DRAFT

NSIC-201

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

M. L. Casada, JBF Associates, Inc.
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Nuclear Safety Information Center

Engineering Technology Division

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CONTENTS

Page

## LIST OF TABLES

# LIST OF FIGURES

## 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 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 funciton 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 San Onofre 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 San Onofre 1.

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

M. L. Casada† A. B. Crawford*
K. H. Harrington† G. T. Mays*

## ABSTRACT

A review of the operating experience of the San Onofre 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 San Onofre 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 San Onofre 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 =

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

2. unit availability =

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

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

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

## 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:

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 (AO)],
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

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 |

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., AOs, 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),

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 | | |

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 Haddam Neck, 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 plants 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 1.6. 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 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 over-voltage/undercurrent condition other than open circuits |
| EC | Undervoltage/overcurrent: component failure produces an under-voltage/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 |
| | <u>Other</u> |
| OA | Declared inoperable: component or system is declared inoperable as required by Technical Specifications but may be capable of partially or completely performing its desired duties when requested (a component/system that is <u>completely</u> failed should not use this code) |
| OB | Flux anomaly: flux characteristics of the reactor core are not as required or desired (e.g., flux spike due to xenon burnout) |
| OC | Test not performed: operator or test personnel fails to perform a required test within the required period |
| OD | Radioactivity contamination: component, system, or area becomes more radioactive than desired or expected |
| OE | Temporary modification: an installation intended for short term use (usually this is for maintenance or modification of installed equipment) |
| OF | Environmental anomaly |
| OG | Airborne release |
| OH | Waterborne release |
| OI | Operator communication |
| OJ | Operator incorrect action |
| OK | Procedure or record error |

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 San Onofre 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

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

1. <u>Increase in heat removal by the secondary system</u>

   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. <u>Decrease in heat removal by the secondary system</u>

   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. <u>Decrease in reactor coolant system flow rate</u>

   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. <u>Reactivity and power distribution anomalies</u>

   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. <u>Increase in reactor coolant inventory</u>

- 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. <u>Decrease in reactor coolant inventory</u>

- 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. <u>Radioactive release from a subsystem or component</u>

- 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. <u>Anticipated transients without scram</u>

- 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.

rate,' were used as primary flags for reviewing the forced shutdowns (and power reductions). Once the generic type of event was identified, the particular initiating event was determined from the details associated with the shutdown. For example, if the reactor shuts down 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

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 |

equipment failure should not be expected to equal those identified as equipment failures with the NSIC categories. Other NSIC categories, such 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.

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) |

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 SAN ONOFRE 1

## 4.1 Summary of Operational Events of Safety Importance

This study reviewed the operational history of San Onofre 1 to indicate those areas of plant performance that have compromised plant safety. The review included a detailed examination of plant shutdowns, power reductions, reportable events, and special environmental reports. The criteria used to show degradations in plant safety were

1. events that initiated a design basis event (DBE), and
2. events that compromise safety functions designed to mitigate the propagation of DBE initiating events.

Shutdowns and power reductions indicated the number and types of DBEs entered. Reportable events and special environmental reports indicated DBEs and the number of times each engineered safety function was lost. The results of the operational review identified 25 DBEs entered and 14 losses of safety function.

## 4.2 General Plant Description

San Onofre Nuclear Generating Station Unit 1 is a Westinghouse pressurized water reactor (PWR) with 430 MWe net maximum capacity. The owners are Southern California Edison, who serves as the operating agent, and San Diego Gas and Electric Company. The architect/engineer is Bechtel Corporation. The plant is subject to license DPR-13, issued on March 27, 1967, pursuant to docket number 50-206.

San Onofre Unit 1 achieved initial reactor criticality on June 14, 1967 by means of boron dilution. The generator first synchronized to the grid on July 17, 1967. On January 1, 1968, commercial operation commenced at a maximum net power of 385 MWe. Full net power of 430 MWe was first achieved on December 27, 1968 and the plant was licensed for 450 MWe-gross on September 20, 1969.

Located near San Clemente in San Diego County, California, the plant lies entirely within Camp Pendleton Marine Reservation (Fig. 4.1). Thus, the immediate surrounding area is sparsely populated. The nearest city of 50,000 or more people is Oceanside, California located seventeen miles southeast of the plant with a population of 80,000. Within thirty miles of the plant, there are four cities and a population of 410,000. Within fifty miles, there are twenty-nine cities and a population of 3,600,000. The plant is sixty-two miles southeast of Los Angeles and fifty-one miles northwest of San Diego.

## 4.3 Availability and Capacity Factors

Table 4.1 contains the availability and capacity factors for San Onofre 1. San Onofre 1 began commercial operation of January 1, 1968. The reactor availability in the period 1968 through 1980 was above 70% except for four years: 1968, 1973, 1977, and 1980. In 1968, the reactor shut down for over six months due to cable tray fires which occurred on February 7 and March 12. The reactor shut down for approximately two and one-half months in 1973 to repair a turbine blade failure. In 1977, the reactor shut down for one month due to reactor coolant pump inspection, and steam generator tube inspection and plugging. Steam generator tube repairs caused the reactor to shut down for six months in 1978. In the thirteen years of commercial operation, the reactor availability has averaged 71.2% while unit availability averaged 68.3%. Unit capacity (MDC) was the same as unit capacity (DER) for San Onofre 1 and averaged 65.9%.

Table 4.1. Availability and capacity factors for San Onofre 1

| | 1967 | 1968 | 1969 | 1970 | 1971 | 1972 | 1973 | 1974 | 1975 | 1976 | 1977 | 1978 | 1979 | 1980 | Total |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Reactor availability | 62.5 | 42.3 | 76.7 | 83.7 | 94.5 | 79.3 | 63.7 | 86.9 | 87.9 | 72.2 | 64.3 | 81.7 | 90.4 | 22.6 | 71.2 |
| Unit availability | 36.4 | 41.5 | 75.8 | 83.0 | 93.4 | 77.8 | 62.8 | 86.1 | 87.4 | 70.2 | 63.7 | 80.2 | 90.2 | 22.3 | 68.3 |
| Unit capacity (MDC)[a] | 17.8[c] | 34.4[c] | 69.8[c] | 81.3[c] | 88.0[c] | 75.2[c] | 60.3 | 83.5 | 86.2 | 65.5 | 61.1 | 70.1 | 87.9 | 21.3 | 65.9 |
| Unit capacity (DER)[b] | 17.8[c] | 34.4[c] | 69.8[c] | 81.3[c] | 88.0[c] | 75.2[c] | 60.3 | 83.5 | 86.2 | 65.5 | 61.9 | 70.1 | 87.9 | 21.3 | 65.9 |

[a]MDC = maximum dependable capacity.

[b]DER = design electrical rating.

[c]Used (MWe) gross.

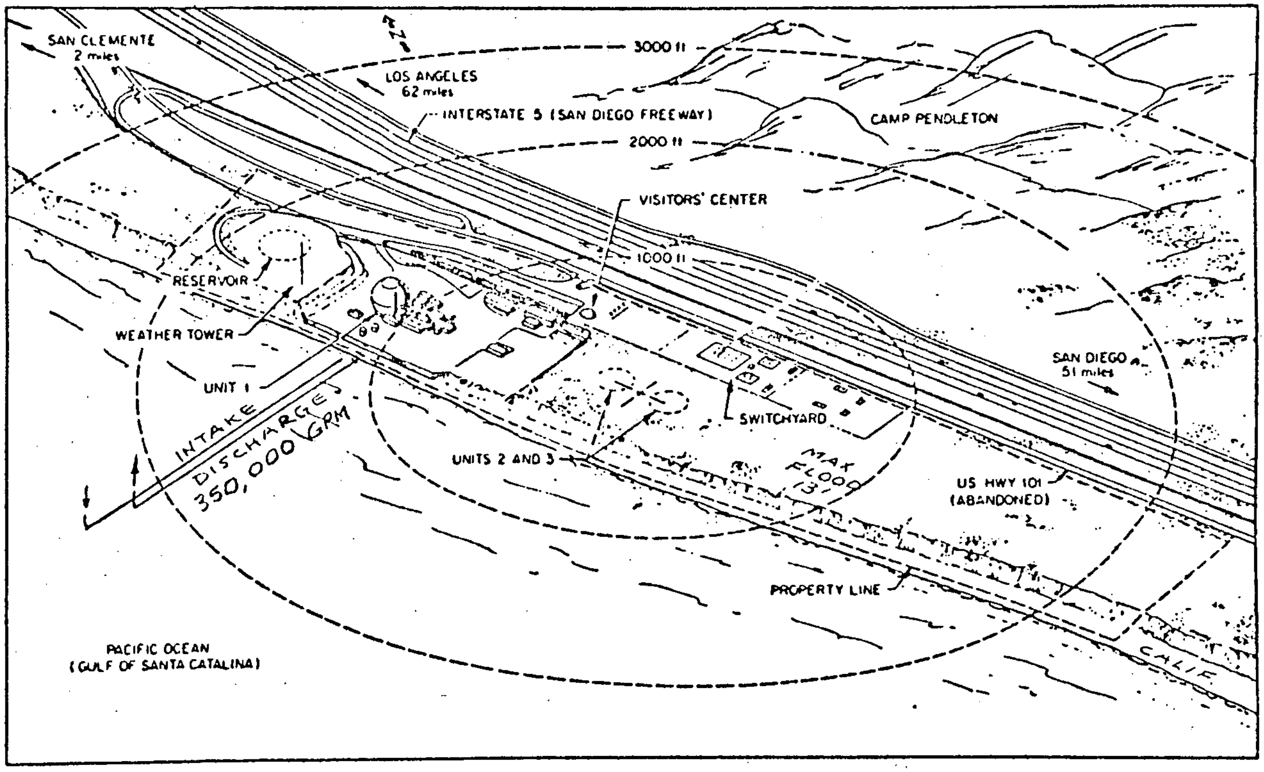


Figure 4.1 San Onofre Plant Site

## 4.4 Forced Reactor Shutdowns and Forced Power Reductions

Tables A.1.1 through A.1.14 in Appendix A provide a comprehensive summary of forced shutdowns and forced power reductions at San Onofre 1. Tables 4.2 and 4.3 summarize Tables A.1.1 through A.1.14 for forced shutdowns and forced power reductions, respectively. The duration of the event is rounded to the nearest hour for forced outages. All power reductions are defined as outages of zero hours duration for computing unit capacity and availability factors and forced outages rates.

### 4.4.1 Review of reactor shutdowns and power reductions

There were ninety-nine forced shutdowns and forty-four power reductions for the reporting period 1967 through 1980 with an average of eight forced outages per year. The average time the unit was shut down due to these occurrences was 1423 h/year (2 mo/year). Two outages, in 1968 (cable tray fires) and in 1980 (steam generator repair), were responsible for 8770 h of downtime for 44% of the total forced outage downtime experienced through 1980. If these two events were not considered, the average downtime due to forced shutdowns was 797 h/year. Approximately one-fourth of these occurrences were identified as DBE-initiating events as defined in Sect. 3.1.1.

#### 4.4.1.1 Yearly summaries for San Onofre 1.

1967

On June 14, 1967, San Onofre unit 1 achieved initial criticality by boron dilution. The generator was first synchronized to the grid on July 17. During turbine overspeed testing on July 17, salt water leakage into the condenser was noted. The salt water inleakage resulted in significant

Table 4.2. Forced shutdown summary for San Onofre 1

| | 1967 | 1968 | 1969 | 1970 | 1971 | 1972 | 1973 | 1974 | 1975 | 1976 | 1977 | 1978 | 1979 | 1980 | Total |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| I. Forced shutdowns | | | | | | | | | | | | | | | |
| 1. Total number | 9 | 9 | 9 | 2 | 12 | 12 | 3 | 4 | 4 | 9 | 10 | 5 | 6 | 5 | 99 |
| 2. Total hours down | 1668 | 5105 | 2122 | 75 | 426 | 661 | 2083 | 1130[d] | 143 | 178 | 761 | 495 | 857 | 4581 | 20285 |
| 3. Cause[a] | | | | | | | | | | | | | | | |
| a. Equipment failure | 7 (1524) | 7 (5056) | 8 (929)[c] | 1 (2) | 8 (156) | 11 (499) | 3 (2083) | 3 (583)[d] | 2 (6) | 6 (145) | 6 (83) | 2 (14) | 4 (459) | 2 (19) | 70 (11558) |
| b. Maintenance or testing | | | 1 (1193) | 1 (73) | 4 (270) | 1 (162) | | 1 (547) | 1 (127) | 1 (6) | 3 (674) | | 2 (398) | 1 (4152) | 16 (7602) |
| c. Regulatory restriction | | | | | | | | | | | | 1 (472) | | 1 (372) | 2 (844) |
| d. Administration | | | | | | | | | | | | | | | |
| e. Operational error | 2 (144) | 2 (49) | | | | | | | | 1 (20) | 1 (4) | 1 (5) | | | 7 (222) |
| f. Other | | | | | | | | | 1 (10) | 1 (7) | | 1 (4) | | 1 (38) | 4 (59) |
| II. Total number of DBE related shutdowns (these are included in totals of part I) | 2 (12) | 4 (179) | 2 (130) | | 2 (18) | 4 (67) | 1 (1974) | 1 (7) | | 3 (36) | 3 (50) | 2 (9) | | 1 (38) | 25 (2520) |
| III. System involved | | | | | | | | | | | | | | | |
| 1. Reactivity control (RB) | 3 (12) | 4 (179) | 1 (86) | | | 1 (4) | 1 | 2 (62)[b] | | | 3 (50) | | | | 15 (393) |
| 2. Reactor vessels and appurtenances (CA) | 2 (408) | 5 (4426) | 4 (614) | 1 (180)[b] | 1 (180)[b] | | | 1 (5) | 2 (131)[b] | 2 (12) | 1 (646)[b] | 1 (10) | | 1 (11) | 20 (6943) |
| 3. Containment heat removal systems and controls (SB) | | | | | | | | | | | 1 (27) | | | | 1 (27) |
| 4. High-pressure safety injection systems and controls (SF-C) | | | | | | | 1 (1974)[b] | 1 (516)[b] | | | 1 (25) | | 1 (234) | | 4 (2749) |
| 5. Offsite power systems and controls (EA) | | | | 1 (6) | | | | | | 1 (7) | | 1 (4) | | | 3 (17) |
| 6. AC onsite power systems and controls (EB) | | | | 1 (73) | 2 (62) | | | | | | | | 1 (133) | | 4 (268) |
| 7. Onsite power systems and controls (composite AC and DC) (ED) | | | | | | | 1 (1) | | 1 (2) | | 2 (6) | | | | 4 (9) |
| 8. Chemical, volume control, and liquid poison systems and controls (PC) | | 1 (152) | | | | | | | | | | | | | 1 (152) |
| 9. Turbine-generators and controls (HA) | 1 (1080) | | 4 (1257)[b] | 1 (2) | 5 (49) | 7 (221) | 1 (1974)[b] | 1 (516)[b] | | 4 (52) | 1 (3) | 1 (5) | | 1 (8) | 27 (5167) |
| 10. Main steam supply system and controls (HB) | 1 (216)[b] | | 1 (1193)[b] | | | 3 (344) | 1 (108) | 1 (547) | 1 (127)[b] | 1 (161) | 1 (646)[b] | 2 (476)[b] | 1 (394) | 1 (4152) | 14 (8304) |
| 11. Main condenser systems and controls (HC) | | | | | 1 (17) | 1 (162)[b] | 1 (1974)[b] | 1 (516)[b] | | | | | 1 (82)[b] | | 5 (2751) |
| 12. Circulating water systems and controls (HF) | 2 (216)[b] | | 1 (165) | | | | | | 1 (10) | | | | | | 4 (391) |
| 13. Condensate and feedwater systems and controls (HH) | | | | | 2 (284)[b] | 2 (254)[b] | | | | 1 (6) | | 1 (4) | 1 (82)[b] | 1 (38) | 8 (668) |
| 14. Reactor trip (IA) | 2 (144) | | | | 1 (8) | | | 1 (55)[b] | | | 1 (4) | | 1 (4) | | 4 (215) |
| 15. Engineered safety feature instrument systems (IB) | | | | | | | | | | | | | 1 (10) | | 1 (10) |
| 16. Not applicable/unknown | | | | | | | | | | | | | | 1 (372) | 1 (372) |

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

[b]Other systems involved. Maximum possible hours listed as breakdown in hours for each system unknown.

[c]Includes continuation outage of 160 h.

[d]Includes continuation outage of 516 h.

Table 4.3. Power Reduction Summary for San Onofre 1

| | 1967 | 1968 | 1969 | 1970 | 1971 | 1972 | 1973 | 1974 | 1975 | 1976 | 1977 | 1978 | 1979 | 1980 | Total |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| I. Power reductions | | | | | | | | | | | | | | | |
| 1. Total number | | | 4 | | | 1 | 2 | 3 | | 12 | 3 | 7 | 10 | 2 | 44 |
| 2. Cause | | | | | | | | | | | | | | | |
| a. Equipment failure | | | 3 | | | | 2 | 3 | | 6 | 1 | 3 | 2 | 1 | 21 |
| b. Maintenance or testing | | | | | | | | | | 5 | 2 | 3 | 7 | 1 | 18 |
| c. Regulatory restriction | | | | | | | | | | | | | | | 0 |
| d. Operator training/license exam | | | | | | | | | | | | | | | 0 |
| e. Administrative | | | | | | | | | | | | | | | |
| f. Operational error | | | | | | 1 | | | | | | 1 | 1 | | 3 |
| g. Other | | | | | | | | | | 1 | | | | | 1 |
| II. Total number of DBE related power reductions (included in totals of part 1) | | | | | | | 1 | | | | | | | | 1 |
| III. System involved | | | | | | | | | | | | | | | |
| 1. Reactivity control (RB) | | | 1 | | | 1 | 1 | | | 1 | | 2 | | | 6 |
| 2. High-pressure safety injection system (SF-C) | | | | | | | | | | | | 1 | | | 1 |
| 3. Reactor trip (IA) | | | | | | | | 2 | | | | | | | 2 |
| 4. Offsite power systems and controls (EA) | | | | | | | | | | 1 | | | | | 1 |
| 5. Onsite systems and controls (composite AC and DC) (ED) | | | 3 | | | 1 | 1 | | | 2 | | | | | 7 |
| 6. Turbine-Generators (HA) | | | | | | | | | | | | | 1 | 1 | 2 |
| 7. Main condenser systems and controls (HC) | | | | | | | | | | 7 | 2 | 2 | 8 | 1 | 20 |
| 8. Circulating water systems and controls (HF) | | | | | | | | | | | 1 | 1 | 1 | | 3 |
| 9. Condensate and feedwater systems and controls (HH) | | | | | | | | 1 | | 1 | | 1 | | | 3 |

turbine blade damage. All of the blades in the last four stage rows required replacement. The outage lasted ~1080 h and accounted for almost 2/3 of the forced outage down time for 1967.

On September 11, the unit shut down for ~192 h due to excessive vibration in the circulating water pump and a leaking pressurizer safety valve. The vibration was caused by pulling in an excessive amount of marine growth into the circulating water system.

The unit shut down on July 9 and December 12 because malfunctioning or defective electrical equipment was resulting in control rod drops. On October 19, an indication of a fully inserted control rod from a bad connector in the position indication circuitry resulted in a unit shutdown. These are recurring events in San Onofre's operating history. Section 4.4.3.4 discusses indication of dropped rods.

On December 5, the reactor tripped on an incorrectly set overpower set point. This event is part of a recurring sequence of events dealing with turbine overspeed setpoint settings (see Sect. 4.4.3.2)

With one circulating water pump out of service, the unit shut down when an inadvertent closure of the tsunami gate occurred. A shorted limit switch caused the closure.

1968

There were nine forced outages accounting for 5105 h of downtime in 1968. The first outage occurred on February 7 when a fire started in the cable tray containing the cables leading to sphere electrical penetration EPC-4. The cause of this event was electrical and mechanical overloading of power cables for the pressurizer heaters. A month later, on March 12, another cable failure resulted in a fire. The cause of this cable failure was a higher than design thermal loading of circuits that supply the pressurizer heaters. These two events shut down the reactor for 4618 h and are discussed in more detail in Sect. 4.5.2.2.

In 1968 there were four events of dropped or bottomed control rods (March 4, March 9, September 9, and September 26 which shut down the reactor for a total of 180 h (see Sect. 4.4.2.6).

Beginning December 9, an increased temperature in valve discharge piping indicated leakage through a pressurizer relief valve. Observation continued until December 28 when the unit was shut down. It remained down for 244 h. Pressurizer relief valves are discussed in more detail in Sect. 4.4.3.1.

1969

In 1969, there were nine forced outages accounting for 2122 h of downtime. There was one major outage this year which commenced on June 20. The unit shut down for 1193 h to allow inspection of the turbine-generator and to make modifications to the steam generator moisture separators.

On March 22 and October 10, the unit shut down for a total of 394 h to replace several leaking pressurizer relief valves. On October 9, the unit was removed from operation for 165 h when the tsunami gate slipped from its anchor bolts and fell into the intake tunnel.

On April 7, a load runback resulted from an unexplained false dropped rod signal. This false signal necessitated a power reduction to investigate the problem. On October 14, a relay malfunction caused several control rods to drop. The unit was down for 86 h.

1970

Only two forced outages interrupted operation in 1970. The first, on May 29, was a reactor trip due to turbine acceleration which was faster than desired. The second, on November 23, was to complete the relocation of the 220 kV switchyard. The total forced outage downtime for the year was seventy-five h.

1971

The reactor availability factor of 94.5% for 1971 was the best achieved during the time period covered by this review. There were twelve forced outages totaling 426 h downtime. This year was the first in which the unit shut down to repair leaking condenser tubes. The unit shut down on June 27 for 17 h and October 27 for 104 h for this cause.

On November 1, 3, and 5, the unit was shut down three times to obtain additional data on turbine overspeed characteristics (see Sect. 4.4.3.2).

1972

There were twelve forced outages totaling 661 h in 1972. The initial repair of leaking steam generator tubes occurred during this year. On July 8, a leak in 'C' steam generator tube was detected. Sampling began and continued until the leakage was ~100 gal/d. On July 19, the unit shut down for 177 h to repair the leak. On September 12, analysis determined that the 'A' steam generator was experiencing primary to secondary side leakage. Sampling began and continued until the leakage was over 100 gal/d. The unit was then shut down for 155 h on October 13 for repairs of the tube leaks.

On April 30, during startup, the reactor and turbine tripped on a steam generator high level signal. The high level resulted from failure of the positioner on the main feedwater regulating valve. This event is discussed in more detail in Sect. 4.4.2.1.

On February 25, March 4, and March 24, the unit shut down for a total of 23 h because of additional turbine overspeed problems and testing. Indication of dropped control rods continued to occur causing a unit shutdown on September 20 and power reduction on December 9.

1973

There were only three forced outages in 1973. However, total downtime was 2083 h. The major outage for the year started on October 21 and accounted for 1974 h downtime. Bearing vibration and salt water leakage indicated turbine problems existed. While shutting the unit down, a safety injection actuation and reactor trip occurred. Investigation revealed turbine blade failure.

On January 6, the unit shut down for 108 h to locate and repair 'A' steam generator tube leaks. For more information on this and subsequent steam generator tube leaks refer to Sect. 4.5.3.2. On January 10, the reactor tripped when No. 4 vital bus was transferred from the backup power supply to the normal power supply.

On February 20, a power reduction occurred resulting from an indication of a dropped rod. On August 17, another power reduction was initiated when failure of an invertor caused a load transfer.

1974

In 1974, there were four forced outages totaling 614 h. The longest outage began on April 27 when the unit shut down for 547 h to repair steam generator and reheater tube leaks and to repair a leaking pressurizer safety valve.

On June 11 and June 14, the unit initiated a power reduction as a result of automatic load runbacks. Problems with inverters which supply power to nuclear power range monitors caused the runbacks. On October 21, the unit was tripped due to dropped control rods.

1975

Of the four forced outages totaling 143 h, the longest outage of 1975 commenced on June 11. The unit shut down for 127 h to repair pressurizer relief valves and to plug leaking steam generator tubes.

On February 19, an inverter failed causing a spike in a pressurizer level channel. This forced the unit down for 4 h. On April 22, the reactor tripped while transferring the No. 4 vital bus back to its normal power supply.

1976

Nine forced outages totaled 178 h in 1976. On July 30, the unit shut down for 101 of the 1978 h to repair leaking steam generator tubes in 'C' steam generator.

On March 23 and March 29, the unit reduced a power to investigate salt water in-leakage into the condenser. On five days (January 8, January 26, January 29, February 6, and July 2), the unit reduced power in order to plug leaking condenser tubes.

The unit again shut down on February 9 and reduced power on September 25 and 26 because of problems with inverters. The events on September 25 and 26 also caused erroneous indications of dropped control rods.

On April 17 through 19, there were four events involving turbine overspeed. The unit was shut down a total of 42 h.

1977

During 1977, there were ten forced outages totaling 761 h. On September 9, the reactor shut down for reactor coolant pump inspection, steam generator inspection, and plugging of leaking tubes in the steam generators. On July 1 and 2, the unit reduced power in order to plug leaking tubes in the south half of 'B' condenser.

There were three events of actual dropped rods on April 14, May 18, and June 9. The unit shut down a total of 50 h during these events.

1978

There were five forced outages totaling 495 h of downtime in 1978. On April 5, the unit shut down for 472 h (95% of the year's forced downtime) in order to inspect the steam generators. This shutdown was required by NRC.

On March 2 and June 9, the unit reduced power in order to plug leaking condenser tubes. On March 22 and May 18, the unit reduced power due to erroneous dropped rods indications.

1979

The 1979 reactor availability factor was 90.4%. This was the second best in the operating history of San Onofre. There were six forced outages totaling 857 h. On April 5, the unit shut down for 82 h to repair a condenser tube leak and to repair the feedwater flow straighteners. In addition, there were seven other occasions (February 23, August 30, September 7, November 29, twice on November 30, and December 2) involving condenser tube leaks which necessitated the unit to reduce power to repair or plug the leaking tubes.

On June 1, the unit went down for 394 h to plug leaking steam generator tubes. On September 14, the unit shut down for 234 h in order to repair refueling water pump suction piping and to replace several pipe sections on the safety injection line. On November 7, the unit shut down for 133 h due to loss of 480 V bus No. 1 (see Sect. 4.5.3.1).

1980

The 1980 reactor availability factor was 22.6%. This was the worst year for San Onofre 1 in terms of power production. Although there were only five forced outages, the five accounted for 4581 h of downtime. After completing refueling on July 12, the unit remained shut down for 4152 h to complete steam generator tube repair.

On January 26, the unit shut down for 392 h in order to effect TMI modifications. On February 17, the unit reduced power to repair a salt water leak into the condenser.

On January 16, the unit shut down for 38 h due to a steam flow/feedwater flow mismatch caused by a construction worker who accidently struck the closing circuit control relay to the east feedwater pump.

4.4.1.2 Systems involved. There were sixteen systems involved in the 141 forced outages and power reductions (Table 4.2). For several of these events, more than one system was involved in the event. In these cases, the outage time was charged to each system involved. Four systems which each were involved in twenty or more events were:

(1) Turbine-Generators and Controls - Twenty-eight events involving the turbine generator and controls system totaled 4651 h. Two outages (July 17, 1967, October 21, 1973) dealing with turbine blade failures and one dealing with modifications of moisture separators (June 20, 1969) accounted for the majority (90%) of the outage time for this system.

(2) Main Condenser and Controls - Twenty-four events involved the main condenser and controls totaling 2235 h of downtime. Most of these events were power reductions for the purpose of repairing minor condenser tube leaks. The outages of July 17, 1967 and October 21, 1973, resulted form major salt water in-leakage resulting in turbine blade damage.

(3) Reactivity Control - The reactivity control system accounted for twenty-one events and 263 h of downtime. Almost all of these events dealt with dropped rods or indications of dropped rods. This was a recurring problem throughout the operating history of San Onofre 1.

(4) Reactor Vessel and Appurtenances - Twenty events totaled 6943 h of downtime because of failures involving the reactor vessel and appurtenances system. The two events (February 7, 1968, March 12, 1968) in the 1968 dealing with the cable fires accounted for 70% of the outage time.

Two other systems were involved with a significant number of forced shutdowns:

(1) Main Steam Supply - Fourteen times totaling 8304 h, and

(2) Condensate and Feedwater - eleven times totaling 668 h.

The additional ten systems each were involved fewer than nine times and averaged four occurrences per system. Nine of these ten systems had less than 400 h charged to them with an average of 135 h. The tenth, High Pressure Safety Injection, was involved four times for 2233 h.

4.4.1.3 <u>Causes of forced reactor shutdowns and forced power reductions.</u> Equipment failure accounted for seventy outages and nineteen power reductions (Table 4.2). The total forced outage downtime through 1980 was 11,042 h. Steam generator problems, leaking pressurizer relief valves, turbine blade failures, dropped control rods, failure of inverters, and turbine overspeed problems accounted for most of the equipment failures.

Maintenance and testing accounted for seventeen outages (7626 h) and eighteen power reductions. Modifications to the moisture separators, relocating the 220 kV switchyard, inspections, condenser tube leaks, and steam generator tube leaks accounted for most of these events.

The unit shut down twice due to regulatory required inspections of the steam generators and for TMI modifications. During 1978 the unit reduced power for extended periods due to an administrative decision to defer fuel depletion.

There were only seven events related to operational error totaling 222 h. There were primarily two areas involved. The first dealt with incorrect settings for turbine overspeed while the second dealt with false indications of dropped rods due to introduction of spurious signals. The only event of major importance was the unexpected opening of a feedwater pump safety injection valve during maintenance which occurred on August 28, 1978. Opening the valve resulted in dilution of the refueling water storage tank (RWST). The unit reduced load and added boric acid to the RWST to compensate for the dilution.

There were five events related to other causes. Two of these were related to brush fires around San Clemente which resulted in a loss of one train of offsite power on January 21, 1976 and January 22, 1977. The others primarily dealt with marine growth in the circulating water system. The event of January 16, 1980, of steam flow/feedwater flow mismatch resulted from an inadvertent action of a construction worker.

4.4.1.4 Non-design basis events. There were 117 non-DBEs of which seventy-five were forced shutdowns and forty-two were power reductions (Table 4.4). These events can be classified into the following categories:

(1) pressurizer relief valves (seven events),

(2) steam generator tubes (nine events),

(3) leaks in the condenser (twenty-four events),

(4) turbine overspeed (fourteen events),

(5) power buses and inverters (twenty-five events), and

(6) others.

The first five categories deal with recurring events and are discussed in Sect. 4.4.3 or Sect. 4.5.3.

### 4.4.2 Review of design basis events

There were twenty-five design basis events (DBEs) accounting for 19% of the total number of events and resulted in twenty-four forced outages (Table 4.5). A forced power reduction (February 20, 1973) resulted in a DBE when an indication of a rod dropped was received. The total downtime associated with the outages was 2568 h or one-third of the total downtime which occurred during the San Onofre 1 operating history through 1980. Seven DBE category types occurred with only three categories having more than one event.

4.4.2.1 D1.2 Feedwater system malfunctions that result in an increase in feedwater flow. On April 30, 1972, The unit had just completed a normal reactor startup and was operating on line at 55 MWe with load being slowly increased. The auxiliary feedwater regulators were in service and controlling steam generator water levels. As the feedwater block

Table 4.4. NSIC primary category summary for non-DBE shutdowns and power reductions for San Onofre 1

| | 1967 | 1968 | 1969 | 1970 | 1971 | 1972 | 1973 | 1974 | 1975 | 1976 | 1977 | 1978 | 1979 | 1980 | Total |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1. Equipment failures | 2 | 3 | 8 | 2 | 2 | 4 | 1 | 2 | 1 | 7 | 5 | 4 | 5 | 3 | 49 |
| 2. Instrumentation and control anomalies | 2 | | 1 | | 1 | 1 | | 2 | | 1 | | 2 | | | 10 |
| 3. Non-DBE reductions in coolant inventory (leaks) | 2 | 2 | 2 | | 4 | 3 | 1 | 2 | 1 | 8 | 4 | 2 | 10 | 1 | 42 |
| 4. Fuel/cladding failure | | | | | | | | | | | | | | | |
| 5. Maintenance error | | | | | | | | | | | | | | | |
| 6. Operator error | 1 | | | | | 1 | 1 | | 1 | | 1 | 1 | 1 | 1 | 8 |
| 7. Procedural/administrative error | | | | | 3 | | | | | | | | | | 3 |
| 8. Regulatory restriction | | | | | | | | | | | | 1 | | 1 | 2 |
| 9. External events | | | | | | | | | 1 | 2 | | | | | 3 |
| 10. Environmental operating constraint-tech. specs. | | | | | | | | | | | | | | | 0 |
| TOTAL | 7 | 5 | 11 | 2 | 10 | 9 | 3 | 6 | 4 | 18 | 10 | 10 | 16 | 6 | 117 |

Table 4.5. DBE initiating events at San Onofre 1

| | DBE category | 1967 | 1968 | 1969 | 1970 | 1971 | 1972 | 1973 | 1974 | 1975 | 1976 | 1977 | 1978 | 1979 | 1980 | Total |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1. Feedwater system malfunctions that result in an increase in feedwater flow | D1.2 | | | | | | 1 | | | | | | | | | 1 |
| 2. Loss of external electric load | D2.2 | | | | | 2 | 2 | | | | | | | | | 4 |
| 3. Turbine trip | D2.3 | | | 1 | | | | | | | 3 | | 1 | | | 5 |
| 4. Loss of normal feedwater flow | D2.7 | | | | | | | | | | | | | | 1 | 1 |
| 5. Single and multiple reactor coolant pump trips | D3.1 | | | | | | | | | | | | 1 | | | 1 |
| 6. Control rod maloperation | D4.3 | 2 | 4 | 1 | | | | | 1 | | | 3 | | | | 11 |
| 7. Inadvertent operation of ECCS during power operation | D5.1 | | | | | | | 1 | | | | | | | | 1 |
| Total | | 2 | 4 | 2 | 0 | 2 | 3 | 1 | 1 | 0 | 3 | 3 | 2 | 0 | 1 | 24 |

valve to 'C' steam generator was opened, flow increased rapidly causing the turbine and reactor to trip from high level in 'C' steam generator. The high level resulted from failure of the positioner on the main feedwater regulating valve, The primary coolant temperature experienced a temperature drop of 91°F. Safety injection equipment was actuated ~12 min after the reactor trip. As a result, feedwater flow to the steam generators ceased and the cooldown in the primary system terminated.

Nine minutes after actuation, the safety injection system was secured. The operation of the safety injection system was normal. However, since the minimum reactor coolant system pressure was significantly above the main feed pump shut-off head, no borated refueling water was delivered to the reactor coolant system through the normal safety injection flow path. Approximately 900 gallons of borated water were delivered to the coolant system through the normal charging path. An analysis to evaluate the thermal effects of this incident concluded that: (1) fast fracture was not a matter of concern, and (2) integrity would be maintained if a loss-of-coolant accident occurred.[14,15]

4.4.2.2 <u>D2.2 Loss of external electric load.</u> There were four events associated with the category - loss of external electric load. The first loss of load event occurred on June 22, 1971. Both the Chino and Santiago lines relayed and the generator tripped on overspeed. On July 12, 1971 the unit tripped from a generator out-of-step condition. In the third event, the main exciter motor failed mechanically when being started on July 27, 1972. The problem was determined to be rubbing between the rotor and stator. On July 29, 1972, the unit tripped from loss of main generator field.

4.4.2.3 D2.3 Turbine trip. There were five events associated with the category - turbine trip. On January 9, 1969, the unit was removed off line after a routine test of the stop valves produced a spurious partial turbine trip. Investigation revealed that the turbine auto stop oil supply pressure was low due to a badly scored seat and disc in the autostop oil dump valve. During startup, twice on April 17, 1976 and once on April 18, 1976, a spurious turbine trip resulted in a reactor trip. The cause was an incorrect setting of an overspeed trip device. On September 12, 1978, the reactor and turbine tripped while performing bearing low oil pressure tests.

4.4.2.4 D2.7 Loss of normal feedwater flow. On January 16, 1980, the unit tripped from steam flow/feedwater flow mismatch. The trip was caused by a construction worker who accidently struck the closing circuit control relay to the east feedwater pump normal discharge valve. This resulted in a rapid decrease in feedwater flow and the resultant trip.[16]

4.4.2.5 D3.1 Single and multiple reactor coolant pump trips. On March 8, 1978, the reactor tripped from a loss of coolant flow signal. The low flow condition occurred when a power system disturbance resulted in low voltage at the reactor coolant pump motors. The disturbance was due to a fault occurring on the San Diego Gas and Electric Company power grid which interties to San Onofre.

4.4.2.6 D4.3 Control rod maloperation. There were eleven forced outages due to dropped rods. The problems with dropped rods were of four types:

(1) relay malfunctions (3),

(2) slave cycler malfunctions (2),

(3) gripper coil malfunctions (3), and

(4) connector malfunctions (1).

On two additional occasions, October 21, 1974 and April 14, 1977, rods dropped for unknown reasons.

On July 9, 1967, a timer relay malfunctioned which caused rod drops in subgroup 4 of control rod group No. 1. On September 9, 1968, all five rods on subgroup 7 slipped from 117 to 40 steps due to BF relay failures. On August 14, 1969, control rods associated with subgroup 7 dropped into the core due to an intermittently open contact of a BF type relay in the rod control logic.

On March 4, 1968, cleaning of the rod controls slave cycler contacts was in progress and the subgroup No. 6 slave cycler had been replaced with a shutdown cycler so that the contacts could be cleaned. An attempt was made to cycle the rods one step and subgroup No. 6 was dropped to 160 steps. The slave cycler was examined and no reason for failure was found. On September 26, 1968, the control rods on subgroup 8 dropped into the core while a slave cycler failure alarm was being cleaned. This occurred when an operator manually opened, closed, and reopened the half-power contactor while attempting to clean a slave cycler failure alarm. The cause was due to improper operation of the slave cycler clutch.

On March 9, 1968, the unit was shut down when physics data indicated a bottomed control rod. Investigation revealed that the polarity of the movable griper coil had been reversed during repairs to penetration EPC-4. On May 18 and June 9, 1977, a gripper coil failed causing four rods to drop which caused the unit to be manually tripped.

On December 12, 1967, five control rods in subgroup 7 dropped because of bad connectors in the vessel head. The defective electrical components were replaced.

4.4.2.7 D5.1 Inadvertent operation of ECCS during power operation. On October 21, 1973, the unit was being removed from service to investigate turbine problems indicated by bearing vibration and salt water leakage when a safety injection actuation occurred and the reactor tripped. During the unit shutdown, rapid cooldown of the primary cooling system resulted in the unnecessary initiation of the safety injection system.

Blowdown on all three steam generators was commenced at 12:15 a.m. and a unit load decrease was initiated. At 1:18 a.m., the turbine was in an unloaded status and a no load trip alarm was received. At 1:19 a.m. the turbine stop valves closed, and since feedwater control was still on automatic, the feedwater regulator valves went to 80% open. As a result reactor coolant temperature and pressure began decreasing rapidly.

The operator changed reactor and feedwater control to manual in an attempt to increase temperature. Steam generator levels were increasing rapidly and pressurizer level had decreased to 10%. At 1:21 a.m., the safety injection system initiated and the reactor tripped. Approximately 1300 gal of borated water from the refueling water storage tank entered the primary system through the charging pumps.

Safety injection had functioned as required. However, in restoring the system to standby, the Loop B safety injection valve was partially inoperable. The valve mounting bolts and a nearby pipe hanger had failed. Further investigation indicated the damage to the Loop B piping was due to a water hammer.

The cause of the overcooling event was assessed as failure to place feedwater control in manual prior to removing the turbine from service. Additionally, changes in steam generator control upon turbine trip were considered.[17]

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

There were five major trends associated with forced outages and power reductions. These trends can be classified into one of the following categories of problems concerning:

(1) pressurizer relief valves,

(2) turbine overspeed,

(3) leaks in the condenser,

(4) indications of dropped rods,

(5) power buses and inverters, and

(6) steam generator tubes.

The pressurizer relief valves and the turbine overspeed problems mainly occurred during the first early years of operation. Leaks in the condenser here occurred throughout San Onofre's operating history with most of these resulting in forced power reductions. Problems with power buses and inverters occurred after the first five years of operation. As well, problems pertaining to steam generator tubes surfaced following the first five years of operation. Problems with power buses and inverters and steam generator tubes are discussed in Sects. 4.5.3.1 and 4.5.3.2, respectively.

4.4.3.1 Pressurizer relief valves. Problems with the pressurizer relief valves occurred primarily during the first three years of operation. On October 1, 1967, the unit shut down for 216 h to repair two pressurizer power relief valves and a bypass valve in a steam line. On December 9, 1968, leakage through pressurizer relief valve RV-533 was observed by an increase in temperature in valve discharge piping. Observation continued until a normal shutdown was required on December 28, 1968, lasting 84 h. On March 22, 1969, leakage through pressurizer relief valve RV-533 was observed and the unit was shut down to repair the valve. Other scheduled work was accomplished at the time which caused the reactor to remain offline for 314 h. On August 10, 1969, the unit shut down for 80 h to correct leakage of three pressurizer relief valves. On April 27, 1974, during a shutdown to repair steam generators, repair of a pressurizer relief valve was also accomplished. On June 11, 1975, the unit shut down to repair safety valves RV-532 and RV-533.

4.4.3.2 Turbine overspeed. There were fourteen shutdowns caused by turbine overspeed problems. On June 22, 1971, a fault on the San Onofre-Chino 220 kV line and a concurrent opening of the San Onofre-Santiago 220 kV line breakers caused a partial unloading of the unit. Downward drift in the backup overspeed device set point resulted in a complete electrical unloading of the units at a frequency of 61 Hz. Instrumentation indicated that the turbine-generator reached a maximum of 133% of normal speed. Three weeks later on July 12, 1971, the unit shut down due to a generator 'out-of-step' condition caused by low generator excitation. This resulted in a complete electrical unloading of the unit. Instrumentation indicated that the turbine-generator reached a maximum of 133% of normal speed.

As a result of these two incidents, unit load was restricted to 80% of full power pending an evaluation of turbine overspeed characteristics.

The results of this evaluation indicated that the unit load could be increased to full power provided (1) that condenser backpressure is a minimum of 1.5 in. of mercury absolute and (2) that the emergency overspeed trip was set at 104%. Load was increased to full power on August 9, 1971. Six additional shutdowns (November 1, November 3, and November 5, 1971; February 25, and March 4, 1972; and March 19, 1976) tested the overspeed setting and gave data on overspeed characteristics.

Five events (December 5, 1967, March 24, 1972, April 17, 1976, April 17, 1976, April 18, 1976) resulted from incorrect settings of the overspeed trip. On May 29, 1970, the turbine was observed to be accelerating too fast and the unit was shut down.

4.4.3.3 Condenser tube leakage. Leaking condenser tubes caused five shutdowns and nineteen power reductions at San Onofre. The first time leaking condenser tubes became a problem was on July 17, 1967. However, the recurrence frequency did not increase until 1976 and continued to be high through 1980.

Five shutdowns specifically to repair condenser tube leaks interrupted operation on July 17, 1967: June 27 and October 27, 1971: October 21, 1973: and April 5, 1979. On two occasions, July 17, 1967 and October 21, 1973, salt water inleakage damaged turbine blades and resulted in 1080 and 2490 h of downtime, respectively.

The first power reduction because of condenser tube leaks did not occur until 1976. However, as Fig. 4.2 indicates, condenser leaks caused recurring power reductions through 1980. Power reductions to repair condenser tubes occurred on January 8, January 26, January 29, February 6,

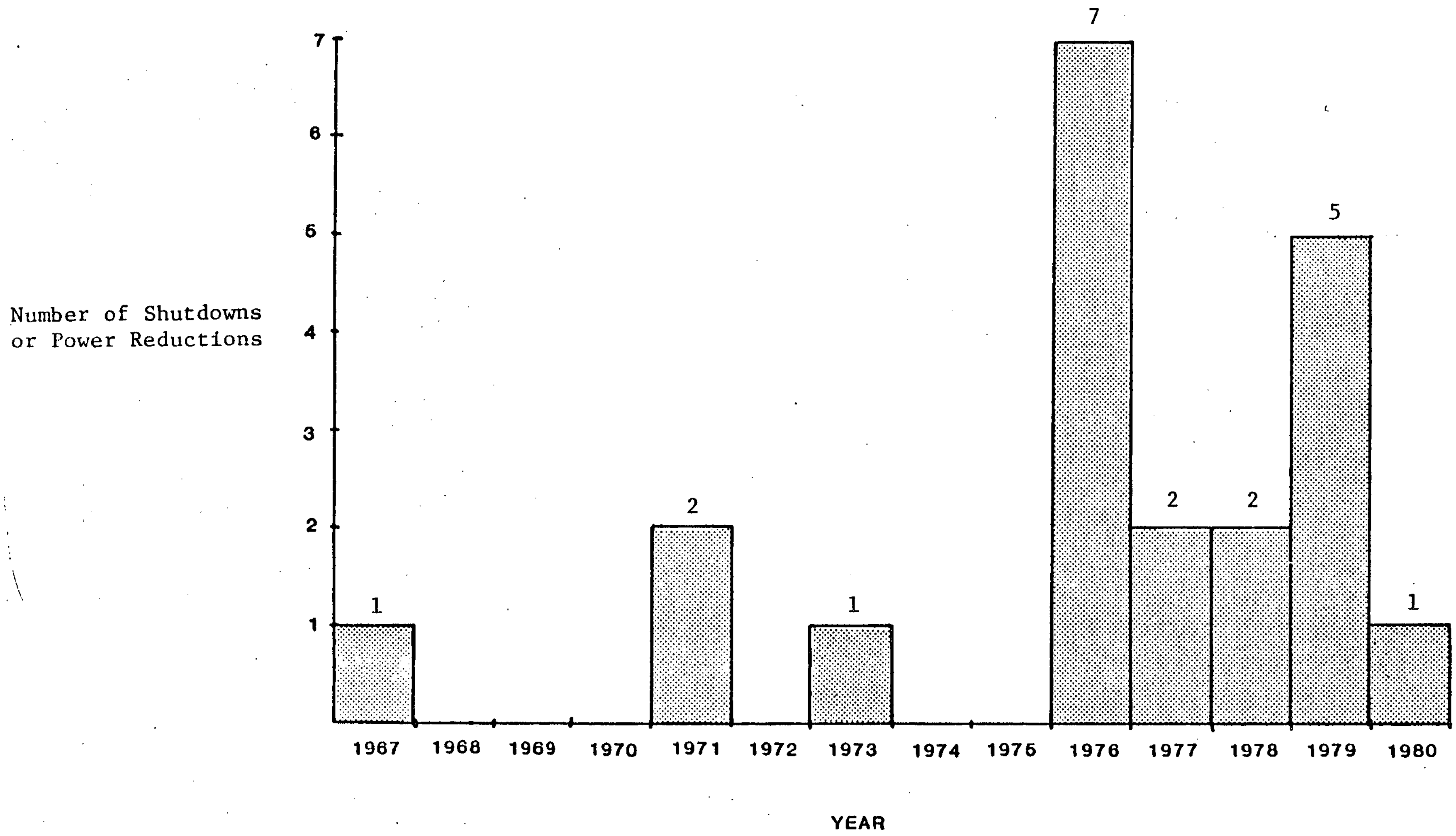


Figure 4.2 Number of Condenser Tube Leakage Events

March 23, March 29, and July 2, 1976; July 1 and July 2, 1977; March 2, and June 9, 1978; February 23, August 30, September 7, November 29, November 30, and December 2, 1979; and February 17, 1980.

4.4.3.4 Indications of dropped control rods. There were 14 events which resulted from indications of dropped control rods. Most of these were power reductions. Seven of these events (December 9, 1972, June 11 and June 14, 1974, February 9 and February 17, 1976; and September 25 and September 26, 1976) were associated with problems related to power buses and inverters and are discussed in Sect. 4.4.3.5. Two of the events (September 20, 1972 and February 20, 1973) are discussed in Sect. 4.4.2.6.

On October 19, 1967, the unit was shut down due to a false indication of dropped control rods. On April 7, 1969, the unit reduced power due to an unexplained false dropped rod signal. On March 22, 1978, an erroneous signal indicated dropped control rods.

On August 7, 1976, the unit reduced power due to a relay coil failure producing an erroneous dropped rod signal. On May 18, 1978, a grounded rod position indication system component caused an erroneous dropped rod signal.

## 4.5 Reportable Events

This study reviewed 294 reportable events from San Onofre Unit 1. The events included miscellaneous reports, abnormal occurrences (AO), station incidents (SI), and licensee event reports (LER) filed by the utility for various equipment failures or technical specification violations. The information in the reportable events was coded as discussed in Sect. 1.3. The tables compiling this coded information are in Appendix A, Part 2.

4.5.1 Review of reportable events from 1967 through 1980

Figure 4.3 illustrates the number of reportable events per year submitted by the Southern California Edison Company. The number of reported events remained relatively constant over the first half of San Onofre's Operating history. The second half tended to have increasing numbers. The years 1967 through 1974 average 17 events per year while the years 1975 through 1980 averaged 26 events per year. Reports of steam generator and condenser tube leakage were major contributors to the trend of increased numbers of reportable events. Peak reporting years were in 1976 and 1980 with 31 and 39 events, respectively.

4.5.1.1 Yearly summaries. The following sections present a summary of reportable events for each year of operation for San Onofre from 1967 through 1980. A single event which occurred in 1966 was also included in this study even though it occurred prior to initial criticality. During installation, a steam generator was dropped one foot during a lift. The steam generator withstood the subsequent hydraulic test, but 32 inconel tubes were deformed and several tube support sheets were crushed. After making repairs and plugging tubes, the steam generator was placed in service.

1967

San Onofre's initial criticality occurred in June 1967. This analysis examined 17 events that were reported during 1967. Two of those events are considered significant, one involving failure of both safety injection pumps and the second a spill of radioactive water from the radwaste system. The first significant event occurred two weeks prior to initial criticality. Both safety injection pumps were declared inoperable

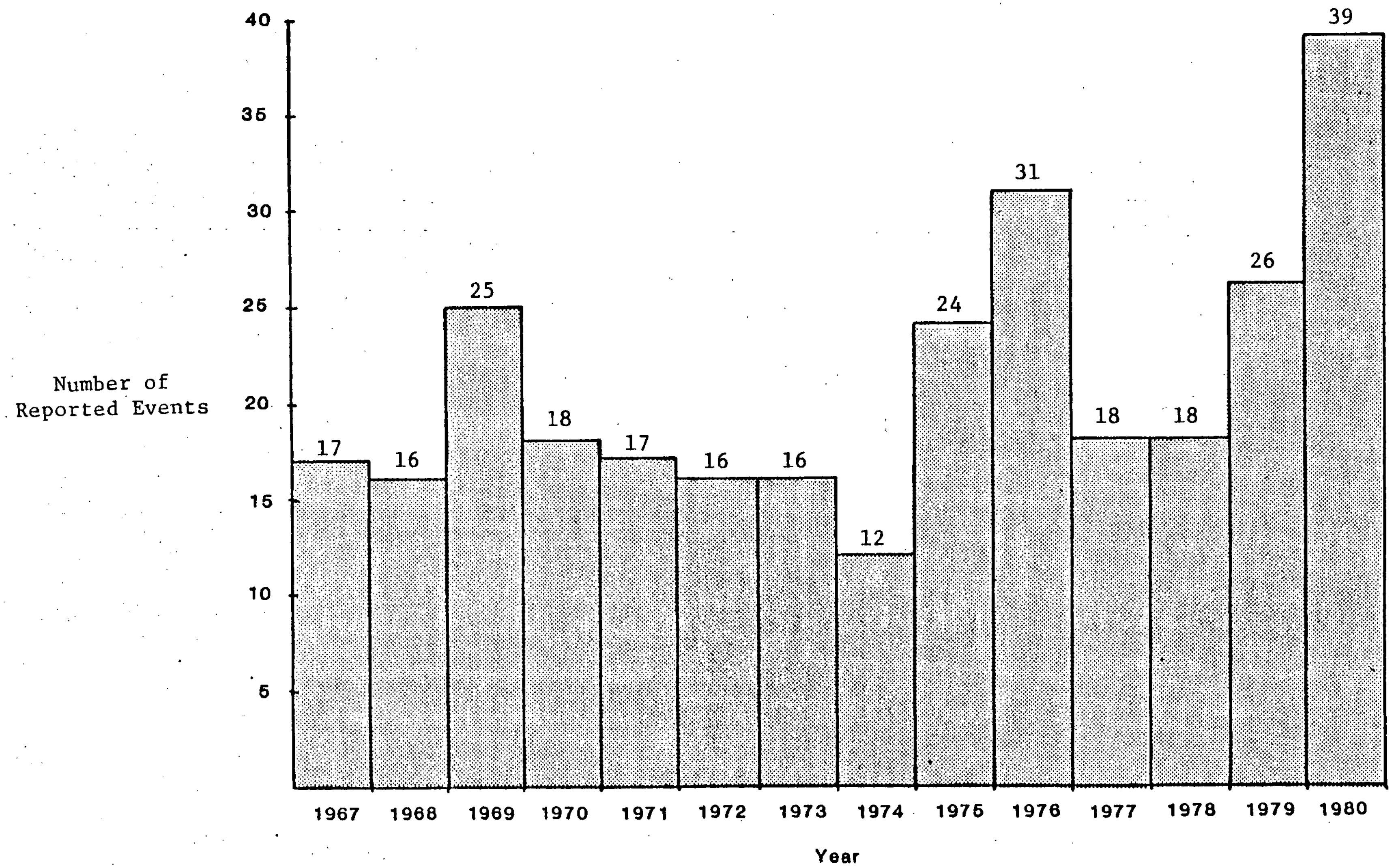


Figure 4.3 Number of Reported Events Per Year at San Onofre 1

when they failed megger tests. This common cause failure is discussed in Sect. 4.5.2.1. The second event involved the overflow of a radwaste tank due to operator error and is described in Sect. 4.5.1.4.

Of the remaining 15 events for 1967, 5 involved control rods or control rod drive mechanisms. These included two instances of inadvertent rod drops, two false or erratic rod position indications and one rod that mechanically jammed.

1968

Fifteen reportable events were reviewed for San Onofre for 1968. Two cable tray fires occurred, the second of which resulted in a five month shutdown. These signifiant events are discussed in Sect. 4.5.2.2. Also significant during the year was an event that prevented adequate boration of the primary system during a shutdown due to crystallization of boron in the lines.

The remaining 12 events were not significant individually. Seven of them however, involved control rods. Five of these events were occasions when rods inadvertently dropped partially or completely into the reactor core. Two control rod anomalies were wiring faults.

1969

Twenty-five reportable events were recorded for 1969. Fourteen of these were 'abnormal occurrences.' Two of the 25 events were identified as significant in this analysis. The first of these involved another instance of boron crystallization inhibiting adequate boration (Sect. 4.5.2.6). The second event, described in Sect. 4.4.2.1, involved a cooling transient due to a feedwater control valve failing open because of an air line failure. The remainder of the events were insignificant and involved a variety of plant systems.

1970

In 1970, San Onofre reported 17 abnormal occurrences. In addition to those events, this review also included a reactor coolant pump event in which the pump flywheel cracked. None of the events examined were deemed significant. Five of the events were false control rod position indications, due to recurring LVDT failures. Three events involved safety injection system pumps or valves.

1971

Seventeen abnormal occurrences at San Onofre were reported in 1971. From these, this review identified two significant events, both involving occurrences of overspeed by the plant turbine. These occurrences required investigation and turbine modification and are described in Sect. 4.4.2.3. Of the remaining 13 events, 4 involved the reactor trip system, 2 resulting in inadvertent reactor scrams due to fabrication and maintenance errors.

1972

San Onofre reported 16 events in 1972. Two significant events occurred. Abnormal occurrence 72-07 reported an overcooling event due to an excessive feedwater flow rate. This was caused by a regulating valve failure and is described in Sect. 4.4.2.1. The second significant event was a recurrence of overspeed problems with the turbine, experienced subsequent to a loss of generator field.

Of the remaining 14 events, three events describe primary to secondary leakage in the steam generator. These are the first instances of steam generator tube leakage reported for San Onofre.

1973

In 1973, reportable events at San Onofre were reported as station incidents (SI). Sixteen such events were reported for the year. Two of those events were considered significant. SI 73-08 was a significant event involving loss of off-site power and subsequent failure of an emergency diesel generator. This event is described in Sect. 4.5.2.3. The second significant event was another overcooling event due to excessive feedwater flow. The transient resulted in actuation of the safety injection system. This event is described in more detail in Sect. 4.4.2.7.

In addition to the diesel generator failure in SI 73-08, four other diesel generator failures were reported during the year (SI 73-03, SI 73-04, SI 73-7, and SI 73-12). Prior to 1973 only two diesel generator failures had been reported, both in 1972.

1974

Twelve station incidents were reported by San Onofre for 1974. Of these twelve reported events, only one was deemed significant. SI 74-06 involved leakage of cooling coils in containment, causing flooding and failure of a number of reactor vessel detector thimbles. This resulted in a reactor trip and is discussed in Sect. 4.5.2.7. One of the other events reported in 1974 was a sabotage threat against San Onofre which the FBI reported. The FBI received a telephone call from a Los Angeles man who claimed that a car with several persons was in route to San Onofre to 'blow up' the generating station. The threat later proved to be a hoax.

1975

San Onofre reported 24-events in 1975, three of which were significant events. During a design review required by the NRC, a single valve failure was identified that could have potentially defeat the safety injection system. This discovery was reported as SI 75-06 and is discussed in Sect. 4.5.2.1. On consecutive days in August, the plant reported failure of two different diesel generators due to loss of cooling. These events are described in Sect. 4.5.2.4. Two other diesel generator failures (SI 75-02 and SI 75-04) were also among the rest of the 24 events for 1975.

1976

Eleven events were reported to the NRC as Licensee Event Reports (LERs) in 1976. The review also examined station incidents (SI) that San Onofre continued to record in its annual operating report. Table A2.11 shows only those SIs that describe events in addition to the LER events.

Twenty non-LER events were included in the review of 1976 operating experience, including 17 station incidents. Of these, one was deemed significant. It was a loss of off-site power event caused by a brush fire. The event (SI 76-02) is described in Sect. 4.5.2.3.

The 31 total events for 1976 included five reports describing failure of an inverter in the on-site power system. Three of these failures occurred on the same day and are discussed in Sect. 4.5.2.1. An additional five reports involved minor problems with spent fuel shipping casks. These included cracked lifting lugs, minor leaks and administrative errors in handling and shipping.

1977

Eighteen events were included in the review of 1977 operating experience. Fourteen of the eighteen were LERs. None of the events were considered significant. The problems presented by the reports were varied, including continued steam generator tube leakage, administrative errors and maintenance and design deficiencies.

1978

In 1978, 17 events were reported. Of the 17, one LER described a significant event. The event involved failure of two inverters for the containment spray actuation system.

Four of the remaining 16 events reported occurrences involving the high pressure safety injection system, while an additional four involved feedwater system equipment.

1979

San Onofre submitted 26 LERs in 1979. Review of these reports identified no significant events. Five of the LERs involved events concerning loss of or failure to collect required environmental data such as drinking water samples and sea temperatures. These included loss of four environmental radiation dosimeters due to vandalism. In November, four LERs reported the discovery of pipe supports missing in three different systems.

1980

In 1980, San Onofre submitted 39 Licensee Event Reports, the largest number of reports in any year. Four of those events were considered significant. The first of these was a cooling transient due to loss of a feedwater pump. The event was due to construction activity and is described in Sect. 4.4.2.4.

A second significant event occurred in March 1980 when all the salt water cooling pumps available for the plant failed. The multiple failure was due to a number of causes and is described in Sect. 4.5.2.5. During refueling in April 1980, the third significant event occurred. Due to a seal failure, source range nuclear instrumentation was flooded and failed. This event is described in Sect. 4.5.2.7.

The final significant event of 1980 occurred in November and involved loss of on-site AC power to all station auxiliary loads. The event was caused by an operator error and is described in Sect. 4.5.2.4.

4.5.1.2 <u>Systems involved in reportable events.</u> A summary involved in reportable events is given in Table 4.6. The table totals the system codes listed in the 'system' category of Appendix A.2 and gives the number of times a reported event involved each system by year.

The most frequently reported systems were:

1. coolant recircualtion systems (29),
2. reactivity control systems (28),
3. condesate and feedwater systems (22),
4. onsite power systems (composite ac and dc) (19),
5. emergency generator systems (19), and
6. circulating water systems (18).

Eighteen of the 29 coolant recirculation system failures were damage to or leaks in the steam generators. The only other major contributors from this system were the reactor coolant pumps which accounted for six of the reports.

The majority (18 of 28) of reports involving the reactivity control system involved false indications of dropped rods. The remaining ten reports concerned actual incidents of dropped rods and stock control rod drives.

Table 4.6. Summary of systems involved in reportable events at San Onofre 1

| System | 1966 | 1967 | 1968 | 1969 | 1970 | 1971 | 1972 | 1973 | 1974 | 1975 | 1976 | 1977 | 1978 | 1979 | 1980 | Total |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Reactor vessel internals (RA) | | | | | | | | | | | 1 | | | | | 1 |
| Reactivity control (RB) | | 5 | 7 | 5 | 5 | 2 | 1 | 1 | 1 | 1 | | | | | | 28 |
| Reactor core (RC) | | | | | | | 1 | | | | | | | | | 1 |
| Reactor vessel (CA) | | 1 | | 2 | | | | | | | | 1 | | | | 4 |
| Coolant recirculation (CB) | 1 | 2 | 1 | | 1 | 1 | 3 | 1 | 1 | 2 | 4 | 3 | | 3 | 6 | 29 |
| Residual heat removal (CF) | | | | | | | | | | | | | | | 1 | 1 |
| Reactor containment (SA) | | 1 | | | | | | | | | | 2 | | | 1 | 4 |
| Containment isolation (SD) | | | | 1 | | | 1 | 1 | | 1 | | | | 3 | 1 | 8 |
| Low pressure safety injection (SF-B) | | 1 | 1 | 2 | | | | | | 1 | | | | | | 5 |
| High presure safety injection (SF-C) | | | | | 3 | | | 1 | | | | 1 | 4 | 1 | 3 | 13 |
| Control room habitability (SG) | | | | | | | | | | | | | | | 1 | 1 |
| Containment spray (SH-B) | | | | | | | | | | | | 2 | 3 | 1 | | 6 |
| Auxiliary feedwater (SH-C) | | | | | | 1 | | | | | | | | 1 | | 2 |
| Reactor trip (IA) | | | | 2 | 1 | 4 | 1 | | 3 | 2 | | | | 2 | | 15 |
| Engineered safety feature instrument (IB) | | | | | | | | | | | 1 | | | | 1 | 2 |
| Safety related display instrumentation (ID) | | | | | | | | | | | 2 | | | | 1 | 3 |
| Other instruments, safety (IE) | | | | | | | | | | | | | | | 4 | 4 |
| Other instruments, non-safety (IF) | | | | | | | | | | 1 | 2 | 2 | | 1 | | 6 |
| Offsite power (EA) | | | | | | 1 | | 2 | 1 | 1 | | | | | 1 | 6 |
| AC onsite power (EB) | | | 2 | | | | | | | | 1 | | | 1 | 1 | 5 |
| DC onsite power (EC) | | | | | | | | | | | | | | 2 | | 2 |
| Onsite power (composite AC an DC) (ED) | | | | 3 | | 1 | 1 | 2 | 1 | 3 | 6 | 1 | 1 | | | 19 |
| Emergency generator (EE) | | | | | | | 2 | 5 | 1 | 4 | 1 | 1 | 2 | 1 | 2 | 19 |
| Spent-fuel storage (FB) | | | | | | | | | | | 5 | 2 | | | | 7 |
| Fuel handling (FD) | | | | | | | | | | 1 | | | | | 1 | 2 |
| Station service water (WA) | | | | | | | | | | | | | | | 1 | 1 |
| Cooling for reactor auxiliaries (WB) | | | | | | | | | 1 | | | 1 | | 2 | 1 | 5 |
| Compressed air (PA) | | | | | | | | | | | | | | | 2 | 2 |
| Chemical, volume control and liquid poison (PC) | | | 1 | 2 | 2 | | 1 | | 1 | 1 | 1 | 1 | | | 1 | 11 |
| Air conditioning, heating, cooling and ventilation (AA) | | 1 | | 1 | | | | | | | | | | | | 2 |
| Turbine generators (HA) | | 1 | | 1 | 1 | 4 | 3 | | | | 1 | | | | | 11 |
| Main steam supply (HB) | | 1 | | | 2 | 1 | | | 1 | | 1 | | | 2 | 3 | 11 |
| Circulating water (HF) | | 2 | 3 | 2 | | | | | | 3 | | | 2 | 1 | 5 | 18 |
| Condensate and feedwater (HH) | | | | 2 | 1 | | 2 | 1 | 1 | 1 | 2 | | 4 | 3 | 5 | 22 |
| Liquid radioactive waste management (HA) | | 2 | | | | | | | | | | | | 1 | | 3 |
| Process and effluent radiological monitoring (MC) | | | | | | | | | | | | | 3 | | | 3 |
| Area monitoring (BA) | | | | | | | | | | 1 | | | | | | 1 |
| Airborne radioactivity monitoring (BB) | | | | 1 | | | | | | 2 | | 1 | | 3 | 1 | 8 |
| System code not applicable (ZZ) | | | | 1 | 2 | 3 | | 4 | 1 | | 2 | | | | | 13 |
| Total | 1 | 17 | 15 | 25 | 18 | 18 | 16 | 18 | 12 | 25 | 31 | 18 | 19 | 28 | 43 | |

All 19 of the composite ac and dc onsite power system events were inverter failures. The remaining three systems were not attributable to any single type or class of failures.

4.5.1.3 Causes of reportable events. Table 4.7 lists the causes of reportable events by year. The largest group of events (181) were those caused by inherent equipment failure. Human errors (93) totaled only about one-half the number of equipment failures. These errors included administrative, design, fabrication, installation, maintenance, and operator errors.

The remaining twenty events resulted from natural and other phenomena which were outside normal operations. On seven occasions, rain storms were the cause of blocked salt water intake. Earthquakes were reported 11 times. A brush fire knocked out offsite power lines once and there was a report of a sabotage threat in the twentieth event.

4.5.1.4 Radioactivity release summary of reportable events. Table 4.8 gives a summary by year of the total radioactivity released from San Onofre Unit 1. Only three radioactivity releases resulted in reportable events. There were no releases in excess of the Tech. Spec. limits.

The first reportable event reporting a radioactivity release occurred while pumping out a radwaste tank on June 28, 1967. Radioactive liquid was being pumped through a monitor tank which had a faster pump-out rate and a low-level cutoff which must be reset manually. The operator was called away during the operation and could not reset the cutoff, resulting in the tank's overflowing. The overflow was routed to the reactor auxiliary building sump, which on high level is automatically pumped to the decontamination drain tank. This tank also overflowed causing the sump to

Table 4.7. Summary of causes of reportable events at San Onofre 1

| Cause | 1966 | 1967 | 1968 | 1969 | 1970 | 1971 | 1972 | 1973 | 1974 | 1975 | 1976 | 1977 | 1978 | 1979 | 1980 | Total |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Administration | | | | | | 1 | | | | | | 2 | | 3 | 1 | 7 |
| Design | | 2 | 2 | 2 | | | | | 1 | 3 | 2 | 2 | 1 | | 5 | 20 |
| Fabrication | | 2 | | 1 | | 1 | | | | | 2 | 2 | | | 2 | 10 |
| Inherent failure | | 7 | 10 | 20 | 16 | 10 | 14 | 9 | 10 | 17 | 18 | 8 | 13 | 14 | 15 | 181 |
| Installation | | 2 | 2 | | | | | 1 | | | 2 | 2 | | 5 | 5 | 19 |
| Maintenance | 1 | 1 | 1 | 1 | | 2 | | | | | 1 | 1 | 1 | | 6 | 15 |
| Operator | | 3 | | | 1 | | 2 | 2 | | 3 | 2 | 1 | 1 | 4 | 3 | 22 |
| Weather | | | 1 | 1 | 1 | 3 | | 4 | | 1 | 4 | | 2 | | 2 | 19 |
| Cause not applicable | | | | | | | | | 1 | | | | | | | 1 |

Table 4.8. Summary of radioactivity released at San Onofre 1

| | 1967[a] | 1968 | 1969 | 1970 | 1971 | 1972 | 1973 | 1974 | 1975 | 1976 | 1977 | 1978 | 1979 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **Airborne** | | | | | | | | | | | | | |
| Total noble gases | 4.02E+00[b] | 4.83E+00[b] | 2.56E+02[d] | 4.2E+02 | 7.67E+03 | 1.91E+04 | 1.10E+04 | 1.78E+03 | 1.11E+03 | 4.16E+02 | 1.54E+02 | 1.81E+03 | 6.05E+02 |
| Total I-131 | NA | NA | NA | NA | NA | 4.43E-05 | 4.20E-01 | 1.88E-04 | 4.50E-03 | 4.13E-03 | 1.81E-04 | 2.21E-04 | 1.22E-04 |
| Total halogens | NA | NA | NA | NA | NA | 4.42E-05 | 6.5E-01 | 2.31E-04 | 2.45E-01 | 4.48E-03 | 1.81E-04 | 2.21E-04 | 1.22E-04 |
| Total particulates | NA | NA | NA | NA | NA | 4.30E-04 | 1.2E+00 | 8.74E-05 | 3.58E-02 | 1.82E-05 | 4.83E-06 | 2.50E-03 | 2.10E-05 |
| Total tritium | NA | NA | 2.47E+00 | 2.09E+01 | 5.36E+01 | 2.81E+02 | 2.69E+02 | 9.14E+01 | 3.43E+01 | 4.72E+01 | 7.57E+01 | 5.75E+01 | 2.82E+01 |
| **Liquid** | | | | | | | | | | | | | |
| Total mixed product | 3.17E-01[c] | 1.64E+00[c] | 8.00E+00[e] | 7.60E+00 | 1.54E+00 | 3.03E+01 | 1.60E+01 | 5.04E+00 | 1.22E+00 | 7.43E+00 | 9.84E+00 | 1.18E+01 | 1.10E+01 |
| Total tritium | NA | NA | 3.53E+03 | 4.80E+03 | 4.57E+03 | 3.48E+03 | 4.07E+03 | 3.81E+03 | 4.00E+03 | 3.39E+03 | 1.79E+03 | 2.50E+03 | 2.32E+03 |
| Total noble gases | NA | NA | NA | 5.59E+01 | 3.06E+01 | 5.43E+00 | 5.36E+01 | 3.37E+00 | 4.77E+00 | 1.26E+01 | 4.81E+00 | 1.08E+01 | 1.81E+01 |
| **Solid** | | | | | | | | | | | | | |
| Total | NA | 1.13E-01 | 4.05E+00 | 1.05E+01 | 1.19E+00 | 7.97E+01 | 3.81E+02 | 2.30E+02 | 2.60E+01 | 6.98E+02 | 6.02E+01 | 7.17E+00 | 9.24E+01 |

NA – not available.

[a]Period covers June through December 1967.

[b]Reported as total liquid release.

[c]Reported as total gaseous release.

[d]Reported as total beta and gamma activity of gaseous releases.

[e]Reported as total beta and gamma activity of liquid releases.

backup through the floor drains, flooding the lower levels of the auxiliary building.[18]

The second radioactivity release event occurred during November 1967. Draining primary loops to replace defective resistance temperature deflectors transferred considerable radioactive crud to the liquid waste tanks. Spills near the coolant drain tank caused smears to reach 538,820 DPM/sq. ft.

Finally, during the testing of the spent fuel cask and air pallet system, radioactive contamination was spread to clean areas of the plant. A temporary clean area became contaminated by the scatter of dry material during the handling of the spent fuel cask head. A smear survey revealed that contamination had been spread to the maintenance shop, the turbine plant, and the administration building. Only one employee, the supervisor of plant maintenance, had evidence of personal contamination. A decontamination team cleaned all contaminated areas.[19]

4.5.1.5 Environmental impact summary of reportable events. There were 30 environment events reported at San Onofre other than the radioactivity release events. Sixteen of the events resulted from the loss of or failure to take environmental data. This lack of data resulted from numerous causes and were not attributable to a single type of failure. Eleven earthquakes were sensed at San Onofre. None of the earthquakes caused equipment damage. Two of the events were impingement of fish or debris on the intake screens. The final event was a brush fire which knocked out offsite power lines and is discussed in Sect. 4.5.2.3.

### 4.5.2 Review of significant events

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

1. an event in which the failure or failures initiated a design basis event (DBE) as listed in Table 3.1, or
2. an event in which the failure or failures compromised a function of the engineered safety features.

Twenty events at San Onofre met the significance criteria above. Table 4.9 summarizes the significance categories from Table 3.3 for these events. The total in the table is 29 because 9 of the events required two significance categories to describe the event. The events designated as significant are discussed below and are grouped as:

1. loss of safety injection system function,
2. equipment disabled by cable tray fires,
3. loss of offsite power,
4. loss of emergency AC power supply,
5. failure of multiple salt water cooling pumps,
6. loss of boric acid injection paths, and
7. instrumentation channels fail due to flooding.

In addition, six of the significant events were DBEs reported as reportable events and are discussed in Sect. 4.4.2. Even though this report did not review the events reported in 1981, one event involved the failure of

Table 4.9. Summary of significant event categories at San Onofre 1

| | 1966 | 1967 | 1968 | 1969 | 1970 | 1971 | 1972 | 1973 | 1974 | 1975 | 1976 | 1977 | 1978 | 1979 | 1980 | Total |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| S1 – Two or more failures in redundant systems | | 1 | 1 | | | | | 1 | | 2 | 1 | | | | 2 | 7 |
| S2 – Two or more failures due to a common cause | | 1 | 3 | | | | | 1 | 1 | | | | | | 1 | 8 |
| S3 – Three or more failures | | | 2 | | | | | | | | | | | | 1 | 3 |
| S7 – Greater threat to plant safety with different plant conditions or the advent of another credible occurrence | | | | | | | | | | | | | | | 1 | 1 |
| S9 – Other | | | | 1 | | 2 | 2 | 1 | | 1 | | | | | 1 | 10 |
| Total | | 2 | 6 | 1 | | 2 | 2 | 3 | 1 | 3 | 1 | | | | 6 | 20(29)[a] |

[a]A total of 20 events were identified as significant with more than one significant category assigned per event for some events.

the safety injection system and is discussed in Sect. 4.5.2.1. Table 4.10 summarizes all significant events discussed.

4.5.2.1 Loss of safety injection system function. Two reported events presented the potential failure of the safety injection system at San Onofre had there been a demand or single failure. In addition, a 1981 report gave details concerning an actual failure of the safety injection system when a demand was present. This event was of such importance that it was included in this analysis even though it falls outside the time period covered.

In the last month prior to power operations in 1967, both safety injection recirculation pumps were found failed. Moisture penetration of the motor windings caused low megger readings. The pumps were removed from service, the windings were dried out and potential sources of in leakage were sealed.[20]

In 1975, Southern California Edison Company (SCE) and Westinghouse reviewed the design of San Onofre's safety injection system. On February 28, 1975, Westinghouse notified SCE of a potential single failure which could prevent the safety injection system from satisfying its design requirements for some accident situations. Specifically, failure of one of the two feedwater pump discharge valves to close might have resulted in a portion of both trains of the safety injection flow being diverted to the steam generators.

Safety injection for San Onofre required realignment of the main feedwater pumps. Rather than taking suction from the condensate discharge, they received borated water from the refueling water storage tank, pumped by two safety injection pumps. The feedwater pumps' normal dis-

Table 4.10. Tabulation of reports categorized as significant for San Onofre 1

| NSIC accession No. | Report No. | Significance category | Description | Section discussed in |
|---|---|---|---|---|
| | Ltr 7/15/67 | S1, S2 | Both SI recirc pumps had low megger readings because of moisture. | 4.5.2.1 |
| 23360 | Ltr 2/26/08 | S2, S3 | Cable tray fire. | 4.5.2.2 |
| 24817 | Ltr 4/8/68 | S2, S3 | Cable tray fire. | 4.5.2.2 |
| 29676 | Ltr 11/25/08 | S1, S2 | No flow from boric acid transfer pumps due to boron crystallization. | 4.5.2.6 |
| 30060 | Ltr 10/27/69 | S9 | Boric acid transfer pump plugged due to boron crystallization. | 4.5.2.6 |
| | Monthly report 6/71 | S9 | Loss of two offsite power lines. | 4.5.2.3 |
| | Monthly report 7/71 | S9 | Turbine overspeed. | 4.4.3.2 |
| 71397 | AO 72-07 | S9 | Feedwater regulating valve failure. | 4.4.2.1 |
| 74190 | Ltr 8/28/72 | S9 | Turbine overspeed. | 4.4.3.2 |
| 81591 | SI 73-08 | S1, S2 | Loss of offsite power. | 4.5.2.3 |
| 87016 | SI 73-13 | S9 | Excessive feedwater flow. | 4.4.2.1 |
| 94768 | SI 74-06 | S2 | Flooded detector thimbles. | 4.5.2.7 |
| | SI 75-06 | S9 | Potential single failure of safety injection. | 4.5.2.1 |
| 106454 | SI 75-17 | S1 | Diesel generator No. 1 overheats. | 4.5.2.4 |
| 106455 | SI 75-18 | S1 | Diesel generator No. 2 overheats. | 4.5.2.4 |
| | SI 76-02 | S1 | Loss of offsite power due to brush fire. | 4.5.2.3 |
| 154455 | LER 80-02 | S9 | Construction worker bumped a relay tripping a feedwater pump. | 4.4.2.4 |
| 155475 | LER 80-06 | S1, S3 | All salt water cooling pumps failed. | 4.5.2.5 |
| 158278 | LER 80-15 | S7 | Onsite and offsite power lost. | 4.5.2.3 |
| 156982 | LER 80-16 | S1, S2 | Source range monitor flooded. | 4.5.2.7 |

charge to the steam generators must close and injection valves must open to provide a path to the three reactor vessel cold legs.

The Westinghouse analysis showed that if one of the two motor-operated valves in the normal discharges from the feedwater pumps did not close, portions of the flow from both pumps could be diverted to the steam generator. For accidents where the secondary system pressure was lower than the reactor coolant pressure, the situation offered the greatest potential for failure of safety injection.

Administrative controls were initiated to mitigate the potential for this single point failure of the safety injection system. Design changes were scheduled for the next refueling outage. Subsequent requirements by the NRC resulted in additional single failure analyses of the safety injection system and extensive modifications.[21-22]

Even though it was beyond the time scope of this study, one event occurred in 1981 which is significant enough to note. On September 3, 1981, the unit was operating normally at 390 MW(e), 87% reactor power. A voltage regulator failure caused erratic instrument indications and the operators manually tripped the ractor. In the transient, following the manual trip, a valid safety injection actuation signal was received. However, both safety injection valves HV 851 failed to open and prevented safety injection. Tests confirmed that the valves did not meet design requirements and would not open under design delta P. Design changes and NRC approval were to be sought before leaving cold shutdown.[23]

4.5.2.2 Equipment disabled by cable tray fires. On two occasions, San Onofre experienced cable tray fires which affected power and control cables to a number of plant systems. On February 7, 1968, a fire occurred in cables leading to a penetration. Prior to the event, the unit was in power operation. The 480-V bus ground alarm initiated, indicating a 100% ground on the No. 1 480-V bus. A loud noise was heard and a security officer reported a fire at the southeast side of the containment sphere. Heating and ventilating alarms sounded. Fire fighters controlled the fire within 5 min. The No. 1 and No. 3 480-V buses were connected resulting in both buses indicated grounding. By opening various breakers, the ground was cleared when the Group C pressurizer heater breaker was opened. At this point, a power reduction was initiated to remove the unit from service.

Inspection revealed that the penetration and 65 cables were damaged by the fire. The cables were located in two cable trays one over the other. No damage was found inside containment, either to the connectors or cables. The cause of the fire was determined to be overloaded and overheated cables for the pressurizer heaters. The damaged cables were replaced with larger size cables and ventilation of the penetration area was improved. The total down time for the unit was 286 h and 44 min.

On March 12, 1968, the second cable tray fire occurred. While at power operations, various alarms sounded, including a 480-V system ground. The operator attempted to identify the cause of the alarms. Within 5 min, smoke was reported coming from the No. 2 480-V switch-gear room. Fire was observed in three cable trays in the room and 13 min into the event the reactor was manually tripped. Assistance in fighting the fire was provided by the Marine Corps fire department and the fire was extinguished 35 min after being reported.

During the fire, the diesel generator feed cables were identified as the source of the ground on the No. 2 480-V bus but since the ground was between the breakers and the bus, the entire bus was removed from service. This disabled a large number of undamaged electrical items. In fighting the fire, the fire pumps failed to start and alternate pumps were made available.

A plant cooldown was started and since power was lost to the boric acid injection pump, the boric acid transfer pump was utilized for boration. Four hours into the cooldown, the cooldown was halted because the boron concentration was decreasing rather than increasing due to blockage of the transfer pump flow by boric acid crystals. Alternate boration was provided and the cooldown completed. Damage from the March 12th fire included 185 electrical circuits, sections of three cable trays, 18 control transformers, knife switches in pressurizer heat cabinet, heating and ventilating annunciator panel and smoke damage to certain other equipment.

Investigation after the second fire altered the assessment of the cause of the two fires. Undersize cabling for the pressurizer heaters remained a contributing factor but also considered important was mechanical overloading of cables in crowded cable trays. This contributed to the thermal overloading that caused the fires. Corrective actions were extensive, requiring addition of cable trays, rerouting of cables, cable upgrading and replacement, and changes in electrical fault isolation. The unit returned to service Sept. 8, 1968 (Ref. 24).

4.5.2.3 Loss of offsite power. AC power is essential for maintaining the reactor in a safe condition for most operating situations. Two complete losses of offsite power were reported for San Onofre, one resulting in no source of AC power being immediately available. Both of these events are described below, along with a partial loss of offsite power that resulted from common cause faults of multiple power lines.

On June 7, 1973, a loss of offsite power occurred at San Onofre with subsequent failure of the No. 1 diesel generator. The reactor was in a refueling outage and the C auxiliary transformer was down for maintenance. The station's auxiliary power needs were supplied by the main transformer and the A and B auxiliary transformers. The main generator neutral current transformers had been shorted and grounded in preparation for performing the high potential test on the main generator.

At 1:56 a.m., June 7, the east vacuum pump was started. At that instant the Unit 1 main transformer relayed due to C phase differential protection. This caused a loss of offsite power, de-energizing the 4 kV and 480 V buses. At 1:59 a.m., the No. 1 and No. 2 diesel generators were started and connected to the 480 V auxiliary buses. Essential equipment including the residual heat removal pumps were restarted.

At 2:50 a.m., the No. 1 diesel generator voltage control failed causing the No. 2 diesel to trip on overload. The No. 1 diesel was removed from service. The No. 2 diesel was restarted and the bus re-energized. It was determined that the unit differential relay operation was due to the ground wire for the main generator current transformers being applied to the differential relay terminals. This situation was corrected and the main generator was re-energized without further incident. At 6:55 a.m., the station auxiliary power system was returned to normal and the diesel generator was stopped.

In addition to the maintenance error which disabled offsite power, there was a failed capacitor in the No. 1 diesel generator. The capacitor was replaced and the diesel was operable at 6:25 p.m. on June 7, 1973 (Ref. 25).

On January 21, 1976, the reactor tripped at 4:19 a.m. due to concurrent loss of two offsite power lines. The Santiago-San Onofre and Chino-San Onofre 220 kV lines both relayed open due to a brush fire beneath them. This common cause loss of two lines recurred intermittently during the day as the lines were returned to service and relayed again due to the fire. The system disturbances also opened other lines including two 138 kV lines to San Onofre. Offsite power to the plant was maintained throughout the day and the reactor returned to critical operation at 9:29 a.m. and full load operation at 4:30 p.m.[26]

For a 4 min period on April 22, 1980, all AC power to plant equipment was unavailable. The unit was in cold shutdown for refueling, with AC power to the site supplied through one 220 kV circuit breaker in the switchyard to the A and B auxiliary transformers. Auxiliary transformer C and the No. 1 diesel generator were out of service for maintenance. Both safeguard load sequencing systems were also out of service.

At 11:07 a.m., a test technician performing routine relay testing on auxiliary transformer C failed to block open a set of relay contacts which subsequently tripped the 220 kV circuit breaker supplying power to the site. This resulted in a loss of power to all the 4 kV and 480 V buses. The inservice diesel generator (No. 2) was manually started by the control room operator. However, offsite poer was restored at 11:11 a.m. prior to loading the diesel. After verifying the trip was not due to equipment failure, the 220 kV circuit breaker was reclosed.

During the 4 min period, no power was available to the charging pump, residual heat removal pumps, component cooling water pumps or the salt water cooling pumps. However, no significant temperature increases were noted during the event.[27]

4.5.2.4 Loss of emergency AC power supply. A potential loss of emergency AC power was revealed when failure of both diesel generators was experienced in the semi-annual test conducted in August 1975. The test requires both diesel generators be operated at rated load for 1 h. This is in addition to weekly and refueling outage testing.

During the test conducted on August 12, 1975, the No. 1 diesel generator tripped due to high cooling water temperature after 35 min of operation. The radiator heat exchanger air passages were blocked with corroded portions of the radiator cooling fans. To correct the problem the radiator was cleaned and the diesel generator tested satisfactorily.

On the next day, August 13, 1975, the semi-annual test of the two diesel generators was conducted again. On this occasion, the No. 2 diesel generator tripped due to high cooling water temperature after 20 min of operation. Investigation revealed that the cap closure on the coolant system standpipe was leaking. The cooling system was not maintaining sufficient pressure and coolant to operate properly. The leakage was routed to a drain and was not observable to operating personnel. A repair of the leak was made, and the drain line was modified to allow identification of leakage.[28]

4.5.2.5 Failure of multiple salt water cooling pumps. On March 10, 1980, San Onofre experienced failure of three salt water pumps to provide cooling water to the component cooling water heat exchangers. At 9:15 p.m. low flow and low discharge pressure alarms were received for the south salt water cooling pump, currently in operation. The north cooling water pump was automatically started and operators dispatched to the pump area.

The operators reported that the discharge valve for the north salt water pump had not opened as required and they attempted to open it. Meanwhile, the auxiliary salt water pump was started from the control room. However, a low flow condition was apparent on that pump also. Examination indicated a failure to prime and the pump was stopped. During this period, a load reduction was initiated as required by technical specifications.

To re-establish salt water cooling flow, the screen wash pumps were started and aligned to discharge to a component cooling water heat exchanger. By 9:30 p.m. flow was sufficient to reduce the component cooling water temperature, that had been rising. By 10:00, the operators were able to restore adequate priming to the auxiliary salt water pump and placed it in service. Unit shutdown procedures were terminated at this point. At 12:05 that evening, the discharge valve for the north salt water pump was opened and that pump returned to service. The failure of the south salt water pump was due to shaft failure from worn bearings. No failure causes were reported for the discharge valve and the failure of the auxiliary pump to prime.[29]

4.5.2.6 Loss of boric acid injection paths. During the shutdown following the fire of March 12, 1968, cooldown was halted due to failure to inject boron for reactivity control (see Sect. 4.5.2.3). The cause given was suction line blockage by boric acid crystals. This caused failure of the boric acid injection pump and both boric acid transfer pumps.

On October 14, 1968, the unit again experienced blockage of the boric acid injection pump and both boric acid transfer pumps. Periodic recirculation flow and improved heat tracing were identified as necessary for adequate system operation.[30]

A similar event occurred on July 15, 1969 when the boric acid injection pump discharge became clogged with boric acid crystals. Temperatures in the discharge piping were found to be 110°F at one point, while precipitation occurs a temperatures below 130°F. Installation of a recirculating pump and raising the heat tracing system setpoint was accomplished to correct the problem.

Other occurrences of single failures due to boric acid crystallization were reported (SI 74-03, LER 76-07, and LER 77-06). However, no losses of injection capability were noted.

4.5.2.7 Instrumentation channels fail due to flooding. On two occasions at San Onofre, failure of multiple instrumentation channels occurred due to flooding.

Failure of two power range instrument channels due to flooding caused an inadvertent reactor trip on July 7, 1974. The reactor trip occurred due to an indication of overpower on nuclear instrument channels 1206 and 1207. Three other instrument channels failed also.

At 10:15 a.m., a routine auto start test of the turbine plant cooling water pumps was conducted. A flow disturbance during the test failed two gaskets in a control rod drive mechanism cooling fan. There was no knowledge of the failure at this time. By 10:50 erratic instrument readings and control rod drive mechanism temperatures indicated a cooler leak. As action to secure the leakage was taken, the reactor tripped due to the overpower indication. Other channels indicated no over power condition actually existed. However, due to the number of channels affected, boration of the reactor was initiated.

A total of 3400 gal of water leaked through the cooler gaskets. Of this amount, ~140 gal collected in the detector thimbles. All affected instrumentation was inspected and replaced as necessary. A design change was accomplished to provide additional drainage for the coolers involved.[31]

During refueling operations on April 20, 1980, two source range instrumentation channels failed. Although alternate channels were available for indication of neutron level, no audible alarm was operable as required. Refueling operations were suspended to allow repairs to be accomplished.

Investigation revealed leakage through the seal between the reactor vessel and reactor cavity. The leak allowed water to enter the neutron detectors and PVC amps. The refueling canal was drained, the seal repaired. The detectors and alarms repaired and replaced in service.[32]

### 4.5.3 Recurring problems in reportable events

As seen from a review of the reported events at San Onofre, there were five problems which recurred over portions of the operating history. These five problem areas were:

1. inverters and vital bus power,
2. steam generator tube leaks,
3. dilution of primary coolant,
4. tsunami gate closure, and
5. erroneous control rod indications.

Erroneous control rod indications are discussed in Sect. 4.4.3.4.

4.5.3.1 <u>Inverters and vital bus power.</u> There were 21 occurrences of the momentary loss of vital bus power causing five shutdowns and nine power reductions. As illustrated by Fig. 4, these events occurred over the most of the period between 1967 and 1980. Inverter failures caused all but one of the losses of vital bus power. The exception was a phase-to-phase short in 1979.

The first vital power loss occurred on March 5, 1969. The No. 1 vital bus transferred to its back-up power supply when the No. 1 inverter experienced a fuse failure. This interruption resulted in a nuclear dropped rod signal and initiated a unit load decrease.

A turbine runback occurred on November 24, 1969 when blown fuses failed inverter No. 3. The load runback was terminated manually and the blown fuses were replaced. Maintenance found no cause for the blown fuses.

On December 18, 1969, the failure of No. 2 vital bus inverter initiated a turbine runback. The inverter failure was triggered by a failed power switch in the control rod position recorder. The unit resumed 450 MW(e) operation the same day.

The No. 4 inverter failed on March 23, 1971, indicating a dropped rod. Investigation revealed a shorted capacitor in the transformer circuit. The capacitor was replaced and inverter No. 4 returned to operation.

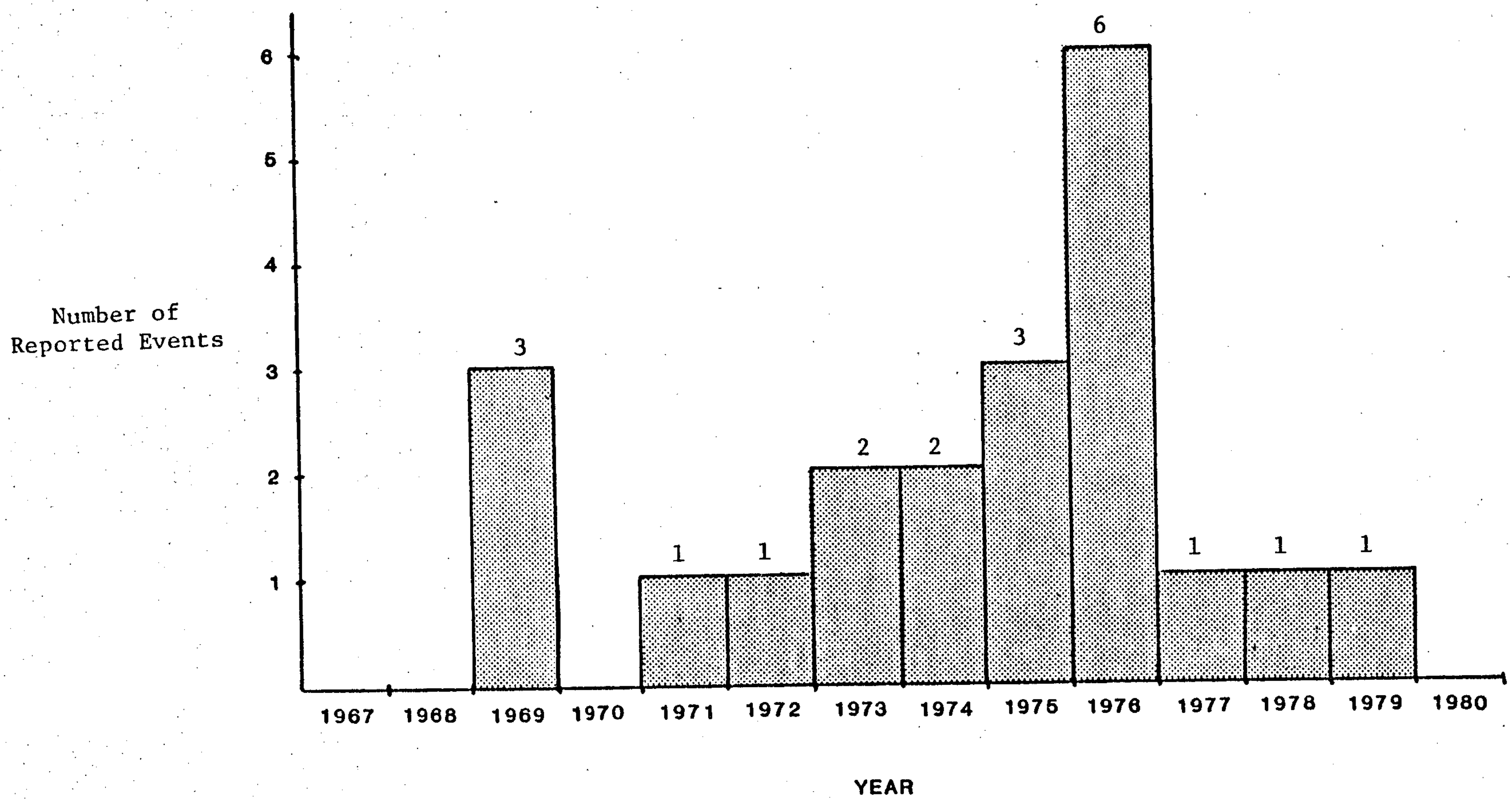


Figure 4.4 Number of Partial Vital Power Losses

The No. 3 vital bus transferred from the No. 3 inverter to its emergency power source on December 9, 1972. The transfer caused a momentary power loss to nuclear instrumentation channel 1207 leading to a nuclear dropped rod indication. Investigation revealed that the inverter low voltage power supply had failed.

On January 10, 1973, the reactor tripped when No. 4 vital bus was transferred from the backup power supply to the normal power supply. The control power switch to the No. 4 voltage regulator was inadvertently bumped and opened.

On August 17, 1973, the No. 2 inverter, which normally supplies power to power range channel 1205 and the rod position voltage regulator, failed and the load transferred to the backup power source. The resultant voltage transient caused a power reduction with indication of a dropped rod.

The unit reduced power due to a momentary power loss to nuclear power channel 1207 on June 11, 1974. The spike recurred three more times. However, an operator in continuous attendance at the control console prevented further power reductions. On June 14, 1974, the No. 1 inverter, which normally supplies power to nuclear power range channel 1208, failed and its load transferred to the backup source. The resultant voltage transient caused a spike on channel 1208 which initiated a 'Nuclear Dropped Rod Stop.' Investigation revealed that a shorted silicon controlled rectifier in the inverter caused the inverter input fuse to open and thus deenergizing the inverter.

While testing a pressurizer level channel on February 19, 1975, a second level channel spiked due to failure of the No. 2 inverter causing the unit to trip. Investigation revealed that a telecommunications crew, working in the DC switchgear room adjacent to the inverters, bumped the inverter cabinet with the foot of an extension ladder and tripped the No. 2 inverter. The No. 2 inverter would not return to service after the failure. Continued investigation indicated that there was a failed component in the undervoltage logic circuit board that should have caused the inverter to trip. Testing of the circuit breaker verified that the undervoltage trip device was striking. Apparently, bumping the cabinet was all that was required to cause the circuit breaker to operate since the logic circuit board had failed at some prior time.

On April 22, 1975, the No. 2 inverter was returned to service and the No. 2 vital bus was transferred to the No. 2 inverter, which is the normal power supply. During the transfer it was noted that No. 4 vital bus had transferred to its backup power supply. When the No. 4 vital bus was returned to its normal supply, the reactor tripped. Investigation revealed that permissive circuit P-7 was momentarily de-energized during the transfer putting the 'at power' trips in service.

An internal short caused an oil-filled capacitor to expand and rupture in the No. 3 inverter on October 19, 1975. A discussion with the manufacturer determined that the failure was random and the voltage rating was more than adequate for the service.[33] The inverter was repaired and vital bus No. 3 returned to inverter power.

On February 9, 1976, a power reduction was initiated when No. 2 inverter was found with zero output, a 'fuse open' alarm, and an obviously failed oil filled capacitor. When vital bus No. 2 transferred to its backup power, the ensuing voltage transient caused a spike in nuclear power channel 1205 which initiated a nuclear dropped rod signal.

On February 17, 1976, the No. 4 inverter failed and the load transferred to its backup source. The resultant voltage transient caused a spike on channel 1206 initiating a nuclear dropped rod alarm. No power reduction ensued as unit was operating at less than 70% of full load.

On August 23, 1976, vital bus No. 1 transferred to its backup power source. This transfer was the result of a component failure in the No. 1 inverter. The momentary loss of power precipitated a transfer of all three steam generator feedwater level control systems from their normal power supplies which are fed from vital bus No. 1 to their backup supplies. The backup positive 15 V power source had malfunctioned. The result was a loss of steam and feedwater flow signals to the three steam generator level control systems. Under these conditions the steam/feed water flow mismatch reactor trip was inoperable.

On September 25, 1976, vital bus No. 3 transferred to its backup source resulting in a power reduction. An inspection revealed a failed oil-filled capacitor in inverter No. 3. The ensuing voltage transient caused a spike in the output of a nuclear powr range channel, which initiated the nuclear dropped rod runback circuit. A half hour later, vital bus No. 2 transferred to its backup source resulting in a power reduction. The transfer resulted from a fuse failure in No. 2 inverter. The spike in channel 1205 initiated the nuclear dropped rod circuit. Inspection revealed failed components in the No. 2 inverter. Another half hour later, vital bus No. 3 transferred to its backup source a second time resulting in a third power reduction. Investigation revealed a failed capacitor in the No. 3 inverter. This event also was a direct result of a capacitor failure in the AC output filter circuit. Inspection revealed a terminal lug on a cable at one of the inductor terminal evidenced heating and arcing. The cable was improperly terminated resulting in high resistance between the cable and the lug.

During normal operation on June 14, 1977, an inverter dc input fuse opened. A single capacitor bank failed resulting in the failure of an additional capacitor bank and the fuse opening. The redundant supply was available and after all affected parts were replaced, normal power resumed through the inverter.

On June 7, 1978, the same inverter which failed on June 14, 1977 failed again as the result of a capacitor failure. The capacitor failure precipiated the failure of a silicon controlled rectifier and the dc input fuse. All affected components were replaced.

Only one loss of vital power was not the result of an inverter failure. On November 7, 1979, the unit shut down due to the loss of the 480 V bus No. 1. The cause of this event was a rodent bridging two energized phases of the bus.

4.5.3.2 <u>Steam generator tube leaks.</u> After the first five years of operation, steam generator tube leakage became a problem. Beginning in 1972, steam generator tube leaks began recurring, causing 9 forced shutdowns and 17 reportable events.

The single event prior 1972 occurred during construction in 1966. A steam generator was dropped one foot during a lift after it was inside containment. The generator withstood the hydraulic test, but 32 Inconel tubes were deformed and several tube sheets crushed. The damage was repaired and tube plugged. Calculations showed that the plugged tubes would not change plant characteristics.

Leaks from steam generators were identified on February 2, July 19, and October 13, 1972: January 5 1973; April 25, 1974; April 13 and June 11, 1975; July 1, 1979: and July 12, 1980. The most extensive repairs were made during the shutdown which commenced on July 12, 1980 and continued for 4152 h.

Prior to 1976, the problem of steam generator tube leaks was considered to be tube wall thinning around the antivibration bars. However, it was discovered that the antivibration bars were wearing and their integrity compromised. In conjunction with a visual inspection of the 'C' steam generator on September 30, 1976, an antivibration bar was removed for inspection. In several locations, the tube had worn into the bar. This problem was corrected by the design and installation of additional antivibration bars with rectangular cross section.[19]

In July 1980, Southern California Edison (SCE) installed leak tight brazed sleeves on tubing indentified with significant intergranular attack. Intergranular attack was occurring at the top of the tubesheet for tubes located in the central region of the steam generators. SCE encountered difficulty with implementation of the brazed sleeve joints in deep sludge and developed an alternate design utilizing mechanical joints.[34]

Other problems associated with steam generator tube leaks included dilution of the primary system. The next section discusses separately this recurring problem.

4.5.3.3 Dilution of primary coolant. On four occasions during extensive steam generator repair, water from the secondary side leaked to the primary side reducing the boron concentration of the primary side. Operators closely monitored boron concentration, reactor water level and other indicators and stopped the positive reactivity insertion.

The first event occurred on October 1, 1977. At the conclusion of eddy current testing, 13 tubes in the 'B' steam generator were explosively plugged. The water level on the secondary side was raised to shield employees working in the area. Approximately one-half hour later the operator noticed the secondary side level had decreased. An additional tube was leaking. The total change in boron concentration was 75 ppm from 1684 to 1609 ppm.[35]

The final three dilutions of the primary system occurred during the steam generator repairs in 1980. All involved leakage past the plugs with inflatable seals used to keep water and foreign material from entering the reactor coolant system. On July 6, a boron dilution of 400 ppm from 3357 to 2957 ppm resulted during decontamination. On September 1, a 35 ppm dilution occurred and on September 22, a 51 ppm dilution. SCE took several corrective actions to improve seal performance.[36-38]

4.5.3.4 Tsunami gate closure. The tsunami gate is the salt water intake stop gate. Its closure cuts off salt water cooling flow. There were three occasions when San Onofre experienced the closure of the tsunami gate. On November 1, 1967, a shorted limit switch caused the gate to close with one circulating water pump out of service. Rupture of the accumualtor reservoir tank caused the gate to fail on February 9, 1968. Finally on October 9, 1969, the tsunami gate slipped from its annular bolts and fell into the intake tunnel.

### 4.6 Evaluation of Operating Experience

This analysis studied 143 shutdowns and power reductions, 294 reportable events and other miscellaneous documentation concerning the operation of San Onofre Nuclear Generating Station Unit 1. The objective was to indicate those areas of plant operation which have compromised plant safety. This review identified two problems which should be of continued concern.

The first area involved the safety injection system. Section 4.5.2.1 detailed three events in which the safety injection system was actually or potentially lost. Another reason for concern are the number of random failures associated with the safety injection system. During safety injection, part of the feedwater system is used to deliver water to the reactor core. If the number of times the safety injection system was involved were combined with the number of times the feedwater system was involved, the combination would have the largest population of failures in reported events (see Table 4.5 in Sect. 4.5.1.2).

The second area for potential continued failures involved inverter failures. A single inverter failure has little consequence. However, with the high frequency of failures as exhibited at San Onofre the potential for a combination of failures is greatly enhanced. Several of the inverter failures were attributed to oil-filled capacitor failures. The capacitors continued to fail despite the vendors assurance that a similar failure was random (see Sect. 4.5.3.1).

Steam generator and condenser tube leaks continued to plague San Onofre through 1980. Associated with major steam generator repairs, dilution of the reactor coolant boron concentration also continued to recur during 1980. Other problems such as dropped control rods, pressurizer relief valves, turbine overspeed and tusunami gate closure occurred only over short periods of time and were solved.

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Appendix A: San Onofre 1

Part 1. Forced Shutdown and Power Reduction Tables

Table A1.1 1967 Forced Shutdowns and Power Reductions for San Onofre 1

| No. | Date (1967) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 7-9 | N/A | 0 | | While performing operator training start-ups with the reactor, unexplained rod drops occurred in subgroup 4 of control rod group No. 1. Investigation revealed that a timer relay was malfunctioning. | A | 3 | Reactor (RB) | Relays | D4.3 |
| 2 | 7-17 | 1080 | 13 | LTR 8/67 | During turbine overspeed testing, salt leaks were noted in the condenser. Inspection of turbine revealed significant blade damage. All of the blades in the four last stage rows were replaced. | A | 1 | Steam & Power (HA) | Turbines | N1.1.4 |
| 3 | 9-11 | 192 | | LTR 11/20/67 | Unit shutdown due to excessive circulating water pump vibration and leaking pressurizer spray valve. Upon investigation, high circulating water pump pressure and high pressure differential across the condenser were also detected. Also the system was pulling in an excessive amount of marine growth following a heat treat of the circulating water system the previous day. | A | 1 | Steam & Power (HF) Reactor Coolant (CA) | Pumps Valves | N3.1 |
| 4 | 10-1 | 216 | | LTR 10/9/67 | Repair of leak on east main steam line maintenance block valve bypass valve. Also two pressurizer power relief valves were also repaired. | A | 1 | Steam & Power (HB) Reactor Coolant (CA) | Valves | N3.1 |

Table A1.1 (Continued)

| No. | Date (1967) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 5 | 10-10 | 144 | 30 | | Reactor tripped by inadvertent signal in the variable low pressure trip channel. Occurred while circuitry was being tested with one channel placed in the trip mode for testing. | G | 3 | Instrumentation & Controls (IA) | Instrumentation & Controls | N2.4 |
| 6 | 10-19 | | 75 | | Unit taken off line due to false indication of inserted control rod. | A | 1 | Reactor (RB) | Instrumentation & Controls | N2.4 |
| 7 | 11-1 | 24 | 90 | LTR 12/11/67 | With one circulating water pump out of service, the unit was manually tripped due to an inadvertent closure of the tsunami gate. A shorted limit switch was found to be the cause of the closure. | A | 3 | Steam & Power (HF) | Circuit Closers/ Interrupters (Switches) | N1.1.4 |
| 8 | 12-5 | | 85 | | Reactor tripped on incorrectly set overpower set point. | G | 3 | Instrumentation & Controls (IA) | Instrumentation & Controls | N6.1 |
| 9 | 12-12 | 12 | | LTR 1/15/68 | Unit was manually tripped when five control rods in subgroup 7 were dropped. Defective electrical components replaced. | A | 2 | Reactor (RB) | Instrumentation & Controls | D4.3 |

Table A1.2 1968 Forced Shutdowns and Power Reductions for San Onofre 1

| No. | Date (1968) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 1-24 | 72 | | | Unit taken off line for containment inspection after increased radiation level was detected in the sphere. Inspection of the sphere revealed two leaking RTDs on the "A" coolant loop and the packing glands on the pressurizer spray valves leaking. | A | 1 | Reactor Coolant (CA) | Valves Instrumentation & Controls | N3.1 |
| 2 | 2-7 | 286 | 95 | LTR 2/26/68 | Reactor manually tripped after fire was observed in sphere electrical penetrations. Probable cause was overloaded power cables for the pressurizer heaters. | A | 2 | Reactor Coolant (CA) | Heaters, Electric Electrical Conductors (Cable) | N1.1.4 |
| 3 | 3-4 | 6 | | | Cleaning of the rod control slave cycler contacts was in progress and the subgroup #6 slave cycler had been replaced with a shutdown cycler so that the contacts could be cleaned. An attempt was made to cycle the rods one step and subgroup #6 was dropped to 160 steps. The slave cycler was examined and no reason for the failure was found. | G | 3 | Reactor (RB) | Control Rod Drive Mechanisms | D4.3 |
| 4 | 3-9 | 43 | | LTR 4/68 | Unit taken off line when physics data indicated a bottomed control rod. Investigation revealed that the the polarity of the movable gripper coil had been reversed during repairs to penetration EPC-4. | G | 3 | Reactor (RB) | Control Rod Drive Mechanisms | D4.3 |

Table A1.2 (Continued)

| No. | Date (1968) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 5 | 3-12 | 4332 | 90 | LTR 4/8/68 | Unit manually tripped after a fire occurred in a cable tray in the switchgear area. Probable cause was overloaded pressurizer heater cables. | A | 2 | Reactor Coolant (CA) | Heaters, Electric Electrical Conductors (Cable) | N1.1.4 |
| 6 | 9-9 | 43 | | | While preparing to roll the turbine all five rods in subgroup 7 slipped from 117 to 40 steps and the reactor tripped immediately. Probable cause was traced to BF relays in rod control system racks. | A | 3 | Reactor (RB) | Control Rod Drive Mechanisms | D4.3 |
| 7 | 9-19 | 152 | | LTR 12/68 | Unit removed off line to repair the letdown isolation valve packing, replace three failed reactor coolant system RTDs, adjust the load runback arming point and perform other minor maintenance. Also reactor coolant pump "A" was found rotating backwards as a result of the anti-rotation pawls not properly engaging. These were modified on all three pumps. | A | 1 | Auxiliary Process (PC) Reactor Coolant (CA) | Valves Instrumentation & Controls Pumps | N1.1.4 |
| 8 | 9-26 | 87 | 33 | LTR 12/68 | Unit manually tripped when control rods in subgroup 8 dropped into the core while a slave cycler failure alarm was being cleaned. (This occurred when an operator manually opened, closed, and reopened the half-power contactor while attempting to clear a slave cycler failure alarm.) The cause of the slave cycler failure was due to improper operation of the slave cycler clutch. | A | 2 | Reactor (RB) | Control Rod Drive Mechanisms | D4.3 |

Table A1.2 (Continued)

| No. | Date (1968) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 9 | 12-28 | ~84 | 100 | LTR 2/69 | Leakage through the pressurizer relief valve RV-533 was observed on 12-9-68 by increased temperature in valve discharge piping. Continued observation mode until a normal shutdown was initiated. | A | 1 | Reactor Coolant (CA) | Valves | N3.1 |

Table A1.3 1969 Forced Shutdowns and Power Reductions for San Onofre 1

| No. | Date (1969) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| | 1-1 | ~160 | | | Continuation of outage of 12-29-68, repair of relief valve. | A | 4 | Reactor Coolant (CA) | Valves | N3.1 |
| 1) | 1-9 | 44 | | LTR 2/69 | Unit removed off line after a routine test of the stop valves produced a spurious partial turbine trip alarm. Investigation revealed that the turbine auto-stop oil supply pressure was low due to a badly scored seat and disc in the auto-stop oil dump valve. | A | 1 | Steam & Power (HA) | Valves | D2.3 |
| 2) | 3-5 | 0 | | | Power reduction. Vital bus inverter failure. | A | 5 | Electric Power (ED) | Inverters | N1.1.4 |
| 3) | 3-8 | 14 | 75 | | Reactor tripped due to relay malfunction in turbine control valve servomotor. (Replaced the turbine control valve control oil plungers as the original design permitted the control valve servomotors to oscillate at high frequency.) | A | 1 | Steam & Power (HA) | Relays | N1.1.1 |
| 4) | 3-22 | 314 | 100 | | Unit shutdown to replace pressurizer safety valves and repair components contributing to reactor coolant system leakage. | A | 1 | Reactor Coolant (CA) | Valves | N3.1 |
| 5) | 4-7 | 0 | 100-50 | | Power reduction. Load runback was experienced due to unexplained false dropped rod signal. The load runback pressure switch had drifted. | A | 5 | Reactor (RB) | Instrumentation & Controls | N2.4 |

Table A1.3 (Continued)

| No. | Date (1969) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 6) | 4-29 | 6 | | | While heat treating the circulating water system a routine stop valve test was performed. The left hand stop valve failed to reopen and the unit was removed from service. The left hand stop valve auto-stop oil solenoid dump valve had a piece of foreign material in it. | A | 1 | Steam & Power (HA) | Valves | N1.1.4 |
| 7) | 6-20 | 1193 | | | Unit shut down to inspect the turbine-generator and modify the steam generator moisture separators. | B | 1 | Steam & Power (HA) (HB) | Turbines Heat Exchangers (Steam Generators) | N1.1.1 |
| 8) | 8-10 | 80 | | | Unit shut down to correct leakage of three pressurizer relief valves. | A | 1 | Reactor Coolant (CB) | Valves | N3.1 |
| 9) | 8-14 | 86 | | | Reactor tripped when control rods associated with subgroup 7 dropped into the core due to an intermittent open contact of a BF type relay in the rod control logic. Reactor cool down followed to allow work on the pressurizer relief valves. | A | 3 | Reactor (RB) Reactor Coolant (CA) | Instrumentation & Controls Valves | D4.3 |
| 10) | 10-9 | 165 | | LTR 10/24/69 | Unit removed from service due to the intake stop gate (tsunami gate) which had slipped from its anchor bolts and dropped into the intake tunnel. | A | 1 | Steam & Power (HF) | Valves | N1.1.4 |

Table A1.3 (Continued)

| No. | Date (1969) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 11) | 10-28 | 60 | | LTR 2/12/70 | Unit shutdown to repair the root valves of the pressurizer instrument column. | A | 1 | Reactor Coolant (CA) | Valves | N1.1.4 |
| 12) | 11-24 | | | | Power reduction. Vital bus inverter failure. | A | 5 | Electric Power (ED) | Inverters | N1.1.4 |
| 13) | 12-18 | | | | Power reduction. Vital bus inverter failure. | A | 5 | Electric Power (ED) | Inverters | N1.1.4 |

Table A1.4 1970 Forced Shutdowns and Power Reductions for San Onofre 1

| No. | Date (1970) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 5-29 | 2 | 0 | LTR 7/10/70 | During startup, the turbine was manually tripped causing a reactor trip when it was noted that the turbine acceleration was faster than desired. | A | 2 | Steam & Power (HA) | Turbines | N1.1.4 |
| 2 | 11-23 | 73 | | | Unit removed from service to complete relocation of the 220 KV switchyard. | B | 1 | Electric Power (EB) | Circuit closers/ Interrupters (Switchgear) | N1.1.1 |

Table A1.5 1971 Forced Shutdowns and Power Reductions for San Onofre 1

| No. | Date (1971) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1 | 3-18 | 8 | | | A reactor trip was experienced due to a spurious signal from Channel I variable low pressure trip circuit while Channel III was in a tripped position for maintenance. | A | 3 | Instrumentation & Controls (IA) | Instrumentation & Controls | N2.4 |
| 2 | 5-1 | 180 | | | Unit was removed from service to repair a pressurizer spray valve flange leak, replace five reactor coolant system RTDs and to plug reheater tube leaks. | B | 1 | Reactor Coolant (CA) Steam & Power (HH) | Valves Instrumentation & Controls Pipes, Fittings | N3.1 |
| 3 | 6-22 | 6 | | LTR 8/12/71 | Unit was tripped by backup overspeed protection when a 220 KV line relayed. | A | 3 | Electric Power (EA) | Circuit Closers/ Interrupters (Switchgear) | D2.2 |
| 4 | 6-25 | 28 | | | Unit removed from service to repair the HP turbine extraction drain piping, leak test main condenser tubes, and inspect equipment in the containment sphere. | B | 1 | Steam & Power (HA) | Pipes, Fittings | N3.2 |
| 5 | 6-27 | 17 | | | Unit removed from service to repair condenser tube leaks. | A | 1 | Steam & Power (HC) | Pipes, Fittings | N3.2 |
| 6 | 7-9 | 32 | | | Unit was removed from service to allow switchyard construction work. | B | 1 | Electric Power (EB) | Circuit Closers/ Interrupters (Switchgear) | N1.1.4 |

Table AI.5 (Continued)

| No. | Date (1971) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 7 | 7-12 | 12 | | | Unit tripped from a generator out-of-step condition. | A | 3 | Steam & Power (HA) | Heat Exchangers (Steam Generators) | D2.2 |
| 8 | 7-24 | 30 | | | Unit removed from service to allow switchyard construction work. | B | 1 | Electric Power (EB) | Circuit Closers/ Interrupters (Switchgear) | N1.1.1 |
| 9 | 10-27 | 104 | | | Unit removed from service to repair reheater tube leaks and apply epoxy to the condenser tube inlet ends. | A | 1 | Steam & Power (HH) | Pipes, Fittings | N3.2 |
| 10 | 11-1 | 3 | | | Unit manually tripped to obtain additional data on turbine overspeed characteristics. | A | 2 | Steam & Power (HA) | Turbines | N7.0 |
| 11 | 11-3 | 3 | | | Unit manually tripped to obtain additional data on turbine overspeed charteristics. | A | 2 | Steam & Power (HA) | Turbines | N7.0 |
| 12 | 11-5 | 3 | | | Unit manually tripped to obtain additional data on turbine overspeed charteristics. | A | 2 | Steam & Power (HA) | Turbines | N7.0 |

Table A1.6 1972 Forced Shutdowns and Power Reductions for San Onofre 1

| No. | Date (1972) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1) | 2/25 | 7 | | | Unit removed off line for balancing of turbine shaft and mechanical overspeed trip testing. | A | 1 | Steam & Power (HA) | Turbines | N1.1.4 |
| 2) | 3/4 | 4 | | | Unit removed off line to conduct turbine overspeed tests. | A | 1 | Steam & Power (HA) | Turbines | N1.1.4 |
| 3) | 3/24 | 12 | | | Unit removed from service to reset the turbine mechanical overspeed trip set point. | A | 1 | Steam & Power (HA) | Turbines | N2.3 |
| 4) | 4/29 | ~12 | | | Unit removed from service to perform modifications to the main steam control valves. | A | 1 | Steam & Power (HB) | Valves | N1.1.1 |
| 5) | 4/30 | 27 | 12 | LTRs 5/9/72 5/30/72 | During startup, the reactor and turbine were tripped on a high steam generator level signal. Failure of positioner of main feed-water regulating valve. | A | 3 | Steam & Power (HH) | Valves | D1.2 |
| 6) | 5/18 | 162 | | | Unit removed from service to inspect the turbine control valves, complete reheater and condenser repairs, and perform miscellaneous maintenance. | B | 1 | Steam & Power (HA) (HC) (HH) | Valves Heat Exchangers | N3.2 |
| 7) | 7/19 | 177 | | | Unit removed from service to repair a tube leak in steam generator C. (On 7/8 a slight increase in radio-active concentration, channel 1216; was noted.) Analysis indicated leak was on "C" steam generator. Sampling began with leak rates calculated through 7/18 when leakage was approximately 100 gal/d. | A | 1 | Steam & Power (HB) | Heat Exchangers (Steam Generators) | N3.1 |

Table A1.6 (Continued)

| No. | Date (1972) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 8) | 7/27 | 33 | 0 | | During startup, the main exciter motor failed mechanically when being started. The problem was determined to be rubbing between the rotor and stator. | A | 3 | Steam & Power (HA) | Generators and Motors | D2.2 |
| 9) | 7/29 | 3 | 100 | LTR 8/28/72 | Unit tripped from loss of main generator field. | A | 3 | Steam & Power (HA) | Generators | D2.2 |
| 10) | 9/20 | 4 | 62 | | During a "Control Rod Exercise Test" an automatic load limit runback was initiated from "Nuclear Dropped Rod" circuitry. The reactor was manually tripped and the control rod circuitry inspected. Contactors and relays cleaned but no abnormal conditions were found. | A | 2 | Reactor (RB) | Instrumentation & Controls | N1.1.4 |
| 11) | 9/20 | 65 | | LTR 9/29/72 | The reactor was manually tripped after a high temperature alarm was received on the east main feed pump inboard motor bearing. Excessive thrust bearing clearance was found in the pump. | A | 2 | Steam & Power (HH) | Pumps | N1.1.4 |
| 12) | 10/13 | 155 | | | On 9/12/72, analysis determined that the "A" steam generator was experiencing primary to secondary side leakage. Sampling began and continued until 10/13 when leakage was over 100 gal/d. Thus unit removed from service to repair "A" steam generator tube leakage. | A | 1 | Steam & Power (HB) | Heat Exchangers (Steam Generators) | N3.1 |
| 13) | 12/9 | 0 | 100-60 | | Power reduction. A transfer of No. 3 vital bus from the No. 3 inverter to its emergency power source caused a momentary power loss to NIS channel 1207 initiating a "Nuclear Dropped Rod" indication. | G | 5 | Reactor (RB) Electric Power (ED) | Instrumentation & Controls Electrical Conductors (BUS) | N6.3 |

Table A1.7 1973 Forced Shutdowns and Power Reductions for San Onofre 1

| No. | Date (1973) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1) | 1/6 | 108 | 100 | LTR 2/15/73 | The unit was removed from service for locating and repairing "A" steam generator tube leaks. (On 11/2/72, analysis indicated a leak existed between steam generator primary and secondary systems.) Sampling began and continued until leak rate was approximately 100 gal/d. | A | 1 | Steam & Power (HB) | Heat Exchangers (Steam Generators) | N3.1 |
| 2) | 1/10 | 1 | 0 | | During startup the reactor tripped when No. 4 vital bus was transferred from the backup power supply to the normal power supply. It is believed that the control power switch to No. 4 voltage regulator was inadvertently opened by being bumped. | A | 3 | Electric Power (ED) | Electrical Conductors (BUS) Circuit Closers/ Interrupters (Switches) | N6.3 |
| 3) | 2/20 | 0 | ?-? | | Power reduction. While unit load was being reduced for a condenser tube cleaning outage, observation of the rod position recorder indicated that a control rod had gone from 200 to 145 steps on the recorder. | A | 5 | Reactor (RB) | Control Rod Drive Mechanisms | D4.3 |
| 4) | 8/17 | 0 | 100-75 | | Power reduction. Initiated when failure of an inverter caused a load transfer. (Voltage transient) | A | 5 | Electric Power (ED) | Generators (Inverters) | N1.1 |
| 5) | 10/21 | 1974 | 100 | LTR 10/22/73 10/31/73 | Unit was being removed from service to investigate turbine problems, indicated by bearing vibration and salt water leakage when a safety injection actuation and reactor trip was experienced. Investigation revealed turbine blade failure. | A | 3 | Steam & Power (HA) (HC) Engineered Safety Features (SF-C) | Turbines Heat Exchangers (Condensers) Instrumentation & Controls | D5.1 |

Table A1.8 1974 Forced Shutdowns and Power Reductions for San Onofre 1

| No. | Date (1974) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DEE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| | 1/1 | 516 | | | Continuation of the October 21, 1973 outage. | A | 4 | Steam & Power (HA) (HC) | Turbine Heat Exchangers (Condensers) | D5.1 |
| 1) | 4/27 | 547 | 100 | | Repair of steam generator and reheater tube leaks and repair of leaking pressurizer safety valve. Prior to shutdown there were indications of a partial loss of fan capacity in generator hydrogen gas blower. | B | 1 | Steam & Power (HB) | Heat Exchangers (Steam Generators) | N3.1 |
| 2) | 6/11 | 0 | 100-40 | | Power reduction. Automatic load limit runback initiated by a momentary spike of a nuclear power channel. | A | 5 | Instrumentation & Controls (IA) | Instrumentation & Controls | N2.4 |
| 3) | 6/14 | 0 | 100-40 | | Power reduction. Automatic load limit runback initiated when an inverter which supplies power to nuclear power range channel 1208 failed and its load was transferred to the backup source. | A | 5 | Instrumentation & Controls (IA) | Generators (Inverters) | N1.1.4 |
| 4) | 7/7 | 55 | 100 | LTR 7/15/74 | Trip from indicated overpower condition caused by water intrusion into detectors of two power range channels due to gasket failure on cooler of rod drive cooling fan. | A | 3 | Reactor (RB) Instrumentation & Controls (IA) | Control Rod Drive Mechanisms Instrumentation & Controls | N3.1 |

Table A1.8 (Continued)

| No. | Date (1974) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 5) | 7/9 | 0 | 70-35 | | Power reduction. While increasing load there appeared to be a flow restriction in the feedwater line to steam generator "C". Upon investigation, the flapper of the associated check valve was found detached from the arm and lying in the check valve body thus causing a flow restriction. | A | 5 | Steam & Power (HH) | Valves | N1.1.4 |
| 6) | 8/20 | 5 | 100 | LTR 8/20/74 | Spurious trip on indicated pressurizer high level while testing level channels. | A | 3 | Reactor Coolant (CA) | Instrumentation & Controls | N2.4 |
| 7) | 10/21 | 7 | 0 | | While returning to full load, unit was manually tripped because of dropped rods (control bank 2 slipped into core). No cause found. | A | 2 | Reactor (RB) | Control Rod Drive Mechanisms | D4.3 |

Table A1.9 1975 Forced Shutdowns and Power Reductions for San Onofre 1

| No. | Date (1975) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1) | 2/19 | 4 | 100 | | While testing a pressurizer level channel, a second level channel spiked due to a failure in the No. 2 inverter. Trip from pressurizer high level. | A | 3 | Reactor Coolant (CA) | Generators (Inverters) | N1.1.4 |
| 2) | 4/22 | 2 | | | While transferring the No. 4 vital bus back to its normal power supply the reactor tripped. | A | 3 | Electric Power (ED) | Electrical Conductors (BUS) | N6.3 |
| 3) | 5/21 | 10 | 100 | | Reactor manually tripped from restricted circulating water flow caused by seaweed fouling intake structure. | H | 2 | Steam & Power (HF) | Pipes, Fittings | N9.2 |
| 4) | 6/11 | 127 | | | Shutdown to repair pressurizer safety valves. Also plugged leaking steam generator tube. | B | 1 | Steam & Power (HB) Reactor Coolant (CA) | Heat Exchangers (Steam Generators) Valves | N3.1 |

Table A1.10 1976 Forced Shutdowns and Power Reductions for San Onofre 1

| No. | Date (1976) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1) | 1/8 | 0 | 100-50 | | Power reduction. Plug 3 leaking tubes in north half of "A" condenser. Also heat treat circulating water tunnels. | B | 5 | Steam & Power (HC) | Heat Exchangers (Condensers) | N3.2 |
| 2) | 1/21 | 7 | | | Loss of off-site power due to fire burning in San Clemente area. | H | 3 | Electric Power (EA) | Electrical Conductors | N9.0 |
| 3) | 1/22 | 0 | 100-60 | | Power reduction. Load reduced as a precautionary measure when brush fires reached vicinity of 220 kv transmission lines. | H | 5 | Electric Power (EA) | Electrical Conductors | N9.0 |
| 4) | 1/26 | 0 | 100-10 | | Power reduction. Leaking condenser tube in north half of "A" condenser resulted in an increased turbine exhaust backpressure. | B | 5 | Steam & Power (HC) | Heat Exchangers (Condensers) | N3.2 |
| 5) | 1/29 | 0 | 100-50 | | Power reduction. Plug leaking condenser tube in north half of "A" condenser. | B | 5 | Steam & Power (HC) | Heat Exchangers (Condensers) | N3.2 |
| 6) | 2/6 | 0 | 100-50 | | Power reduction. Condenser tube leakage corrected | B | 5 | Steam & Power (HC) | Heat Exchangers (Condensers) | N3.2 |
| 7) | 2/9 | 4 | 100 | | Spurious spike in pressurizer level while second channel was in test. | A | 3 | Reactor Coolant (CA) | Instrumentation & Controls | N2.4 |
| 8) | 3/19 | 0 | 100-25 | | Power reduction. Replace internals in "C" steam generator feedwater line check valves. | A | 5 | Steam & Power (HH) | Valves | N1.1.4 |

Table A1.10 (Continued)

| No. | Date (1976) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 9) | 3/23 | 0 | 100-50 | | Power reduction. Investigate condenser salt water in-leakage. | A | 5 | Steam & Power (HC) | Heat Exchangers (Condensers) | N3.2 |
| 10) | 3/29 | 0 | 100-50 | | Power reduction. Investigate condenser salt water in-leakage. | A | 5 | Steam & Power (HC) | Heat Exchangers (Condensers) | N3.2 |
| 11) | 4/17 | 9 | 33 | | During startup, spurious turbine trip resulted in a reactor trip. | A | 3 | Steam & Power (HA) | Turbines | D2.3 |
| 12) | 4/17 | 7 | 33 | | During startup toward full power, spurious turbine trip resulted in a reactor trip. | A | 3 | Steam & Power (HA) | Turbines | D2.3 |
| 13) | 4/18 | 20 | 15 | | Turbine trip due to incorrect setting of an overspeed trip device. | G | 3 | Steam & Power (HA) | Instrumentation & Controls | D2.3 |
| 14) | 4/19 | 6 | 75 | | Load was being reduced for preparation of overspeed testing when a faulty thermocouple indicated thrust bearing temperature on east feedwater pump was increasing. Unit removed for overspeed tests and repairs to "B" steam generator feedwater line check valve. | B | 3 | Steam & Power (HB) | Valves | N1.1.4 |
| 15) | 6/28 | 8 | 100 | LER 76-004 | While unit at reduced load for heat treating the circulating water system, the "C" loop reactor coolant flow transmitter failed resulting in trip from reactor coolant full load low flow indication. | A | 3 | Reactor Coolant (CA) | Instrumentation & Controls | N1.1.4 |
| 16) | 7/2 | 0 | 100-75 | | Power reduction. Plug leaking condenser tube. | B | 5 | Steam & Power (HC) | Heat Exchangers (Condensers) | N3.2 |

Table A1.10 (Continued)

| No. | Date (1976) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 17) | 7/14 | 16 | 100 | | Repair a control oil leak on the No. 2 turbine control valve. | A | 1 | Steam & Power (HA) | Valves | N1.1.4 |
| 18) | 7/30 | 101 | 100 | LER 76-006 | Repair leaking steam generator tubes on "C" steam generator. | A | 1 | Steam & Power (HB) | Heat Exchangers (Steam Generators) | N3.1 |
| 19) | 8/7 | 0 | 100-60 | | Power reduction. Relay coil failed which resulted in an erroneous indication of a dropped control rod. | A | 5 | Reactor (RB) | Relays | N1.1.4 |
| 20) | 9/25 | 0 | 76-70 | | Power reduction. Failures of No. 2 and No. 3 inverters supplying vital power buses resulted in an erroneous indication of a dropped control rod. | A | 5 | Electric Power (ED) | Generators (Inverters) | N1.1.4 |
| 21) | 9/25 | 0 | 76-70 | | Power reduction. No. 3 inverter failed resulting in an erroneous indication of dropped control rod. | A | 5 | Electric Power (ED) | Generators (Inverters) | N1.1.4 |

Table A1.11 1977 Forced Shutdowns and Power Reductions for San Onofre 1

| No. | Date (1977) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1) | 4/14 | 26 | 90 | | Received two rod bottom lights. Manually tripped the unit. | A | 1 | Reactor (RB) | Instrumentation & Controls | D4.3 |
| 2) | 4/21 | 27 | 100 | | Failure of both reactor cavity cooling fans. Manually tripped the unit. | A | 1 | Engineered Safety Features (SB) | Blowers (Fans) | N1.1.4 |
| 3) | 5/18 | 14 | 100 | | Failed gripper coil on control rod group-- dropped four rods. Manually tripped the reactor. | A | 2 | Reactor (RB) | Control Rod Drive Mechanisms | D4.3 |
| 4) | 6/9 | 10 | 100 | | Failed moveable gripper coil on shutdown rod group - dropped four rods. Manually tripped the reactor. | A | 2 | Reactor (RB) | Control Rod Drive Mechanisms | D4.3 |
| 5) | 6/10 | 4 | 90 | | Inadvertent trip during routine weekly testing of power range NIS instrumentation. An overtrip channel had been reset prior to testing another channel. | G | 3 | Instrumentation & Controls (IA) | Instrumentation & Controls | N6.1 |
| 6) | 7/1 | 0 | ?-50 | | Power reduction. Plug leaking tube in south half of "B" condenser. | B | 5 | Steam & Power (HC) | Heat Exchangers (Condensers) | N3.2 |
| 7) | 7/2 | 0 | ?-50 | | Power reduction. Plug additional tube in south half of "B" condenser. | B | 5 | Steam & Power (HC) | Heat Exchangers (Condensers) | N3.2 |
| 8) | 8/17 | 0 | 100-60 | | Power reduction. Repair a ground in circulating water pump motor leads. | A | 5 | Steam & Power (HF) | Motors | N1.1.4 |
| 9) | 9/9 | 646 | 95 | LER 77-013 | Reactor coolant pump inspection, S/G inspection, S/G plugging. | B | 1 | Reactor Coolant (CA) Steam & Power (HB) | Pumps Heat Exchangers (Steam Generators) | N3.1 |

Table A1.11 (Continued)

| No. | Date (1977) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 10) | 10/6 | 2 | 0 | | Reactor trip breaker undervoltage relay stuck in de-energized position. | A | 3 | Electric Power (ED) | Relays | N1.1.4 |
| 11) | 10/6 | 4 | 0 | | Reactor trip breaker undervoltage relay stuck in de-energized position. | A | 3 | Electric Power (ED) | Relays | N1.1.4 |
| 12) | 11/19 | 25 | 100 | | Repair SIS recirculation valve. | B | 1 | Engineered Safety Features (SF-C) | Valves | N1.1.4 |
| 13) | 12/29 | 3 | 95 | | Repair steam leak on turbine drain line. Turbine generator off-line. Reactor remained critical. | B | 9 | Steam & Power (HA) | Pipes, Fittings | N3.2 |

Table A1.12 1978 Forced Shutdowns and Power Reductions for San Onofre 1

| No. | Date (1978) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1) | 1/10 | 0 | 100-84 | | Power reduction. Repair ground on circulating water pump motor leak. | B | 5 | Steam & Power (HF) | Motors | N1.1.4 |
| 2) | 3/8 | 4 | 95 | | Reactor tripped from a loss of coolant flow signal. The low flow condition occurred when a power system disturbance (off-site) resulted in low voltage at the reactor coolant pump motors. | H | 3 | Electric Power (EA) | Pumps | D3.1 |
| 3) | 3/22 | 0 | 100-? | | Power reduction. Erroneous signal indicated a dropped control rod. | A | 5 | Reactor (RB) | Instrumentation & Controls | N2.4 |
| 4) | 3/29 | 0 | 100-46 | | Power reduction. Repair a condenser tube leak. | B | 5 | Steam & Power (HC) | Heat Exchangers (Condensers) | N3.2 |
| 5) | 4/5 | 472 | 100 | | Steam generator inspector. (Regulatory Requirement) | D | 1 | Steam & Power (HB) | Heat Exchangers (Steam Generators) | N8.0 |
| 6) | 5/18 | 0 | 100-66 | | Power reduction. Grounded rod position indication system component caused on erroneous dropped rod signal. | A | 5 | Reactor (RB) | Instrumentation & Controls | N1.1.4 |
| 7) | 6/9 | 0 | 100-58 | | Power reduction. Repair condenser tube leaks. Maintained reduction to defer fuel depletion. | B,F | 5 | Steam & Power (HC) | Heat Exchangers (Condensers) | N3.2 |

Table A1.12 (Continued)

| No. | Date (1978) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 8) | 8/15 | 0 | 100-83 | | Power reduction. Repair reheater drain line. | A | 5 | Steam & Power (HH) | Pipes, Fittings | N1.1.4 |
| 9) | 8/18 | 0 | 100-33 | LER 78-010 | Power reduction. Reclose a feed-water pump safety injection valve. Valve unexpectedly opened during maintenance. | G | 5 | Engineered Safety Features (SF-C) | Valves | N6.3 |
| 10) | 9/12 | 5 | 100 | | Reactor and turbine tripped while performing bearing low oil pressure trip test. | G | 3 | Steam & Power (HA) | Turbines | D2.3 |
| 11) | 11/6 | 4 | | | Spurious steam-feedwater flow mismatch. | A | 3 | Steam & Power (HB) (HH) | Instrumentation & Controls | N2.4 |
| 12) | 11/10 | 10 | | | Instrument malfunction indicating a low flow condition of the reactor coolant system. | A | 3 | Reactor Coolant (CA) | Instrumentation & Controls | N1.1.4 |

Table A1.13 1979 Forced Shutdowns and Power Reductions for San Onofre 1

| No. | Date (1979) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1) | 2/23 | 0 | 100-70 | | Power reduction. Plug leaking condenser tubes. | A | 5 | Steam & Power (HC) | Heat Exchangers (Condensers) | N3.2 |
| 2) | 4/5 | 82 | 100 | LER 79-002 | Repair a major condenser tube leak and the feedwater flow straighteners. | A | 1 | Steam & Power (HC) (HH) | Heat Exchangers (Condensers) Pipes, Fittings | N3.2 |
| 3) | 4/22 | 0 | 100-33 | | Power reduction. Repair of a steam leak in turbine steam extraction line. | B | 5 | Steam & Power (HA) | Turbines Pipes, Fittings | N3.2 |
| 4) | 5/14 | 4 | 100 | | Unit trip from 2 out of 3 variable low pressure trip channels while performing Delta T and TAVE tests. | B | 3 | Instrumentation & Controls (IA) | Instrumentation & Controls | N6.3 |
| 5) | 6/1 | 394 | 95 | LERs 79-008 79-010 | Steam generator tube leak - tubes plugged. | B | 1 | Steam & Power (HB) | Heat Exchangers (Steam Generators) | N3.1 |
| 6) | 8/29 | 10 | 95 | | Replace low voltage power supply on #2 sequencer. | A | 1 | Instrumentation & Controls (IB) | Relays | N1.1.4 |
| 7) | 8/30 | 0 | 100-50 | | Power reduction. Condenser tube leak. | B | 5 | Steam & Power (HC) | Heat Exchangers (Condensers) | N3.2 |
| 8) | 9/7 | 0 | 100-60 | | Power reduction. Condenser tube leak. | B | 5 | Steam & Power (HC) | Heat Exchangers (Condensers) | N3.2 |

Table A1.13 (Continued)

| No. | Date (1979) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 9) | 9/14 | 234 | 100 | LER 79-016 79-013 | Repair refueling water pump suction piping and replace pipe section on safety injection line. | A | 1 | Engineered Safety Features (SF-C) | Pipes, Fittings | N1.1.4 |
| 10) | 10/23 | 0 | 100-25 | | Power reduction. Decreasing condenser vacuum from an open interconnection between condenser and the hotwell. | G | 5 | Steam & Power (HC) | Heat Exchangers (Condensers) | N1.1.4 |
| 11) | 11/7 | 133 | 100 | LER 79-017 | Loss of 480V Bus No. 1. | A | 2 | Electric Power (EB) | Electrical Condustors (BUS) | N1.1.4 |
| 12) | 11/29 | 0 | 100-65 | | Power reduction. Condenser tube leak. | B | 5 | Steam & Power (HC) | Heat Exchangers (Condensers) | N3.2 |
| 13) | 11/30 | 0 | 100-80 | | Power reduction. Condenser tube leak. | B | 5 | Steam & Power (HC) | Heat Exchangers (Condensers) | N3.2 |
| 14) | 11/30 | 0 | 100-80 | | Power reduction. Condenser tube leak. | B | 5 | Steam & Power (HC) | Heat Exchangers (Condensers) | N3.2 |
| 15) | 12/12 | 0 | 100-60 | | Power reduction. Locate condenser tube leakage. | B | 5 | Steam & Power (HC) | Heat Exchangers (Condensers) | N3.2 |
| 16) | 12/16 | 0 | 100-80 | | Power reduction. Repair south circulating water pump. | A | 5 | Steam & Power (HF) | Pumps | N1.1.4 |

Table A1.14 1980 Forced Shutdowns and Power Reductions for San Onofre 1

| No. | Date (1980) | Duration (Hrs) | Power (%) | Reportable Event | Description | Cause | Shutdown Method | System Involved | Component Involved | DBE(D)/ NSIC(N) Event Category |
|---|---|---|---|---|---|---|---|---|---|---|
| 1) | 1/16 | 38 | 100 | LER 80-002 | Unit tripped from steam flow/feed-water flow mismatch trip caused by construction worker who accidently struck the closing circuit control relay to the east feedwater pump normal discharge valve. | H | 3 | Steam & Power (HH) | Relays | D2.7 |
| 2) | 1/26 | 372 | 100 | | Unit off line for TMI modification. | D | 1 | NA | NA | N8.0 |
| 3) | 2/11 | 0 | ?-60 | | Power reduction. Turbine low governor oil pressure. | A | 5 | Steam & Power (HA) | Turbines | N1.1.4 |
| 4) | 2/12 | 8 | 60 | | Unit off line to repair the turbine governor control oil pressure system. | A | 1 | Steam & Power (HA) | Turbines | N1.1.4 |
| 5) | 2/17 | 0 | 100-66 | | Power reduction. Repair salt leak in the condenser. | B | 5 | Steam & Power (HC) | Heat Exchangers (Condensers) | N3.2 |
| 6) | 3/6 | 11 | 100 | LER 80-011 | Unit manually off line to replace pressurizer relief tank rupture diaphragm. | A | 1 | Reactor Coolant (CA) | Valves | N1.1.4 |
| 7) | 7/12 | 4152 | 0 | LER 80-014 | Steam generator tube repair. Continuation of refueling outage after refueling complete. | B | 4 | Steam & Power (HB) | Heat Exchangers (Steam Generators) | N1.1.4 |

Appendix A: San Onofre 1

Part 2. Reportable Event Coding Sheets

Table A2.1 Coding Sheet for Reportable Events at San Onofre 1 - 1966

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 66-01 | 11057 | | 4/66 | A | CB | II | | A | AS | G | C7 | Steam generator was dropped during construction. |

Table A2.2 Coding Sheet for Reportable Events at San Onofre 1 - 1967

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 67-01 | 23143 | 5/19/67 | 1/26/68 | A | RB | J | | C | AG | C | N | A control rod drive mechanism jammed due to broken latch pin. |
| 67-02 | 31250 | 5/18/67 | 6/15/67 | A | SA | | | C | AY | G | N | Containment integrity was broken by opening both airlock doors. |
| 67-03 | 31250 | 5/67 | 6/15/67 | A | HB | II | | C | AY | H | N | Borated water was admitted to turbine side of steam generator during testing. |
| 67-04 | | 6/2/67 | 7/15/67 | A | SF-B | DD | | C | ED | E | S1,S2 | Both safety injection recirc pumps had low megger readings because of moisture. |
| 67-05 | 31251 | 6/28/67 | 7/15/67 | B | MA | BB | I | B | BS,OD | H | N | Radwaste tank overflowed due to operator inattention. |
| 67-06 | | 7/9/67 | 8/67 | B | RB | J | P | B | EF | D | N | Timer relay malfunction caused control rod drops. |
| 67-07 | 63382 | 7/17/67 | 8/67 | B | HA | NN | | C | AD | C | N | Turbine blade damage during turbine overspeed test from salt water leak. |

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 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 67-08 | | 7/67 | 8/67 | D | CB | DD | | C | ED | D | N | Low megger readings on RCPs caused by wet nitrogen. |
| 67-09 | 31253 | 9/11/67 | 10/9/67 | B | CA | CC,OO | | B | AU | D | N | Leaking pressurizer safety valve. |
| 67-10 | 31254 | 10/67 | 11/20/67 | B | HF | DD | | B | HC,HD | D | N | Excessive vibration of main circulating pumps due to mussels in sea-water intake lines. |
| 67-11 | 31254 | 10/67 | 11/20/67 | B | RB | J | K | B | EG | D | C7 | False indication of control rod in core. |
| 67-12 | 31254 | 10/67 | 11/20/67 | B | AA | Z | | B | AE | B | N | Reactor-head ventilation duct collapsed. |
| 67-13 | 31255 | 11/1/67 | 12/11/67 | B | HF | OO | K | B | ED | D | C7 | Closure of tsumani gate because of shorted limit switch caused a scram. |
| 67-14 | 31255 | 11/67 | 12/11/67 | B | MA | BB | | B | BS,OD | H | N | Radioactive spills near coolant drain tank. |
| 67-15 | 31256 | 12/12/67 | 1/15/68 | B | RB | J | | B | AS | E | C7 | Five control rods dropped. |
| 67-16 | 31256 | 12/67 | 1/15/68 | B | RB | J | K | B | EF | D | C7 | Control rod indicators were erratic. |
| 67-17 | 31256 | 12/67 | 1/15/68 | B | CB | II | | B | HB | B | N | Steam generator "B" had a high $\Delta T$. |

Table A2.3 Coding Sheet for Reportable Events at San Onofre 1 - 1968

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 68-01 | | 1/20/68 | 2/8/68 | B | HF | U | | B | HB | I | N | Condenser flow blocked by marine debris. |
| 68-02 | 31257 | 1/68 | 2/8/68 | B | RB | J | K | B | EE | D | N | Control rod indicator light lit due to open circuit. |
| 68-03 | 23360 | 2/7/68 | 2/26/68 | B | EB | G | | B | BY | B,E | S2,S3 | Fire in cable tray. |
| 68-04 | | 2/9/68 | 3/68 | B | HF | P | | B | AG | D | C7 | Intake stop gate accumulator reservoir tank ruptured. |
| 68-05 | | 3/4/68 | 4/68 | B | RB | J | | C | AL | D | C7 | Control rod subgroup slipped into the reactor. |
| 68-06 | 31259 | 3/9/68 | 4/68 | B | RB | J | | B | BD | G | N | Miswired control rod drive. |
| 68-07 | 24817 | 3/12/68 | 4/8/68 | B | EB | G | | B | BY | E | S2,S3 | Fire in cable trays. 5 month shutdown. |
| 68-08 | | 8/68 | 9/68 | D | SF-B | OO | E | C | AZ | D | N | Faulty flow comparator failed to close LPCI valve. |
| 68-09 | | 9/7/68 | 10/68 | D | RB | J | | C | AS | D | C7 | Two control rods dropped into the core during tests. |
| 68-10 | 42736 | 9/9/68 | 12/68 | B | RB | J | P | B | AQ | D | N | 5 control rods slipped causing scram. |
| 68-11 | 42736 | 9/22/68 | 12/68 | D | CB | DD | | B | CA | D | N | During shutdown coolant pump rotated backwards. |

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 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 68-12 | 42736 | 9/26/68 | 12/68 | B | RB | J | | B | AS | D | C7 | Control rod group dropped into reactor due to clutch failure. |
| 68-13 | | 10/1/68 | 11/68 | B | RB | J | | B | AS | D | C7 | Control rod subgroup 8 dropped. |
| 68-14 | 29676 | 10/14/68 | 11/25/68 | B | PC | Z | | B | HC | B | S1,S2 | No flow in boron transfer pumps due to boron crystallization. |

Table A2.4 Coding Sheet for Reportable Events at San Onofre 1 - 1969

| Number* | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 69-01 | 34905 | 1/7/69 | 2/69 | D | CA | OO,CC | | A | AU | D | N | Leaking pressurizer safety valve. |
| 69-02 | 34905 | 1/9/69 | 2/69 | D | HA | NN | | B | AU | D | N | Oil leak in turbine auto stop valve. |
| 69-03 | | 3/5/69 | 4/69 | B | HF | P | | B | AL | D | N | Traveling screen shear pin failure. |
| 69-04 | | 3/5/69 | 4/69 | B | ED | S | | B | EE | D | C7 | Inverter failure on No. 1 vital bus. |
| 69-05 | 38753 | | 7/3/69 | B | RB | J | K | B | EG | D | N | Excessive turbine load runback due to spurious signal from control rod position indicator. |
| 69-06 | 38751 | | 7/22/69 | B | SD | OO | | C | AX | G | N | Leak in two containment isolation valves. |
| 69-07 | 38750 | | 7/22/69 | B | SF-B | | C | C | EG | D | N | Partial failure of flow comparator of safety injection system. |
| 69-08 | 39060 | 7/15/69 | 10/27/69 | B | PC | DD | | B | BV | B | S9 | Boric acid injection pump plugged due to crystallization of boric acid. |

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 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 69-09 | | 8/13/69 | 9/69 | B | IA | | U | B | EF | D | N | RTD failure caused unit trip. |
| 69-10 | | 8/13/69 | 9/69 | B | HH | OO | | B | AY | D | N | FW control valve failed open due to air line failure. |
| 69-11 | | 8/14/69 | 9/69 | B | RB | J | P | B | EF | D | N | Control rod dropped due to intermittent relay operation. |
| A069-06 | | 10/3/69 | 11/69 | B | BB | AA | N | B | EG | D | N | Stack gas monitor power supply failed. |
| A069-07 | | 10/6/69 | 11/69 | B | AA | E,MM | | B | HB | D | N | Control rod shroud cooling fan belt failure was caused by entanglement with nearby tubing. |
| A069-08 | 39059 | 10/9/69 | 10/24/69 | B | HF | OO | | B | AD | B | C7 | Tsumani gate broke from attachment plate. |
| A069-09 | | 10/14/69 | 11/69 | B | SF-B | QQ | | C | EE | D | N | Safety injection recirc valve motor winding failed. |
| A069-10 | | 10/22/69 | 11/69 | B | RB | | K | B | EF | D | C7 | Grounded LVDT caused erratic CR position indication. |

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 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| A069-11 | | 10/27/69 | 11/69 | B | ZZ | | | | OF | I | N | A felt at San Onofre. |
| A069-12 | | 10/27/69 | 11/69 | B | PC | OO | | B | AG | D | N | Vapor seal head tank fill valve failed open. |
| A069-13 | 42745 | 10/28/69 | 2/12/70 | B | CA | OO | | B | AU | C | N | Leaking root valves of pressurizer instrument column. |
| A069-14 | | 10/29/69 | 11/69 | B | HH | QQ | | B | AD | D | N | Feedwater pump discharge valve motor operator fell off. |
| A069-15 | | 11/20/69 | 12/69 | B | ED | S | | B | EE | D | C7 | No. 3 inverter failed due to blown fuses. |
| A069-16 | | 11/29/69 | 12/69 | B | RB | | K | B | EG | D | C7 | Erroneous rod bottom indication given by failed bistable. |
| A069-17 | | 11/29/69 | 12/69 | B | RB | | K | B | EG | D | C7 | Erroneous rod bottom indication given by failed bistable. |
| A069-18 | | 12/10/69 | 1/70 | B | IA | S | I | B | EE | D | N | Pressurizer level indicator failed upscale due to blown fuse. |
| A069-19 | | 12/18/69 | 1/70 | B | ED | AA | | B | EE | D | C7 | Inverter for vital bus #2 failed. |

*AO - Abnormal Occurrence.
Other events numbered sequentially.

Table A2.5 Coding Sheet for Reportable Events at San Onofre 1 - 1970

| Number* | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| A070-01 | | 1/7/70 | 2/70 | B | PC | DD | | B | HB | D | N | North charging pump shaft broke. |
| A070-02 | | 2/16/70 | 3/70 | B | RB | | K | B | EG | D | C7 | Failed LVDT caused erroneous CR position indication. |
| A070-03 | | 2/17/70 | 3/70 | B | RB | | K | B | EG | D | C7 | Failed LVDT caused erroneous CR position indication. |
| A070-04 | | 2/5/70 | 3/70 | B | HB | MM | | B | AU | D | C7 | Steam generator tube leak. |
| A070-05 | | 4/8/70 | 4/70 | B | RB | | K | B | EG | D | C7 | Failed LVDT caused erroneous CR position indication. |
| A070-06 | | 4/13/70 | 4/70 | B | HB | QQ | | C | AZ | D | N | Turbine control valves suddenly and spuriously changed positions. |
| A070-07 | 3200 | 5/29/70 | 7/10/70 | B | HA | NN | | B | BF | H | N | Turbine manually tripped due to over acceleration. |
| A070-08 | | 5/29/70 | 6/70 | B | RB | | K | B | EG | D | C7 | Failed LVDT caused erroneous CR position indications. |
| A070-09 | | 7/27/70 | 8/70 | B | IA | | L | B | EG | D | N | A momentary spike occurred on flux channel. |
| A070-10 | | 8/24/70 | 9/70 | B | SF-C | F | | C | BF | D | N | West safety injection recirculation pump breaker spuriously tripped. |

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 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| A070-11 | | 9/12/70 | 10/70 | B | ZZ | | | | OF | I | N | A moderate earthquake was felt at San Onofre. |
| A070-12 | | 9/10/70 | 10/70 | B | PC | DD | | B | HB | D | N | North charging pump shaft cracked. |
| A070-13 | 58006 | 10/19/70 | 11/16/70 | B | SF-C | OO | | C | BB | D | N | A safety injection system isolation valve stuck open. |
| A070-14 | 58006 | 11/4/70 | 11/16/70 | B | SF-C | OO | | C | AD,ED | D | N | Damaged safety injection system pump discharge valve. |
| A070-15 | | 11/21/70 | 12/70 | B | RB | | K | B | EG | D | C7 | Failed LVDT caused erroneous CR position indication. |
| A070-16 | | 11/27/70 | 12/70 | B | HH | | U | B | EG | D | N | Failed thermocouple on on the east FW pump give an erroneous signal. |
| A070-17 | | 12/3/70 | 1/71 | B | ZZ | | | | | D | N | Contractors leased van damaged. |
| 70-01 | 60821 | | 12/15/70 | D | CB | DD | | C | AV | D | N | Cracked reactor coolant pump flywheel. |

*AO - Abnormal Occurrence.
Other events numbered sequentially.

Table A2.6 Coding Sheet for Reportable Events at San Onofre 1 - 1971

| Number* | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| A071-01 | | 2/9/71 | 2/71 | B | ZZ | | | | OF | I | N | A moderate earthquake was felt at San Onofre. |
| A071-02 | | 3/18/71 | 3/71 | B | IA | | M | B | EF | C | N | Cold soldered joints in a low pressure trip circuit caused scram. |
| A071-03 | | 3/23/71 | 3/71 | B | ED | S | | B | BG | D | C7 | No. 4 inverter failed due to capacitor failure. |
| A071-04 | | 4/15/71 | 4/71 | B | IA | | M | B | EH | D | N | A failed capacitor in a low pressure trip circuit caused a drift in the trip set point. |
| A071-05 | | 4/27/71 | 4/71 | B | CB | DD | U | B | EG | D | N | A RCP thermocouple gave an erroneous indication due to static charge of unknown origin. |
| A071-06 | | 6/22/71 | 6/71 | B | EA,HA | | | B | BG | D | S9 | Loss of two offsite power lines caused turbine overspeed. |
| A071-07 | | 6/26/71 | 6/71 | B | RB | J,S | | B | EE | D | N | Fuse failure caused partial immobility of two CRDs. |

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 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| A071-08 | | 7/7/71 | 7/71 | B | HA | QQ | | C | ED | D | N | Turbine load limit valve motor failed during a test. |
| A071-09 | | 7/12/71 | 7/71 | B | HA | T | | A | EC | G | N | Gradual decay of generator excitation because maintenance took too long. |
| A071-10 | | 7/12/71 | 7/71 | B | HA | NN | | C | BW | D | S9 | Spurious turbine overspeeds occurred. |
| A071-11 | | 8/31/71 | 8/71 | B | ZZ | | | | OF | I | N | A small earthquake was recorded at San Onofre. |
| A071-12 | | 9/21/71 | 9/71 | B | IA | | M | B | EH | D | N | A zener diode failure in a low pressure trip circuit caused a drift in the trip set point. |
| A071-13 | | 9/23/71 | 9/71 | B | RB | J,S | | B | EE | D | N | Fuse failure caused partial immobility of two CRDs. |
| A071-14 | | 9/30/71 | 9/71 | B | ZZ | | | | OF | I | N | A small earthquake was recorded at San Onofre. |
| A071-15 | | 11/9/71 | 11/71 | B | SH-C | DD | | C | OC | A | N | Aux FWP tests were scheduled monthly instead of bi-weekly. |
| A071-16 | | 11/13/71 | 11/71 | B | IA | | M | B | EH | G | N | A bad soldered joint in a low pressure trip circuit caused a drift in the trip set point. |

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 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| A071-17 | 55346 | 12/26/71 | 2/25/72 | D | HB | 00 | | C | BA | D | N | Two steam relief valves failed to open on test. |

*AO - Abnormal Occurrence.

Table A2.7 Coding Sheet for Reportable Events at San Onofre 1 - 1972

| Number* | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| A072-01 | | 1/72 | 1/28/72 | C | EE | QQ | | C | BB | D | N | DG fuel line isolation valve failed to close. |
| A072-02 | 69325 | 1/72 | 1/28/72 | C | SD | OO,QQ | | C | AG,AH | D | N | An isolation valve failed to close. |
| A072-03 | 55249 | 1/72 | 2/11/72 | C | RC | R | | C | BX | H | N | Cladding failure in two fuel rods. |
| A072-04 | | 2/12/72 | 2/72 | C | PC | QQ | | C | ED | D | N | Seal water filter bypass valve failed open during a test due to a shorted valve operator. |
| A072-05 | | 2/2/72 | 2/72 | C | CB | MM | | B | AU | D | C7 | Steam generator tube leaked. |
| A072-06 | | 4/7/72 | 4/72 | B | HA | QQ | | C | EF | D | N | Turbine control valve spuriously changed positions. |
| A072-07 | 71397 | 4/30/72 | 5/9/72 | B | HH | QQ | | B | AB | D | S9 | Excessive feedwater flow due to regulating valve failure. |
| A072-08 | 84464 | 5/2/72 | 5/72 | B | HA | OO,QQ | P | B | AG,AZ | D | N | Two turbine control valves in main steam system failed. |
| A072-09 | | 6/5/72 | 6/72 | B | IA | L | | B | OJ | H | N | Operator bumped into a flux monitor causing a unit runback. |

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 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 72-01 | 78514 | 7/8/72 | 2/15/73 | B | CB | II | | B | AU | D | C7 | Steam generator C tubes leaked. |
| 72-02 | 78514 | 7/18/72 | 2/15/73 | B | EE | N | | C | EE | D | N | Diesel generator had broken wire in exciter field. |
| 72-03 | 74190 | 7/29/72 | 8/28/72 | B | HA | NN,T | | B | BF,EE | D | S9 | Turbine overspeed when unit tripped due to loss of generator field. |
| 72-04 | 78514 | 9/12/72 | 2/15/73 | B | CB | II | | B | AU | D | C7 | Steam generator A tubes leaked. |
| 72-05 | 75547 | 9/20/72 | 9/29/72 | B | HH | DD,D | A | B | AB | D | N | High temperature alarm from feedwater pump due to bearing problem. |
| 72-06 | 78514 | 9/20/72 | 2/15/73 | D | RB | J | | C | AS | D | C7 | Four control rods slipped. |
| 72-07 | 78514 | 12/9/72 | 2/15/73 | B | ED | AA | | B | EE | D | C7 | Inverter on vital bus No. 3 failed. |

*AO - Abnormal Occurrence.
Other events numbered sequentially.

Table A2.8 Coding Sheet for Reportable Events at San Onofre 1 - 1973

| Number* | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| SI73-01 | | 1/5/73 | 8/28/73 | B | CB | II | | B | AU | D | C7 | Steam generator A tubes leaked. |
| SI73-02 | | 1/10/73 | 8/28/73 | B | ED | AA | T | B | EE | H | C7 | Operator inadvertently bumped a switch de-energizing the inverter to vital bus No. 4. |
| SI73-03 | 79373 | 2/13/73 | 3/8/73 | B | EE | N | U | C | HB,BL | D | N | Overheating of diesel generator No. 2 due to stuck cooling water thermostats. |
| SI73-04 | 79468 | 2/16/73 | 3/12/73 | B | EE | N | | C | AD | D | N | Failure of fuel injection pump for diesel generator No. 1. |
| SI73-05 | | 2/20/73 | 8/28/73 | B | RB | J | T | B | AY | D | C7 | A switch spuriously opened in the CRD positioner and a CR slipped into the core. |
| SI73-06 | | 2/21/73 | 3/8/73 | B | ZZ | | | | OF | I | N | A moderate earthquake was felt at San Onofre. |
| SI73-07 | | 5/8/73 | 8/28/73 | B | EE | P | | B | AQ | D | N | Diesel generator failed to start because of a clogged filter in the air start mechanism. |

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 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| SI73-08 | 81591 | 6/7/73 | 7/6/73 | D | EA,EE | LL,N | | B | ED | D | S1,S2 | Loss of offsite power due to transformer trip. Also one diesel generator failed. |
| SI73-09 | | 7/14/73 | 2/25/74 | D | ZZ | | | | OF | I | N | A small earthquake was sensed at San Onofre. |
| SI73-10 | | 8/6/73 | 2/25/74 | B | ZZ | | | | OF | I | N | A small earthquake was sensed at San Onofre. |
| SI73-11 | | 8/17/73 | 2/25/74 | B | ED | AA | | B | EE | D | C7 | Inverter No. 2 failed. |
| SI73-12 | 87066 | 10/16/73 | 11/20/73 | B | EE | N | | C | AB | E | N | Failure of fuel pump for diesel generator. |
| SI73-13 | 87016 | 10/21/73 | 10/22/73 | B | HH,SF-C | OO,QQ | | B | HA,HH | H | S9 | During reduction in power, unit tripped due to excessive feed-water flow. Safety injection system was actuated. Motor operator of safety injection valve damaged by water hammer. Reason for shutdown was turbine failure. |
| SI73-14 | | 10/28/73 | 2/25/74 | D | ZZ | | | | OF | I | N | A small earthquake was sensed at San Onofre. |

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 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| SI73-15 | | 11/29/73 | 2/25/74 | D | EA | F | | B | EE | D | N | Electrical disturbance in offsite power grid. |
| SI73-16 | | 12/18/73 | 1/14/74 | D | SD | OO | | C | AW | D | N | Containment isolation valve failed leaky. |

*SI - Station Incident.

Table A2.9 Coding Sheet for Reportable Events at San Onofre 1 - 1974

| Number* | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| SI74-01 | | 1/14/74 | 2/8/74 | D | HB | Z,KK | | B | AD | D | N | Main steam line knee support found misplaced. |
| SI74-02 | | 2/26/74 | 8/14/74 | B | IA | | M | B | EH | D | N | Loop "B" low pressure trip set point high. |
| SI74-03 | | 4/3/74 | 8/14/74 | B | PC | OO | | B | HL | D | N | Boron crystallization prevented operation of BAT discharge valve. |
| SI74-04 | | 6/11/74 | 8/14/74 | B | IA | | L | B | EG | D | N | Spurious high flux indication caused turbine runback. |
| SI74-05 | | 6/14/74 | 8/14/74 | B | ED | AA | P | B | ED | D | C7 | No. 1 inverter failed due to shorted SCR. |
| SI74-06 | 94768 | 7/7/74 | 7/15/74 | D | WB | U | L | B | AU | B | SZ | Leaking cooling coils flooded detector thimbles, causing a reactor trip. |
| SI74-07 | | 7/9/74 | 2/12/75 | B | HH | PP | | B | AL | D | N | FW check valve to steam generator "C" failed. |
| SI74-08 | | 4/25/74 | 8/14/74 | B | CB | II | | B | AU | D | C7 | Tube leak in steam generators "A" and "C". |
| SI74-09 | | 7/24/74 | 2/12/75 | B | ZZ | | | | | Z | N | FBI reported a sabotage threat. |
| SI74-10 | 95378 | 8/13/74 | 9/30/74 | B | EE | N,DD | | C | AG | D | N | Diesel geneator cooling pump seized. |

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 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| SI74-11 | | 8/20/74 | 2/12/75 | B | IA | | I | B | EG | D | N | Spurious trip from pressurizer level indicator. |
| SI74-12 | | 10/21/74 | 2/12/75 | B | RB | J | | B | AS | D | C7 | Subgroup 8 control rods dropped into the core. |

*SI - Station Incident.

Table A2.10 Coding Sheet for Reportable Events at San Onofre 1 - 1975

| Number* | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| SI75-01 | | 1/13/75 | 8/20/75 | B | IA | | C,M | B | EH | D | N | Low pressure trip set-point drift. |
| SI75-02 | 93568 | 1/14/75 | 1/22/75 | B | EE | N | | C | AO | D | N | Diesel generator failed due to separation of pulley from drive shaft. |
| SI75-03 | | 2/12/75 | 8/20/75 | B | HF | | U | B | EG | D | N | Circulating water ΔT monitor failed. |
| SI75-04 | 100574 | 2/12/75 | 3/12/75 | B | EE | N,DD | | C | HB | B | N | Diesel generator failed due to insufficient flow of fuel transfer pump. |
| SI75-05 | | 2/19/75 | 8/20/75 | B | ED | AA | | B | BG | D | C7 | No. 2 inverter failed tripping the reactor. |
| SI75-06 | | 2/28/75 | 3/31/75 | B | SF-B | OO | | B | OK | B | S9 | Potential SI failure due to single valve failure. |
| SI75-07 | | 3/15/75 | 6/16/75 | C | SD | OO | | C | BB | D | N | Leakage of containment isolation valve. |
| SI75-08 | | 4/2/75 | 8/20/75 | C | FD | L | | B | AL | D | N | Fuel transfer system was damaged. |
| SI75-09 | | 4/22/75 | 8/20/75 | B | ED | AA | | B | BG | D | C7 | No. 2 inverter failed tripping the reactor. |
| SI75-10 | | 4/25/75<br>5/2/75 | 8/25/75 | B | IA | | L | B | EH | D | N | Overpower trip set point drifted. |
| SI75-11 | | 4/13/75 | 8/25/75 | C | CB | II | | C | AU | D | C7 | Steam generator "A" tubes leaked. |

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 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| SI75-12 | | 4/18/75 | 8/25/75 | B | EA | F | | C | BG | H | N | Operator deenergized two 4 kV buses during a trip. |
| SI75-13 | | 5/21/75 | 8/25/75 | B | HF | P | | B | AQ | I | N | Heavy influx of sea-grass clogged the traveling screens. |
| SI75-14 | 104065 | 6/19/75 | 7/7/75 | B | BA,BB | | N | B | OC | H | N | Air and drinking water samples not taken. |
| SI75-15 | | 6/11/75 | 8/25/75 | B | CB | II | | B | AU | D | C7 | Steam generator "C" tube leaked. |
| SI75-16 | | 7/9/75 | | B | HF | | T | B | BB | D | N | Chlorination micro-switch failed to discontinue chlorination. |
| SI75-17 | 106454 | 8/12/75 | 9/12/75 | B | EE | N,U | | B | AR | D | S1 | Diesel generator #1 overheats due to blocked cooling. |
| SI75-18 | 106455 | 8/13/75 | 9/12/75 | B | EE | N,FF | | B | AW | B | S1 | Diesel generator #2 overheats due to loss of cooling water. |
| SI75-19 | | 8/20/75 | | B | PC | DD | | C | AG | D | N | South boric acid transfer pump bearing seized. |
| SI75-20 | | 8/20/75 | | B | RB | J | C | B | BC | D | C7 | Control rod group 3 skipped steps. |
| SI75-22 | | 10/19/75 | | B | ED | AA | C | B | BY | D | C7 | Oil-filled capacitor in the No. 3 inverter failed. |

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 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| SI75-23 | | 11/2/75 | | B | HH | Z | | B | AL | D | N | FW flow straightening device became loose. |
| SI75-24 | 108558 | 11/11/75 | 11/25/75 | B | BB | | N | B | OC | H | N | Offsite particulate filters not collected for 2 weeks. |

*SI - Station Incident.

Table A2.11 Coding Sheet for Reportable Events at San Onofre 1 - 1976

| Number* | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER76-01 | 114077 | 3/22/76 | 3/23/76 | B | FB | O | | B | AO | C | N | Crack in shipping cask lifting device. |
| LER76-02 | | 5/3/76 | 2/25/77 | B | FB | O | | B | AO | C | N | Crack in shipping cask lifting device. |
| LER76-03 | 115452 | 5/20/76 | 6/18/76 | B | HB | II,OO | E | B | AU | D | N | Leak in instrument valve caused low SG flow indication. |
| LER76-04 | 115878 | 6/28/76 | 7/14/76 | B | CB | | E | B | EG | D | N | Loop "C" flow transmitter failure caused spurious trip. |
| LER76-05 | | 6/1/76 | 7/26/76 | B | ID | AA | L | B | ED | D | N | Axial monitoring system failed due to shorted power supply. |
| LER76-06 | 117666 | 7/30/76 | 8/25/76 | B | CB | II | | B | AU | D | C7 | Steam generator tube failed. |
| LER76-07 | 117154 | 7/31/76 | 8/25/76 | D | PC | DD | | C | HC | B | C4 | Boric acid transfer line was blocked by boron crystallization. |
| LER76-08 | 118184 | 8/23/76 | 9/16/76 | B | ED | AA | | B | EE | D | C7 | Inverter No. 1 failed causing loss of feedwater control. |
| LER76-09 | 119784 | | 11/16/76 | C | RA | KK | | C | AI,HD | B | N | Thermal shield flexure supports failed. |
| LER76-10 | 119783 | | 11/15/76 | C | CB | II | | C | AU | D | C7 | SG tubes had defects. |

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 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER76-11 | 120269 | | 12/1/76 | C | CB | II,KK | | C | AB | D | C7 | SG antivibration bars were worn. |
| SI76-01 | | 1/1/76 | 2/25/77 | B | ZZ | | | | OF | I | N | A small earthquake was sensed at San Onofre. |
| SI76-02 | | 1/21/76 | 2/25/77 | B | EA | G | | B | BG | I | S1 | Loss of offsite power due to brush fire. |
| SI76-03 | | 2/9/76 | 2/25/77 | B | IB | | I | C | EG | D | N | Spurious pressurizer high level signal tripped reactor. |
| SI76-04 | | 2/9/76 | 2/25/77 | B | ED | AA | C | B | BY | D | C7 | Oil-filled capacitor in No. 2 inverter failed. |
| SI76-05 | | 2/17/76 | 2/25/77 | B | ED | AA | | B | BG | D | C7 | No. 4 inverter failed tripping the reactor. |
| SI76-06 | | 3/5/76 | 2/25/77 | B | FB | JJ | | B | OD | H | N | Contamination was spread from spent fuel cask area. |
| SI76-07 | | 3/27/76 | 2/25/77 | B | FB | JJ | | B | OC | H | N | Spent fuel cask was shipped without knowledge of amount inside. |
| SI76-09 | | 4/17/76 | 2/25/77 | B | HA | | C | B | BF | G | N | Turbine tripped due to low overspeed setting. |
| SI76-11 | | 4/17/76 | 2/25/77 | B | HH | D,DD | | B | BL | D | N | FW pump vibration caused high bearing temperature. |

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 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| SI76-12 | | 4/19/76 | 2/25/77 | B | HH | PP | | B | AL | D | N | FW check valve seat became disconnected. |
| SI76-15 | | 7/2/76 | 2/25/77 | B | FB | JJ | | B | AX | D | N | Shipping cask neutron shield leaked borated glycol. |
| SI76-16 | | 7/26/76 | 7/25/77 | B | EB | G | | B | BY | E | N | Brief construction fire scorched electric cable jackets. |
| SI76-21 | | 8/23/76 | 2/25/77 | B | ID | AA | K | B | EG | D | C7 | Rod bottom indicator power supply failed causing load runback. |
| SI76-23 | | 9/25/76 | 2/25/77 | B | ED | AA | C | B | BY | D | C7 | Oil-filled capacitor in No. 3 inverter failed. |
| SI76-24 | | 9/25/76 | 2/25/77 | B | ED | AA | | B | BG | D | C7 | No. 2 inverter failed. |
| SI76-25 | | 9/25/76 | 2/25/77 | B | ED | AA | C | B | BY | D | C7 | Oil-filled capacitor in No. 3 inverter failed. |
| SI76-26 | | 10/18/76 | 2/25/77 | B | ZZ | | | | OF | I | N | A moderate earthquake was felt at San Onofre. |
| 76-1 | 111601 | | 3/1/76 | B | IF | | U | B | EG | D | N | Offshore temperature sensors failed. |

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 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 76-2 | 115877 | | 6/28/76 | B | EE | KK | | B | AV | E | N | Voids found in diesel generator building wall. |
| 76-3 | 115878 | | 7/15/76 | B | IF | | U | C | EG | I | N | Offshore temperature sensors failed. |

*LER - Licensee Event Report.
SI - Station Incident. Numbers omitted are which were identified as LERs and SIs.
Other events numbered sequentially.

Table A2.12 Coding Sheet for Reportable Events at San Onofre 1 - 1977

| Number* | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER77-01 | 122203 | 1/17/77 | 2/4/77 | D | BB | | | B | OC | A | N | Airborne samples not collected. |
| LER77-02 | 122204 | 1/17/77 | 2/7/77 | D | SH-B | Z | | C | AR | C | N | Corrosion in containment spray piping. |
| LER77-03 | 122132 | 1/24/77 | 2/18/77 | D | WB | Z,DD | | A | AU | A | N | Leak in RHR pump cooling water line due to line being cut into. |
| LER77-04 | 122133 | 2/15/77 | 3/2/77 | D | SH-B | Z | | C | AO | E | N | Two leaks found in containment spray system. |
| LER77-05 | 123082 | 2/20/77 | 3/4/77 | D | SA | | | C | AX | G | N | Leak in containment vessel due to drilled holes. |
| LER77-10 | 125174 | 4/17/77 | 5/13/77 | B | CB | OO | | B | AU | D | N | Reactor coolant system leak through valve packing. |
| 77-01 | 124898 | 4/21/77 | 5/11/77 | B | IF | | U | C | AL | D | N | Offshore temperature sensor found to be missing. |
| LER77-06 | 125175 | 4/26/77 | 5/25/77 | B | PC | DD | | B | HC | D | C4 | Failure of boric acid transfer pump due to boron precipitation. |
| LER77-07 | 125181 | 5/10/77 | 5/25/77 | B | EE | N,KK | | C | AL | E | N | Fuel oil bypass line supports omitted on diesel generators. |
| 77-02 | 126491 | 6/1/77 | 6/29/77 | B | IF | | U | C | AL | D | N | Offshore temperature sensor found to be missing. |

Table A2.12 (continued)

| Number* | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 77-03 | | 6/16/77 6/26/77 6/27/77 | 7/12/77 | B | FB | OO | | B | AY | D | N | Spent fuel shipping cask drain valve found open. |
| LER77-08 | 125703 | 6/14/77 | 7/1/77 | B | ED | QQ,S | | B | BB | D | C7 | Inverter deenergized by fuse failure. |
| LER77-09 | 129554 | 7/29/77 | 8/25/77 | B | CA | CC | | B | OK | B | N | Pressurizer heat-up rate nonconservative. |
| 77-04 | 127978 | 7/30/77 | 8/18/77 | C | FB | OO | | B | AY | H | N | Spent fuel shipping cask drain valve found open. |
| LER77-12 | 128944 | 8/9/77 | 9/8/77 | B | SA | FF | | B | OC | C | N | Deficiency found in electrical penetration documentation. |
| LER77-11 | 128890 | 8/24/77 | 9/7/77 | B | SF-C | DD | | B | BH | B | N | Charging pump circuitry change fails to eliminate single failure effect. |
| LER77-13 | 129778 | 9/19/77 | 10/3/77 | D | CB | II,KK | | C | AB | D | C7 | Tube wall thinning due to anti-vibrational bars. |
| LER77-14 | 130702 | 10/1/77 | 10/14/77 | D | CB | II,KK | | A | BV,AU | D | C7 | Inadvertent dilution of reactor coolant due to steam generator leak. |

*LER - Licensee Event Report.
Other events numbered sequentially.

Table A2.13 Coding Sheet for Reportable Events at San Onofre 1 - 1978

| Number* | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER78-01 | 135753 | 1/16/78 | 2/10/78 | B | HF | DD,F | P | B | BF | D | N | Saltwater cooling pump breaker trips open when other pump is down for maintenance. |
| LER78-02 | 136374 | 1/30/78 | 2/23/78 | B | SH-B | OO | | C | BB | D | N | Containment spray valve fails to close in testing. |
| LER78-03 | 137341 | 3/15/78 | 3/28/78 | B | SH-B | | C | B | BF | D | N | Two containment spray actuation system inverters tripped. |
| LER78-04 | 138234 | 3/28/78 | 4/25/78 | B | EE | N | | C | AG | D | N | Diesel generator fails to start due to lack of lubrication. |
| LER78-05 | 138253 | 4/18/78 | 5/10/78 | D | SF-C | OO | | C | AE,AH | D | N | Safety injection valve opens slowly due to distorted valve shaft. |
| LER78-06 | 138254 | 4/26/78 | 5/10/78 | B | SF-C | | | B | HB | B | N | Reanalysis of small break LOCA indicated lower than anticipated flow. |
| LER78-07 | 139587 | 6/7/78 | 6/29/78 | B | ED | S | C | B | BF | D | C7 | Input fuse to inverter fails. |
| LER78-08 | 140322 | 7/18/78 | 8/9/78 | B | EE | N | | C | AG | D | N | Diesel generator fails to start due to binding of linkage. |
| LER78-09 | 140321 | 8/1/78 | 8/9/78 | B | HH | | E | B | AL | D | N | Flow straightener became dislodged and lodged against feedwater flow orifice plate. |

Table A2.13 (continued)

| Number* | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER78-10 | 140353 | 8/18/78 | 9/14/78 | B | HH,SF-C | OO | | B | BH | G | N | Feedwater pump safety injection valve inadvertently opened. |
| LER78-11 | 141389 | 10/23/78 | 11/6/78 | C | HH | GG | | C | HH,AH | D | N | Shock absorbers found inoperable. |
| LER78-12 | 141196 | 10/23/78 | 11/21/78 | C | SF-C | Z | | C | AR | D | N | Leaks found in charging pump discharge line. |
| LER78-13 | 142839 | 11/25/78 | 12/20/78 | B | HH | | E | B | AL | D | N | Flow straightener dislodged. |
| LER78-14 | 145212 | 12/5/78 | 1/3/79 | B | SH-B | OO | | A | AY | H | N | Hydrazine additive pump recirc valve left open. |
| 78-01 | | 9/13/78 | 10/13/78 | B | HF | P | | B | AQ | I | N | Large quantity of fish impinged on intake screens. |
| 78-02 | | 9/18/78 | 10/18/78 | | MC | P | | B | AL | I | N | Settling plates damaged by increased aquatic activity. |
| 78-03 | | 6/30/78 | 10/13/78 | B | MC | | U | B | EG | D | N | Two temperature sensors failed. |
| 78-04 | | 10/10/78 | 12/19/78 | B | MC | | U | B | EG | D | N | Two temperature sensors failed. |

*LER - Licensee Event Report.
Other events numbered sequentially.

Table A2.14 Coding Sheet for Reportable Events at San Onofre 1 - 1979

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER79-01 | 147264 | 1/9/79 | 2/7/79 | B | SD | QQ | | C | BB | D | N | Containment air unit cooling water valve failed to close. |
| LER79-02 | 150075 | 4/4/79 | 5/4/79 | B | HH | | E | B | AL | D | N | Flow straightener dislodged. |
| LER79-03 | 149445 | 4/4/79 | 5/3/79 | B | SD | OO | | C | BB | D | N | Containment air unit cooling water valve failed to close. |
| LER79-04 | 151481 | 1/9/79 | 4/19/79 | B | BB | | N | B | OK | A | N | Loss of benthic survey data not reported. |
| LER79-05 | 151480 | 4/2/79 | 4/27/79 | B | IF | | U | B | AG | D | N | Sea temperature data lost due to instrument malfunction. |
| LER79-06 | 149247 | 4/13/79 | 5/11/79 | B | BB | | N | B | AL | D | N | Four environmental radiation dosimeters lost due to vandalism. |
| LER79-07 | 150036 | 4/23/79 | 5/22/79 | B | HB | | E | B | EG | D | N | Steam flow indicator failed. |
| LER79-08 | 150253 | 6/4/79 | 6/18/79 | D | CB | II | | C | AU | D | C7 | Leaking steam generator tubes. |
| LER79-09 | 150252 | 6/5/79 | 6/19/79 | D | HH | GG | | C | HH,AH | D | N | Feedwater line shock absorber fails. |
| LER79-10 | 152656 | 6/5/79 | 6/15/79 | D | HB | II | | C | AO | E | N | Cracks in steam generator feedwater nozzle to reducer welds. |

Table A2.14 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER79-11 | 151831 | 8/6/79 | 8/31/79 | B | BB | | N | B | OC | A | N | Drinking water sample lost. |
| LER79-12 | 151834 | 8/5/79 | 8/30/79 | B | HF | | | B | OC | A | N | Fish impingement data not collected. |
| LER79-13 | 152654 | 9/6/79 | 9/11/79 | B | SH-B | Z | | C | AO | D | N | Cracks in suction piping for containment spray pumps. |
| LER79-14 | 152206 | 8/29/79 | 9/28/79 | B | EC | | R | C | EE | D | N | 15 VDC power supply failed disabling one diesel generator and one safety injection train. |
| LER79-15 | 154812 | 9/25/79 | 10/9/79 | B | SD | FF | | A | AH | H | N | Containment personnel air lock left open for 10 seconds. |
| LER79-16 | 154815 | 9/15/79 | 10/5/79 | D | SF-C | Z | | C | AO,AV | D | N | Crack-like indication in safety injection line weld. |
| LER79-17 | 153395 | 11/7/79 | 11/21/79 | B | EB | | T | B | BF,ED | D | C7 | Short circuit in 480 V bus due to rodent. |
| LER79-18 | 153387 | 11/9/79 | 11/26/79 | D | MA | KK,FF | | C | AL | E | N | Pipe support found missing. |
| LER79-19 | 153668 | 11/16/79 | 11/30/79 | B | EE | KK | | C | AL | E | N | Pipe supports found missing. |

Table A2.14 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER79-20 | 153669 | 11/26/79 | 12/7/79 | D | WB | KK | | C | AL | E | N | Pipe supports found missing. |
| LER79-21 | 153670 | 11/14/79 | 12/12/79 | B | EC | | P | B | BF | D | N | Load sequencer power supply trips off. |
| LER79-22 | 153819 | 11/30/79 | 12/17/79 | D | WB | KK | | C | AL | E | N | Pipe supports found missing. |
| LER79-23 | 153873 | 10/10/79 | 12/11/79 | B | CB,IA | CC | M | C | EI | H | N | Low pressure trip safety setting too low on one pressurizer pressure channel. |
| LER79-24 | 153936 | 12/3/79 | 1/31/80 | B | HH | | E,I | B | OK | H | N | Modifications made to equipment w/o approved design change. |
| LER79-25 | 153496 | 10/11/79 | 1/11/80 | B | CB,IA | | F,U | B | EG | D | N | T-AVG converter fails. |
| LER79-26 | 153937 | 12/26/79 | 1/10/80 | B | SH-C | DD,D | | C | AB,AK | H | N | Steam driven aux. feed-water pump bearing fails. |

Table A2.15 Coding Sheet for Reportable Events at San Onofre 1 - 1980

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER80-01 | 154456 | 1/10/80 | 1/25/80 | B | HF | Z,KK | | C | BC | E | N | Piping supports fail to meet seismic criteria. |
| LER80-02 | 154455 | 1/16/80 | 1/28/80 | B | HH | OO | P | B | BF | G | S9 | Construction worker bumped a relay tripping a FW pump. |
| LER80-03 | 154623 | 1/9/80 | 2/5/80 | B | SD,PA | QQ,HH | | B | BB | D | C8 | Service water containment isolation valve fails to close due to clogged air line. |
| LER80-04 | 155198 | 1/19/80 | 2/13/80 | B | HH | | E | B | AL | D | N | Flow straightener dislodged. |
| LER80-05 | 155571 | 2/18/80 | 2/29/80 | B | HF | H | | B | HB | I | N | Condenser cooling water flow partially blocked by seaweed. |
| LER80-06 | 155475 | 3/10/80 | 3/24/80 | B | HF | DD | | B | HC | D | S1,S3 | All salt water cooling pumps failed. |
| LER80-07 | 155983 | 2/24/80 | 3/20/80 | B | SF-C | Z | | B | HE | D | N | Crack found in charging pump discharge line, due to cavitation. |
| LER80-08 | 156134 | 3/18/80 | 4/9/80 | B | HF | KK | | A | HD,AR | D | N | Salt water cooling system pipe supports fail. |
| LER80-09 | 155441 | 3/31/80 | 4/2/80 | B | BB | | N | B | OC | I | N | Benthic data not taken when required. |

Table A2.15 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER80-10 | 156333 | 3/18/80 | 4/9/80 | B | SF-C | DD,D | | B | AB | D | N | Charging pump thrust bearing fails. |
| LER80-11 | 156334 | 3/10/80 | 4/8/80 | B | PC | CC | | B | AD | H | N | Pressurizer relief tank rupture disk ruptures due to overfilling. |
| LER80-12 | 156205 | 3/24/80 | 4/21/80 | B | EE | N,DD | | B | BF | D | N | Diesel generator fuel oil transfer pumps fail due to water flooding due to failure of sump pumps. |
| LER80-13 | 157052 | 3/29/80 | 4/28/80 | B | SG | S | | B | AM | G | N | Control room emergency air treatment system dampers fail closed due to wrong fuse. |
| LER80-14 | 166740 | 4/14/80 | 7/22/80 | C | CB | II | | C | AU | D | C7 | 9 steam generator tubes found to be leaking. |
| LER80-15 | 158278 | 4/22/80 | 5/5/80 | D | EA | | P | A | BF | G | S7 | 4 kV and 480 V onsite and offsite power lost. |
| LER80-16 | 156982 | 4/20/80 | 5/20/80 | C | IE | FF | S | B | EG | D | S1,S2 | Source range nuclear instrumentation fails during refueling due to seal leakage. |

Table A2.15 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER80-17 | 156956 | 5/5/80 | 5/14/80 | C | FD | PP | | B | AU | D | N | Erroneous boron results from reactor refueling cavity due to contamination of chemistry sample. |
| LER80-18 | 156925 | 5/6/80 | 5/16/80 | C | IE | | F | A | AE | B | N | Incore flux thimble bent. |
| LER80-19 | 158666 | 5/10/80 | 5/22/80 | D | CF | KK | | C | AL | E | N | Pipe support found to be missing. |
| LER80-20 | 158570 | 5/13/80 | 5/28/80 | D | HH,HB | GG | | C | AH | H | N | Failed hydraulic snubbers found. |
| LER80-21 | 158571 | 5/15/80 | 5/29/80 | D | HH | KK | | C | HD,HH | D | N | Feedwater pipe supports found to have failed. |
| LER80-22 | 156924 | 5/7/80 | 5/20/80 | D | CB | Z | | C | AV | C | N | Pinhole found in reactor coolant pump seal injection pipe. |
| LER80-23 | 157678 | 5/23/80 | 6/18/80 | D | WB | U | | A | AO | G | N | Flaw found in component cooling water heat exchanger weld. |
| LER80-24 | 158572 | 5/16/80 | 5/29/80 | D | HB | Z | | C | AO | E | N | Indications found in main steam piping welds during radiographic examination. |

Table A2.15 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER80-32 | 160238 | 7/17/80 | 8/18/80 | D | PA,WA | QQ | | B | BB | B | N | Containment isolation valve in service water system failed to close due to desiccant contamination of instrument air. |
| LER80-33 | 160315 | 9/2/80 | 9/16/80 | D | IB | | P | C | EG | B | N | Sequencer relays fail to reset SIS properly during test. |
| LER80-34 | 160322 | 9/1/80 | 9/17/80 | D | CB | II | | A | AU | G | C7 | Reactor coolant dilution due to steam generator leak. |
| LER80-35 | 160510 | 9/18/80 | 10/10/80 | D | ID | | T | C | ED | B | N | Containment isolation panel reset switches found defective. |
| LER80-36 | 160162 | 9/22/80 | 10/3/80 | D | CB | II | | A | AU | D | C7 | Reactor coolant dilution due to seal failure during steam generator decontamination. |
| LER80-37 | 160875 | 11/13/79 | 10/27/80 | B | IF | | U | B | AG | D | N | Ocean thermal monitoring data not taken due to motor failure. |
| LER80-38 | 161910 | 11/22/80 | 12/9/80 | D | EB | F | | B | AY | H | N | AC power to all station auxiliaries lost due to operator opening wrong breaker. |
| LER80-39 | 162578 | 10/12/80 | 12/23/80 | D | EE | N | | C | AK | B | N | Diesel generator turbocharger thrust bearings worn due to inadequate lubrication. |

Table A2.15 (continued)

| Number | NSIC Accession Number | Event Date | Report Date | Plant Status | System | Equipment | Instrument | Component Status | Abnormal Condition | Cause | Significance Category | Comment |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| LER80-32 | 160238 | 7/17/80 | 8/18/80 | D | PA,WA | QQ | | B | BB | B | N | Containment isolation valve in service water system failed to close due to desiccant contamination of instrument air. |
| LER80-33 | 160315 | 9/2/80 | 9/16/80 | D | IB | | P | C | EG | B | N | Sequencer relays fail to reset SIS properly during test. |
| LER80-34 | 160322 | 9/1/80 | 9/17/80 | D | CB | II | | A | AU | G | C7 | Reactor coolant dilution due to steam generator leak. |
| LER80-35 | 160510 | 9/18/80 | 10/10/80 | D | ID | | T | C | ED | B | N | Containment isolation panel reset switches found defective. |
| LER80-36 | 160162 | 9/22/80 | 10/3/80 | D | CB | II | | A | AU | D | C7 | Reactor coolant dilution due to seal failure during steam generator decontamination. |
| LER80-37 | 160875 | 11/13/79 | 10/27/80 | B | IF | | U | B | AG | D | N | Ocean thermal monitoring data not taken due to motor failure. |
| LER80-38 | 161910 | 11/22/80 | 12/9/80 | D | EB | F | | B | AY | H | N | AC power to all station auxiliaries lost due to operator opening wrong breaker. |
| LER80-39 | 162578 | 10/12/80 | 12/23/80 | D | EE | N | | C | AK | B | N | Diesel generator turbocharger thrust bearings worn due to inadequate lubrication. |