB) | Transformers | N9.2 |
| 6 | 8/10 | 100 | 95 | | Problems with speed control of gas turbine generator. Outage extension. | A | N/A | Electric Power (EE) | Turbines | N1.1.4 |
| 7 | 8/22 | 0 | 65 | | Power reduction. Adjust control rod pattern. | H | 5 | Reactor (RB) | Control Rods | N1.1 |
| 8 | 12/1 | 45 | 100 | | Problems with main turbine generator pressure regulator causing it to open. | A | 1 | Steam & Power (HA) | Instrumentation & Controls | N1.1.4 |
| 9 | 12/17 | 51 | 100 | | Malfunction of isolation condenser isolation valve, cleanup isolation valve. | A | 1 | Reactor Coolant (CE) | Valve Operators | N1.1.4 |

Table A1.8 1977 Forced Shutdowns and Power Reductions for Millstone 1

| No. | Date (1977) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 1/3 | 49 | 100 | | Malfunction of main generator electric pressure regulator due to plugged line. | A | 3 | Steam & Power (HB) | Instrumentation & Controls | N2.1 |
| 2 | 1/26 | 13 | 100 | | Increasing coolant conductivity due to demineralizer malfunction. | A | 2 | Reactor Coolant (CG) | Demineralizers | N1.1.4 |
| 3 | 3/6 | 36 | 100 | | Repair of small steam leak on main line drain line to main condenser. | A | 1 | Steam & Power (HB) | Pipes, Fittings | N1.1.4 |
| 4 | 4/7 | 9 | 100 | | While performing reactor water level surveillance, an inadvertent reactor scram occurred. | A | 3 | Instrumentation & Controls (IA) | Valves | N2.4 |
| 5 | 4/23 | 0 | 100 | | Power reduction. Repair steam leak on extraction non-return valve. | A | 5 | Steam & Power (HA) | Valves | N1.1.4 |
| 6 | 5/14 | 11 | 100 | | Steam leak on an extraction non-return valve. | A | 3 | Steam & Power (HA) | Valves | N1.1.4 |
| 7 | 6/13 | 8 | 0 | | Faulty test solenoid prevented MS valve from returning to open position. | A | 3 | Steam & Power (HB) | Instrumentation & Controls | N2.1 |
| 8 | 6/14 | 132 | 0 | | Mechanical pressure regulator swing tripped turbine. | A | 2 | Steam & Power (HB) | Circuit Closers/ Interrupters | N2.4 |
| 9 | 6/19 | 6 | 100 | | Mechanical pressure regulator did not take control - bypass valves failed open. | A | 3 | Steam & Power (HB) | Circuit Closers/ Interrupters | N2.4 |

Table A1.8 (Continued)

| No. | Date (1977) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 10 | 7/12 | 10 | 100 | | Bent actuator on main steam isolation valve allowed valve to close. | A | 3 | Reactor Coolant (CD) | Circuit Closers/ Interrupters | N2.4 |
| 11 | 7/22 | 37 | 100 | | False signal tripped breaker on lube oil pump for recirculating MG set. | A | 3 | Reactor Coolant (CB) | Instrumentation & Controls | N2.4 |
| 12 | 8/6 | 37 | 100 | | Loss of plant air due to loss of cooling water to air compressors. | A | 3 | Auxiliary Process (PA) | Blowers | N1.1.4 |
| 13 | 8/27 | 0 | 100 | | Power reduction. Plug tubes in main condensers. | B | 5 | Steam & Power (HC) | Heat Exchangers (Main Condenser) | N1.1.4 |
| 14 | 9/21 | 0 | 90 | | Power reduction. Condenser tube maintenance. | B | 5 | Steam & Power (HC) | Heat Exchangers (Main Condenser) | N1.1.4 |
| 15 | 11/22 | 66 | 100 | | Leak in feedwater heater. | A&B | 1 | Reactor Coolant (CH) | Heat Exchangers | N1.1.4 |
| 16 | 11/29 | 55 | 50 | | Automatic pressure relief valve lifted prematurely. | A&B | 2 | Reactor Coolant (CC) | Valves | D6.1 |
| 17 | 12/13 | 269 | 80 | | Manual scram in response to a second hydrogen explosion in offgas stack. | A | 2 | Radioactive Waste Management (MB) | Other | N1.1.4 |

Table A1.9 1978 Forced Shutdowns and Power Reductions for Millstone 1

| No. | Date (1978) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 1/31 | 14 | 90 | | Steam leak on 2 inch steam line. | A | 1 | Reactor Coolant (CC) | Pipes, Fittings | N1.1.4 |
| 2 | 5/19 | 22 | 100 | | Malfunction of level control system for moisture separator tripped turbine. | A | 3 | Reactor Coolant (CC) | Instrumentation & Controls | D2.3 |
| 3 | 5/29 | 16 | 100 | | Broken air supply tube to moisture separator level controller tripped turbine. | A | 3 | Steam & Power (HA) | Pipes, Fittings | D2.3 |
| 4 | 7/14 | 100 | 100 | | Replaced containment isolation valve cable splices with those that are environmentally qualified. | B | 1 | Electric Power (ED) | Electrical Conductors | N10 |
| 5 | 7/20 | 1 | 100 | | Malfunction of moisture separator drain tank level controllers tripped turbine. | A | 4 | Steam & Power (HB) | Instrumentation & Controls | D2.3 |
| 6 | 12/12 | 44 | 100 | | Main steam line flow instrumentation check out. | A | 3 | Instrumentation & Controls (IA) | Instrumentation & Controls | N2.4 |
| 7 | 6/10 | 0 | 100 | | Power reduction. Plug main condenser tubes. | A | 5 | Steam & Power (HC) | Heat Exchangers (Main Condenser) | N1.1.4 |
| 8 | 7/3 | 0 | 90 | | Reduced power to plug main condenser tubing. | A | 5 | Steam & Power (HC) | Heat Exchangers (Main Condenser) | N1.1.4 |

Table A1.9 (Continued)

| No. | Date (1978) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 9 | 7/25 | 0 | 80 | | Reduced power to plug main condenser tubing. | A | 5 | Steam & Power (HC) | Heat Exchangers (Main Condenser) | N1.1.4 |
| 10 | 7/26 | 0 | 80 | | Reduced power to plug main condenser tubing. | A | 5 | Steam & Power (HC) | Heat Exchangers (Main Condenser) | N1.1.4 |
| 11 | 7/27 | 0 | 80 | | Reduced power to plug main condenser tubing. | A | 5 | Steam & Power (HC) | Heat Exchangers (Main Condenser) | N1.1.4 |
| 12 | 7/29 | 0 | 97 | | Reduced power to plug main condenser tubing. | A | 5 | Steam & Power (HC) | Heat Exchangers (Main Condenser) | N1.1.4 |
| 13 | 8/18 | 0 | 95 | | Reduced power to plug main condenser tubing. | A | 5 | Steam & Power (HC) | Heat Exchangers (Main Condenser) | N1.1.4 |
| 14 | 8/28 | 0 | 40 | | Reduced power to plug main condenser tubing. | A | 5 | Steam & Power (HC) | Heat Exchangers (Main Condenser) | N1.1.4 |
| 15 | 9/4 | 0 | 70 | | Reduced power for main condenser maintenance. | A | 5 | Steam & Power (HC) | Heat Exchangers (Main Condenser) | N1.1.4 |
| 16 | 12/25 | 0 | 75 | | Reduced power for main condenser maintenance. | A | 5 | Steam & Power (HC) | Heat Exchangers (Main Condenser) | N1.1.4 |

Table A1.10 1979 Forced Shutdowns and Power Reductions for Millstone 1

| No. | Date (1979) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 1/6 | 269 | 100 | | Stress corrosion cracking of clean-up return line. | A | 1 | Reactor Coolant (CG) | Pipes, Fittings | N1.1.4 |
| 2 | 1/19 | 30 | 70 | | Change automatic pressure relief valve topworks. | A | 1 | Engineered Safety Features (SF) | Valve Operators | N1.1.4 |
| 3 | 1/22 | 0 | 97 | | Reduced power for main condenser maintenance. | B | 5 | Steam & Power (HC) | Heat Exchangers (Main Condenser) | N1.1.4 |
| 4 | 2/26 | 33 | 100 | | Safety relief valve lifted prematurely and failed to reseat. | A | 2 | Reactor Coolant (CC) | Valves | D6.1 |
| 5 | 3/10 | 0 | 97 | | Reduced power for main condenser maintenance. | B | 5 | Steam & Power (HC) | Heat Exchangers (Main Condenser) | N1.1.4 |
| 6 | 3/17 | 10 | 100 | | MSIV position indicating problems. | A | 9 | Reactor Coolant (CD) | Instrumentation & Controls | N2.4 |
| 7 | 6/28 | 0 | 15 | | Turbine trip due to feed-water regulating valve lock-up. | A | 3 | Reactor Coolant (CH) | Valves | D2.3 |
| 8 | 7/2 | 30 | 90 | | Loss of both plant air compressors. | A | 3 | Auxiliary Process (PA) | Blowers (Compressors) | N1.1.4 |
| 9 | 8/15 | 0 | 70 | | Reduced power for main condenser maintenance. | B | 5 | Steam & Power (HC) | Condenser | N1.1.4 |

Table A1.10 (Continued)

| No. | Date (1979) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DRE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 10 | 12/19 | 52 | 100 | | Main generator loss of excitation. | A | 3 | Steam & Power (HA) | Generators (Main Generator Exciter) | D2.3 |

Table A1.11 1980 Forced Shutdowns and Power Reductions for Millstone 1

| No. | Date (1980) | Duration (H[illegible]) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 1/5 | 6[illegible] | [illegible]-40 | | Reduced power to 40% due to isolation condenser out of service. | A | 5 | Reactor Coolant (CE) | Heat Exchangers | N1.1.3 |
| 2 | 6/25 | 13 | [illegible] | | Electric pressure regulator malfunction induced APRM scram. | A | 3 | Reactor Coolant (CC) | Instrumentation & Controls | D2.1 |

Appendix A: Millstone 1

Part 2. Reportable Event Coding Sheets

Table A2.1 Coding Sheet 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 | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| RS-70-4 | 58005 | 11/8/70 | 11/17/70 | B | EE | DD,NN | M | C | BK | B | - | Gas turbine generator failed during test due to LTA pressure in lube oil pump. |
| AO-70-5 | 60225 | 12/3/70 | 12/14/70 | B | CE | H,OO | M | C | BJ | B | - | Isolation condenser isolated during tests due to design error (NS 12:3 p 248). |
| RS-70-6 | 58011<br>57474 | 11/19/70 | 11/25/70 | B | CD | OO(2) | - | C | OJ | G,H | - | Steam isolation valve failure improper maintenance procedure and operator error combined (Reactor Shutdown). |
| RS-70-06 | 59927 | 12/9/70 | 12/19/70 | B | CC | Z | - | B | AH | B | E3 | Excessive pipe movement during transient due to design error. |
| RS-70-06 | 59927 | 12/9/70 | 12/19/70 | B | CD | OO(2) | - | B | AL | E | | MSIV closed due to missing spring in solenoid valve. |
| RS-70-06 | 59927 | 12/9/70 | 12/19/70 | B | HE | OO | - | B | AC | D | - | Steam bypass valve caused malfunction of pressure control. |
| AO-70-7 | 59595 | 12/23/70 | 1/2/71 | B | RB | J | K | C | AM | B | - | Approved wiring change not tested resulting in loss of full rod control |
| AO-70-8 | 59595 | 12/23/70 | 1/2/71 | B | HC | H,Z | - | C | AO | E | - | Welding error caused decrease in condenser vacuum (Reactor Shutdown) |

Table A2.1 (continued)

| Number | NSIC Accession Number | Event Date* | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| AO-70-9 | 59594 | 12/29/70 | 1/6/71 | B | HE | QQ | - | B | AI | D | - | Main steamline bypass valve failure caused by broken linkage on valve operator (Reactor Shutdown). |
| AO-70-10 | 59593 | 12/31/70 | 1/8/71 | B | EE | T,DD | - | C | HK | E | - | Slow turbine start due to reinstallation error in lube oil pump disc. link. |

Table A2.2 Coding Sheet 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 | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| AO-71-1 | 62296<br>63204 | 1/19/71 | 2/1/71 | B | SF-D<br>SF-B | 00 | T | C | EI | E | S7 | Four simultaneous valve failures render ECCS inoperable. Reactor shut down. Cause-torque switches set too low. |
| AO-71-2 | 61450 | 1/19/71 | 1/25/71 | B | SF-B | 00 | - | C | AK | D | - | Galled threads on valve cause turbine control valve to close. |
| RS-71-17 | 62298 | 2/12/71 | 3/9/71 | B | SF, IA | 00 | I | A | OJ | H | - | Error during maintenance on level indicator. Reactor scram on low level. |
| AO-71-5 | 62279 | 2/21/71 | 3/3/71 | B | EE | S,T,NN | T,P | B | | D | - | Gas turbine generator fails to start |
| AO-71-6 | 62300 | 3/25/71 | 4/2/71 | B | CC | 00,NN | - | C | BI | B | - | Loss of pressure to valve during test reactor shut-down. |
| AO 71-7 | 63140 | 3/31/71 | 4/8/71 | B | SF-B | X,00 | - | C | ED | D | - | Motor failed on LPCI valve due to short in windings. |
| AO-71-8 | 63139 | 4/22/71 | 4/27/71 | B | EE | NN,00 | T | A | OK | A,G | - | Gas turbine generator inoperable due to a switch being left in the wrong position. |
| AO-71-9<br>AO-71-10 | 63125 | 5/1/71<br>5/2/71 | 5/12/71 | B | CG, WA<br>EB | D,U<br>X,00 | - | B | AU,AT | D | S2 | Three simultaneous failures in motor control center due to moisture. |
| AO-71-11 | 64435 | 5/25/71 | 6/4/71 | B | CH | 00 | K | B | HD | B,D | - | NS 12:6 (1971) p. 619 feedwater control valve problems. |

Table A2.2 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| AO-71-12 | 64436 | 5/27/71 | 6/4/71 | B | CJ,EE | OO,NN | T | B | EE,ED | E,D | SI | NS 12:6 (1971) p. 619 Failure of mechanical and electric valve regulator (Reactor Shutdown also, short circuit in speed switch on gas turbine generator. |
| - | 64810 | 6/24/71 | 7/12/71 | B | EA | - | P | B | EG,OF | F | - | Offsite relay tripped reactor inadvertently during thunderstorms (Reactor Shutdown). |
| AO-71-13 | 65552 | 8/13/71 | 8/19/71 | B | SA ,IA | - | T | C | - | E,G | - | Drywell high pressure switch had loose hold down screws. |
| AO-71-14 | 66469 | 8/30/71 | 9/9/71 | B | CD | OO | - | B | AG | D | - | Plunger in air slide valve resisted movement. |
| - | 67211 | 9/22/71 | 9/22/71 | B | WA,SF | DD | - | C | OA | D | - | Insufficient head service water pumps. |
| AO-71-15 | 67459 67460 | 9/17/71 | 9/28/71 | B | SF-B | X | - | C | AH | B | | 2 motors burned out for same test |
| AO-71-16 | 67459 | 9/18/71 | 9/28/71 | B | SF-B | OO | - | C | AC | D | - | LPCI outboard cont. spray valve fails due to pins in wrong position. |
| AO-71-17 | 67461 | 9/27/71 | 9/30/71 | B | SF,WA | DD | - | C | OA | D | - | Emergency service water pumps provide LTA head. |
| AO-71-17 | 68264 | 9/21/71 | 9/31/71 | B | SF,WA | DD | - | C | OA | B | - | Service water pumps LTA head. |
| AO-71-18 | 67993 | 9/23/71 | 10/4/71 | B | HE | NN,OO | - | B | AB | B | - | Turbine bypass valve fails Brittle fracture of studs on linkage, reactor scrammed. |

Table A2.2 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| AO-71-19 | 67992 | 9/29/71 | 10/8/71 | B | HE | NN,OO | M | B | - | G | - | Failure of steam turbine bypass valve, reactor shutdown. See NS 13:2 p. 137. |
| AO-71-20 | 66996 | 10/10/71 | 10/22/71 | B | CC,HE | NN,OO | M | B | AP,BL | B | S7 | Failure of steam turbine bypass valve reactor blowdown (see NS 13:2 p. 137). |
| AO-71-21 | 68303 | 10/13/71 | 10/20/71 | D | IE | DD | N | B | AA,BL | E | - | Two stack gas sample pumps fail off, third tagged out. |
| AO-71-22 | 68304 | 10/13/71 | 10/20/71 | D | SF | OO | - | C | AP | B | - | Rigid air-lines should have been flexible. |
| AO-71-23 | 68305 | 10/16/71 | 10/20/71 | D | CF | OO | - | C | AP | G | - | Valve actuator failed due to loose parts from vibration. |
| AO-71-24 | 68788 | 11/2/71 | 11/11/71 | B | EE | NN | T | C | AL | E/C | - | Gas turbine generator fails to start due to loose connection. |
| - | 68307 | - | 11/16/71 | B | RB | DD | - | C | AT | E | - | Pump in standby liquid control system leaked. |
| AO-71-25 | 68308 | 11/30/71 | 12/9/71 | B | EE | NN,T | U | C | HB,OJ | H | S8 | Cold lube oil caused gas turbine trip off. Op. did not turn on oil htrs. |
| AO-71-26 | 69199 | 12/10/71 | 12/20/71 | B | CE | H | - | C | AN | B | - | Stripped threads on yoke sleeve caused isol. cond. valve to fail closed. |
| AO-71-27 | 69318 | - | 12/27/71 | B | TB | - | M,P | B | AL | G | - | Electric pressure regulator fails (Reactor Shutdown). |

Table A2.3 Coding Sheet 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 | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| AO-72-1 | - | 2/3/72 | 2/15/72 | B | CC,IB | - | M | C | OA | B | - | Main steamline differential pressure sensors do not meet manufacturers specs. |
| AO-72-2 | 39368 | 2/4/72 | 2/15/72 | B | IB | QQ,P | M | B | AQ | G | - | Filter clogged in electric pressure regulator pilot valve (Reactor Shutdown). |
| AO-72-3 | 39314 | 2/4/72 | 2/15/72 | B | EE | NN | - | B | EF | E | - | Gas turbine vibration monitor fails. |
| AO-72-4 | 39195 | 2/8/72 | 2/29/72 | B | CC | QQ | M | C | EI | G | - | Set point drift in pressure relief valve controller. |
| AO-72-5 | 39317 | 2/11/72 | 2/15/72 | B | SF-B | QQ | M | B | EG | D | - | Failure of LPCI low pressure switch. |
| AO-72-6 | 39177 | 2/18/72 | 2/28/72 | D | CC | H | M | C | EH | D | - | Set point drift causes pressure sensors in condenser to trip too high. |
| AO-72-7 | 39177 | 2/19/72 | 2/28/72 | D | CC | - | M | C | EH | D | - | Set point drift in main steamline pressure sensors. |
| AO-72-8 | 39218 | 2/23/72 | 3/2/72 | B | CC | - | M | C | AI | B | - | ΔP sensor failed,reactor shutdown. |
| AO-72-9 | 55348 | 3/3/72 | 3/13/72 | B | SF-B | OO | T | C | - | D | - | Cleanup auxiliary pump bypass valve inoperable. |
| AO-72-10 | 55347 | 3/6/72 | 3/13/72 | B | CE | QQ,DD | T | B | EI | D | - | Loss of monitoring capability of cont. isol. valves. |

Table A2.3 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| AO-72-11 | 55353 | 3/9/72 | 3/20/72 | B | EE | NN | T | B | EG | G | - | Gas turbine generator fails to start-faulty speed switch. |
| AO-72-12 | 70045 | 3/11/72 | 4/20/72 | B | IA | - | T | B | EH | D | - | Set point drift of level sensors. |
| AO-72-13 | 70045 | 4/12/72 | 4/20/72 | B | IA | - | T | B | EH | D | - | |
| AO-72-14 | 70045 | 4/18/72 | 4/20/72 | B | CC | - | M,T | C | EH | D | - | Set point drift of pressure sensors in main steamline. |
| AO-72-15 | 71716 | 6/8/72 | 6/8/72 6/19/72 | B | CE | QQ,H, OO | - | C | - | G | - | Improper assembly of valve. |
| AO-72-16 | 72424 | 6/24/72 | 7/5/72 | B | IA | - | T | C | EH | D | - | Set point drift of high-flow switch. |
| AO-72-17 | 72561 | 7/13/72 | 7/14/72 | B | IA | - | M | C | EI | D | S3 | Three pressure sensors trip above tolerance limit. |
| - | 72562 | 6-12-72 | 7-13-72 | B | ZZ | KK | - | B | AE | D | - | Pipe hangers shifted. |
| AO-72-18 | 72774 | 7/19/72 | 7/20/72 | B | SF | - | P | C | EI | G | S3 | 4 of 5 time delay relays do not meet specs. |
| - | 73637 | - | 8/9/72 | B | EE | N | M,T | C | EH | D | - | Set point drift causes DG to fail to start. |
| AO-72-19 | 73638 | 8/10/72 | 8/11/72 | B | RA | - | I | C | EH | D | - | Set point drift of reactor vessel low level switch. |
| AO-72-20 | 75492 | 8/15/72 | 8/23/72 | B | CE | H | T | C | EH | D | - | Isolation condenser flow switches set point drift. |

Table A2,3 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| AO-72-21 | 75398 | 8/31/72 | 8/31/72 | B | CB | DD,FF | - | B | AW | D | - | Excessive coolant rate leak due to bad pump seals. |
| AO-72-22 | 75078<br>75399<br>75755<br>77519 | 9/1/72 | 9/11/72 | B | HC | H | - | B | AR,AU,OF | B | S7 | Main condenser tubes leak salt water into coolant seals. |
| AO-72-23 | 75400 | 9/11/72 | 9/13/72 | C | IA | - | T,I | C | EI | G | - | Low-low water level detector out of calibration. |
| AO-72-24 | 75345 | 9/25/72 | 9/6/72 | B | SC | OO | - | C | AO,AV | E | - | Crack in header of atmospheric control. |
| AO-72-25 | 75795 | 10/12/72 | 10/25/72 | C | CE | H | T,M | C | EH | D | - | Isolation condenser pressure switch set point drift on 3/4 sensors. |
| AO-72-26 | 75910 | - | 10/21/78 | C | CH | Z | - | C | AV | - | - | 2 feedwater sparger leaks. |
| AO-72-27 | 75902 | - | 10/28/72 | C | CA | OO | - | C | OA | G | - | Penetration leaks and isolation valve leaks during testing. |
| - | 76011 | 8/25/72 | 9/23/72 | B | EA | LL | - | B | - | - | S7 | Plane crashes into power transformer. |
| | 76071 | | 10/13/72 | C | CE | H | T | C | EH | D | - | Set point drift of 3 isolation cond. switches. |

Table A2.4 Coding Sheet 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 | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| - | 77956 | - | 1/26/73 | B | RB | I,A | - | B | AT,AR | B,G | S7 | Accumulators for control rod drive found to be bad, all replaced (see NS 14:4 p. 373). |
| AO-73-1 | 80136 | 3/7/73 | 3/8/73 | B | RB | I | - | C | BI | D | - | Excessive scram time 2/145 C.R.'s. |
| AO-73-2 | 79560 | 3/14/73 | 3/21/73 | B | RB,EE | S,T,H, NN | P | B | EI | G | - | Scram on blown CRD fuse; gas-turbine fails to start. |
| AO-73-3 | 80118 | 3/21/73 | 3/27/73 | B | CH | DD,D | - | B | AB | G | - | Inboard bearing failure. |
| AO-73-4 | 80274 | 4/5/73 | 4/12/73 | B | CE | H,QQ | T | B | BC,HA, OK | G | S7 | Isolation condenser isolated on high flow due to incorrect maintenance. |
| AO-73-5 | 80275 | 4/5/73 | 4/13/73 | B | EE | T,NN | - | C | OJ,OA | H | - | Gas turbine generator fails to start after isolation condenser failure. Operator made wrong adjustment during startup. |
| AO-73-6 | 80728 | 4/18/73 | 5/18/73 | D | CF | QQ | - | B | AD,AI,AM | B | - | Valve operator broke on shutdown cooling system isolation valve improper sized motor. |
| - | 80710 | - | 5/21/73 | D | CH | Z | - | C | AV | D | - | Report on spargers crack in feedwater system. |
| - | 84795 | 7/18/73 | 7/18/73 | C | ZZ | GG | - | C | AT | G | - | Shock suppressors leaked hydraulic fluid, deterioration of seals. |

Table A2.4 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| - | 82972 | - | 7/13/73 | C | CB | OO | - | B | BA | D | - | Recirc. vent valve line leaks. |
| AO-73-8 | 82957 | 8/2/73 | 8/2/73 | C | SF-B | - | M,T | C | EH | D | - | Set point drift on DP sensors of LPCI. |
| - | 82966 | - | 8/8/73 | C | RB | J | - | C | CA | D | - | Control rod uncoupled. |
| - | 82968 | - | 8/6/73 | C | CH | Z | M | C | AU | D | - | Leak in pressure sensing line. |
| AO-73-9 | 83225 | 8/13/73 | 8/20/73 | C | IA | - | I | B | AM | | - | Level sensors read wrong due to mismatch in ventilation around sensor |
| - | 84017 | - | 8/30/73 | C | SF-D | DD,QQ | - | C | BA,EC | D | - | Failure of core spray valve to open. |
| AO-73-10 | 84491 | 9/21/73 | 10/1/73 | B | IA | OO | I | B | AT | D | - | Mismatch in level indicators due to valve leak. |
| AO-73-11 | 84497 | 9/22/73 | 10/1/73 | B | CH | D | - | A | BL | G | - | Condensate booster pump bearing over heats-rework bearings. |
| - | 84551 | - | 11/4/73 | B | SH-B | - | T,I | C | - | D | - | Level switch failed to trip. |
| AO-73-12 | 88079 | 12/30/73 | 1/7/74 | B | CE | Q | - | B | AU | D | - | Steam leak in isolation condenser flange (Reactor Shutdown). |

Table A2.5 Coding Sheet 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 | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| - | 93695 | 1/6/74 | 2/1/74 | B | CA | OO | - | B | AB | D | - | Pressure relief valve lifts due to worn preload ass'y. |
| - | 89258 | - | 3/8/74 | B | ZZ | GG,FF | - | B | AT | B | - | Snubbers leak. |
| AO-74-1 | 89340 | 3/6/74 | 3/15/74 | B | CB | BB,OO,FF | - | B | AU | D | - | Drywell leak due to seal packing failure (Reactor Shutdown). |
| - | 89663 | - | 4/1/74 | B | CB | DD,QQ,OO | P | B | BB,OK | G | - | Pump suction valve on recirc. pump fails to close. |
| AO-74-2 | 92184 | 5/17/74 | 5/28/74 | B | CH | DD,D | - | B | AB | B | - | Inboard bearing FWCI condensate pump 2nd failure. |
| AO-74-4 | 95140 | 8/28/74 | 9/5/74 | B | SH-D | FF | - | C | AW | D | - | Tear in seal at Rx bldg-turbine bldg interface. |
| AO-74-6 | 95426 | 9/18/74 | 9/25/74 | C | CB | Z | - | C | AO | B,D | - | Weld cracks in recirc. loop discharge bypass lines. |
| AO-74-8 | 97077 | 10/31/74 | 11/6/74 | C | WA, SF-B | OO | - | C | BA | B | - | Emergency service water valve fails due to manual overtorqueing. |
| AO-74-9 | 97076 | 11/4/74 | 11/12/74 | B | CD | QQ,P | - | C | HC | G | - | Foreign material in air slide valve disables valve operator (Reactor Shutdown). |

Table A2.5 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| AO-74-10 | 97506 | 11/15/74 | 11/18/74 | B | CD | QQ,P | - | C | HC | C | - | Air slide valve clogs-second time in one month (Reactor Shutdown). |
| AO-74-11 | 98161 | 12/13/74 | 12/19/74 | B | CG | QQ | - | A | AB,BC | D | - | Misalignment of gear train on valve motor op. |
| AO-74-12 | 98582 | 12/20/74 | 12/30/74 | B | SD | H | I | C | EH | D | — | Set point drift on level sensor in isolation condenser. Set point lock installed. |
| AO-74-13 | 98581 | 12/20/74 | 12/30/74 | B | HC | H | M | C | EH | D | - | Set point drift on condenser vacuum swtiches |
| AO-74-14 | 98705 | 12/23/74 | 1/2/75 | B | SF-B | MM | I | B | HC,BT | G | - | Level sensor tubing blocked. Blowout tube installed. |
| AO-74-15 | 98706 | 12/28/74 | 1/2/75 | B | ZZ | CG | - | C | AT | E | | 2 snubbers had no fluid. Improper installation. |

Table A2.6  Coding Sheet 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 | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| - | 99684 | - | 1/24/75 | B | WC | M | - | B | OF | D | S9 | Hydrogen explosion in acid day tank due to moisture. |
| AO-75-2 | 93569 | 1/25/75 | 1/30/75 | B | CC | Z | E | B | AD | D | - | Flow indicator reading decrease due to weld crack in sensing tube. |
| AO-75-4 | 93515 | 1/29/75 | 1/31/75 | B | EE | NN,T | P | B | EF | D | - | Spurious operation of relay causes turbine startup sequence to begin. |
| AO-75-5 | 101408 | 3/27/75 | 4/3/75 | B | WF,MA | V | - | B | AW,OD | E | - | Wiring error caused flow of contaminate into boiler system. |
| AO-75-6 | 101701 | 3/30/75 | 4/8/75 | B | MA | - | - | B | OD,OK | A | S8 | Inadvertent discharge of radioactive liquid due to procedure error. |
| AO-75-7 | 102297 | 4/18/75 | 4/24/75 | B | EE | T,C | P | B | EB | D | - | Oscillator board trips inverter operation. |
| AO-75-8 | 103148 | 5/20/75 | 5/29/75 | B | EE | NN,T,QQ | T,U | C | EI,BL | G | - | Incorrect valve position plus temp. sensor error cause gas turbine trip. |
| AO-75-9 | 104066 | 5/20/75 | 7/11/75 | B | CC | OO | - | B | BP,BB | D | - | Safety/relief valve fails to reset - no cause found (Reactor Shutdown). |
| AO-75-10 | 103074 | 5/21/75 | 5/28/75 | B | ZZ | GG,FF | - | A | BT | B,E | - | No fluid in two snubbers due to improper assembly. |

Table A2.6 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| AO-75-11 | 103879 | 6/26/75 | 7/2/75 | B | CC | - | T,M | C | AR,AU | G | - | Moisture from leak causes switch failure. |
| AO-75-14 | 105835 | 7.14/75 | 7/22/75 | B | CE | H | E,T | C | EH | D | - | Set point drift in flow sensors on isolation condenser. |
| AO-75-15 | 104868 | 7/22/75 | 8/1/75 | B | CC | MM | M | B | HD | B | - | Variation in diff. pressure data due to sensing tube failure. |
| AO-75-16 | 106457 | 8/16/75 | 8/22/75 | B | IA | - | M,T | C | EH | D | - | Set point drift on vessel high pressure switch. |
| AO-75-17 | 105547 | 8/18/75 | 8/26/75 | B | CC | QQ | - | C | AQ | G | - | Safety/relief valve bellows integrity could not be verified due to grit in air lines. |
| AO-75-18 | 106343 | 9/8/75 | 9/15/75 | B | IB | BB | M,T | C | EH | D | - | Set point drift of dry-well high pressure switches. |
| AO-75-19 | 107502 | 10/12/75 | 10/21/75 | C | IA | - | U,T | C | EH | D | - | Set point drift on steam tunnel temp. sensors. |
| AO-75-20 | 107501 | 10/13/75 | 10/22/75 | C | IE | - | N,T | C | EH | D | - | Set point drift of ventilation radiation monitor. |
| AO-75-21 | 108248 | 11/13/75 | 11/25/75 | B | CB | Z | - | B | AO,AI,AW | B | - | Instrument pipe weld crack due to fatigue. |
| AO-75-22 | 108520 | 11/24/75 | 12/3/75 | B | HC | - | M,T | C | EH | D | - | Set point drift in condenser low vacuum switch. |

Table A2.6 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| AO-75-23 | 108889 | 12/9/75 | 12/18/75 | B | IB | BB | M,T | C | EH | D | - | Set point drift of dry-well high pressure switches. |
| AO-75-24 | 108888 | 12/19/75 | 12/23/75 | B | CD | - | P,T | C | BC | G | - | Valve position switch out of position caused MSIV close relay to fail. |
| AO-75-25 | 109191 | 12/17/75 | 12/23/75 | B | IB | - | P | C | EH | D | - | Relay set point drift. |
| AO-75-26 | 109454 | 12/30/75 | 1/7/76 | B | WA | P | - | C | AQ | G,A | - | Grit in strainer caused emergency service water pump trip. |
| AO-75-27 | 109459 | 12/23/75 | 1/7/76 | B | SF-B | - | M,T | C | EH | D | - | Set point drift in LPCI pressure switch. |
| RO-75-28 | 110932 | | 12/28/76 | B | CG | JJ | - | B | AR | D | - | Hole in cleanup filter sludge tank. |

Table A2.7 Coding Sheet 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 | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| RO-76-2 | 111648 | 1/26/76 | 2/25/76 | B | SA | F | - | C | AA | D | - | Vacuum breaker fails due to teflon bushing out of round. |
| RO-76-3 | 111658 | 2/10/76 | 3/9/76 | B | EC | H | M,T | C | EH | D | - | Set point drift on condenser low vacuum switch. |
| RO-76-4 | 111647<br>120080 | 2/12/76 | 3/5/76<br>9/28/76 | B | CD | H | - | B | AR | A,B | S7 | Isolation condenser tube failure due to corrosion cracking<br>NS 17:4 pp. 494-495. |
| RO-76-5 | 112163 | 2/12/76 | 3/12/76 | D | CH | QQ | M | B | AI | D | - | Feedwater regulation valves lock due to diaphragm failure in regulator. |
| RO-76-7 | 112308 | 2/19/76 | 3/15/76 | B | CC | NN | M,T | C | EH | D | - | Instrument drift caused turbine relays to trip. |
| RO-76-8 | 1111911 | 2/29/76 | 3/15/76 | B | EE | NN,T | - | C | BC | G | - | Gas turbine governor out of adjustment. |
| RO-76-10 | 112309 | 3/8/76 | 3/22/76 | B | EE | NN,T | - | C | BC | G | - | Gas turbine generator governor out of adjustment, (Reactor Shutdown). |
| RO-76-11 | 112162 | 3/9/76 | 3/23/76 | B | ZZ | GG | - | C | BT,AU | G | - | Shock suppressor low on fluid. |
| RO-76-16 | 113541 | 4/22/76 | 5/4/76 | B | CD | OO | - | C | AG | D | - | Primary containment isolation valves fail to close due to internal binding. |

Table A2.7 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| RO-76-17 | 113540 | 4/23/76 | 4/30/76 | B | RB | R | - | B | OG | B | C3 | Excess stack gas release due to fuel clad perforations. |
| RO-76-23 | 114643 | 5/28/76 | 6/11/76 | B | CD | H | - | B | BU | E | - | Increase in chloride ion concentration in cond. return leg. |
| RO-76-27 | 11572[illegible] | 6/23/76 | 7/9/76 | B | IA | - | M,T | B | EH | D | - | Set point drift in pressure switch. |
| RO-76-28 | 116269 | 7/14/76 | 7/30/76 | D | RC | QQ | - | C | CA | E | - | Isolation condenser inboard isolation valve inoperable due to previous fix for leak. |
| RO-76-29 | 116780 | 8/10/76 | 8/24/76 | B | EE | NN,T | - | C | EA,BF | B | S8,S7 | Gas turbine generator trips out on incorrect AC feed. Complete caps. |
| RO-76-30 | 117678 | 8/31/76 | 9/8/76 | B | EE | NN,T | - | C | BF | D | - | Gas turbine generator trips on overspeed. Speed switch faulty. |
| RO-76-31 | 117679 | 8/13/76 | 9/9/76 | D | IA | - | L | C | EH | D | - | Set point drift in intermediate range monitors. |
| RO-76-32 | 118795 | 9/25/76 | 10/8/76 | B | SF-B | PP | - | B | AW | G | - | Packing leak in LPCI testable check valve. |
| RO-76-33 | 120533 | 10/28/76 | 11/23/76 | C | RB | R | - | C | AD | D | - | Neutron source rod broken. |
| RO-76-34 | 120436 | 11/12/76 | 11/24/76 | C | RB | I | - | C | OJ,OK | A,H | S8 | Inadvertent criticality due to operator error during shutdown margin test. |

Table A2.7 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| RO-76-35 | 120272 | 11/9/76 | 12/3/76 | C | CC | - | U,T | C | EH | D | - | Set point drift of steam tunnel temp. switch. |
| RO-76-36 | 120228 | 11/22/76 | 11/27/76 | C | SF-D | Z | - | C | AO,AV | B | - | Weld joint leak in head spray system due to stress corrosion. |
| RO-76-37 | 120670 | 12/7/76 | 12/30/76 | B | IA | - | M,T | C | EH | D | - | Set point drift of dry-well pressure switches. |
| RO-76-38 | 120669 | 12/18/76 | 12/30/76 | B | SH-B | - | M,T | C | EH | D | - | Set point drift of containment spray pressure switches. |
| RO-76-40 | 121055 | 12/14/76 | 1/12/77 | B | CC | - | M,T | C | EH | D | - | Set point drift in main steam line flow switch. |
| RO-76-42 | 121034 | 12/17/76 | 1/14/77 | B | CE | QQ | T | C | BB,EI | G | - | Isolation condenser steam supply valve fails to close due to incorrect torque switch setting (Reactor Shutdown). |
| RO-76-43 | 121035 | 12/18/76 | 1/14/77 | B | CE | QQ | T | C | BB,EI | G | - | Isolation condenser condensate return valve torque switch miscalibrated. |
| RO-76-44 | 121036 | 12/18/76 | 1/14/77 | B | CG | QQ,X | - | C | AH,BA,OK | G | - | Procedural error caused overload of motor on valve. |

Table A2.8 Coding Sheet 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 | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| RO-77-2 | 122206 | 1/8/77 | 2/7/77 | B | BB | AA,S | N | B | EA | B | S1 | Both stack gas monitors inoperable due to blown fuse in power supply |
| RO-77-3 | 122174 | 1/12/77 | 2/10/77 | B | IA | - | M,T | C | EH | D | - | Set point drift in dry-well pressure swtich. |
| RO-77-4 | 122136 | 1/25/77 | 2/23/77 | B | SA | QQ,MM | - | B | AQ,BB | G | - | Drywell vent bypass valve fails to close due to grit in air operator line |
| RO-77-5 | 122137 | 1/26/77 | 2/24/77 | B | CB | M | - | B | BU | G | - | High coolant conductivity due to improper rinsing of resin in demineralizer (Reactor Shutdown). |
| RO-77-6 | 122138 | 1/28/77 | 2/28/77 | B | CE | H | - | A | OF | I | - | Impending storm conditions halted maintenance of isolation condenser. |
| RO-77-7 | 122429 | 2/1/77 | 3/1/77 | B | EE | Y,T | - | C | OA,AT,AV | D | - | Diesel generator in-operable due to fuel oil leak-cracked nipple. |
| RO-77-8 | 122139 | 2/1/77 | 2/28/77 | B | CE | H | - | A | BU | B | - | High chloride ion concentration in isolation condenser. |
| RO-77-9 | 123022 | 2/11/77 | 3/11/77 | B | IA | - | I,T | C | EH | D | - | Set point drift in reactor water level switch. |
| RO-77-10 | 123023 | 2/14/77 | 3/11/77 | B | IA | BB | M,T | C | EH | D | - | Set point drift in dry-well pressure swtich. |
| RO-77-11 | 129462 | 2/14/77 | 3/14/77 | B | CC | - | M,T | C | EH | D | - | Set point drift in main steam line pressure switch. |

Table A2.8 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| RO-77-12 | 123024 | 2/15/77 | 3/14/77 | B | HC | H | M,T | C | EH | D | - | Set point drift in condenser vacuum switch |
| RO-77-13 | 124872 | 3/18/77 | 3/30/77 | B | SF-B | - | - | C | BV | A | - | Low concentration in standby liquid control system due to low temp in storage tank |
| RO-77-14 | 125214 | 5/3/77 | 5/17/77 | B | IA | - | M,T | C | EH | D | - | Set point drift in reactor low pressure pump start permissive switch |
| RO-77-15 | 125177 | 5/14/77 | 6/7/77 | B | SH-D | S | - | B | EH,OA | D | - | Standby gas treatment system circuit blown fuse. |
| RO-77-16 | 125594 | 6/14/77 | 6/27/77 | B | CB | Z | - | A | AW | D | - | Leak in recirc loop drain - cause unknown |
| RO-77-17 | 130698 | 6/17/77 | 6/17/77 | B | CC | OO | - | C | AY | - | - | Safety/relief valve opens - no cause known |
| RO-77-18 | 126008 | 6/18/77 | 7/1/77 | B | CC | OO,P | - | B | AT | D | - | Safety/relief valve seat leakage due to filter failure |
| RO-77-19 | 126489 | 6/15/77 | 7/15/77 | B | CE | H | M,T | C | EH | D | - | Set point drift in isolation condenser actuation press. switch |
| RO-77-20 | 143504 | 7/14/77 | 7/26/77 | B | BB | DD | - | C | AC | D | - | Cooling fan deterioration causes loss of stack sampler |
| RO-77-21 | 143468 | 7/13/77 | 8/5/77 | B | CH | OO | - | B | AC | D | S7 | Degradation of valve diaphragm causes loss of full HPCI Capability |

Table A2.8 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER 77-22 | 143507 | 7/21/77 | 8/19/77 | B | CC | OO,G | A | B | ED | D | - | Safety/relief valve bellows alarm received due to wiring short circuit. |
| RO-77-23 | 143469 | 8/5/77 | 8/31/77 | B | IC | F | - | B | AB | B | - | Three vacuum breakers fail to close due to friction |
| RO-77-24 | 143470 | 8/6/77 | 8/30/77 | B | IX | OO | - | B | AC | B | - | Reactor scram on loss of instrument air. Gasket failure (Reactor shutdown) |
| LER 77-25 | 143509 | 8/7/77 | 8/30/77 | B | CC | OO | K,T | C | AC | D | - | Pressure switch on bellows monitor fails |
| LER 77-27 | 143511 | 9/9/77 | 10/7/77 | B | EE | MM | - | C | EG | D | - | Spurious noise halts gas turbine startup tests. |
| LER 77-28 | 143512 | 9/12/77 | 10/11/77 | B | SD | H | M,T | B | EH | D | - | Set point drift in isolation condenser pressure switches |
| RO-77-29 | 143561 | 9/27/77 | 10/14/77 | B | EE | N,MM | - | B | AT | D | - | Diesel generator inoperable due to fuel oil leak |
| RO-77-30 | 143471 | 10/12/77 | 11/2/77 | B | SH-D | FF | - | C | AV | D | - | Standby gas treatment system inoperable due to leak in seal |
| LER 77-32 | 143555 | 11/01/77 | 11/15/77 | B | BB | DD | - | B | AA | D | - | Stack sample pump trips due to normal pump |
| LER 77-34 | 143546 | 10/28/77 | 11/23/77 | B | CC | OO | A | B | ED | E | - | Bellows leakage alarm sounds due to wiring short circuit |

Table A2.8 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| RO-77-35 | 143472 | 10-31-77 | 11-29-77 | B | CE | QQ | - | B | OA | - | - | Isolation condenser steam supply valve inoperable due to unknown cause |
| LER 77-36 | 143547 | 11/7/77 | 12/2/77 | B | IA | - | I,T | C | AG,AQ | G | - | Low-low level switch fails to trip due to grit. |
| RO 77-37 | 144121 | 11/30/77 | 3/3/78 | C | ZZ | GG | - | C | OA | E | - | Two snubbers declared inoperable |
| RO 77-38 | 143485 | 11/10/77 | 12/9/77 | B | SF-B | QQ | - | C | ED | G | S8 | Maintenance foreman inadvertently short circuits valve operator on LPCI valve |
| RO 77-39 | 144187 | 12/10/77 | 12/12/77 | B | EE | N | - | C | OA | - | - | Diesel generator declared inoperable - cause unknown |
| RO 77-40 | 144186 | 12/13/77 | 12/14/77 | B | SC | - | - | - | - | - | S9 | Two hydrogen explosions in off-gas system reactor shutdown |
| RO 77-43 | 133684 | 12/19/77 | 1/18/78 | D | BB | KK | N | A | BC | H | - | Stack gas sampler and structure damaged due to hydrogen explosion. |

Table A2.9 Coding Sheet 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 | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| RO-78-1 | 134979 | 1/16/78 | 2/14/78 | B | CC | - | M,T | C | EH | D | - | Setpoint drift in main steam line low pressure switches |
| RO-78-2 | 134961 | 1/17/78 | 2/14/78 | B | SH-D | P | - | C | AA | D | - | Pressure drop limit exceeded due to normal wear of filter |
| RO-78-3 | 136464 | 2/14/78 | 2/28/78 | B | HA | QQ | P | C | EH | D | - | Set Point drift in turbine control time delay relay |
| RO-78-4 | 142572 | 3/10/78 | 3/24/78 | C | IB | OO | - | B | BB | - | - | Safety/relief valve fails to reseat. Cause unknown |
| RO-78-5 | 137251 | 3/19/78 | 4/3/78 | C | CB | OO,Z | - | A | AO,AI,AT | B | - | Fatigue caused weld leak in recirc. discharge valve vent line |
| RO-78-6 | 137252 | 3/20/78 | 4/3/78 | C | CD | OO | - | A | AT | D | - | Containment isolation valve excess leakage |
| RO-78-7 | 137504 | 3/11/78 | 4/10/78 | C | CE | H,Z | | A | BS | G | - | Pipe movement in isol. condenser due to water in steam lines |
| RO-78-8 | 137504 | 4/17/78 | 4/20/78 | B | SF-B SF-D | KK | - | B | AH | B | - | Degradation of core spray and LPCI piping-support structure due to poor design |
| RO-78-9 | 138250 | 4/3/78 | 4/25/78 | B | CC | - | M,T | C | EH | D | - | Set point drift in steam tunnel temp. switches |

Table A2.9 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| RO-78-10 | 138838 | 4/24/78 | 5/23/78 | B | RB | J,OO | - | C | BI | E | - | Scram time too slow for control rods. Due to tight packing on valve |
| RO-78-11 | 139384 | 5/8/78 | 6/6/78 | B | IA | - | M,T | C | EH | D | - | Set point drift in containment pressure switch |
| RO-78-12 | 139640 | 5/19/78 | 6/16/78 | D | EE | NN,T | - | C | BI,EI | D | - | Gas turbine fails to complete startup sequence due to incorrect governor setting |
| RO-78-13 | 139817 | 5/29/78 | 6/20/78 | D | CE | H | - | B | BL | G | - | Steam trap blow by causes isolation condenser to be removed from system. |
| RO-78-14 | 139876 | 6/13/78 | 7/12/78 | B | EE | NN,T | T | B | BW | D | - | Gas turbine trips out due to overspeed switch failure |
| RO-78-16 | 140261 | 7/5/78 | 8/4/78 | B | HC | H | M,T | B | EH | D | - | Set point drift in condenser low vacuum switch |
| LER 78-16 | 140386 | 7/25/78 | 8/9/78 | B | SH-A | - | A | B | OK | G | S8 | Procedural error allowed containment to be purged without high radiation monitor. |
| RO-78-17 | 140032 | 7/10/78 | 8/4/78 | B | FB | - | N | C | EI | G | - | Calibration error in spent fuel storage air monitors. |
| RO-78-18 | 141762 | 9/5/78 | 10/4/78 | B | IA | - | I | C | AK | G | - | Lack of lubrication causes low low water level sensors to fail |

Table A2.9 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| RO-78-19 | 141763 | 9/12/78 | 10/10/78 | B | IA,HA | NN | P | C | EH | D | - | Set point drift in turbine control time delay relay |
| RO-78-20 | 148110 | 10/10/78 | 10/10/78 | B | WC | M | - | B | OA | - | - | Chlorine capacity of demineralizer less than tech. specs. |
| RO-78-21 | 141139 | 9/14/78 | 10/12/78 | B | EE | NN | T | C | BF | D | - | Faulty speed switch trips gas turbine |
| RO-78-22 | 141149 | 9/14/78 | 10/13/78 | B | ZZ | QQ,BB | - | B | AQ | G | - | Drywell vent bypass valve fails to close due to dirt in air operator. |
| RO-78-23 | 141726 | 10/11/78 | 11/6/78 | B | IA | BB | M,T | C | EH,AK | D | - | Setpoint drift in drywell high pressure switch |
| RO-78-24 | 141729 | 10/19/78 | 11/16/78 | B | CE | QQ,H | T | B | BF | G | - | Spurious opening of isolation condenser isolation valve due to set point drift |
| RO-78-26 | 142286 | 11/6/78 | 11/30/78 | B | IA | BB | M,T | C | EH | D | - | Set point drift in drywell high pressure switch |
| RO-78-27 | 142285 | 11/6/78 | 12/1/78 | B | IA | - | I | C | EH | D | - | Set point drift in reactor low level switch |
| RO-78-29 | 142849 | 11/22/78 | 12/22/78 | B | EE | NN,T | G | C | OA,AA | G | - | Gas turbine inoperable when maint. crew had to repair damaged indicator light socket |
| LER-78-30 | 146505 | 12/18/78 | 1/17/79 | B | ZZ | BB | - | - | OK,BU | A | - | Excess oxygen in drywell due to procedural error |

Table A2.10 Coding Sheet 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 | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER-79-1 | 146724 | 1/6/79 | 1/18/79 | D | CG | Z | - | A | AO,AV | D | - | Stress corrosion cracking of cleanup system return line weld (reactor shutdown) |
| LER-79-2 | 146643 | 1/7/79 | 2/6/79 | D | SH-D | - | P | B | BC,BD | E | - | SBGT system "A" fails to start due to misalignment of startup relay |
| LER-79-3 | 146644 | 1/8/79 | 2/7/79 | D | CC,ID | - | M,T | C | EH | D | - | Setpoint drift in main steam line pressure switch |
| LER-79-4 | 147428 | 1/25/79 | 2/23/79 | B | RB | I | - | C | AG | - | - | Control rod sticks. Cause unknown. |
| LER-79-5 | 147414 | 2/26/79 | 2/26/79 | B | CC | OO | - | D | AB | D | S7 | Pressure relief valve lifts prematurely and fails to seat due to steam cutting of disc (reactor shutdown) |
| LER-79-6 | 147295 | 2/1/79 | 3/2/79 | B | RB | I | - | B | OJ | H | S7 | Two control rods inoperable due to operator error |
| LER-79-7 | 147413 | 2/14/79 | 3/9/79 | B | EE | NN,T | T | C | BD | D | - | Gas turbine generator fails to start due to faulty speed switch. |
| LER-79-8 | 147415 | 2/14/79 | 3/14/79 | B | CE,ID | QQ | K | A | AD | D | - | RCIC valve loses position indication due to failure of valve operator casing |

Table A2.10 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER-79-9 | 148229 | 2/22/79 | 3/22/79 | D | CC | OO | G | C | OA,AV | D | - | Safety/Relief valve bellows monitor inoper, due to crack in air tube fitting |
| LER-79-10 | 149268 | 4/3/79 | 4/18/79 | B | RC | - | - | B | OJ | H | - | Core thermal power exceeds tech specs due to operator error |
| LER-79-11 | 149267 | 3/17/79 | 4/17/79 | B | CD | OO | P | C | BB,BC | D | - | MSIV relay fails to deenergize due to maladjustment of relay limit switch (reactor shutdown) |
| RO-79-11 | 152303 | 5/14/79 | 5/18/79 | C | SA | Z | - | A | AA | D | - | Leak rate of penetrations exceeded |
| LER-79-12 | 149544 | 4/23/79 | 5/21/79 | B | BA | - | N | C | EH | D | - | Set point drift in reactor building radiation monitor |
| LER-79-15 | 150110 | 5/28/79 | 6/22/79 | C | RB | J | K | C | AH | D | - | Control rod out block function fails due to worn limit switch |
| LER-79-16 | 150109 | 5/30/79 | 6/29/79 | C | IA | - | M,T | C | EH | D | - | Set point drift in reactor low pressure switch. |
| LER-79-17 | 150108 | 6/2/79 | 6/28/79 | C | IA | - | U,T | C | EH | D | - | Set point drift in steam tunnel temperature switch |

Table A2.10 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER-79-18 | 150748 | 7/13/79 | 7/26/79 | B | SF-B | - | - | A | OK,BU | A | S8 | Concentration in Standby liquid control system LTA due to procedural error |
| LER-79-19 | 150782 | 6/28/79 | 8/3/79 | D | CE | OO | - | C | AA,AU | D | - | Containment isolation valve leakage due to normal wear |
| - | 150990 | 2/16/79 | 3/19/79 | B | IA | - | G | B | BU,OK | A | - | Abnormal oxygen levels in containment due to cancellation of shutdown |
| LER-79-20 | 151245 | 7/11/79 | 8/9/79 | B | IA | - | I,T | C | EH | D | - | Set point drift in reactor low-low level switch |
| LER-79-21 | 151244 | 7/12/79 | 8/10/79 | B | IA | - | I,T | C | AQ | G | - | Low low water level switch fails to trip due to grit on shft assembly |
| LER-79-22 | 151211 | 7/23/79 | 8/22/79 | B | SC | QQ | - | B | BB | - | - | Containment vent bypass valve fails to close - cause unknown |
| LER-79-23 | 151207 | 8/8/79 | 8/24/79 | B | SC | QQ | - | B | BB | - | - | Same as above |
| LER-79-24 | 151425 | 7/31/79 | 8/30/79 | B | BA | - | N | C | EH | D | - | Set point drift in refueling floor radiation monitor |
| LER-79-25 | 151913 | 9/13/79 | 9/25/79 | B | RB | - | - | B | OK | B | - | Total peaking factor in error due to design oversight |

Table A2.10 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER-79-26 | 151912 | 9/14/79 | 9/27/79 | B | SF | - | - | B | OK | B | S7 | Potential existed for loss of power to ECCS to go undetected-design error |
| LER-79-27 | 151758 | 8/27/79 | 9/24/79 | B | AB | - | - | B | OC | G | - | System not tested acc'd to schedule |
| LER-79-28 | 151911 | 8/28/79 | 9/25/79 | B | CC | QQ | - | C | BB,AR | D | - | Pressure suppression chamber vent bypass valve fails to close due to rust buildup |
| LER-79-29 | 151930 | 9/4/79 | 10/4/79 | B | CE | QQ,H | T | C | BB | D | - | Isolation condenser isolation valve fails to close due to faulty microswitch |
| LER-79-30 | 152940 | 10/9/79 | 11/8/79 | B | EE | KK,N | - | B | AL | E | - | Service water system pipes to DG not restrained due to installation error |
| LER-79-31 | 153647 | 10/16/79 | 11/15/79 | B | SF-C | KK | - | B | OA,AL | E | - | Feedwater coolant injection system declared inoperable due to missing support structure |
| LER-79-32 | 153362 | 11/6/79 | 12/3/79 | B | IA | - | M,T | C | EH | D | - | Set point drift in drywell pressure switches |
| LER-79-33 | 153360 | 11/13/79 | 12/4/79 | B | IA,CC | - | M,T | C | EH | D | - | Set point drift in main steam line delta P switch |
| LER-79-34 | 153749 | 11/15/79 | 12/14/79 | B | SF-B | QQ,X | - | C | OA,ED | D | - | LPCI Valve inoperable due to electrical fault in motor controller |

Table A2.10 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER-79-35 | 153906 | 12/4/79 | 12/20/79 | B | IA | - | M,T | C | EH | D | - | Set point drift in main steam low pressure switch |
| RO-79-36 | 153942 | 12/19/79 | 1/18/80 | D | CE | Z,H | - | B | HH | D | - | Water hammer in isolation condenser piping |

Table A2.11 Coding Sheet 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 | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER-80-01 | 154444 | 1/3/80 | 2/1/80 | B | CB | DD | - | B | BW | - | - | Pump speed mismatch during recirc pump runback |
| LER-80-001 | 155196 | 1/31/80 | 2/14/80 | B | - | - | - | - | - | B | - | MAPLHGR wrong |
| LER-80-3 | 154614 | 1/4/80 | 2/4/80 | B | CB | KK | - | B | AH | B | - | Structural Deficiency in isolation cond. system supply line (Reactor shutdown) |
| LER-80-4 | 155195 | 1/21/80 | 2/14/80 | B | IA,CC | - | E,T | C | EH | D | - | Set point drift in main steam line high flow switch |
| LER-80-5 | 156005 | 2/20/80 | 3/17/80 | B | SF-D | KK | - | C | AC,HH | D | - | Core spray supports damaged by water hammer. |
| LER-80-6 | 156157 | 4/7/80 | 4/21/80 | B | IA | - | M,T | C | OK | E | - | Pressure sensor isolated due to installation error |
| LER-80-8 | 158283 | 6/5/80 | 7/2/80 | B | SA | - | M,T | C | EH | D | - | Set point drift in two reactor pressure switches |
| LER-80-9 | 158276 | 6/5/80 | 7/2/80 | B | IA | - | M,T | C | EH | D | - | Set point drift in reactor protection low pressure switches |
| LER-80-10 | 159128 | 6/13/80 | 7/11/80 | B | RA | KK | - | B | AC | D,E | - | Bolts on penetration base plate faulty due to installation error |
| LER-80-11 | 160233 | 8/3/80 | 8/15/80 | B | BB | OO | N | C | BC,OA | G | - | Off gas radiation monitoring system inoperable due to valve misalignment |

Table A2.11 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER-80-12 | 159289 | 7/25/80 | 8/22/80 | B | IC | - | N | C | EI | G | - | APRM reads low due to calibration error |
| LER-80-13 | 160459 | 9/8/80 | 10/3/80 | B | IA | - | M,T | C | EH | D | - | Set point drift in high pressure switch |
| LER-80-14 | 160060 | 10/05/80 | 10/16/80 | C | CD | OO | - | C | - | D | - | Leak rate test failure of two MSIV's |
| LER-80-15 | 160923 | 10/7/80 | 10/24/80 | C | IA | - | I,T | C | AQ | G | - | Set point drift in low water level scram switch - grit. |
| LER-80-16 | 161870 | 10/23/80 | 11/6/80 | C | CB | DD,KK | - | C | AH | C | - | Cracks found in jet pump support beams |
| LER-80-18 | 161470 | 11/5/80 | 11/19/80 | C | CC | Z | - | C | AO | D | - | Weld failure in two main steam lines |
| LER-80-19 | 161471 | 11/6/80 | 11/20/80 | C | CC | Y | - | C | AO | D | - | Condenser Nozzle weld cracks due to stress corrosion |

Appendix B (Continued)

| Report No. | Event date | Event description and problem solution |
|---|---|---|
| AO 76-8 | 2-29-76 | GTG did not start due to improper governor setting. Governor readjusted. |
| AO 76-10 | 3-3-76 | 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 dircuit breaker. Breaker indicator bulb replaced. |
| LER 79-7 | 2-14-79 | GTG fails to start due to faulty speed switch. Switch replaced. |

Appendix B. Gas turbine generator failures

| 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. |