REVIEW OF THE OPERATING EXPERIENCE HISTORY OF THE GINNA NUCLEAR POWER PLANT FOR THE NUCLEAR REGULATORY COMMISSION'S SYSTEMATIC EVALUATION PROGRAM

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INTRODUCTION

The Systematic Evaluation Program Branch (SEPB) of the Nuclear Regulatory Commission (NRC) has the responsibility for conduct of the Systematic Evaluation Program (SEP) in an effort to determine the safety margins of the design and operation of the eleven oldest operating commercial nuclear power plants in the United States. These eleven plants are being reevaluated in terms of present NRC licensing requirements and regulations. The SEP program as well is to:

- establish documentation which 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;
- provide the capability for making integrated and balanced decisions with respect to any required backfitting; and
- provide for the early identification and resolution of any potential safety deficiency.

The SEP Program is evaluating specific safety topics (called the Topic List) and is based on an integrated review of the overall ability of a plant to respond to certain design basis events (challenges), 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.

In this report the operating experience of the Ginna 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 Program. The results of this report will be used by SEPB in performing the integrated assessment of overall plant safety for this plant.

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

1. SCOPE OF REVIEW

The assessment of the operating experience review for Ginna covered the time from initial criticality through and including 1979. The review included the following aspects of operation: availability and capacity factors; review of forced shutdowns and power reductions; reportable events; events of environmental importance and radioactivity releases; an evaluation of the operating experience in total.

1.1 Availability and Capacity Factors

Both reactor availability and unit availability factors were compiled for all years. Starting with 1974, the unit capacity factors using the design electrical rating (DER - net MWe) and the maximum dependable capacity (MDC - net MWe) were compiled as well. Data for the capacity factors was 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. Unit capacity (DER) =

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

4. Unit capacity (MDC) =

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

1.2 Review of Forced Shutdowns and Power Reductions

The forced shutdowns and power reductions were reviewed and data 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 Appendices).

The following data was compiled annually for the forced shutdowns and power reduction:

1. date of occurrence;
2. duration in hours;

3. noting if the shutdowns were also a reportable event, e.g., a licensee event report (LER) or abnormal occurrence report (AOR);
4. a summary description of the events associated with the shutdown or power reduction;
5. cause of the shutdown (Table 1.1);
6. method of shutdown (Table 1.1);
7. the system directly involved with the shutdown or power reduction (Table 1.2);

Table 1.1 Cause of Forced Shutdown or Power Reduction and Method of Shutdown

Cause	
Equipment failure	A
Maintenance or testing	B
Refueling	C
Regulatory restriction	D
Operator training and license exams	E
Administrative	F
Operational error	G
Other	H

Method of Shutdown	
Manual	1
Manual Scram	2
Automatic scram	3
Continuation	4
Load Reduction	5
Other	9

Table 1.2 Systems Involved With the Forced Shutdown or Power Reduction

System Description	Code
Reactor	RX
Reactor Vessel Internals	RA
Reactivity Control Systems	RB
Reactor Core	RC
Reactor Coolant System & Connected Systems	CX
Reactor Vessels & Appurtenances	CA
Coolant Recirculation Systems & Controls	CB
Main Steam Systems & Controls	CC
Main Steam Isolation Systems & Controls	CD
Reactor Core Isolation Cooling Systems & Controls	CE
Residual Heat Removal Systems & Controls	CF
Reactor Coolant Cleanup Systems & Controls	CG
Feedwater Systems & Controls	CH
Reactor Coolant Pressure Boundary Leakage Detection Systems	CI
Other Coolant Subsystems & Their Controls	CJ
Engineered Safety Features	SX
Reactor Containment Systems	SA
Containment Heat Removal Systems & Controls	SB
Containment Air Purification & Cleanup Systems & Controls	SC
Containment Isolation Systems & Controls	SD
Containment Combustible Control Systems & Controls	SE
Emergency Core Cooling Systems & Controls	SF
Control Room Habitability Systems & Controls	SG
Other Engineered Safety Feature Systems & Their Controls	SH
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 & Controls	EA
AC Onsite Power Systems & Controls	EB
DC Onsite Power Systems & Controls (Composite AC & DC)	EC
Onsite Systems & Controls (Composite AC & DC)	ED
Emergency Lighting Systems & Controls	EF
Other Electric Power Systems & Controls	EG
Fuel Storage and Handling Systems	FX
New Fuel Storage Facilities	FA
Spent Fuel Storage Facilities	FB
Spent Fuel Pool Cooling & Cleanup Systems & Controls	FC
Fuel Handling Systems	FD

System Description (Cont'd.)	Code
Auxiliary Water Systems	WX
Station Service Water Systems & Controls	WA
Cooling Systems for Reactor Auxiliaries & Controls	WB
Demineralized Water Make-up Systems & Controls	WC
Potable & Sanitary Water Systems & Controls	WD
Ultimate Heat Sink Facilities	WE
Condensate Storage Facilities	WF
Other Auxiliary Water Systems & Their Controls	WG
Auxiliary Process Systems	PX
Compressed Air Systems & Controls	PA
Process Sampling Systems	PB
Chemical, Volume Control, & Liquid Poison Systems & Controls	PC
Failed Fuel Detection Systems	PD
Other Auxiliary Process Systems & Their Controls	PE
Other Auxiliary Systems	AX
Air Conditioning, Heating, Cooling & Ventilation Systems & Controls	AA
Fire Protection Systems & Controls	AB
Communication Systems	AC
Other Auxiliary Systems & Their Controls	AD
Steam and Power Conversion Systems	HX
Turbine-Generators & Controls	HA
Main Steam Supply System & Controls (Other Than CC)	HB
Main Condenser Systems & Controls	HC
Turbine Gland Sealing Systems & Controls	HD
Turbine Bypass Systems & Controls	HE
Circulating Water Systems & Controls	HG
Condensate and Feedwater Systems & Controls (Other Than CH)	HH
Steam Generator Blowdown Systems & Controls	HI
Other Features of Steam & 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 & Effluent Radiological Monitoring Systems	MC
Solid Radioactive Waste Management Systems	MD
Radiation Protection Systems	BX
Area Monitoring Systems	BA
Airborne Radioactivity Monitoring Systems	BB

8. the component directly involved with the shutdown or power reduction (Table 1.3); and
9. categorization of the shutdown or power reduction. Each shutdown or power reduction was placed into one of two sets of categories. The shutdowns and power reductions were first evaluated against design bases events (DBE) as described in Chap. 15 of the Standard Review Plan.[1] If the shutdown or power reduction could not be categorized as a DBE initiating event, then it was placed into one of a series of NSIC categories. For further discussions of these two sets of categories, use of the categories, and a listing of them, see Sect. 3.1 and following.

The listings for the cause, shutdown method, system involved, and component involved along with their respective codes are those used in NRC's Gray Book[2] series for shutdowns. Note that the information listed under the "system involved" column in the data tables in the appendices indicates (1) a general classification of systems (fully written out) and (2) a specific system within the general classification which is coded with two letters.

1.3 Review of Reportable Events

The operating events as reported in LERs and LER predecessors, e.g., AORs, unusual event 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 time period predating LERs

Table 1.3 Components Involved With the Forced Shutdown or Power Reduction

Component Type	Component Type Includes:
Accumulators	Scram Accumulators Safety Injection Tanks Surge Tanks
Air Dryers	
Annunciator Modules	Alarms Bells Buzzers Claxons Horns Gongs Sirens
Batteries & Chargers	Chargers Dry Cells Wet Cells Storage Cells
Blowers	Compressors Gas Circulators Fans Ventilators
Circuit Closers/Interrupters	Circuit Breakers Contactors Controllers Starters Switches (other than sensors) Switchgear
Control Rods	Poison Curtains
Control Rod Drive Mechanisms	
Demineralizers	Ion Exchangers
Electrical Conductors	Bus Cable Wire
Engines, Internal Combustion	Butane Engines Diesel Engines Gasoline Engines Natural Gas Engines Propane Engines

Component Type (Cont'd.)	Component Type Includes
Engines, Internal Combustion	Butane Engines Diesel Engines Gasoline Engines Natural Gas Engines Propane Engines
Filters	Strainers Screens
Fuel Elements	
Generators	Inverters
Heaters, Electric	
Heat Exchangers	Condensers Coolers Evaporators Regenerative Heat Exchangers Steam Generators Fan Coil Units
Instrumentation and Controls	
Mechanical Function Units	Mechanical Controllers Governors Gear Boxes Varidrives Couplings
Motors	Electric Motors Hydraulic Motors Pneumatic (Air) Motors Servo Motors
Penetrations, Primary Containment Air Locks	
Pipes, Fittings	
Pumps	
Recombiners	
Relays	
Shock Suppressors and Supports	
Transformers	Steam Turbines Gas Turbines Hydro Turbines

Component Type (Cont'd.)	Component Type Includes
Valves	Valves Dampers
Valve Operators	
Vessels, Pressure	Containment Vessels Drywells Pressure Suppression Pressurizers Reactors Vessels

was reviewed. Any documents that contained LER-type information (equipment failure, abnormal event, etc.) 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. an LER report 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 1.4);
7. type of equipment involved with the reportable event (Table 1.5);
8. type of instrument involved with the reportable event (Table 1.5);
9. status of the component (equipment) at the time of the occurrence (Table 1.6);
10. abnormal condition associated with the reportable event, e.g., corrosion, vibration, leak, etc. (Table 1.6);
11. cause of the event (Table 1.6); and
12. significance of the reportable event. Each reportable event was screened using criteria as a step in the evaluation process (See Sect. 3.2 and following for further discussion of the criteria, the use of the criteria, and a listing of the criteria.)

Table 1.4 Data Collected for Reportable Events — Plant Status and System Involved

PLANT STATUS	
A	Construction
B	Operation
C	Refueling
D	Shutdown

SYSTEM	
A	Chemical and Volume Control
B	Component Cooling
C	Condensate Purification
D	Condenser Cooling
E	Containment
F	Containment Air Cooling
G	Containment Filtering
H	Containment Hydrogen Control
I	Containment Isolation
J	Containment Purge
K	Containment Spray
L	Core Reflooding
M	Electric Power
N	Emergency Cooling/LPSI
O	Emergency Electric Power
P	Engineered Safety Features
Q	Fire Protection
R	Hydraulic
S	Main Cooling
T	Pneumatic
U	Radiation Monitoring
V	Reactor Control
W	Reactor Protection
X	Safety Injection/HPSI
Y	Secondary Cooling/Aux.
Z	Secondary Cooling/Feedwater
AA	Secondary Cooling/Steam
BB	Service Water
CC	Shutdown Cooling
DD	Waste Disposal
EE	Ventilation
FF	Reactor Internals

Table 1.5 Data Collected for Reportable Events — Equipment Involved and Instrument Involved

EQUIPMENT		INSTRUMENTATION	
A	Accumulator	A	Alarm
B	Air Drier	B	Amplifier
C	Battery and Charger	C	Electronic Function Unit
D	Bearing	D	Failed Fuel Detection Instrument
E	Blower and Dampers	E	Flow Sensor
F	Breaker	F	In-Core Instrument
G	Cables and Connectors	G	Indicator
H	Condenser	H	Intermediate Range Instrument
I	Control Rod	I	Level Sensor
J	Control Rod Drive	J	Meteorological Instrument
K	Cooling Tower	K	Position Instrument
L	Crane	L	Power Range Instrument
M	Demineralizer	M	Pressure Sensor
N	Diesel Generator	N	Radiation Monitor
O	Fastener	O	Recorder
P	Filter/Screen	P	Relay
Q	Flange	Q	Seismic Instrument
R	Fuel Element	R	Solid State Device
S	Fuse	S	Start-Up Range Instrument
T	Generator	T	Switch
U	Heat Exchanger	U	Temperature Sensor
V	Heater		
W	Internal Combustion Engine		
X	Motor		
Y	Nozzle		
Z	Pipe and Pipe Fitting		
AA	Power Supply		
BB	Pressure Vessel		
CC	Pressurizer		
DD	Pump		
EE	Recombiner		
FF	Seal		
GG	Shock Absorber		
HH	Solenoid		
II	Steam Generator		
JJ	Storage Container		
KK	Support Structure		
LL	Transformer		
MM	Tubing		
NN	Turbine		
OO	Valve		
PP	Valve, Check		
QQ	Valve Operator		

Table 1.6 Data Collected for Reportable Events — Component Status, Abnormal Condition, and Cause

	COMPONENT STATUS
A	Maintenance and Repair
B	Operation
C	Testing
	ABNORMAL CONDITION
A	Age
B	Airborne Release
C	Concentration
D	Corrosion
E	Crack
F	Crud
G	Environmental Anomaly
H	Erosion
I	Exposure
J	Fatigue
K	Fire
L	Instrument Calibration
M	Instrument Set Point Drift
N	Leak
O	Liquid Level
P	Lubrication
Q	Open/Short Circuit
R	Operator Communication
S	Operator Incorrect Action
T	Procedures
U	Records
V	Sampling
W	Smoke
X	Stress
Y	Stress Corrosion
Z	Vibration
AA	Waterborne Release
BB	Wear
CC	Weld
	CAUSE
A	Administrative Error
B	Design Error
C	Fabrication Error
D	Inherent Failure
E	Installation Error
F	Lightning
G	Maintenance Error
H	Operator Error
I	Weather

1.4 Events of Environmental Importance and Radioactivity Releases

Based upon reviewing forced shutdowns, power reductions, reportable events (environmental LERs), and operating reports, any significant or recurring environmental problems were summarized.

The routine radioactivity releases were tabulated as well, and releases where limits were exceeded were reviewed and discussed.

1.5 Evaluation of Operating Experience

Based upon the review involving screening, categorizing, and compiling data, the operating history of the plant was evaluated. Judgments and conclusions were made regarding safety problems, operations, trends (recurring problems), or potential safety concerns.

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 report to provide specific plant and equipment details when necessary.

2. SOURCES OF INFORMATION UTILIZED IN THE REVIEW

Several sources of information and 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[2] from 1974 through 1979 (the first Gray Book was issued in May 1974). Prior to 1974, the annual or semiannual reports were used to compile the availability factors only.

2.2 Forced Reactor Shutdowns and Power Reductions

The review of the forced power reductions involved checking the following sources for completeness of details and accuracy:

1. *Nuclear Power Plant Operating Experience for 19XX*, for the years 1973, 1974–75, 1976, 1977, and 1978 (Refs. 3, 4, 5, 6, and 7). The report for 1979 has not been published. However, since the work on the section of these reports on outages has been performed by NSIC since 1973, the draft copy of this report for 1979 was available;
2. The Gray Book - NUREG-0020 Series;[2]
3. Annual or semiannual reports from the time of startup through 1977. For 1977 through 1979, monthly operating reports were used because the utilities were no longer required to file annuals. 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 the reportable events. The material on the NSIC computer file consists of the appropriate bibliographic material, title, 100-word abstract, and keywords. When it was necessary to obtain additional information on the event, the original LER (or equivalent) was consulted by (1) examining those full-size copies on file at NSIC (for the years 1976 through 1979); (2) the microfiche file of docket material at NSIC; or (3) the appropriate operating report (semiannual, annual, or monthly report).

2.4 Environmental Events and Radioactivity Releases

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 the report *Radioactive Materials Released from Nuclear Power Plants — Annual Report 1977*, NUREG-0521. This report presents year-by-year comparisons for plants in a number of different categories (solid, gas, liquid, noble gas, tritium, etc.). The data for 1978 was taken from the report *Radioactive Materials Released from Nuclear Power Plants — Annual Report 1978*, NUREG/CR-1497, which was published in March 1981. The data for 1979 was 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 ~35 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 Ginna 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-1970's, it was not unusual for licensees to submit on their own or at NRC (AEC's) request 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. Events described in the following publications often contained details, evaluations, or assessments other than those provided in the reportable event (or shutdown) as a result of further analyses and examination:

1. *Reports to Congress on Abnormal Occurrences*, NUREG-0020 series;
2. "Power Reactor Event Series" (formerly Current Event Series) published by NRC;
3. "Operating Experience Section" of the *Nuclear Safety* journal; and
4. NRC's Office of Inspection and Enforcement's (I&E) publications
    a. Operating Experience Bulletins
    b. IE Bulletins
    c. IE Circulars
    d. IE Information Notices.

3. CRITERIA AND CATEGORIZATION FOR THE EVALUATIONS OF THE OPERATING HISTORY

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

The shutdowns were evaluated against the design basis events (DBE's) as set forth in Chap. 15 of the SRP. The DBE's are those postulated disturbances in process variables or postulated malfunctions or failures of equipment for which the plants are to be designed to withstand and for which the licensees are expected to analyze and include in safety analysis reports (SAR). In the SAR, the effects of anticipated process disturbances and postulated component failures are to be examined to determine their consequences and to evaluate the capability built into the plant to control or accomodate 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. Reactivity and power distribution anomalies,
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 that are to be considered by the licensee in the SAR are presented in Table 3.1

Those shutdowns identified as DBE 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 non-DBE shutdowns were examined as part of a trends analysis.

The reportable events were screened using the criteria presented in Sect. 3.2 (and following) and were categorized according to their significance. The information collected on the reportable events (as outlined in Tables 1.4 through 1.6) was used to analyze trends for all reportable events — those identified as significant or non-significant.

Table 3.1 Initiating Event Descriptions for Design Basis Events as Listed in Standard Review Plan Chapter 15 (Revision 3)

1. Increase in Heat Removal by the Secondary System

   1.1 Feedwater system malfunctions that result in a decrease in feedwater temperature
   1.2 Feedwater system malfunctions that result 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 PWR

2. Decrease in Heat Removal by the Secondary System

   2.1 Steam pressure regulator malfunction or failure that results in decreasing steam flow
   2.2 Loss of external electric load
   2.3 Turbine trip (stop valve closure)
   2.4 Inadvertent closure of main steam isolation valves
   2.5 Loss of condenser vacuum
   2.6 Coincident loss of onsite and external (offsite) a.c. power to the station
   2.7 Loss of normal feedwater flow
   2.8 Feedwater piping break

3. Decrease in Reactor Coolant System Flow Rate

   3.1 Single and multiple reactor coolant pump trips
   3.2 BWR recirculation loop controller malfunctions that result in decreasing flow rate
   3.3 Reactor coolant pump shaft seizure
   3.4 Reactor coolant pump shaft break

4. Reactivity and Power Distribution Anomalies

   4.1 Uncontrolled control rod assembly withdrawal from a subcritical or low power startup 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
   4.4 Startup 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

Table 3.1 (continued)

4.6 Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant of a PWR
4.7 Inadvertent loading and operation of a fuel assembly in an improper position
4.8 Spectrum of rod ejection accidents in a PWR
4.9 Spectrum of rod drop accidents in a BWR

5. Increase in Reactor Coolant Inventory

5.1 Inadvertent operation of ECCS 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 2.1 through 2.6 and item 1.2

6. Decrease in Reactor Coolant Inventory

6.1 Inadvertent opening of a pressurizer safety or relief valve in a PWR or a safety or relief valve in 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 2.7, 2.8, and 1.3

7. Radioactive Release from a Subsystem or Component

7.1 Radioactive gas waste system leak or failure
7.2 Radioactive liquid waste system leak or failure
7.3 Postulated radioactive releases due to liquid tank failures
7.4 Design basis fuel handling accidents in the containment and spent fuel storage buildings
7.5 Spent fuel cask drop accidents

8. Anticipated Transients Without Scram

8.1 Inadvertent control rod withdrawal
8.2 Loss of feedwater
8.3 Loss of a.c. 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

The review approach with respect to operational events (forced shutdowns and reportable occurrences) consisted primarily of a three-step process: 1) compile information on the events, 2) screen the events for significance using selected criteria and guidelines, and 3) evaluate 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.

The reviewers worked semiindependently (brief exchanges of ideas and information) and then were brought together periodically for discussion and final resolution as to how events were to be categorized and how the criteria were to be used consistently.

3.1 Significant Shutdowns and Power Reductions

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

3.1.1 Criteria for significant shutdowns and power reductions

As indicated previously, the occurrences identified as design basis events were used as criteria to categorize and note significant shutdowns. These events are listed in Table 3.1 as they are found in SRP Chap. 15.

3.1.2 Use of criteria for determining significant shutdowns and power reductions

The generic DBE initiating event types, e.g., "increase in heat removal by the secondary system" or "decrease in reactor coolant system," 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

due to a feedwater regulator valve failing open, the shutdown is a DBE generic type 1 event. Specifically, based upon 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 DBE's, 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 design basis event 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 closures of main steam isolation valves.

3.1.3 Non-DBE shutdown and power reduction categorization

Those non-DBE shutdowns 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 for non-DBE shutdowns taken from the Gray Book (listed in Table 1.1) are limited and very general, while NSIC cause categories

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 upon 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 upon proper procedures and improper understanding or misinterpretation on the part of the operator of what is to be done → incorrect action)
- N 6.3 Inadvertent action (purpose and action not related, e.g., 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

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 Tech Specs

are more specific. Thus, as an example, the number of Gray Book causes noted as equipment failure should not be expected to equal those identified as equipment failures with the NSIC categories. Other NSIC categories, such as component failure, could be classified as an equipment failure if the only available designations for cause were those listed in the Gray Book.

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 and are termed significant. Those criteria in Table 3.4 indicate 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 upon (1) engineering judgment; (2) the systems, equipment, or components involved; or (3) whether the safety of the plant was compromised. The final evaluation for significance considered whether a DBE was initiated or whether a safety function was compromised such that the system could not mitigate the propagation of events for which it was designed. Thus, the number of events categorized finally as significant was reduced considerably by these steps in the review process.

Table 3.3 REPORTABLE EVENT CRITERIA — SIGNIFICANT

SIGNIFICANCE CATEGORY	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 which could have been a greater threat to plant safety with different plant conditions, the advent of another credible occurrence, or 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 FOR CONDITIONAL SIGNIFICANCE	EVENT DESCRIPTION
C1	A single failure occurs in a non-redundant system.
C2	Two apparently unrelated failures occur during the same event.
C3	A problem results in an off-site radiation release or personnel exposure.
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 ESF actuation occurs during an event.
C7	A particular occurrence is recognized as having a significant recurrence rate.
C8	Other.

Those events involving radioactivity releases were automatically categorized as a conditionally significant 3 event and held for discussion in the respective "environmental and release" sections of the report.

3.2.3 Non-significant reportable events

Those reportable events not identified as significant or conditionally significant were categorized as non-significant (with an "N" in the significance column in the coding sheets in the appendices). These events and the events rejected during the additional review step as noted above were further reviewed by compiling a tabular summary of the systems (Table 1.4) to detect trends and recurring problems. The systems selected yield meaningful information concerning the system's ability to mitigate accident sequences or mitigate the effects of such accident sequences.

4. OPERATING EXPERIENCE REVIEW OF GINNA

4.1 Summary of Operational Events of Safety Importance

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

Shutdowns and power reductions indicated the number and types of DBEs entered. The reportable events and special environmental events indicated the number of times each engineered safety function was compromised. The results of the analyses identified 23 DBE's entered. Additionally, two events were identified where loss of safety system function occurred in some engineered safety features.

4.2 General Plant Description

R. E. Ginna Nuclear Station is a Westinghouse pressurized water reactor (PWR) of 470-MWe net maximum dependable capacity, owned by Rochester Gas & Electric Corporation (RGE) and located in Ontario, New York. The Architect/Engineer was Gilbert Associates, and the constructor was Bechtel. The condenser cooling method is once-through, and Lake Ontario is the condenser cooling water source. The Plant is subject to license DPR-18, issued September 19, 1969, pursuant to Docket Number 50-244. The date of initial reactor criticality was November 9, 1969, and commercial generation of power began July 1, 1970.

The nearest city is Rochester, New York, 17 miles away. The population within 30 miles is 840,000 and within 50 miles, 1,200,000.

4.3 Availability and Capacity Factors

Table 4.1 contains Ginna's availability and capacity factors. The rated net power was boosted from 420 to 470 MWe on March 3, 1972, to more fully utilize the installed generating capacity. From mid 1972 until mid 1973 there was a regulatory limit of 1266 MWt (83% of design value of 1520 MWt). The reactor availability from 1970 through 1979 stayed above 70% except for 2 years, 1974 and 1976, when major outages were necessary for repairs (see Sect. 4.4). The ten full years of operation, 1970 through 1979, averaged 78.1% reactor availability and 74.6% plant availability. Capacity factors were not available prior to 1975. The MDC and DER capacity factors from 1975 through 1979 averaged 69.5 and 68.5%, respectively.

Table 4.1 Availability and Capacity Factors for Ginna

	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979
Reactor Availability	69.3	72.9	77.6	72.0	95.3	63.9	81.5	69.0	87.0	86.9	74.8
Unit Availability	34.3	69.4	75.9	69.2	95.0	62.4	76.7	58.2	85.5	80.6	72.8
Unit Capacity (MDC)*	ND†	ND	ND	ND	ND	ND	73.9	49.9	73.6	78.2	71.9
Unit Capacity (DER)‡	ND	ND	ND	ND	ND	ND	73.9	47.9	70.6	78.2	71.9

*MDC = Maximum Dependable Capacity

†ND = No Data

‡DER = Design Electrical Rating

4.4 Review of Reactor Shutdowns and Power Reductions

Table A.1 provides a comprehensive summary of information concerning shutdowns and power reductions at Ginna. Some information is still missing (denoted by blanks), however, and some was assumed (denoted by "A/"). More complete information was provided when events generated reports; in such instances, more detailed descriptions are in Sect. 4.5.

Tables 4.2 and 4.3 of forced shutdowns and power reductions summarize Table A.1. Causes of forced shutdowns, item I.3 in Table 4.2 and item I.2 in Table 4.3, are dominated at Ginna by equipment failures. Shutdowns reported to be caused by operator errors amount to only 10% of the total, and no power reductions are attributed to operator error. More than one system is often involved in a shutdown or power reduction and in some cases the cause of the shutdown is not ascertainable. Therefore, the totals for cause, shutdown method, and system in Tables 4.2 and 4.3 are not comparable.

Table 4.2 Forced Shutdown Summary for Ginna

	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	Total
I. Forced Shutdowns												
1. Total Number	11	27	14	12	5	9	13	14	6	4	3	118
2. Total hours down	79.4	1435	169	86.8	434	3291	392	2026	380	281	1107.5	9681.7
3. Cause*												
A. Equipment Failure	6.(70.7)	17(613.7)	9(127.4)	10(62.5)	4(292)	7(3000)	11(336)	12(1486)	6(380)	3(249)	1(416)	86(7029.3)
B. Maintenance or Testing	-	6(820.7)	1(17.3)	1(24)	-	2(291)	1(44)	3(540)	-	-	1(.5)	15(1737.5)
D. Regulatory Restriction	-	-	-	-	-	-	-	-	-	-	1(691)	1(691)
E. Operator Training/ License Exam	-	-	-	-	-	-	-	-	-	-	-	-
F. Administrative	-	-	-	-	-	-	-	-	-	1(32)	-	1(32)
G. Operational Error	5(8.7)	4(1.0)	2(1.8)	1(.3)	-	-	-	-	-	-	-	12(11.8)
H. Other	-	-	-	-	1(142)	-	1(12)	-	-	-	-	2(154)
4. Shutdown Method												
1. Manual	2	10	-	4	2	5	7	10	6	2	2	50
2. Manual Scram	1	-	4	4	-	-	1	2	-	-	-	12
3. Automatic Scram	8	15	9	4	3	4	5	2	-	2	1	53
4. Continuation	-	-	-	-	-	-	-	1	-	-	-	1
II. Total number of SRP Related Shutdowns (These are included in Totals of Part I)	-	1	6	1	1	-	5	5	3	1	-	23

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

Table 4.2 (continued) Forced Shutdown Summary For Ginna

	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	Total
III. System Involved												
1. Reactivity Control Systems	-	1	1	2	-	2	1	4	2	-	1	14
2. Coolant Recirculation Systems and Controls (CB)	1	4	2	3	-	1	-	-	-	2	1	14
3. Main Steam Isolation Systems & Controls (CD)	-	-	2	-	-	-	-	-	-	-	-	2
4. Residual Heat Removal Systems & Controls (CF)	-	-	-	-	-	-	-	-	1	-	-	1
5. ECCS & Controls (SF)	1	2	-	-	-	-	-	1	-	-	-	4
6. Reactor Trip Systems (IA)	-	2	2	-	-	-	-	-	-	1	-	5
7. Engineered Safety Feature Instrument Systems (IB)	-	1	-	-	-	-	-	-	-	-	-	1
8. Safety-Related Display Instrumentation (ID)	-	1	-	-	-	-	-	1	-	-	-	2
9. Other Instrument Systems Required for Safety (IE)	-	1	1	-	-	-	-	-	-	-	-	2
10. Offsite Power Systems and Controls (EA)	-	1	-	1	1	-	1	-	-	-	-	4
11. AC Onsite Power Systems & Controls (EB)	-	1	-	-	-	-	-	-	-	-	-	1
12. DC Onsite Power Systems & Controls (EC)	-	1	-	-	-	-	-	-	-	-	-	1
13. Composite AC & DC Systems (ED)	-	-	-	-	-	2	1	1	-	-	-	4
14. CVCS & Liquid Poison Systems & Controls (PC)	-	-	1	-	1	-	-	-	-	-	-	2
15. Turbine Generators & Controls (HA)	-	6	-	3	-	1	4	5	-	-	-	19
16. Main Steam Supply System & Controls (HB)	-	-	1	1	-	2	4	2	3	1	1	15
17. Main Condenser Systems & Controls (HC)	2	-	-	-	-	-	1	-	-	-	-	3
18. Circulating Water Systems & Controls (HG)	-	-	-	-	1	-	-	-	-	-	-	1
19. Condensate and Feedwater Systems & Controls (HH)	7	11	6	1	2	1	2	-	-	-	2	32

Table 4.3 Power Reduction Summary for Ginna

	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	Total
I. Power Reductions												
1. Total number	-	1	-	-	5	5	7	11	3	4	7	43
2. Cause												
A. Equipment Failure	-	-	-	-	-	1	2	8	3	4	5	23
B. Maintenance or Testing	1	-	-	-	5	4	4	3	-	-	1	18
D. Regulatory Restriction	-	-	-	-	-	-	-	-	-	-	-	-
E. Operator Training/License Exam	-	-	-	-	-	-	-	-	-	-	1	1
F. Operational Error	-	-	-	-	-	-	-	-	-	-	-	-
G. Other	-	-	-	-	-	-	1	-	-	-	-	1
3. System Involved												
1. Reactivity Control System (RB)	-	-	-	-	-	-	-	-	-	-	1	1
2. Other Coolant Subsystems & Their Controls (CJ)	-	-	-	-	-	-	-	-	-	-	1	1
3. Emergency Core Cooling Systems & Controls (SF)	-	-	-	-	-	1	-	-	-	-	-	1
4. Offsite Power Systems & Controls (EA)	-	-	-	-	-	-	1	-	-	-	-	1
5. AC Onsite Power Systems & Controls (EB)	-	-	-	-	-	-	-	1	-	-	1	2
6. Air Cond., Heating, Cooling, & Ventilation (AA)	-	-	-	-	-	-	1	-	-	-	-	1
7. Turbine-Generators & Controls (HA)	-	1	-	-	1	-	1	1	-	1	-	5
8. Main Steam Supply System & Controls (HB)	-	-	-	-	-	-	-	-	-	-	3	3
9. Main Condenser Systems & Controls (HC)	-	-	-	-	-	3	-	-	-	-	-	3
10. Condensate and FW Systems & Controls (HH)	-	-	-	-	4	1	4	9	3	3	1	25

4.4.1 DBE Initiating Events

Of the 118 total forced shutdowns and power reductions accumulated at Ginna, 23 fell into DBE initiating event categories as shown in Table 4.4. None of these events initiated any sequence that led to any significant economic loss or safety hazard to the plant or the environs.

The categorized events were dominated by control rod malfunctions, steam generator tube leaks, and spurious closures of the main steam isolation valves (MSIVs). Seven of the eight control rod malfunctions generated individual reports (the exception is the first one, July 5, 1971), as did all five steam generator tube failures and one MSIV spurious closure (June 23, 1975). Control rod failures and steam generator tube failures are thoroughly discussed as recurring reportable events in Sect. 4.5.2.

The only turbine trip that caused shutdown occurred on January 27, 1970 because of "the loss of the electro-hydraulic (EH) governor pump pressure."[8] The only loss-of-offsite-power event occurred on October 21, 1973, and is described in detail in Sect. 4.5.1.

All three loss-of-normal-feedwater events occurred within 2 months of each other on July 11 and 13 and September 5, 1971. The two in July most likely are related (instrument bus #1 inverter failures), but not enough information is given in the semiannual report to be conclusive. The September event occurred when the containment ventilation was reset causing the feedwater valves to close by mistake.[9]

Table 4.4 DBE Initiating Events at Ginna

Description	DBE Category	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	Total
Loss of Offsite Power	D2.2	–	–	–	–	1	–	–	–	–	–	–	1
Turbine Trip	D2.3	–	1	–	–	–	–	–	–	–	–	–	1
Spurious Closure of Main Steam Isolation Valves	D2.4	–	–	2	1	–	–	2	–	–	–	–	5
Loss of Normal Feedwater Flow	D2.7	–	–	3	–	–	–	–	–	–	–	–	3
Control Rod Malfunction	D4.3	–	–	1	–	–	–	1	4	2	–	–	8
Steam Generator Tube Failure	D6.3	–	–	–	–	–	–	2	1	1	1	–	5
Total		0	1	6	1	1	0	5	5	3	1	0	23

Note that the trend of total number of DBEs per year bears no correlation with other trends, such as plant performance as measured by total number of shutdowns per year or total downtime per year.

4.4.2 Trends and safety implications of shutdowns and power reductions

Over the more than 10 years of operation at Ginna, forced downtime averaged nearly 40 d/year. Annual downtime was dominated by single, large events such as the 114-d shutdown in 1974 to overhaul the #2 low-pressure turbine. Total annual hours of forced down-time followed no trend. Years 1970, 1974, 1976, and 1979 each recorded more than 1100 h forced out of service; the other 6 years each recorded less than 450 h forced out of service.

Four out of every five forced shutdowns were caused by equipment failures. Operator errors that were reported to have caused shutdowns fell from five in the 2 months in 1969 and four in the first full year of operation (1970) to one in 1972 and none thereafter. Some fine tuning of the feedwater system and its controls made it easier to operate after 1970, a year in which all four operator-error shutdowns were because of feedwater regulation problems. Today, the general availability and use of simulators helps alleviate this problem.

All of the 118 forced shutdowns that did not fall into DBE categories were put in NSIC categories * (see Sect. 3.1.1). In NSIC event categories, leaks curtailed operations most often, with 31 shutdowns caused by leaks in the secondary system and 14 caused by leaks in the primary system. Thirty eight

*Except eight forced shutdowns [12/6/69, 12/14/69, 12/15/69, 12/30/69 (2), 1/3/70, 1/4/72, and 7/28/72] that were not documented sufficiently to enable categorization.

equipment failures in normal operation (equipment other than instruments and controls) forced shutdowns or power reductions.. The third major type of NSIC event was instrumentation and control anomalies, with 31 leading to shutdowns or forced reductions. No fuel/cladding failures required shutdowns or forced reductions, and no environmental operating constraints curtailed operations either. Nine events had an undetermined category.

Causes of the forced reductions in reactor power were evenly split between equipment failures and maintenance or testing restrictions. Forced reductions in power were reported only once at Ginna before the fourth year of operation. After 1973 began, however, forced reductions were reported nearly five times per year. They usually lasted only a few hours, and seldom was power reduced more than 50%. Power reductions under 30% were included only if the associated failure had safety significance.

A discussion of shutdowns and power reductions for each year 1969 through 1979 follows.

1969

The Ginna PWR went critical at 0530 on November 9, 1969, and the generator was first synchronized on December 2. Low-power physics testing in December showed excellent agreement between experimental and predicted values (Ref. 8, p. 2).

Operator errors dominated the causes for forced outages, inducing 5 of the 11 recorded incidents (Table A.1.1). The steam and power system was involved nine times.

Two shutdowns were caused by Technical Specification restrictions. On December 16, safeguard valve 850B on the RHR pump suction line from the containment sump failed to open during a test. Both discs of this double-disk valve were bowed from internal pressure. When the RHR system is put in service, the water between the disks is heated to about 285°F, creating a large internal pressure. (The valve is tested for 750 psig in this space.) The recommendation was made that any double-disc valves that are closed during RHR operation be tested immediately after shutting down the system.[10]

The other shutdown caused by a Technical Specification restriction occurred when the discharge valve on the 1B motor-driven auxiliary feedwater pump (AFWP) was "found to be inoperable" (Ref. 8, p. 17). No cause has been documented.

During the automatic scram on December 3, one shutdown rod (E-11) failed to drop. Exercise of the shutdown bank freed the rod. This situation did not occur again and is not considered safety significant.

Two recurring problems first surfaced during this report period. A condenser hotwell problem was investigated on December 14, but no results were reported (Ref. 8, p. 17). Feedwater control problems caused eight forced shutdowns in the first 7 months of Ginna's operation. The first four shutdowns because of feedwater control problems occurred in this report period. On December 3, three shutdowns resulted from operator errors in aligning feedwater valves. The shutdown on December 30 occurred in switching to the auxiliary feedwater. After June, 1970, manual control of the feedwater flow caused no problems; some fine tuning of the feedwater control system made it easier to operate.

1970

In 1970, the reactor and control system proved its capacity to operate under the most severe transient conditions: four automatic scrams from 100% power (420 MWe) occurred, and the plant responded as designed each time.

This year had 27 forced shutdowns (plus 1 power reduction), the most in the 10 years of operations at Ginna (Table 4.2). The shutdowns are attributed partly to the operator's lack of familiarity with the new plant; ten of the 27 forced shutdowns were caused by maintenance, testing, and operator errors. Only 12 of these errors occurred over the following 9-year period (1971 through 1979). The remaining 17 shutdowns were caused by failures of new equipment.

Two lengthy shutdowns accounted for 80% of the forced downtime this year. On May 4, the unit went down for 34 d to repair the #2 low-pressure turbine blades (Table A.1.2). On September 30, the unit went down for 14 d to repair, among other things, a leak in a reactor coolant temperature detector.

One incident at the beginning of 1970, only 2 months after the initial criticality, exemplified the problems in maintenance and testing that plagued Ginna in its early years. Following a spurious safety injection signal on January 3, one train of the high-pressure coolant injection system did not operate. After dirty contacts on a relay were found and cleaned, a test signal activated the train. However, at this point, motor operator valve (MOV)-878B failed closed, thereby stopping the SI flow from entering the cold leg. After adjusting the valve's arms and contacts, operators stroked it successfully seven times.[11]

The control rod step counters malfunctioned on January 17, requiring a manual shutdown. The ability to safely shut down the reactor was not compromised.

The first of six MSIV failures over the 10 years of operation at Ginna occurred during the low-pressure turbine failure on May 14, when MSIV-1B did not close on a manual signal from the control room. None of these failures compromised the safety of the plant.

A recurring problem with the packing of the pressurizer spray valves first surfaced on May 2, at which time the reactor was shut down to repack these valves. On July 5, packing was added to pressurizer control valve (PCV)-431A, but design capability of these valves was never compromised.

The first of many shutdowns to repair leaking pressurizer relief valves was required on July 12; others followed on November 15 and December 12. These leaks never jeopardized the safe operation of Ginna.[12]

Another problem recurring over the life of the plant first occurred on January 27 when the EH turbine governor lost pressure, causing the turbine to trip (this was an DBE Category 2.3 initiating event). The manual feedwater control was too slow in responding to the consequent steam generator level transient, and the reactor scrammed on lo-lo steam generator level and was down for 25 min (Ref. 8, p. 18). On June 19, a manual turbine trip to repair the governor caused an unintentional reactor trip. During the September 6 shutdown, unspecified repairs on the EH governor control system were made. None of the 12 EH governor-related shutdowns over the life of Ginna compromised the safe operation of the plant.

Feedwater manual control problems and leaking condenser tubes continued to plague Ginna in 1970. Brief shutdowns caused by manual feedwater control errors occurred on January 27 (discussed previously) and 28, March 30, and

June 19. Condenser tubes were plugged during two shutdowns in December, at which time other necessary maintenance was concurrently accomplished.

1971

The number of forced outages dropped to 14 in 1971 (Table A.1.3), with no reported power reductions. The unit was generally base loaded at full load during this period, as it was throughout the following years. Total forced downtime was only 169 h, second lowest in Ginna's history. On February 3 and November 12, the previously mentioned recurring problem with the packing in the pressurizer spray valves occurred causing forced outages for repairs of 27.2 h total duration. The series of failures leading to a shutdown on June 30 is described in detail in Sect. 4.5.2. On July 11, the inverter on the 125-V dc connection to instrument bus #1 led to an automatic scram. This problem surfaced four more times in the period 1974 through 1976.

During the refueling outage commenced February 27, all fuel assemblies were tested using the wet-dry sipping technique. Defective fuel was found in region 3.

1972

The Atomic Energy Commission (AEC) granted permission to increase power from 1300 to 1520 MWt on March 1. The 1520-MWt level was reached on April 12.

This year saw the least forced downtime in Ginna's history: 12 forced shutdowns for 87 h total (Table A.1.4). The EH turbine governor system caused shutdowns again on February 24 and March 8. Shutdowns accounting for 28% of this year's total downtime were necessary on March 20 and September 5 to repair packing on pressurizer spray valves. Malfunctions in the drive

and logic supply power for the control rods caused shutdowns on June 23 and July 27, respectively. A leak in the block valve downstream of the power-operated relief valve necessitated a shutdown on December 14, 1972.

During the refueling outage commenced April 14, inspections found defective fuel: some rods were bowed, some were collapsed and some leaking. Sixty-one new fuel assemblies replaced all of region 3, part of region 2, and 3 assemblies in region 1.

On June 24 the turbine generator returned to service using an interim set of conditions. The maximum nuclear power was not to exceed 1266 MWt (83% of rated power). Other conditions required various tests to be performed at various power levels.

1973

Only five forced shutdowns occurred in 1973. Five power reductions also occurred, all for required maintenance (Table A.1.5).

A new problem surfaced on June 9 when the bearing shoes on the 1A main feedwater pump needed to be replaced. Power reductions were also necessary on August 19 and September 10 to repair the B main feedwater pump.

Tubes in the main condenser leaked again and caused a shutdown for repairs on January 12 and a 38% power reduction on March 9.

Further repairs on the turbine EH control system necessitated a 50% power reduction in November.

Two lengthy shutdowns occurred this year. On July 22, an 8.5-d shutdown began to repair a disconnected flow transmitter on the B auxiliary feedwater pump control valve. The disconnection occurred during the only waterhammer event in Ginna's history severe enough to cause a shutdown.[13] In

October, Ginna experienced its sole loss-of-offsite-power event, described in detail in Sect. 4.5.1. Ginna was restarted 6 d later.

1974

This year Ginna suffered the most forced downtime in its history. Of the total 3291 h out of service, 83% were caused by a blade failure in the same low-pressure turbine that caused a 5-week shutdown in May and June 1970. The maintenance overhaul began January 1 and ended April 25 (Table A.1.6). Refueling was completed during this outage.

Instrument bus inverters failed twice, on April 27 and July 26, both causing a low steam generator level and low feedwater flow. The same event happened in July 1971.

A repair was attempted on June 29 to stop a leak on a charging pump filter vent line. After the 43-h repair, the reactor was restarted. It was shut down 14 h later, and the repair job was redone correctly.

The 1A main feedwater pump impeller failed on May 20, forcing a power reduction. This was the fifth power reduction or shutdown caused by mechanical failure of a main feedwater pump.

On November 2, the reactor was manually shut down to begin an inspection of the steam generator tubing. This inspection, lasting more than 11 d, was the first of seven outages forced by steam generator tube degradations or inspections. RG&E's practice throughout the life of Ginna has been to plug leaks in the steam generators and condenser as soon as they are observed as opposed to continuing operation with leakage. Though there had been no leakage at this time, two tubes were plugged.

The recurring problem with the condenser tubes forced three power reductions of 52% in December. Freon checks of the 1B condenser tubes were made. Since changing the secondary chemistry from phosphate to all volatile treatment (AVT) during the last shutdown for steam generator tube inspection, condenser tube integrity problems have caused more frequent shutdowns. Whereas in the five years of operation previous to November 1974 in which Ginna has been shutdown five times for condenser tube inspections, plugging, and repairs, future years saw an average of three shutdowns per year because of condenser tube problems. However, no technical evidence has been documented to demonstrate this link.

1975

Condenser tube leaks continued to plague Ginna in 1975 (Table A.1.7). Power reductions were necessary on July 30 and December 21 and 23 to do freon checks for leaks. On December 23, one leak was found in the B condenser.

Another familiar problem caused three shutdowns in May 1975. The EH turbine control system malfunctioned and forced two shutdowns in May. The reactor was taken down to repair the control system on May 31. Two more shutdowns were to occur because of this system in 1976.

The unit was at 95% power on June 17 when a reactor trip was caused by the loss of the 1A inverter and the corresponding loss of the 1A instrument bus. The inverter failed when the pulse drive printed circuit card failed.

1976

An unusually difficult year for Ginna was 1976. Twenty-six forced outages and power reductions were recorded, only two less than during 1970, the first full year of operation. Equipment failures accounted for 20

of these events. A total of 2026 h forced out of service this year was second only to 1974, when 2737 h were needed for one job alone — the maintenance overhaul of the #2 low-pressure turbine; 36% of the downtime in 1976 was attributed to blade failures on the same turbine [January 29, July 29, and August 7 (Table A.1.8)]. Another large contributor was steam generator tube leaks, accounting for 30% of the forced downtime this year (continued from the December 30, 1975, outage and on April 24).

Three shutdowns were caused by EH turbine control problems. On April 18 water leaking into the oil from the oil cooler forced two shutdowns. On May 22 another fluid leak in the system forced another shutdown.

Other recurring problems affecting Ginna in 1976 were an instrument bus inverter failure on June 3 and a main feedwater pump impeller failure on September 10. A 46% power reduction for 1 week was necessary to repair the main feedwater pump.

A total of ten condenser tubes were plugged during four forced outages. Erosion from the bottom side of the tubes was caused by steam impingement from the steam dump lines located below the affected tubes (September 25 and December 11, 14, and 17).

1977

This year was the first of three successive years of few forced shutdowns and power reductions and high unit availability. Only six forced shutdowns and three forced power reductions occurred in 1977 (Table A.1.9).

Condenser tube leaks forced two 50% power reductions — one in January and one in March. The recurring problem with steam generator tube leaks forced an 8-d shutdown beginning July 5 for inspections and repairs.

A new problem surfaced in 1977. Shaft seals on steam generator snubbers were found leaking. Inoperable snubbers forced shutdowns on August 2 and November 2.

1978

Four forced shutdowns and four forced power reductions occurred in 1978, totaling only 281 h down. A steam generator tube leak on January 25 accounted for 73% of this downtime (Table A.1.10).

Other recurring problems were condenser tube leaks (January 11 and 14 and September 15) and packing leaks around the pressurizer spray and block valves (February 20).

1979

Three forced shutdowns and seven forced power reductions occurred in 1979. Repairs beginning December 2 on a steam generator tube leak accounted for 38% of the total downtime this year (Table A.1.11).

A gasket leak around the B steam generator handhole forced brief power reductions on October 27 and November 10 and 24.

Nearly a month was spent out of service starting July 6 because of an NRC requirement to inspect welds on the feedwater nozzles at the steam generators.

No condenser tube leaks caused shutdowns or power reductions in 1979; this problem may have been alleviated, although 1980 data are necessary for confirmation. Note that of the 22 shutdowns caused by condenser tube leaks or degradation, 17 occurred after the switch to all volatile treatment (AVT) was made on November 2, 1974.[14] Similarly, all shutdowns caused by steam

generator tube degradations occurred after this switch. However, there is no conclusive technical evidence to link these events.

4.5 Review of Reportable Events

Appendix A.2 is a table of all reportable events at Ginna. A few pieces of information, usually dates, are missing; blanks in Table A.2 indicate this. Dashes indicate "not applicable." Occasionally reviewers were able to deduce facts from the sketchy reports; these assumptions are denoted "A/." When the reportable event involved a forced shutdown, it is denoted in the "comment" column by the words "reactor shutdown." These events, then, appear in the shutdown Table A.1 as well.

Of the 23 DBE-related shutdowns in Table A.1, 14 generated reports and thus appear in Table A.2 also. Of these, seven involved control rod malfunctions, five involved steam generator tube leaks, one loss of offsite power, and one spurious closure of the MSIVs. These are of varying degrees of significance, so the overlap between Tables A.1 and A.2 does not in itself indicate particularly significant events.

From the plant startup in November 1969 through 1973, events were simply reported in letters to the AEC Directorate of Licensing and, for the purposes of Table A.2, are numbered chronologically. Starting in 1974 the events were reported as "Abnormal Occurrences" labeled chronologically with an "AO" prefix. In 1975, events were reported as "Unusual Events" (UEs) and "Reportable Occurrences" (ROs) as well as Abnormal Events. In 1976 and 1977, events were reported as ROs only. Ginna began reporting events as Licensee Event Reports in 1978, and the other designations were dropped.

The reporting pattern over the life of Ginna is shown in Figure 4.1, with the trend toward more reported events. The years 1970 through 1973 averaged 10 reported events per year whereas the years 1974 through 1979 averaged over 23 per year. The recurring failures discussed in Sect. 4.5.2.2.1 contribute to this trend. But even if these recurring failures are subtracted from the total number of reported events, the upward trend is still marked. (The year 1978 set capacity factor records and, thus, is below this trend.)

4.5.1 Significant events at Ginna

As the initial step is identifying significant events among Ginna's reported events, the reported events were screened by the significance categories listed in Section 3.2. The events meeting at least one of the significance criteria in the first step were reviewed relative to either

(1) the initiation of a DBE from Ginna's FSAR Chapter 14 (listed in Table 4.5), or

(2) the compromise of a safety function design to mitigate one or more of the DBEs.

The events identified from this review were termed significant.

Two significant events, summarized in Table 4.6 resulted from the review of Ginna's reportable events.

These events were:

1) loss of offsite power, and

2) MSIVs closure.

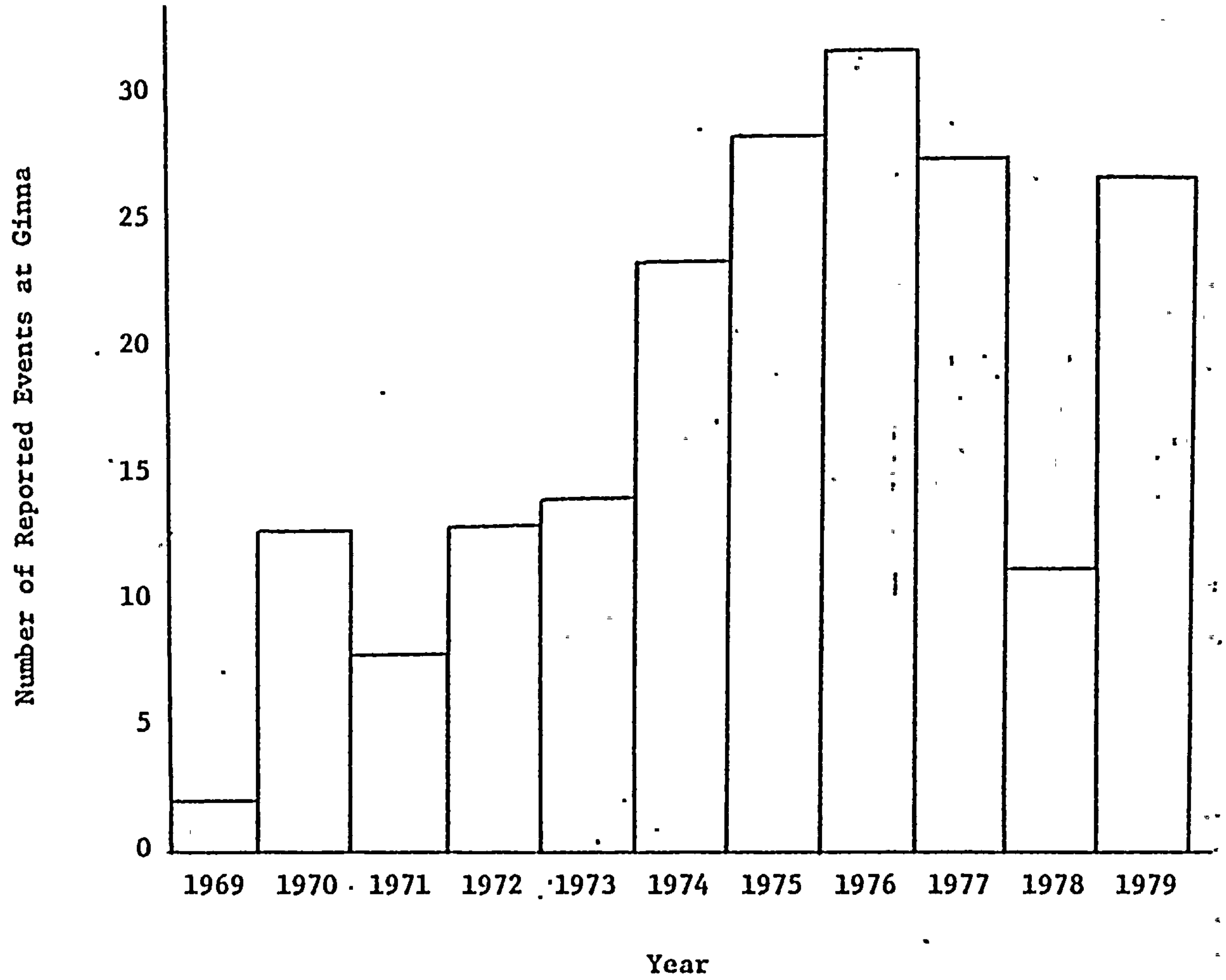


Figure 4.1 Number of Reported Events at Ginna

Table 4.5 GINNA FSAR CHAPTER 14 DESIGN BASIS EVENTS (DBEs)

14.1.1	Uncontrolled RCCA Withdrawal (k < 1).
14.1.2	Uncontrolled RCCA Withdrawal (at power).
14.1.3	Malpositioning of the Part Length Rods.
14.1.4	RCCA Drop.
14.7.5	CVCS Malfunction.
14.7.6	Loss of Reactor Coolant Flow.
14.7.7	Startup of an Inactive Reactor Coolant Loop.
14.7.8	Loss of External Electrical Load.
14.1.9	Loss of Normal Feedwater.
14.1.10	Excessive Heat Removal Due to Feedwater Temperature Decrease.
14.1.11	Excessive Load Increase.
14.1.12	Loss of All AC Power to the Station Auxiliaries.
14.2.5	Steam Pipe Rupture.
14.2.6	Rupture of CRD Housing (RCCA Ejection).
14.3	Primary System Pipe Rupture.

Table 4.6 SIGNIFICANT EVENTS[1] FOR GINNA

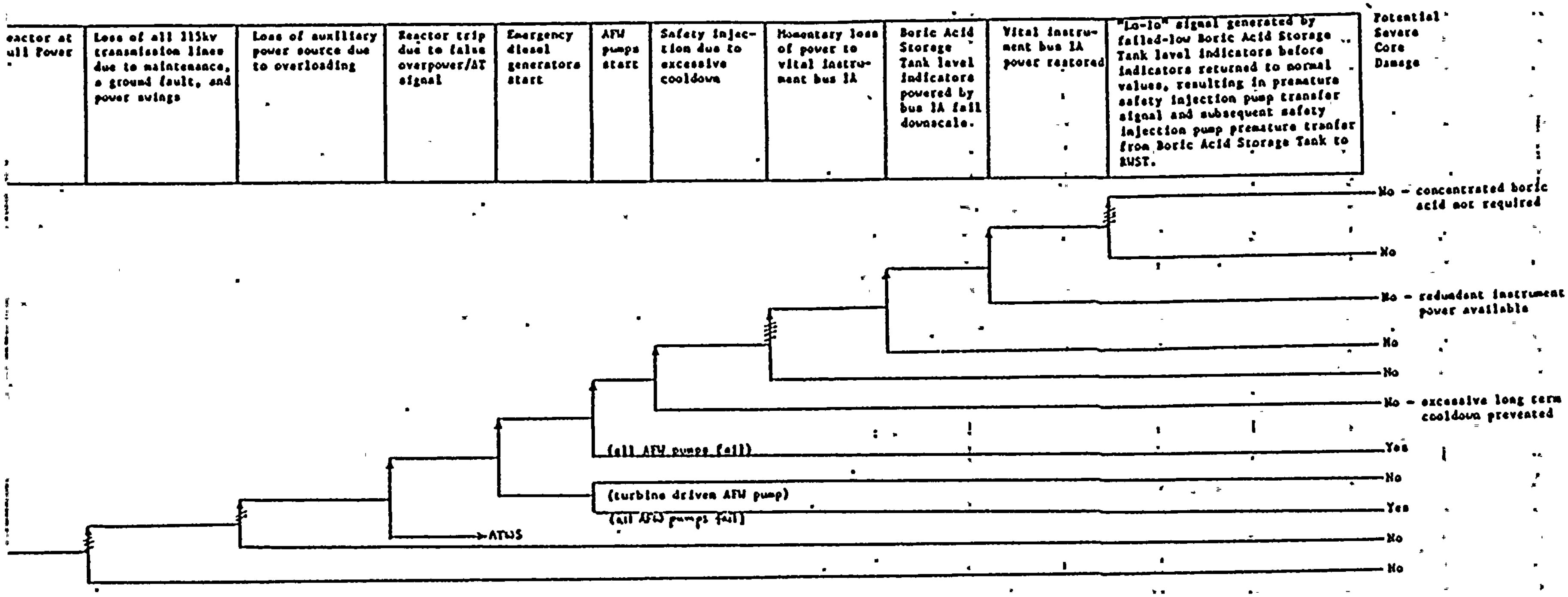
Significance Category	Number of Significant Events[2] and Number of Times Category Assigned											Total Times Assigned[3]
	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	
S1												
S2							1					1
S3					1							1
S4												
S5					1							1
S6												
S7					1							1
S8												
S9											Total	4
Total No. of Significant Events					1		1					2

1. See Table 3.3 for significance criteria descriptions.
2. Blanks indicate 0.
3. See text for explanation.

The total number of significant events, tallied in the bottom row in Table 4.6, does not equal the total in the far right column because two events, 71-02 and 73-09, required multiple significance categories.

4.5.1.1 Loss of offsite power leads to excessive cooldown rate. On October 21, 1973, the reactor was at 91% power, with one of four 115-kV transmission circuits out of service because of construction. Auxiliary power was provided from the station switchyard and an external 34.5 kV line. A second 115-kV line, sagging from the increased load, flashed over to an underbuilt 34.5 kV line, and its circuit opened. The consequent 230-MWe power swing on the remaining two 115-kV lines caused them both to trip, resulting in complete loss of generator output ability. The loss of all 115-kV transmission lines overloaded a supply circuit for the remaining auxiliary power source, causing it to trip and resulting in a total loss of offsite power plus a turbine trip and a reactor trip (Fig. 4.2).

According to an RGE letter on October 23, 1973, to the AEC Directorate of Regulatory Operations, Region 1 (NSIC 87031), disturbances on the instrument buses generated a double low level in both steam generators which started the steam driven auxiliary feedwater pump. This injected cold water into the steam generators and caused a rapid cooldown of the reactor coolant system (RCS) (an 85°F drop in the cold leg in 10 min, see NSIC 85369) which exceeded the Technical Specification limit. (TS 3.1.2.1.b states "for temperature above an indicated temperature of 290°F the [cooldown] rate shall not exceed 100°F/hr.") The rapid cooldown lowered the RCS pressure, and SI was automatically initiated on low pressure/low level in the pressurizer.



NSIC 85370, 87031 — Actual Occurrence for Loss of Offsite Power, Excessive RCS Cooldown, Safety Injection, and Failure of a Vital Instrument Bus at R. E. Ginna

Figure 4.2 ACCIDENT SEQUENCE PRECURSOR STUDY LOSS OF OFF-SITE POWER EVENT TREE FOR EVENT 10/21/73

At this point, RGE's letter of October 31, 1973 (NSIC 85370), picks up the sequence of events: power was lost for 38 s to instrument bus 1A, causing a premature transfer of the SI pump suction from the boric acid storage tanks to the refueling water storage tank. Operator error was suspected, but no malfunction was found. The concentrated boric acid was not necessary for maintaining subcriticality, and no core damage occurred.

4.5.1.2 Both MSIVs close spuriously. On June 23, 1975, the unit was operating at 88% of full power when a reactor trip occurred caused by a double low level in the "B" steam generator. The "A" and "B" main steam isolation valves had failed closed. The steam flow impinged on the leading edge of the MSIV's disks and forced the valves closed. This was a common cause failure. The same event had occurred 17 d earlier (June 6, 1975) and caused a shutdown (see Appendix A.1), but no failure cause was found at that time and no report was made. After the June 23, 1975, event, the rubber piston seat in the MSIV actuators was removed to increase the piston stroke, and 1 in. of metal was removed from the valve disk stops to permit positioning of the disk's leading edge out of the steam flow path. This event was mitigated without incident both times.

4.5.1.3 Conditionally Significant Events of Importance. The following two conditionally significant events are described in further detail because they represent partial loss of important safety system functions.

4.5.1.3.1 Safety injection train B fails to actuate on demand. Four failures occurred after an accidental SI signal on January 3, 1970. The reactor was operating at less than 1% thermal power, and repair personnel were calibrating one of six pressure transmitters associated with the safety injection logic system when the accidental signal was given. The reactor scrammed as designed,

but the B SI train did not respond. A relay failed in the open position. After relay contacts were cleaned, the B train actuated properly. The second and third failures in this event were malfunctions of two different position indicators on isolation valves inside containment. Finally, the motor-operated valve on the 2-in. SI line to the B cold leg inside containment did not operate because of an open limit switch contact. This event was flagged S3 because the four failures constituted a breach of the second line of assurance: the B SI train failures reduced the capability to mitigate loss-of-coolant accident (LOCA) as analyzed in FSAR Chap. 14.3.

4.5.1.3.2 Loss of concentrated boric acid injection capability. On April 27, 1971, two concurrent tests turned up a design error that was not detectable by standard tests. A safety injection signal was given during a simulated station blackout. These two conditions were not normally tested concurrently. When station blackout occurs, the voltage is momentarily lost on all 480-V safeguard buses (Nos. 14, 16, 17, 18). Instrument bus 1B (1 of 4) is tied to 480-V bus 14 [with a backup supply from the main lighting panel (see FSAR p. 8.2-5 and Figure 8.2-5)]. Instrument bus 1B feeds the concentrated boric acid tank level channels LC 102 and LC 172 (NSIC 66927, RO-71-02). Therefore, when the 1B bus voltage was lost, a downscale indication was shown on LC 102 and LC 172. Low-level indication from these two channels actuates an interlock which prevents the concentrated boric acid tank outlet valves (826A, B, C, and D in FSAR Figure 6.2-1) from opening, thereby preventing all direct flow from the chemical volume control system (CVCS) concentrated boric acid tanks to the suction of the SI pumps. The power supply for LC 102 and LC 172 was changed to a battery-supplied uninterrupted bus.

The design purpose of the concentrated boric acid injection system is to provide protection from a main steam pipe break and potential return to criticality and power. During hot functional testing prior to initial criticality, the fact that the momentary loss of bus 14 would cause this problem was not discovered because two different tests were involved.

4.5.2 Trends and safety implications of reportable events in addition to those categorized as significant

An overview of all the reportable events is given in Table 4.7. The system categories are those used in coding these events, and the number of times per year that a system is involved in a reported failure is tabulated.

The most frequently reported systems are high-pressure safety injection (HPSI) (24), main cooling (23), chemical and volume control (19), reactor protection (16), and emergency electric power (14). Fourteen of the twenty-four safety injection/HPSI failures were caused by recurring failures during startup tests of the HPSI pumps powered by the emergency buses (detailed in Sect. 4.5.2.2.1). The CVCS failures did not begin occurring until 1974;

Table 4.7 Summary of Systems Involved in Reportable Events by Year

	System	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	Total
A.	Chemical and Volume Control	–	–	1	–	–	3	4	2	6	2	2	20
B.	Component Cooling	–	–	–	–	–	–	–	1	–	–	–	1
C.	Condensate Purification	–	–	–	–	–	–	–	–	–	–	–	0
D.	Condenser Cooling	–	–	–	–	–	–	–	–	–	–	–	0
E.	Containment	1	2	–	–	–	–	–	–	–	1	2	6
F.	Containment Air Cooling	–	–	–	–	–	–	–	–	1	–	–	1
G.	Containment Filtering	–	–	–	–	–	–	1	–	–	–	–	1
H.	Containment Hydrogen Control	–	–	–	–	–	–	–	–	–	–	–	0
I.	Containment Isolation	–	1	–	–	–	1	–	1	–	–	–	3
J.	Containment Purge	–	–	–	–	–	1	–	3	–	–	–	4
K.	Containment Spray	–	–	–	–	–	–	–	–	2	–	1	3
L.	Core Reflooding	–	1	–	–	–	2	–	1	–	–	1	5
M.	Electric Power	–	–	–	–	2	2	–	1	–	–	1	6
N.	Emergency Cooling/LPSI	1	–	–	–	–	–	–	–	–	–	–	1
O.	Emergency Electric Power	–	–	1	1	–	1	3	2	2	2	2	14
P.	Engineered Safety Features	–	1	–	–	–	–	1	3	2	–	6	13
Q.	Fire Protection	–	–	–	–	–	–	–	–	–	1	–	1
R.	Hydraulic	–	–	–	–	–	–	–	–	1	–	–	1
S.	Main Cooling	–	2	–	–	–	3	4	2	5	2	5	23
T.	Pneumatic	–	–	–	–	–	–	–	–	–	–	–	0
U.	Radiation Monitoring	–	–	–	–	–	–	–	1	–	–	–	1
V.	Reactor Control	–	–	–	1	1	–	1	1	5	–	1	10
W.	Reactor Protection	1	–	–	3	1	–	4	7	–	1	–	17
X.	Safety Injection/HPSI	–	1	2	–	3	3	4	2	5	–	4	24
Y.	Secondary Cooling/Aux FW	–	1	–	–	1	1	1	–	–	–	–	4
Z.	Secondary Cooling/Feedwater	–	–	1	–	4	1	1	–	–	1	3	11
AA.	Secondary Cooling/Steam	–	–	–	1	1	1	2	–	–	–	–	5
BB.	Service Water	–	1	–	–	–	–	–	–	1	–	2	4
CC.	Shutdown Cooling	1	–	–	1	–	1	–	2	1	–	1	7
DD.	Waste Disposal	–	–	1	3	–	1	–	–	–	–	–	5
EE.	Ventilation	–	–	–	–	–	–	–	1	–	–	–	1
FF.	Reactor Internals	–	–	–	2	–	–	1	2	–	–	–	5
	TOTAL[1]	4	10	6	12	13	21	27	32	31	10	31	197

[1]More than one system category was often assigned to each event. Thus, the totals exceed the actual number of events reported.

since then they have averaged over three per year. Fourteen of the nineteen CVCS failures involved cracks in piping and associated welds, all apparently caused by vibrations of the charging pumps.

4.5.2.1 Causes of reported events. Causes of the 183 reported events are shown in Table 4.8. Only one cause was attributed to each event.

Inherent failures account for 44% of all reported events. Roughly 50% of these were equipment failures caused by age and normal wear, and the other 50% had undetermined causes, which means that nearly a quarter of all reported events had undetermined causes. These events with undetermined causes represent a significant part of the operating experience data base at Ginna and should not be simply ignored by labeling them "inherent failure" with cause unknown.

The 30 administrative errors were generally attributed to inadequate or nonexistent procedures or training and inadequate records and administrative control. Note that preoperational errors, design and installation errors, account for nearly 23% of reported events. Greater care taken during construction can significantly reduce failures during operation. Also, note that failures attributed directly to operator error were not reported until 1975 and then averaged 2.2 per year thru 1979. The LER reporting system formally implemented in 1975, introduced new cause categories for reporting purposes — operator error was one of these. However, all human errors — administrative, design, fabrication, installation, maintenance, and operational — accounted for 56% of the total number of reported events. Each year they consistently accounted for about one half of all events reported. No significant trend in the reporting of human-induced errors is found.

Table 4.8 CAUSES OF REPORTED EVENTS

CAUSE	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	TOTAL
A. Administrative Error	-	3	1	3	2	5	6	3	2	1	5	31
B. Design Error	-	-	1	-	2	1	1	6	2	-	3	16
C. Fabrication Error	-	1	1	1	-	-	-	1	3	-	1	8
D. Inherent Failure	2	3	4	6	5	10	12	14	9	4	11	80
E. Installation Error	-	4	-	-	2	3	4	4	3	1	4	25
F. Lightning	-	-	-	-	-	-	-	-	-	-	-	0
G. Maintenance Error	-	-	-	2	1	3	1	2	3	-	-	12
H. Operator Error	-	-	-	-	-	-	3	-	4	3	1	11
I. Weather	-	-	-	-	-	-	-	-	-	-	-	0
Total	2	11	7	12	12	22	27	30	26	9	25	183

4.5.2.2 <u>Trends in conditionally significant events</u>. Though only two events in Ginna's life were labeled "significant," many events portended problems that possibly could adversely affect the safe operation of the plant. Eighty-four of these were labeled conditionally significant (C) - and are summarized and discussed in Table 4.9 and text that follows. Two of these conditionally significant events were discussed earlier in Sect. 4.5.1.3.

The thirteen C3 events — offsite radiation releases are discussed separately in Sect. 4.6. The fifteen C4 events — design, manufacturing, or installation deficiences, might have been prevented by diligently applied, routine quality assurance measures. The six C5 events — forced outages lasting longer than 10 d, all occurred in the second half of Ginn'a life. Four were shutdowns to plug failed steam generator tubes (AO 75-07, RO 75-13, 76-15, and 79-22). Both C6 events — engineered safety feature (ESF) actuation, did not involve any loss of coolant; event 70-01 was an SI subsystem failure in a test, and event 70-01 called on the SI because of overcooling of the primary system by the auxiliary feedwater in this loss-of-offsite-power incident.

The 34 C7 events — recurring failures, are discussed in detail in Sect. 4.5.2.2.1. The 12 C8 events — other conditionally significant events, are discussed in Sect. 4.5.2.2.3 and 4.5.2.2.4.

<u>4.5.2.2.1 Recurring Failures — C7</u>. Failures recurring over the life of the plant were labeled C7. Four problems are identified with a historical recurrent rate greater than or equal to 0.5 per year; and a fifth area, leaking snubber reservoirs (Table 4.10) is included because of a recurrence rate of 1 per year for the last 3 years.

Table 4.9 CONDITIONALLY SIGNIFICANT EVENTS AT GINNA[1]

Conditionally Significant Categories	Number of Events[2]											
	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	Total[3]
C1												
C2												
C3		1	3	3		3	1	1			1	13
C4			1		2	2	1	4	2		4	16
C5							2	2			2	6
C6		2			1							3
C7	1	1	1	1	2	2	4	5	9	3	5	34
C8	1		1		2	2	2	1		1	2	12
Total[3]	2	3	6	4	7	8	10	13	11	4	14	84

1. See Table 3.4 for conditionally significant criterion descriptions.
2. Blanks indicate 0.
3. The two totals here are equal, unlike for the significant events in section 4.5.1. No attempt was made here to differentiate the total number of conditionally significant events from the total number of times the CS category was used (84), as was done for the significant events. More than one of the conditionally significant categories could be assigned to each event.

Table 4.10 Recurring Failures - C7

Event	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	Total
SI Pump(s) - failure to start	–	–	–	–	2	2	3		2		1	10
Steam generator tube failures	–	–	–	–	–	–	2	2	2	1	2	9
Control Rod Drive Malfunctions												8
Rod (s) fail to drop	1	–	–	–	–	–	–	–	–	–	–	
Rod(s) fail dropped	–	–	–	1	–	–	1	3	–	–	–	
Automatic control fails	–	–	–	–	–	–	–	–	2	–	–	
Diesel Generator(s) fail to start	–	–	1	–	–	–	–	–	2	1	1	5
Snubber Reservoir Leaking	–	–	–	–	–	–	–	–	1	1	1	3
Total C7 Events	1	1	1	1	2	2	5	5	9	3	5	35

All ten of the SI pumps' failures to start were caused by the emergency bus breaker problems discussed in Sect. 4.5.2.2.2.

Steam generator tube failures became a problem in 1975 concurrent with the switch to all volatile chemical control (system modification 74-45, started during shutdown November 2, 1974); tube thinning and corrosion problems have not yet been solved.

None of the eight control rod drive mechanism (CRDM) malfunctions reported over Ginna's life compromised its safety function. Water from a leaking fitting in a feedwater flow indicator entered the power cabinet and grounded out the control power, causing rods G-5 and G-9 to drop on June 23, 1972 (72-09); the cabinet was dried and the leaking fitting was replaced. Rod G-5 did not move with its bank during a rod control system test in 1975, though the rod drop was not impeded upon a trip signal; the lift coil had an open circuit, preventing normal driving (UE75-08). Then in April and July of 1976, rods G-5 and G-9 dropped: the alarm indication was a stationary "A" regulation failure, but in both instances no cause could be determined and the rods tested normally. Based on similar instances at other Westinghouse facilities, certain electronic components in the rod control circuitry were replaced with upgraded components (76-13 and 76-18). In August of this year rods G-3 and G-11 partially dropped into the core. This was the first occurrence with these two rods, but the fifth such occurrence associated with the 2BD power cabinet. Again, no abnormalities were found (76-21). Polarity checks conducted in January, 1978, indicated that 27 movable coils and 2 lift coils had polarity reversals, resulting in too short an interval between stationary coil mechanism latch-in and movable coil mechanism drop-out. Since these polarity reversals were corrected, Ginna has had no further rod drop problems.

Either of the two station diesel generators failed during tests only five times over Ginna's life; they never failed on demand. This failure rate of 0.25 per year per diesel is remarkably low.

Steam generator or main steam line snubbers were found with low or no reservoir level in 77-13, 78-06, and 79-20. Excessive seal leakage on these vibration-damping components had not yet been remedied.

Finally, note the trend over the years at Ginna of increasing numbers of recurring failures. These five recurring failures over the period 1975 through 1979 occurred at a combined rate of over 5 per year. This fact focuses future maintenance efforts on recurring failures.

4.5.2.2.2 Emergency bus breaker failures. Breakers in three positions marked 1, 2, and 3 in Fig. 4.3 failed 14 times in Ginna's life during tests of safety injection pump "C" (see FSAR Fig. 8.2-4 for details), as shown in Table 4.11.

Table 4.12 lists the 14 breaker failures between the diesel generator "B", buses 14 and 16, and HPSI pump "C".

From this series of failures, apparently the conservative design of three 50% HPSI pumps is partially nullified by the unreliability of the circuit breakers. From 1973 through 1979, the three breakers failed an average of two tests per year. The HPSI pumps are tested monthly (TS 4.5.2.1) — this is the relevant test for breaker position numbers 1 and 2. The diesel generators are started in a monthly test (TS 4.6.1.a) — this is the relevant test for breaker position 3 only. The diesel generators are also tested with a simulated

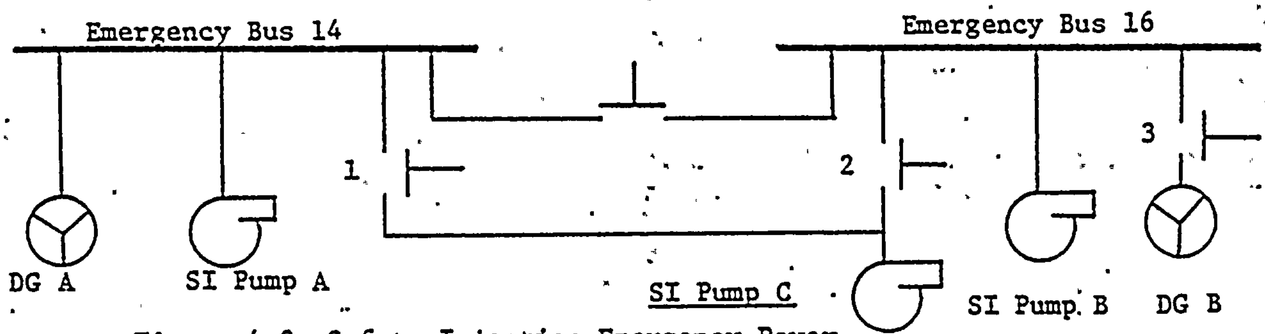


Figure 4.3 Safety Injection Emergency Power Supply

Table 4.11 Summary of Failures by Year

	Reportable Event Numbers						Total
Breaker No. 1				77-01 77-07		79-01	3
Breaker No. 2	73-03 73-04	74-04 74-14	AO 75-02 AO 75-03 AO 75-05				7
Breaker No. 3			UE 75-05	77-19	78-07	79-18	4
TOTAL							14

Table 4.12 Emergency Bus Breaker Failures

Report Number	Event Date	Event Description and Problem Solution
73-03	6/11/73	SI pump "C" fails to start in test. Problem not found (Reportable event 73-04).
73-04	6/12/73	SI pump "C" fails to start in test from bus 16. The cell switch in the breaker cubicle (breaker position No. 2 in Figure 4.2) was bent. Breaker was adjusted.
74-04	4/6/74	SI pump "C" fails to start in test from bus 16. The trip bar switch in the breaker control relay was out of adjustment. It was adjusted and the capacitor in the control circuit was replaced.
74-14	8/7/74	SI pump "C" fails to start from bus 16 in test. No cause found.
AO 75-02	2/14/75 (Report date)	SI pump "C" fails to start from bus 16 in test. Weak spring in breaker found and replaced.
AO 75-03	2/14/75 (Report date)	SI pump "C" fails to start from bus 16 in test. Loose wire in breaker found and repaired.
AO 75-05	2/26/75 (Report date)	SI pump "C" fails to start from bus 16 in test. Lockout solenoid was sluggish, did not complete its stroke. No repair done.
UE 75-05	9/15 (Report date)	Breaker in position No. 3 fails open during test of DG "B". No cause found.
77-01	1/3/77	SI pump "C" fails to start in test from bus 14. The contact assembly in the breaker in position No. 1 (a W model DB-50, 600 V AC) was replaced.
77-07	6/29/77	SI pump "C" fails to start in test for bus 14. The entire breaker in position No. 1 (W DB-50, 600 V AC) was replaced.
77-19	9/14/77	Breaker in position No. 3 fails open during test of DG "B". Secondary contact finger was bent. A new secondary contact section was installed. The bent secondary contact finger is believed to be caused by improperly returning the breaker to service after being in the "rolled out" position. Auxiliary operators retrained.
78-07	8/16/78	Breaker in position No. 3 fails closed and when finally opened it fails open during test of DG "B". Adjusted the W model DB-75, 600 V AC 3000 amp breaker.
79-01		SI pump "C" fails to start in test from bus 14. Breakers intermittently failed open. Alarm switch in the breaker's control wiring was replaced.
79-18		Breaker in position No. 3 fails open in test of DG "B". Though a relay (W BFDL-22) was suspected, no change was made.

SI signal during every refueling outage (TS 4.6.1.b) — this is the relevant test for all three breaker positions. Therefore, each breaker is tested 13 times per year, or, over 7 years, 91 times. Assuming tests were done per Technical Specification frequency and assuming one refueling outage per year, test failure probabilities for 1973 through 1979 are:

Breaker position	Total No. of failures in 7 years	Total No. of tests in 7 years	Test failure probability (%)
1	3	91	≈3
2	7	91	≈7
3	4	91	≈4

The trend of the data indicates that this problem is not yet solved.

4.5.2.2.3 Common cause failures. Table 4.13 lists ten common cause. failures that were reported at Ginna. Two failures are considered significant events (71-02 and UE 75-03) and are described in detail in Sect. 4.5.1. Two control rods dropped when water leaked into their shared cabinet on June 23, 1972, and again on March 5, 1975. That problem was solved by welding a leak-tight cover to that cabinet. Three common cause events (72-09, UE 75-01, and UE 75-03) turned up during operations, and three common cause failures (78-06, 79-12, and 79-15) were identified during inspections. With the exception of 71-2, discussed in Sect. 4.5.1, none of these common cause failures significantly affected the safety of the plant.

4.5.2.2.4 Other conditionally significant events. Twelve conditionally significant events, flagged C8, portended significant problems but did not qualify as S or C1-7 events.

Table 4.13 Common Cause Failures

Report Number	Event Date	Significance Category	Event Description and Problem Solution
71-02	4/27/71	C4	Redundant level channels in BAT fail down-scale due to momentary loss of instrument Bus 1B voltage during coincident station blackout and SI tests, causing BAT-to-SI valves to close. Level channels' power supply switched to DC source.
72-09	6/23/72	C7	Two control rods drop due to a water leak into a shared power cabinet.
73-11	12/21 (report date)	C4, C8	Both motor-driven auxiliary feedwater pumps airbound in test due to bubble in common header from condensate supply. Cause of bubble uncertain. The pumps were vented every 8 hours until corrective action (unknown) was taken.
74-11	6/26/74	C8	Five of the eight solenoid valves which control the MSIV's fail in test due to over-heating. Could not close MSIV's. Over-heating due to solenoids being left energized too long in test. No corrective action taken.
UE 75-01	3/5/75	C7	See 72-08 above. Cover welded onto cabinet.
UE 75-03	6/23/75	S2	Both MSIV's fail closed due to steam im-pinging on leading edge of valves' disks. Valves ground down so they sit out of steam flow path.
78-06	7/13/78	C7, 8	Two snubbers on main steam system fail due to constant vibration of system. No corrective action was taken other than re-placing failed scrubbers with tested spares.
79-12	3/27/79	C8	Six anchor bolts for piping supports for "safety equipment" not up to specifications. Corrective action see IE Bulletin 79-02.
79-15	7/24/79	C4, 8	Ten piping supports not properly installed in containment spray, residual heat re-moval, and service water systems. Supports re-worked and re-analyzed (See IE Bulletin 79-14).

Five of the C8 events were common cause failures and were discussed in Sect. 4.5.2.2.3. Of the other seven, two similar events were reported in 69-02 and 74-06, both involving a failed valve in the 10-in. line between the containment sump and the residual heat removal pumps. On December 16, 1969, both disks of a valve were bowed outward from internal pressure caused by heatup of water trapped inside the valve by ambient reactor coolant water. Similarly, on April 22, 1974, water trapped between two valves in the same line was heated, and the consequent pressure seated the valve so tightly that it would not open on a signal from the control room. Operating procedures should preclude this problem.

On June 30, 1971, when a third condensate pump was put in service, a series of failures occurred that led to a condensate line water hammer and a reactor scram (71-04). The safety of the plant and public was not endangered, but the event sequence deserves some attention. Increased condensate pump outlet pressure, resulting from the added condensate pump, caused the hydrogen cooler temperature control valve superstructure to buckle, and the valve failed closed. The hydrogen cooler temperature rose, and the normal condensate bypass valve consequently closed. This caused an emergency feed valve to open, supplying relatively cold condensate directly to the main feedwater pumps' (MFWP) suction, which induced severe vibrations in the MFWP's suction line and tripped MFWP A. MFWP B was tripped manually, tripping the turbine and then the reactor in the normal logic sequence. This is the only condensate line water hammer reported at Ginna.

On July 22, 1973, a water hammer in the feedwater line to the B steam generator damaged several supports and insulation at various locations along the line (73-06). Failure of a small pin which prevents rotation of the control plug on the B feedwater control valve caused the control plug to separate from the threaded stem of the valve, inducing rapid flow variations, causing the water hammer. According to a RG&E letter to the AEC Directorate of Licensing dated August 21, 1973, the valve manufacture was consulted and subsequent investigation revealed that the original stem on the B feedwater control valve had been damaged during construction. A replacement stem had been installed, but the recommended torquing procedure had not been followed during reassembly of the new plug on the stem. Had the plug been torqued properly, the holding pin would not have failed. RG&E has since stated that a design change has been made in this valve operator.

On June 26, 1974, following maintenance on the main steam isolation valves, the four solenoids on each of the MSIVs were left in the energized position. During subsequent testing, five of the eight solenoids failed; but, after they were allowed to cool, they operated correctly. Inadequate maintenance procedures were responsible for this failure (74-11).

Both discharge valves for one of three 50%-capacity SI pumps were found closed in a test on June 25, 1975 (AO 75-10). The recognized unreliability of the HPSI emergency bus breakers (Sect. 4.5.2.2.2) makes this operator error more significant.

The administrative oversight reported in UER 75-06 is significant because it represents a deficiency in quality assurance procedures.

A modification was made in which a steam generator level signal input was transferred to a bus that was already part of the lo-lo steam generator level protection system and the reactor trip logic for steam flow—feedwater flow mismatch. On discovery of the error only hours later, the system was returned to its original configuration.

An actuator in a control room ventilation damper was originally installed incorrectly and was not discovered until March 8, 1976 (76-11). Previous tests had assumed that actuators had been properly installed. Proper tests verify functionability of a system and not just its components.

4.6 Events of Environmental Importance

A summary by year of the total radioactivity released from Ginna is shown in Table 4.14, organized by airborne and liquid effluents and by off-site shipments of solid waste. Because of varying reporting requirements in 1969, 1970, and 1978, some data are not available. The 1979 data has not been published as of April 1981. An overall increase over 10 years is apparent in the activity released as airborne tritium and as airborne noble gases, with the notable exception of 1972 which had the largest releases of noble gases.

Note the spike of mixed fission and activation products released via the liquid pathway in 1970. Approximately 88% of this mixed activity was discharged in March and April 1970 because of increased activity in the primary coolant leaking from the plunger leakoffs of the charging pumps. Increased primary loop activity detected at 100% power in March was first evidence of leaky fuel. On March 25, during full power operation, and

Table 4. 14 Summary of Radioactivity Released From Ginna

EFFLUENT (CURIES)	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979
AIRBORNE:										
Total Noble Gases	1.00E+01[1]	3.20E+01[1]	1.18E+04	5.76E+02	7.57E+02	1.04E+04	5.52E+03	3.20E+03	9.27E+02	6.84E+01
Total I-131	NA[2]	NA	2.85E-02	5.31E-04	2.82E-04	NA	3.17E-02	2.54E-02	1.02E-02	1.91E-02
Total Halogens (Including I-131)	NA	NA	3.00E-02	9.8 E-04	4.46E-04	2.69E-02	3.71E-02	2.63E-02	NA	2.98E-02
Total Particulates ($T_{1/2}$>8 day)	5.00E-02[1]	1.70E-01[1]	7.81E-05	3.3 E-05	4.19E-05	2.00E-02[1]	8.95E-05	7.03E-05	1.43E+04	5.50E-03
Total Tritium	NA	NA	8.8 E-03	1.15E+00	3.56E-01	NA	2.36E+01	5.00E+01	4.38E+01	5.67E+00
LIQUID:										
Total Mixed Fission & Activation Products	1.0 E+01	9.0 E-01	3.0 E-01	7.5 E-02	1.00E-01	4.21E-01	6.90E-01	6.47E-02	6.07E-02	1.27E-01
Total Tritium	1.08E+02	1.54E+02	1.19E+02	2.86E+02	1.95E+02	2.61E+02	2.42E+02	1.19E+02	2.42E+02	1.26E+02
Dissolved Noble Gases	NA	NA	ND[3]	3.03E-04	ND	NA	3.93E+02	ND	NA	9.05E-03
SOLID:										
Total	4.64E+00	4.73E+01	1.41E+03	5.99E+02	6.14E+02	1.07E+02	9.78E+01	6.90E+02	6.27E+02	3.92E+02

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[1] From NUREG-0521, Radioactive Materials Released from Nuclear Power Plants.
The rest of the data was compiled from RG&E's semi-annual and annual operating reports.

[2] Not available.

[3] None detected.

again on April 24, during startup after a month-long scheduled maintenance outage, the maximum concentration of liquid released (2.4 E-07 μCi/cc, excluding tritium) exceeded the technical specification limit of 1.0 E-07 μCi/cc but did not exceed the maximum permissible concentration (MPC) isotopic limits; isotopic analysis showed Iodine-131 to be the only isotope discharged at a concentration greater than one-tenth of its MPC. (This value for I-131 could not be found.) At no time did the concentration of tritium in the discharge canal exceed the technical specification limit of 3 E-03 μCi/cc.

Four radiation exposure incidents and nine offsite releases are summarized in Table 4.15. Thirteen of these appear in Table A.2, all marked C3. One incident was reported in a letter to the AEC on June 11, 1971, in which two concurrent yet independent leaks were surmised to have exceeded technical specification release limits (71-03). Another incident was reported in a letter to the AEC on February 9, 1973: a maintenance error on the RCS letdown line caused a 47-h unplanned release of radioactive gases on June 10, 11, and 12, 1972 (72-07). Note that 1972 airborne noble gas releases (Table 4-14) were the highest ever at Ginna, exceeding 1971 and 1973 releases by over a factor of 100.

The four overexposures all resulted from administrative errors and lax health physics procedures. No equipment failures caused overexposures. Of the nine offsite releases, five occurred during operations and four during maintenance. The CVCS was involved in all but one of these incidents.

Table 4.15 Radiation Over-exposures and Releases

Event Report	NSIC Number	Event Date	Description
70-07	49878	8/19/70 (report date)	A chemist received 4.2 rems in a 2-month period. Procedures shortened.
Letter to AEC (6/11/71)		6/1-6/5/71	Xenon leak from a pressure control valve (on the volume control tank) to the auxiliary building. Waste evaporator vent concurrently vented I-131 to auxiliary building during shutdown of charging system. A continuous recording iodine monitor was installed on the stack vent. Amounts released uncertain.
71-04	64850	7/21/71 (report date)	A steam fitter received 3.09 rems in one quarter.
71-05	67990	10/6/71 (report date)	6.7 mCi of I-131 released to atmosphere through the plant vent during a 17 hr. period. Monitor failed. Boric acid concentrator drain flushed.
72-05	70029	4/20/72	I-131 releases in excess of Tech. Specs. coincident with processing of reactor coolant from the C-holdup tank through the boric acid evaporator.
Letter to AEC (2/9/73)		6/10-6/12/72	Krypton-85 released for 47 hours (unplanned) due to RCS drain valves left open during maintenance. Maximum release rate was 8.4 E-04 Ci/sec; maximum concentration was 2.0 E-05 μCi/cc.
72-11	77411	12/28/72 (report date)	228 Ci released in 24 hours, 6% of Tech. Spec. limit, during the first spent resin removal operation (from spent resin tanks). Procedures modified.
Notification No. 107 Letter to AEC	89263 90662	2/20/74 (report date) 3/25/74	40 contractor maintenance personnel received over 3 rems in January while repairing leaks in the spent fuel pit.
74-08	94198	5/11/74	Leak in one of two mixed-bed demineralizers (CVCS) to auxiliary building. The highest release rate was 1.63% of Tech. Spec. limit.

Table 4.15 (continued) Radiation Over-exposures and Releases

Event Report	NSIC Number	Event Date	Description
74-10	94522	6/21/74 (report date)	A primary sample system flow indicator was reinstalled incorrectly, leak filled chemical drain tank, high level alarm failed. In 2 hours, 0.9 Ci of noble gases was released.
Letter to NRC	109653	5/19/75 (report date)	One maintenance worker's uptake above limits. The dose to the lungs from Co and Zr was 9.6 rems for the first year. Improper use of masks due to administrative error. $10,000 fine proposed.
Letter to RGE	105548	8/23/75 (report date)	
76-19	115728	7/16/76 (report date)	11 low level releases were made from the main water treatment demineralizer (CVCS) neutralizing tanks without the continuous use of a gross activity monitor due to a design modification omission. Maximum concentration released was 2.25 E-07 µCi/cc.
79-02	146698	1/4/79	Cracked charging pump cylinder leaks to plant vent. Amount unspecified.

4.7 Evaluation of Operating Experience

Reactor availability at Ginna has been among the industry's highest, averaging 78.1% over the ten full years of operation. Equipment failures dominated the shutdowns, accounting for 73% of the total causes; however, equipment failures accounted for only 44% of the licensee event reports and their predecessors (hereafter referred to as reportable events), a difference that will be discussed in more detail in this section.

Whereas trends in shutdowns and power reductions generally followed trends in reactor availability, reportable events had two distinct phases (Fig. 4.1). Through 1973, reportable events were generated consistently at a rate of about ten per year; then, in 1974, these rates doubled and averaged about 25 per year through 1979. The exceptional year, 1978, with only nine reportable events, set the record for capacity factor at 78.2%. This is the only reason found for the low number of reportable events in 1978.

Two points of view provided by the analysis of shutdowns and power reductions and of the reportable events complement one another in assessing the operational history of the plant. The analysis of the shutdowns is most useful in identifying accident sequences, whereas the analysis of reportable events is most useful in providing specific information and in identifying specific failures *within* accident sequences. The distinction arises from the

fact that the plant is designed to shut down *whenever* an unsafe operating condition exists, such as when an - accident sequence has been initiated; whereas an event report is generated to fulfill a specific technical specification requirement, such as a limit bounding the accident sequence or a component failure that limit the plant's capability to respond to an accident sequence. Reportable events tend to be more component oriented, identifying single events or occurrences. Whereas Ginna has experienced 118 shutdowns, reportable events totaled 181. Reportable events cover a wider range of events, including causes not categorized in the shutdown tables (such as test-induced failures during a shutdown and all design and fabrication errors). Reportable events cover all actual failures, including releases, major outages and major damages, and all SIs, plus potential failures identified through operations or reanalyses with impact on plant safety. Nearly one in four reportable events had undetermined causes (Sect. 4.5.2.1).

Conclusions about the overall operational safety of the plant do not vary when taken from shutdowns and from reportable events. Safety significant events consistently are identified in both ways. The two points of view are complementary in individual cases. For example, the MSIV spurious closures were reported by shutdowns on both June 6 and 23, 1975, but only by reportable event on the latter date, when the cause was surmised and corrective action was taken. For review purposes, the complementary aspects of the two points of view are apparent from the interwoven nature of Sects. 4.4 and 4.5 on shutdowns and reportable events, respectively.

One result of the reportable events different from that of the shutdowns concerns preoperational sources of error; design and installation errors account for 23% of all reportable events (Sect. 4.5.2.2). Shutdowns do not acknowledge this source of error explicitly.

Human factors cause a higher percentage of reportable events, by a factor of 2, than of shutdowns. This difference in human causes — 56% of the reportable events to 27% of the shutdowns — is discussed below.

For reportable events, human factors include administrative, design, fabrication, installation, maintenance, and operator errors — in short, all those causes listed in Table 4.8 except inherent failures, lightning, and other weather. For shutdowns, human factors include maintenance or testing, administrative, or operational errors (item I.3 of Table 4.2). This categorization scheme itself tends to attribute human factors as cause to more reportable events than to shutdowns. In spite of this inherent bias in the categorization scheme; same valid conclusions can be drawn.

Operator errors as reported causes for shutdowns were reduced to one in 1972 and none thereafter, and no evidence exists in the shutdown reports to suggest that operator-error-induced shutdowns occurred after 1972 but were reported as having other causes. Human factors as defined above caused shutdowns throughout the plant's life, though again the first years of operation were the worst here, with over one-third (10 of 28) occurring in 1970 alone.

Reportable events, on the other hand, attribute over half of their causes to human factors. No reportable events were attributed to operator errors through 1974; however, in 1975 through 1979, they averaged over two per year (Table 4.8). Human factors averaged 7 per year through 1974 and from 1975 through 1979 averaged 13.4 per year. This increase, however, simply reflects the overall trend of increased numbers of reported events (Fig. 4.1); human factors accounted for roughly 50% of all reportable event causes each year from 1970 through 1979. The difference in causes for shutdowns and reportable events indicates that while human factors can frequently incapacitate components and can influence tests and maintenance operations that in turn generate reportable events, they are much less likely to incapacitate systems that in turn generate reactor shutdowns. Also, as indicated before, 27% of reportable events were generated by preoperational causes (design, fabrication, and installation errors) all of which are caused by human factors.

Tests helped identify many potential safety hazards at Ginna. Tests uncovered significant faults as reported in (1) AO 75-10, when both of SI pump C's outlet valves were found closed, and (2) 76-11, when 25% of the control room ventilation filter capability was found inoperable. Concurrent tests run on April 27, 1971, uncovered a highly significant fault in the emergency power supply for all four boric acid tank (BAT)-to-SI valves. Never did failures resulting from tests jeopardize the safety of the plant or nullify the design capability of a safety system.

Inspections turned up the recurring main steam and steam generator snubber leak problems as reported in 77-13, 78-06, and 79-20. Inspections also found potential common cause failures involving safety equipment anchor bolts (79-12) and piping supports (79-15). System modifications did not cause problems except for one in 1975 when an inspection found that no separation of redundant function was provided for a modified safety system (UE 75-06).

Although 1979 was a very good year at Ginna, recurring problems with condenser and steam generator tube failure are not yet solved. Problems here have consistently caused shutdowns from 1975 through the present. Condenser tube failures were identified through power reductions; reportable events never were involved.

Other recurring problems leading to shutdowns seem to be solved: EH turbine-generator governor control problems, not a factor since May 22, 1976; instrument bus inverter failures, not a factor since June 3, 1976; manual feedwater control problems, never a factor except in the first 8 months of operation; main feedwater pump impeller failures, no occurrence since September 10, 1976; MSIV spurious closures, not a factor since June 23, 1975; pressurizer spray valve packing leaks, not a factor since September 5, 1972; and turbine blade failures, not a factor since August 7, 1976. Control rod malfunctions, though they have not caused a shutdown since November 17, 1977, have not been resolved.

No environmental considerations ever constrained operations at Ginna. Only once, in March and April 1970, did the liquid effluent require isotopic analysis and comparison with maximum permissible concentrations before being released to the discharge canal.

REFERENCES

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2. *Operating Units Status Report Licensed Operating Reactors*, NUREG-0020 May 21, 1974 issue through January 1980 issue, Vol. 4, No. 1.
3. *Nuclear Power Plant Operating Experience During 1973*, OOE-ES-004, U.S. Atomic Energy Commission, December 1974.
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6. *Nuclear Power Plant Operating Experience 1977*, NUREG-0483, U.S. Nuclear Regulatory Commission, February 1979.
7. *Nuclear Power Plant Operating Experience 1978*, NUREG-0618, U.S. Nuclear Regulatory Commission, December 1979.
8. Semiannual Report No. 1, Rochester Gas & Electric Corporation, Ginna Station, Docket 50-244, July 13, 1970, p. 18.
9. Semiannual Report No. 4, Rochester Gas & Electric Corporation, Ginna Station, Docket 50-244, February 25, 1972, p. 5.
10. *Nuclear Safety*, Vol. 11(3), May—June 1970, p. 241.
11. *Nuclear Safety*, Vol. 11(4), July—August 1970, p. 328.
12. The block valve leaked enough to cause shutdowns, however, on December 14, 1972, and February 20, 1978.
13. *Nuclear Safety*, Vol. 15(1), January—February 1974, p. 88.
14. Semiannual Report No. 10, Rochester Gas & Electric Corporation, Ginna Station, Docket 50-244, (no date), pp. 3, 23.

Appendix A: Ginna

Part 1. Shutdown and Power Reduction Tables

Table A1.1
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1969)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	11-29	1.3	2		High condenser pressure. Valve to bistable left closed after calibration.	G	3	Steam & Power (HC)	I & C	N6.4
2)	12-3	5.1	20	Ltr 12-12-69	Double-low SG level. Suction valve to 1B MFWP mistakenly closed. (One shutdown rod failed to drop — see letter.)	G	3	Steam & Power (HH)	Valves	N6.1
3)	12-3	.8	20		Double low SG level. FW bypass valve was not manually reset.	G	3	Steam & Power (HH)	Valves	N6.4
4)	12-3	.7	20		Double low SG level. FW bypass valves were prematurely put in automatic.	G	3	Steam & Power (HH)	Circuit controllers/ interrupters	N6.2
5)	12-4	.8	24		Loss of both RCPs. Field from generator was manually removed before 4160 V busses were switched.	G	3	Reactor Coolant (CB)	Pumps	N5.1
6)	12-6	.7	22		Double low SG level. Excessive vibration of seal water differential pressure mercoid tripped MFWP.	A	3	Steam & Power (HH)	Pumps	[illegible]
7)	12-14	15.0	30		To check condenser hotwell "problem".	A	2	Steam and Power (HC)	Heat Exchanger	
8)	12-15	.7	2		Low seal water differential pressure to MFWP.	A	3	Steam and Power (HH)	Pumps	

Table A1.1
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1969)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
9)	12-16	36.9	2		Tech. Spec. requirement to shut down due to failure of safeguard valve MOV-850B to open fully during monthly surveillance test. Both valve disks were bowed out due to pressure internal to the valve.	A	1	Engineered Safety Feature (SF)	Valves	N1.2.1
10)	12-30	.4	1		FF/SFHH with low SG level. 1A AFWP was not available and steam driven AFWP was slow starting. 1B AFWP did not supply enough water for the load.	A	3	Steam and Power (HH)	Pumps	
11)	12-30	17.0	2		Tech. Spec. requirement to shutdown due to failure of valve on discharge of 1B motor-driven AFWP.	A	1	Steam and Power (HH)	Valves	

Table A1.2
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1970)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1-3	1.8	2		SI due to 2/3 signals of low pressure in B main steam line. During calibration of pressure channel 483 I&C workers closed a leaking valve in pressure channel 478. (Note: B train SI failed to operate. See Monthly Report #1, p. 52.)	B	3	I&C (ID)	I&C	N5.2
2)	1-3	15.7	2		MOV-8783 on SI line to B cold leg was found inoperable.	A	1	Engineered Safety Features (SF)	Valves	
3)	1-17	6.4	2		Steam leak thru gasket on turbine stop valve.	A	1	Steam and Power (HA)	Valves	N3.2
4)	1-17	1.6	2		Malfunction of the rod step counters.	A	1	Reactor (RB)	I&C	N2.1
5)	1-27	.4	50		Double-low SG level due to:		3			
					A. Turbine trip due to loss of FW governor pump pressure, plus	A		Steam and Power (HA)	Pumps	D2.3
					B. Manual FW control was too slow in responding.	C		Steam and Power (HH)	I&C	N6.1
6)	1-27	.5	50		Overtemperature delta T signal coincident with overtemperature delta T testing — cause undetermined.	B	3	I&C (IA)	I&C	N1.2.4

Table A1.2
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1970)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
7)	1-28	.9	23		Overtemperature delta T due to dirty contact on Channel IV delta T relay in 'B' reactor trip logic.	B	3	I&C (IA)	I&C	N2.1
8)	1-28	.3	40		FF/SFMM and lo SG level. FW valve A closed on high SG level and operator could not manually control the resulting level drop.	G	3	Steam and Power (HH)	I&C	N6.1
9)	2-22	.2	10		During 4160 V bus underfrequency tests, both busses were inadvertently tied together.	B	3	Electric Power Systems (EB)	Electrical Conductors	N6.2
10)	3-2	40	70		To repack the PZR spray valves.	A	1	Reactor Coolant (CB)	Valves	N1.1.1
11)	3-30	[illegible]	100		FF/SFMM and lo SG level. Operator could not manually control SG level oscillations subsequent to manual takeover of FW control.	G	3	Steam and Power (HH)	I&C	N6.1
12)	5-14	816	85		Following a 100% full power trip test and a turbine inspection, plant remained shutdown for:	B	1			
					a) Turbine repairs on #2 LP section blading (replaced all 10th stage blades and 4 11th stage blades; damage due to foreign object).			Steam and Power (HA)	Transformers	N1.1.4
					b) Replacement of all RTDs.			I&C (ID)	I&C	N2.1
				TWX 5-18-70 Ltr 5-22-70	c) MSIV-1B did not close upon manual signal from control room. The packing gland was adjusted and the valve closed.			Steam and Power (HH)	Valves	N1.1.4

Table A1.2
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1970)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
13)	6-18	14.9	0		SI relay terminal screws were loose, relay fluttered causing B trip breaker to open.	A	3	Engineered Safety Feature (SF)	Relays	N2.1
14)	6-19	.3	15		Double-lo SG level. While the FW was on manual control the turbine was intentionally tripped to prepare for EH governor repair. The resulting pressure transient caused the double-low level.	G	3	Steam and Power (HH)	16C	N6.1
15)	7-5	4	83	Ltr 7-15-70 Ltr 2-17-71	Power reduction. Turbine taken out to repair a steam leak in a turbine high pressure gage line. Other maintenance included: added packing to PZR spray valve PCV-431A, checked the stroke on PZR spray valve PCV-431B, and adjusted the turbine trip pilot valve.	B	5†	Steam and Power (HA)	Pipes, fittings	N3.2
16)	7-12	24.7	0		PZR pressure relief valve MOV-516 leakage rate required shutdown.	A	1	Reactor Coolant (CB)	Valves	N3.1
17)	7-25	2.6	27		FF/SFHH and low SG level due to HFWP A trip.	A	3	Steam and Power (HH)	Pumps	

Table A1.2
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1970)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
18)	9-5	14.5	37		Low SG level plus low FW flow occurred after turbine trip, which was done manually due to apparent turbine vibrations. Further investigation revealed the cause to be FW oscillation to the B SG due to loose packing of the 1-B MFW control valve.	A	3	Steam and Power (HH)	Valve	N1.1.4
19)	9-6	24	50		Repairs on:	A	1			
					A. EH control system.			Steam and Power (HA)	Mechanical function units	N2.1
					B. MFWP A suction relief valves.			Steam and Power (HH)	Valves.	N1.1.4
20)	9-30	335.1	100		To repair a leak on a reactor coolant temperature detector, plus: replacement of RTDs, repair of steam and water leaks, corrective maintenance of B MFWP, replacement of plunger in 1B and 1C charging pumps, replacement of connecting rod and repack seals in 1A and B phosphate pumps, installation of doors and gates in high radiation areas, change filters and reinforce filter framework on plant vent system HEPA filter bank, and found 5 fractured anchor bolts on the B RCP support and replaced all of the 56 anchor bolts for the steam generators and RCPs (see report with letter 12-21-70).	A	1	I&C (IE)	I&C	N3.1

Table A1.2
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1970)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
21)	10-29	1.6	100		Fault on trunk line #23 caused back-up distance relay to trip, causing turbine trip.	A	3‡	Electric Power Systems (EA)	Relay	N1.1.4
22)	11-1	.4	100		Generator automatic voltage regulator failure caused turbine trip.	A	3	Steam and Power (HA)	Circuit closers/ interrupters	N2.1
23)	11-2	7.6	100		Generator automatic voltage regulator failure caused turbine trip.	A	3	Steam and Power (HA)	Circuit closers/ interrupters	N2.1
24)	11-15	8.7	0		PZR relief valve leakage. Valve was repacked.	A	1	Reactor Coolant (CB)	Valves	N3.1
25)	11-23	1.3	0		Test for battery ground. Ground located on PCV-4316 control power.	B	3	Electric Power Systems (EC)	Batteries and chargers	N1.2.4
26)	12-12	38.7	0		Maintenance and repairs:	A	1			
					A. PZR power relief guard valve MOV-515 repacked.			Reactor Coolant (CB)	Valve	N3.1
					B. Plugged tubes in condenser A.			Steam and Power (HC)	Heat exchangers	N3.2
27)	12-31	76.8	0		Repair of:	A	1			
					A. Moisture separator reheaters.			Steam and Power (HC)	Heat exchangers	N1.1.4
					B. Condenser tube leaks.			Steam and Power (HC)	Heat exchangers	N3.2

*Unit left shutdown for scheduled maintenance.

†Scheduled test/unscheduled repairs encountered.

‡Turbine generator removed from service only. Reactor was controlled by boration with the D control rod bank at 140-180 steps out. Power level corresponded to 10(exp -7) amps on the intermediate power range channel.

Table A1.3
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1971)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1-13	12.3	100		Manual trip of MSIVs.			Reactor Coolant (CB)		D2.4
2)	2-3	9.9	100		To repack PZR spray valve PCV-431A.	A	2	Reactor Coolant (CB)	Valves	N1.1.1
3)	5-14	10.0	80		Low SG level. Repaired leak at 2B reheater turbine intercept valve flange.	A	2	Steam and Power (HH)	Valves	N3.2
4)	5-28	61.6	0		Repairs:	A	2			
					A. Charging pump filter drain leak.			Auxiliary Process Systems (PC)	Pipes, fittings	N3.1
					B. Repacked FW valves.			Steam and Power (HH)	Valves	N3.2
5)	6-30	7.7	100	Ltr 7-9-71	Upon boosting the MFWP suction pressure by manually placing the third condensate pump in service:	A	3			
					A. The hydrogen cooler temperature control valve superstructure failed under the increased condensate header pressure and the valve failed closed. The normal condensate bypass valve went closed due to high hydrogen cooler temperature. The emergency feed valve opened, supplying relatively cold condensate directly to the MFWP suction, inducing severe vibrations in the MFWP suction lines, tripping the MFWP A. The B MFWP was tripped manually, tripping turbine and then reactor in the normal logic sequence.			Steam and Power (HH)	Valves	N1.1.1

Table A1.3
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1971)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
5) (cont'd)					B. 1A-MSIV failed to close upon signal from control room. Solenoid operators were dirty.			Reactor Coolant (CD)	Circuit closers/ interrupters	N2.1
6)	7-5	7.6	100		Voltage was lost to control rods. Control rods dropped, caused low PZR pressure and reactor trip. A. Voltage regulator failed on the A rod drive generator set, plus B. A reverse current relay failed.	A	3	Reactor (RB)	Control rod drive mechanisms Relays Circuit closers/ interrupters	D4.3
7)	7-11	4.3	100		Low SG level and low FW flow caused by loss of instrumentation to B FW control logic (Inverter on instrument bus #1 failed) causing FW control valve to fail closed.	A	3	Steam and Power (HH)	I&C	D2.7
8)	7-13	10.2	100		SG A low level and low feed flow.		3	Steam and Power (HH)		D2.7
9)	7-14	1.4	100		Spurious indication of RCP breaker trip due to faulty coil.	A	3	I&C (IA)	Circuit closers/ interrupters	N2.1
10)	9-5	19.8	100		"When containment ventilation was reset the feedwater valves mistakenly closed, resulting in a SG low level/ low feed flow trip and a spurious safeguards actuation" (Semi-Annual Report #4, pp. 5).	A	3	Steam and Power (HH)	Circuit closers/ interrupters	D2.7

Table A1.3
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1971)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
11)	9-6	5.1	75		Spurious closure of MSIVs caused double low SG level.	A	3	Steam and Power (HB)	Circuit closers/ interrupters	D2.4
12)	9-6	.3	0		During investigation for battery ground, reactor coolant flow trip signal was given.	G	3	I&C (IA)	Circuit closers/ interrupters	N6.3
13)	11-12	17.3	0		To repack PZR spray valves.	B	2	Reactor Coolant (CB)	Valves	N1.1.1
14)	12-15	1.5	100		During testing of B reactor trip breaker, bypass breaker was not clear of breaker test switch and and turbine trip circuitry was completed.	G	3	I&C (IE)	Circuit closers/ interrupters	N6.3

Table A1.4
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1972)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1-4	<24	100			B	1			
2)	2-24	2.7	100		Blown fuse in EH control caused turbine governor valves to start to close. Operator then tripped turbine.	A	3	Steam and Power (HA)	Mechanical function units	N2.1
3)	3-8	.8	60		Hot shutdown due to "problem" with EH system.	A	2	Steam and Power (HA)	Mechanical function units	N2.1
4)	3-20	17.5	92		Hot shutdown to repair packing on PZR spray valve PCV-431B.	A	1	Reactor Coolant (CB)	Valves	N1.1.1
5)	4-14	.3	86		FF/SFMM plus low SG level due to accidental closing of MSIVs.	G	3	Steam and Power (HB)	I&C	D2.4
6)	6-23	6.2	36	Ltr 6-30-72	Rods G-5 and G-9 dropped into core. Water had leaked into CRDM cabinets, grounding out their power.	A	3	Reactor (RB)	Control rod drive mechanisms	N5.1
7)	7-13	2.5	83		Faulty speed channel card on turbine control system caused control valves to close.	A	2	Steam and Power (HA)	Circuit closers/ interrupters	N2.1
8)	7-14	1.6	83		FF/SFMM plus low SG level caused by MFWP trip due to loss of AC oil pump.	A	3	Steam and Power (HH)	Pumps	N1.1.4

Table A1.4
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1972)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
9)	7-27	4.5	83	Ltr 8-7-72	Manual trip due to loss of logic supply power for RCCS.	A	2	Reactor (RB)	Control rod drive mechanisms	N2.2
10)	7-28	2.6	30		Loss of 4160 V supply during synchronization procedures.		2	Electric Power Systems (EA)	Generators	
11)	9-5	7.0	83		PZR spray valve leak repair.	A	1	Reactor Coolant (CB)	Valves	N1.1.1
12)	12-14	17.1	82.5		PZR PORV block valve leak repair.	A	1	Reactor Coolant (CB)	Valves	N3.1

Table A1.5
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1973)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
8)	10-21	142	91	73-9 73-10 Ltr 10-31-73 Ltr 11-15-79	Loss of offsite power. When one of the four 115 kv transmission lines was down for construction of a new substation, a second 115 kv line, sagging due to the increased load (station output was 435 MWe net or 91% power), flashed over to an underbuilt 34.5 kv line and its circuit opened. The consequent 230 MWe power swing on the remaining two 115 kv lines caused them both to trip, causing complete loss of generator output ability. A turbine and reactor trip followed immediately.	H	3	Electric Power Systems (EA)	I&C	D2.2
9)	11-4	20	91-48		Power reduction. Repair on EH control system.	B	5	Steam and Power (HA)	Mechanical function units	N2.1
10)	12-11	55	91		Repair weld leak on charging pump discharge filter.	A	1	Auxiliary Process Systems (PC)	Filters	N1.1.3

Table A1.5
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1973)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1-12	31	83		To repair condenser tube leaks and condenser baffle plate.	A	1	Steam and Power (HH)	Heat exchangers	N3.2
2)	3-9	13	83-45		Power reduction. Tube repair on A and B condensers.	B	5	Steam and Power (HH)	Heat exchangers	N3.2
3)	6-9/10	12	83-45		Power reduction. Replaced outboard bearing shoes on 1A MFWP.	B	5	Steam and Power (HH)	Pump	N1.1.4
4)	7-22	203	95	73-6 73-7	FF/SFMM plus low SG level due to disconnected flow transmitter on B AFWP control valve.	A	3	Steam and Power (HH)	Pipes, fittings	N2.1
5)	7-31	3	20		Circulating water pumps tripped due to faulty trip circuit.	A	3	Steam and Power (HG)	Circuit closers	N2.1
6)	8-19	96	90-47		Power reduction. B MFWP bearing repair.	B	5	Steam and Power (HH)	Pump	N1.1.4
7)	9-10	24	90-50		Power reduction. B MFWP repair.	B	5	Steam and Power (HH)	Pump	N1.1.4

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Table A1.6
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1974)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/NSIC(N) Event Category
1)	1-1	2737	91		Blade failure on No. 2 LP turbine. Refueling was accomplished while maintenance overhaul of turbine was in progress.	A	1	Steam and Power (HA)	Turbines	N1.1.4
2)	4-27	9	43		Instrument bus inverter failed resulting in trip.	A	3	Electric Power (ED)	Generators	N2.2
3)	5-18	9	71		Steam leak on main steam to 1A reheater.	A	3	Steam and Power (HB)	Pipes, fittings	N3.1
4)	5-20	102	70-45		Power reduction. Failure of 1A-MFWP impeller.	A	5	Steam and Power (HH)	Pump	N1.1.4
5)	6-21	139	71		Repair gasket leak on PZR manway.	A	1	Reactor Coolant (CB)	Vessels, pressure	N3.1
6)	6-29	43	70		Repair leak in charging pump filter vent line.	A	1	Reactor (RB)	Pipes, fittings	N3.2
7)	7-2	47	70		Repair leak in charging pump filter vent pipe socket weld.	A	1	Reactor (RB)	Pipes, fittings	N3.2
8)	7-26	16	70		Instrument bus inverter failed resulting in trip.	A	3	Electric Power (ED)	Generators	N2.2
9)	8-24	21	91		Repair feedwater heater tube leaks.	B	3	Steam and Power (HH)	Heat exchangers	N3.2
10)	9-18	6	92-75		Power reduction. Accumulator boron concentration below technical specifications.	B	5	Engineered Safety Features (SF)	Accumulator	N7.0

Table A1.6

FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1974)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/NSIC(N) Event Category
11)	11-2	270	100		Inspect steam generator tubing.	B	1	Steam and Power (HB)	Heat exchangers	N3.1
12)	12-11	6	100-48		Power reduction. Freon check of 1-B condenser for tube leaks.	B	5	Steam and Power (HC)	Heat exchangers	N3.2
13)	12-15	10	100-48		Power reduction. Freon check of 1-B condenser for tube leaks.	B	5	Steam and Power (HC)	Heat exchangers	N3.2
14)	12-21	14	100-48		Power reduction. Freon check of 1-B condenser for tube leaks.	B	5	Steam and Power (HC)	Heat exchangers	N3.2

Table A1.7
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1975)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1-4	<24	100-71		Repaired steam supply line of 1B reheater.	A	5	Steam and Power (HH)	Heat exchangers	N3.2
2)	3-5	9	100	UE 75-01	Rods dropped due to water leakage into rod control cabinets during maintenance.	A	1	Reactor (RB)	I&C	D4.3
3)	3-10	*	100	AO 75-07	SG tube leak prior to refueling and maintenance outage.	A	1	Steam and Power (HB)	Heat exchangers	D6.3
4)	5-19	9	25		Hotwell level control failed causing low SG level.	A	3	Steam and Power (HC)	I&C	N2.1
5)	5-21	44	55		E.H. control failure: vibration-induced wear caused short circuit.	A	1	Steam and Power (HA)	Electrical conductors	N2.1
6)	5-26	6	55		E.H. control valve position limiter malfunction-control circuit card failed due to short on 5/19/75.	A	2	Steam and Power (HA)	Circuit closers	N2.1
7)	5-31	24	50		A. E.H. unit repair.	A	1	Steam and Power (HA)	Mechanical function units	N2.1
					B. MOV on main feedwater line repair.			Steam and Power (HH)	Valves	N1.1.4
8)	6-6	12			MSIV malfunction. (See similar event on 6-23-75.)	A	3	Steam and Power (HB)	Valves	D2.4

Table A1.7
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1975)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
9)	6-17	4	95		Loss of instrument bus inverter due to failure of the pulse drive printed card circuit.	A	3	Electric Power Systems (ED)	Generators	N2.1
10)	6-17	97	30	LTR 7-17-75	Manual turbine trip due to excessive vibration of the FW piping due to hunting by the FW control valves.	A	1	Steam and Power (HH)	Transformers	N2.1
11)	6-23	87	88	UE 75-03	MSIVs malfunctioned and went closed.	A	3	Steam and Power (HB)	Valves	D2.4
12)	7-20	6	100-14		Power reduction. Repaired leak in the drain line on the high pressure turbine.	B	5	Steam and Power (HA)	Transformers	N3.2
13)	7-24	12	100		Lightning strike in switchyard caused turbine trip.	H	3	Electric Power Systems (EA)	Relay	N9.2
14)	7-30	14	99-48		Power reduction. Freon check for tube leaks in 1A condenser water boxes. (Ref. 1)	B	5	Steam and Power (HH)	Heat exchangers	N3.2
15)	8-3	<24	100-56		Power reduction. Load dispatcher requested rollback to maintain transmission line capabilities.	G	5	Electric Power Systems (EA)	None	N9.0
16)	10-10	44	100		To replace power cables for lake intake heaters.	B	1	Steam and Power (HF)	Electrical conductors	N1.1.4
17)	10-26	<24	100-70		Power reduction. Loss of #1 generator transformer cooling fans.	A	5	Power Auxiliary Systems (AA)	Blowers	N1.1.4
18)	12-21	15	100-46		Power reduction. Freon check for tube leaks in 1-B-1 condenser water box. No leaks found.	B	5	Steam and Power (HH)	Heat exchangers	N3.2

Table A1.7
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1975)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
19)	12-23	15	100-46	-	Power reduction. Freon check for tube leaks in 1-B-1 condenser water box. One leak found.	B	5	Steam and Power (HH)	Heat exchangers	N3.2
20)	12-30	.44	·100	RO 75-13	SG tube leak.	A	1	Steam and Power (HB)	Heat exchangers	D6.3

*Refueling outage started 4 days earlier than scheduled due to leak.

†No leaks found. The source of sodium was traced to a leaking heating coil in the sodium hydroxide tank.

‡Other maintenance included: 1. Leak in B AFW isolation valve seal was sealed using the Furmanite process.
2. Replaced scored and leaking accumulator cylinder of hydraulic snubber on B feedwater line.
3. Replaced cable and connector at the reactor head to bank B CRDM.

Table A1.8
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1976)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
Continued)	NA	307	NA		Continuation from 12/30/75 outage due to B steam generator tube leakage.	A	4	Steam and Power (HB)	Heat exchangers	—
1)	1-29	66	100		No. 2 low pressure turbine blade failure.	A	1	Steam and Power (HA)	Turbines	N1.1.4
2)	4-7	<24	<1		Reactor trip due to maintenance error on overpower delta-T instrumentation.	B	3	I&C (ID)	I&C	N5.1
3)	4-12	<24	40-27		Power reduction. Turbine runback due to failure of Power Range N-44 High Voltage Power Supply.	A	5	Electric Power (EB)	Circuit closers/ interrupters	N2.2
4)	4-14	*	70-50		Power reduction. Vibration in HFWP A.	A	5	Steam and Power (HH)	Pump	N1.1.4
5)	4-16	5	50	76-13	Plant placed in hot shutdown to retrieve two dropped control rods, G-5 and G-9. Four printed circuit cards replaced, but no defect found.	A	1	Reactor (RB)	Control rod drive mechanisms	D4.3
6)	4-18	5	50		E.H. governor problems due to water in oil from oil cooler leak.	A	1	Steam and Power (HA)	Circuit closers/ interrupters	N2.1
7)	4-18	40			E.H. governor problems due to water in oil from oil cooler leak.	A	1	Steam and Power (HA)	Circuit closers/ interrupters	N2.1
8)	4-24	308	99	76-15	B steam generator tube leakage.	A	1	Steam and Power (HB)	Heat exchangers	D6.3

Table A1.8
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1976)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/NSIC(N) Event Category
9)	5-7	10	45		Gasket failure in 1-A moisture separator reheater steam line.	A	1	Steam and Power (HB)	Pipes, fittings	N3.2
10)	5-12	159	90-50		Power reduction. 1A MFWP impeller failure.	A	5	Steam and Power (HH)	Pump	N1.1.4
11)	5-22	5	100		Fluid leak in EH system.	B	1	Steam and Power (HA)	Mechanical function units	N1.1.4
12)	6-3	8	100		1-B inverter failure caused low SG level.	A	3	Electric Power (ED)	Generators	N1.1.4
13)	7-4	7	100	76-18	Dropped 2 control rods (G-5 and G-9, again).	A	1	Reactor (RB)	Control rod drive mechanisms	D4.3
14)	7-6	<24	100-81		Power reduction. Condensate pump isolated for repairs.	A	5	Steam and Power (HH)	Pump	N1.1.4
15)	7-29	<24	100-50		Power reduction. #2 low pressure turbine had excessive bearing vibration. Rotor lost some metal.	A	5	Steam and Power (HA)	Turbine	N1.1.4
16)	8-4	50	100	76-21	Dropped 2 control rods, G-3 and G-11.	A	1	Reactor (RB)	Control rod drive mechanisms	D4.3
17)	8-7	668	100		Blade failure on #2 low pressure turbine.	A	2	Steam and Power (HA)	Turbines	N1.1.4

Table A1.8
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1976)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
18)	9-10	168	96-50		Power reduction. B MFWP impeller failure.	A	5	Steam and Power (HH)	Pump	N1.1.4
19)	9-25	<24	95-50		Tube leak was found and plugged in 1B2 condenser. Cause was steam impingement from the steam dump lines located below the affected tubes.	A	5	Steam and Power (HH)	Heat exchangers	N3.2
20)	10-8	511	100	76-24	Leak in stainless steel pipe between boric acid tanks and SI pumps. Five sections of pipe were replaced.	B	2	Engineered Safety Features (SF)	Pipes, fittings	N3.2
21)	12-11	<24	100-45		Power reduction. 1B-1 condenser checked for leak.	B	5	Steam and Power (HH)	Heat exchangers	N3.2
22)	12-12	<24	100-45		Power reduction. Loss of condenser vacuum due to ejector steam supply valve "problem".	A	5	Steam and Power (HH)	Valve	N1.1.4
23)	12-14	<24	100-46		Power reduction. 1B-2 condenser checked for leaks.	B	5	Steam and Power (HH)	Heat exchangers	N3.2
24)	12-17	†	100-46		Power reduction. 1B-2 condenser checked for leaks.	B	5	Steam and Power (HH)	Heat exchangers	N3.2
25)	12-17	12	46	76-28	Dropped control rod F-12. The stationary hold coil was found open at the reactor head assembly plug.	A	1	Reactor (RB)	Control rod drive mechanisms	D4.3

*Until the next shutdown.

†Forced reduction ran over into the following incident.

Table A1.9
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1977)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DOE(D)/ NSIC(N) Event Category
1)	1-11	14	100-46		Power reduction. 1-B-2 condenser tube leak.	A	5	Steam and Power (HH)	Heat exchangers	H3.2
2)	1-25	14	100-50		Power reduction. 1-B-1 condenser tube leak.	A	5	Steam and Power (HH)	Heat exchangers	H3.2
3)	3-21	12	100-45		Power reduction. 1-B-1 condenser tube leak.	A	5	Steam and Power (HH)	Heat exchangers	H3.2
4)	5-23	15	35	77-04	Inoperable control rods, K-7 and L-8 due to installation error.	A	1	Reactor (RB)	Control rod drive mechanisms	D4.3
5)	7-5	188	100	77-08	B steam generator tube leak.	A	1	Steam and Power (HB)	Heat exchangers	D6.3
6)	8-2	81	100		Two snubbers inoperative on steam generator A due to shaft seal leakage.	A	1	Steam and Power (HB)	Shock suppressors and supports	H1.1.4
7)	11-2	48	100		One snubber inoperative on steam generator B due to shaft seal leakage.	A	1	Steam and Power (HB)	Shock suppressors and supports	H1.1.4
8)	11-17	30	100	77-23	Inoperable control rods due to random failure of two p.c. cards.	A	1	Reactor (RB)	Control rod drive mechanisms	D4.3
9)	12-3	18	100		Leaking pipe tap - residual heat removal system.	A	1	Reactor Coolant (CF)	Pipes, fittings	H3.1

Table A1.10
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1978)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1-11	<24	100-48		Power reduction. Condenser tube leaks were repaired.	A	5	Steam and Power (HH)	Heat exchangers	N3.2
2)	1-14	<24	100-48		Power reduction. Condenser tube leaks were repaired.	A	5	Steam and Power (HH)	Heat exchangers	N3.2
3)	1-25	212	100	78-03	B steam generator tube leak. (Also, CRDMs rewired; repaired the B loop sample valve gasket leak; SG snubber A-7 rebuilt; replaced the PZR pressure and level control transmitters with W terminal blocks.)	A	1	Steam and Power (HB)	Heat exchangers	D6.3
4)	2-20	19	100		Packing leak on PZR spray valve 431-B. (Also, PZR block valve repacked.)	A	1	Reactor Coolant (CB)	Valves	N1.1.1
5)	7-15	<24	100-46		Power reduction. Turbine runback.	A	5	Steam and Power (HA)	Turbines	N1.1.4
6)	8-26	18	100		Burned out coil in reactor trip logic.	A	3	I&C (IA)	I&C	N2.1
7)	9-15	<24	100-48		Power reduction. Condenser tube leak repair.	A	5	Steam and Power (HH)	Heat exchangers	N3.2
8)	12-7	32	100		Low oil level in reactor coolant pump motor required manual turbine trip. Reactor tripped on lo-lo SG level. Lube oil added to RCP motor.	F	3	Reactor Coolant (CB)	I&C	N7.0

Table A1.11
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1979)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/NSIC(N) Event Category
1)	1-10	<24			Power reduction. Loss of indication of seal flow on the 1A RCP.	A	5	Reactor Coolant (CO)	I&C	N2.1
2)	4-9	512			Power reduction. During annual overhaul of A condensate pump, B condensate pump discharge valve spuriously reduced condensate pressure at MFWP suction, caused, reduced FW flow.	B	5	Steam and Power (HH)	Valve	N1.1.4
3)	4-11	5.5			Power reduction. Relay failed in the main transformer cooling fan.	A	5	Electric Power (EB)	Relay	N2.1
4)	7-6	691	100	79-13	NRC required shutdown to inspect FW nozzle welds at SGs.	D	1	Steam and Power (HH)	Pipes, fittings	N8.1
5)	8-5	.5	20		Loss of condenser vacuum during test caused reactor trip.	B	3	Steam and Power (HH)	Heat exchangers	N1.2.4
6)	8-31	<24	100	79-17	Power reduction. To bring boric acid tank concentration back to within specifications.	D	5	Reactor (RB)	Accumulators	N7.0
7)	10-27	.3	100-1		Power reduction, B SG handhole gasket leak. Reactor maintained critical at 1% power to aid repair by maintaining vacuum on secondary side.	A	5	Steam and Power (HB)	Heat exchangers	N3.1
8)	11-10	<24	100-20		Power reduction. B SG handhold gasket leak.	A	5	Steam and Power (HB)	Heat exchangers	N3.1

Table A1.11
FORCED OUTAGES AND POWER REDUCTIONS

No.	Date (1979)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
9)	11-24	<24	100-20		Power reduction. B SG handhole gasket leak.	A	5	Steam and Power (HB)	Heat exchangers	N3.1
10)	12-2	416	100	79-22	Repairs on:	A	1			
					A. Tube leak in B SG.			Steam and Power (HB)	Heat exchangers	N3.1
					B. Nozzle on PZR relief line.			Reactor Coolant (CB)	Heat exchangers	N1.1.4
					C. Correct low boron concentration in boric acid tanks.			Reactor (RB)	Accumulators	N7.0

Appendix A: Ginna

Part 2. Reportable Event Coding Sheets

Table A2.1 Coding Sheet for Reportable Events for [illegible]

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
69-01	40509	12/3	12/12*	B	W	J,S	[illegible]	B	D,F	D	C7	1 rod fails to drop on scram (reactor shutdown).
69-02	40510	12/16	12/23*	B	E,N,CC	00	-	C	X	D	C8	Valve fails shut between sump and RHR pump on test. (see 74-06 and NS 11(3) 241).

*From Semiannual Report No. 1.

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
70-01		1/3	[a]	B	X	OO	–	B	Q	E	S3,C6	"B" train SI did not actuate on test signal (see NS 11(4)328). HG-6 relay contacts fail open.
70-02		1/8		D	S	–	U	A	T,V	A	N	RCS temperature exceeds Tech. Spec. cold shutdown limit.
70-03		1/30		B	Y,BB	OO	–	C	T	A	N	Service water valves to a AFP "improperly inspected." A/ left closed.
70-04		3/6		B	E	FF	–	C	N	D	N	Excessive leakage thru containment personnel air lock.
70-05	46960		5/18	C	AA	FF,OO	–	C	N	D	N	MSIVs fail open during test (see 70-6).
70-06	47292	5/14	5/22	B	AA	FF,OO	–	C	N	D	C7	MSIV fails open during test (reactor shutdown; see 70-5).
70-07	49878		8/19	B	–	–	–	–	V	A	C3	Overdose.
70-08	57237		10/14	D	L	OO	–	B	E	C	N	Core deluge valve fails open during shutdown.
70-09	56044		10/14	D	I,P	OO	–	B	T	E	N	Containment isolation valve fails open during shutdown.
70-10	56975		10/23	D	E	OO	–	B	T	E	N	1 of 2 sump valves fails open during startup.
70-11	61003		12/10	D	S	O,DD	–	A	Y	E	N	Main coolant pump studs cracked.

[a] From Semiannual Report No. 1.

Table A2.3 Coding Shee[illegible] Reportable Events for Ginna — 1971

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
71-01	61321		2/24	B	O	N,X	–	B	E	C	C7	DG fails to start during training.
71-02	63222 66927	4/27	4/29	D	X	–	I	C	Q	B	S1, 2, 3, 4 C4	All 4 BAT-to-SI valves fail closed during test.*
71-03	64820	6/30	7/9	B	Z	00	–	B	X	D	C8	Startup of a third condensate pump led to 4 failures in series and a shut-down.
71-04	64850		7/21	B	–	–	–	–	T	A	C3	Overdose.
71-05	67990		10/6	B	DD	–	N	B	B	D	C3	Iodine waste gas monitor fails to detect release.
71-06	68302		12/14	B	X	–	P	C	N	D	N	SI pump fails to start on test. Faulty relay.

*Also identified as a precursor to potential severe core damage in a NSIC study "Accident Sequence Precursors" for Matt Taylor of NRC/PAS.

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment.
72-01	69330		1/27	D	H	-	T	C	Q	D	N	Trip switch fails to actuate on test.
72-02	65012		2/7	B	V	-	L	C	H	D	N	Setpoint drift.
72-03	58877		2/7	B	O	-	P	C	Q	D	N	One of two under-frequency protection channels fails on test.
72-04	69712	3/20	3/30	D	AA	00	-	C	X	D	N	MSIV fails open (A/ during test).
72-05	70029		4/28	D	DD	P	-	B	B	G	C3	Excess release thru aux. bldg. floor drains.
72-06	71407		5/15	D	CC	DD	-	B	O,T	A	N	RHR pump air bound during cold shutdown.
72-07	72421		6/30	B	FF	R	-	B	J,H,X	D	N	Failed fuel found during examination.
72-08	72422	6/23	6/30	B	W	J	E	B	N,Q	G	C7	Two rods drop due to water leak in same cabinet (reactor shutdown).
72-09	73245	7/27	8/7	B	W	J,AA	-	B	Q	D	N	Voltage spike in 15V DC power supply rendered rods not drivable (reactor shutdown).
72-10	75908		10/30	C	FF	R	-	B	X	C	N	Loose fuel plug found during refueling.
72-11	77411		12/28	B	DD	P	-	A	B	A	C3	Release during resin handling.

Table A2.5 Coding Sheet [illegible]eportable Events for Ginna — 1973

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
73-01	81270		5/31	B	V	–	C,L	B	Q	E	N	Loose solder joint in nuclear power averaging unit while at power – rods started withdrawing.
73-01A	81477		6/8	B	H	–	T	C	F	D	N	Armature found failed during test of solenoid coil undervoltage device.
73-02	81464		6/18	B	AA	–	E,T	C	U	A	N	Incorrect trip point found during recheck.
73-03	81505		6/11	B	X	DD	T	C	Q	D	C7	SI pump fails to start on test.
73-04	81505		6/12	B	X	DD	T	C	Q	D	C7	SI pump fails to start on test.
73-05	82680		7/27	D	H,X	–	I,R,T	C	Q	E	N	Low PZR level test signal fails to actuate SI.
73-06	83262	7/22	8/21	B	Z	G,00	–	B	Q	G	C4, C8	FW control valve fails. Consequent water hammer damages supports (see NS 15(1)88).
73-07	83162	7/22	8/1	B	Z	Z	E,M	B	Z	B	C4	Water hammer disconnects a flow orifice and D/P gage from FW piping (reactor shutdown; see 73-06).
73-08	83834		9/14	B	Z	II	E,M	B	Q	D	N	SG flow channel fails downscale.

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
73-09	85369 85370 87031	10/21	10/23	B	H	G	–	B	T	A	S3,S7	LOOP at full power brings on AFW which overcools primary system.
73-10	87031	10/21	10/23	B	Z	G,AA	C	B	Q	D	S5 C6	BAT cut out too soon (upon SI after LOOP) due to loss of power to instrument bus 1A* (reactor shutdown).
73-11	87030		12/21	B	Y	DD	–	C	O	B	C4,C8	2 of 3 AFWP air-bound in test due to common header from condensate supply.

*Also identified as a precursor to potential severe core damage in a NSIC study "Accident Sequence Precursors" for Matt Taylor of NRC/PAS.

Table A2.6 Coding Sheet [illegible]ortable Events for Ginna — 1974

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
74-01	88090	1/18	1/28	D	S	II	–	A	C	A	N	SG water drained back into reactor during shutdown for maintenance (see NS 15(3) 337).
74-02	88534		2/19	D	S	II,MM	–	C	H	D	N	SG tube wastage.
74-03	89162		2/22	D	I	OO	–	C	F,H	D	N	Excessive leakage in test of containment purge and exhaust dampers.[1]
Notification No. 107	89263 90662	Jan.	3/25	D	–	–	–	–	T	A	C3	Maintenance workers received doses in excess of 3 rem/qtr.
74-04	89868 91662	4/6	4/16	D	X	F,DD	P	C	Q	G	C7	Failure of SI pump to start manually from one bus in test.
74-05	90623	4/18	4/29	D	A	Z	–	C	H,Y	E	N	Leak in drain line on the CVCS letdown system.
74-06	93704	4/22	4/23	D	CC	OO	–	B	T	A	C8	Valve in RHR system from sump fails closed (see 69-02).
74-07	93705	4/26	4/26	B	H	–	P	B	Q	D	N	Underfrequency relay fails.
74-08	94198	5/11	5/21	B	DD	H	–	B	B	D	C3	Radioactivity leaks into auxiliary building from mixed bed demineralizer.
74-09	94594	5/30	6/10	B	H	–	P	C	Q	D	N	Underfrequency relay fails in test.
74-10	94522	6/13	6/21	B	S	OO	A,E	B	B	G	C3	Radioactive release offsite due to faulty alarm in chemical drain tank.

Tab (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
74-11	94784	6/26	7/8	B	AA	HH,OO	–	C	X	D	C8	5 of 8 solenoid valves which control the 2 MSIVs fail during test due to overheating – could not close MSIVs.
74-12	94753	6/29	7/18	B	A	Z	–	B	H,CC	E	N	Leak in CVCS pipe weld.
74-13		7/3	7/18	B	A	Z	–	B	H,CC	E	N	Leak in weld on charging pump filter bypass line.
74-14	95033	8/7	8/15	B	X	DD	–	C	Q	D	C7	SI pump fails to start on one bus in test.
74-15	95594	8/15	8/23	B	Z	OO	–	C	F	D	N	SG blowdown isolation valve fails open in test due to foreign object in valve seal.
74-16	95922	9/18	9/30	B	L	A,PP	–	B	C,H	D	N	Primary coolant leaks into accumulators thru check valve.
74-17	96363	9/25	10/4	B	L	A	–	B	O,U	B	C4	Volume per inch in accumulators less than in Tech. Specs.
74-18	97717 97807		11/19	D	J	OO	–	B	H	A	N	Containment purge exhaust valve leakage high during test.
74-19	98164		12/13	B	O	–	P	C	F	G	N	Undervoltage relay fails due to scale (crud) on armature.

Table (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
74-20	94523 98719		12/18	B	Y	DD	–	C	T	A	N	AFW test deficient procedure.
74-21		12/11	12/11	B	X	–	I	B	Q	D	N	BAT level transmitter fails low.

Table A2.7 Coding Sheet [illegible]portable Events for Ginna – 1975

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
AOR 75-01	93279		1/17	B	Y	DD	H	C	P,T	A	N	Lube oil supply regulator drifts, oil pressure drops, and turbine driven AFWP trips during test.
AOR 75-02	100041		2/14	B	X	F	–	C	Q	D	C7	One SI pump fails to start manually in first test. One spring in breaker contact was weak.
AOR 75-03	100041		2/14	B	X	F	–	C	Q	D.	C7	Loose wire causes SI pump to fail to start on test.
AOR 75-04	100040		2/21	B	AA	OO	–	C	BB	D	N	Solenoid valves fail to operate a MSIV in test (see 70-5,6).
AOR 75-05	100276		2/26	B	X	F	–	C	F	D	C7	Breaker fails in power supply to one SI pump in test.
AOR 75-06	100924		3/25	D	A	OO	–	B	T	A	N	RCS deborated during shutdown.
AOR 75-07	101700	3/10	4/7	B	S	II,HH	–	C	D	B	C4, 5, 7	SG tube wall thinning due to phosphate corrosion found during inspection (reactor shutdown).
AOR 75-08	101732 .102290		4/17	C	G	P	–	C	A	D	N	Charcoal filters for post accident iodine show aging.

Tab[illegible] (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
AOR 75-09	102555		4/22	B	W	J	–	B	F	E	N	Borated water leaks out caps on CRD rod travel housings.
AOR 75-10	103687 104052		6/25	B	X	OO	–	C	S	H	C8	Both discharge valves for SI pump C found closed in test (pump C is 1 of 3 50% capacity pumps).
AOR 75-11	103686		6/30	B	S	–	N	B	S	H	N	RCS sampling schedule incorrect.
AOR 75-12	104954		8/5	B	O	–	P	C	F	D	N	Undervoltage relay fails test.
AOR 75-13	105297		8/22	B	P	OO	–	C	S	H	N	Operator mistakenly aligned 2 valves to dump refueling water into sump during testing (see NSIC-144, p. 52).
UER 75-01	100898	3/5	3/17	B	W	J	–	B	Q	G	C7	Two control rods drop due to a water leak into their common cabinet (reactor shutdown; see 72-08).
UER 75-02	102795		5/8	A	A	Z	–	B	CC	E	N	Weld in RCS letdown line leaks.
	105548 109653		5/19	D	–	–	–	–	T	A	C3	6 workers used inadequate masks. $10K fine proposed.

Table (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
UER 75-03	104209	6/23	7/15	B	AA	OO	-	B	F	D	S2	Both MSIVs close spuriously due to steam impinging on leading edge of valves' disks (reactor shutdown).
UER 75-04	105549		8/21	B	W	AA	U,T	C	H	D	N	Zener diode fails in test. The overpower delta T trip bistable set point drifted nonconservatively.
UER 75-05	106344		9/15	B	O	F,H	-	C	Q	D	C7	DC supplied bus 16 breaker fails open upon manual try.
UER 75-06	106618		9/29	B	Z	H	I	B	T	A	C8	SG level indicator and control signal transferred to new bus with no separation of redundant function provided.
UER 75-07	106619		9/29	B	W	A,I	K	B	L,T	A	N	Alarm setpoint too high for deviation between control rod indicator and bank counter.
UER 75-08	106620		10/1	B	V	G	-	C	Q	D	N	One rod not moving with bank in test.
UER 75-09	107765		11/4	B	S,FF	R	-	B	T	A	N	Iodine activity in RCS exceeds limit due to thermal-and pressure - induced stressed during startup.

Table (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
UER 75-10	108519		12/2	B	O	N	C	C	Q	D	N	DC wattmeter indicates no response to governor control switch.
RO 75-11	108802		12/15	B	A	Z	-	B	N,CC	E	N	Weld leaks in charging pump discharge drain line.
RO 75-12	109455		1/8/76	B	A	Z	-	B	N,CC	E	N	Weld leaks in charging pump discharge drain line (see 75-11).
RO 75-13	110324	12/30	1/26/76	B	S	II,MM	-	D	N	D	C5,C7	Leaking SG tubes activate blowdown isolation with high iodine count (reactor shutdown).

Table A2.8 Coding Sheet [illegible]ortable Events for Ginna — 1976

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
76-01	110356	1/11	2/6	D	J	OO	–	C	G,H	G	N	Excessive leakage thru containment purge supply and exhaust valves during heatup.
76-02	110355	1/12	2/6	D	B	DD	–	B	X,Z	D	N	Component cooling pump fails during cold shutdown.
76-03	110354	1/15	2/6	D	CC	Q	–	C	H,T	G	N	Bolts on flange on RHR suction piping from sump not tightened in maintenance.
76-04		1/29		B	H	S,AA	L	B	Q	D	N	One of four power-range channels fails during load decrease for shutdown (see 74-14).
76-05		2/8		D	J	OO	–	C	H	D	N	Excessive leakage thru containment purge supply and exhaust valves in test.
76-06	111419	2/6	2/19	C	N	–	P	C	Q	C	N	Undervoltage relay fails in test.
76-07	111785 113975	2/7	3/1	D	P	QQ	P	C	T	D	N	Sump discharge valve fails open in test due to relay latch failure.
76-08	112137	2/27	3/11	C	S	II,HH	–	C	D	B	C7	SG tubes plugged.
76-09	112165	2/27	3/22	C	O	F,N	P	C	Q	D	N	DC breaker to bus 18 trips 6 seconds too early.
76-10	112164 115062	3/22	6/21	D	J	FF,OO	–	B	H	D	N	Containment purge supply valves fail open.

Table (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
76-11	112138	3/8	3/19	C	EE	QQ	–	C	U	E	C4, C8	Control room ventilation damper precludes use of one of four charcoal filters.
76-12	113287 114205	4/23	5/25	B	I	FF,KK	–	C	N	D	C4	Excessive leakage thru personnel hatch air lock.
76-13	113785	4/16	5/11	B	W	I	R	B	BB	D	C7	Two rods drop spuriously during power escalation (reactor shutdown).
76-14	113784	4/12	5/10	C	W	AA	L	C	Q	D	N	One channel of four power-range high voltage power supplies fails at power (see 74-4).
76-15	114204	4/24	5/24	B	S	II,HH	–	B	N,X	B	C5, C7	SG tube leak trips air ejector and blowdown activity alarms (reactor shutdown).
76-16		5/20		B	V	J	K	C	Q	D	N	Rod position deviation channel found inoperable (see UER 75-07).
76-17	111864 115729	6/18	7/15	B	L	A	–	C	C,T	A	N	Accumulator level falls below Tech. Spec. limits.
76-18	115895	7/4	7/19	B	W	I,S	–	B	Q	B	C7	Two rods drop (reactor shutdown; see previous events: 76-13, UER 75-01; and 72-08).

Table [illegible] (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
76-19	115728	7/8	7/16	B	U	H	–	B	AA	B	C3	11 releases made without proper monitoring.
76-20	115727	7/9	7/16	B	O	F,H	T	C	BB	D	N	Second DG not started when first DG was inoperable.
76-21	116779	8/4	8/19	B	W	S	–	B	Q	B	C7	Two rods drop (reactor shutdown; see same events in same CRDM cabinets 76-18, 76-13, UER 75-01, and 72-8).
76-22	116890	8/5	8/19	A	P,FF	R	–	B	U	B	C4	LOCA DBA changes by Westinghouse.
76-23	117969	8/19	9/14	C	X	Z	–	C	U	A	N	Wrong schedule piping installed during last refueling shutdown.
76-24	118969	10/8	10/22	B	X	Z	–	B	E,N,Y	D	C4, C5	Leaks in SI suction piping (reactor shutdown; see NSIC-144, p. 100).
76-25	119758	10/10	11/9	D	CC	Z	–	B	N,CC	E	N	Leak in RHR return line while shutdown (see 74-5).
76-26	120427	11/16	12/8	B	P	G	C	B	CC	D	N	One of two redundant NaOH valve controllers fails to respond.
76-27	121059	12/16	1/14/77	B	A	Z	–	B	N,CC	E	N	Pinhole leak in weld in letdown diverter line to CVCS holdup tank.
76-28		12/17		B	W	G,J	–	B	Q	D	N	One rod drops due to open stationary hold coil while at power (reactor shutdown).

Table [illegible] (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
76-29	121060	12/17	1/11/77	B	W,FF	R	-	B	T	A	N	Event 76-28 caused quadrant-to-average power tilt ratio to exceed Tech. Spec. limit.
76-30	121033	12/17	1/12/77	B	A	H,Z	-	B	N,CC	E	N	Leak in letdown line weld to mixed bed demineralizer.
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

Table A2.9 Coding Sheet [illegible]ortable Events for Ginna — 1977

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
77-01	121663	1/3	1/27	B	X	F	–	C	Q	D	C7	SI pump C fails to start on bus 14 due to a faulty breaker.
77-02	122135	2/17	2/23	B	P,V	I	–	B	U	B	C4	Loss-of-flow transient DBA analysis changed by Westinghouse.
77-03	125040	5/2	5/16	C	S	II,HH	–	C	D	C	C7	SG tubes plugged after inspection (see NUREG-0090, Vol., 1, No. 1, p. 17).
77-04	132719	5/23	6/20	D	V	G,J	–	A	S	E	N	Cables on two control rods reversed during refueling (reactor shutdown).
77-05	125593	5/20	6/27	C	A	H	–	B	C	H	N	Release of chloride from old bed of resin exceeds RCS Tech. Spec. limit.
77-06	126094	6/19	7/7	B	A	DD	–	B	BB	D	N	Charging pump varidrive smoking; 2 of 3 pumps out of service.
77-07	132720	6/29	7/27	B	X	F	–	C	Q	D	C7	SI pump fails to start on bus 14.
77-08	126884	7/5	7/18	D	S	II,HH	–	B	E,Y	D	C7	SG tube leaks trip RCS activity alarms (reactor shutdown; see 77-03).
77-09	143450	7/12	8/2	D	A	Z	–	D	N,CC	G	N	Leak in weld in charging pump discharge relief piping.

Table [illegible] (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
77-10	143451	7/13	7/27	A	BB	Z,KK	–	B	U	B	C4	Design error in redundant service water discharge line support.
77-11	143455	7/13	7/27	B	F	E	–	B	T	A	N	Change in containment fan cooler Tech. Specs.
77-12	143460	7/13	7/29	B	P,X	A,PP	–	B	N	D	N	SI pump discharge check valve leaking.
77-13	143461	8/2	8/25	B	S	II,KK	–	B	N,O,BB	D	N	2 of 8 SG snubbers show excessive seal leakage.
77-14	143463	8/9	8/18	B	S,V	I	–	B	C	H	N	Flux difference (control rods plus boron dilution) target band departure.
77-15	143462	8/22	9/2	B	K	PP	–	B	C	D	N	Check valve leakage from RWST to NaOH tank.
77-16	143464	8/23	9/2	B	X	OO	–	A	S	G	N	1 of 2 boric acid flow paths to RCS isolated.
77-17	143465	8/24	9/2	B	K	JJ	–	B	C,T	A	N	Change in calibration procedure for NaOH concentration.
77-18	143466	9/4	9/30	B	A	Z	–	B	N,CC	G	N	Charging pump discharge relief valve pinhole leak.
77-19	143545	9/14	10/6	B	O	F	–	C	S	H	C7	DG bus 16 breaker fails to close in test (see 77-01, 77-07).
77-20	143467	9/20	10/19	B	A	Z	–	B	N,CC	E	N	Charging pump drain line pinhole leak.

Table (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
77-21		10/7		B	O,R	N	–	C	T	H	C7	DG governor hydraulic actuator speed setting misread and maladjusted.
77-22		11/2		B	S	FF,KK	–	B	N	E	C7	One of eight SG snubbers' reservoirs found empty.
77-23	144122	11/16	3/21/78	B	V	–	K,R	B	Q	D	C7	Control rod urgent failure rod stop. 2 p.c. cards failed.
77-24		11/18		B	V	–	K,R	B	Q	D	C7	Control rod urgent failure rod stop. 2 p.c. cards failed.
77-25		11/29		B	A	Z	–	B	H,CC	C	N	Pinhole leak in non-regenerative Hx outlet piping.
77-26		12/2		B	CC	Z	–	C	H,CC	C	N	RHR pump leak on flow orifice inlet weld in test.

Table A2.10 Coding Sheet for Reportable Events for Ginna — 1978

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
78-01	134505	1/11	2/8	B	A	Z	–	B	J,N,CC	D	N	Leak in charging pump discharge relief pipe weld.
78-02	134488	1/11	2/10	B	H	J	S	C	T	A	N	Failure to verify rod stop permissive and low range power range trip as required by Tech. Specs.
78-03	134506	1/25	2/8	B	S	II,MM	–	B	N,Y	D	C7	SG tube leak trips air ejector and blowdown activity monitors. (reactor shutdown; see 77-3).
78-04	137333	3/4	3/31	B	O	G,N	–	C	Q	E	N	DG fuel transfer pump unavailable due to poor connection.
78-05	137856	4/19	5/2	C	A,S	P,OO	–	A	C,S	H	N	Reactor makeup water let into reactor erroneously during refueling. Positive reactivity insertion.
78-06	139904	7/13	8/3	B	Z	Z,KK,PP	–	B	O,BB	D	C7,C8	Two snubbers on MS system fail due to constant vibration. (see 79-20).
78-07	140738	8/16	9/14	B	O	F,N	–	C	Q	D	C7	DG bus 16 breaker fails on test (see 79-18). This is bus 16's tenth such failure.

Table (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
78-08	140235	8/23	9/5	B	E	R	-	A	S	H	N	Auxiliary building door opened during fuel handling.
78-09	145262	12/26	1/9/79	B	Q	-	A	A	R	H	N	Proper fire watch not done during halon system repair.

Table A2.11 Coding Sheet [illegible]portable Events for Ginna — 1979

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
79-01	146697	1/3	1/24	B	X	G	A	C	Q	E	C7	SI pump fails to start from bus 14 breaker in test due to loose wire in circuit breaker control wiring (see 78-7).
79-02	146698	1/4	1/24	B	A	DD	–	B	E,J	D	C3	Crack found in charging pump piston cylinder.
79-03	146699	1/4	1/24	B	A	DD	–	A	P,WW	B	C4	Second charging pump (during 79-02 repair) varidrive speed control lost at low speeds.
79-04	147367	2/6	3/6	B	O	N	–	C	O	H	N	DG trips due to low level in fuel tank in test.
79-05	147368	2/6	3/16	B	E	OO	–	C	X	B	C4	Containment recirculation sump outlet valve fails to open in test.
79-06	148569	3/21	4/3	D	S	II,MM	–	A	H,Y	D	C7	SG tubes plugged after inspection (see 77-3).
79-07	149252	4/2	4/30	B	V	–	H	B	T	A	N	Intermediate range monitors not tested when required.
79-08	152012 152297	4/6	8/21	B	X	OO	–	A	CC,H,Y	D	N	Leak found in boric acid flow control valve nipple.
79-09	149253	4/16	4/30	B	P	PP,JJ	–	B	C,H	D	N	Leak from RCS thru blender boric acid supply check valve (see 79-10).

Table (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
79-10	149254	4/20	4/30	B	P	PP,JJ	-	B	C,H	D	H	Leak from RCS thru blender boric acid supply check valve (see 79-9).
79-11	149705	5/11	5/25	B	S	OO	-	B	U	C	C4	PZR relief valve capacity overstated/ underdesigned.
79-12	150418 154821	3/27	7/10	A	P	Z,KK	-	A	T	E	C4,C8	Six anchor bolts for piping supports for "safety equipment" found not up to specs.
79-13	150702	7/6	7/23	D	Z	Y,Z	-	C	T	B	C5	SG FW nozzle to elbow welds inspection per IE Bulletin 79-13. (reactor shutdown)
79-14	150904	7/9	8/8	D	S	CC	A	B	H	D	H	One of three redundant overpressurization alarms has set point drift.
79-15	150905	7/24	8/1	A	P,K,BB, CC	Z,KK	-	A	T	E	C4,C8	Ten piping supports not properly installed in CS, RHR, and SW systems.
79-16	150944	8/4	8/16	D	L,M,P	F	-	B	T	A	S2	Reactor operated with AC power incorrectly supplied to some valves.
79-17	152178	8/31	9/14	B	X	JJ	-	B	C	A	H	BAST concentration exceeds Tech. Spec. limit (power reduction).
79-18	152194	9/13	10/10	B	O	N	P	C	Q	D	C7	DG output breaker to bus 16 fails open (see 78-7).

Tab[illegible] (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
79-19	152357	9/19	10/19	A	BB	Z,KK	-	A	T	E	N	Loose nuts found on service water piping supports to AFWP.
79-20	152755	10/9	11/7	A	Z	Z,KK	-	C	O,BB	D	C7	HS snubber found with no accumulator level (see 78-6).
79-21	152982	10/18	11/16	B	P,Z	OO	-	C	T	A	N	Crossover valves between AFWPs not stroked in test. (new requirement).
79-22	153682	12/1	12/14	B	S	II,MM	-	B	D,N	D	C5,C7	SG tube leaks trip air ejector and blowdown activity monitors. (reactor shutdown; see 77-3).
79-23	154315	12/7	12/21	D	S	Y,CC	-	B	E,CC	D	N	Linear indications found in PZR relief nozzle-to-safe-end weld.
79-24	154294	12/17	12/28	B	X	JJ,OO	-	B	C	D	N	Backflow thru blender boric acid check valve between reactor makeup and BAST. (see 79-9, -10)
79-25	153941	12/23	1/22/80	B	E	JJ	-	B	T	A	N	Containment airlock not tested when required.

[illegible]