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Nuclear Safety Information Center

Engineering Technology Division

REVIEW OF THE OPERATING EXPERIENCE HISTORY OF BIG ROCK POINT THROUGH 1980 FOR THE NUCLEAR REGULATORY COMMISSION'S SYSTEMATIC EVALUATION PROGRAM

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FOREWORD

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

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

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

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

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

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ABSTRACT

A review of the operating experience of the Big Rock Point nuclear power plant from initial criticality through 1981 was performed by the staff of the Nuclear Safety Information Center for the Nuclear Regulatory Commission's Systematic Evaluation Program (SEP). Under the SEP, the safety margins of the design and operation of ten of the older operating commercial nuclear power plants in the United States are being reevaluated.

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

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1. SCOPE OF REVIEW

The assessment of the operating experience review for Big Rock Point covered the time from initial criticality through 1981. The data collection and evaluation included the following aspects of operation: availability and capacity factors, forced shutdowns and power reductions, reportable events, events of environmental importance and radioactivity releases, and evaluation of the operating experience in total. Tables at the end of Chap. 1 show the codes assigned to operational aspects of forced shutdowns, power reductions, and reportable events. These codes are used in the reporting of data collected during the review of operating experience.

1.1 Availability and Capacity Factors

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

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

1. reactor availability =

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

2. unit availability =

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

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

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

Reserve shutdown hours are the amounts of time the reactor is not critical or the unit is shutdown for administrative or other similar reasons when operation could have been continued.

1.2 Review of Forced Shutdowns and Power Reductions

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

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

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

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

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

The listings for the cause, shutdown method, system involved, and component involved along with their respective codes are those used in the NUREG-0020 series[3] ("Gray Books") on shutdowns. Note that the information listed under the "System involved" column in the data tables in the appendixes indicates (1) a general classification of systems (fully written out) and (2) a specific system, which is coded with two letters, within the general classification.

1.3 Review of Reportable Events

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

The following information was recorded for each reportable event reviewed:

1. LER number or other means of identification of report type,
2. NSIC accession number (a unique identification number assigned to each document entered into the NSIC computer file),
3. date of the event,
4. date of the report or letter transmitting the event description,
5. status of the plant at the time of the occurrence (Table 1.4),
6. system involved with the reportable event (Table 1.2),
7. type of equipment involved with the reportable event (Table 1.5),
8. type of instrument involved with the reportable event (Table 1.5),
9. status of the component (equipment) at the time of the occurrence (Table 1.4),
10. abnormal condition associated with the reportable event (e.g., corrosion, vibration, leak) (Table 1.6),
11. cause of the reportable event (Table 1.4), and
12. significance of the reportable event.

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

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

1.4 Events of Environmental Importance and Releases of Radioactivity

Any significant or recurring environmental problems were summarized based on the review of forced shutdowns, power reductions, reportable events (environmental LERs), and operating reports. Routine radioactivity releases were tabulated as well, and releases where limits were exceeded were reviewed and are discussed in Sect. 4.5.1.5.

1.5 Evaluation of Operating Experience

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

Table 1.1. Codes and causes of forced shutdown or power reduction and methods of shutdown

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

Table 1.2. Codes and systems involved with the forced shutdown, power reduction, or reportable event

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

Table 1.2 (continued)

System	Code
Fuel storage and handling systems	FX
New fuel storage facilities	FA
Spent-fuel storage facilities	FB
Spent-fuel pool cooling and cleanup systems and controls	FC
Fuel handling systems	FD
Auxiliary water systems	WX
Station service water systems and controls	WA
Cooling systems for reactor auxiliaries and controls	WB
Demineralized water makeup systems and controls	WC
Potable and sanitary water systems and controls	WD
Ultimate heat sink facilities	WE
Condensate storage facilities	WF
Other auxiliary water systems and controls	WG
Auxiliary process systems	PX
Compressed air systems and controls	PA
Process sampling systems	PB
Chemical, volume control, and liquid poison systems and controls	PC
Failed-fuel detection systems	PD
Other auxiliary process systems and controls	PE
Other auxiliary systems	AX
Air conditioning, heating, cooling, and ventilation systems and controls	AA
Fire protection systems and controls	AB
Communication systems	AC
Other auxiliary systems and controls	AD
Steam and power conversion systems	HX
Turbine-generators and controls	HA
Main steam supply systems and controls (other than CC)	HB
Main condenser systems and controls	HC
Turbine gland sealing systems and controls	HD
Turbine bypass systems and controls	HE
Circulating water systems and controls	HF
Condensate cleanup systems and controls	HG
Condensate and feedwater systems and controls (other than CH)	HH
Steam generator blowdown systems and controls	HI
Other features of steam and power conversion systems (not included elsewhere)	HJ
Radioactive waste management systems	MX
Liquid radioactive waste management systems	MA
Gaseous radioactive waste management systems	MB
Process and effluent radiological monitoring systems	MC
Solid radioactive waste management systems	MD

Table 1.2 (continued)

System	Code
Radiation protection systems	BX
Area monitoring systems	BA
Airborne radioactivity monitoring systems	BB
Other	XX
Not applicable	ZZ

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

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

Table 1.3 (continued)

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

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

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

Table 1.5. Codes for equipment and instruments involved in reportable events

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

Table 1.6. Codes used for reportable events—abnormal conditions

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

Table 1.6 (continued)

AW Leak, external: defect in a component results in a leak from the system that is contained in an onsite building

AX Leak to environment: leak not resulting from a cracked or broken component

AY Was opened/transfers open: component is/was opened by error or spuriously opens

AZ Was closed/transferred closed: component is/was wrongly closed by error or spuriously closes

BA Fails to open: component is in the closed state and fails to open on demand (e.g., the circuit breaker "fails to open" when an overcurrent occurs)

BB Fails to close: component is in the open state and fails to close on demand

BC Malposition or maladjustment: component is out of desired position (e.g., normally open valve is closed) or adjusted improperly (not for instrument drift or out of calibration)

BD Failure to start/turn on: component fails to start on demand

BE Stopped/failed to continue to run: component fails to continue running when it has previously started

BF Tripped: component automatically trips on or off (desired or undesired) (e.g., the turbine tripped because of overspeed, the circuit breaker tripped because of overspeed, or the circuit breaker tripped because of overload)

BG Deenergized/power removed: component on system loses its driving potential but not necessarily electrical power [e.g., (1) a fuse blows and there is no power to a sensor, and the sensor is deenergized; (2) a valve closes off the steam supply to a turbine, and the turbine has no driving power]

BH Energized/power applied: component or system gains its driving potential but not necessarily electrical power (e.g., valve is opened allowing steam to turn a turbine)

BI Unacceptable response time: component does not respond to a demand within a desired time frame but does not otherwise fail (e.g., a diesel generator fails to come to full speed within the time constraint)

BJ High pressure: higher than normal or desired pressure exists in a component or system (does not include instrument misindications)

Table 1.6 (continued)

BK	Low pressure: lower than normal or desired pressure exists in a component or system (does not include instrument misindication)
BL	High temperature: component experiences a higher than normal or desired temperature
BM	Low temperature: component (or system) experiences a lower than normal or desired temperature
BN	Freezing: fluid medium (e.g., water) freezes in or on a component
BO	Excessive thermal cycling: frequent changes in temperature that could result in metal fatigue or cracking
BP	Unacceptable heatup/cooldown rate: heatup or cooldown rate exceeds limits
BQ	Thermal transient: system experiences an undesired or unstable thermal transient or thermal change
BR	Excessive number of pressure cycles: system experiences an undesired number of significant pressure changes (e.g., pressure pulses as from a positive displacement pump)
BS	High level/volume: higher than normal or desired level or volume exists (actual or potential) in a component, such as tank or sump, or area, such as auxiliary building (not for instrument misindication)
BT	Low level/volume: lower than normal or desired level or volume exists in a component (not for instrument misindication)
BU	Abnormal concentration/pH: an abnormal (either high or low) concentration of a chemical or reagent exists in a fluid system or an abnormal pH exists (does not include abnormal boron concentrations)
BV	Abnormal boron concentration: process system control rod has an abnormal boron concentration from burnup, dilution, or overaddition
BW	Overspeed: speed in excess of design limits
BX	Cladding failure: cladding of a component fails (e.g., the cladding of a fuel pellet is breached, and radioactive fuel leaks out)
BY	Burning/smoking: component is on fire or smoking
BZ	Engaged: component engages or meshes (this is not to be used when a component binds or becomes stuck or jammed)
CA	Disengaged/uncoupled: component disengages, loses required friction, or is no longer meshed (as in gears), for example, the clutch on the motor disengages from the shaft (this should not be used for dropped control rods)

Table 1.6 (continued)

Electric/instruments

EA	Excessive electrical loads: electrical loads exceed design rating
EB	Overvoltage/undercurrent: component failure produces an overvoltage/undercurrent condition other than open circuits
EC	Undervoltage/overcurrent: component failure produces an undervoltage/overcurrent condition other than shorts
ED	Short circuit/arcing/low impedance: electrical component shorts or arcs in the circuit or has a low impedance including shorts to ground
EE	Open circuit/high impedance/bad electrical contact: electrical component has a structural break, or electrical contacts fail to contact and fail to pass the desired current
EF	Erratic operation: component (especially electrical or instrument) behaves erratically or inconsistently (if an instrument produces a bad but constant signal, use "EG", if an instrument produces an inconsistent signal use "EF")
EG	Erroneous/no signal: electrical component or instrument produces an erroneous signal or gives no signal at all (not for out-of-calibration error)
EH	Drift: a change in a setting caused by aging or change of physical characteristics (does not include personnel errors or a physical shift of a component)
EI	Out of calibration: component (particularly instruments) become out of adjustment or calibration (does not include drift)
EJ	Electromagnetic interference: abnormal indication or action resulting from unanticipated electromagnetic field
EK	Instrument snubbing: dampening of pulsating signals to an instrument

Hydraulic

HA	High flow: higher than normal or desired flow exists in a component/system (does not include instrument misindication (see code EG)
HB	Low flow: lower than normal or desired flow exists in a component/system (does not include instrument misindication)
HC	No flow or impulse: fluid flowing through a pipe, filter, orifice, or trench or the fluid in an impulse line (e.g., instrument sensing line) is blocked completely or decreased due to some foreign material, crud, closed (either partially or completely) valve or damper, or insufficient flow area

Table 1.6 (continued)

HD	Flow induced vibration
HE	Cavitation
HF	Erosion
HG	Vortex formation
HH	Water hammer
HI	Pressure pulse/surge
HJ	Air/steam binding
HK	Loss of pump section
HL	Boron precipitation
	<u>Other</u>
OA	Declared inoperable: component or system is declared inoperable as required by Technical Specifications but may be capable of partially or completely performing its desired duties when requested (a component/system that is <u>completely</u> failed should not use this code)
OB	Flux anomaly: flux characteristics of the reactor core are not as required or desired (e.g., flux spike due to xenon burnout)
OC	Test not performed: operator or test personnel fails to perform a required test within the required period
OD	Radioactivity contamination: component, system, or area becomes more radioactive than desired or expected
OE	Temporary modification: an installation intended for short term use (usually this is for maintenance or modification of installed equipment)
OF	Environmental anomaly
OG	Airborne release
OH	Waterborne release
OI	Operator communication
OJ	Operator incorrect action
OK	Procedure or record error

2. SOURCES OF INFORMATION

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

2.1 Availability and Capacity Factors

The availability and capacity factors were either extracted or calculated from data given in the Gray Books[3] from 1974 through 1981 (the first Gray Book was issued in May 1974). Prior to 1974, annual or semiannual reports were used to compile availability factors only.

2.2 Forced Reactor Shutdowns and Power Reductions

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

1. *Nuclear Power Plant Operating Experience for 19XX*, for the years 1973-1979 (Refs. 4—11). The report for 1981 has not been published. However, because work on the section on outages in these reports has been performed by NSIC since 1973, the draft copy of this report for 1981 was available.
2. NUREG-0020 series[3] (Gray Books).
3. Annual or semiannual reports of the Big Rock Point plant from the time of startup through 1977. For 1977 through 1981, monthly operating reports were used because the utilities were no longer required to file annual reports. The review of power reductions involved primarily the annual, semiannual, and monthly reports.

2.3 Reportable Events

The NSIC computer file of LERs was the primary source of information in reviewing reportable events. Material on the NSIC computer file consists of the appropriate bibliographic material, title, 100-word abstract, and keywords. When additional information on the event was needed, the original LER (or equivalent) was consulted by examining (1) those full-sized copies on file at NSIC (for the years 1976–1981); (2) the microfiche file of docket material at NSIC; or (3) the appropriate operating report (semiannual, annual, or monthly).

Two computer files on RECON (a computer retrieval system containing ~40 data bases operated at ORNL) were used extensively. Printouts were obtained from the files for Big Rock Point to provide coverage on many types of "docket material," including reportable events, where the licensee may have been in correspondence with NRC [or the Atomic Energy Commission (AEC)] concerning a particular event. Licensees are often requested to submit additional information or perform further analysis. Before the LERs came into existence in the mid-1970s, it was not unusual for licensees to submit, on their own or at the request of NRC or AEC, more than one letter transmitting information on a particular event. Thus, these printouts provided additional sources of information on reportable events.

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

1. *Reports to Congress on Abnormal Occurrences*, NUREG-0090 series[12];
2. "Power Reactor Event Series" (formerly Current Event Series) published bimonthly by NRC;
3. "Operating Experiences," a section of each issue of the *Nuclear Safety* journal; and
4. the publications of NRC's Office of Inspection and Enforcement (IE), such as operating experience bulletins, IE bulletions, IE circulars, and IE information notices.

2.4 Environmental Events and Releases of Radioactivity

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

The data for radioactivity releases were compiled primarily from *Radioactive Materials Released from Nuclear Power Plants — Annual Report 1977* (Ref. 13). This report presents year-by-year comparisons for plants in a number of different categories (such as solid, gas, liquid, noble gas, and tritium). Data for 1978 were taken from *Radioactive Materials Released from Nuclear Power Plants — Annual Report 1978* (Ref. 14). Data for 1979, 1980, and 1981 were compiled from the annual environmental reports submitted by Big Rock Point.

2.5 Use of Computer Files on RECON and Special Publications

Two computer files on RECON (a computer retrieval system containing ~40 data bases operated at ORNL) were used extensively for another purpose in addition to those indicated thus far. Printouts were obtained from the files for Big Rock Point to provide coverage on other types of 'docket material' besides reportable events where the licensee may have been in correspondence with NRC [or the Atomic Energy Commission (AEC)] concerning a particular event. Licensees are often requested to submit additional information or perform further analysis. Before the LERs came into existence in the mid-1970s, it was not unusual for licensees to submit on their own or at the request of NRC or AEC more than one letter transmitting information on a particular event. Thus, these printouts provided additional sources of information on reportable events.

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

1. *Reports to Congress on Abnormal Occurrences*, NUREG-0090 series[13];
2. 'Power Reactor Event Series' (formerly Current Event Series) published bimonthly by NRC;
3. 'Operating Experiences,' a section of each issue of the *Nuclear Safety* journal; and
4. the publications of NRC's Office of Inspection and Enforcement (IE), such as operating experience bulletins, IE bulletions, IE circulars, and IE information notices.

3. TECHNICAL APPROACH FOR EVALUATIONS OF OPERATING HISTORY

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

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

Shutdowns were evaluated against the DBEs found in Chap. 15 of the *Standard Review Plan.*[1] The DBEs are those postulated disturbances in process variables or postulated malfunctions or failures of equipment that the plants are designed to withstand and that licensees analyze and include in safety analysis reports (SARs). The SAR provides the opportunity for the effects of anticipated process disturbances and postulated component failures to be examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures and situations (or to identify the limitations of expected performance).

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

1. ensure that a sufficiently broad spectrum of initiating events has been considered,
2. categorize the initiating events by type and expected frequency of occurrence so that only the limiting cases in each group need to be quantitatively analyzed, and
3. permit the consistent application of specific acceptance criteria for each postulated initiating event.

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

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

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

Reportable events were screened using the criteria presented in Sect. 3.2 and were categorized according to their significance. The information collected on the reportable events was used to analyze trends for all reportable events, both significant and not significant.

3.1 Significant Shutdowns and Power Reductions

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

3.1.1 Criteria for significant shutdowns and power reductions

As indicated previously, the occurrences identified as DBEs were used as criteria to categorize and note significant shutdowns. These events are listed in Table 3.1 at the end of Sect. 3 as they are found in Chap. 15 of the *Standard Review Plan.*[1]

3.1.2 Use of criteria for determining significant shutdowns and power reductions

Generic design-basis initiating events such as "increase in heat removal by the secondary system" or "decrease in reactor coolant system flow rate," were used as primary flags for reviewing the forced shutdowns (and power reductions). Once the generic type of event was identified, the particular initiating event was determined from the details associated with the shutdown. For example, if the reactor shuts down because of an increase in heat removal because a feedwater regulator valve failed open, the shutdown is a generic type 1 DBE. Specifically, based on the initiating event (valve failed open), it is a 1.2 DBE — "feedwater system malfunction that results in an increase in feedwater flow." Some shutdowns were readily identifiable as specific DBEs, such as tripping of a main coolant pump, a 3.1 DBE. Once categorized as a DBE, the shutdown was considered significant regardless of the resulting effect on the plant (because a DBE had been initiated).

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

3.1.3 Non-DBE shutdown and power reduction categorization

Those shutdowns that were not DBEs were assigned NSIC categories (Table 3.2) to provide more information on the failure or error associated with the shutdown. With these categories, more specific types of errors and failures could be examined through tabular summaries to focus the reviewer's attention on problem areas (safety related or not) that were not revealed by the DBE categories.

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

3.2 Significant Reportable Events

3.2.1 Criteria for significant reportable events

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

3.2.2 Use of criteria for determining significant reportable events

The reportable events were all reviewed, applying the two sets of criteria for significance rather liberally. A number of significant events and conditionally significant events were noted. The events initially identified as significant or conditionally significant were analyzed and evaluated further based on (1) engineering judgment; (2) the systems, equipment, or components involved; or (3) whether the safety of the plant was compromised. The final evaluation for significance considered whether a DBE was initiated or whether a safety function was compromised so that the system as designed could not mitigate the progression of events. Thus, the number of events finally categorized as significant was reduced considerably by these steps in the review process.

3.2.3 Reportable events that were not significant

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

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

1. Increase in heat removal by the secondary system

   1.1 Feedwater system malfunction that results in a decrease in feedwater temperature
   1.2 Feedwater system malfunction that results in an increase in feedwater flow
   1.3 Steam pressure regulator malfunction or failure that results in increasing steam flow
   1.4 Inadvertent opening of a steam generator relief or safety valve
   1.5 Spectrum of steam system piping failures inside and outside of containment in a pressurized-water reactor (PWR)
   1.6 Startup of idle recirculation pump[a]
   1.7 Inadvertent opening of bypass resulting in increase in steam flow[a]

2. Decrease in heat removal by the secondary system

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

3. Decrease in reactor coolant system flow rate

   3.1 Single and multiple reactor coolant pump trips
   3.2 Boiling-water reactor (BWR) recirculation loop controller malfunction that results in decreasing flow rate
   3.3 Reactor coolant pump shaft seizure
   3.4 Reactor coolant pump shaft break

4. Reactivity and power distribution anomalies

   4.1 Uncontrolled control rod assembly withdrawal from a subcritical or low-power start-up condition (assuming the most unfavorable reactivity conditions of the core and reactor coolant system), including control rod or temporary control device removal error during refueling
   4.2 Uncontrolled control rod assembly withdrawal at the particular power level (assuming the most unfavorable reactivity conditions of the core and reactor coolant system) that yields the most severe results (low power to full power)
   4.3 Control rod maloperation (system malfunction or operator error), including maloperation of part length control rods

Table 3.1 (continued)

4.4 Start-up of an inactive reactor coolant loop or recirculating loop at an incorrect temperature.
4.5 A malfunction or failure of the flow controller in a BWR loop that results in an increased reactor coolant flow rate
4.6 Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant of a PWR
4.7 Inadvertent loading and operation of a fuel assembly in an improper position
4.8 Spectrum of rod ejection accidents in a PWR
4.9 Spectrum of rod drop accidents in a BWR

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

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

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

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

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

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

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

8.1 Inadvertent control rod withdrawal
8.2 Loss of feedwater
8.3 Loss of ac power
8.4 Loss of electrical load
8.5 Loss of condenser vacuum
8.6 Turbine trip
8.7 Closure of main steam line isolation valves

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

Table 3.2. NSIC event categories for non-DBE shutdowns

- N 1.0 Equipment failure
  - N 1.1 Failure on demand under operating conditions
    - N 1.1.1 Design error
    - N 1.1.2 Fabrication error
    - N 1.1.3 Installation error
    - N 1.1.4 End of design life/inherent failure/random failure
  - N 1.2 Failure on demand under test conditions
    - N 1.2.1 Design error
    - N 1.2.2 Fabrication error
    - N 1.2.3 Installation error
    - N 1.2.4 End of design life/inherent failure/random failure
- N 2.0 Instrumentation and control anomalies
  - N 2.1 Hardware failure
  - N 2.2 Power supply problem
  - N 2.3 Setpoint drift
  - N 2.4 Spurious signal
  - N 2.5 Design inadequacy (system required to function outside design specifications
- N 3.0 Non-DBE reductions in coolant inventory (leaks)
  - N 3.1 In primary system
  - N 3.2 In secondary system and auxiliaries
- N 4.0 Fuel/cladding failure (densification, swelling, failed fuel elements as indicated by elevated coolant activity)
- N 5.0 Maintenance error
  - N 5.1 Failure to repair component/equipment/system
  - N 5.2 Calibration error
- N 6.0 Operator error
  - N 6.1 Incorrect action (based on correct understanding on the part of the operator and proper procedures, the operator turned the wrong switch or valve – incorrect action)
  - N 6.2 Action on misunderstanding (based on proper procedures and improper understanding or misinterpretation on the operator's part of what was to be done – incorrect action)
  - N 6.3 Inadvertent action (purpose and action not related, for example, bumping against a switch or instrument cabinet)
- N 7.0 Procedural/administrative error (incorrect operating or testing procedures, incorrect analysis of an event – failure to consider certain conditions in analysis)
- N 8.0 Regulatory restriction
  - N 8.1 Notice of generic event
  - N 8.2 Notice of violation
  - N 8.3 Backfit/reanalysis

Table 3.2 (continued)

N 9.0	External events
	N 9.1 Human induced (sabotage, plane crashes into transformer)
	N 9.2 Environment induced (tornado, severe weather, floods, earthquake)
N 10.0	Environmental operating constraint as set forth in Technical Specifications

Table 3.3. Reportable event criteria — significant

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

Table 3.4. Reportable event criteria — conditionally significant

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

4. OPERATING EXPERIENCE REVIEW OF BIG ROCK POINT

4.1 Summary of Operational Events of Safety Importance

The operational history of Big Rock Point 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 analyses identified twenty-three DBEs entered. Additionally, four events were identified in which a loss of safety system function occurred.

4.2 General Plant Description

Big Rock Point Nuclear Power Plant is a General Electric boiling water reactor owned and operated by Consumers Power Company. The plant is located four miles northeast of Charlevoix, Michigan, on the Little Traverse Bay of Lake Michigan. There are no large population centers within sixty miles of the plant. Traverse City, with a population of 18,300, is the largest urban area near the plant at a distance of forty-five miles.

The reactor has a licensed thermal power of 240 MWt and a design electric rating of 72 MWe. Big Rock Point achieved initial criticality on September 27, 1962. The turbine generator was first sychronized to the tranmission system on December 8, 1962. The plant reached full temporary licensed power of 157 MWe on March 21, 1963. A permanent forty-year operating license for 240 MWe became effective on May 1, 1964.

4.3 Availability and Capacity Factors

Table 4.1 presents the Big Rock Point availability and capacity factors [reactor availability, unit availability, unit capacity using the maximum dependable capacity (MDC) and unit capacity using the design electrical rating (DER)]. For 1966 through 1980, the reactor availability factor averaged 74.6% while the unit capacity factors, DER and MDC, averaged 57.9 and 60.7%, respectively. From 1966 through 1968, and 1972 through 1980, the average unit availability factor was 69.1%.

Availability and capacity factors were low during 1965, 1976, and 1979. The unit was shut down during the first seven months of 1965 for analyses, testing, and repair work on the thermal shield hold-down assemblies. The lower values for 1976 were due to a refueling outage, installation of the reactor depressurization system, and modification of the emergency core cooling system. In 1979, the plant performance was low because of a 5000-hour shutdown to correct problems with the inlet diffusers.

4.4 Forced Reactor Shutdown and Forced Power Reduction

4.4.1 Review of forced reactor shutdowns and forced power reductions

From startup in September 1962, through December 31, 1980, Big Rock Point experienced 123 forced shutdowns and sixty-six forced power reductions. Tables A1.1 through A1.18 present a compilation of data describing

Table 4.1. Availibity and capacity factors for Big Rock Point

Average	1962-1963	1964	1965[a]	1966[a]	1967[a]	1968[a]	1969[a]	1970[b]	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980
Reactor availability	d	d	14.8	75.1	83.7	81.5	89.7	93.5	96.7	80.0	80.0	70.8	60.3	51.4	74.1	78.9	24.0	79.2
Unit availability	d	d	14.6	73.6	81.8	80.2	d	d	d	79.9	79.9	70.3	59.8	50.1	73.4	77.9	23.5	78.9
Unit capacity (MDC)[c]	d	d	13.2	60.5	75.7	68.8	67.3	64.6	59.3	70.7	67.9	54.3	46.7	39.2	63.4	71.9	20.6	71.5
Unit capacity (DER)[e]	d	d	13.0	59.7	74.6	67.8	66.4	63.7	58.5	69.7	67.0	53.5	46.1	38.7	57.2	63.6	18.0	64.1

[a] November to November

[b] November 1969 to December 1970

[c] MDC - Maximum Dependable Capacity

[d] No Data (ND)

[e] DER - Design Electrical Rating

each forced shutdown and power reduction. Limited information was available for 1962 through 1965. Tables 4.2 and 4.3 summarize the forced shutdowns and forced power reductions.

The consequence of some of these shutdowns and power reductions was solely the inability to produce power. However, many of the events have safety implications. Some of the shutdowns were design basis events (DBEs). DBEs are postulated failure events which result in system transients, challenging one or more safety systems. Because they challenge safety systems and are the initiating events in postulated accident sequences, DBEs warrant special attention.

4.4.1.1 Yearly summaries

The following is a discussion of forced shutdowns, forced power reductions and other important events by year for 1962 through 1980.

1962-1963

Criticality was first achieved on September 27, 1962, twenty-nine months after initial ground breaking. The initial full power rating of 157 MWt was reached on March 21, 1963.

During pre-startup in December 1962, resins were inadvertently introduced into the primary coolant water and thus into the control rod drive water. Although attempts were made to remove the resins, troubles were still encountered with the rod drives and it was necessary to remove the fuel from the reactor vessel for cleaning. This event revealed problems with the reactor inlet diffuser design and the cap screws from a tube and channel assembly.

Table 4.2. Big Rock Point forced shutdown summary

	1962-1963	1964	1965	1966	1967	1968	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	Total
I. Forced shutdowns																			
1. Total number	19	3	3	13	9	13	9	10	9	12	2	1	3	3	1	5	4	4	123
2. Total hours down				514	543	259	233	190	210	341	97	1045	3516	3299	88	1941	5239	335	17,850
3. Cause[a]																			
A. Equipment failure	6	2	4	12 (490)	9 (537)	11 (232)	9 (209)	8 (108)	6 (157)	10 (320)	3 (97)	3 (1045)	2 (95)	2 (84)	1 (88)	4 (1919)	5 (5237)	4 (39)	101 (10,657)
B. Maintenance or testing	9	1			1 (6)	1 (21)				1 (13)									13 (40)
D. Regulatory restriction													1 (3421)	1 (3215)			1 (2)	1 (296)	4 (7234)
C. Operational error	2					1 (6)	1 (24)		1 (18)	1 (8)									6 (56)
H. Other				1 (24)				2 (82)	2 (35)							1 (22)			6 (163)
4. Shutdown method																			
1. Manual	3		2	12	6	12	9	7	6	9	2	1	3	2	1	2	2		79
2. Manual scram	2															1	1		4
3. Automatic scram	14	3	1	1	3	1		3	3	3				1		2	1	4	40
4. Continuation			1		1		1				1	2					2	1	9
II. Total number of DBE related shutdowns (These are included in Totals of Part I)	3		1	1	1	1		2	4	3				1		2		1	20
Reactor vessel internals (RA)																	1		1
Reactivity control systems (RB)	1		1	1	2	5	3			3		1				2			19
Reactor vessel (CA)						2													2
Coolant recirculation systems (CB)	1			2		1	1		2	2							1		10
Main steam systems (CC)				1	2													1	4
Reactor coolant cleanup systems (CG)											1								1
Feedwater systems (CH)	1			8			1			2	1								13
Reactor containment systems (SA)																	1		1
Containment heat removal systems (SB)								1			1		1			1			4
Emergency core cooling systems (SF)														1					1
Core spray system (SF-D)							1	1											2
Reactor trip systems (IA)	7	1			2	1		1										3	15
Engineered safety feature instrument systems (IB)																	1		1
Electric power systems (EX)	1									1									2
Offsite power systems (EA)			1	1				2	2							1			7
AC onsite power systems (EB)									1							1			2
Steam and power conversion systems (HX)					1														1
Turbine-generators (HA)	3		2		2		4	3	3	1			1	1	1				21
Main steam supply systems (HB)	1						1			1									3
Main condenser systems (HC)						1	1	1	1				1						5
Turbine bypass systems (HE)	4	2				1				1							2		10
Condensate cleanup systems (HC)						1													1
Condensate and feedwater systems (HH)												1							1
Gaseous radioactive waste management systems (MB)					1	1		1		1				1					5

[a]When available, the number of hours associated with the cause of the shutdown is in parentheses.

Table 4.3. Forced power reduction summary for Big Rock Point

	1962-1963	1964	1965	1966	1967	1968	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	Total
I. Power Reduction																			
1. Total Number	0	0	0	5	9	6	2	1	2	6	10	7	6	1	2	3	0	6	66
2. Cause																			
A. Equipment failure				5	9	6	2		2	5	7	7	4	1	1	1		6	56
B. Maintenance or testing								1		1	3								5
F. Administrative													1						1
H. Other													1		1	2			4
II. Total number of DBE related power reductions (these are included in Totals of Part I				1					1									1	3
III. Systems Involved																			
RX Reactor																			
RB Reactivity control													1			1			2
RC Reactor core				3	1	3	2			1	3	2							15
CX Reactor coolant										1									1
CB Coolant recirculation and controls				1					2	2								1	6
CC Main steam system and controls				1	2													1	4
CG Reactor coolant cleanup and controls										2	2							1	5
CH Feedwater systems and controls				1		3						1		1	1				7
IX Instrumentation and controls																			
IB Engineered safety feature instrumentation											1								1
ID Safety-related display instrumenation												1							1
EX Electric power																			
EA Offsite power and controls													1			1			2
EB AC onsite power and controls											2								2
WX Auxiliary water																			
WB Cooling system for reactor auxiliaries and controls											2								2
HX Steam and power conversion																			
HA Turbine-generators and controls					6								2		1			2	11
HC Main condenser and controls												3	1						4
HE Turbine bypass and controls													2					1	3
MX Readioactive waste management																			
MA Liquid radioactive waste management								1											1
XX Other systems																1			1

1964

There were three forced shutdowns in 1964. All three were due to equipment failures. Two of these were due to spurious opening of the turbine bypass valve. The third was due to a spurious trip of a channel 2 picoammeter during testing of channel 1.

Power operation ceased in February and resumed on May 21 following a shutdown to reload the eighty-four fuel bundles and to inspect the turbine generator. The generator required rewinding of the generator field.

The reactor shut down on July 13 for a gamma scan of the core and reconstitution of a forty-four bundle core. Another shutdown occurred from August 26 to September 15 to inspect core internals in an attempt to find the cause of observed flux oscillations.

On September 18, a scram occurred due to a spurious opening of the turbine bypass valve. During startup and routine control rod tests following the scram, evidence of galling of the rod drives was noted. The plant shut down for examination of internals which revealed that foreign particles had lodged between the index tube and the upper guide sleeve of the control rod drives. While inspecting the control rod blades, six thermal shield hold-down studs were found to be cracked. The unit remained shut down for the remainder of the year to investigate methods of repair.

1965

In 1965, there were three forced shutdowns. All three of these (and one continuation of a forced shutdown) resulted from equipment failures.

The 1964 shutdown for thermal shield modifications continued until September 4, 1965. After startup, the reactor operated at full power for the remainder of the year. On September 17, a load rejection occurred due to a relaying malfunction. The plant shut down for several days to repair turbine steam leaks and modify twenty-two control rods.

1966

Twelve of the thirteen forced shutdowns in 1966 resulted from equipment failures. The thirteenth was due to a storm. All five of the forced power reductions resulted from equipment failures.

On August 3, the plant was shut down for the eighth time because of tube leakage in the high-pressure feedwater heater. At this time, the heater tube sheet was blanked and the water box divider tube removed. Permanent piping allowed the feedwater to bypass the heater and thus eliminate the tube leak problem.

Fuel cladding failures necessitated reducing the plant power level as follows: to 79% from February 10 until the refueling outage in April; to 85% on June 2; to 71% after the feedwater heater outage on June 18; and to 46% from July 26 until the refueling outage in September. Following refueling, the plant was operated at reduced power for the rest of the year because of the failure of the seals on the No. 2 reactor recirculating pump.

1967

Nine forced shutdowns and nine power reductions were required in 1967. Eight of the forced shutdowns resulted from equipment failures and one was due to maintenance and testing. All nine of the power reductions were due to equipment failures. The plant operated at full power until January 20, when the reactor scrammed due to difficulty with the initial pressure regulator (IPR). Power was reduced several times during February to make repairs on the IPR. The IPR functioned satisfactorily after the repairs were made on February 17.

After the May refueling, the plant operated continuously until October 26 except for two shutdowns for operator training and examination. Failure of fuel elements necessitated a power reduction.

1968

There were thirteen forced shutdowns and six power reductions in 1968. Eleven of the shutdowns resulted from equipment failures, one was due to maintenance and testing, and one was due to an operational error. All six of the power reductions were due to equipment failures. The plant was operated in the 'all-rods-out' core configuration until the refueling outage in February. The plant resumed operation on March 15. However, due to problems with the lower bearing in recirculating pump No. 2, it was necessary to resume operation with only one recirculation pump in operation. During the approach to critical, following the pump repairs in early April, control rod drive B-4 could not be withdrawn and had to be replaced. Plant operation was resumed on April 9 but the load was derated until the June 21 refueling because old fuel bundles had been used to reconstitute a full eighty-four bundle core. Following the startup, the plant operated around 90% of full power with no significant difficulties until December 14 when fuel cladding failures forced a power reduction to 83%.

1969

There were nine forced shutdowns in 1969 of which eight resulted from equipment failures. An operational error caused the ninth. Both of the power reductions were due to equipment failures.

The plant continued to operate at reduced load due to fuel cladding failures until the 'all-rods-out' coast down started on April 10. Refueling began on April 18 and power operation resumed on May 9. Due to premature failure of several 'E' fuel bundles, the plant operated around 69% power for the rest of the year.

High conductivity of the primary coolant due to previous overheating of the demineralizer resin caused one shutdown. The remainder of the forced shutdowns were due to steam or cooling system leaks.

Members of the company-wide union were on strike from April 8 to June 30. Refueling operations, necessary maintenance, and operation of the plant were performed by supervisory personnel, engineers and technicians during this period.

1970

Eight of ten forced shutdowns in 1970 resulted from equipment failures and two from faults in the transmission line external to the plant. The only power reduction occurred as a result of maintenance operations.

The plant continued to operate at reduced power because of the fuel failures in 1969. During the six-week refueling outage starting on February 13, a turbine inspection was conducted, a control rod drive support structure was installed, portions of the redundant core spray system were installed, and the containment leak rate test was conducted. On June 28 and December 3, high pressure trips occurred because of lead rejections caused by faults in the 138 kV transmission line. Both load rejections were the result of severe storm conditions in the area.

1971

There were nine forced shutdowns in 1971. Six resulted from equipment failures, one was due to an operational error, one due to a local storm, and the other was apparently due to sabotage. The two power reductions were caused by equipment failures.

The plant continued to operate at 70% due to premature failure of several 'E' fuel bundles. On January 23, the plant shut down to repair turbine condenser leaks. During testing of the containment isolation valves prior to startup, the main steam isolation valve failed to close. The cause was a defective solenoid due to moisture in the instrumentation. The instrument air dryer and four similar valves were replaced during the February refueling outage. Also, during the February outage, installation of the redundant core spray system, begun in 1970, was completed and two in-core detector assemblies were replaced.

Members of the company-wide union were on strike from May 12 to September 1 and the plant was once again operated by the supervisory presonnel, engineers, and technicians.

On May 12, the reactor scrammed due to load rejection resulting from a fault in the 139 kV transmission line. The fault resulted from a cut guy line and a corner strain pole cut approximately half way through the thickness of the pole. Another scram occurred on September 28 due to load rejection caused by a local storm.

Two forced power reductions and one forced shutdown were caused by failure of recirculating pump seals.

1972

There were twelve forced shutdowns in 1972. Ten of these resulted from equipment failures, one occurred during testing, and one was due to an operational error. Five of the six forced power reductions resulted from equipment failures. The sixth occurred during maintenance.

A turbine trip occurred on January 25 following a line fault on the offsite 138 kV electric system which was not cleared by the Big Rock Point relaying scheme. As a result, the plant became momentarily isolated from the rest of the 138 kV transmission grid with essentially no load. Concurrently, the redundant 46 kV offsite power supply was also lost due to unusual weather conditions. The diesel generator started and supplied plant loads.

The plant operated in the coastdown mode from January 4 until the refueling shutdown on March 18. During the shutdown, the clean-up system heat exchangers containing Cufenloy tube bundles were replaced. The new heat exchangers utilized stainless steel tube bundles in an attempt to eliminate crud deposits on the fuel cladding and thus decrease the rate of premature cladding failures.

After several miscellaneous equipment failures caused power reductions or shutdowns, power increased to 83% on July 6 and remained at this level most of the time until December 30. On December 30, increased activity levels in the off-gas due to fuel cladding failures required a power reduction to 68%.

1973

There were only two forced shutdowns in 1973. Both of these resulted from equipment failures. There were ten forced power reductions. Seven power reductions resulted from equipment failures, two occurred during maintenance, and one during testing.

The plant operated in the coastdown mode until March 3 when it was shut down for refueling. Fuel crud levels were lower than previous cycles.

In the middle of April, power operation resumed at 92% of full power. Power remained at this level until December 3 when it was reduced to 76% due to high off-gas activity from fuel cladding failures. Another power reduction (to 70%) was necessary on December 6 because of high off-gas activity.

On December 8, the unit was forced off-line due to a packing failure on the reactor steam drum level instrument valve. The plant remained down for the rest of the year to repair the leaking emergency condenser.

1974

There were one forced shutdown and seven forced power reductions in 1974. All resulted from equipment failures.

After completing the repairs to the emergency condenser, the plant was returned to service at 50% power. On January 12, a special operational test was successfully completed on the emergency condenser to assure adequate cooling capacity. The power was raised to 70% and it was maintained at this level until the refueling outage starting on March 23.

Following startup after the refueling outage, the power was increased in an attempt to reach 100%. However, the stop on the initial pressure regulator was reached at 98%. After three hours operation at this power level, a reduction to 93% was initiated due to flooding of the intermediate pressure feedwater heater and condenser vacuum upset. Further power reductions were necessary on May 17 and 20 due to fuel cladding failure.

On October 6, an incore detector failed leaving only ten operational. The plant was put into coastdown mode from November 5 through November 26 at which time the administrative limit was reevaluated and raised from 80 to 90% of the technical specifications thermal-hydraulic limits. Operation was resumed at 83% on November 26 and maintained at this level for the remainder of the year.

1975

Of the three forced shutdowns in 1975, two resulted from equipment failures and one was due to regulatory restrictions. There were six forced power reductions. Four resulted from equipment failures, one was administrative, and one was for modification of an external substation.

At the start of the year, the power was being maintained at 83%. On January 7, power was reduced to 80% due to encroachment on the 90% maximum average planar linear heat generation rate limit on 'F' type fuel. On January 16, approximately one week prior to the scheduled semiannual outage, the plant shut down when studies revealed that there was a design deficiency in the instrumentation for the post-incident cooling system. Modifications, as a result of the special task force investigation, were completed by the first week of June, at which time power operations were resumed. The power fluctuated around 80% until October 18 when a power reduction was required to permit modifications to the Livingston substation. While at reduced power the initial pressure regulator (IPR) failed and load was carried at 68% with the synch-governor control until Ocotber 24 when repairs on the IPR controls were completed. On October 30, the IPR failed again due to a malfunction in the control system involving a valve bellows.

1976

There were three forced shutdowns in 1976. Two resulted from equipment failures and one was due to regulatory restrictions. The single forced power reduction resulted from equipment failure.

Work during the six-month outage starting on January 31 included refueling and installation and startup of the reactor depressurization system. Also included were several minor modifications to the emergency core cooling system (ECCS). On July 28, the 14th cycle began but a power limit of 88% of rated power was imposed due to loss of coolant accident (LOCA) peak clad temperature restrictions. Operation from August throughout the remainder of the year was essentially continuous.

1977

Only one forced shutdown was required in 1977. This shutdown and one of the two forced power reductions resulted from equipment failures. The second power reduction was necessary in order to investigate noise in the No. 2 reactor feed pump.

Operations continued at 85% of full power for the first part of the year with only two minor power reductions. When the plant shut down July 23 for refueling, it had accumulated 343 days of continous operation. After operations resumed, power generation was interrupted only once through the remainder of the year when turbine control problems resulted in an eighty-eight hour outage in October.

1978

There were five forced shutdowns in 1978. Four resulted from on-site equipment failures and one was due to a substation wiring error. One of the three forced power reductions resulted from on-site equipment failure. The remaining two were due to substation and relaying difficulties.

The plant operated most of the year around 90% of full load. Rod drive problems accounted for 1765 hours of outage time. Two shutdowns and two power reductions involved substation or tone relaying troubles.

1979

Of the four forced shutdowns in 1979, three resulted from equipment failures and one was due to regulatory restrictions. There were no forced power reductions.

The plant operated at 82% of full power until the start of the refueling outage on February 3. During the outage the welds of the new core spray ring were reworked.

On April 17 at low power, the turbine bypass valve failed to open causing a high pressure trip. During subsequent testing, a reactor inlet diffuser vibration problem was discovered and reactor vessel repairs were made. On November 4, power operation was resumed. Another shutdown was required on November 6 to replace a recirculating pump, repair incore flange leaks, and repair leaks in the turbine bypass drain line.

On December 31, a shutdown began to address Three Mile Island (TMI) 2 concerns. Modifications provided indication of the relief valve position, the ability to manually reset containment isolation valves, and a radiation monitor for assessing core damage.

1980

All four forced shutdowns and all six of the forced power reductions in 1980 resulted from equipment failures. The short-term lessons-learned changes required by NUREG/0598 (Ref. 14) because of the TMI 2 accident were completed and the plant returned to operation on January 13. A forced shutdown occurred on January 15 due to failure of the initial pressure regulator (IPR). The plant operated at about 88% of full load until the refueling outage which commenced on October 31. During this period, there were six forced power reductions including one which was caused by the loss of a reactor recirculating pump.

4.4.1.2 Systems involved

Tables 4.2 and 4.3 present yearly summaries of the forced shutdowns and forced power reductions that occurred at Big Rock Point. As indicated in the tables, the systems involved in forced shutdowns and power reductions were dominated by three systems. These systems were involved in approximately 80% of the 189 forced outages and power reductions. The three systems are the reactor system (38 events), reactor coolant system (51 events), and the steam and power conversion system (60 events).

Each of these system categories contains subsystems. Over half of the reactor system forced shutdowns and power reductions were due to failures in the reactivity control system (RB). Mechanical failures of the control rod drives or leaks in the control rod drive system caused most of these failures. The reactor core system (RC) was not responsible for any forced shutdowns but was responsible for fifteen of the power reductions. These reductions were required due to fuel cladding failures (see Sect. 4.4.3.1).

The forced shutdown and power reductions involving the reactor coolant systems were dominated by failures in the coolant recirculation system (CB) and the feedwater systems (CH). Seventeen outages and power reductions were attributed to the recirculation sytem with eight of these resulting from leaking recirculating pump seals. Leakage in the feedwater heater caused seven of the twenty feedwater system events.

Of the sixty forced shutdowns and power reductions in the steam and power conversion system, thirty-three occurred in the turbine generator and controls system (HA). Of these, thirteen were due to steam leaks in various parts of the system and ten were due to difficulties with the initial pressure regulator. Six of the initial pressure regulator outages occurred within the first two months of 1967. After cleaning and adjusting the regulator on February 17, no failures were reported until 1970.

4.4.1.3 Causes of forced reactor shutdowns and forced power reductions

As well as presenting yearly summaries of forced shutdowns and power reductions for systems, Tables 4.2 and 4.3 also present yearly summaries for causes of these events. Equipment failures dominated the causes for forced shutdowns and power reductions (81%). Approximately half of the equipment failures were due to leaks in piping, heat exchanger tubes, valve packings, and pump seals. Only fourteen percent were caused by human errors with the majority of these caused by maintenance and resting errors. The remaining causes ('others' in the tables) accounted for five percent of the events and were adverse environmental conditions.

4.4.1.4 Non-design basis events

There were 175 force shutdowns or forced power reductions which were not categorized as DBE initiating events. Table 4.4 lists the number of these per year by NSIC category. Seventy-four of the 175 forced shutdowns or forced power reductions were assigned to NSIC event category 1.0 event types - Equipment Failure; nineteen were category 2.0 event types - Instrument and Control Anomalies; fifty-one were category 3.0 event types - Non-DBE Reduction in Coolant Inventory; sixteen were category 4.0 event types - Fuel/Cladding Failure; one was a category 5.0 event type - Maintenance Error; four to 6.0 - Operator Error; one to 7.0 - Procedural/Administrative Error; four to 8.0 - Regulatory Restriction; and five to 9.0 - External Event.

4.4.2 Review of design basis events

Design basis events (DBEs) are transients which challenge the safe operation of a plant and the ability of engineered safety features to safely shut the plant down. Big Rock Point has experienced twenty-three forced shutdowns or forced power reductions caused by DBE initiating events. Table 4.5 gives the number of these events by DBE type for each year. This section discusses the forced shutdown and forced power reductions in each DBE category.

4.4.2.1 DBE category 2 - decrease in heat removal The seventeen events in category 2 were of five types:

1. DBE 2.1 Steam pressure regulator malfunction or failure that resulted in decreasing steam flow (3).
2. DBE 2.2 Loss of external load (9).
3. DBE 2.3 Turbine trip (3).
4. DBE 2.5 Loss of condenser vacuum (1).
5. DBE 2.7 Loss of normal feedwater flow (1).

All of these events were followed by a safe reactor shutdown. In one type 2.3 event (11/26/71), the turbine generator tripped. However, there was no scram since the condenser and turbine bypass valves were able to handle the load. All other category 2 events resulted in automatic scrams.

Table 4.4. Non-DBE initiating event summary

Description	1962-1963	1964	1965	1966	1967	1968	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	Total
1.0 Equipment Failures	4	2	3	2	10	10	5	[illegible]	3	8	3	4	3	1	1	4	4	2	74
2.0 Instrumentation and Control Anomalies	8			1	1			2		1	2				1			3	19
3.0 Non-DBE Reductions in Coolant Inventory (Leaks)				10	5	4	7	2	3	5	3	3	4	1	1		1	3	52
4.0 Fuel/Cladding Failure				3	1	3	2			1	3	2							15
5.0 Maintenance Error					1														1
6.0 Operator Error	2					1	1												4
7.0 Procedural/Administrative Error		1																	1
8.0 Regulatory Restriction													2	1			1		4
9.0 External Events											2		1			2			5
10.0 Environmental Operating Constraint - Tech Specs																			
Total	14	3	3	16	18	18	15	9	6	15	13	9	10	3	3	6	6	8	175

Table 4.5. DBE initiating event summary

DBE category	Description	1962–1963	1964	1965	1966	1967	1968	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	Total
2.1	Steam pressure regulator malfunction or failure that results in decreasing steam flow					1									1				1	3
2.2	Loss of external electrical load			1	1				2	2	1						2			9
2.3	Turbine trip	1								2										3
2.5	Loss of condenser vacuum	1																		1
2.7	Loss of normal feedwater flow										1									1
3.1	Single and multiple reactor coolant pump trips				1					1									1	3
4.3	Control rod	1					1				1									3
Total		3	0	1	2	1	1	0	2	5	3	0	0	0	1	0	2	0	2	23

Two of the three DBE 2.1 events (steam pressure regulator malfunction) were caused by failures of the initial pressure regulator in 1967, 1976, and 1980.

Four of the nine scrams caused by a DBE 2.2 event (loss of external electrical load) (138 kV line) were caused by electrical storms, four were due to relaying malfunctions, and one was due to wiring errors at an offsite substation. The ninth was apparently due to sabotage. A guy wire had been cut and a corner strain pole had been sawed approximately halfway through.

Two of the three DBE 2.3 events (turbine trip) occurred in 1971. Accidental tripping of the 2400-volt station power breaker (September 22, 1971) caused a loss of most major equipment and subsequent turbine trip and rod scram. Failure of the linkage arm of the turbine trip solenoid on November 26, 1971 caused a turbine trip. The load was carried by the bypass valve and condenser until the reactor was manually shut down for repairs to the linkage. The fourth DBE 2.3 event occurred on February 20, 1963, and the only information available is that there was a momentary generator loss.

The single type 2.5 event (loss of condenser vacuum) occurred in 1963. A low vacuum scram resulted from loss of station power.

The only type 2.7 event was caused by inadequate feedwater supply during a load rejection test on July 6, 1972. This caused a reactor and turbine trip.

4.4.2.2 <u>DBE Category 3 - decrease in reactor coolant system flow rate.</u> All three events in category 3 were caused by the loss of one of the reactor coolant pumps (Type 3.1 event 'single and multiple reactor coolant pump trips'). Two of these were due to leaky seals (See also 4.4.3.2). The cause of the third one is not known.

4.4.2.3 <u>DBE category 4 - reactivity and power distribution anomalies.</u> All three events in DBE Category 4 were type 4.3 - control rod maloperation. All occurred at low power. The first was on February 17, 1963. Demineralizer resin which had accidentally been released into the primary system caused a malfunctioning of the rod collet fingers. The second event occurred on October 13, 1968. While returning to power, a rod would not move from notch 15. After shutdown, the rod was exercised using increased hydraulic pressure and it functioned properly. The assembly was removed from the reactor. However, there was no apparent reason for the malfunction. The third occurred on November 12, 1972, when a short period scram occurred because of high notch worth in the withdrawal sequence. A new control rod withdrawal sequence was developed to minimize this difficulty.

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

4.4.3.1 <u>Summary of events relating to fuel element cladding failure</u>. Big Rock Point is a high power density reactor which has been involved in a development program for high performance fuel elements. Its license permits insertion of powdered or pelletized fuel elements with Inconel, Incoloy, and Zircoloy cladding.

Big Rock Point has experienced considerable difficulties with failed fuel element cladding and these experimental elements accounted for a number of the failures. The problem was costly but did not pose any real safety problems. The power was reduced as necessary to keep the off-gas activity within acceptable limits.

Crud buildup was found during most of the early fuel inspections. Chemical analysis showed that the crud consisted mainly of zinc, nickel, iron and copper, which are constituents of the feedwater heater tube material. Therefore the copper nickel tubes were replaced with stainless steel in 1968. However, crud buildup and tube failures continued. These were attributed to 'hide out' of inventory material in the system and from 'fluffing' of the demineralizers plus new material from the cleanup heat exchangers. These heat exchangers were replaced in April 1972.

Flow tests indicated flow pattern difficulties which caused regions of higher power in the fuel rods. To correct this, fuel channel-orifice hardware on sixty-nine of the eighty-four fuel support-tube-and-channel assemblies were replaced in 1972 and 1973. The combination of the aforementioned corrective actions resulted in lower off-gas activity levels and a reduction in fuel element failures.

The first indication of possible leaking fuel cladding was evident on September 4, 1965 when the off-gas activity started to increase. From mid-September through mid-October, the off-gas activity rose consistently on an exponential curve to a rate of about 15,000 μCi/s, and then showed signs of leveling off. This level had remained essentially constant since November 1, 1965. However, on February 10, 1966, the off-gas activity rate of release reached 50,000 μCi/s. This increasing release rate indicated an increased deterioration of the fuel cladding. Therefore, power was reduced from 70 to 60 Mwe (net) to reduce fuel deterioration.

During the April refueling outage, four (out of thirty) bundles were found to have gross defects. One rod had approximately eight inches of fuel missing below the middle spacer. There was no significant amount of crud buildup.

Following the startup in May 1966, the activity in the off-gas continued to increase. Due to high off-gas activity, power was reduced to 64 MWe on June 2, to 54 MWe on June 19, and to 35 MWe on July 26.

During the September 1966 shutdown, dry sipping located eleven failed bundles in the central core region. Four developmental Incoloy-800 clad bundles were the primary source of activity. These failures were not expected since the lead bundle of this group only had an approximate 700 MWd/T exposure (designed for 15,000 Mwd/T). Three other elements failed grossly due to longitudinal splits in cladding or to circumferential cracks at pellet interfaces. The other identified elements had very low leakage signals and were visually inspected. It appears that failure was due to intergranular stress corrosion, similar to that experienced with other stainless steel clad material.

Prior to the May 19, 1967 shutdown for refueling, the off-gas activity rate had been steady at 800 μc/s. This indicated no gross fuel failures in the core. Dry sipping indicated a 19 mil Incoloy 800 clad developmental bundle (D-4) as a leaker. This bundle was eliminated from subsequent core loading. It was also noted that several fuel elements had 1-3 mils of crud buildup since October 1966.

In early December 1967, plant load was reduced after off-gas activity rates had increased from 13,000 to 21,200 μc/s. This reduction was made to preserve fuel integrity.

Refueling was started February 11, 1968. Dry sipping results showed twenty-nine out of thirty-three reload-2 'C' fuel bundles leaked. These are vibratory packed powdered fuel. Impurities on the fuel particles reacted chemically with the cladding to form local blisters of Zirconium hydride. These blisters breached the cladding. The copper, nickel tubes of the feedwater heater were replaced with stainless steel tubes during this shutdown.

On June 9, 1968, off-gas activity again started to increase. The activity rate rose from 3400 to 14,000 μc/s indicating fuel failure. Plant load was reduced on June 12 and 13 to 57 MWe (gross) and 52 MWe (gross), respectively. A power increase to 60 MWe (gross) was made on June 21, 1968 to satisfy requirements set forth in the centermelt fuel program. Off-gas activity increased to 11,000 μc/s during this power increase. Shutdown for refueling began after this test. Two standard stainless steel clad fuel elements, two Zirconium-clad powder elements, and two of six centermelt fuel elements had leaked. During refueling, forty-one 'E' bundles were loaded. These were pellet UO_2 rather than the 'C' powder UO_2 elements that failed early.

During the last week of October 1968, off gas activity rate increased from 3700 to 12,500 μc/s indicating clad failure in the new core. Power was reduced to 68 MWe (gross) on December 14, to 62 MWe (gross) on January 2, 1969 and to 53 MWe (gross) on February 18, 1969 to lower off-gas activity.

Refueling started April 18, 1969 and eighty-two of eighty-four assemblies were dry sipped. Nine failed assemblies were found. Intermediate performance centermelt assembly (D-50) severly failed. Also, three out of thirty 'B' assemblies, and two of ten 'C' UO_2 powder assemblies failed. A big surprise was that three of the forty-one 'E' UO_2 pellet assemblies failed. All failures occurred in the same location in the core - a hot corner on the side closest to the center of the core. Significant crud accumulation and crud spalling was evidenced on all of the failed assemblies. Hot cell examinations were conducted on two fuel rods from the intermediate performance centermelt assembly D-50. The cause of the severe clad deterioration was accelerated corrosion on the rods outside surface driven by local overheating. Since the preliminary investigations showed accelerated corrosion due to high cladding temperatures, subsequent power operation was limited to 165 MWt.

The stainless steel cleanup heat exchangers replaced their copper nickel predecessors in 1972. The poor core flow distribution was corrected in 1972 and 1973. Since then, the crud buildup has decreased and the number of fuel failures have decreased considerably even while operating at higher power levels.

4.4.3.2 Summary of reactor recirculating pump failures. The shaft seals of the reactor recirculating pumps failed ten times between 1966 and 1980. A new type cartridge seal was installed during the February 1968 refueling outage. This had to be replaced shortly after startup due to inadequate seal leak off flow. However, the new type seal cartridges were easier to replace and resulted in shorter duration shutdowns. Until the middle of 1972, the rate of seal failure remained about the same. There have only been two losses of reactor recirculating pumps since 1972.

4.5 Reportable Events

This review of the operating history at Big Rock Point included a study of 324 reportable events which were submitted to the AEC and NRC concerning technical specification violations and limited conditions of operation. These reports came in the form of letters, telegrams, abnormal occurrences (AO's), reportable occurrences (RO's), and licensee event reports (LERs). The reports were reviewed and coded as per Sect. 1.3 and are arranged by year in Part 2 of Appendix A.

4.5.1 Review of reportable events from 1966 to 1980

Although Big Rock Point achieved initial criticality in 1962, this review found no reports prior to 1966 containing reportable event type of information. Events prior to 1966 were obtained from letter correspondence between the AEC and Consumers Power Company. Figure 4.1 illustrates a histogram of reports filed by Big Rock Point for 1966 to 1980. Environmental reports are discussed in Sect. 4.5.1.4.

4.5.1.1 Yearly summaries. The following sections present a summary of reportable events for each year at Big Rock Point.

Prior to 1966

Big Rock Point had trouble with the control rod drives beginning in 1962 when the reactor first went critical. An accidental resin release in December 1962 revealed a design deficiency in the condensate demineralizer. Galling of the control rod drive index tubes also caused malfunctions in the drive system. In 1963, metal chips were present in the guide sleeve windows. After several occurrences, the index tubes were replaced

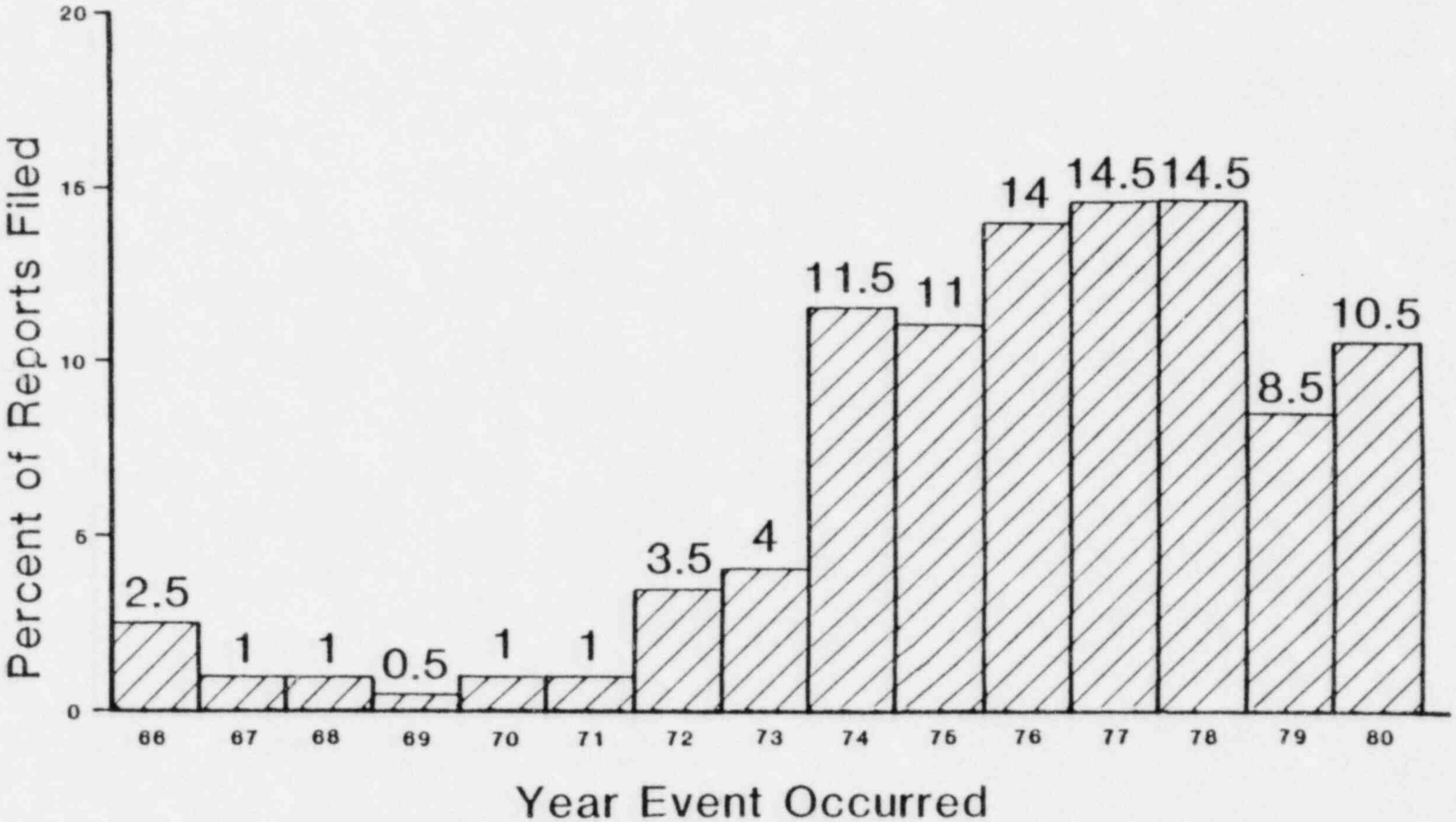

Figure 4.1 The number of reported events per year at Big Rock Point.

with 304 stainless steel index tubes. This resolved the problem of galling. Also in 1963, loose bolts from the fuel-channel-support tubes fell into the control rod drives, causing the control rods to jam. To alleviate the problem of falling bolts, an additional flow distributor was added along with welding of 'keepers' on the cap screws and inserting stabilizer blocks on all unused fuel channels.

On three separate occasions in December 1963, a safety system malfunction involving a scram annunciation was not followed by a scram action. It was impossible to tell which sensor caused the trip. However, it was postulated that vibration on a remote pannel caused a spurious trip of the steam drum low water level sensors. Tests conducted on the system showed the vibration to be of sufficiently short duration to trip the annunciator relays but not the actual scram relays. Additionally, some 'non-fail-safe' failures of certain transistors in the safety system scram logic circuits occurred. However, other (redundant) transistors in the circuit would have had to fail to negate the system.

Failed fuel cladding became a problem in 1965. The off-gas activity rose consistently until it reached 15,000 μc/s, where it leveled off. This level remained essentially constant until 1966.

1966

The first year reportable events were submitted for Big Rock Point was 1966. The problems involving the control rod drives and the fuel elements continued. Four developmental fuel bundles were the primary contributors to the high off-gas activity. These failures were not expected since the fuel had only reached half of its design life.

A design error in the scram dump tank caused several control rods to drift out of the core. The scram dump tank was being pressurized by water leakage through line seals from the insert header to the withdraw header. When the control rod drive pumps were operating, the leakage would pressurize the scram dump tank enough such that the collet piston locking device would open. This allowed the control rods to drift. Installation of a vent line between the scram dump tank and the reactor vessel corrected this design error.

1967

The control rod drive problems and leaking fuel elements were still presenting problems in 1967. Several drives stuck when bolts from the grid bar assembly became lodged. As a result, sixty-eight of the seventy grid bar assembly bolts were replaced.

No gross fuel failures occurred this year. In December, power was reduced after the off-gas activity started to increase. Reducing power preserved fuel integrity.

1968

There was only one incident of a stuck control rod in 1968. A bolt lodged in the drive mechanism and prohibited rod movement. This bolt remained from early test work where torque wrenches broke the upper-grid bolts prior to their replacement.

During refueling in February, dry sipping showed twenty-nine of thirty-three reload-2 'C' fuel bundles leaking. These bundles were vibratory packed powdered fuel. The off-gas activity again increased in June. During the June refueling, pellet UO_2 rather than powdered UO_2 was loaded into the core. An indication of a clad failure in the new core occurred in October when the off-gas activity again increased. The off-gas activity continued to increase into 1969.

1969

Only one reportable event was recorded during 1969. An alarm circuit on a recorder failed to warn the reactor operator of high coolant temperature.

Power was reduced in January and again in February in order to reduce off-gas activity. Refueling in April revealed nine failed assemblies. All of the failures occurred in the same location in the core, a hot corner on the side closest to the center of the core. All of the failed assemblies had evidence of significant crud accumulation and crud spalling. Hot cell examinations on two of the fuel rods showed that the accelerated corrosion on the rod surface was driven by local overheating. Since preliminary investigations revealed accelerated corrosion due to high cladding temperatures, the power was temporarily limited to 165 MWt. The reloading of pellet UO_2 and derating the thermal output of the core solved the problem of leaking fuel elements.

1970

Only three events were reported in 1970. The only event of importance was a diode failure which caused the diesel generator to fail in developing proper voltage.

1971

In 1971, four events were reported and two of these involved control rods. The other two events involved the replacement of a section of cleanup system piping due to cracks and the failure of the diesel generator to run. The diesel failed due to high temperature in the cooling water system. The centrifugal cooling water pump had lost its prime as a result of leakage at the pump shaft packing. The leak depleted the priming water supply during the two weeks between tests. A manual valve for water makeup was left open and steps were taken to provide priming water makeup during a loss of station power.

1972

The number of reportable events for 1972 totaled eleven. Valve failures represented 50% (five out of ten events) of the equipment failures. The diesel generator experienced two failures during this time period. The diesel failed to start when the set points on the lube oil pressure switch were low, and on another occasion, the diesel failed to achieve rated voltage. A significant event also occurred in the electric power system when offsite power was lost during a storm. A trip coil in an oil circuit breaker burned out. A more detailed description is provided in Sect. 4.5.2 which discusses the significant reportable events.

1973

The first year for reporting events as abnormal occurrences (AO's) was 1973. The number of reportable events increased to eighteen. Five of these events were due to setpoint drifts. An administrative error occurred during draining of the fuel pool for relining of the pool. A spent fuel rod was found on the pool floor, and draining was halted. The fuel rod had been on the fuel pool floor since the last refueling outage. The rod was stored temporarily in a fuel transfer cask. Procedures for exercising closer control of all spent fuel operations were implemented.

1974

The number of reportable events doubled from 1973 to 1974 (eighteen to thirty-seven). Along with the continuing problems of control rods becoming stuck, on three occasions the control rods were withdrawn too quickly. Other problems occurring during the year included valve failures in the engineered safety features system. The occurrences involved valves that were leaking, tagged out, or not tested as per technical specifications.

No significant events occurred during 1974, but an event considered noteworthy occurred on July 15. During refueling operations, the supply root valves in the post-incident system were closed and tagged out. The valves had previously been considered part of the fire system. Analysis showed the valves were really common to both systems. Had the post-incident system been required, the operator would have had over two hours to take corrective action before the water level dropped to the reactor flange level. This event was a technical specification violation and operators are now required to check the root valves prior to refueling.

1975

Three of the thirty-four reportable events which occurred at Big Rock Point in 1975 were noteworthy and none were significant. The first event occurred in January when a design deficiency was discovered in the reactor level sensors and pressure sensors. The sensors were not qualified to meet the high temperature specifications for LOCA conditions.

The second event concerned several valves in the reactor cleanup system that were rated lower than the design limits required. Five valves were found to be deficient in either their temperature or pressure requirements.

A procedural error was responsible for the third event. Reactor pressure was reduced for work on the condenser and personnel inadvertently removed the accumulator to a control rod drive system. The accumulator is required for a scram when the reactor pressure is below 450 psi. The cause was a misunderstanding of an operations memo concerning the shutdown margin. The operations memo was revised to clarify the operating requirements.

1976

The number of reportable events increased to forty-four in 1976. Thirteen of the events were attributable to the reactor depressurization system (RDS), which was installed during the year. The RDS is a part of the ECCS and is used to rapidly reduce the pressure of the primary system during LOCA conditions. The reduction in pressure permits the core spray system to spray water into the reactor vessel.

One event considered noteworthy in 1976 involved the RDS. The RDS test procedures were being reviewed when it was realized that the monthly on-line tests were not adequate to meet technical specifications. See Sect. 4.5.1.2.2 for further discussion of this event.

Another event considered noteworthy involved the emergency power system. The diesel generator was supplying 95 kW to busses 1A and 1B following a breaker that tripped due to an overload. The diesel subsequently tripped on high cooling water temperature. The diesel cooling deficiencies were corrected and it was tested with a fire pump load. The plant was in cold shutdown at the time of the event.

Overall, the diesel generator was involved in sixteen of the forty-four reportable events for 1976. Nine of these events were failures of the diesel generator to start within time limits as set forth in the technical specifications.

1977

The RDS accounted for fifteen of the fifty-two reportable events submitted in 1977. On nine occasions, the specific gravity of an RDS battery was low. Even though the specific gravity was below technical specifications, the battery was still able to perform its function.

The diesel generators were again responsible for a proportion of the reportable events (eight of fifty-two). As in the previous year, the generators failed to start within the time limit set forth by the technical specifications.

No significant events occurred in 1977, however, two noteworthy events did. On August 5, one of the noteworthy events occurred in the emergency power system. The event involved the diesel generator. However, the generator was not held accountable. The generator was operating properly when automatic and manual transfers of power to the '2B' bus failed. The auxiliary switch, which was installed in 1976 to ensure proper operation of the generator's output breaker, was not wired properly. Normal station power was available during this incident.

The second event occurred on April 21 involving inadequate testing procedures. A review of the ten-year inspection plan revealed several instances where the minimum number of inspections had not been performed to meet the 25% criteria for the first quarter of the ten-year plan. The ten-year plan was revised to correct its deficiencies.

1978

Failures in the RDS again resulted in a substantial number of reportable events in 1978 (ten of forty-seven). Five of the events in the RDS resulted from the failure of one of the four RDS channels being inoperable. The RDS was also involved in a significant event. During a maintenance activity on the control circuitry of the RDS, the fire pump control switches were placed in the inhibit position. The fire pumps provide initial flow to the ECCS system. If the pumps had been required, the operator would have had to realize the switch was in inhibit and then manually initiate the pumps. Therefore, the system was no longer automatically operable. The fire pump control consoles have been marked with instructions for the use of the inhibit condition. See Sect. 4.5.2 for further details.

On April 7, a significant event occurred involving the reactor protection channels. Two of the reactor protection system channels failed during a loss of offsite power. The failure was attributed to a binding level sensor switch/pointer mechanism on a scale plate inside the cover because of inadequate testing. All four level sensors were repaired and retested prior to plant startup. For further details, see Sect. 4.5.2.

An event worth mentioning occurred in the control rod drive system. A control rod was removed and the reactor mode switch was not placed in the shutdown position as required by technical specifications. This condition existed for several hours until the drive was reinstalled. The incident was reviewed with all repairmen prior to the January 1979 refueling.

<u>1979</u>

The number of reportable events decreased for the second straight year in 1979 (twenty-nine events). The containment isolation system accounted for nine of the reportable events while the reactor coolant system was responsible for eight. Seven of the reportable events were due to valve failures and nine of the reportable events concerned leaks. None of the reportable events involved the RDS.

1980

In 1980, the engineered safety feature (ESF) instrumentation accounted for twelve of the thirty-four reportable events. All of these ESF related events involved set point drift of a level sensor. No events were categorized as significant during 1980.

The RDS appeared to be functioning properly as there were six reportable events in 1980 and zero in 1979, as compared to thirteen, fifteen, and ten in 1976, 1977, and 1978, respectively. All but one of the events involving the RDS in 1980 were reported when the specific gravity of the RDS batteries fell below technical specifications. The remaining event involved one of the RDS channels being removed from service.

4.5.1.2 <u>Review of reportable events by systems.</u> Table 4.6 presents a compilation of reportable events by system and year. Subsystems having a small number of reportable events were combined into the broader system titles where applicable. The code used for the reactor depressurization system (RDS), also known as the automatic depressurization system (ADS), was SF-A. The RDS is a part of the ECCS and was installed in 1975. For the emergency condenser, the system code for reactor core isolation cooling system and controls, CE, was used.

Approximately 77% of the reportable events involved the the following systems: reactivity control system (13.9%), RDS (13.2%), reactor coolant system (12.9), emergency power (12.4%), containment isolation (11.4%), instrumentation and controls (8.8%), and radioactive waste management (4.1%). Radioactive waste management, reactor coolant, and instrumentation and controls are general system categories. The other four systems are unique subsystems with a sufficient number of reportable occurrences such that they were considered seperately.

4.5.1.2.1 <u>Reactivity control system.</u> The reactivity control system accounted for 13.9% of the reportable events. The control rods and control rod drives were involved with most of the occurrences for this system (thirty-five of forty-seven). Jamming of control rods due to galling of the index tubes or lodging of loose parts in the drive system accounted for thirteen occurrences. The other major contributor for the reactivity control system concerned the CRD's. The withdrawal time was less than the technical specifications requirement (six occurrences).

Trouble with the control rod drives was noted during rod performance checks on December 18, 1962. One control rod continued to move downward

Table 4.6. Summary of systems involved in reportable events

System	1966	1967	1968	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	Total
Reactor	3	4	4			2	3	1	10	9	3	6	8	3	2	58
Reactor coolant	3		2	1		1	1	2	1	4	1	7	7	8	6	44
Engineered safety features							2	5	8	8	19	25	15	13	12	107
Instrumentation and controls					1			2	5	1	2	1	4	2	12	30
Electric power	1		1		1	1	3	1	5	4	18	9	5	2	2	53
Fuel handling								1	1	2			2			6
Other auxiliary systems										1		1	4			6
Steam and power	1				1			2		2	1		3	1		11
Radiation protection									1			1	1			3
Radioactive waste management		1					2	3	4	2			2			14
No system applicable								1	2	2		2				7
TOTAL	8	5	7	1	3	4	11	18	37	35	44	52	51	29	34	339

out of the core after the demand signal was turned off. Examination of this drive indicated that resins in the drive had prevented proper operation of the collet fingers. The resins were introduced into the primary system when several of the outlet diffusers shifted allowing the resins to leak through. Failure of one of the outlet strainers then premitted resin release into the feedwater system. On February 17, 1963, another rod drive would not relatch. Inspection of the drive revealed nothing apparently wrong. However, the drive was rebuilt anyway since it was in the core position where the resin was deposited earlier. After the reactor was cleaned, no drive failures due to resin deposits occurred.

One of the major concerns for the control rod drive system appeared several years after the resin deposits were cleaned up. On June 22, 1966, several rods again drifted out of the core. It was determined that the scram dump tank was being pressurized by leakage of water through line seals from the insert header to the withdrawal header. When the control rod drive pumps were operating, the leakage pressurized the scram dump tank. This pressure buildup was enough to open the collet piston locking device, thereby allowing the rods to drift. Therefore, a vent line was installed between the scram dump tank and the reactor vessel. No occurrence of this type has occurred since this modification.

4.5.1.2.2 Reactor depressurization system. The reactor depressurization system accounted for 13.2% of the reportable events. This system was incorporated into Big Rock Point's ECCS in 1976. A large majority of the reportable events were due to the specific gravity of the RDS batteries (twenty-six of forty-five) being below the technical specifications limit. The RDS instrument channels were involved in eleven of the reportable events. Several events occurred as a result of a new system being installed. The first event occurred on September 7, 1976. Procedures for the fire pump actuation tests had not been developed. Testing of the fire pumps had been done during the initial checkout of the system, but the method used was not feasible during operation. Procedures were developed and testing was completed. The next two reportable events occurred on December 9, 1976. During a review of surveillance test procedures as a result of a minor test malfunction, it was realized that the monthly on-line test method was subjecting the system to violation of the single failure design criteria for inadvertent operation. The procedure development was inadequate due to insufficient knowledge of the actuation system. Subsequent review indicated that through weekly verification of the continuious automatic test circuitry and additional testing during refueling operations, adequate testing would be accomplished. The last event of interest occurred on February 15, 1978. During maintenance activity on the control circuitry, the fire pump control switches were placed in the inhibit position. Therefore, the system was no longer automatically operable. The fire pump control switches on the RDS console have been marked with specific instructions for the use of the inhibit position.

4.5.1.2.3 <u>Reactor coolant system and connected systems.</u> The eleven reactor coolant system (RCS) and connected systems accounted for 12.9% of the reportable events. The emergency condensor and reactor core coolant cleanup systems accounted for 50% (twenty-two of forty-four). Valves, piping and welds were the most common equipment failures (twenty-six events or 58.7%). The most common occurrence involved a leak or a crack which had not propagated through-wall (fourteen events or 31.8%). The next major contributor was weld related failures (six events or 13.6%).

Another important event involved the RCS and connected systems, specifically the coolant recirculation and controls system. During inspection for leakage in the control rod drive room on April 20, 1979, noise was heard in the primary system with the recirculation pump in service. On June 9, 1979, it was discovered that a diffuser dislodged from the No. 1 recirculation inlet, while on June 13 a loose diffuser on the No. 2 recirculation inlet was found. Based on geometry factors and flow data, flow blockage did not occur.

4.5.1.2.4 Emergency power. The emergency power system accounted for forty-two (12.4%) of the 339 reportable events. The diesel generators were responsible for 95% of the emergency powers reportable events. The major contributors to these system failures were the failure of the diesels to run (five times), failure to start (nine times), and unacceptable response time (sixteen times). The unacceptable response time represents failure of the diesel to start within the time limits required by the technical specifications. The failures to start were mainly due to design, maintenance, or operator errors. One of the failures to run over the full mission occurred while the diesel generator was supplying a load to buses 1A and 2B following an overload tripping of breaker 52-2A on May 16, 1976. The diesel generator tripped on high cooling water temperature. The cooling pump shaft was scored and the inlet screen was partially plugged.

On August 5, 1977, another event occurred involving a diesel. The diesel generator was operating properly when automatic and manual a transfers of power to the '2B' bus failed to close the generator output breaker. The cause of the failure was improper wiring of the auxiliary switch.

4.5.1.2.5 <u>Containment isolation.</u> The containment isolation system accounted for forty-one (11.4%) of the reportable events. Valve failures contributed to twenty-two of the occurrences for this system, while ten of these failures resulted in leaks. Of all the system failures, eighteen were inherent failures and ten were a result of administrative error. No significant events occurred involving this system.

4.5.1.2.6 <u>Instrumentation and controls.</u> Instrumentation and controls accounted for thirty (8.8%) of the reportable events. The majority of these events involved engineered safety feature instrumentation (nineteen events). Two equipment failures, a valve failure and a cable failure, were experienced. All remaining failures were attributable to instrument errors, and of these, failure of level sensors dominated. Most of these events involving level sensors were due to set point drift of the sensors. Overall, setpoint drift accounted for fifteen of the reportable events.

One event occurred which was categorized as significant. On April 6, 1978, two reactor protection channels failed to operate during a loss of offsite power (LOOP). The level sensors switch/pointer mechanism was binding on a scale plate inside the cover. After the new covers were installed, they were not adequately tested. All four sensors were repaired and retested prior to plant startup. Further details of this event are discussed in Sect. 4.5.2.

Two noteworthy events also occurred in this system. The first event occurred on January 16, 1975 when a design review of the existing core spray switches revealed that eight reactor pressure switches and eight reactor water level switches did not meet the high temperature specifications for the design basis LOCA. Due to these deficiencies, it was not known whether the core spray or backup core spray systems would automatically operate under all postulated accident conditions. However, manual actuation of the core spray system was available.

Another noteworthy event occurred on September 11, 1978 when a control rod was removed and the reactor mode switch was not placed in the shutdown position required by technical specifications. This condition existed for several hours until the drive was reinstalled.

4.5.1.2.7 Radioactive waste management. Radioactive waste management accounted for fourteen (4.1%) of the reportable events. A majority of the events occurred in the radiological monitoring subsystem (ten events). Valve failures due to design errors, inherent failures, or administrative errors were involved in six of fourteen or 43% of the events for the radioactive waste management system. None of the reportable events for this system threatened plant safety.

4.5.1.3 Cause of reportable events. Each reportable event was categorized by the cause codes listed in Table 1.4. The number of reports attributed to each cause is found by year in Table 4.7 and is graphically depicted in Fig. 4.2.

These cause codes can be divided into two groups, non-human causes and human causes. The non-human category includes inherent failure, lightning, and weather. The human failure category includes all the remaining codes. Human failure can be further subdivided into two groups: out-of-plant personnel error and in-plant personnel error. Out-of-plant personnel errors involve administrative, design and fabrication errors which generally concern the reactor or component vendor, the A/E, or the utility

Table 4.7. Cause of reportable events at Big Rock Point by year

Cause		1966	1967	1968	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	Total	Percent of total
Administrative error	(A)			1					4	8	13	7	7	3	2	3	48	14.8
Design error	(B)	2		1		1		3	2	5	7	6	5	6	5		43	13.3
Fabrication error	(C)	1						2		3	2	1				1	10	3.1
Inherent error	(D)	3	5	4	1	2	3	4	4	11	7	21	30	28	17	26	166	51.3
Installation error	(E)	1								3	2		4	2	2	1	15	4.6
Maintenance error	(G)		1						3	4	3	8		3	1	3	26	8.0
Operator error	(H)							1		3	2	2	1	5			14	4.3
Weather	(I)	1						1									2	0.6
TOTAL		8	6	6	1	3	3	11	13	37	36	45	47	47	27	34	324	
Percent of total		2.5	1.9	1.9	0.3	0.9	0.9	3.4	4.0	11.4	11.1	13.9	14.5	14.5	8.3	10.5		100.0

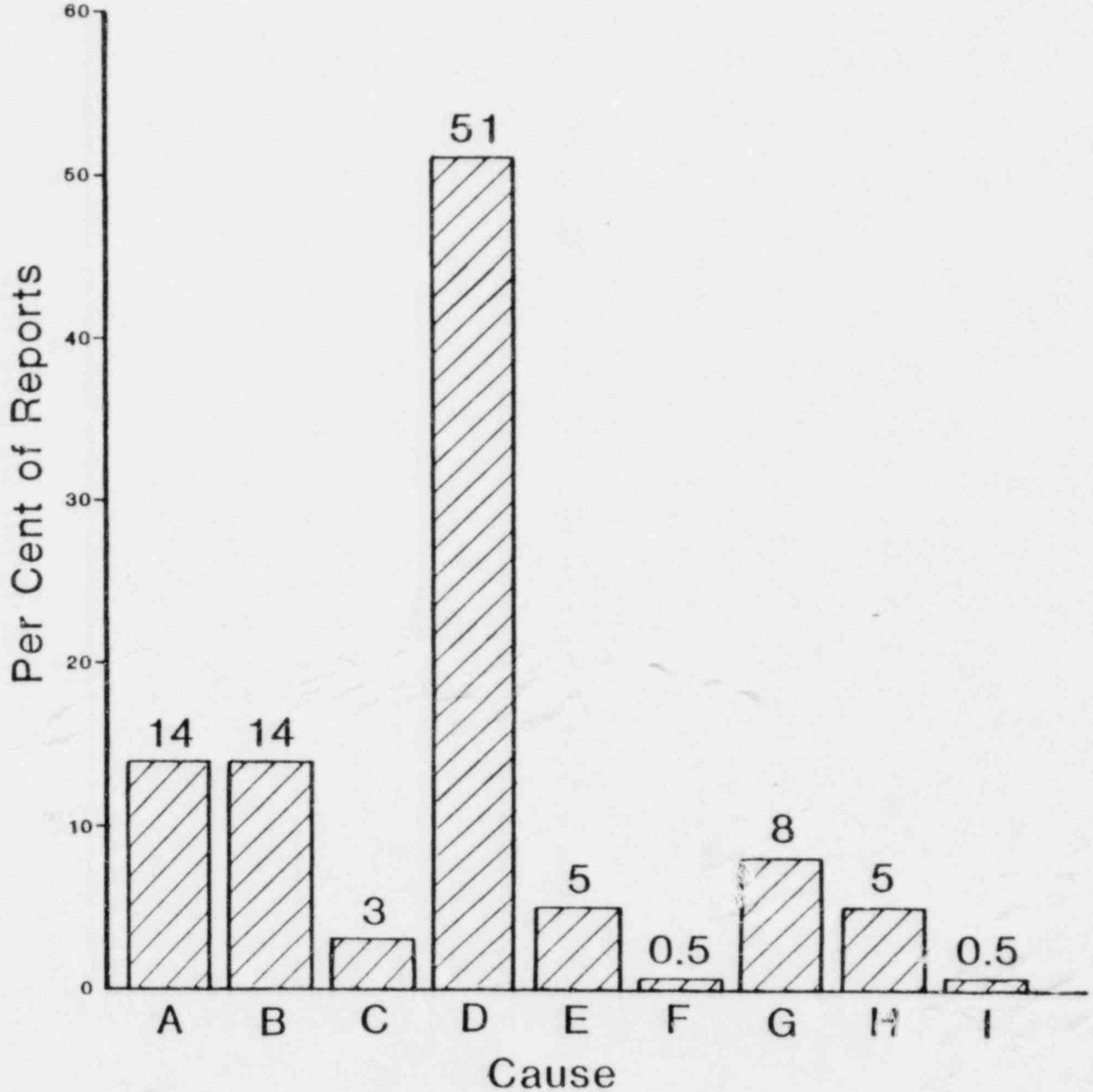

Figure 4.2 Causes of reportable events at Big Rock Point

management. In-plant personnel errors concern hands on human involvement such as installation, maintenance or operator errors and in most cases pertains to the plant operating staff itself.

The number of reports were evenly divided between non-human and human causes with each group contributing 168 and 156 reports respectively. Out-of-plant human errors contributed 101 reports while in-plant human error resulted in only fifty-five reports. Thus about 2/3 of the human errors were caused by people removed from the plant.

4.5.1.4 Events of environmental importance. A summary of radioactivity releases from Big Rock Point is shown in Table 4.8. The table gives the airborne and liquid releases and the solid waste shipped for the years 1966 through 1979.

Seven events have occurred at Big Rock Point which involved or could have involved radioactivity release [illegible] ersonnel exposure. These events are listed in Table 4.9. Only four involved actual releases beyond the plant boundary or possible personnel exposure. Two events involved radioactivity releases in gaseous or liquid form. Four events concerned onsite releases that could have caused a radiological hazard.

Table 4.8. Summary of radioactivity released from Big Rock Point

Effluent	1966	1967	1968	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979
Arborne:														
Total nobe. gas	6.8E+05[c]	ND	ND	ND	2.80E+05	2.84E+05	2.58E+05	2.30E+05	1.88E+05	5.06E+04	1.52E+04	1.34E+04	1.89E+04	6.67E+03
Total I-131		ND	ND	ND	IH	IH	IH	4.60E+00	9.01E-02	2.19E-02	IH	1.40E-03	2.87E-03	2.99E-04
Total halogens (including I-131)		ND	ND	ND	1.3E-01[a]	6.1E-01[a]	1.5E-01[a]	4.70E+00	3.55E-01	2.67E-01	5.0E-02[b]	2.01E-01	1.46E-01	7.86E-03
Tot... particulates ($T_{1/2}$ 78d)		ND	ND	ND	IH	IH	IH	3.70E-01	9.70E-02	9.80E-02	IH	8.67E-03	6.04E-03	1.60E-03
Total tritium		ND	ND	ND	ND	ND	ND	1.97E+01	5.07E+01	7.39E+00	2.41E+00	1.08E+01	8.32E+00	3.15E+00
Liquid:														
Total mixed fission and activa- activation products	6.8[d]	ND	ND	ND	4.70E+00	3.50E+00	1.10E+00	2.70E+00	1.07E+00	2.07E+00	7.70E-01	3.92E-01	2.74E-01	9.03E-01
Total tritium		ND	ND	ND	5.40E+01	1.03E+01	1.04E+01	1.97E+01	5.07E+01	5.73E+00	2.41E+00	8.83E+00	4.05E+00	5.45E+00
Dissolved noble gases		ND	ND	ND	ND	ND	ND	1.70E-02	0.00E+00	7.24E-03	ND	0.00E+00	0.00E+00	5.45E-04
Solid:	~122	ND	ND	ND	ND	ND	1.05E+00	2.13E+07	1.99E+02	1.57E+05	3.69E+00	9.68E+02	2.56E+01	2.77E+02

[a]Halogens and particulates including I-131

[b]Reported as I-131 and particulates

[d]Total airborn

[e]Total liquid

IH - Included in Halogens

ND - No data

4.5.2 Review of significant events. A tabulation of the number of each type of significant event appears in Table 4.10. Significant event codes are defined in Table 3.3. Each reportable event considered significant is identified in Table 4.11. The events which degraded a safety function or initiated a DBE are: three losses of offsite power, and one involving the emergency core cooling system (ECCS).

4.5.2.1 Loss of offsite power with radioactivity release. On August 9, 1966, a violent storm caused the 138 kV breaker to open. The turbine bypass valve opened too slowly, thus the reactor scrammed on high pressure before the turbine could be run back to supply house loads. The turbine continued to supply station loads until it was manually tripped four minutes later. When station power was lost (i.e., the turbine tripped), the bypass valve opened before the d.c. operated isolation valve closed. This caused the turbine rupture diaphram to rupture. The plant airborne activity became high enough to warrant a local evacuation, and was finally cleared four hours later by the turbine building ventilation system. Prior to March of 1968 only one offsite line existed. Thus, every load rejection represented a complete loss of offsite power.

4.5.2.2 Loss of offsite power followed by several component failures. On January 25, 1972, a severe winter storm caused the transmission lines to became ice laden (LTR March 3, 1972). High winds on the following day caused the generation of several momentary line faults when the conductors moved relative to one another. Protective circuits operated successfully on twelve occasions to clear these faults. However, on the thirteenth fault, a trip coil burned out in an oil circuit breaker and the circuit breaker failed to open. Protective relays in the substation

Table 4.10. Number of significant events at Big Rock Point

Significance category	1966	1967	1968	1969	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	Total number assigned
S_3 Two or more failures due to common cause													1			1
S_7 An event which could have been a greater threat	1						1									2
S_9 Other													1			1
Total	1						1						2			4

Table 4.11. Tabulation of reports categorized as significant

NSIC accession No.	Report No.	Description
S2 – Common cause/Common mode failure during the source event.		
138236	RO-78-18	Two steam drum level sensors became stuck during a load rejection.
S7 – An event that could have been a greater threat to plant safety.		
11038	Operations report	Bypass valve problems during a loss of load.
39024	LTR 3/3/72	Several independen failures occur during a loss of offsite power.
S9 – Other events considered significant.		
135891	RO-78-08	Both fire pumps not automatically operable. Therefore, ECCS not automatic.

operated to clear the fault, but in doing so they momentarily isolated the generator from the load and it tripped on overspeed. The reactor subsequently tripped on high flux. Since the fault occurred on the distribution side of the substation, a load rejection signal was not sent to the circuit breaker protecting the generators. Thus, a turbine runback was never initiated. The 138 kV line circuit breaker was manually opened because the line became intermittenly de-energized over a twenty minute period. This resulted in an undervoltage signal and an automatic transfer to the 46 kV alternate source. During the transfer however, a stuck contact on an instantaneous overcurrent relay in the 46 kV bus protection relay scheme, coupled with the operation of the undervoltage bus fault detector relay, caused the circuit breaker serving the 46 kV line to trip. This de-energized the 46 kV line back to Big Rock Point. Normally, the bus fault detector would have reopened had the fault cleared within a few cycles, however, the fault lasted sixty-nine cycles. Thus both offsite power sources were lost. The diesel generators started and provided power to essential loads. Within twenty minutes, full potential was provided to the 138 kV line. When attempts were made to reclose the breaker, a false tripping signal was generated by audio tone relay equipment and the breaker immediately retripped. The audio tone trip was defeated and the 138 kV line was restored. The tone controls were reconnected and the plant loads were transferred to the 138 kV source. The diesel generator started and assumed plant loads.

The two emergency condenser valves, MO-7063 and MO-7053, automatically opened during the transient in order to control reactor pressure. Approximately two and one half hours later, it was decided to close the valves to conserve reactor pressure. Valve MO-7063 failed to operate. An investigation revealed an improperly set torque switch caused the valve's motor operator to burn out.

A design error in the spent fuel pool piping configuration and valve alignment was also discovered. When normal power was lost, the spent fuel pit, the radwaste and treated waste pumps ceased to operate and the spent fuel pit to radwaste isolation valves automatically closed. Due to the valving and piping arrangement, an 11-1/2 ft head was established between the fuel pool and treated receiver tanks.

When the isolation valves were reopened, a siphoning action from the fuel pool to the clear waste receiver tanks was created. The situation was discovered when the operator realized the radiation level in the fuel pool area was gradually increasing. Corrective actions were taken to eliminate the creation of a hydraulic head.

4.5.2.3. Failure of the ECCS to auto-initiate or auto-transfer. On February 15, 1978, both fire pumps were unavailable in the automatic mode due to a maintenance error on the reactor depressurization system control circuitry (RO-78-08). The fire pump control switches were inadvertently placed in the inhibit mode with both pumps shutdown. The fire pumps provide initial flow to the ECCS system. Should the pumps have been required, the operator would have had to realize the switch was in inhibit and then manually initiate the pumps. The fire pump control switches on the RDS panel have been marked with specific instructions for use of the inhibit position.

4.5.2.4. Two reactor protection channels fail during a loss of off-site power. An event in which a common mode failure was involved occurred on April 7, 1978 (RO-78-18). The 138 kV transmission line was lost which resulted in a load rejection. The cause for this event was not discussed in the report. The reactor scrammed on low condenser vacuum. During the transient however, one of two low drum level scram sensors in both of the reactor protection channels became stuck at the +5 in drum level. An investigation revealed that the switch/pointer mechanism was binding on a scale plate inside the instrument's cover. A new cover and scale plate had been installed a month and a half earlier but the problem was not detected during the test. All four sensors were repaired and tested prior to plant startup. This event represented a degradation of the reactor protection system.

Inspection of the diffuser over the No. 2 recirculation pump 20 in diameter inlet revealed that the single lower attachment was loose. This would allow that diffuser to move on its upper attachments in a hinge fashion and make contact with the large baffle. This probably was the source of the vibration type noise first noticed on April 20, 1979.

Based on geometry factors and flow data, it is not believed that flow blockage occurred, however, this does represent an initiated event to core blockage.

A total of four failed bolts were missing: three from the No. 1 diffusers and one from the No. 2 diffuser. One well-worn bolt piece was retrieved during the outage and other well-worn pieces are believed to have been retrieved in prior years dating back to 1974.

4.5.3 Trends and safety implications of reportable events

As an additional step in the overall evaluation process, the events at Big Rock Point were examined to find discernible recurring events that indicate potential safety problems. The four types of recurring events found were:

1. diesel generator failures,
2. emergency condenser failures,
3. control rod drive problems, and
4. failed fuel elements.

4.5.3.1 Diesel generator failures. There were seventeen failures of the emergency power system of Big Rock Point. This greatly exceeds what one would expect based on industry wide emergency power system failures. All but one of these events involved the failure of the single diesel generator.* One event, however, involved a failure of the '2B' (emergency power) bus. No emergency power failures occurred during a loss of offsite power.

Eight of the diesel generator failures were failures to start on demand. There was no single dominant cause for these failures which occurred from 1971 through 1980. In addition to the failures to start, there were eight failures to run. Four of the failures to run were caused by high coolant temperature from cooling water pump failures. The remaining four events were caused by voltage regulator, armature, and fuel transfer pump (twice) failures.

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*Information available during this review suggested a second 'standby' diesel generator was available as early as 1972. This generator must be manually initiated at its site, which is about 220 yards from the turbine building. Once the engine is stablized, the operator must energize the generator and manually load to '2B' bus. The relationship of this generator to the plant, however, is not known (ie, intended use, date of installation, test frequency, etc.). Thus, it was not included as part of the emergency power system.

4.5.3.2 Emergency condenser. Failures in the emergency condenser were also reported in LERs. Two of these failures were system failures. This system utilizes two condensing loops to provide a heat sink during a number of transients. One loop is sufficient to remove decay heat within a few minutes of shutdown. However, should an outlet valve in one loop fail to open upon demand the ability of the single loop to provide for reactor cooling is unknown. The system could either settle out at a higher mass flow rate through the operating loop or the resistance to flow could be too great and the remaining loop would vapor lock (i.e., the pressure drop due to friction could be greater than the head created by the density gradient). Both events identified as failures involved the failure of emergency condensers' valves to open upon demand.

4.5.3.3 Control rod drive problems. The control rods and the CRDs experienced difficulty in the earlier years at Big Rock Point. Reoccurring problems involved: the control rods drifting out of the core, galling of the control rod index tubes, jamming of the rods, and the withdrawal times less than technical specifications limit.

Trouble with the control rod drives was noted during the rod performance checks on December 18, 1962. One control rod continued to move downward, out of the core after the demand signal was turned off. Examination of this drive indicated that resins in the drive had prevented proper operation of the collet fingers. The resins were introduced into the primary system when several of the outlet diffusers shifted allowing the resins to leak through. Failure of one of the outlet strainers then permitted resin release into the feedwater system. On February 17, 1963, another rod drive would not relatch. Inspection of the drive revealed nothing apparently wrong. However, the drive was rebuilt anyway since it was in the core position where the resin was deposited earlier. After the reactor was cleaned, no drive failures due to resin deposits occurred.

On June 22, 1966, after the resin deposits were cleaned up, several rods again drifted out of the core. It was determined that the scram dump tank was being pressuized by leakage of water through line seals from the insert header to the withdrawal header. When the control rod drive pumps were operating, the leakage pressurized the scram dump tank. This pressure buildup was enough to open the collet piston locking device, thereby allowing the rods to drift. Therefore, a vent line was installed between the scram dump tank and the reactor vessel. No occurrence of this type has occurred again since this modification.

The first incident of index tube galling occurred during a scram test on February 20, 1963. Flow measurements indicated high leakage through the drive system. Some resin was still present, but a large number of metal chips were also present in the guide sleeve windows. After a number of such occurrences, nitrided 304 SS index tubes were installed in place of several 17-4 PH SS index tubes. On October 31, 1965, four drive systems stuck due to metal particles. None of the previously modified drives were among the four. Therefore, all remaining drives were modified and no galling of index tubes has been reported since this modifications.

Jamming of control rods due to galling of the index tubes or lodging of loose parts in the drive system accounted for thirteen of the occurrences. The first occurrence of a control rod jamming occurred on December 18, 1962. Several control rod drives jammed when loose bolts lodged on top of the core support plate. The bolts were the same type as those used to bolt together the fuel channels and their support tubes. As a result, all Zircaloy support-tube-and-channel assemblies were modified by staking the cap screws. A drive in the same core position jammed on May 26, 1963. Additional modifications included an additional flow distributor along with welding of 'keepers' on the cap screws.

Loose bolts continued to cause the control rods to stick. On December 25, 1967, several drives stuck when bolts from the grid bar assembly became lodged. As a result, sixty-eight of the seventy grid bar assembly bolts were replaced. On April 6, 1968, another loose bolt in the control rod drive mechanism caused a control rod to jam. The bolt remained from the previous year when torque wrenches broke off several of the upper-grid bar assembly bolts prior to replacement.

The control rods became jammed on several other occasions, however, their occurrences were infrequent.

The last major contributor to the reportable events in the control rod drive system involved the withdrawal time being less than the technical specifications limit. The first three occurrences were in 1974, with two occurring in 1975, and the last one occurred in 1978.

4.5.3.4 Failed fuel elements. Failed fuel cladding became a problem in 1965. The off-gas activity rose consistently until it reached 15,000 μc/s, where it leveled off. This level remained esentially constant until 1966. The primary contributors to the high off-gas activity were four developmental fuel bundles that failed. These failures were not expected since the fuel had only reached half of its design life.

No gross fuel failures occurred in 1967. In December, power was reduced after the off-gas activity started to increase. Reducing power preserved fuel integrity. During refueling in February 1968, dry sipping showed twenty-nine of thirty-three reload-2 'C' fuel bundles leaking.

These bundles were vibratory packed powdered fuel. The off-gas activity again increased in June. During the June refueling, pellet UO_2 rather than powdered UO_2 was loaded into the core. An indication of a clad failure of the new core occurred in October when the off-gas activity again increased. The off-gas activity continued to increase into 1969.

Power was reduced in January and again in February of 1969 in order to reduce off-gas activity. Refueling in April revealed nine failed assemblies. All of the failures occurred in the same location in the core, a hot corner on the side closest to the center of the core. All of the failed assemblies had evidence of significant crud accumulation and crud spalling. Hot cell examinations on two of the fuel rods showed that the accelerated corrosion on the rod surface was driven by local overheating. Since preliminary investigations revealed accelerated corrosion due to high cladding temperatures, the power was temporarily limited to 165 MWt. The reloading of pellet UO_2 and derating the thermal output of the core solved the problem of leaking fuel elements.

4.6 Evaluation of Operating Experience

The major sources of information utilized during this evaluation were (1) forced shutdowns and power reductions and (2) reportable events. Two significant areas were identified in the review of shutdowns and power reductions. These are failed fuel elements and loss of the 138 kV line. Failed fuel was mainly a problem during the 1960's. This problem was solved by replacing powdered fuel with fuel pellets which resulted in derating the core thermal power, changing heat exchanger tube material to reduce corrosion which contributed to crud build-up on the fuel elements, and modifying reactor core flow patterns. The 138 kV line had been lost nine times at what appears to be a constant rate. Two of these events were complete losses of offsite power, a typical number for a plant operating for 18 years. The first complete LOOP occurred prior to the installation of the 47 kV line and little is known about this event including the duration of the LOOP. The second event (see Sect. 4.5.2) which was well documented, involved failures in other systems during the transients, however none of these failures impacted the plants' recovery. Offsite power was also restored within 20 minutes, thus minimizing the significance of this event.

There were no significant problems identified through the search of LERs. Events caused by human errors contributed about half of the reports. Several emergency power failures were identified. However, these events were judged to be failures without adequate knowledge of the emergency power system, and it is possible the number of emergency power failures will be substantually reduced once this information is obtained. It does appear, however, that the 'onsite' emergency power system only loads the '2B' bus automatically. This possibility should be investigated.

The emergency condenser was considered failed on two occasions. Both of these events involved the failure of a dc operated emergency condenser outlet valve to open, thus rendering one of two emergency condenser loops inoperable. Based on the information available we were unable to predict how the emergency condenser would respond to a transient given one loop failed and thus assumed it would become vapor locked. This potential failure should also be investigated in greater depth.

Overall the operation of Big Rock Point has been satisfactory from a safety point of view. A concern was identified about the ability of the emergency power to respond given a loss of offsite power. There were no incidents identified, however, where the emergency power failed to respond adequately given an actual undervoltage on the 138 kV lines. Again, no period was identified were the operation of Big Rock Point posed a great threat to the general public.

REFERENCES

1. Nuclear Regulatory Commission, Instructions for Preparation of Data Entry Sheets for Licenses Report (LER) File, NUREG-0161, July 1977.
2. Nuclear Regulatory Commission, 'Accident Analysis for the Review of Safety Analysis Reports for Nuclear Power Plants,' Chapter 15 of Standard Review Plan, NUREG-0800 (July 1981).
3. Nuclear Regulatory Commission, Licensed Operating Reactors – Status Summary Report, NUREG-0020, May 21, 1974, issue through Vol. 5, No. 1, (January 1981).
4. U.S. Atomic Energy Commission, Nuc'ear Power Plant Operating Experience During 1973, OOE-ES-004 (December 1974).
5. Nuclear Regulatory Commission, Nuclear Power Plant Operating Experience 1974–1975, NUREG-0227 (April 1977).
6. Nuclear Regulatory Commission, Nuclear Power Plant Operating Experience 1976, NUREG-0366 (December 1979).
7. Nuclear Regulatory Commission, Nuclear Power Plant Operating Experience 1977, NUREG-0483 (February 1979).
8. Nuclear Regulatory Commission, Nuclear Power Plant Operating Experience 1978, NUREG-0618 (December 1979).
9. Nuclear Regulatory Commission, Nuclear Power Plant Operating Experience 1979, NUREG/CR-1496 (ORNL/NUREG/NSIC-180) (May 1981).
10. Nuclear Regulatory Commission, Nuclear Power Plant Operating Experience 1980, in publication.
11. Nuclear Regulatory Commission, Radioactive Materials Released from Nuclear Power Plants – Annual Report 1977, NUREG-0521 (January 1979).

12. Nuclear Regulatory Commission, Radioactive Materials Released from Nuclear Power Plants - Annual Report 1978, NUREG/CR-1497 (March 1981).

13. Nuclear Regulatory Commission, Reports to Congress on Abnormal Occurrences, NUREG-0090.

14. NUREG-0578, 'TMI Lessons Learned Task Force Status Report and Short-Term Recommendations,' July 1979.

Appendix A: Big Rock Point

Part 1. Forced Shutdowns and Power Reduction Tables

Table A1.1 1962 and 1963 Forced Shutdowns and Power Reductions at Big Rock Point

No.	Date (1962) to 1963	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	12/62 to 3/63		<1		Spurious period or flux trip.	B	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N2.4
2)	12/62 to 3/63		<1		Spurious period or flux trip.	B	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N2.4
3)	12/62 to 3/63		<1		Spurious period or flux trip.	B	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N2.4
4)	12/62 to 3/63		<1		Spurious period or flux trip.	B	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N2.4
5)	12/62 to 3/63		<1		Spurious period or flux trip.	B	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N2.4
6)	12/62 to 3/63		2		Low drum level (control on manual during testing).	B	3	Steam & Power (HA)	Turbine	
7)	12/62 to 3/63		2		Inadvertent simultaneous closure of reactor recirculating pump discharge valves.	C	3	Reactor Coolant (CB)	Valves	N6.0
8)	12/62 to 3/63		29		Low drum level (drum level transient during IPR adjustment).	B	3	Steam & Power (HA)	Instrumentation & Controls	N2.0

Table A1.1 (Continued)

No.	Date (1962) to 1963	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
9)	12/62 to 3/63		50		Low vacuum scram resulting from loss of station power.	A	3	Electric Power (EX)		D2.5
10)	12/62 to 3/63		58		High neutron flux (resulting from pressure transient caused by improper response of bypass valve during genertor trip tests).	B	3	Steam & Power (HE)		N1.2
11)	12/62 to 3/63		58		Main steam bypass system.	A	2	Steam & Power (HE)		N1.1
12)	12/62 to 3/63		6		Main steam bypass system.	A	2	Steam & Power (HE)		N1.1
13)	12/3/62		2		Accidental jarring of steam drum water level control panel.	G	3	Steam & Power (HE)	Instrumentation & Controls	N6.3
14)	2/17/63	2	33		Low steam drum level.		3	Steam & Power (HB)	Vessels	
15)	2/17/63	~20	6		Malfunction of rod drive B-5 due to collet finger assembly.		1	Reactor (RB)	Control Rod Drive Mechanism	D4.3
16)	2/20/63		50		Momentary loss of generator.	B	3	Steam & Power (HA)	Generators	D2.3
17)	4/12/63	4			Failure of a seal ring in the feedwater check valve at the steam drum.	A	1	Reactor Coolant (CH)	Valves	N1.1

Table A1.1 (continued)

No.	Date (1962) to 1963	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
18)	10/27/63				When resetting a channel scram annunciation, a rod scram occurred.	A	3	Instrumentation & Controls (IA)	Control Rods	N2.0
19)	11/5/63				Electrical short due to a water leak.	A	1	Instrumentation & Controls (IA)	Control Rods	N2.0

Table A1.2 1964 Forced Shutdowns and Power Reductions at Big Rock Point

No.	Date (1964)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	5/31	750	29		Spurious opening of the bypass valve.	A	3	Steam & Power (HE)	Valves	N1.1
2)	7/1				Spurious trip of channel 2 picoammeter coincided with test of channel 1.	B	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N7.0
3)	9/18		65		Spurious opening of the turbine bypass valve.	A	3	Steam & Power (HE)	Valves	N1.1

Table A1.3 1965 Forced Shutdowns and Power Reductions at Big Rock Point

No.	Date (1965)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	9/17	24	97		Turbine trip and reactor scram due to a loss of 138 Kv load due to a relaying malfunction.	A	3	Electric Power (EA)	Relays	D2.2
2)	9/30	18	97		Steam leak in turbine stage drain line.	A	1	Steam & Power (HA)	Pipes	N1.1
3)	10/30	?	97		Repair minor steam leaks under the turbine.	A	1	Steam & Power (HA)	Pipes	N1.1
4)	10/30	?	97		Modify 22 control rods. (10 others were modified during the shutdown in August.)	A	4	Reactor (RB)	Control Rods	N1.1

Table A1.4 1966 Forced Shutdowns and Power Reductions at Big Rock Point

No.	Date (1966)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1/18	120	97		Repair leaking tube in high pressure feedwater heater.	A	1	Reactor Coolant (CH)	Heat Exchangers	N3.1
2)	2/2		97-26		Power reduction. Tighten packing on valve in vent line from reactor to steam drum and repair recycle valve controls on No. 1 & 2 reactor feed pumps.	A	5	Reactor Coolant (CC) (CH)	Valves Instrumentation & Controls	N3.1 N2.0
3)	2/10		97-83		Power reduction. Fuel cladding failure.	A	5	Reactor (RC)	Fuel Elements	N4.0
4)	3/22	48	83		Remove valve in vent line from reactor to steam drum.	A	1	Reactor Coolant (CC)	Valves	N3.1
5)	4/1	48	83		Repair 4 leaking tubes in high pressure feedwater heater.	A	1	Reactor Coolant (CH)	Heat Exchangers	N3.1
6)	5/11	48			Replace cracked tee in control rod drive system.	A	1	Reactor (RB)	Pipes, Fittings	N1.1.1
7)	5/26	48			Repair 4 leaking tubes in high pressure feedwater heater.	A	1	Reactor Coolant (CH)	Heat Exchangers	N3.1
8)	6/2		97-89		Power reduction. Fuel cladding failure.	A	5	Reactor (RC)	Fuel Elements	N4.0
9)	6/18	34	89		Repair 4 leaking tubes in high pressure feedwater heater.	A	1	Reactor Coolant (CH)	Heat Exchangers	N3.1
10)	7/1	~8	75		Repair leaking tubes in high pressure feedwater heater.	A	1	Reactor Coolant (CH)	Heat Exchangers	N3.1

Table A1.4 (Continued)

No.	Date (1966)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
11)	7/13	~8	75		Repair leaking tubes in high pressure feedwater heater.	A	1	Reactor Coolant (CH)	Heat Exchangers	N3.1
12)	7/20	~8	75		Repair leaking tubes in high pressure feedwater heater.	A	1	Reactor Coolant (CH)	Heat Exchangers	N3.1
13)	7/26		75-49		Power reduction. Fuel cladding failure.	A	5	Reactor (RC)	Fuel Elements	N4.0
14)	8/3	24	49		Blank off high pressure feedwater heater tube sheet to eliminate tube leakage.	A	1	Reactor Coolant (CH)	Heat Exchangers	N3.1
15)	8/8	24	49	LTR 12/20/66	138 Kv breaker opened during a storm. The bypass valve opened but did not prevent pressure build-up and reactor scrammed on high pressure.	H	3	Electrical Power (EA)	Circuit Closures/ Interrupters	D2.2
16)	11/10		~95-55		Power reduction. Seals on No. 2 reactor recirculating pump failed.	A	5	Reactor Coolant (CB)	Pumps	D3.1
17)	11/12	~24	55		Examination and removal of No. 2 reactor recirculating pump.	[illegible]	1	Reactor Coolant (CB)	Pumps	N1.1
18)	12/15	72	69		Reinstallation of No. 2 reactor recirculating pump.	A	1	Reactor Coolant (CB)	Pumps	N1.1

Table A1.5 1967 Forced Shutdowns and Power Reductions at Big Rock Point

No.	Date (1967)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1/20	~12	97		Pressure transient caused by erratic operation of the turbine admission valve caused by the initial pressure regulator.	A	3	Steam & Power (HA)	Instrumentation & Controls	D2.1
2)	1/20	~156	Low		Reactor startup was delayed because rod drive E4 could not be withdrawn. Installed new drive.	A	4	Reactor (RB)	Control Rod Drive Mechanisms	N1.1
3)	1/26	~336	Low		Failure of turbine shaft-driven oil pump which operates the turbine admission valve.	A	1	Steam & Power (HA)	Pumps	N1.1
4)	2/10	26	97		Replaced defective control rod drive D-1.	A	1	Reactor (RB)	Control Rod Drive Mechanisms	N1.1
5)	2/12		53-0		Power reduction. Repair turbine initial pressure regulator.	A	5	Steam & Power (HA)	Instrumentation & Controls	N1.1
6)	2/13	7	?-0		Power reduction. Repair turbine initial pressure regulator.	A	5	Steam & Power (HA)	Instrumentation & Controls	N1.1
7)	2/14		?-0		Power reduction. Repair turbine initial pressure regulator.	A	5	Steam & Power (HA)	Instrumentation & Controls	N1.1
8)	2/16	7	97-0		Power reduction. Repair turbine initial pressure regulator.	A	5	Steam & Power (HA)	Instrumentation & Controls	N1.1

Table A1.5 (Continued)

No.	Date (1967)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
9)	2/17		?-0		Power reduction. Repair turbine initial pressure regulator.	A	5	Steam & Power (HA)	Instrumentation & Controls	N1.1
10)	3/10	8	96-0		Power reduction. Inspect the generator exciter brushes.	A	5	Steam & Power (HA)	Generators	N1.1
11)	3/10	~6	0		Error during instrument work.	B	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N5.0
12)	3/27	~6	96		Repair steam leaks n packing gland of butterfly valve on discharge of No. 2 recirculating pump.	A	1	Steam & Power (HX)	Pipes, Fittings	N3.1
13)	3/27	~1	0		Short period when attempting to raise reactor pressure.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N2.0
14)	4/14	21	96		Leaks in packing of the isolation valve for the west steam reference line to the drum level instrumenta-	A	1	Reactor Coolant (CC)	Valves	N3.1
15)	10/26	24	96		Repair steam leak in the bonnet of the high pressure bleeder trip valve.	A	1	Reactor Coolant (CC)	Valves	N3.1
16)	11/25	7	96		Replace offgas filter due to high differential pressure.	A	1	Radioactive Waste Management (MB)	Filters	N1.1.4

Table A1.5 (Continued)

No.	Date (1967)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
17)	12/?		100-78		Power reduction. Offgas activity to pressure fuel integrity.	A	5	Reactor (RC)	Fuel Elements	N4.0
18)	12/?		78-13		Power reduction. Make temporary repairs to stop steam leaks on the turbine trip valve to the high pressure heater.	A	5	Reactor Coolant (CC)	Valves	N3.1
19)	12/?		82-13		Power reduction. Make temporary repairs to stop steam leaks on the turbine trip valve to the high pressure heater.	A	5	Reactor Coolant (CC)	Valves	N3.1

Table A1.6 1968 Forced Shutdowns and Power Reductions at Big Rock Point

No.	Date (1968)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	OBE(D)/ NSIC(N) Event Category
1)	1/?		82-?		Power reduction. Repack No. 1 & 2 reactor feed pumps.	A	5	Reactor Coolant (CH)	Pumps	N1.1
2)	1/?		82-?		Power reduction. Repack No. 1 & 2 reactor feed pumps.	A	5	Reactor Coolant (CH)	Pumps	N1.1
3)	1/?		82-?		Power reduction. Repack No. 1 & 2 reactor feed pumps.	A	5	Reactor Coolant (CH)	Pumps	N1.1
4)	4/4	24			Reinstall No. 2 recirculating pump.	A	2	Reactor Coolant (CA)	Pumps	N1.1
5)	4/6		Low		Install new shaft seal cartridge in No. 2 recirculating pump.	A	1	Reactor Coolant (CA)	Pumps	N1.1
6)	4/7	~48	Low		Control rod drive B4 could not be withdrawn from the fully inserted position. It was replaced.	A	1	Reactor (RB)	Control Rod Drive Mechanisms	N1.1
7)	4/23	6	79		The high-sphere pressure sensors were accidentally bumped.	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N6.3
8)	6/3	~24	83		Repair leaks in unions adjacent to the explosive valves on the reactor poison system.	A	1	Reactor (RB)	Pipes, Fittings	N1.1
9)	6/12		83-75		Power reduction. Fuel cladding failure.	A	5	Reactor (RC)	Fuel Elements	N4.0

Table A1.6 (Continued)

No.	Date (1968)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
10)	6/13		75-68		Power reduction. Fuel cladding failure.	A	5	Reactor (RC)	Fuel Elements	N4.0
11)	9/10	~20	97		Repack No. 2 reactor recirculating pump butterfly valve.	A	1	Reactor Coolant (CB)	Valves	N3.1
12)	9/21	~21	97		High delta P in stack off-gas filter.	B	1	Radioactive Waste Management (MB)	Filters	N1.1.4
13)	10/13	~20	95		Repair 2 steam leaks and replace the high-pressure heater drain valve diaphram.	A	1	Steam & Power (HG)	Pipes, Fittings Valves	N3.1
14)	10/?	~8	Low		While returning to power, control rod B-5 could not be moved from notch 15.	A	1	Reactor (RB)	Control Rod Drive Mechanisms	D4.3
15)	10/?	~12			Inspect and replace O-rings in control rod flanges.	A	1	Reactor (RB)	Control Rod Drive Mechanisms	N1.1
16)	10/?	~12			Inspect and replace O-rings in control rod flanges.	A	1	Reactor (RB)	Control Rod Drive Mechanisms	N1.1
17)	11/6	23	95		Packing leak on the main steam bypass isolation valve.	A	1	Steam & Power (HE)	Valves	N3.1
18)	12/14		95-89		Power reduction. Fuel cladding failure.	A	5	Reactor (RC)	Fuel Elements	N4.0
19)	12/14	17	95		Packing leak on the steam supply to the condenser air ejectors.	A	1	Steam & Power (HC)	Pipes, Fittings	N3.1

Table A1.7 1969 Forced Shutdowns and Power Reductions at Big Rock Point

No.	Date (1969)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1/2		89-81		Power reduction. Fuel cladding failure.	A	5	Reactor (RC)	Fuel Elements	N4.0
2)	1/17	24	81		Steam leak in turbine stage drain heater.	A	1	Steam & Power (HA)	Heat Exchangers	N3.1
3)	2/18		81-70		Power reduction. Fuel cladding failure.	A	5	Reactor (RC)	Fuel Elements	N4.0
4)	3/1	29	68		Steam leak in valve packing on the air ejection supply line.	A	1	Steam & Power (HA)	Valves	N3.1
5)	3/3	~10	68		Excessive cooling water leakage at the D-3 control rod drive flange.	A	1	Reactor (RB)	Control Rod Drive Mechanisms	N1.1
6)	3/3	~18			Replace 3 control rod drives. Replace shaft seals on No. 1 reactor recirculating pump.	A	4	Reactor (RB)	Control Rod Drive Mechanisms	N1.1.4
7)	6/7	~24	69		Repack outside gland on the butterfly valve in the No. 1 reactor recirculating loop.	A	1	Reactor Coolant (CB)	Valves	N3.1
8)	6/21	~24	69		Repair 4 steam leaks in the turbine pipe tunnel area.	A	1	Steam & Power (HB)	Pipes, Fittings	N3.1
9)	7/7	~24	69		Repair leaks in the turbine stage drains and in the B-3 control rod drive cooling water connection.	A	1	Steam & Power (HA) Reactor (RB)	Pipes, Fittings	N3.1 N1.1

Table A1.7 (Continued)

No.	Date (1969)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
10)	8/11	~48	69		Repair steam leaks and inspect for known leakage in the turbine main condenser and core spray heat exchanger.	A	1	Steam & Power (HC) Engineered Safety Features (SF-D)	Heat Exchangers	N3.1 N1.1
11)	10/20	24	69	LTR 2/20/70	High conductivity of the primary coolant caused by previous mal-operation which resulted in over-heating the resin in the cleanup demineralizer.	G	1	Reactor Coolant (CH)	Deminer-alizers	N6.0
12)	11/5	~8	69		Repair steam leak in the turbine stage drain line to the inter-mediate pressure heater.	A	1	Steam & Power (HA)	Pipes, Fittings	N3.1

Table A1.8 1970 Forced Shutdowns and Power Reductions at Big Rock Point

No.	Date (1970)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1/8	~24	70		Replace off gas filter.	A	1	Radioactive Waste Management (MB)		N1.1
2)	3/30	~8			Turbine problems.	A	1	Steam & Power (HA)	Turbines	N1.1
3)	3/31	~8			Turbine problems.	A	1	Steam & Power (HA)	Turbines	N1.1
4)	4/1	~8	70		Minor adjustments to the turbine initial pressure regulator.	A	1	Steam & Power (HA)	Instrumentation & Controls	N2.0
5)	4/24		70		Leaking core spray heat exchanger tube.	A	1	Engineered Safety Features (SF-D)	Heat Exchangers	N1.1
6)	6/28	72	70		A fault in the 138 Kv transmission line caused a load rejection due to a severe storm. The reactor tripped on high pressure.	H	3	Electric Power (EA)	Circuit Closures/ Interrupters	D2.2
7)	10/5		70-7		Power reduction. Replace solenoid valve assembly on the dirty sump discharge isolation valve.	B	5	Radioactive Waste Management (MA)	Valves	N1.1
8)	10/7	24	70		Repack main steam bypass valve.	A	1	Steam & Power (HC)	Valves	N3.1

Table A1.8 (Continued)

No.	Date (1970)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
9)	11/13	24	70	LTR 12/1/70	Plug 3 tubes in the post incident heat exchanger.	A	1	Engineered Safety Features (SB)	Heat Exchangers	N3.1
10)	11/14	4	Low		Erratic operation of the period amplifier in the channel 4. Log N neutron monitoring equipment caused a short period scram.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N2.4
11)	12/3	10	70		A fault in the 138 Kv transmission line caused a load rejection due to a severe storm. The reactor tripped on high pressure.	H	3	Electric Power (EA)	Circuit Closures/ Interrupters	D2.2

Table A1.9 1971 Forced Shutdowns and Power Reductions at Big Rock Point

No.	Date (1971)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1/23	40	70		Repair turbine condenser tube leaks and a steam leak in the intermediate - pressure heater line.	A	1	Steam & Power (HC)	Heat Exchangers (Condensers)	N3.1
2)	2/2	216	70-63		High seal temperature on the No. 2 recirculating pump.	A	5	Reactor Coolant (CB)	Pumps	N1.1
3)	4/20	24	70		Repair steam leak from the packing of the butterfly valve located on the discharge of the No. 1 reactor recirculating pump.	A	1	Reactor Coolant (CB)	Valves	N3.1
4)	4/29	24	70		Make adjustments to the turbine initial regulator.	A	1	Steam & Power (HA)	Instrumentation & Controls	N1.1
5)	5/12	21	70		Load rejection due to a fault in the 138 Kv transmission line caused by a corner strain pole which had been cut half way through and a guy wire which had been cut.	H	3	Electric Power (EA)	Circuit Closures/ Interrupters	D2.2
6)	6/2	25	70		Steam leak in the turbine stage drain piping to the high pressure heater.	A	1	Steam & Power (HA)	Pipes, Fittings	N3.1
7)	9/22	18	70		Loss of all major rotating equipment due to accidental tripping of the 2400 volt station power relays.	G	3	Electric Power (EB)	Circuit Closures/ Interrupters	D2.3
8)	9/28	14	70		High flux scram following loss of the 138 Kv transmission line attributed to a local storm.	H	3	Electric Power (EA)	Circuit Closures/ Interrupters	D2.2

Table A1.9 (Continued)

No.	Date (1971)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
9)	10/18		70-53		Power reduction. Failure of No. 2 reactor recirculating pump seals necessitated pump shutdown.	A	5	Reactor Coolant (CB)	Pumps	D3.1
10)	10/23	33	57		Shutdown to replace the No. 2 recirculating pump seal cartridge.	A	1	Reactor Coolant (CB)	Pumps	N1.1
11)	11/26	11	57		Failure of the linkage arm of the turbine trip solenoid caused a turbine and generator trip.	A	1	Steam & Power (HA)	Instrumentation & Controls	D2.3

Table A1.10 1972 Forced Shutdowns and Power Reductions at Big Rock Point

No.	Date (1972)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1/25	80	55	LTR 3/3/72	Turbine trip on overspeed due to no load. This was caused by the Big Rock Point relaying scheme not clearing when a system line fault occurred.	A	3	Electric Power (EX)	Relays	D2.2
2)	2/11	8	53		Adjust the initial-pressure regulator which would not regulate the turbine control valves effectively at low power.	A	1	Steam & Power (HB)	Instrumentation & Controls	N2.0
3)	5/15	60	70		Primary coolant leakage at the B 5 control rod drive flange. During maintenance the teflon O-ring had been replaced with a new type silver plated inconel O-ring.	A	1	Reactor (RB)	Pipes, Fittings	N1.1.3
4)	5/18-5/19		70-?		Several power reductions to isolate a leak into the component cooling water system. The leak was traced to the No. 1 reactor recirculating water pump seal cooling water heat exchanger.	A	5	Reactor Coolant (CB)	Heat Exchangers	N3.1
5)	5/19		70-67		Power reduction. Shutdown No. 1 reactor recirculating water pump due to leaking heat exchanger.	A	5	Reactor Coolant (CB)	Heat Exchangers	N3.1
6)	6/10	15	67		Replace No. 1 reactor recirculating pump seal heat exchanger.	A	1	Reactor Coolant (CB)	Heat Exchangers	N1.1
7)	6/17	~20	67		Replace seal cartridge on No. 1 reactor recirculating pump.	A	1	Reactor Coolant (CB)	Pumps	N1.1

Table A1.10 (Continued)

No.	Date (1972)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
8)	6/18	~8	71		Replace leaking offgas rupture diaphram.	A	1	Radioactive Waste Management (MB)	Pipes, Fittings	N1.1
9)	7/6	3	83		Low drum level scram due to inability to maintain an adequate feedwater supply during a load rejection test.	B	3	Reactor Coolant (CH)		D2.7
10)	7/27		83-70		Power reduction. Scram valves were inadvertently opened while working on a scram valve solenoid. This caused rod drive E-1 to fully insert.	B	5	Reactor (RB)	Control Rod Drive Mechanisms	N1.1
11)	7/29	30	83		Repack the turbine main steam bypass valve.	A	1	Steam & Power (HE)	Valves	N3.1
12)	9/30	40	83		Repair steam leak on the turbine high pressure extraction line.	A	1	Steam & Power (HA)	Pipes, Fittings	N3.1
13)	11/6		83-13		Power reduction. Pump bearing failure caused the clean-up system pump to fail.	A	5	Reactor Coolant (CG)	Pumps	N1.1
14)	11/8		83-13		Power reduction. Replace clean-up system pump due to bearing failure.	A	5	Reactor Coolant (CG)	Pumps	N1.1
15)	11/12	~8	Low		Short period scram because of a high notch worth in sequence during withdrawal of control rod B-5.	C	3	Reactor (RB)	Control Rod Drive Mechanisms	D4.3
16)	11/23	33	83		Excessive leakage through the O-ring on control rod drive C-5.	A	1	Reactor (RB)	Control Rod Drive Mechanisms	N1.1

Table A1.10 (Continued)

No.	Date (1972)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
17)	12/16	26	83	LTR 3/23/73	Repair a leak in the feedwater line blank/flange.	A	1	Reactor Coolant (CH)	Pipes, Fittings	N3.1
18)	12/30		?-68		High activity in plant off gas. (Fuel cladding failures.)	A	5	Reactor (RC)	Fuel Elements	N4.0

Table A1.11 1973 Forced Shutdowns and Power Reductions at Big Rock Point

No.	Date (1973)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1/20	25	66		Leak in the packing of the reactor cleanup system discharge valve to the No. 1 reactor recirculating pump discharge piping.	A	1	Reactor Coolant (CG)	Valves	N3.1
2)	5/3		91-39		Power reduction. System substation work.	B	5	Electric Power (EB)	Other (XX)	N9.0
3)	5/12		91-39		Power reduction. System substation work.	B	5	Electric Power (EB)	Other (XX)	N9.0
4)	5/21		91-83		Power reduction. Flux tilting test to determine location of leaking fuel bundles.	B	5	Reactor (RC)	Fuel Elements	N4.0
5)	6/29		91-87		Power reduction. In-core detectors No. 12 and No. 14 were alarming. Later tests indicated that no thermal limits had been exceeded and these were recalibrated.	A	5	Instrumentation & Controls (IB)	Instrumentation & Controls	N2.3
6)	7/20		92-3		Power reduction. Leak in component coolant line to the motor thrust bearing of No. 2 recirculating pump.	A	5	Auxiliary Water (WB)	Pipes, Fittings	N3.1
7)	8/16		92-13		Power reduction. Leak in flex line from heat exchanger on the recirculating pump.	A	5	Auxiliary Water (WB)	Pipes, Fittings	N3.1

Table A1.11 (Continued)

No.	Date (1973)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
8)	9/19		92-13		Power reduction. Cleanup pump stopped and could not be re-started. Reduced power to enter the recirculating pump room to isolate the cleanup system.	A	5	Reactor Coolant (CG)	Pumps	N1.1
9)	9/22		92-13		Power reduction. Enter recirculating pump room to valve the cleanup system into service after having replaced cleanup pump.	A	5	Reactor Coolant (CG)	Pumps	N1.1
10)	12/3		92-76		Power reduction. High offgas release rate.	A	5	Reactor (RC)	Fuel Elements	N4.0
11)	12/6		76-70		Power reduction. High offgas release rate.	A	5	Reactor (RC)	Fuel Elements	N4.0
12)	12/8	72	70		Packing failure on the level instrumentation lower root valve at east end of reactor steam drum.	A	1	Reactor Coolant (CH)	Instrumenta-tion & Controls	N2.0
13)	12/8				Leaking tubes on the emergency condenser and modify baffle plater.	A	4	Engineered Safety Features (SB)	Heat Exchangers	N1.1

Table A1.12 1974 Forced Shutdowns and Power Reductions at Big Rock Point

No.	Date (1974)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
	(continuation)									
	12/8/73	253			Repair emergency condenser. Modify baffles in inlet water box.	A	4	Engineered Safety Features (SB)	Heat Exchangers	N1.1
1)	5/5		98-93		Power reduction. Flooding of intermediate pressure feedwater heater and condenser vacuum upset.	A	5	Reactor Coolant (CH)	Heat Exchangers	N1.1
2)	5/17		95-83		Power reduction. Fuel cladding failure.	A	5	Reactor (RC)	Fuel Elements	N4.0
3)	5/20		83-70		Power reduction. Fuel cladding failure.	A	5	Reactor (RC)	Fuel Elements	N4.0
4)	6/2	48	70		Steam leak on 3 in. drain line from HP section of turbine to HP feedwater heater.	A	1	Steam & Power (HH)	Pipes, Fittings	N3.1
5)	6/5	744		UE74-07 UE74-08	Control rod drives stuck. Other maintenance performed.	A	4	Reactor (RB)	Control Rod Drive Mechanisms	N1.1
6)	9/28	12	83-64		Power reduction. Remove No. 1 condensate pump for replacement of two upper motor thrust bearings.	A	5	Steam & Power (HC)	Pumps	N1.1
7)	10/6		83-?		Power reduction. Failure of another in-core detector. This reduced the number of operational detectors to 10. Plant was placed in coastdown mode.	A	5	Instrumentation & Controls (ID)	Instrumentation & Controls	N1.1

Table A1.12 (Continued)

No.	Date (1974)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/NSIC(N) Event Category
8)	11/21	7	83-13		Power reduction. Repair turbine intermediate pressure extraction line to intermediate pressure feedwater heater.	A	5	Steam & Power (HC)	Pipes, Fittings	N3.1
9)	11/23	11	88-13		Power reduction. Repair turbine intermediate pressure extraction line to intermediate pressure feedwater heater.	A	5	Steam & Power (HC)	Pipes, Fittings	N3.1

Table A1.13 1975 Forced Shutdowns and Power Reductions at Big Rock Point

No.	Date (1975)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1/7		83-80		Power reduction. Encroachment of the 90% MAPLHGR limit on "F" type fuel.	F	5	Reactor (RB)	Fuel Elements	N8.0
2)	1/16	3421	80	AO-1-75 (1-27-75)	Unit was shut down when it was found that design and QA deficiencies existed in instrumentation for the post incident cooling system.	D	1	Engineered Safety Features (SB)	Instrumentation & Controls	N8.3
3)	9/25	48	80-70		Power reduction. Repair a ground in a wiring junction box to No. 2 condensate pump motor.	A	5	Steam & Power (HC)	Electrical Conductors	N1.1
4)	10/19		80-42		Power reduction. Modifications to the Livingston substation.	H	5	Electric Power (EA)	Transformers	N9.0
5)	10/19		42-11		Power reduction. The turbine bypass valve opened partially due to failure of the initial pressure regulator. Turbine governor control was also unresponsive.	A	5	Steam & Power (HE)	Instrumentation & Controls	N1.1
6)	10/30	6	80-7		Power reduction. Leak in HP stage drain line from HP turbine to HP heater. IPR failed during power reduction.	A	5	Steam & Power (HE) (HA)	Pipes, Fittings	N3.1 N1.1
7)	11/13	45	83		Plug leaking tubes in main condenser.	A	1	Steam & Power (HC)	Heat Exchangers	N3.1

Table A1.13 (Continued)

No.	Date (1975)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
8)	12/3		72-?		Power reduction. Attempt to repair leak in high pressure turbine casing reducer.	A	5	Steam & Power (HA)	Pipes, Fittings	N3.1
9)	12/6	50	74	AO-75-27	Repair leak in high pressure turbine casing reducer and perform control rod drive testing.	A	1	Steam & Power (HA)	Pipes, Fittings	N3.1

Table A1.14 1976 Forced Shutdowns and Power Reductions at Big Rock Point

No.	Date (1976)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1/31	3215	~70		Installation of the Reactor Depressurization system and minor modification to the ECCS.	D	1	Engineered Safety Features (SF)	Pipes, Fittings	N8.0
2)	7/28	18	Low		Pinhole leak in valve on air ejector system.	A	1	Radioactive Waste Management (MB)	Valves	N3.1
3)	8/11	66	88		The TG Initial Pressure regulator failed resulting in high flux and a reactor trip.	A	3	Steam & Power (HA)	Circuit Closures/ Interrupters	D2.1
4)	11/22	24	88-69		Power reduction. Repack No. 1 reactor feed pump inboard shaft seal.	A	5	Reactor Coolant (CH)	Pumps	N1.1

Table A1.15 1977 Forced Shutdowns and Power Reductions at Big Rock Point

No.	Date (1977)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	2/6	7	85-42		Power reduction. Repair steam leak on the turbine extraction line of the IP heater.	A	5	Steam & Power (HA)	Pipes, Fittings	N3.1
2)	4/4	24	85-7		Power reduction. Investigate abnormal noise in No. 2 reactor feed pump.	H	5	Reactor Coolant (CH)	Pumps	N1.1
3)	10/29	88	80		Turbine control problems.	A	1	Steam & Power (HA)	Turbines	N2.0

Table A1.16 1978 Forced Shutdowns and Power Reducations at Big Rock Point

No.	Date (1978)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1/13	484			Repairs to control rod drive B4.	A	1	Reactor (RB)	Control Rod Drive Mechanisms	N1.1
2)	3/20		90-?	RO 78-16	Power reduction. Investigate source of water leakage. Visual inspection indicated that it was from CRD cooling flange O-rings.	A	5	Reactor (RB)	Pipes, Fittings	N1.1
3)	4/7	43	90	RO 78-18	Faulty tone relaying equipment resulted in the opening of the 199 OCB even though the 138 KV power line remained energized. Reactor scrammed on low condenser vacuum.	A	3	Electric Power (EB)	Circuit Closures/ Interrupters	D2.2
4)	4/15		~90-~50	RO 78-21	Power reduction. Modification to the Emmett Substation.	H	5	Electric Power (EA)		N9.0
5)	4/25		90-51		Power reduction. Loss of tone relaying equipment due to a brush fire off site.	H	5	Other (XX)		N9.0
6)	5/31	22	90		Wiring error during modification to an offsite substation resulted in tripping breaker to 138 KV line.	H	3	Electric Power (EA)	Circuit Closures/ Interrupters	D2.2
7)	9/4	111	Low	RO 78-035	Unacceptable test results for containment supply ventilation valve leak rate test. Valves were repaired.	A	1	Engineered Safety Features (SB)	Valves	N1.1
8)	9/9	1281	Low	LER 78-038	Control rod drive problems - high temperature encountered.	A	2	Reactor (RB)	Control Rod Drive Mechanisms	N1.1

Table A1.17 1979 Forced Shutdowns and Power Reductions at Big Rock Point

No.	Date (1979)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	2/2	18	87	LER 79-001	Replace valve disc with modified design, after unacceptable leak rate test on containment ventilation valve.	A	2	Engineered Safety Features (SA)	Valves	N1.2
2)	4/17	315	Low	LER 79-018	High pressure reactor trip caused by the turbine bypass valve failing to open.	A	3	Steam & Power (HE)	Valves	N1.1
3)	4/17	4847		LER 79-020	Correct inlet diffuser vibration problem in reactor vessel and repair leak in CRD housing.	A	4	Reactor (RA)	Diffusers	N1.1
4)	11/6	54	Low		Replace recirculating pump seal.	A	1	Reactor Coolant (CB)	Pumps	N1.1
5)	11/6	3			Repair leaks in turbine bypass drain line.	A	4	Steam & Power (HE)	Pipes, Fittings	N3.1
6)	12/31	2			Regulatory shutdown for checking relief valve position. Manual reset of containment isolation and radiation monitors.	D	1	Instrumentation & Controls (IB)	Instrumentation & Controls	N8.0

Table A1.18 1980 Forced Shutdowns and Power Reductions at Big Rock Point

No.	Date (1980)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/NSIC(N) Event Category
	(continuation) 12/31/79	296			Regulatory shutdown to implement requirements of NUREG-0578.	D	4	Instrumentation & Controls (IB)	Instrumentation & Controls	N8.0
1)	1/13	4	Low		Failure of intermediate power range monitor.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N2.0
2)	1/13	5	Low		Failure of intermediate power range monitor.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N2.0
3)	1/13	15	Low		Intermediate power range trip on period due to prompt effect.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N2.0
4)	1/15	15			Failure of intermediate pressure regulator.	A	3	Steam & Power (HA)	Instrumentation & Controls	D2.1
5)	1/18				Power reduction. Intermediate pressure regulator test.	A	5	Steam & Power (HA)	Instrumentation & Controls	N1.1
6)	4/17	27	88-?		Power reduction. Repair piping in high pressure turbine drain line.	A	5	Steam & Power (HA)	Pipes, Fittings	N3.1

Appendix A: Big Rock Point

Part 2. Reportable Event Coding Sheets

Table A1.18 (Continued)

No.	Date (1980)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
7)	4/30	9	88-?		Power reduction. Piping repair downstream of turbine bypass valve.	A	5	Steam & Power (HE)	Pipes, Fittings	N3.1
8)	5/9		88-?		Power reduction. Loss of reactor recirculating pump.	A	5	Reactor Coolant (CB)	Pumps	D3.1
9)	9/20	10	88-?		Power reduction. Repair to intermediate pressure turbine drain line.	A	5	Steam & Power (HA)	Pipes, Fittings	N3.1
10)	9/27	3	88-?		Power reduction. Take cleanup pump out of service for repairs.	A	5	Reactor Coolant (CG)	Pumps	N1.1

Table A2. 1 Coding Sheet for Reportable Events at Big Rock Point - 1966

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
66-1	14892	050166	122066	C	RB	Z,J	-	B	AI,AG	B,C	C4	Cracking in CRD hyrraulic system and two CRs fail to withdraw (24646).
66-2	10568	062266	070466	D	RB	I,J	-	B	AU	B	N	Leak into CRD would unlock collet allowing CR to drift out.
66-3	11038	080866	122066	E	EC,HE	F	-	B	BJ,OD	I,F	S7	Loss of offsite power and rupture of condenser rupture diaphragm (14893).
66-4	16521	120066	120066	-	CE	DD	-	A	AK	E	N	New approach to recirculation pump maintenance.
66-5	-	120066	061781	-	CG	Z	-	A	BO,AV	D	N	Parts of clean-up system piping replaced due to cracks.
66-6	23393	120066	122066	B	CH	MM	-	B	AU	D	N	Feedwater heater tube failures.

Table A2. 2 Coding Sheet for Reportable Events at Big Rock Point - 1967

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
67-1	16522	012067	010067	E	BE	J	-	B	BR,AG	D,G	N	Reactor scram due to pressure transient, restart was inhibited by jammed CRD.
67-2	27476	050067	050067	C	RC	R	-	C	AQ	D	C7	Fuel element leaking due to crud (27477).
67-3	19274	091267	110667	-	HC	OO	-	-	AW,OD	D	C3	Off gas system had leaky diaphram, exposure to worker fixing it.
67-4	22828	122567	010868	B	BE	I,J	-	B	AG	D	C7	CRD rod F-5 would not withdraw but would insert.
67-5	24201	122567	030568	E	BE	I,J	-	E	AG	D	C7	Rod F-5 jammed by piece of steel.

Table A2. 3 Coding Sheet for Reportable Events at Big Rock Point - 1968

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
68-1	30032	040068	043068	E	HC	H	-	B	AU	D	C7	2 fuel bundles leaked.
68-2	25305	040668	042368	D	RB	I,J	-	B	AG	D	C7	Rod B-4 would not withdraw
68-3	31307	062468	071168	D	CB	DD,OO	-	A	AW,OD	A	C3	Personnel overexposed during repair of recirculation pump.
68-4	33048	073068	070068	-	CB	BB	-	B	BU	D	N	High steam drum conductivity.
68-5	61319	110068	121671	E	HC	H	-	B	AQ,BL	B	N	Crud buildup causes fuel failure.
68-6	-	113068	020469	-	EE	N	-	C	BD,CA	B	N	DG linkage pin designed wrong.
68-7	31010	120068	122768	E	HC	H	-	B	AU	D	C7	Fuel elements leak and power reduced due to off-gas.

Table A2. 4 Coding Sheet for Reportable Events at Big Rock Point - 1969

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
69-1	-	102069	022070	R	CX	-	A,O	R	RG	D	N	Alarm circuit on recorder failed to warn RO of high coolant temperature.

Table A2. 5 Coding Sheet for Reportable Events at Big Rock Point - 1970

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
70-1	42001	020070	022470	-	IA	-	I	-	BC	B	N	Moved water level sensors to area of lower radiation for accessibility.
70-2	57230	080670	100870	B	EE	N,T	C	C	EG	D	C1	Diode failure caused DG to fail to develop proper voltage.
70-3	60903	111370	120170	D	HC	H,MM	-	D	AU,OH	D	C3	Condenser tube leaks and noncondensible gas drawn into cooling water.

Table A2. 6 Coding Sheet for Reportable Events at Big Rock Point - 1971

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
71-1	64240	020071	061771	C	CG	Z	-	A	EO,AY	D	N	Section of cleanup system piping replaced due to cracks (65548).
71-2	74353	030271	032671	C	RB	I,J	-	C	ED,AG	D	N	CRD stuck in inserted position due to roller being stuck in drive.
71-3	63790	052671	060771	B	RE	I	-	B	AG	D	N	Control rod C-3 would not withdraw but would insert.
71-4	65547	071571	081[illegible]71	B	EE	N	-	C	BE,BL	D	C1	DG fails to run due to high cooling water temperature.

Table A2. 7 Coding Sheet for Reportable Events at Big Rock Point - 1972

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
72-1	39024	012572	030372	E	EA	00	-	B	EF	I	S7	Off-site power lost during storm and switchgear malfunctioned.
72-2	71399	032872	051172	C	RB	00	-	C	AM	B	C4	Liquid poison system explosive valve fails to fire.
72-3	70037	040072	041972	C	RC	-	F	A	AR	B	N	Outer encapsulation of neutron sources fail.
72-4	73801	052572	062372	B	EE	N	-	C	ED,BC	H	C1	Diesel generator fails to start due to low pressure set point.
72-5	72453	061072	062672	B	MC	00	-	C	CA	D	N	Off gas isolation valve fails to seal.
72-6	75136	072972	091372	E	RE	G	-	-	BL	B	N	Failure of startup channels due to faulty cable.
72-7	74355	082872	090172	E	MC	Z	-	C	EU	D	N	Off gas system holdup time shorter than expected (75077).
72-8	75973	083172	092672	E	SD	00	-	-	BB	C	N	Containment isolation valve fails to open due to faulty solenoid.
72-9	77446	112372	122072	B	SD	00	-	B	BB	C	N	Containment isolation valve fails to open due to solenoid failure.
72-10	77861	121672	032373	B	CE	H	-	B	AX	D	C3	Leak into emergency condenser secondary yields radiation release.

Table A2. 8 Coding Sheet for Reportable Events at Big Rock Point - 1973

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
-	79595	030073	032673	C	BC	F	-	-	AL	B	N	Cobalt target rods become loose.
-	80131	030373	040573	E	CD	OO	-	-	EB	D	N	MSIV packing was binding the valve stem.
A07308	74830	040573	050873	C	-	-	T	-	EH	D	N	Time delay relay switch set point drift.
A07303	74830	040573	050873	C	CE	OO	-	-	BA	G	N	Emergency condenser outlet valve fails to open.
A07307	74830	040573	050873	C	HC	-	T	-	EH	D	N	High condensor pressure switch set point drift.
A07304	74830	040573	050873	C	SD	-	T	-	EH	D	N	Reactor enclosure high pressure switch set point drift.
A07306	74830	040573	050873	C	SD	-	T	-	EH	D	N	Reactor building vacuum relief pressure switch set point drift.
A07305	74830	040573	050873	C	SFC	-	T	-	EH	D	N	High reactor pressure scram switch set point drift.
A07313	80732	041973	051873	B	EE	N	-	C	EL,AW	D	C1	DG shutdown due to high coolant temp.
-	74354	050073	050273	-	MC	-	-	E	OD	A	C3	Radiation level at control fence is high.
-	87052	101773	112773	B	SD	OO	-	C	AU	D	N	Sphere vent valve operator reserve nitrogen supply leaked.
-	85568	102373	112073	E	FA	R	-	-	CK	A	N	Spent fuel rod found on spent fuel pool floor (91119).
A07311	85590	110173	110273	D	MC	OO	-	-	BC	A	N	Stack off-gas isolation valve left open.

Table A2. 8 (continued)

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
A07312	85590	110273	111473	D	HH	-	T	E	EI	G	N	Calibration errors.
A07314	85573	110273	111473	D	ID	-	L	C	EI,OC	A	N	Instrument calibration errors on neutron-monitoring system.
A07313	85590	110373	111373	D	ID	-	L	E	EF	G	N	Calibration errors.
A07316	87053	111173	112673	B	SF	H	-	-	AU,AE	D	N	Leak in emergency condenser tubes, divider plate warped (87091).
-	88106	113073	012174	D	MC	OO	-	B	OA,AW	B	N	Off-gas isolation valve still leaking.

Table A2. 9 Coding Sheet for Reportable Events at Big Rock Point - 1974

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
-	88330	010874	021174	D	BE	P	N	B	EC	E	N	Stack gas effluen. monitor installed wrong.
A07401	89266	030174	031274	B	ID	-	L	B	EG	D	N	Neutron flux level instrumentation malfunctions.
A07402	89319	030774	031874	B	EE	N	-	C	ED	G	C1	Diesel generator fails to start.
A07403	89745	032374	040374	D	BB	J	-	C	EI	B	N	CRD withdrawal time less than limit.
A07404	89747	033174	040474	D	IB	-	-	C	CC	A	N	Reactor protection logic system test performed 5 days late.
UE7402	90650	033174	043074	D	BB	I,J	-	B	HD,AG	D	N	Control rod blade lower roller came loose and CRD stuck.
A07405	-	040474	040574	C	CE	U	-	-	CK	H	N	Failed to check core spray heat exchanger as required.
UE7404	91120	040674	050774	D	BC	-	F	B	BO	C	N	Anomalies in cobalt distribution in target rods.
A07406	90374	040774	041774	C	SFD	-	T	B	AW	D	N	Backup core spray system pressure switch leaks water.
-	90576	041074	041874	C	BE	I	-	-	AC	C	N	Fabrication error on several control rods.
A07407	90577	041174	042374	C	EE	N	-	C	CA	G	C1	Diesel generator starting motor mechanism fails.
A07408	-	041174	041674	-	HC	OO	-	B	BC	E	N	Off-gas drain valve improperly adjusted.
A07409	91000	042374	050374	C	HC	OO	-	C	CA	B	N	Off-gas isolation valve fails to seat properly.

Table A2. 9 (continued)

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
A07404	91121	042674	050674	C	SD	OO	-	C	AX,AL	E	N	Vent valve leaks due to improper installation.
A07410	-	050074	050674	D	SD	OO	-	C	AX,BC	G	N	Containment vent valve flange bolt not tightened and leaks.
A07411	91150	050174	051374	C	SD	-	T	C	EH	B	N	Containment vacuum pressure switch set point drifts.
A07412	91151	050374	051374	D	-	-	-	-	CK	A	N	Startup checklist not completed just prior to critical approach.
A07413	91147	050474	051674	D	RB	J	-	C	EI	D	N	CRD withdrawal time less than limit.
A07414	91667	050774	052374	B	ID	-	F	A	CJ	H	N	During irradiation of flux wires, reactor power increased.
A07415	92611	053174	061074	B	EE	N,DD	-	C	EE,AR	D	N	DG transfer pump fails due to key on pump shaft corroding.
UE7406	92438	060074	061074	-	SD	E		C	HA	B	N	High flow on plant exhaust fan.
A07416	92612	060374	061374	D	RB	J	-	C	EI	D	N	CRD withdrawal time still less than limit.
A07417	92613	060474	061474	D	IB	-	T	C	AR	D	N	Scram dump tank level switch fails.
UE7407	94371	060474	070574	D	RB	J	-	C	AG	D	N	CRDs stick due to wedged rollers and bolts (UE7408).
UE7410	94750	060774	072674	C	RA	O	-	A	AD	G	N	Reactor baffle plate latching bolts shear.
A07419	94393	060874	072674	D	RB	J,DD	-	B	AT	D	N	Water to CRD pump exceeded drain capacity resulting in flooding.

Table A2. 9 (continued)

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
UE7409	90651	060874	070374	C	RC	R	-	A	AR	D	N	Neutron source material in vessel accelerated fuel degradation (94372).
AO7418	-	061274	061374	C	MX	R	-	C	CK	H	N	Dry sipped wrong fuel bundle.
AO7421	-	070774	072374	C	-	-	-	-	CK	A	N	Failed to report UE7410 within 30 days.
AO7422	94751	071274	072674	C	SD	OO	-	A	CK,OC	A	N	Solenoid valves replaced but not tested for integrity.
AO7420	94752	071574	072574	C	SHB	OO	-	A	OK	A	C8	Post-incident system supply root valves tagged out during refueling.
UE7411	94915	071874	081674	C	IC	-	P	B	EY	D	N	Relay burned causing coil to overheat closing isolation valves.
AO7423	95542	091774	092774	B	SD	OO	-	C	CK	A	N	Test fixture on containment emergency escape lock left installed on lock.
UE7412	97138	101574	111474	-	HC	-	C	E	OK	B	C4	Off-gas flow recorder to be rescaled to conform with correct specifications.
UE7413	97513	102274	112174	-	FD	R	-	C	CK	A	N	Higher enriched fuel than expected inserted in a fuel rod.
AO7424	97496	110774	111874	B	EE	C	R	C	EA	C	C1	Defective diode causes battery charger to fail.
AO7425	97742	111474	112674	B	EE	N	-	C	ED	A	C1	DG did not start due to corroded battery terminals.

Table A2.10 Coding Sheet for Reportable Events at Big Rock Point - 1975

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
AO7501	93277	011675	012775	E	IE	-	I,M	B	CA	B	C4	Reactor water level sensors and pressure sensors design deficient.
UE7501	99662	011775	021175	D	RE	J	-	A	AG	G	N	CRD jammed in fully inserted position.
AO7503	93505	011875	013175	D	RB	J	-	C	EI	D	N	CRD withdrawal time less than limit.
UE7502	93506	012275	013175	D	SD	Z	-	A	OD,OH	A	C3	Radioactive water poured down floor drain.
AO7502	100044	012375	022175	D	RA	O	-	A	AL	B,E	C4	Bar beam clamplock bolt missing on locking device for lower grid bars.
AO7504	93504	012475	020375	-	FD	R	-	-	CK	A	N	Safety evaluation of dry sipping technique to be reperformed.
AO7505	-	013075	021075	D	CC	Z	-	C	AV	E	N	Weld defect in main steam line.
AO7507	100043	020675	022475	D	CE	Z	-	B	AC	C	C4	Emergency condenser outlet pipe cracked.
AO7508	101151	031875	033175	D	SD	OO	-	C	AU,BC	G	N	Containment vent valve leaks during test.
AO7509	102299	041075	042175	D	EE	N	-	C	ED	D	C1	DG fails to start.
RO7603	112729	051675	040576	C	SFD	Z	-	-	AO	G	N	Defective weld in core spray piping.
AO7511	103070	052075	053075	D	HC	-	T	C	EH	B	C4	Condenser pressure switch cannot be set low enough.
AO7512	103186	052175	060275	D	SFD	OO	-	C	AL	B	C4	Core spray valve operator lock nuts loose.

Table A2.10 (continued)

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
A07513	103187	052375	060275	D	CE	OO	-	C	AL	B	C4	Emergency condenser outlet valve operator lock nut loose.
A07514	103206	052675	060575	D	SHA	OO	-	C	AU,AI	D	N	Containment vent supply valve leaks.
A07515	103482	053075	061075	D	SD	OO	-	C	AU	A,C	N	Containment isolation valve leaks.
A07516	104210	053075	071775	D	SI	OO	-	-	OK	A	N	Valve inspections and repair procedures being reviewed.
UE7503	103702	060675	063075	D	SHE	DD	-	-	OK	B	C4	Procedures for post-incident system conflict with core spray system.
A07522	106452	070075	092275	E	EB	F	-	A	OK	A	N	During construction power moved from one panel to another.
A07517	104809	071875	073175	B	MA	-	-	-	OC	H	N	Discharge canal water not analyzed due to no sample taken.
UE7504	105553	072575	082575	B	RB	J	-	C	AG	D	N	CRD would not withdraw further.
A07518	105842	082575	090475	E	RE	I	-	-	CK	A	N	Control rod worth calculations contain errors.
UE7505	106453	083075	092375	E	MC	C	-	C	EG	D	N	Off-gas monitor failed to trip on signal.
A07519	106299	090075	091875	B	CG	OO	-	B	CK	B	C4	Valves rated lower than design limits require.
A07520	106297	090975	091975	E	EF	AA	-	B	CK	A	N	Load added to light panel due to unapproved circuit change.

Table A2.10 (continued)

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
A07521	106298	090975	091975	E	SD	OO	-	-	CK	A	C8	Containment isolation valves not tested properly due to plant drawing errors.
A07523	106986	092575	100975	E	FX	B	-	-	OK	A	N	Unlicensed reactor fuel received.
A07526	108251	102475	112475	B	AB	Z	-	A	OK	A	N	Changes to fire system without authorization.
A07524	100082	110075	111775	-	EX	F	-	-	CK	A	N	No analysis performed on additional load to breaker.
A07525	108250	111375	112475	D	HC,RB	H,A	-	A	CK	H	C8	Reactor pressure reduced for work on condenser and accumulator to CRD removed in violation.
A07527	108805	120675	121675	D	RB	J	-	C	OK	H	N	CRD scram tests performed without use of written procedures.
A07529	108807	120675	121675	D	RB	Z	-	C	AU,AI	D	N	CRD pump discharge piping leaks.
A07528	108806	120775	121675	D	RB	J	-	C	EI	D	N	CRD withdrawal time less than limit.
A07530	109196	121875	122675	B	-	-	-	-	CK	A	N	Construction crew began digging without a work package.

Table A2.11 Coding Sheet for Reportable Events at Big Rock Point - 1976

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
T7601	110357	011976	020276	E	SD	-	M,T	C	OA	B	C1	Pressue sensors have too low design pressure rating.
LER7602	111650	020276	030176	C	ED	F,OO	-	A	CK,BG	H	N	Power supply to core spray valves not tagged out as required.
RO7604	113200	032476	041576	C	EE	N	-	C	BE	D	C1	DG tripped due to high cooling water temperature.
RO7605	113277	032776	040976	C	CE	OO,Y	-	C	AV,AR	C	N	Surface cracks on steam drum relief valve nozzles (116898).
RO7606	113550	041776	050376	C	SHA	OO	-	C	AU,BC	D	N	Containment vent supply valves leak.
RO7607	113982	042876	051276	C	SD	OO	-	C	BB	D	N	Resin sluice line isolation valve failed to close.
RO7609	115066	051676	060976	D	EE	N,F	-	B	EA	B	N	DG breaker interlock did not function automatically due to wrong fuse.
RO7608	115042	051676	060976	C	EE	N	-	B	BE	D	C1	Emergency DG tripped while supplying load due to high cooling water temperature.
RO7610	115453	052876	062576	D	RB	-	T	C	EH	D	N	Set point drift on CRD accumulator pressure switch.
RO7611	-	060576	070276	D	EE	N	-	-	OK	A,B	N	DG control circuit completed without review, wrong fuse size used.
RO7612	115737	061976	070276	D	SHA	OO	-	C	AU,AQ	D	N	Containment vent supply valves leak during test.

Table A2.11 (continued)

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
R07613	115880	062076	071976	D	IB	OO	I	C	CK	G	N	Reactor water level instruments in error due to equalizing valve left open.
R07614	115881	062176	071976	D	SFA	OO	-	C	AW	D	N	Reactor depressurizing system valves leak.
R07615	116535	063076	073076	D	EC	C	-	B	EC	D	N	Battery charger fails and battery voltage reduced.
R07616	116880	070476	080476	D	RB	J	-	C	AG	G	N	CRD fails to withdraw.
R07617	116881	071876	080476	D	SFA	C	-	C	BU	G	C7	Specific gravity of station battery acid low due to addition of water.
R07618	116786	072276	081976	D	EE	N	-	C	BI	D	C7	Starting time of DG exceeds limit.
R07619	116787	072276	081976	D	SFA	C	-	C	BU	G	C7	Water added to RDS battery and lowers its specific gravity.
R07620	116788	072976	081976	B	SFA	C	-	C	BU	G	C7	Water added to RDS battery and lowers its specific gravity.
R07621	117676	080576	090376	B	EE	N	-	C	BI	D	C7	DG failed to start within time limit.
R07623	117677	080576	090376	E	EE	N	-	B	CK,OC	H	C8	DG returned to operable status without retesting.
R07622	-	081276	090776	D	EE	N	-	C	BD	D	N	DG failed to start due to battery cable faults.
R07624	-	081376	090776	D	IB	-	S	B	EG	G	N	Power range neutron monitor had wrong polarization.

Table A2.11 (continued)

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
RO7625	119154	090276	100176	E	EE	N	-	C	BI	D	C7	DG failed to start within time limit.
RO7626	119162	090776	100676	B	SFA	DD	-	C	CK,OC	A	C8	Fire pump actuation test associated with RDS not performed.
RO7627	119465	092776	102676	B	SA	Z	-	B	CK,OC	A	N	Expansion joints at containment penetrations not inspected as per tech specs.
RO7628	119464	100476	102676	B	SFA	C	-	C	EU	G	C7	Water added to RDS battery and lowers its specific gravity.
RO7629	119516	101476	102876	B	SF	Z	-	B	CK	B	N	Errors found in allowable leak rate limit calculations.
RO7630	119749	102176	112276	B	SFA	C	-	C	BU	G	C7	Water added to RDS battery and lowers its specific gravity.
RO7634	119750	102776	112376	B	SFA	-	C	C	OA	D	N	RDS system channel removed from service for maintenance.
RO7631	119751	102876	112376	B	EE	N	-	C	BI	D	C7	DG failed to start within time limit.
RO7632	120270	110476	120176	B	EE	N	-	C	BI	D	C7	DG failed to start within time limit.
RO7633	120271	110476	120176	B	SFA	C	-	C	BU	D	C7	RDS battery has low specific gravity.
RO7636	120680	111876	121776	E	EE	N	-	C	BI	D	C7	DG fails to start within time limit.
RO7637	120676	113076	123076	B	RB	-	C	C	OC	A	N	Liquid poison circuit test not performed.

Table A2.11 (continued)

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
RO7638	120677	120276	123076	B	EE	N	-	C	BI	D	C7	DG fails to start within time limit.
RO7639	120679	120276	123076	B	SFA	C	-	C	BU	D	C7	RDS battery has low specific gravity.
RO7640	120678	120376	123076	B	HH	-	N	C	CK	[illegible]	N	Condensate radiation monitor flow inadequate due to surveillance procedures.
RO7643	121053	120976	010777	B	SFA	C	-	C	EU	D	N	Low specific gravity in ADS battery "B."
RO7642	120675	120976	122276	B	SFA	-	C	C	CK	A	C8	Insufficient knowledge of RDS actuation system violated tech specs.
RO7641	120674	120976	122276	B	SFA	-	-	-	CK	A	C8	RDS test procedures inadequate to cover tech specs.
RO7644	121052	122076	012077	B	EE	N	-	C	EI	B	N	DG fails timing test by 4 sec.
RO7645	121523	122776	012677	B	EE	N	-	C	EI	B	N	DG fails timing test. Modifications made to fuel governor lub oil supply.
RO7646	121524	122876	012677	B	EE	N	-	C	ED	D	C1	DG fails to start; the starter failed.

Table A2.12 Coding Sheet for Reportable Events at Big Rock Point - 1977

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
RO7701	121525	010377	012677	B	EE	N	-	C	BD	B	C1	DG fails to auto-start.
RO7702	122184	010477	020777	B	SFA	C	-	C	BU	D	C7	Low specific gravity in RDS batteries.
RO7705	122186	010777	020877	B	EE	N	-	A	CB	-	N	DG out of service 8 hrs to modify fuel oil lub governor.
RO7703	122201	011877	021677	B	SD	FF	-	B	AA	D	N	Damper lock broken on stock fan. Air supply to damper worn through.
RO7706	122202	011977	021677	B	SFA	-	C	C	CB	-	N	RDS out twice for 24 hour period for maintenance.
RO7704	122187	012777	020877	B	SFA	-	T	B	CF	B	N	RDS switches not environmentally qualified.
RO7707	123020	021777	031777	B	SFA	C	-	C	BU	D	C7	Low specific gravity in RDS battery cells. Tech specs change submitted.
RO7708	123798	022477	032377	B	SFA	C	-	C	BU	D	C7	Low specific gravity in RDS battery cell.
RO7709	124103	031777	041477	B	SFA	C	-	C	BU	D	C7	Low specific gravity in RDS battery cell.
RO7710	124901	032477	042177	B	EE	N	-	C	BI	D	N	Diesel fails start test by 0.8 sec.
RO7711	125210	033077	042877	B	SFA	G	-	B	AL	E	N	Loose connectors on uninterupted power supply.
RO7714	125032	042177	051777	B	-	-	-	C	CC	A	C8	Several tests missed due to poor 10 yr plan.

Table A2.12 (continued)

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
RO7713	125340	042177	051677	B	BB	-	N	B	BT	B	N	Operation of both air ejector radiation monitors degraded.
RO7712	125339	042177	051677	B	SFA	C	-	C	BU	D	C7	Specific gravity of RDS low.
RO7717	125342	050377	051677	B	SFA	-	C	B	EG	D	N	One RDS channel power supply fails.
RO7715	125341	050377	051677	B	SFA	-	C	C	OC	A	N	RDS channels not tested after one failed.
RO7716	125180	050577	060377	B	CF	U,Z	-	B	CJ	A,E	N	Defective hose installed in post incident systems heat exchanger.
RO7718	125549	051877	061777	B	EE	N	-	C	EI	D	N	DG fails starting test by 1.3 sec.
RO7719	125550	052677	061777	B	EE	N	-	C	EI	D	N	DG fails starting test by 1.9 sec.
RO7720	125312	052777	060877	B	RC	R	-	B	CK	B	N	MAPLHGR limits nonconservative for single loop operators.
RO7721	126492	061677	071577	B	SFA	C	-	C	BU	D	C7	Specific gravity low on RDS battery cells.
RO7722	128318	072077	081977	B	EA	LL	-	C	AC	D	N	Bushing inselator on power transformer fails.
RO7723	128317	072877	081977	C	SFA	-	I	C	EH	D	N	Set point drift in steam drum level sensor.
RO7724	127981	080277	081677	C	CE	Z	-	C	AO	E	N	Poor weld in emergency condenser pipe.
RO7726	128945	080477	090277	C	IA	-	M	A	EH	D	N	Set point drift in RDS primary system pressure sensor (134062).

Table A2.12 (continued)

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
R07727	128946	080577	090277	C	EE	F	-	C	EB	E	C8	Transfer of DG power to "2B" bus fails (134063).
R07733	128947	080677	090277	C	CF	U,Z	-	B	CJ	A	N	Defective hose installed in post accident heat exchanger (132710).
R07734	128948	081077	090277	C	CG	Z	-	C	AC	E	N	Two Bid welds in demineralizer piping fail.
R07735	129548	081277	090977	C	SB	-	E	A	EG	D	N	Containment spray flow transmitters fail.
R07728	128222	081277	082577	C	SD	OO	-	C	AX	D	N	Cleanup sluice system valve leaks excessively.
R07729	128221	081277	082577	C	SD	PP	-	C	AX	D	N	Rod drive check valves leak.
R07730	128220	081377	082577	C	SD	Z,PP	-	C	CK	A	N	Design deficiency in CRD system could compromise containment.
R07731	128223	081477	082577	C	CG	PP	-	C	AQ	D	N	Demineralized water line check valve leaks.
R07738	130024	081677	092977	C	CH	PP	-	C	AQ,AT	O	N	Crud buildup results in feedwater check valve leakage.
R07736	129829	082377	092377	C	RB	J	-	C	EW	D	N	3 CRDs withdraw too quickly.
R07737	130025	082977	092977	C	SFD,AB	N,G	-	C	BD	D	C1	Diesel fire pump fails to start due to loose cables.
R07725	129827	090477	091677	C	SD	OO	-	B	EB	D	N	Containment isolation valve fails to close (130907).

Table A2.12 (continued)

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
R07732	129828	090977	092377	C	SFD	-	-	-	EI	B	N	Study indicates some uncertainty in core spray distribution.
R07739	130913	092977	102777	E	SFA	C	-	C	EU	D	C7	Specific gravity low on RDS battery cell.
R07740	130908	100477	110177	D	SFD	-	M,T	C	EH	D	N	Six of eight core spray pressure switches set points drift.
R07741	130997	102077	111877	E	EE	N	-	C	EI	D	C7	Diesel generator fails starting test.
R07742	130998	102077	111877	D	SFA	C	-	C	EU	D	C7	Specific gravity low on RDS battery cell.
R07743	131705	103077	112977	D	RB	-	T	B	CJ	H	N	CRD removed with reactor mode switch in run.
R07744	130883	103177	110977	C	CX	OO	X	B	AY	H	C3	Reactor coolant backs up into plant heating system.
R07745	131706	103177	112977	D	RB	OO	-	C	BI	D	N	CR withdrawal speed excessive.
R07746	131791	111177	120977	E	RB	J	-	C	ED	D	N	CRD malfunctions.
R07747	132949	111777	120977	E	SFD	LL	-	-	CK	A	C7	Defective procedure capable of reducing ECCS capability.
R07751	133612	121577	011378	E	-	OO,F	-	C	CC	A	N	Surveillance schedule skipped.
R07749	133610	121677	011378	E	RB	Z	-	B	AP	D	N	One drop/sec leak in weld between valve and pipe weld.
R07750	133611	122277	011378	E	SFA	C	-	C	EU	D	C7	Specific gravity low on RDS battery cell.

Table A2.13 Coding Sheet for Reportable Events at Big Rock Point - 1978

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
R07806	136471	010878	030778	B	SFA	-	C	B	EG	D	N	One of four ADS channels defective.
R07801	134504	010978	020878	B	SFC,AB	N,G	-	C	EI	D	C1	Diesel fire pump fails to start within 20 sec.
R07809	136472	011478	030778	B	SFA	-	C	B	EG	D	N	One of four ADS channels made inoperable for troubleshooting.
R07812	136474	011778	030778	B	MA	HH	-	B	AM	B	N	Solenoid in rad waste system not qualified.
R07802	134981	011978	021778	D	EC	C	-	C	EU	D	C7	Low specific gravity in RDS battery.
R07803	134273	012078	020178	D	SD	OO	-	C	EC	D	N	Containment isolation valve leaks excessively.
R07805	136476	020378	030178	B	MB	-	E	B	BB	G	N	Low flow in off-gas system.
R07807	136470	020978	030778	B	EE	N	-	C	BF	D,B	N	Diesel generator trips after 25 min.
R07808	135891	021578	021578	B	AB,SFA	-	T	A	CJ	H	S8	Both fire pumps unavailable with ADS out for maintenance.
R07810	136477	021778	030178	B	CC	-	I	B	AM	B	N	Nonqualified flow switches.
R07811	136473	021778	030778	B	RB	HH	-	B	AM	E	N	One scram pilot valve unfit for conditions.
R07804	136475	022078	030178	B	EE	N	-	C	BI	D	C7	Diesel generator exceeds starting time by 15 seconds.
R07813	137025	022378	032378	B	SD	-	I	B	AM	E	N	Marginal electrical circuitry for Big Rock.

Table A2.13 (continued)

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
RO7815	136980	030978	032878	B	SPA	-	C	E	EG	D	N	One of four ADS channels out of service for troubleshooting.
RO7814	136981	031178	032878	B	SPA	-	C	E	EG	D	N	One of four RDS channels fails.
RO7817	137503	032078	041978	B	RB	-	T	A	EC	G	N	Control switch on CPD pilot valve incorrectly set.
RO7816	137502	032078	041978	B	SD	FF	-	B	AW	D	N	Containment leak rate exceeds limits.
RO7818	138236	040778	050278	B	IA	-	I	B	EG	D	S2	Failure of two RPS channels during LOOP.
RO7819	138237	040778	050278	D	IA	-	M	B	EH	D	N	One RPS vacuum sensor drifts slightly.
RO7820	138238	041178	050278	B	BB	OO	-	B	AQ	D	N	Crud causes off-gas flow to be low.
RO7821	138295	041578	051578	B	EA	LL	-	B	CE	-	N	Modification to substation.
RO7822	138296	041778	051578	B	EA	LL	-	B	-	-	N	Loop with runback.
RO7823	138821	050478	060178	B	AE,SFA	C	-	C	EU	D	C7	Low specific gravity in diesel fire pump starting tatteries.
RO7824	139646	051278	061278	B	RB	-	M	C	EH	D	N	Drift in CRD accumulator level switch.
RO7825	139645	052578	061678	B	AE,SFA	C	-	C	BU	D	C	Low specific gravity in diesel fire pump starting t tteries.
RO7826	141046	053178	063078	D	HH	JJ	-	B	ET	H	C6	Condensate storage tank level drops below tech specs limit after scram.

Table A2.13 (continued)

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
RO7827	141048	053178	063078	C	RE	J	-	E	AG	H	N	Failure of a CR to withdraw beyond position 20.
LER7828	141049	060578	063078	B	CE	PP	-	B	AG	G	C1	Emergency condenser outlet valve inoperable.
LER7829	141050	060678	063078	E	CI	FF	-	B	AC	D	N	Excessive leakage of primary coolant.
PO7830	139900	071278	080778	B	SFA	-	C	B	EG	D	N	An RDS channel failed.
RO7831	140383	081078	082278	E	CG	Z	-	B	AO	B	N	Minor defect in reactor cleanup system piping.
RO7832	140350	081978	091578	E	HH	PP	-	E	AW	D	C3	Failed check valve causes demineralized water to leak into containment.
RO7833	140701	082978	092778	E	FI	OO	-	C	AO	D	N	Reactor and fuel pit drain line valve leaks.
RO7834	140704	083178	092778	B	SFA	C	-	C	BU	D	C7	Low specific gravity in RDS batteries.
RO7835	140211	090478	091278	E	SD	OO	-	C	AW	D	N	Containment valve leaks.
RO7836	140735	090578	100578	D	HH	JJ	-	B	BT	H	N	Condensate storage tank level drops below tech specs limit.
RO7837	140213	090678	091278	C	CD	OO	-	C	AC	D	N	An MSIV failed to close.
LER7838	141519	090978	100978	B	RE	J	-	B	EL	B	N	Reactor scrammed due to high CRD temperature.
LER7840	141524	091078	100978	D	EC	AA	-	B	AC	D	N	Containment relief valve invertor fuse blows.
LER7839	141522	091178	100978	D	IE	-	T	B	CJ	H	C8	CR removed with reactor not in shutdown mode.
LER7841	141192	092278	100678	C	FD	L	-	B	AB	D	N	Refueling cask trip line fails.

Table A2.13 (continued)

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
RO7842	141844	092878	102678	D	RB	J	-	C	CK	A	N	CRD coupling test may be deficient.
LER7844	141453	101878	103178	E	CG	Z	-	B	AI	D	C8	Crack in nonisolatable 3" pipe (148063).
RO7845	141485	102078	110278	D	IB	G	-	B	AM	B	N	Nonlock qualified cables and connectors.
LER7840	141703	102578	111078	D	RB	J	-	C	BI	D	N	One CR exceeds scram limit.
LER7847	142288	103178	112978	B	RB	J	-	C	BL	D	N	One CRD temperature exceeds limit. See 153975.
RO7848	142209	110878	120678	B	CE	U	-	B	EL	A	N	Emergency condenser shell side temperature exceeds limit (153836).

Table A2.14 Coding Sheet for Reportable Events at Big Rock Point - 1979

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
LER7901	147305	020179	022679	B	SD	OO	-	B	BC	D	N	Containment ventilation valves out of adjustment.
LER7902	147304	020379	030279	C	HC	OO	-	B	AA	D	N	Condenser hotwell valve fails.
LER7903	147303	020479	030279	C	SD	PP	-	C	AW	D	N	Containment isolation valves leak excessively.
LER7904	147302	021979	030579	C	CE	Z	-	C	AV	E	N	Bad welds in emergency condenser inlet line.
LER7906	147300	022179	030579	C	CG	Z	-	C	AV	B	N	Cracks in reactor cleanup system piping (153974).
LER7905	147301	022179	030579	C	RE	-	T	C	EE	D	C8	Seven of 32 CRD accumulator level switches fail.
LER7910	148200	022179	032179	C	SHE	Z	-	C	AM	A	N	Backup hose for post incident heat exchanger to short.
LER7907	147299	022279	030579	C	CE	Z	-	C	AO	Z	N	Weld does not meet present requirements.
LER7908	148201	022279	032179	C	EE	N	-	C	BI	D	C7	Diesel generator exceeds starting time.
LER7909	148199	022379	032179	C	CH	PP	-	C	AU	D	N	Feedwater check valve leaks.
LER7911	148337	030179	032179	C	RB	PP	-	C	AC	D	N	CRD pump check valve leaks.
LER7913	148339	030279	032779	C	CG	OO	-	C	AV	D	N	Reactor cleanup system sluice valves leak excessively.
LER7912	148338	030279	032779	C	SD	PP	-	C	AW	D	N	Containment isolation check valve leaks.

Table A2.14 (continued)

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
LER7914	149434	031279	041179	C	EE	N	-	C	EC	D	C7	DG output voltage zero.
LER7916	149432	031379	041979	C	SFD	F	-	A	BC	G	N	Fire pump broken, damaged during maintenance.
LER7915	149433	031979	041979	C	SD	S	-	B	EE	B	N	Fuse blown on an inverter (also see LER79021).
LER7917	149977	041079	051079	C	SH	GG	-	C	AM	B	N	Inadequate snubbers.
LER7918	149976	041779	050279	C	CA	Q	-	C	AO	E	N	Leak between CRD housing and reactor vessel (153935).
LER7919	149731	041879	050279	E	RB	FF	-	C	AW	D	N	CRD flanges leak.
LER7920	150275	060979	062279	C	CE	Y	-	B	AD	D	C8	Recirculation diffusers break off.
LER7921	151824	061679	071379	C	SD	S	-	B	EE	B	N	Fuse blows on an inverter (also see LERs 7840 and 7915).
LER7922	151825	082279	090579	C	IA	-	I	B	EG	B	C4	Common mode problem with RPS and ECCS (153973).
LER7923	152017	091179	101079	C	SD	OO	-	C	AW	D	N	reactor and fuel pit drain line valve leaks.
LER7924	152016	091379	101079	C	IB	-	T	C	EH	D	N	Minor drift in RDS switches.
LER7925	154266	103079	110979	C	SD	OO	-	B		D	N	Power supply to containment vent valves fails.
LER7926	154265	110179	120379	D	SPA	C	-	C	BU	D	C7	Low specific gravity in RDS batteries.
LER7927	154264	111579	121179	E	CE	OO	-	B	AU	D	N	Emergency condenser outlet valve leaks (154558).
LER7928	154263	121379	122679	E	SD	OO	-	B	-	B	N	Containment isolation valve might not close given worst LOCA.

Table A2.14 (continued)

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
LER7929	153498	121579	011580	E	SC	DD	-	A	OI	A	N	Both plant vent fans unavailable.

Table A2.15 Coding Sheet for Reportable Events at Big Rock Point - 1980

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
LER8001	154457	011480	021280	B	CE	OO	-	B	AW	C	N	Emergency condenser outlet valves leak through seal.
LER8003	154906	012380	022180	B	SFA	-	C	B	EG	D	N	One channel of RDS removed from service.
LER8002	154904	012480	022180	B	SFA	C	-	C	BU	D	C7	Low specific gravity in RDS batteries.
LER8004	154459	020180	021280	B	CF	Z	-	B	AD	G	N	Control air tubing to auxiliary system cooling valve broken.
LER8007	155473	022480	031980	B	RE	DD	-	B	AP	D	N	A CRD pump failed.
LER8005	155482	030180	040280	B	IE	-	I	B	EH	D	N	RDS level sensor drifts.
LER8006	157053	040380	050180	B	IE	-	C	B	EH	D	N	ADS channel set point drift.
LER8009	156984	041580	051980	B	SD	FF	-	C	CC	A	N	Leak test not performed.
LER8010	157074	042380	052380	B	IE	-	I	B	EH	D	N	ADS level channel drift.
LER8011	158574	050180	053080	B	SFA	C	-	C	BU	D	C7	Low specific gravity in RDS batteries.
LER8012	157076	050780	052380	B	IE	-	I	B	EH	D	N	Reactor level indicator drifts.
LER8013	156954	050980	052380	B	SD	OO	-	B	-	A	N	NRC dictated failure.
LER8014	158226	051280	061180	B	IB	-	I	B	EH	D	N	ADS channel drifts.
LER8015	158267	051580	061180	B	IB	-	I	B	EH	D	N	ADS channel drifts.
LER8016	158268	052780	061180	B	IE		I	B	EH	D	N	ADS channel drifts.
LER8017	158777	070380	080180	B	IE	-	I	B	EH	D	N	ADS test channel drifts.
LER8018	159088	071080	080480	B	SFA	C	-	C	BU	D	C7	Low specific gravity in RDS batteries.

Table A2.15 (continued)

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
LER8042	161911	120680	121880	E	BE	Z	-	C	AO	E	N	Crack in weld.
LER8043	161913	121280	122380	D	CG	Z	-	C	AR	D	N	Crack in reactor cleanup system pipe.

Table A2.15 (continued)

NUMBER	NSIC ACCESSION NUMBER	EVENT DATE	REPORT DATE	PLANT STATUS	SYSTEM	EQUIPMENT	INSTRUMENT	COMPONENT STATUS	ABNORMAL CONDITION	CAUSE	SIGNIFICANCE CATEGORY	COMMENT
LER8019	160291	071380	081280	B	SD	-	M	B	EE	D	N	Containment pressure sensor fails.
LER8021	159288	072980	082580	B	IB	-	I	B	EH	D	N	Set point drift in level transmitter.
LER8022	159256	080780	090580	B	SFA	C	-	C	BU	D	C7	Low specific gravity in RDS batteries.
LER8023	159293	081280	082680	B	SD	-	-	-	-	-	N	Procedures revised for small LOCA.
LER8024	159257	081480	091280	B	IB	-	I	B	EH	D	N	Set point drift in level transmitter.
LER8028	160071	090980	101080	B	IB	-	I	B	EB	D	N	Sensor channel D of RDS fails.
LER8030	160043	091980	101780	B	SD	OO	-	C	AB	D	N	Containment leak rate excessive.
LER8029	160072	091980	101080	B	SFA	C	-	B	BU	D	N	ADS battery fails to hold a charge.
LER8031	160171	092380	102080	B	IE	-	I	A	EH	G	N	ADS level switch set below limit.
LER8035	161469	102480	111980	B	CF	JJ	-	B	AO	A,D	N	Anion resin tank leaks.
LER8034	161674	110180	111980	B	CD	OO	-	B	AC	D	N	An MSIV failed to close on first attempt.
LER8038	161983	111880	121280	D	CH	PP	-	C	AU	D	N	Feedwater check valve leaks.
LER8036	161980	111830	121280	D	EE	N	-	C	ED	G	C1	DG fails tc reach rated voltage.
LER8037	161982	111880	121280	D	EE	N	-	C	EE	D	C1	DG fails to run.
LER8039	162017	112780	122280	D	SD	OO	-	C	AW	D	N	Containment isolation valve leaks.