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The In-Plant Reliability Data Base for Nuclear Power Plant Components: Data Collection and Methodology Report

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Prepared for the
U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Division of Risk Analysis
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FOREWORD

Since 1978 the American National Standards Institute/Failure Incidents Reports Review (ANSI/FIRR) committee has sponsored a voluntary program of visits to nuclear power stations for record collection at the plant site and the conversion of these records into a comprehensive data base to be applied to risk and reliability analyses. Additional funding is provided by the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research. Oak Ridge National Laboratory (ORNL) is the prime contractor and is primarily responsible for coordination of data coding and analysis. Science Applications, Inc., as a subcontractor to ORNL, provides technical assistance in the development of the In-Plant Reliability Data (IPRD) system and the IPRD generic systems descriptions.

As secretariat to the ANSI/FIRR committee, the Institue of Electrical and Electronics Engineers (IEEE) Office of Standards provides assistance in data handling and storage. The IEEE Subcommittee on Reliability (SC-5) is responsible for selection and coordination of the data collection team and plant visits as well as providing technical assistance.

ACKNOWLEDGMENT

We appreciate the guidance and advice from the NRC Project Manager, Dr. James W. Johnson of the Office of Nuclear Regulatory Research. We also appreciate the fine work contributed by the members of the Computer Science Division of ORNL, especially Betsy Horwedel. We would like to thank Joseph R. Fragola of Science Applications, Inc. for his valuable technical assistance. Special thanks go to SAI-Oak Ridge Office for its substantial contributions to this document.

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ABSTRACT

The development of a component reliability data base for use in nuclear power plant probabilistic risk assessments and reliability studies is presented in this report. The sources of the data are the in-plant maintenance work request records from a sample of nuclear power plants. This data base is called the In-Plant Reliability Data (IPRD) system. Features of the IPRD system are compared with other data sources such as the Licensee Event Report system, the Nuclear Plant Reliability Data system, and IEEE Standard 500. Generic descriptions of nuclear power plant systems formulated for IPRD are given.

Keywords: in-plant data, IPRD, reliability, data base, nuclear power plant, probabilistic risk assessment, generic system description, component.

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1. INTRODUCTION

1.1 Background

Since the publication of the Reactor Safety Study WASH-1400 (Ref. 1) in 1975, there has been a growing interest in and need for the use of probabilistic techniques to assess the safety and reliability of nuclear power generating stations. Both the interest and the need have intensified in the aftermath of the Three Mile Island accident, contributing to the subsequent initiation of the Interim Reliability Evaluation Program (IREP), a major risk assessment of four nuclear power plants. It is anticipated that the IREP program will be the forerunner of the National Reliability Evaluation Program (NREP) effort involving the application of probabilistic techniques to assess the reliability of all U.S. nuclear power stations in operation, under construction, and in the design stage.

One of the major difficulties in performing a defensible risk assessment has been the lack of adequate equipment failure data. The lack of hard data in developing the component failure rate estimates has been a major contributor to the large uncertainties and was highlighted by the well-publicized "Lewis Committee Report."[2] Other studies, including WASH-1400, have clearly identified the large uncertainties inherent in using current data banks. Furthermore, some studies have raised the question of the applicability of generic failure and repair rates to specific facilities. In some cases, it is suspected that plant-specific environmental and design conditions and operational practices may justify or require modifying generic data to ensure applicability.

1.2 Objectives

The major objective of the program described in this report has been to provide an improved data base for use in probabilistic risk assessments. The method for accomplishing this objective has been to create a detailed data base for selected components at various operating nuclear power generating stations. All components of a particular type are included. The source of this detailed data is the in-plant maintenance files. Thus, the resulting data base has been named the "In-Plant Reliability Data" (IPRD) system.

In addition to providing failure rates and component down times for use in probabilistic risk assessments, potential uses of the IPRD include:

- revising of component test intervals and allowable down times;
- identifying generic problems and recurring failures;
- identifying the variables (e.g., environment, operating mode, system, maintenance policy, etc.) that control component failure rates;
- providing, for a sample of components and plants, an extensive data base against which to compare existing data sources (e.g., LERs and NPRDS) to assess the degree to which these data sources accurately reflect the actual component reliability;
- correlating current incidents with previous failures;
- identifying trends and patterns in the failure characteristics of particular components or aggregations of components; and
- identifying failure mechanisms over time for use in defining the aging requirements for component qualification.

This report is the first in a series of publications on the IPRD system. It contains a detailed description of the data base, the procedures used to collect and process the data, and working definitions of the terms used in the encoding schemes. Subsequent publications will focus on the display and analysis of failure and repair data for particular components. It is anticipated that each component or related group of components will constitute a separate publication. Currently, publications on pumps and valves are planned.

1.3 Existing Data Sources

Before describing the features of the IPRD, it will be useful to examine briefly three other major existing sources of nuclear reliability data. These data sources are: (1) The Nuclear Plant Reliability Data System (NPRDS) (Ref. 3); (2) The EG&G Idaho Studies,[4-9] which calculated component failure rates using Licensee Event Report (LER) data; and (3) Institute of Electrical and Electronics Engineers (IEEE) Standard 500 (Ref. 10).

1.3.1 The NPRDS

The NPRDS (Ref. 3) provides utilities, The Nuclear Regulatory Commission (NRC), Architect-Engineers (AEs), and suppliers with reliability data for safety-related components and systems of nuclear power plants. At the time the IPRD program was started, the NPRDS was maintained by Southwest Research Institute (SRI) under contract to the Edison Electric Institute and under the direction of American Nuclear Society (ANS) Standards Committee ANS 58.20. Since January 1982 the NPRDS has been under the direction of the Institute of Nuclear Power Operations (INPO). The following information on NPRDS describes the data base prior to 1982. Information on file in the NPRD system is derived from standard format input reports prepared by staffs of nuclear utilities. Participation in the system is voluntary. Some quality assurance was performed on the data by SRI to eliminate obvious errors, and requests for corrections are sent to utilities when necessary.

The NPRD system includes design data and other descriptive information on each unit, such as unit owner; rating; type; Nuclear Steam Supply system supplier; AE; and critical and commercial service dates, submitted by the utility on a standard report form. It also includes engineering data and descriptive information for safety-related systems and components for each unit, such as unit name, owner, component or system code designation, safety class, manufacturer model and serial number, operating environment, drawing number, and operation and testing data, submitted on a standard report form and updated as required. A quarterly operating report, submitted by the utility on a standard report form, includes such information as unit name, owner, on-line time in hours, reactor critical hours, standby and shutdown hours, and number of failure reports for the quarter. A report of failure form, submitted quarterly by the utility on a standard report form, includes such information as unit name, owner, failed component or system code, component identification number, date of failure, failure number, failure start and end times, failure description, cause and corrective action, and failure classification.

The limitations of the NPRDS discussed below are being rectified by INPO. However, prior to 1982 some of the major sources of inconsistency in NPRDS were:

1. definition of systems;
2. definition of reportable scope (i.e., the size and makeup of the component population for which engineering and failure data were to be reported);
3. designation of boundaries between components and ancillary equipment;
4. interpretation of reporting instructions and data element codes;
5. definition of a reportable component failure;
6. variations in the skill and training of individuals responsible for reporting data to NPRDS; and
7. variations in the amount of effort spent in collecting and correcting the data.

1.3.2 The LER system

For each licensed nuclear power plant, certain events or "reportable occurrences" must be reported to the NRC in accordance with Regulatory Guide 1.16 (Ref. 11). In order to collect, collate, store, retrieve, and evaluate information concerning these events, the then Atomic Energy Commission (AEC) established in 1973 a computer-based data file of information extracted from licensee reports. This data bank is known as the LER file.

The reportable events, known as LERs, are for the most part deviations from a plant's tech specs and as such they should include most failures of safety-related components. Though the LER system was not designed to provide reliability information, enough information is available from LERs for component failures to make the LER data base attractive as a source of reliability data. Therefore, beginning in FY 1978, the NRC funded a project, the goal of which was to produce gross component failure rate estimates from the LERs. The project was carried out at the Idaho National Engineering Laboratory (INEL) by its contractor EG&G. To date, reports have been produced on pumps,[4] control rods and drive mechanisms,[5] diesel generators,[6] valves,[7] primary containment penetrations,[8] and instrumentation.[9]

The major limitations of the LER system derive from the fact that it was not designed to be used as a source of reliability data. It also suffers from the data inconsistency problem similar to that described for NPRDS. The list below summarizes the weaknesses of the LER system as a reliability data base:

1. The LER data base does not contain plant population data. The standard estimates of failure rate per unit time require knowledge of the number of similar items and their exposure times. If failure rates per demand are required, even more detailed information on the components is necessary. This information is missing in the LER data base. The EG&G Idaho study[4–9] utilized reactor plant Final Safety Analysis Reports (FSARs) to obtain counts of pumps. The LER data could then be utilized to obtain gross failure rate estimates of safety-related pumps and valves for selected plants.
2. LERs are not submitted for every plant component. Failures of nonsafety components may or may not be reported.
3. LERs may not be submitted for every type of failure even if the component affected was a safety class component.
4. The reporting of data by different individuals in different organizations (see the discussion in Sect. 1.3.1, NPRDS) results in considerable inconsistencies in the LER data base just as it does for NPRDS.

1.3.3 IEEE Standard 500

Published in 1977, IEEE Standard 500 contains failure data for generic electrical and electronic equipment. The failure data include failure rate ranges, failure modes, and environmental factors. Although a substantial portion of the data contained was based either directly or indirectly on in-plant experience, much of the data was obtained by a consensus approach, a "Delphi study," using ~200 experts throughout the

United States. The data from this Delphi study was therefore supplemented with data obtained from nuclear facilities, fossil-fueled generating plants, transmission grids, and other industries.

This effort represents a positive contribution in that failure rate information is tabulated in readily usable form for selected components. The primary limitations of IEEE data sources are: (1) the limited component types addressed and (2) the difficulty in extracting the estimates based on actual operating experience from those based on the Delphi evaluation.

1.4 History and Features of IPRD System

In this section, a brief overview of the IPRD system will be presented. A more detailed discussion of the data base can be found in Chap. 2, and a description of the data collection and encoding methodology is given in Chap. 3.

In 1972, the IEEE published the first edition of Standard 352 (Ref. 12), which contains a basic methodology necessary to conduct a reliability analysis. At the time of publication, it was recognized that this methodology was incomplete without a supporting data base. Work began to produce a data base to support IEEE's Standard 352 and this culminated in 1977 with the publication of IEEE Standard 500. This data source was limited to electrical, electronic, and sensing devices. This limitation was due in part to the fact that the organizational scope of IEEE was limited to these devices. However, it was recognized by the nuclear community that the components within the IEEE scope were not the only sources of concern from a plant risk and availability standpoint. In addition to the electrical components, other significant components were mechanical in nature and were therefore within the scope of the American Society of Mechanical Engineers (ASME). Because the scope of the components of interest was within ASME and the data base development and reliability skills resided within the IEEE, it was decided that the future data collection effort be sponsored cooperatively under the umbrella of the American National Standard Institute (ANSI) federation.

It was this desire to expand the work on data to mechanical devices which led to the creation of a data subcommittee of the ANSI Failure and Incidents Reports Review (FIRR) Committee. This subcommittee was charged specifically to arrange for visits to operating nuclear plants and to collect in-plant information. It was also the responsibility of this committee to store this collected information until it could be used. To ensure that proper experience and expertise was applied to the effort, and that these stored data records could be readily accessed, the IEEE was granted the secretariat of the FIRR Committee. The IEEE Reliability Subcommittee (SC-5) was granted overall technical cognizance of the data collection, storage, and encoding effort.

In 1978, this subcommittee began collecting failure and repair data from nuclear plant maintenance and repair files. This ongoing data collection effort has, for the most part, relied on industry volunteers for its manpower requirements. Some financial support was obtained from the NRC initially to cover administrative costs of the data collection effort and later to establish a consistent data base on nuclear component reliability behavior for use in reliability and probabilistic risk assessments (PRA). In this manner the raw in-plant maintenance records from which the IPRD system has been developed were collected.

The data base established in this study is unique in that it is a comprehensive collection of component population, failure, and repair data for a sample population of operating nuclear generating units. The sample includes data from both pressurized-water reactors (PWRs) and boiling-water reactors (BWRs), and will ultimately include the major nuclear steam suppliers, a variety of AEs, and various operating utilities. For the selected component types (including pumps and valves), the data extracted from the plant maintenance logs will contain the majority of the failure and repair information for nonsafety as well as safety-class components from the time of commercial operation to the time when the data were collected. Preventive maintenance, casual maintenance, and some contractor-performed maintenance actions are not included. The sections that follow discuss briefly the features that make the IPRD a unique source of nuclear reliability data.

1.4.1 Complete component failure histories

Unlike either the NPRD or LER systems, IPRD includes an essentially complete failure history (not just catastrophic failures) for all components included in the data base. Moreover, this history is traceable to the original in-plant source documents, should further investigation or checking be required. All failures are coded with one of three severity levels:

1. Catastrophic — total inability of the component to perform its intended function.
2. Degraded — component operates at less than its specified performance level.
3. Incipient — component functions as designed but the failure will probably progress to the degraded or catastrophic level if left unattended.

The number of failures in different severity levels should be due, at least in part, to the plant maintenance policies. For example, one would expect a priori that a plant using a "run-to-fail philosophy" would tend to have more catastrophic failures while an extensive preventive maintenance program would tend to discover more failures at the incipient level before they progress to the degraded or catastrophic levels. Thus, the effect of different plant maintenance policies can be investigated as well as the effectiveness of the various preventive maintenance programs. Since the IPRD system includes these noncatastrophic failures it has the potential for providing information useful in establishing the optimum level of preventive maintenance in a plant. However, not all plants list incipient failures, and therefore only limited comparisons between plants can be made.

The completeness of the IPRD failure history for a given component also makes it a potentially useful data source for studying incipient and degraded failures, for looking at time-dependent effects such as burn-in and wear-out, and for estimating the denominators required in computing the maintenance contribution to human error rates. The human error rate denominators, which are the number of opportunities for making an error, are derived from the number of maintenance actions. Since IPRD includes corrective maintenance records, it can provide more accurate estimates for the human error rate denominator than other data bases that include only the fraction of maintenance resulting from catastrophic failures.

1.4.2 Nonsafety and safety-class components

Unlike NPRDS or LERs, IPRD includes nonsafety as well as safety-class components. In addition to increasing the size of the component population, this feature eliminates the plant-to-plant inconsistencies inherent in selection of reportable components based on safety class. Furthermore, it will allow studies to be performed comparing the behavior of safety- with nonsafety-class components.

1.4.3 Population data

Unlike the LER system, IPRD includes detailed, plant-specific population data, derived from plant records. Population information is included for every component in the data base, whether or not a failure has occurred for that component. Detailed information on component characteristics was included when available. Operational parameters such as duty cycle, operating mode, and number of annual operations are not readily available from the plant's records. Engineering judgement was used to estimate these parameters where necessary. In addition, the component population classification scheme employed by IPRD is designed to facilitate computerized data aggregation at any level desired (e.g., by plant, unit, system, component size, environment, or operational mode). The desired aggregation is accomplished through use of a system code and hierarchical number assigned to each component in the population.

The hierarchical number gives information on the design of the component and the environment in which the component operates. Together, the system and hierarchy codes provide a mechanism whereby failure and repair data for similarly designed components operating in similar environments can be grouped together. This is important when calculating failure rates and mean repair times because, typically, any one component has too few failures to accurately estimate a failure rate or mean repair time. To reduce the statistical uncertainty surrounding the estimated mean values, it is usually desirable to pool data from components in different systems and/or plants. Flexible data aggregation capability also facilitates the investigation of characteristics that are controlling the components' reliability behavior.

1.4.4 Generic systems definitions

IPRD uses a detailed set of generic systems definitions to allow proper aggregation and comparison of components performing the same functions in different plants. The Tennessee Valley Authority's Unique Identification (UNID) of structures, systems, and components descriptions had not yet been developed when the IPRD program initiated the generic system description effort. Although a set of systems codes was developed for NPRDS, these codes are not generic, nor were they judged to be adequate for the purpose of this study. Presently, NPRD is improving the classification of components in the systems. The systems definitions used in IPRD were developed specifically for this study and are described in detail in Chap. 2 and Appendix A.

1.4.5 Repair data

The existence of repair time information is another unique feature of IPRD. These data are important for probabilistic risk assessment studies and for work on revision of allowed down times in plant technical specifications. Neither LERs nor NPRDS contain repair time data.

1.4.6 Consistent data selection and encoding

In contrast to NPRDS and LER systems, where the selected set of reportable data as well as the quality of the reported data varies considerably from plant to plant, the IPRD data are selected, encoded, and checked by a small group of engineers. This does not eliminate biases which are bound to exist when engineering judgment is used, but it is expected that inconsistencies in the IPRD system will be reduced considerably.

1.5 Data Collection, Analysis, and Reporting Schedule

The initial phase of this project was to encode the pump data from two nuclear generating units. Pumps were selected because the size of the population (250 to 300 for a typical plant) is large enough to give a representative number of different components on which to test and perfect the encoding methodology, yet small enough to be reasonable for a pilot study.

The second phase of the project increased the size of the data base in two ways. The number of units was increased from two to six, and valves were included as well as pumps.

The third phase of the project will be concerned with data analysis and will concentrate on the calculation, analysis, and comparison of statistics such as failure rates and mean repair times.

The fourth phase will be devoted to further enlargement of the data base to include other components and plant units.

2. IPRD DATA BASE

2.1 General

The IPRD system contains population, failure, and repair data for selected components. For each component the data base contains the entire failure and repair history from the date of the reactor's commercial operation to the data collection date. The data base contains three distinct record types:

1. Population record: contains information about the design, operating environment, operating mode, and functional name of the component. One population record is constructed for each component.
2. Failure record: contains the failure report number, date, failure mode, failure cause(s), failure severity level, and failure description. One failure record is entered into the data base for each failure.
3. Repair record: contains the repair time, crew size, repair category, and repair description. One repair record is encoded for each repair action.

Thus, each component has one population record and a pair of records (failure and repair) for each failure. The population records are matched to the failure and repair records by the component identification (ID) number. The population record is created whether or not the component has experienced any failures.

In general, the various fields are encoded using coding schemes that allow the user to select from a fixed list of possibilities. There are certain limitations inherent when using fixed coding schemes. First, this method limits the user's flexibility since the user must select from a fixed list of codes. Second, there is a need for engineering interpretation of the failure and repair descriptions.

The major advantages of this encoding scheme are (1) that the variability for a given field is limited to that provided in the list of possible options and (2) data can be quickly and accurately retrieved from the data base. For example, a listing of all bearing failures can be produced by listing all failure records with that particular failure cause code. Since the primary purpose of the data base is to produce failure rates and mean repair times for various groupings of components, the data base has been constructed to facilitate data aggregation and retrieval.

Where possible, existing coding schemes have been adopted as they are or with some modification for use in this study. The failure cause codes for pumps were taken from the EG&G Idaho analysis of LER pump failure data.[4] The component type codes were adopted from the six-character component type codes used by the NPRDS and the LER system. Vendor codes are identical to those used by NPRDS and the LER systems.

In those cases where no coding scheme existed or where existing schemes were deemed to be inadequate, new coding schemes were developed. For example, a new code was developed for generic systems because the list of systems used by NPRDS and the LER system was found to be too limited for this project. Hence, a more comprehensive list was developed.

A detailed description of the in-plant data collection and encoding process, including a discussion of quality assurance methods and estimation techniques for nonobjective data can be found in Chap. 3. The rest of this chapter is devoted to a description of the individual data elements in the IPRD data base.

2.2 Coding Conventions

Each field in each record of the data base is completed using one of the codes available for that field. If the information does not exist, the field is left blank or zero-filled. This is done, for example, in the hierarchical number on the population card when information for a particular qualifier is unavailable. The only "free format" fields in the data base are: (1) the plant-specific component ID field on the population record, (2) the failure report number on the failure and repair records, and (3) the description fields on each record.

2.3 Access to IPRD Records

The IPRD program is a pilot program and as such is developmental in nature. The program was not intended to utilize the industrial data records for such purposes as identifying preferred vendors. The pilot program was seen from the outset as a method to supplement the aggregate information contained in IEEE 500, the EG&G reports, and other sources. For this reason, the published data base is limited to aggregate information such as aggregate mean values and error bounds. This limitation does not significantly decrease the value of the results since the aggregate information is of interest in a data base for risk assessments. The computer data records are recognized as a research data source and are accessible for the purpose of noncommercial research projects through the auspices of the ANSI/FIRR committee, and/or through the IEEE Nuclear Power Engineering Committee/Reliability Subcommittee (SC-5).

2.4 Population Data

The population record contains information on the design, operating environment, and operating characteristics of each component. For example, a component could be a motor, valve, or pump with driver. A population record is entered for every component in the set of components selected, even if no failures have occurred for that component. Each data field of the population record is described below. Figure 2.1 shows the field arrangement of the population card for pumps. The fields on the population record are:

1. Plant: A plant is a facility which contains one or more nuclear power generating units. Each plant is assigned a unique identifier code.
2. Unit: A unit is a nuclear steam supply, its associated turbine-generator, auxiliaries, and engineered safety features. This code is used to identify each unit in multiunit plants.
3. System: A system is a combination of several pieces of equipment integrated to perform one or more specific functions. A three-digit

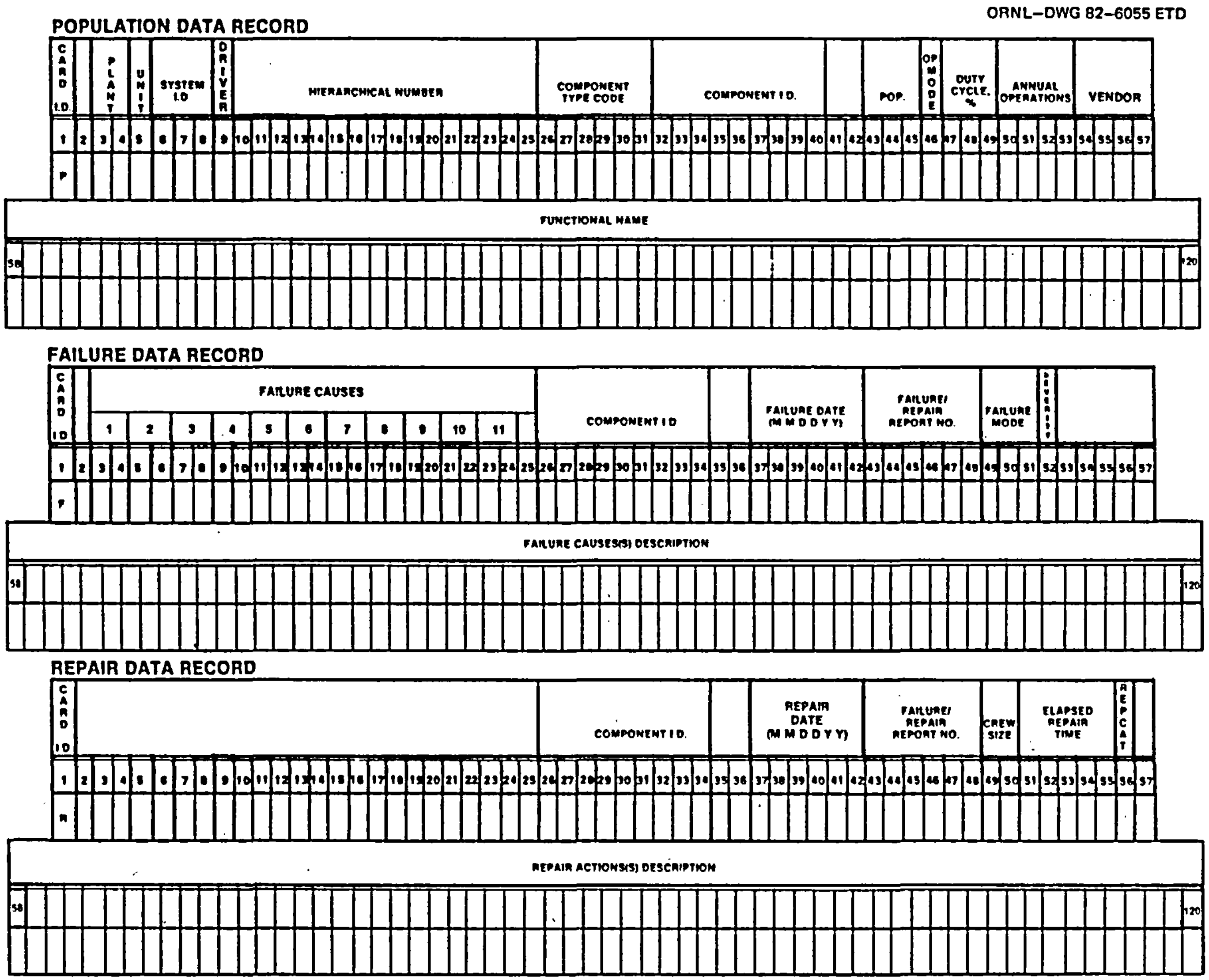

Fig. 2.1. IPRD population, failure, and repair data record forms for pumps.

system code is used to identify the system which contains a particular component. The residual heat removal system is an example.

Each nuclear power plant is divided into a set of systems which provide the necessary functions for operation of the facility. Each plant appears to have a unique definition of system boundaries, and a common nomenclature has evolved. A set of system codes was developed for NPRDS; however, these codes were judged to be inadequate for the purposes of this study. Therefore, to allow proper aggregation and comparison of components performing the same function in different plants, a set of generic systems was defined specifically for use in IPRD.

The IPRD generic systems list contains seven major system groupings. Within each major grouping are various systems and their associated subsystems. The seven major systems groupings are

1. nuclear,
2. engineered safety,
3. containment,
4. electrical,
5. power conversion,
6. process auxiliary, and
7. plant auxiliary.

For the first three major groupings, there are significant differences between the BWR and the PWR designs. Hence, a separate list was prepared for these groupings. All other groupings are essentially the same for BWRs and PWRs. Table 2.1 contains a complete list of the systems and codes. A detailed system description for each system is given in Appendix A, along with other system titles, the major components of each system, and system interfaces.

The approach used to generate the generic systems list was to evaluate several safety analysis reports (SARs). These include the CESSAR System 80 (Ref. 13), RESAR-41 (Ref. 14), B-SAR-205 (Ref. 15), GESSAR-238 (Ref. 16), and Stone and Webster PSAR (Ref. 17), and the specific SARs for William H. Zimmer Unit 1 (Ref. 18), Grand Gulf Units 1 and 2 (Ref. 19), and Oconee.[20] A systems list for the specific vendor and balance-of-plant AE was developed. The specific plant systems listings were correlated by function to generate the generic listing.

4. Driver: For pumps, this field delineates the prime movers and is coded motor (M), diesel (D), and turbine (T).

5. Hierarchical number: The hierarchical number is used to categorize the component. A scheme analogous to that used by IEEE Standard 500 was adopted for this study. The coding scheme is based on developing a set of qualifiers of the form:

$$Q_1.Q_2.Q_3. \dots Q_n .$$

Each successive qualifier categorizes the component more specifically. For example, the hierarchical number categorization scheme for

Table 2.1. IPRD generic systems list

BWR		PWR	
	Nuclear Systems – N		
N01	Reactor core	N01	Reactor core
N02	Control rod drive system	N02	Control rod drive system
N02.A	Control rod drive hydraulic system		
N03	Reactor control system	N03	Reactor control system
N04	Reactor recirculation system	N04	Reactor coolant system
N05	Standby liquid control system	N05	Emergency boration system
N06	Reactor protection system	N06	Reactor protection system
N07	Neutron monitoring/nuclear instrumentation system	N07	Nuclear monitoring/nuclear instrumentation system
N08	Residual heat removal/low-pressure safety injection system	N08	Residual heat removal/low-pressure safety injection system
N09	Reactor water cleanup system	N09	Chemical and volume control system (CVCS)
	Engineered Safety Systems – S		
S01	Reactor core isolation cooling system		
		S02	Engineered safety features actuation system
S03	Engineered safety features	S03	Safety injection system
S03.A	High-pressure coolant injection/core spray system	S03.A	High-pressure safety injection subsystem
		S03.B	Safety injection tank/core flood subsystem
S03.C	Low-pressure coolant injection	S03.C	Low-pressure safety injection subsystem
S03.D	Low-pressure core spray system		
S03.E	Automatic depressurization system		
S04	Remote shutdown system	S04	Remote shutdown system
		S05	Auxiliary feedwater system
	Containment Systems – C		
C01	Primary containment and penetrations		
C02	Reactor building	C02	Reactor building/containment and penetrations
C03	Containment heat removal	C03	Containment cooling system
		C03.A	Ice condenser system
C04	Containment isolation system	C04	Containment isolation
C05	Containment purge system	C05	Containment purge system
C06	Standby gas treatment system		
C07	Combustible gas control system	C07	Combustible gas control system
C08	Containment ventilation system	C08	Containment ventilation system
C09	Reactor building ventilation system		
C10	Containment spray system	C10	Containment spray system
		C11	Penetration room ventilation system

Table 2.1 (continued)

BWR and PWR

Electrical systems – E

E01	Main power system
E01.A	Protective relaying and controls
E02	Plant AC distribution system
E02.A	Essential power system
E02.B	Nonessential power system
E02.C	High-pressure core spray power system
E02.D	Protective relaying and controls
E03	Instrumentation and control power systems
E03.A	DC power system Vital DC power subsystem Plant DC power subsystem
E03.B	Instrument AC power system Vital instrument AC power subsystem Plant instrument AC power subsystem
E04	Emergency power system
E04.A	Diesel-generator fuel oil subsystem
E04.B	Diesel-generator cooling water subsystem
E04.C	Diesel-generator air subsystem
E04.D	Diesel-generator lubrication oil subsystem
E05	Plant lighting system
E05.A	Essential lighting
E05.B	Nonessential lighting
E06	Plant computer
E07	Switchyard
E07.A	DC control power system
E07.B	Protective relaying

Power Conversion Systems – P

P01	Main steam system
P02	Turbine-generator system
P02.A	Electro-hydraulic control subsystem
P02.B	Turbine gland seal subsystem
P02.C	Turbine lubrication subsystem
P02.D	Stator (hydrogen) cooling subsystem
P02.E	Hydrogen seal oil subsystem
P03	Turbine bypass system
P04	Condenser and condensate system
P04.A	Condenser evacuation system
P04.B	Condensate cleanup/polishing system
P04.C	Condensate heater drain subsystem
P05	Feedwater system
P05.A	Feedwater heater drain subsystem
P06	Circulating water system
P07	Steam generator blowdown (PWR)
P08	Auxiliary steam system

Process Auxiliary Systems – W

W01	Radioactive waste system
W01.A	Gaseous radwaste system offgas subsystem (BWR)
W01.B	Liquid radwaste system
W01.C	Solid radwaste system
W02	Radiation monitoring system
W02.A	Plant area radiation monitors
W02.B	Environmental radiation monitors
W02.C	Process radiation monitors
W03	Cooling water systems
W03.A	Reactor building cooling water system
W03.B	Turbine building cooling water system
W04	Service water systems
W04.A	Demineralized makeup water system
W04.B	Station service water system Essential service water system Nonessential service water system
W04.C	Chilled water system
W05	Refueling system
W06	Spent fuel storage system
W06.A	Fuel pool cooling and cleanup system
W07	Compressed air system
W07.A	Service air system
W07.B	Instrument air system
W08	Process sampling system
W09	Plant gas system
W09.A	Nitrogen system
W09.B	Hydrogen system

Plant Auxiliary Systems – X

X01	Potable and sanitary water system
X02	Fire protection system
X02.A	Water system
X02.B	Carbon dioxide system
X03	Communications system
X04	Security system
X05	Heating, ventilating, and air conditioning systems
X05.A	Control room habitability system
X05.B	Turbine building ventilation system
X05.C	Diesel building ventilation system
X05.D	Auxiliary building ventilation system
X05.E	Fuel building ventilation system
X06	Nonradioactive waste system
X06.A	Gaseous waste subsystem
X06.B	Liquid waste subsystem
X06.C	Solid waste subsystem

pumps is shown in Table 2.2. Pumps in general are classified as category 3.1. A positive displacement pump is 3.1.1, while a centrifugal pump is 3.1.2. More qualifiers are added to more specifically define the pump design. The last qualifiers define the operating environment. For example, the number 3.1.2.2.1.3.5.3 refers to a single suction-volute centrifugal pump operating with a head pressure of 100 to 990 psig, a horsepower rating 1000 to 9999 HP, and a fluid temperature greater than 300°F. For electrical and electronic components, the hierarchies in IEEE Standard 500 will be used. For mechanical components, a hierarchical numbering scheme will be developed for each component.

Together, the system and hierarchy codes provide a mechanism whereby failure and repair data for similarly designed components operating in similar environments can be grouped. This is important when estimating component reliability parameters because typically any one component has too few failures to accurately estimate a failure rate or mean repair time. To reduce the uncertainty around the mean estimates, it is usually desirable to pool data from components in different systems and/or plants. Flexible data aggregation capability also facilitates the investigation of the characteristics that are controlling the components' reliability behavior.

6. Component type code: A six-digit code, as displayed in Table 2.3, is used to cover the various types of equipment used in nuclear power plants. This coding is essentially the same as that used in NPRDS.

7. Component ID: This code is that used by the operating utility to identify components. This is the ID used on internal plant equipment lists, failure reports, and/or maintenance logs. This plant-specific ID is retained to assure proper linkage between the IPRD data base and the original plant maintenance data.

8. Component vendor: The vendor code is the four-digit code used by NPRDS.

9. Population: The number of components with the same root component ID number. For example, a system that has four identical pumps in parallel performing the same function with component IDs: P1A, P1B, P1C, and P1D has a population number size of four.

10. Operating mode: This field is used to describe the component's dominant mode of operation. For example, for pumps the three valid operating modes are running, alternating, and standby. The encoding of this data element is based on professional engineering judgment and knowledge of nuclear power plant operations.

11. Duty cycle: The percentage of time that the component is operational. The duty cycle estimate is necessary for computing the per hour failure rates. The duty cycle is affected by the system in which the component is operating and the function that the component performs in that system. Because the operational status of some components is dependent on the plant operating status, two types of duty cycles are used: availability dependent and availability independent. The availability dependent components are either directly proportional to plant availability or the complement of plant availability, that is (1 minus the plant availability with 1 signifying the plant being available 100% of the calendar time). For those components that are operated independently of plant availability status, the duty cycle is the percent of the time the component is operational during the calendar time. The encoding of this data element

Table 2.2. IPRD hierarchical numbers for pumps

- 3.1 Pumps
  - 3.1.1 Positive displacement
    - 3.1.1.1 Reciprocating
      - 3.1.1.1.1 Piston/crank shaft
      - 3.1.1.1.2 Diaphragm
    - 3.1.1.2 Rotary
      - 3.1.1.2.1 Vane
      - 3.1.1.2.2 Lobed impeller (Roots type)
      - 3.1.1.2.3 Gear type
      - 3.1.1.2.4 Axial piston (swash plate)
      - 3.1.1.2.5 Radial plunger
  - 3.1.2 Centrifugal (single or multistage)
    - 3.1.2.1 Turbine
    - 3.1.2.2 Volute (open/closed impeller)
      - 3.1.2.2.1 Single suction
      - 3.1.2.2.2 Double suction

x.y.z. Qualifiers – add digits at the end of the hierarchical number (enter zero if unknown)

.x – head, psig	.x – head, ft
.1 – 1–9	.1 – 2.3–22
.2 – 10–99	.2 – 23–230
.3 – 100–999	.3 – 231–2.3 x 10^3
.4 – >1000	.4 – >2.3 x 10^3

<u>.y – Horsepower rating, HP</u>

.1 – <1
.2 – 1–9
.3 – 10–99
.4 – 100–999
.5 – 1000–9999
.6 – >10000

<u>.z – Fluid temperature, °F</u>

.1 – <150
.2 – 150–300
.3 – >300

Table 2.3. IPRD component types codes

Component types	Code
Accumulators, tanks	ACCUMU
Air dryers	AIRDRY
Annunciator modules	ANNUNC
Batteries	BATTRY
Blowers, fans, compressors	BLOWER
Battery chargers	CHARGE
Circuit breakers, motor starters, fuses	CKTBRK
Control rods	CONROD
Control rod drive mechanisms	CRDRVE
Demineralizers	DEMINX
Electric connectors (cable, bus, wires)	ELECON
Internal combustion engines (gasoline, diesel)	ENGINE
Filters, strainers, screens	FILTER
Fuel elements	FUELXX
Generators, inverters	GENERA
Electric heaters	HEATER
Lifting devices (cranes, hoists, jacks)	HOISTX
Heat exchangers (coolers, heaters, evaporators, steam generators)	HTEXCH
Instruments, controls, sensors	INSTRU
Mechanical function units (governors, gear boxes, varidrives)	MECFUN
Motors (electric, hydraulic, pneumatic)	MOTORX
Penetrations, air locks, hatches	PENETR
Pipes, fittings	PIPEXX
Pumps	PUMPXX
Recombiners	RECOMB
Relays	RELAYX
Shock suppressors and supports	SUPORT
Switches	SWITCH
Switchgear, load control centers, motor control centers, panelboards	SWCHGR
Transformers	TRANSF
Turbines (gas, steam)	TURBIN
Valves	VALVEX
Valve operators	VALVOP
Pressure vessels (reactor vessels, pressurizers, drywells)	VESSEL

is based on professional engineering judgment and knowledge of nuclear power plant operations.

12. Annual operations: The number of times during a year that the component is cycled or demanded (e.g., starts, openings, closings, or energizings). Annual operations are entered as a four-digit integer. This number is needed for computing per demand failure rates for components operated in a cyclic fashion. The cycles or demands result from either the system being required only periodically or from tests of the component. The encoding of this data element is based on professional engineering judgment, knowledge of nuclear power plant operations, and familiarity with NRC technical specifications for periodic testing of safety-related components.

13. Functional name: This field is a free-format field and is provided for entering the component's functional name and any additional information. Examples of functional names are charging pump, containment isolation valve, and lockout relay.

2.5 Failure Data

The format of the failure record is shown in Fig. 2.1. The failure record contains the following data fields.

1. Failure cause: The failure cause code is a two-digit code which gives the circumstances during design, manufacture, or use that have led to failure. Failure causes used for pumps are from the LER pump report[4] and are given in Table 2.4. A maximum of 11 failure cause codes can be assigned to a single failure record.

2. Component ID: (See Sect. 2.4).

3. Failure date: The month, day, and year are encoded using the convention (mmddyy). The date recorded on plant maintenance records could be either the date the failure actually occurred, the date that it was detected, or the date the report was filled out. The actual failure date is the preferred date. Usually, the detection date is the date recorded.

4. Report number: This is a six-character field used to cross reference the failure record with a specific plant maintenance record, thus providing traceability to original data sources. For example, if the plant uses maintenance work orders, the plant maintenance work order number would be entered.

5. Failure mode: The failure mode is the effect by which a failure is observed. For pumps the failure modes are coded in the following manner:

Code	Description
A	Fails to start
B	Fails while running
C	Low output
D	Vibration
E	Leak
F	Other

Each type of component is assigned a unique set of failure modes.

Table 2.4. Pump failure cause codes

Code	Description	Code	Description
00	Unknown	19	Failed fasteners/welds
01	Bearing failure	20	Improper clearances
02	Personnel error	21	Filter/strainer plugged
03	Seal/packing failure	22	Failed internals
04	Design error	23	Corrosion/erosion
05	Loose fasteners/couplings	24	Thermal/overspeed/overload trip
06	Shaft/coupling failure	25	Air/vapor bound
07	Blown fuse	26	Belt drive failure
08	Binding/bound/seized	27	Motor failure
09	Quality control	28	Switch failure
10	(Descriptor not assigned)	29	Damaged seal surface
11	Misalignment	30	Excessive vibration
12	Foreign material contamination	31	High pressure
13	Bellows rupture	32	Brake failure
14	Procedural discrepancy	33	Clutch failure
15	Fabrication error	34	Cracked casing
16	Shaft key failure	35	Leaky fittings
17	Broken diaphragm	36	Failed/faulty mechanical controls
18	Normal wear	37	Control circuit failures

6. Failure severity: The failure severity is a single-character code intended to denote the extent to which the component is unable to perform its designated function. The three levels of severity are catastrophic, degraded, and incipient. Table 2.5 gives the codes and a definition for each.

7. Failure cause description: This field is a free-format field and is provided for entering the description of the failure from the maintenance work request. For example, "pump mechanical seal is leaking."

Table 2.5. Failure severity codes

Code	Description	Clarification
C	Catastrophic	The component is completely unable to perform its function. For example, a pump is frozen and will not operate; valve fails to open on demand, fails to close on demand, or fails to change positions on demand.
D	Degraded	The component operates at less than its specified performance level. A degraded failure does not normally bring about failure of the component to perform its intended function. For example, low flow from pump, or partial opening, closing, or position change of a valve.
I	Incipient	The component performs within its design envelope but exhibits characteristics which, if left unattended, will probably develop into a degraded or catastrophic failure. For example, in a pump, a leak at the mechanical seal of pump, excessive vibration, excessive noise or overheating is classified as incipient. Note that an incipient failure does not normally bring about failure of the component to perform its intended function.

2.6 Repair Data

Repair refers to the restoration or replacement of parts or components of material as necessitated by wear, damage, or failure of parts to maintain the specific part or component in efficient operating condition.[21] The format of the repair record is shown in Fig. 2.1. The repair record contains the following data fields:

1. Component ID: (See Sect. 2.4).
2. Repair date: The month, day, and year are encoded using the convention (mmddyy). The actual repair date is the preferred date but was not usually available. For these cases the failure date is repeated.
3. Report number: This is a six-character field used to cross reference the failure and repair records with a specific plant maintenance record.
4. Crew size: A two-character field is used to record the number of workers used to accomplish the repair.
5. Elapsed repair time: The actual repair time as documented on the plant's maintenance work request is encoded on the IPRD repair card. In most cases the repair time recorded is the calendar time required to repair or replace the component. Depending on the importance of the component to plant operation and the type and severity of the failure, the

repair time may be considerably less than the component's unavailable time. The plant's philosophy of what the repair time includes is important in the analysis of the component's unavailable time as well as in comparing the down times of similar components in several plants.

6. Repair category: A single-character code is used to describe the specific action taken to repair the component. Table 2.6 gives the codes and definitions.

7. Repair action description: This field is a free-format field and is provided for entering the description of the repair from the maintenance work request, for example, "replaced mechanical seal on pump."

Table 2.6. Repair category codes

Code	Description	Clarification
A	Replaced component	The entire component is replaced.
B	Minor repair	An action that is relatively easy to perform, such as adjusting the stuffing nut on a pump seal to reduce fluid leakage. Repair classification is made on the basis of task difficulty and not on the elapsed repair time.
C	Major repair	An action that is difficult or complex to perform, such as a pump overhaul.
D	Reset/adjust	Reset is an action to restore a mechanism, electrical circuit, or device to a prescribed state.[21] For example, reset a tripped circuit breaker.
		Adjust is an action to change the value of some element of the mechanism or the circuit of the instrument or of an auxiliary device to bring the indication to a desired value, within a specified tolerance for a particular value of the quantity measured.[21] For example, increasing the range of a differential pressure transmitter.
E	Recalibrate	The adjustment to a known standard, to reconcile output and input value, usually numerically related by a defined mathematical expression by adjustment of gain, zero set, etc.

3. DATA COLLECTION AND ENCODING

3.1 Data Collection

The data collection effort on which this project is based began in 1978 under the auspices of the ANSI/FIRR committee. One of the purposes of the data collection program was to establish a failure and repair data base from which failure and repair statistics analogous to those in IEEE Standard 500 could be obtained for mechanical devices in nuclear power generating plants. The absence of such data was viewed by the FIRR committee as a serious impediment to any kind of systematic evaluation of nuclear plant component reliability.

The data collection at a plant site typically proceeds as follows. The FIRR committee obtains permission for a site visit from one of the operating commercial nuclear power plants. A team leader is appointed whose responsibility is to assemble a team of volunteers for the site visit and to arrange for facilities such as office space and photocopy equipment which will be needed by the volunteer team. The volunteer team members work for nuclear component vendors, nuclear steam suppliers, AEs, standards organizations, or any of a variety of industrial concerns who have an interest in improving the performance of nuclear generating stations. A typical team consists of one leader and four team volunteers.

During the site visit, the volunteer team locates and copies any plant population lists and attempts to obtain as much detail as possible on description, specifications, and operating conditions of each component. In addition, the maintenance records are reviewed and those records that are judged to be component failures are copied. In this sorting process the following rules are followed:

1. All facility maintenance is excluded. Facility maintenance is that maintenance concerned with housekeeping, plumbing, ambient lighting, etc.

2. All construction and building modifications are excluded.

3. Routine or preventative maintenance items are excluded. Routine or preventive maintenance is any maintenance done on a regular basis such as lubrications, inspections, and calibrations. If, during the course of routine or preventive maintenance, a component failure is discovered by the maintenance crew, that component failure would be included in IPRD. The routine maintenance would be excluded.

4. Spurious conditions are included. A spurious condition is any condition reported as an abnormal component performance but which cannot be verified or reproduced by the maintenance crew. An example of a spurious condition would be a meter reading that is reported as abnormal but is shown by follow-up investigation to be normal.

Use of these four rules results in the collection of maintenance items that are defined as component failures or corrective maintenance items. By definition, corrective maintenance is any repair or replacement action necessary to restore a component from a state of actual degradation (i.e., catastrophic and degraded failures) or from a state of anticipated degradation (i.e., incipient failure).

Plant records are either duplicated using photocopy equipment or hand transcribed. In those cases where computer files exist, the data files are printed out. The plant data are maintained by the secretariat of the FIRR committee and are physically located in the IEEE Standards Office in New York.

3.2 Data Selection

The selection of plant data for inclusion in IPRD is made on the basis of (1) the existence of population data, (2) the quality of failure and repair data and (3) the ability to obtain hierarchical information on components. Secondary considerations include trying to achieve an adequate balance among the four nuclear steam suppliers and the various AEs.

3.3 Data Encoding

3.3.1 Manual data entry

In the early phases of this project, population, failure, and repair data were transcribed onto encoding forms. Data from the forms were key punched to cards and the cards were read into the computer.

3.3.2 Computerized data entry

A computerized data entry system was developed to increase the efficiency and productivity of staff as well as to reduce errors. The data encoded by the clerical staff were

- Population record: plant, unit, component type, component ID, component vendor, functional name;
- Failure record: failure date, report number, failure cause text; and
- Repair record: repair date, crew size, repair time, repair action text.

Data were encoded onto floppy disks using a minicomputer system. Printouts were subsequently produced. Data requiring engineering judgment were encoded by engineers writing on these printouts, as follows:

- Population record: system, hierarchical number, operating mode, duty cycle, annual operations;
- Failure record: failure cause, failure mode, failure severity; and
- Repair record: repair category.

The clerical staff then entered these data into a minicomputer to complete the data base entry.

3.4 Quality Control

Quality control checks data records for accurate and consistent encoding of the data fields. Quality control is performed in several different ways, including:

1. Computer consistency checks: The first level of quality control was performed by computer software that checked for obvious errors during data entry, such as making sure that each code assigned was one of the admissible codes. Other checks that were made include assuring that failure and repair records occurred in pairs and that a population record existed for every pair of failure and repair records.

2. Engineering consistency checks: There were two separate steps to check the engineering judgment used in assigning codes. To ensure that the data were classified accurately and consistently, a validation program was developed. This step consisted of two major parts: individual record checking and cross checking.

In the individual record checking procedure, an engineer reviewed every population, failure, and repair record in the data base for completeness and accuracy of the coding with the descriptor fields, for instance, checking that the failure cause descriptor matched the failure cause code. The cross checking procedure was performed by sorting the data by a specific parameter such as failure cause code, failure mode, severity code, and repair category. In this manner, the individual records were checked for coding consistency against those within the classification. Those records that were inconsistent with the definition of the code classification as well as those that did not appear consistent with the bulk of the records within that classification were corrected.

3. Data encoding checks: A final check was made to assess the accuracy of data encoding. This check was made by taking a random sample of records from the data copied by volunteer teams during the plant visit. The original data was checked against the final IPRD entries. An estimate was made of encoding accuracy. Where applicable the following fields were checked:

A. Failure date,
B. Failure description,
C. Repair description,
D. Repair time, and
E. Report number.

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Appendix A

GENERIC SYSTEM DESCRIPTIONS

A.1 Nuclear Systems – N

The nuclear system grouping is comprised of the reactor core and those systems and subsystems that monitor and control the core's reactivity, remove heat from the core, and otherwise directly support the safe operation of the reactor. The systems and subsystems that make up the nuclear system grouping are shown in Table A.1. The following descriptions of systems are identified as applying to a boiling-water reactor (BWR) or pressurized-water reactor (PWR) or both.

Table A.1. Nuclear systems

	BWR	PWR
N01	Reactor core	Reactor core
N02	Control rod drive system	Control rod drive system
N02.A	Control rod drive hydraulic system	
N03	Reactor control system	Reactor control system
N04	Reactor recirculation system (including reactor vessel and internals)	Reactor coolant system (including reactor vessel and internals)
N05	Standby liquid control system	Emergency boration system[a]
N06	Reactor protection system	Reactor protection system
N07	Neutron monitoring/nuclear instrumentation system	Nuclear monitoring/nuclear instrumentation system
N08	Residual heat removal/ low-pressure safety injection system	Residual heat removal/ low-pressure safety injection system
N09	Reactor water cleanup system	Chemical and volume control system

[a]Westinghouse units have emergency boration as a unique system. Babcock & Wilcox and Combustion Engineering units provide emergency boration as a function of the CVCS.

N01 Reactor core (BWR and PWR)

The reactor core produces heat through nuclear fission. A reactor core contains about 200 to 300 fuel assemblies in a PWR and ~700 in a BWR which are arranged in a lattice fashion approximating an upright cylinder. The exact number of fuel assemblies varies with both the nuclear steam supply system (NSSS) vendor and the generation of the reactor design. Fuel assemblies are constructed of uranium dioxide (UO_2) pellets surrounded by a Zircaloy cladding tube. The lattice structure of the reactor core is a loose one that allows reactor coolant water to flow up and around the fuel assemblies. Empty spaces are also left in the lattice for the insertion of control rods.

Systems that interface with the reactor core are the control rod drive system and the reactor coolant system in PWRs or the main steam, feedwater, and reactor recirculation systems in BWRs. There are also some power and temperature measuring components that may be inserted directly into the reactor core.

N02 Control rod drive system (BWR and PWR)

The control rod drive system is used for power shaping and reactivity control. The control rod drive system is made of ~50 control rod assemblies in PWRs and 185 in BWRs. The neutron absorber in the control rods is boron carbide (B_4C) or silver-indium-cadmium alloys. As with the fuel assemblies, the exact number of control rod assemblies depends on the NSSS vendor and the reactor design. Also included in the control rod drive system is an energy source for the drive mechanisms. For PWRs the energy source is electrical power, and for BWRs it is a charged hydraulic system. When fully inserted into the reactor core, the control rods provide sufficient negative reactivity to stop the core's chain reaction while the reactor is at operating temperatures. Power shaping is accomplished by driving selected control rods to predetermined positions within the reactor core.

In PWRs, control rods are held above the reactor core suspended by electromechanical devices. When a trip signal is received, the electrical current passing through the devices is interrupted causing them to release the control rods. The control rods then drop due to gravity through their guide tubes into the reactor core. Control rod drive stators are typically air cooled by the control rod drive mechanism fans that draw air from the containment through a chiller.

For BWRs, hydraulic pressure is used to force the control rods up into the core from their position below the reactor. The loss of electrical signal will actuate two pilot valves that in turn activate a scram inlet valve and a scram exhaust valve. The scram exhaust valve operates slightly before the inlet valve and exhausts water from the guide tube above the control rod drive mechanism's drive piston. The scram inlet opens to supply pressurized water to the bottom of the drive piston. This dual action rapidly forces the control rod up into the reactor core.

Control rod drive systems in PWRs interface with the plant ac distribution system, the reactor building cooling water system, the dc power system, the reactor protection system, the reactor control system, and the reactor core. In BWRs, the control rod drive system interfaces with the plant ac distribution system, the reactor protection system, the reactor control system, the control rod drive hydraulic subsystem, and the reactor core.

N02.A Control rod drive hydraulic system (BWR). The control rod drive hydraulic system supplies and controls the pressure and flow of water to and from the BWR control rod drives through a hydraulic control unit (HCU). There is one HCU for each control rod drive. The control rod drive system is required to:

1. recharge the scram accumulators (part of the HCUs) during a scram reset or during system startup (a scram discharge volume is provided for all control rod drives to receive the water displaced by the drives during a reactor scram);
2. maintain, during rod insertion or withdrawal, drive pressure sufficiently above reactor vesssel pressure to permit operation of drive mechanisms;
3. supply cooling water to the HCUs; and
4. pressurize the return line slightly above reactor vessel pressure to receive the normal drive discharge to the reactor vessel.

The control rod drive hydraulic system is a one-train system designed to supply pressurized water to the hydraulic control units. The HCUs are connected to a pump discharge by four parallel piping headers: accumulator charging, drive, cooling, and return. Each HCU furnishes, on signal, the pressurized water to a drive unit that positions a control rod. Each HCU has a number of valves that control water flow to and from a drive piston and an accumulator that stores sufficient energy to fully insert its associated control rod at lower reactor vessel pressures. The control rod drive hydraulic system is a Safety Class 2 system, and it is located within the secondary containment.

Water is drawn from the condensate storage tank by one of two redundant pumps. The other pump is an installed spare. The water is filtered on pump suction and again, by one of two drive water filters, on pump discharge. A flow control valve downstream of the drive water filters maintains a constant system flow rate. A charging water header supplies charging flow to each of the HCUs and to the drive header. Pressure in the drive header is maintained by a pressure control valve located between the charging and cooling headers.

Flow through the drive headers to the HCUs and the return flow from the HCUs are stabilized by passing the flow through two parallel, solenoid operated stabilizing valves. The drive header also supplies flow to the cooling header. Cooling header pressure is maintained by a pressure control valve located between the drive and cooling headers. A return header routes water discharged from the HCUs during normal operation and water passing through the cooling header pressure to the reactor vessel. During a scram, water discharged through the HCUs flows into a common header and then into an instrument volume. This piping arrangement makes up the scram discharge system.

The control rod drive hydraulic system interfaces with the following systems:

- plant ac distribution
- reactor protection
- reactor control (rod control and information)
- condensate storage

N03 Reactor control system (BWR and PWR)

The reactor control system provides the means for monitoring and controlling control rod position during normal operating conditions. The objective of the reactor control system is to match reactor power as closely as possible to unit demand while maintaining a controlled power level and balanced flux distribution in the reactor core. In BWRs, this is accomplished by a manual control system in which an operator can selectively position a single control rod or a control rod group. The electrical circuitry, switches, indicators, and alarms that are needed to provide input signals for the control rod drive system make up the reactor manual control system. In PWRs, the control function may be either automatic or manual. While in the automatic mode, the input signal to the control rod mechanisms is produced by comparing the reactor power level with the unit demand as seen by the turbine. The relative difference between these signals controls the motion of the control rod drive mechanism. At low power conditions, the reactor is under manual control by an operator. The operator may also assume manual control at any time while the reactor is in a normal operating mode. The design and capabilities of the PWR reactor control systems vary depending on the NSSS vendor. In reactors designed by Babcock & Wilcox (B&W), this system is called the integrated control system (ICS). In B&W units, the ICS also controls the feedwater and turbine generator systems. Combustion Engineering (CE) uses the title reactor regulating system. The reactor control systems interface with the 120 Vac instrument power system, the control rod drive system, the nuclear instrumentation system, and the nonnuclear instrumentation system of their respective units.

N04 Reactor recirculation system (BWR)

The reactor recirculation system pumps reactor coolant through the core. This is accomplished by two recirculation loops that are external to the reactor vessel but inside the containment. A constant flow rate is maintained by either variable speed pumps (older BWRs) or a flow control valve (newer BWRs). The reactor recirculation system is a Safety Class 1 system. Electrical power for each recirculation loop is provided by separate power trains.

The reactor vessel is also included as a part of the reactor recirculation system. The vessel contains, in addition to the core, the core support structures; the steam separators and dryers; the jet pumps; the control rod guide tubes; distribution lines for feedwater, core sprays, and liquid control; the in-core instruments; and other components. The reactor core is cooled by demineralized water that enters the lower portion of the core and boils as it flows upward through the fuel bundles. Steam leaving the core is dried by the steam separators and dryers located in the upper portion of the reactor vessel. The steam is then directed to the turbine by the main steam system.

A typical recirculation loop (in newer BWRs) contains four motor operated valves and one hydraulically operated valve, a recirculation pump, and twelve jet pumps. Two motor operated valves are used as pump suction and pump discharge valves. A third motor operated valve is used to bypass the larger discharge shutoff valve to warm the pipeline during the hot standby condition. The last motor operated valve is in a line that bypasses both the flow control valve and the discharge shutoff valve. This valve is manually set to adjust bypass flow control. The hydraulically operated flow control valve is located between the recirculation pump and the discharge shutoff valve. Downstream of the bypass line and discharge shutoff valve are the jet pumps. The jet pumps are internal to the reactor vessel and are located between the inner vessel wall and the core shroud. For older BWRs, the flow control valve is omitted and a variable speed recirculation pump is used.

Systems interfacing with the reactor recirculation system are

- main steam and feedwater (through the vessel)
- control rod drive
- reactor core
- plant ac distribution
- dc power (for pump and valve control circuitry)
- instrument air (for flow control valve)
- high- and low-pressure core spray, reactor core isolation cooling, residual heat removal, and reactor water cleanup (all through the vessel)
- reactor building cooling water (recirculation pump motor cooling)
- reactor vessel support

N04 Reactor coolant system (PWR)

The reactor coolant system transports heat from the reactor core to the steam generators where steam is produced for use in the unit's turbine. Depending on the NSSS vendor and the generation of the design, two to four coolant loops are used. Loop design may involve one reactor coolant pump and one steam generator per loop or two reactor coolant pumps and one steam generator per loop. The reactor coolant system is a Safety Class 1 system. Electrical power is provided by two or three separate buses depending on reactor design. The reactor coolant system is contained entirely within the containment building.

In PWRs, reactor coolant pressure is maintained by a pressurizer that is connected to the piping between the reactor vessel outlet and the steam generator inlet (hot leg). Pressure is controlled by altering the saturation temperature and pressure of the steam and water inside the pressurizer. Condensing the steam volume by spraying cooler water into it has the net effect of reducing system pressure. Increasing the steam temperature by using electric heaters to heat the water volume has the effect of increasing system pressure. A system of pressure relief valves is provided to prevent overpressurizing the reactor coolant system.

The steam generators are vertical shell and tube (either straight or U-tube) heat exchangers in which hot reactor coolant water passes through the tubes heating secondary water on the shell side to the boiling point. Steam drying is accomplished inside the steam generators. Steam leaving the steam generators is transported to the unit's turbine by the main steam system.

The reactor vessel is an integral part of the reactor coolant system. The vessel makes up a part of the primary system pressure boundary, and it contains the reactor core, the core support structures, the in-core instruments, and the control rod guide tubes. In the B&W NSSS, vent valves are placed between the upper plenum and the downcomer.

The reactor coolant system interfaces with the following systems:

- reactor core
- control rod drive
- plant ac distribution
- dc power (for pump control circuitry)
- high-pressure safety injection
- residual heat removal
- chemical and volume control
- emergency boration (Westinghouse reactors)
- main steam and feedwater (through the steam generators)
- reactor building cooling water (reactor coolant pump motor cooling)
- reactor vessel support
- reactor cooling system piping

N05 Standby liquid control system (BWR)

The standby liquid control system is designed to operate in the event that an insufficient number of control rods can be inserted into the reactor to shut it down in a normal manner. Although the standby liquid control system does not operate as quickly as the control rods, it does provide a redundant, independent, and diverse method for bringing the fission process to subcriticality and maintaining subcriticality as the reactor cools. The standby liquid control system operates by injecting concentrated boric acid into the coolant water. The system is sized to bring the reactor from rated power to cold shutdown. The standby liquid control system is a manually initiated Safety Class 2 system.

A typical system configuration has two redundant pumping trains drawing from a common boric acid solution tank and discharging into a single line connecting to the reactor vessel. A recirculation line through a test tank is also provided. The standby liquid control system is located within the reactor building and penetrates primary containment. Electrical power is provided by two separate independent buses.

Systems typically interfaced with the standby liquid control system are

- essential ac distribution subsystem
- dc power (for pump and valve control circuitry)

- demineralized water
- reactor core and reactor recirculation (inside the vessel)

N05 <u>Emergency boration system (PWR)</u>

In the event of a steam line break or the spurious actuation of a main steam pressure relief valve in the presence of a stuck rod, PWRs require the rapid injection of a concentrated boron solution to control reactivity and to shut down the reactor. In reactors of Westinghouse design, this function is provided by the emergency boration system. In CE and B&W reactors, boron injection is a function of the chemical and volume control system. The emergency boration system is a Safety Class 2 system that penetrates containment. Electrical power is supplied by the essential ac distribution subsystem.

The emergency boration system is often a single train system with redundant boron injection pumps. The pumps draw suction from a single tank and discharge into a header that connects to the suction side of each of the reactor coolant cold legs. Piping also connects the reactor coolant pump discharge to the boron injection tank. A recirculation loop is provided for testing and recirculation of concentrated boric acid and usually contains a pump and surge tank.

The emergency boration system interfaces with the following systems:

- essential ac distribution subsystem
- reactor coolant
- chemical and volume control
- dc power (for pump control circuitry)
- emergency core cooling system actuation

N06 <u>Reactor protection system (BWR and PWR)</u>

The reactor protection system monitors the reactor, reactor coolant system, and other plant parameters important to safety. Upon an abnormal condition of any one or a combination of the various parameters, it instructs the control rod drive system to shut down the reactor to prevent fuel damage and in some instances to prevent the overpressurization of the reactor coolant system. The parameters typically monitored are

- reactor power and core power balance (axial and radial)
- reactor coolant temperature and pressure
- operation of the reactor coolant or reactor recirculation pumps
- dry well (primary containment) pressure
- feedwater flow
- turbine operation

This list is not inclusive; other parameters may be monitored depending on the NSSS vendor and the generation of the design.

Reactor protection systems are Class 1E systems and typically consist of three or four signal processing channels that are kept electrically independent and physically separate. The channel outputs are combined into a two-out-of-three (2/3) or a two-out-of-four (2/4) voting logic.

The manner in which the 2/4 logic is developed depends on the NSSS vendor. Each signal processing channel monitors all of the plant parameters used by the particular design. When an abnormal condition of any of the parameters is sensed by a channel, the channel goes into a "trip" state. The coincidence of two signals on any of the three or four channels will cause the reactor protection system to initiate a reactor trip or shutdown.

Systems that interface with the reactor protection system are

- nuclear instrumentation or neutron monitoring
- nonnuclear instrumentation
- ac instrument power
- control rod drive

N07 Neutron monitoring/nuclear instrumentation system (BWR and PWR)

The neutron monitoring or nuclear instrumentation system is used to monitor reactor performance and to provide this information to the plant operators, the reactor control system, and the reactor protection system. The neutron monitoring/nuclear instrumentation systems are typically made up of four to six subsystems depending on the NSSS vendor. A source range monitor subsystem provides neutron flux indications during startup and low power operation. The intermediate range monitor system operates during low- and intermediate-power operations. In some designs, it may also input into the reactor protection system. The power range monitor subsystem provides flux indications during full power operation. It is the prime source of neutron flux information for the reactor protection system. These three subsystems measure average reactor power and power imbalances. Some designs incorporate a local power range subsystem, which can provide detailed information about neutron flux throughout the reactor core and under any power level operation. In-core monitoring subsystems are also included and are usually used to calibrate the other subsystems. A rod block or rod withdrawal inhibit subsystem is also common and is used to prevent control rod withdrawals unless reactor coolant flow rates are within established boundaries.

The various sensors, with the exception of the in-core monitors, are mounted on the outside of the reactor vessel. All subsystems connected to the reactor protection system are classified as Class 1E. In-core monitors are typically utilized for fuel management and are not classified 1E.

Systems that interface with the neutron monitoring/nuclear instrumentation system are

- ac instrument power
- reactor protection
- reactor control
- control rod drive

N08 Residual heat removal/low-pressure safety injection system (BWR and PWR)

The residual heat removal/low-pressure safety injection system (RHR/LPSIS) is a versatile system that performs several functions during the various states of reactor operation. Its primary function is to remove heat from the reactor core during normal shutdown, loss-of-coolant-accident (LOCA), and post-LOCA conditions. It may also assist in containment heat removal and containment spray operations. The RHR/LPSIS is a Safety Class 2 system which is powered by the essential ac distribution subsystem. In some designs the RHR/LPSIS is called the decay heat removal/low-pressure safety injection system or the shutdown cooling/low-pressure safety injection system.

The RHR/LPSIS is a two- or three-train system depending on the NSSS vendor and the design generation. Each train of the two-train designs has a 100% capacity and is redundant to the other train. Three-train designs require two of the three trains to operate in order to accomplish their function. Each train consists of a pump and a heat exchanger with their associated valves. In PWRs, the pumps can be aligned to draw suction from the borated refueling water storage tank, the containment sump, or a hot leg (or legs) of the reactor coolant system. In BWRs, suction is from the suppression pool. After passing through the heat exchanger, discharge piping may connect at several points and be valved in or out depending on the operational mode of the RHR/LPSIS.

During a normal shutdown, valve alignment directs flow from the reactor coolant system hot leg (suppression pool in BWRs) through the RHR pumps and heat exchangers into the reactor vessel (reactor recirculation system pump discharge in BWRs). Following a large LOCA, the LPSIS goes into a coolant injection mode when reactor vessel pressure lowers enough to permit operation. In this case, valve alignment directs suction from the borated or refueling water storage tank (still the suppression pool in BWRs) through the pumps and heat exchangers or piping to the reactor vessel. When the water level in the borated or refueling water storage tank falls to a predetermined height, suction in PWRs is changed over to the containment sump and the storage tank is usually valved out to prevent pump cavitation. This phase of operation is known as the recirculation mode. In BWRs, the recirculation mode has the same flow path as the injection mode. In some cases, suction is taken off RHR/LPSIS piping for containment spray or high-pressure coolant injection. In BWRs, suppression pool temperature can be limited by directing a portion of the cooled LPSIS flow into the suppression pool. This is called the containment heat removal mode. The RHR/LPSIS interfaces with the following systems:

- essential ac distribution subsystem
- dc power (for pump and valve control circuitry)
- reactor coolant/reactor recirculation
- reactor building service water, reactor building cooling water or the chemical and volume control (for secondary heat exchanger cooling)
- refueling (in PWRs)
- engineered safety features actuation

Other interfaces may include:

- containment spray system
- reactor core isolation cooling system (in BWRs)
- high-pressure safety injection system

N09 Reactor water cleanup system (BWR)

The reactor water cleanup system is used to recirculate a portion of the reactor coolant through a filter-demineralizer system to remove particulate and dissolved impurities from the reactor coolant. It is also used to remove excess coolant from the reactor system in a controlled fashion. The reactor water cleanup system is a nonsafety system even though a portion of its piping is a part of the reactor coolant pressure boundary. These piping sections can be isolated if necessary.

One of two parallel cleanup recirculation pumps draws suction from reactor recirculation system piping and discharges through a series of regenerative and then nonregenerative heat exchangers. Flow out of the nonregenerative heat exchangers may be passed through one of two parallel filter-demineralizer units, or the filter-demineralizers may be bypassed. From the filter-demineralizers or the by-pass, flow can be directed back through the regenerative heat exchangers and into the feedwater system piping or directly to the condensate system. It is also possible to direct flow into the liquid radwaste system.

The reactor water cleanup system interfaces with the following systems:

- plant ac distribution
- dc power (for pump and valve control circuitry)
- reactor recirculation
- containment isolation
- reactor building closed cooling water (secondary side of nonregenerative heat exchangers)
- feedwater
- condensate storage
- liquid radwaste

N09 Chemical and volume control system (PWR)

The chemical and volume control system (CVCS) controls the volume, purity, and boric acid content of the reactor coolant. The CVCS also provides seal injection water for the reactor coolant pumps. Coolant purity is controlled by continuously purifying a bypass stream of reactor coolant. Adjustments in coolant volume are made automatically to maintain a predetermined level in the pressurizer. A "bleed and feed" technique is used to control the boric acid concentration in the reactor coolant. In B&W and CE units, the CVCS also supplies the emergency boration function.

The CVCS is in general a nonsafety system even though portions of it, such as the emergency boration piping and those sections of piping that form a part of the reactor coolant pressure boundary, are safety grade piping. Electrical power is supplied by the plant ac distribution system.

In B&W units, the chemical addition and boron recovery system, the letdown and purification system, and portions of the high-pressure injection system are the equivalent of the chemical and volume control system.

For coolant purification, a typical flow path through the CVCS is letdown from the reactor coolant system, usually through a regenerative heat exchanger into a series of filters and the demineralizers to the volume control tank. Charging pumps draw suction from the volume control tank and discharge, usually, back through the regenerative heat exchanger into the reactor coolant system cold leg piping. The volume control tank acts as a supply of coolant for increasing reactor coolant inventory and as a storage tank for inventory reduction.

The volume control tank is typically tied to the borated or refueling water storage tank or the boric acid mix tanks through boric acid transfer pumps and filters. Boric acid concentration in the reactor coolant is controlled by diverting the letdown stream to the boron recovery subsystem and sending either concentrated boric acid or demineralized water through the charging pumps.

Reactor coolant pump seal leak off is collected in a header and directed, usually, through a cooling heat exchanger to the volume control tank. From there the charging pumps send a stream of flow through one of two redundant filters and a flow control valve into a header connected to each reactor coolant pump's seals.

Systems that interface with the chemical and volume control system are

- plant ac distribution
- dc power (for pump and valve control circuitry)
- reactor coolant
- reactor building cooling water system (for nonregenerative heat exchanger secondary flow)
- demineralized water
- engineered safety features actuation
- refueling
- liquid radwaste (for sluice from the filters and demineralizers)
- instrument air (for flow control valve)
- gaseous radwaste (vents)

A.2 Engineered Safety Systems — S

The engineered safety systems grouping is made up of those systems, other than containment systems, that are used to mitigate the effects of a reactor accident such as an LOCA. The systems and subsystems that make up the engineered safety systems grouping are listed in Table A.2. The following systems descriptions are identified as relating to a BWR or PWR or both.

S01 Reactor core isolation cooling system (BWR)

The reactor core isolation cooling system (RCIC) provides makeup water to the reactor vessel when the vessel is isolated. The RCIC system

Table A.2. Engineered safety systems

	BWR	PWR
S01	Reactor core isolation cooling system	
S02	Engineered safety features activated by a number of independent control systems	Engineered safety features actuation system
S03	Engineered safety features	Safety injection system
S03.A	High-pressure core injection/spray system	High-pressure safety injection subsystem[a]
S03.B		Safety injection tank/core flood subsystem[a]
S03.C	Low-pressure coolant injection (a functional mode of the residual heat removal system)	Low-pressure safety injection subsystem (a functional mode of the residual heat removal system)[a]
S03.D	Low-pressure core spray system	
S03.E	Automatic depressurization system	
S04	Remote shutdown system	Remote shutdown system
S05		Auxiliary feedwater system

[a] B&W units have these as separate, independent systems.

is an automatically initiated system designated to maintain an adequate water level in the reactor vessel. The RCIC system is a Safety Class 2 system that penetrates primary containment. Electrical power is provided by the dc power system.

The RCIC system's primary source of water is the condensate storage tank. The secondary source is the suppression pool. Valves may also be aligned to provide water from the RHR/LPSIS. A steam turbine-driven pump draws suction from these three sources and discharges through a shutoff valve into the reactor vessel. Steam to operate the turbine is generated in the reactor vessel by decay heat in the reactor core.

Systems that interface with the RCIC are

- Reactor coolant (in the suppression pool and in the form of steam for the turbine)
- dc power (for pump and valve control circuitry plus electrical power for valve operators)

- residual heat removal/low-pressure safety injection
- condensate storage

S02 Engineered safety features actuation system (PWR)

The engineered safety features actuation system (ESFAS) monitors a number of plant parameters important to safety. An abnormal indication of any of these parameters will cause the ESFAS to activate and the corresponding mitigation system to prevent the occurrence of excessive offsite radiation. Typical parameters that are monitored are

- reactor coolant or pressurizer pressure
- reactor building pressure
- refueling or borated water storage tank water level
- steam generator pressure (shell side)

The mitigating actions initiated by the ESFAS are

- high-pressure safety injection
- low-pressure safety injection
- containment isolation
- low-pressure change to recirculation
- containment cooling through ventilation
- containment spray
- steam generator level control (in some systems)

There is no equivalent of the ESFAS in BWRs. Each BWR system that provides the equivalent of one of the above functions has its own actuation and control system

The ESFAS is a Class 1E system that consists of three or four signal processing channels for each parameter which output typically into two 2/3 or 2/4 voting logics. The exact combination of signal processing channels and voting logics for a particular reactor depends on the NSSS vendor. An abnormal value for any of the monitored parameters will cause its signal processing channel to go into a "trip" state for each of the two voting logics. The coincidence of two tripped channels for the same parameter will cause the ESFAS to actuate the associated mitigation system or systems.

Systems that interface with the ESFAS are

- instrument ac power
- nonnuclear instrumentation
- high-pressure safety injection
- residual heat removal/low-pressure safety injection
- containment cooling
- containment spray
- auxiliary feedwater
- any system with valves that must be closed to isolate the containment

S03 Safety injection system (PWR)

The safety injection system is composed of three subsystems: the high-pressure safety injection subsystem, the safety injection tank/core flood subsystem, and the low-pressure safety injection subsystem. In B&W units, these subsystems are treated as individual systems. Each subsystem is discussed below.

S03.A High-pressure safety injection subsystem (PWR). The high-pressure safety injection subsystem (HPSIS) is designed to operate for small LOCAs when reactor coolant pressure has not been significantly reduced. In this circumstance, the HPSIS injects borated water into the reactor coolant system to provide cooling to limit core damage and fission product release and to ensure an adequate shutdown margin. The HPSIS is actuated by the ESFAS and is powered electrically by the essential ac distribution subsystem. The HPSIS is a Safety Class 2 system.

The HPSIS has two or three redundant trains depending on the NSSS vendor. A typical train consists of a high head pump that draws suction from the refueling water storage or volume control tank, again depending on the NSSS vendor. All pumps are started on an initiation signal. The pumps discharge flows into the cold legs or hot legs of each reactor coolant loop. The particular connection point depends on the NSSS vendor. Both B&W and CE units have cold leg connections.

The HPSIS interfaces with the following systems:

- engineered safety features actuation
- essential ac distribution subsystem
- dc power (for pump and valve control circuitry)
- chemical and volume control or refueling
- residual heat removal/low-pressure safety injection (for alternate suction)

S03.B Safety injection tank/core flood subsystem (PWR). The safety injection tank/core flood subsystem is a passive system that requires no external signal or power source to operate. It is designed to rapidly inject cooling water into the reactor vessel when vessel pressure falls below a predetermined level. The safety injection tank/core flood subsystem is a Safety Class 2 system.

In Westinghouse and CE designed reactors, the safety injection tank subsystem is a subsystem of the safety injection system. In B&W units, the core flood system is a separate system that shares common reactor vessel inlet piping with the RHR/LPSIS.

Typically, there is one safety injection tank for each safety injection train. The tanks contain borated water and are pressurized to about 600 psig. Nitrogen is used to provide the charging pressure. The outlet of a safety injection tank is connected to a check valve that directs flow out of the tank. In series with the check valve is a motor operated isolation valve. Under normal operating conditions, the motor operated valves are open and reactor coolant pressure against the check valve outlets is enough to keep the valves closed. In the cold shutdown condition when reactor vessel pressure is not high enough to prevent the check valves opening, the motor operated valves are closed. In the event of a large LOCA, reactor vessel pressure will decrease. When the pressure has lowered below the charging pressure in the safety injection tanks, the check valves will open, injecting cooling water into the vessel. The safety injection tanks contain enough borated water to cover the reactor core.

Systems that interface with the safety injection tank/core flood subsystem are:

- reactor coolant
- nitrogen subsystem of the plant gas system
- borated or refueling water storage tank
- other subsystems of the safety injection system or the RHR/LPSIS
- plant ac distribution (for motor operated valve operation)
- engineered safety features actuation

S03.C Low-pressure safety injection subsystem (PWR). In some plants low-pressure safety injection is a function of the RHR/LPSIS, which is discussed in Sect. N08.

S03 Engineered safety features (BWR)

Reactors of General Electric design do not have a single equivalent of the safety injection system. There are four systems that perform essentially the same functions, but they actuate and operate more or less independently. The four systems are high-pressure core injection/spray (HPCI/S), low-pressure coolant injection, low-pressure core spray (LPCS), and automatic depressurization. Each is described below.

S03.A High-pressure core injection/spray system (BWR). The HPCI/S system maintains an adequate coolant inventory inside the reactor vessel to keep fuel cladding temperatures below fragmentation temperature in the event of small breaks in the reactor coolant pressure boundary. In older BWR designs, an injection system was used to input cooling water into the feedwater lines for core cooling. New BWR designs use a spray sparger mounted above the reactor core for cooling water injection. The HPCI/S system cooling is used to help decrease vessel pressure to allow operation of the low-pressure cooling systems. In the new BWR designs, the HPCI/S system operates independently of all other systems. It is provided its own dedicated power system. The HPCI/S system is a Safety Class 2 system that penetrates primary containment. The HPCI/S system may be used as a backup to the RCIC system.

The HPCI/S system is a single train system that may have redundant pumps. Suction may be drawn from the condensate storage tank or the suppression pool. Pump discharge is through a series of check valves into the feedwater line (old BWRs) or the spray sparger (new BWRs). The HPCI/S system is actuated by its own control system on the occurrence of high containment pressure or low water level in the vessel. A recirculation line is provided for testing.

Systems that interface with the HPCI/S system are

- essential ac distribution subsystem (old BWRs) or HPCI/S power system (new BWRs)
- dc power (old BWRs for pump control circuitry)

- condensate storage
- feedwater (old BWRs)
- suppression chamber

S03.C Low-pressure coolant injection system (BWR). Low-pressure coolant injection is a function of the RHR/LPSIS which is discussed in Sect. N08.

S03.D Low-pressure core spray system (BWR). The low-pressure core spray (LPCS) system is designed to deliver low-pressure cooling water to a spray sparger located over the reactor core. The LPCS system has the ability to cool the fuel by spraying water into each fuel channel. The LPCS system can only be activated after reactor vessel pressure has been reduced. In conjunction with the automatic depressurization system or the HPCI/S system, the LPCS system can maintain fuel cladding below the fragmentation temperature. The LPCS system is a Safety Class 2 system and is powered from the essential ac distribution subsystem.

The LPCS system consists of a single independent pump that draws suction from the suppression pool and discharges through a series of check valves into the low-pressure spray sparger over the reactor core. The LPCS system is actuated by its own control system upon the indication of a reactor coolant pressure boundary breach, but it is prevented from operation until after reactor vessel pressure is reduced. A recirculation line is provided for testing.

The following systems interface with the LPCS system:

- essential ac distribution subsystem
- dc power (for pump control circuity)
- suppression pool
- instrument air (for air operated check valve inside containment)

S03.E Automatic depressurization system (BWR). The automatic depressurization system operates to reduce reactor vessel pressure to permit the operation of low-pressure cooling systems in an LOCA situation in which the HPCI/S system is not maintaining adequate coolant inventory. The automatic depressurization system is a control system that actuates some of the relief valves mounted in the main steam lines. The automatic depressurization system actuates on the indication that a reactor coolant pressure boundary breach has occurred, that the HPCI/S system is not providing sufficient coolant flow, and that the LPCS or the RHR/LPCIS pumps are in operation. The automatic depressurization system is a Class 1E system.

The only system with a direct interface with the automatic depressurization system is the main steam system.

S04 Remote shutdown system (BWR and PWR)

The remote shutdown system, in both BWRs and PWRs, usually consists of a remote control panel from which the systems necessary to bring the plant to a safe cold shutdown can be operated. The controls on the panelboard are redundant to their counterparts in the control room, and they are to be used in the event that the control room is damaged or becomes uninhabitable. The controls on the remote shutdown panel vary between

NSSS designs. In some instances the remote shutdown system is a bunkered system. It is designated as a Class 1E system and interfaces with the following systems:

- control rod drive
- nuclear instrumentation
- nonnuclear instrumentation
- essential ac power

S05 <u>Auxiliary feedwater system (PWR)</u>

The auxiliary feedwater system (or emergency feedwater system) is designed to provide an adequate supply of cooling water to the steam generators so that they can act as heat sinks for decay heat removal from the reactor core in the event of a loss of power, a feedwater line malfunction, a small LOCA, or a main steam line break. The system is a Safety Class 3 system with the exception of the piping between the isolation valves and the connections to the feedwater piping. These sections of piping are Safety Class 2. The auxiliary feedwater system generally penetrates containment. Electrical power is provided by the essential ac distribution subsystem. A turbine-driven pump is supplied steam from taps on the main steam lines. The auxiliary feedwater system is actuated by its own control system. Actuation may be initiated by a loss of ac power, a decrease in feedwater header pressure, a safety injection signal, low level in the steam generators, or a manual signal. During startup, the auxiliary feedwater system is used to increase steam generator pressure by drawing suction from the condensate storage tank. This provides enough steam to start the main feedwater pumps.

The auxiliary feedwater system typically consists of two motor driven feedwater pumps and one steam turbine driven feedwater pump. The two motor driven pumps are sized for 50% capacity each; the turbine driven pump is sized for 100% capacity. Operation of only one pump is needed. All start on the actuation signal. The turbine driven pump will operate as long as steam is available from the main steam lines and dc control power is available. The pumps usually draw suction from two or more sources. Among the sources may be an auxiliary feedwater storage tank, the condensate storage tank, the service water system, and the condenser hot well. The pumps' discharge typically passes through a common header before entering the piping connected to the main feedwater lines. There is at least one auxiliary feedwater line for each main feedwater line. Flow control in each auxiliary feedwater line is established by a flow control valve.

The following systems interface with the auxiliary feedwater system:

- main steam system
- feedwater system
- essential ac distribution system
- dc power system (for pump control circuitry)
- engineered safety features actuation system
- condensate system
- condenser storage system

- demineralized water system (makeup to the auxiliary feedwater storage tank)
- instrument air system (for pneumatic valves)

A.3 Containment Systems - C

The containment systems grouping is made up of the containment (primary and secondary, as applicable) and those systems needed to prevent containment overpressure, to prevent excessive leakage from the containment to the environment, and to provide a habitable atmosphere inside containment. The systems and subsystems that make up the containment systems grouping are listed in Table A.3. The following systems are identified as relating to a BWR or a PWR or both.

Table A.3. Containment systems

	BWR	PWR
C01	Primary containment and penetrations	
C02	Reactor building	Reactor building/containment and penetrations
C03	Containment heat removal (a function of the residual heat removal system)	Containment cooling system
C03.A		Ice condenser system[a]
C04	Containment isolation system	Containment isolation (a function of the ESFAS and the various piping systems that penetrate containment)
C05	Containment purge system	Containment purge system
C06	Standby gas treatment system	
C07	Combustible gas control system	Combustible gas control system
C08	Containment ventilation system	Containment ventilation system
C09	Reactor building ventilation	
C10	Containment spray system	Containment spray system
C11		Penetration room ventilation system

[a]Unique to one series of Westinghouse units.

C01 Primary containment and penetrations (BWR)

The purpose of the primary containment is to contain the radioactive fluids and fission products that may result from postulated accidents inside the containment. Primary containment design in BWRs features two primary variations of the basic design. Primary containment consists of the dry well, a steel and concrete structure that houses the reactor vessel, the reactor recirculation loops, the control rod drive mechanisms and hydraulic subsystem, and other components; the wet well, or suppression pool; and the standpipes connecting the dry well and wet well. In older BWR designs, the dry well was an inverted "light bulb" structure and the wet well was a steel torus (doughnut) positioned below the dry well. Both were housed in the reactor building, or secondary containment. New BWR designs feature a primary containment much like that in PWRs. Dry well is a steel-reinforced concrete structure. The suppression pool is a toroidal chamber in the concrete structure below the dry well, which is a part of the reactor building. Rooms within both the old and new design structures are provided for the associated reactor systems and components.

Penetrations provide controlled access through the primary containment for piping, electrical cabling, other plant equipment, and personnel. Penetrations are normally sealed by double O-ring or double gasket seals. Personnel and equipment penetrations are air lock assemblies in which the inner and outer doors are interlocked to prevent their simultaneous opening. Numerous electrical and piping penetrations are installed. Two personnel hatches and an equipment hatch are typical.

C02 Reactor building (BWR)

The reactor building in BWRs is the secondary containment. During plant operations, such as refueling, when the primary containment must be breached, the reactor building functions as the primary containment. The reactor building houses the primary containment and various reactor support and safety system equipment. The reactor building is more typical of older BWRs. In the newer designs, there is only a containment structure; the primary-secondary distributions are not made.

C02 Reactor building/containment and penetrations (PWR)

As in BWRs, the PWR containments are designed to mitigate the consequences of postulated accidents inside containment by containing the radioactive fluids and fission products that may be produced by the accidents. The containment is usually a cylindrical or hemispherical carbon steel lined, reinforced concrete structure that houses the reactor vessel, reactor coolant system, control rod drive mechanisms, the emergency sump, various instrumentation, and other equipment. In some instances there is an air space, or annulus, between the carbon steel liner and the concrete structure. Numerous electrical and piping penetrations pass through the containment structure. Two personnel hatches, a fuel transfer tube, and one equipment hatch are typical. The electrical and piping penetrations are sealed by double O-ring or double gasket seals. All connecting cabling or piping passes through a penetrating room before attaching to the penetrations. The penetration room contains any gases that may leak through the penetrations during an accident situation. Equipment and personnel hatches are double door air locks interlocked to prevent the simultaneous opening of both doors.

C03 Containment heat removal (BWR)

In BWRs, containment heat removal is a function of the RHR/LPSIS, which is discussed in Sect. N08.

C03 Containment cooling system (PWR)

The containment cooling system is designed to help return containment pressure to a low value following a pipe break inside containment by cooling and condensing the steam generated by the break. The containment cooling system typically works together with the containment spray system and the water in the emergency sump to reduce containment pressure. The containment cooling system is totally housed inside the reactor building. It is a Safety Class 2 system. Electrical power is provided by the essential ac distribution subsystem and the actuation signal is generated by the ESFAS on the occurrence of a 4-psig pressure inside containment.

The containment cooling system may also be known as the reactor building cooling system or a subsystem of the containment atmosphere recirculation system.

A typical system configuration has two to three fans that discharge through cooling coils into a common header. Fan suction is taken from air registers near the roof of the reactor building. After passing through the cooling coils and common header, the fan discharge is directed by ductwork into the lower portion of the reactor building structure. The air then flows upward through the reactor building and around the reactor vessel gaining heat as it rises. As the intake registers, the cycle is started again.

The containment cooling system interfaces with the reactor vessel, the reactor building, and the essential ac distribution subsystem.

C03.A Ice condenser system (PWR). In a number of units of Westinghouse design, an ice condenser cooling system is built into the containment structure. The ice condenser is designed to help limit pressure inside the containment following an LOCA or a steam line break. As with the containment cooling system, pressure reduction is achieved by condensing the steam generated and released to the containment during an accident. A glycol system keeps the ice frozen. The ice condenser system is a Safety Class 3 system.

The ice is kept in "baskets" located in an annular space between the inner containment wall and the containment's steel liner. The ice baskets are also located at a level above the reactor vessel. Channels are provided to allow air flow through and between the ice baskets.

Following an accident, steam released to the containment causes the internal pressure to build. At a designated differential pressure level, doors in the inner containment wall will open allowing the steam to enter the bottom of the annular ice storage space. The steam is forced up through the ice baskets, condensing as it rises. At the top of the annular region, the cooled air is directed back into the inner containment where it further condenses some of the steam inside the containment. Natural circulation keeps the cycle operating until equilibrium is reached.

A glycol refrigeration system having two redundant pumps and heat exchangers keeps the ice in its frozen state.

The containment structure and the essential ac distribution subsystem are the only systems that interface with the ice condenser system.

C04 Containment isolation system (BWR)

The purpose of the containment isolation system is to prevent the release of significant amounts of radioactive materials from the reactor coolant pressure boundary following an accident or certain transients.

This is accomplished by monitoring selected plant parameters and, on the occurrence of an abnormal indication in one, initiating the closure of selected valves in various systems that penetrate containment. The conditions that will initiate the containment isolation system include:

- reactor trips (level 2 or 3)
- high radiation in the main steam line
- high temperature in the main steam line
- high flow in the main steam line
- low pressure in the main steam line
- high dry well pressure
- high radiation at containment ventilation exhaust

The containment isolation system is a class 1E system, and it is powered from separate buses in the dc power system.

Four instrument channels are used for each parameter monitored by the containment isolation system. The output of these channels is fed into two 2/4 voting logics. When an abnormal condition is sensed by one of the channels, it goes into a "tripped" state. The loss of a logic power supply will also cause its associated channel to trip. The coincidence to two or more tripped channels will initiate a containment isolation. The channel outputs are also fed directly to the main steam isolation valves where a trip signal from channels 1 or 4 will activate one solenoid actuation valve. Channels 2 or 3 will activate the other solenoid. Both solenoid valves must operate to trip the isolation valve.

Systems that interface with the containment isolation system are the dc power system and the various systems in which containment isolation valves are located.

C04 Containment isolation (PWR)

Containment isolation is one of the functions provided by the ESFAS in PWRs. This system is discussed in Sect. S02.

C05 Containment purge system (BWR and PWR)

The containment purge system is designed to provide a habitable working environment for personnel inside the containment structure by reducing airborne radioactivity and providing outside air during periods of extended occupancy. The containment purge system is a nonsafety system powered by the plant ac distribution system. The containment purge system works in essentially the same fashion in both BWRs and PWRs.

In some designs, the containment purge system may be a subsystem in a containment atmosphere recirculation and cleanup system.

The containment purge system is usually divided into two subsystems: supply and exhaust. The supply subsystem uses a fan to draw in outside air. The air is filtered before entering the fan and heated after discharge. Then the air is exhausted to the containment. The exhaust subsystem also uses two fans to draw air from the containment. Containment air passes through a multistage filtering system before it enters the fans. Fan exhaust discharges through the unit vent. The exhaust is usually monitored for radiation. On the occurrence of high radiation, selected dampers are closed and the air flow goes into a recirculation mode. Some designs have a filtration system that can be connected to most or all of the containment air systems. The dual action of removing and filtering containment air and bringing in fresh air tends to lower the airborne radioactivity levels inside containment.

Systems that typically interface with the containment purge system are the plant ac distribution system, the containment system, and in some instances a glycol system for heating the fresh air.

C06 Standby gas treatment system (BWR)

The standby gas treatment system limits the release of airborne radioactive materials to the environment under accident conditions. The system is activated by the following signals:

- high dry well pressure
- low reactor vessel water level
- high radiation in the fuel handling area ventilation system
- manual initiation

The standby gas treatment system is a Safety Class 3 system that is supplied electrical power by the essential ac distribution system.

The standby gas treatment system is a two-train system whose principal components are fans and multistage filters. Depending on design, the multistage filters may be on the suction side or the exhaust side of the fans. Flow control is established by a motor operated valve on the side of the fan opposite the filter. A normally isolated cross-tie is sometimes provided between the fans' suction piping. Fan suction is normally drawn from the containment ventilation system.

The essential ac distribution system and the containment ventilation system typically interface with the standby gas treatment system. The dc power system may also be used for fan control circuitry.

C07 Combustible gas control system (BWR and PWR)

The combustible gas control system monitors the hydrogen content inside containment after an LOCA and maintains the concentration at a safe level. The system in newer designs is usually divided into two subsystems: the hydrogen recombiner subsystem, which is a Safety Class 2 system, and its backup, the dilution air subsystem, which is nonsafety. Older designs typically have a Safety Class 3 dilution air system. The recombiner subsystem is supplied electrical power by the essential ac distribution. The dilution air system is supplied by the nonessential portion of the ac distribution system in newer designs. Both BWR and PWR systems operate in essentially the same manner.

In older designs this system may be referred to as a hydrogen purge system.

The recombiner subsystem design usually features two redundant recombiners with their associated fans, after-coolers, and ductwork. The combiners may be either catalytic or electric. The dilution air subsystem has two piping trains, a supply train and an exhaust train, which operate in the same manner as a containment purge system (Sect. C05). The exhaust subsystem may also discharge into the gaseous radwaste system. A hydrogen analyzer continuously monitors containment air and actuates an alarm when the hydrogen concentration reaches a preset level, usually less than 4%. The combustible gas control system is manually initiated.

The plant ac distribution system is the primary interface for the combustible gas control system. The gaseous radwaste system may also interface.

C08 Containment ventilation system (BWR and PWR)

The containment ventilation system circulates air within the containment to maintain the bulk air temperature suitable for personnel and equipment. The containment ventilation system is a nonsafety system and is supplied power from the plant ac distribution system. This system operates continuously under normal plant conditions but is shut off under accident conditions. Ventilation systems in both BWRs and PWRs work in the same fashion with ductwork branching to supply a flow of air to each of the rooms and compartments within the containment.

The containment ventilation system may also be called the reactor building ventilation system, or it may be a subsystem of the containment atmosphere recirculation system. The containment ventilation system may be divided into various subsystems also. In BWRs the subsystems may include the dry well ventilation system. Those PWR containments that have an annulus (the air space between the steel liner and the reinforced concrete) typically have an annulus ventilation system as a subsystem.

The containment ventilation system typically consists of two or three fans with their associated ductwork and dampers. The fans generally discharge into a common header from which the ductwork branches into the various rooms and compartments. Some systems have a filtering unit installed in the ductwork. The system is manually operated from the control room.

The plant ac distribution system is the only system that directly interfaces with the containment ventilation system.

C09 Reactor building ventilation system (BWR)

Older BWR designs may have a reactor building ventilation system that is totally separate from the containment (or dry well) ventilation system. The purpose of this system is to maintain the bulk air temperature in the reactor building (or secondary containment) suitable for personnel or equipment. The system is operated in the same fashion and is designed essentially the same as the containment ventilation system. It is also a nonsafety system.

C10 Containment spray system (PWR and BWR)

The containment spray system provides a water spray to the containment following an LOCA or steam line break to limit containment pressure and to minimize the release of radioactive iodine and particulates to the environment. The containment spray system is a Safety Class 2 system found in all PWR designs. Newer BWR designs have a containment spray that is a subsystem of the RHR/LPSIS (Sect. N08). Electrical power for the containment spray pumps is provided by separate buses of the essential ac distribution subsystem. The containment spray system is normally housed in the auxiliary building, with the exception of the spray headers and nozzles, which are in the containment.

The containment spray system may also be called the reactor building spray system.

There are a number of possible configurations for the containment spray system. It is normally a two- or three-train system that either draws suction directly from the refueling water storage tank and the emergency sump or indirectly from these sources through the RHR/LPSIS or the safety injection system. Most systems have chemical addition tanks from which sodium hydroxide (NaOH) or hydrazine is added to the spray water. These chemicals have the dual effect of improving iodine removal from the containment atmosphere and neutralizing much of the corrosive effects of the boric acid in the refueling storage tank water. Each of the two or three piping trains has a containment spray pump that discharges into a spray header or headers inside the containment. In some designs, the pumps discharge through containment spray heat exchangers. The containment spray system may be manually actuated or it may be automatically actuated by the ESFAS. The occurrence of high containment pressure (about 10 psig) causes the ESFAS to actuate containment spray.

The following systems typically interface with the containment spray system:

- essential ac distribution subsystem
- dc power system
- refueling water (condensate storage system for BWRs)
- emergency sump
- residual heat removal/low-pressure safety injection system
- reactor building cooling water (when heat exchangers are included)
- chemical and volume control (for the addition of sodium hydroxide or hydrazine)

C11 Penetration room ventilation system (PWR)

Those reactor buildings that do not have a steel liner usually have a penetration room through which all connections to electrical or piping penetrations must pass. The penetration room sits outside of and adjacent to the reactor building. After an accident, some leakage will occur around the electrical and piping penetrations. The penetration room ventilation system is designed to prevent this leakage from escaping to the environment. It is a Safety Class 3 system and is powered by the essential ac distribution subsystem.

The penetration room ventilation system is a two-train system. The two redundant fans draw suction from the penetration room and discharge through a multistage filter unit into the unit vent. On the occurrence of high radiation at the vent, the system goes into a recirculation mode or the flow is routed to the gaseous radwaste system. The suction from the penetration room keeps it at a slightly lower pressure than the outside so that any leakage is into the room rather than from the room. The penetration room ventilation system is manually actuated from the control room.

Systems that may interface with the penetration room ventilation system are the essential ac distribution subsystem and, possibly, the gaseous radwaste system.

A.4 Electrical System - E

The electrical system grouping is made up of those plant systems that supply electrical power to either the utility grid or other plant systems or that are purely electrical in nature. These systems are essentially the same in both BWRs and PWRs. The systems are listed in Table A.4.

E01 Main power system

The main power system receives electrical power from the unit generator and directs it to the unit step-up transformer where the electrical voltage is raised to a level compatible with the utility transmission system. The main power system is a nonsafety system.

There are two basic arrangements of the main power system. In the first, an isolated phase bus is used to connect the unit generator to the unit step-up transformer. The second arrangement has a split isolated phase bus and two "half size" unit step-up transformers. In both cases, taps are provided on the isolated phase bus for connections to a unit auxiliary transformer. Some newer designs also incorporate generator breakers that are located between the unit generator and the unit step-up transformers. The isolated phase bus normally is provided with a cooling system. Unit step-up transformers may be either a three-phase transformer or three single-phase transformers.

Protective relaying for the main power system is designed to protect the system's equipment from damage should a fault condition occur. If a fault occurs, the relaying system will initiate the opening of the appropriate circuit breakers to isolate the faulted equipment. Relays that

Table A.4. Electrical systems[a]

E01	Main power
E01.A	Protective relaying and controls
E02	Plant ac distribution system
E02.A	Essential power system
E02.B	Nonessential power system
E02.C	HPCS power system (BWRs)
E02.D	Protective relaying and controls
E03	Instrumentation and control power systems
E03.A	Dc power system
	Vital dc power subsystem
	Plant dc power subsystem
E03.B	Instrument ac power system
	Vital instrument ac power subsystem
	Plant instrument ac power subsystem
E04	Emergency power system
E04.A	Diesel generator fuel oil subsystem
E04.B	Diesel generator cooling water subsystem
E04.C	Diesel generator starting air subsystem
E04.D	Diesel generator lubrication oil subsystem
E05	Plant lighting system
E05.A	Essential lighting
E05.B	Nonessential lighting
E06	Plant computer
E07	Switchyard
E07.A	Dc control power system
E07.B	Protective relaying

[a]The electrical systems for both BWRs and PWRs are essentially identical.

provide instantaneous overcurrent or time overcurrent protection and differential zone protection are typical.

Systems that interface with the main power system are

- turbine generator
- switchyard
- plant ac distribution (through the unit auxiliary transformer)

E02 Plant ac distribution system

The plant ac distribution system is the normal source of electrical power for the plant under normal operating conditions, and it is the preferred choice under accident conditions. The system is divided into two subsystems that are typically kept separated. The essential ac distribution subsystem provides electrical power to those systems that are required to bring the plant to a safe shutdown following an accident and that are needed to mitigate the effects of an accident. This subsystem is designated as Class 1E. The essential ac distribution subsystem is further divided into two or three distinct trains, depending on design. These trains are kept electrically separate and separated physically as much as possible. Redundant equipment in safety related piping systems is supplied power from separate electrical power trains. The nonessential ac distribution subsystem supplies electrical power to the remainder of the plant's systems. It is designated as a nonsafety system. [In some newer BWR designs, a high-pressure core spray (HPCS) power system is provided. This system is dedicated solely to powering the equipment needed for the HPCS system. The HPCS power system is designated Class 1E and is kept separate from all others.]

The plant ac distribution system is a hierarchical arrangement of switchgear at various voltage levels. Both essential and nonessential subsystems are constructed in this manner. There are usually three or four different voltage levels within the plant ac distribution system. The two highest levels are fed from the unit generator through a unit auxiliary transformer when the unit is operating. The unit auxiliary transformer is a tertiary transformer with the two secondary windings having different voltage levels. When the unit is shut down, these two voltage levels are fed from the switchyard through a startup transformer or, in the case of a generator breaker design, from the switchyard through the unit step-up transformers and the unit auxiliary transformers. The startup transformers are also tertiary. The highest of the voltage levels is either 13,800 or 6,900 V. The reactor coolant or reactor recirculation pump motors are fed from here. The next voltage is the 4160-V level. The essential and nonessential distinctions are first made here. Most large pump motors in the plant are supplied from these switchgear. Also, the lower voltage switchgear are supplied from the 4160-V switchgear. In some design, there are essential and nonessential switchgear at this level. Other designs have one set of switchgear with both essential and nonessential loads. In the event of an accident, all loads are shed from the switchgear, but only the essential loads are added back. When normal power is not available, the emergency power system is generally connected at the 4160-V level.

The next voltage level is either 600 or 480 V. Switchgear at this level are fed from 4160-V switchgear through auxiliary transformers. This voltage level is used to supply power to medium to small pump and fan motors and large valve operator motors. Some plants also have switchgear at the 208- or 120-V level. These switchgear are supplied from the 600- or 480-V level through auxiliary transformers. Small valve operator and pump or fan motors are supplied power from these switchgear. Some instrumentation may also be included on these switchgear. The separation between the nonessential and the various trains of the essential subsystems are maintained at these voltage levels.

Switchgear at all of the various voltage levels typically have two incoming breakers through which they are supplied power. Only one of the breakers is usually closed at any one time. Undervoltage relays are supplied to coordinate the operation of the normally closed and normally open incoming breakers. At the 13,800/6,900-V and the 4,160-V levels, synchronizing relays are provided to prevent the paralleling of two voltage sources unless they are synchronized.

Each of the incoming and feeder breakers in the plant ac distribution system have protection from being damaged by electrical faults in other equipment. At the 13,800/6,900- and 4,160-V levels, instantaneous overcurrent and time overcurrent relays provide this protection. At the lower voltage levels, instantaneous magnetic and thermal magnetic coils that actuate the circuit breakers are used. Thermal overloads are also used.

The following systems interface with the plant ac distribution system:

- main power
- switchyard
- emergency power
- dc power (load off the 600/480-V switchgear plus control power for ac circuit breakers)
- plant safety related systems (essential ac distribution subsystem)
- plant nonsafety related subsystems (nonessential ac distribution subsystem)

E03 Instrumentation and control power systems

The instrumentation and control power systems provide either dc or ac power to the instruments and control circuits for the plant. The dc power system is designed to supply either 125 or 250 V dc to its designated loads. The distinction between safety and nonsafety power supplies is kept in these systems. Each system in discussed in the following sections.

E03.A Dc power system. The dc power system supplies an uninterruptible source of dc electrical power to some instrumentation and numerous control circuits in the plant. Those dc loads that are safety related are supplied from a vital dc power subsystem. Other dc loads are supplied from the plant dc power system. (BWRs having an HPCS power system also have a dedicated dc bus for the HPCS system.)

The vital dc power subsystem is composed of four separate panelboards that are supplied electrical power by a battery charger connected in parallel with a 125-V battery. The battery chargers are connected to the 600/480-V switchgear in the essential ac distribution subsystem (Sect. E02). In the event of a loss of ac power, the batteries are sized to carry their associated loads for a number of hours. Control circuits for ac circuit breakers and the inverters for the instrument ac power system (Sect. E03.B) are the primary loads on the dc panelboards. The vital dc power subsystem is a Class 1E system.

The plant dc power subsystem is constructed in much the same manner as the vital dc power subsystem except that it sometimes is arranged to supply 250- as well as 125-V dc power. The loads on the plant dc power subsystem are essentially the same as those on the vital subsystem except that they are nonsafety loads, and the plant subsystem is a nonsafety system.

The dc power system interfaces with the following systems:

- plant ac distribution
- instrument ac power

- most of the safety related plant systems (vital subsystem for control circuitry)
- most of the nonsafety plant systems (plant subsystem for control circuitry)

E03.B Instrument ac power system. The instrument ac power system provides an interruptible source of 120 V, single-phase ac power to the plant's instrumentation and the plant computer. Safety related instrumentation is supplied from the vital instrument ac power subsystem. The plant computer and nonsafety instrumentation is supplied by the plant instrument ac power subsystem. (BWRs having an HPCS power system have a dedicated 120-V ac bus and HCPS instrumentation.)

The vital instrument ac power subsystem is made up of four separate panelboards. Each panelboard is supplied power from an inverter that is connected to a corresponding vital dc power panelboard or from a nonsafety, regulated power supply through a static selector switch. The vital instrument ac power subsystem is a Class 1E system.

The plant instrument ac power subsystem is supplied by inverters or by a regulated, 120-V, single-phase supply. The plant computer must be connected to the regulated supplies. Other nonsafety instrumentation, including the turbine generator electrohydraulic control system, is supplied by an appropriate number of panelboards. The exact number depends upon the particular design. The plant instrument ac power subsystem is a nonsafety system.

Systems interfacing with the instrument ac power system are

- dc power
- plant computer
- reactor protection
- engineered safety features actuation (PWRs)
- turbine generator electrohydraulic control
- other safety related instrumentation (vital subsystem)
- other nonsafety related instrumentation (plant subsystem)

E04 Emergency power system

The emergency power system is designed to supply an adequate amount of electrical power to safely shut down the plant when the normal ac power supply has been lost. The system has two distinct portions: an electrical system and a supporting group of mechanical systems. The electrical portion is designated Class 1E and the mechanical systems are classified as Safety Class 3. (BWRs that have an HPCS power system are provided with a dedicated emergency power system for the HPCS system loads.)

The electrical portion of the emergency power system typically consists of a diesel powered generator, a circuit breaker to connect the generator to the 4160-V switchgear, under-voltage relays on the 4160-V switchgear to actuate the emergency power system, and the load shedding and sequencing circuitry. There are typically two emergency power trains. The supporting mechanical subsystems are discussed in the following sections.

The plant ac distribution system and the dc power system are the only plant systems that interface with the electrical portion of the emergency power system.

E04.A Diesel generator fuel oil subsystem. The diesel generator fuel oil subsystem supplies the fuel to run the generators in the emergency power system. There is a fuel system for each diesel. Each fuel system is sized to hold a 7-d supply of fuel oil when the diesel is operating under maximum loading conditions. The diesel generator fuel oil subsystem is a Safety Class 3 system.

Each fuel system has two storage tanks: a large storage tank and a smaller day tank. Two redundant pumps are supplied to transfer fuel oil from the storage tank to the day tank. The pumps can be manually actuated or the level in the day tank can actuate them. Depending upon the design, the day tank may supply fuel to the diesel via a fuel pump, or gravity feed may be used.

The diesel generator fuel oil subsystem interfaces with the essential ac distribution subsystem and the dc power system.

E04.B Diesel generator cooling water subsystem. The diesel generator cooling water subsystem provides cooling for the diesel generators while they are running. The system may also have a loop for warming the diesel jacket to prevent thermal shock when the diesel starts. Each diesel has a cooling system. The diesel generator cooling water system is a Safety Class 3 system.

Typical system configuration features redundant pumps that discharge into the diesel jacket. Pump suction draws from the diesel jacket through a heat exchanger and a lubrication oil cooler. A surge tank is provided to regulate water inventory. A recirculation loop is also provided with a heater and its own pump for jacket warming. The temperature of the water leaving the diesel jacket is monitored. If an excessive temperature is reached, a diesel shut-off circuit is actuated.

Systems that interface with the diesel generator cooling water subsystem are

- essential ac distribution subsystem
- dc power (for control circuitry)
- station service water (heat exchanger cooling)
- diesel generator lubrication oil subsystem

E04.C Diesel generator starting air subsystem. The diesel generator starting air subsystem supplies compressed air to starting motors on the diesel generators to provide for a rapid start of the diesel. Each diesel has a starting air system. The diesel generator starting air system is a Safety Class 3 system.

This system is typically a two-train system. Each train has an air compressor that exhausts into a grouping of receivers, or compressed air storage tanks. Compressor suction is taken from the atmosphere and filtered before entering the compressor. A cross tie is typically provided between the two trains.

The essential ac distribution subsystem and the dc power system interface with the diesel generator starting air subsystem.

E04.D Diesel generator lubrication oil subsystem. The diesel generator lubrication oil subsystem supplies oil to lubricate the moving parts of the diesel generator. Each diesel has a lube oil system. The system is designated as Safety Class 3.

The system features a loop with a motor driven pump to increase oil pressure enough to permit starting the diesel. This loop is sometimes filtered and heated. Once the diesel has started, it provides lube oil pumps that circulate oil through the engine. The recirculation loop usually has a filter and a lube oil cooler.

The systems that interface with the diesel generator lubrication oil system are the essential ac distribution subsystem, the dc power system, and the diesel generator cooling water subsystem.

E05 Plant lighting system

The plant lighting system provides adequate illumination in the plant for personnel to perform their required tasks. The system is typically divided into essential and nonessential subsystems. The essential subsystem is Class 1E and provides lighting for those areas in the plant in which personnel activity will be required after an accident. The nonessential subsystem is a nonsafety system and is disabled following an accident. Outdoor lighting may also be provided.

The plant lighting panelboards are usually loads on either the vital instrument ac power subsystem or the 120-V level of the nonessential ac distribution system depending on the safety designation of the lighting system.

E06 Plant computer

The plant computer may be used for a number of functions. The exact list of functions depends upon each design. Typical among the functions are

- monitoring and printout of plant parameters
- performance of calculations based on plant parameters
- various information displays through either a printer or a cathode ray tube (CRT)
- in some designs, control of selected balance-of-plant systems

The plant computer is a nonsafety component. Its power is supplied through a regulated, 120-V, single-phase power supply.

E07 Switchyard

The switchyard is the line between the unit and the utility's transmission grid. A number of transmission lines typically distribute the power produced by the unit to transmission substations that further distribute the power. The switchyard is usually connected to the unit through the unit step-up transformer. It may also provide power through a startup transformer when the unit is shut down. The switchyard is not safety related, and in many cases may not be considered a part of the unit.

Many switchyard designs feature a two-bus arrangement with a "breaker-and-a-half" scheme connecting them. This arrangement is very flexible and prevents a single active failure from isolating a line. Differential zone protective relaying as well as instantaneous and time overcurrent relaying is used to protect switchyard equipment. The switchyard is also provided its own dc power system for circuit breaker control circuitry.

The switchyard interfaces with the main power system and sometimes the plant ac distribution system.

A.5 Power Conversion System — P

The power conversion system grouping is made up of the systems and components that are used to transform, or to support the transformation of, heat energy produced by the reactor core into electrical energy. The systems and subsystems comprising the power conversion system grouping are listed in Table A.5. The list of systems is essentially the same for both BWRs and PWRs. The major difference is that under normal operating conditions, the main steam, turbine generator, condensate, and feedwater systems in BWRs use radioactive coolant in either a gaseous or liquid form. The steam-water cycle in PWRs does not use irradiated water.

Table A.5. Power conversion systems[a]

P01	Main steam
P02	Turbine generator
P02.A	Electrohydraulic control subsystem
P02.B	Turbine gland seal subsystem
P02.C	Turbine lubrication subsystem
P02.D	Stator (hydrogen) cooling subsystem
P02.E	Hydrogen seal oil subsystem
P03	Turbine bypass
P04	Condenser and condensate
P04.A	Condenser evacuation
P04.B	Condensate cleanup/polishing
P04.C	Condensate heater drain subsystem
P05	Feedwater
P05.A	Feedwater heater drain
P06	Circulating water
P07	Steam generator blowdown (PWR)
P08	Auxiliary steam

[a]The power conversion systems for both BWRs and PWRs are essentially the same. The major difference is environmental — the BWR system has slightly radioactive steam passing through the main flow path even under normal conditions.

P01 Main steam system

The main steam system is used to transport the steam generated in the reactor vessel in BWRs or the steam generators in PWRs to the turbine generator. The interface with the turbine generator is at the turbine stop valves. Depending upon the design, from two to four steam lines may be used. The main steam system is a nonsafety system that penetrates containment, although portions of the system are safety related (up to the main steam isolation valves).

After leaving containment, several pressure relief valves are connected in parallel to the main steam lines. Groups of these are normally set to open at three succeedingly higher pressure levels. Next in the steam lines are the main steam isolation valves (MSIVs). These valves are usually pneumatically actuated and require two coincident signals from the engineered safety features actuation system to actuate a closure. The MSIVs are Safety Class 3 components. Between the MSIVs and the turbine stop valves are a number of steam taps. There is one tap in each leg for the turbine-bypass-to-condenser lines and the atmospheric dump valves. Taps are also provided to supply steam for the main feedwater pump turbines, the auxiliary or emergency feedwater pump turbines, and the moisture separator reheaters. A connecting line for the auxiliary steam system is also provided.

The main steam system interfaces with the following systems:

- dc power (for MSIV control circuitry)
- reactor vessel (BWRs)
- instrument air
- reactor coolant (through steam generators in PWRs)
- turbine generator
- turbine bypass
- feedwater
- auxiliary feedwater
- auxiliary steam

P02 Turbine generator system

The turbine generator is the device used to transform mechanical energy in the form of moving steam to electrical energy. This is done by using steam to turn the turbine. Connected to the turbine shaft is an electrical generator in which a loop is rotated through a magnetic field, producing an electrical current in the loop. The turbine generator is a nonsafety system that is supported by a number of subsystems. (Note that in BWRs the turbine building is a part of secondary containment.)

The turbine is normally a four-stage device with the stages mounted axially along the turbine shaft. The first stage is a high-pressure turbine; the last three stages are low-pressure turbines. The high-pressure turbine is driven by the main steam system. From the exhaust of the high-pressure turbine, the steam is directed through the moisture separator reheaters and then to the low-pressure turbines. The low-pressure turbines exhaust into the condenser. Taps are provided on the high- and low-pressure turbines to supply steam to the condensate and feedwater heaters.

Generator output is connected to the main step-up transformer through an isolated-phase bus.

The turbine is controlled by an electrohydraulic control (EHC) subsystem. The turbine is controlled by regulating the flow of steam through it. The EHC subsystem accomplishes this by throttling the turbine control valves and intercept valves. To trip the turbine, the turbine stop valves are closed, shutting off the steam supply. Generator voltage is controlled by an exciter system.

Systems that interface with the turbine generator system are

- main steam
- electrohydraulic control subsystem
- turbine gland seal subsystem
- turbine lubrication subsystem
- stator (hydrogen) cooling subsystem
- hydrogen seal oil subsystem
- main power
- feedwater and condensate (through steam supply to the water heaters)
- condenser

hydraulic

P02.A <u>Electrohydraulic control subsystem</u>. The EHC subsystem controls the turbine by regulating the steam flow through the turbine. The EHC subsystem is capable of remote, manual, or automatic starting of the turbine, loading the turbine at a preset rate, and holding load and speed at a preset level. The EHC subsystem typically has three distinct sections: a speed control unit, a load control unit, and a valve positioning unit. Steam flow through the turbine is accomplished by throttling the pneumatically operated turbine control valves and intercept valves. Trips are accomplished by closing the turbine stop valves. Turbines designed by Allis-Chalmers provide a backup mechanical-hydraulic control system.

P02.B <u>Turbine gland seal subsystem</u>. The turbine gland seal subsystem seals the turbine shaft between both turbine casings and between the exhaust hoods and the atmosphere. This prevents air from leaking into the turbine and steam from leaking into the turbine building. The shaft seals are labyrinth-type or pressure-packing glands through which steam is passed outward away from the turbine. The seals exhaust into a gland steam condenser.

P02.C <u>Turbine lubricating subsystem</u>. The turbine lubricating subsystem supplies oil to lubricate the moving parts of the turbine generator. The subsystem is divided into two parts: a lubricating oil section and an oil conditioning system. The lubricating oil section consists of bearing oil pumps, an oil reservoir, and oil coolers in series. The oil conditioning section is made up of a clean oil storage tank, a used oil storage tank, a filtering unit, and an oil transfer pump. Water from the turbine building cooling water system is used to cool the oil in the heat exchangers.

P02.D <u>Stator (hydrogen) cooling subsystem</u>. The stator (hydrogen) cooling subsystem is used to remove heat from the coils of the generator stator. Hydrogen is used to fill the stator housing and keeps the stator windings moisture free. When replacement of the hydrogen is necessary, it is displaced by carbon dioxide (CO_2), and then a fresh supply of hydrogen is added. This system consists of pressure regulators and controls for the hydrogen gas and a carbon dioxide circuit for purging operations.

P02.E Hydrogen seal oil subsystem. The hydrogen seal oil subsystem is used to prevent hydrogen leakage through the generator shaft seals. This system circulates oil through the shaft seals, entraining any hydrogen that should leak into them. The system consists of pumps, storage tanks, and the controls necessary to degasify the oil before returning it to the shaft seals.

P03 Turbine bypass system

The turbine bypass system allows the NSSS to follow an ~50% step load reduction to the turbine generator without causing a reactor trip or lifting the main steam pressure relief valves. The turbine bypass system is a nonsafety system.

The turbine bypass system consists of a pneumatically or electrically operated turbine bypass valve and controls, one or two isolation valves and controls, and associated piping for each main steam line. On the occurrence of a large reduction in electrical load, the turbine bypass valves open, relieving main steam directly to the condenser. Some designs use a group of small turbine bypass valves in parallel rather than a single large valve for each steam line. This helps prevent an uncontrollable cooldown if a valve sticks open. The turbine bypass valves are opened automatically by the turbine EHC subsystem following a large load reduction. During a normal shutdown of the reactor, the turbine bypass valves are opened manually to release steam generated by decay heat in the reactor. As cooldown continues, the turbine bypass valve is throttled closed, eventually transferring the decay heat removed to the residual heat removal/low-pressure safety injection system.

The turbine bypass system interfaces with the following systems:

- main steam
- condenser
- EHC subsystem
- instrument air
- plant ac distribution (for motor operated isolation valves)

P04 Condenser and condensate system

The function of the condenser is to condense steam from the low-pressure turbine exhausts, the feedwater pump turbines, and the turbine bypass system. The condensate system takes condensed steam from the condenser and heater drains and delivers it to the feedwater system. Along the way, the condensate is purified and heated. The condenser and condensate system is a nonsafety system.

The condenser is a triple-shell, single-pass water box in which circulating cooling water is used to condense steam from the turbine. As the steam condenses into a liquid, it is collected in the hotwell sections of the condenser. In the event one train of the circulating cooling water is lost, a condenser circulating water pump is activated to provide water to both sides of the condenser shell.

The condensate system draws the condensed steam from the hotwells through two or three condensate pumps. The pumps discharge through the steam jet air ejectors and gland steam condensers before passing to the condensate cleanup/polishing system. This system may be bypassed. The exact order of these components and systems varies with design. From the condensate cleanup/polishing system, the system water pressure is raised by two or three condensate booster pumps. The discharge from these pumps is passed through a string of condensate heaters. From the condensate heaters, the water enters the feedwater system. A condensate tank provides makeup water to the condenser to maintain hotwell level. A number of flow control and isolation valves are scattered throughout the system. Their operation generally requires instrument air or electrical power. Condensate makeup is provided by the demineralized water system.

The condenser and condensate system interfaces with the following systems:

- turbine generator
- turbine bypass
- feedwater
- turbine gland seal subsystem
- condenser evacuation subsystem
- condensate cleanup/polishing system
- auxiliary feedwater
- auxiliary steam
- demineralized water
- instrument air
- dc power
- plant ac distribution

P04.A Condenser evacuation system. The condenser evacuation system is designed to provide the initial vacuum in the condenser shells during startup, to maintain the vacuum during condenser operation, and to dispose of noncondensable gases collected from the condenser. The loss of condenser vacuum allows buildup of noncondensable gases that inhibit the heat transfer capability of the condenser. The condenser evacuation system is a nonsafety system.

Steam jet air ejectors are used to remove noncondensable gases from the condenser and to maintain the vacuum in the condenser shells. This is done by passing a jet of steam through the condenser shells. The passage of the steam creates a vacuum that draws the noncondensable gases into the jet stream. The air ejectors exhaust into the gaseous radwaste system. The steam jet air ejectors function by using auxiliary steam. Motor driven air removal pumps are provided for initial condenser shell side air removal.

The condenser evacuation system may also be known as the condenser vacuum system or the vacuum system.

Systems interfacing with the condenser evacuation system are

- condensate
- auxiliary steam
- gaseous radwaste
- plant ac distribution

P04.B Condensate cleanup/polishing system. The condensate cleanup/polishing system removes impurities from the condensate water that result from condenser tube leakage. This results in a high purity effluent capable of meeting feedwater and steam generator chemistry standards. This is a nonsafety system.

The condensate cleanup/polishing system consists of several mixed-bed-type demineralizers. The exact number depends upon design; however, sufficient capacity is usually provided for operation at full condensate flow while one of the demineralizers is being regenerated. Differential pressure around the demineralizers is monitored to detect impaired flow. At a preset pressure level, a bypass valve is opened to prevent demineralizer damage.

The condenser cleanup/polishing system interfaces with the following systems:

- condensate
- plant ac distribution

P04.C Condensate heater drain subsystem. The condensate heater drain subsystem collects the steam condenser. The condensate heater drain system is a nonsafety system.

The steam passing through the shell side of the condensate heaters is collected in a common header. Depending upon design, either gravity flow or motor operated pumps return the water to the condenser.

The condenser, the main steam system (extraction steam), the condensate system, and the plant ac distribution system interface with the condensate heater drain system.

P05 Feedwater system

The feedwater system takes condensate from the condensate system, heats it, raises its pressure, and delivers it to the steam generators in PWRs or the reactor vessel in BWRs to be boiled off as steam. The feedwater system is a nonsafety system that penetrates containment.

In general, two steam-turbine-driven main feedwater pumps draw suction from the condensate system. In some plants (particularly the Westinghouse design), electric powered main feedwater pumps are provided. Both pumps are needed for full operation. The pumps discharge through a string of feedwater heaters into a common header. From the header, one feedwater line goes to each steam generator in PWRs. The BWRs generally have two lines that go to the reactor vessel. These lines each contain a feedwater regulation valve that is used to throttle feedwater flow to match unit demand. A parallel loop bypasses the feedwater regulation valves and contains the startup valves. These valves are used to throttle feedwater flow during startup. Both the feedwater regulation valves and the startup valves are pneumatically operated. Control is provided by the feedwater control system except in the case of B&W designs. Here control is provided by the integrated control system. In PWRs, the connections to the auxiliary feedwater system are made in this section of piping. Containment isolation is provided by the flow control valves and check valves.

The systems that interface with the feedwater system are

- condensate
- main steam (including extraction steam)
- reactor core and reactor vessel (BWRs)
- reactor coolant (PWRs)
- auxiliary feedwater (PWRs)
- integrated control (Babcock & Wilcox units)
- feedwater heater drain subsystem
- instrument air
- plant ac distribution

P05.A Feedwater heater drain system. The feedwater heater drain system collects the steam condensed in the feedwater heaters and returns it to the main condenser. In some designs, this system is combined with the condensate heater drain system to form a single heater drain system. All of these are nonsafety systems.

The condensed steam is collected in a common header. It may then be either gravity fed or pumped back to the main condenser.

The condenser, the main steam system (extraction steam), the feedwater system, and the plant ac distribution system interface with the feedwater heater drain system.

P06 Circulating water system

The circulating water system provides cooling water to the condenser to condense the steam exhaust from the turbine. The circulating water system is the ultimate heat sink for the plant. Two designs are prevalent: an open system and a closed system. The open system draws raw water from outside the plant, passes it through the condenser, and discharges it back into the water source at another location. The closed system recirculates cooling water through the condenser to cooling towers and back. The cooling towers transfer heat to the surounding air. Makeup water is provided by a raw water source. The circulating water system is a nonsafety system.

The circulating water system may also be called the condenser circulating water system.

The system consists of several pumps. Again, the exact number depends upon the particular design, flow control, and isolation valves, and for open systems, a system of moving screens on the intake to strain out debris in the raw water. Other water systems may take suction from the circulating water system, depending on design.

The circulating water system interfaces with the following:

- condensate
- environment
- plant ac distribution system
- dc power system (for isolation valves)

P07 Steam generator blowdown system (PWR)

In CE and Westinghouse designs, the steam generator blowdown system is used in addition to the chemical feed section of the feedwater system and the condensate cleanup/polishing system to control the chemical composition and solids concentration of the feedwater in the steam generators. In B&W plants, steam generator design and water chemistry do not require this system. This is a nonsafety system.

Each steam generator is provided with a blowdown line containing a flow control valve and a containment isolation valve. The lines join in a common header. From the header, blowdown steam enters a system of blowdown concentrator reboilers. Main steam is used to evaporate the blowdown. Main steam condensate flows from the reboilers into reboiler receivers (tanks) where it is directed back to first stage reheaters. Bottom liquid represents the difference between the blowdown feed rate and the evaporation rate. This flows into the base of a separator where it is drawn off. Bottom liquid is normally discharged from the plant; however, if radioactivity is detected in the liquid, it is diverted to the liquid radwaste system. Any residual liquids in the separators are periodically blown down to a sludge pot and sent to the solid radwaste system.

Systems interfacing with the steam generator blowdown system are

- main steam
- liquid radwaste
- solid radwaste
- instrument air
- dc power (for valve control circuitry)

P08 Auxiliary steam system

The function of the auxiliary steam system is to supply heating steam through the plant and to recover the condensed steam from the equipment served. At multi-unit plants, this system is often shared by the units. It is a nonsafety system.

Steam for the system may be obtained from taps on the main steam lines when the unit is operating or from a fossil-fired auxiliary boiler when the unit is shut down. Steam flows from these sources through a common header to the various pieces of equipment served in the plant. Condensate is collected and pumped back to the boilers. Surge tanks are provided to hold up extra fluid.

The auxiliary steam system interfaces with the following:

- boron recovery subsystem of the CVCS
- domestic hot water tank
- various evaporators in waste systems
- demineralized water storage tank heaters
- condensate storage tank heaters
- steam jet air ejectors
- turbine gland seal subsystem

A.6 Process Auxiliary Systems — W

The process auxiliary systems grouping is made up of those systems and subsystems that support the plant systems directly involved in the process of safely producing electrical power. The systems and subsystems of the process auxiliary systems groupings are essentially the same for both BWRs and PWRs. They are listed in Table A.6.

W01 Radioactive waste system

The radioactive waste system is used to collect radioactive wastes from the plant and to reduce the concentration of the radionuclides to as low a level as is practicable so that the wastes can be safely released

Table A.6. Process auxiliary systems[a]

W01	Radioactive waste
W01.A	Gaseous radwaste
W01.A1	Offgas subsystem (BWRs)
W01.B	Liquid radwaste
W01.C	Solid radwaste
W02	Radiation monitoring
W02.A	Plant area radiation monitors
W02.B	Environmental radiation monitors
W02.C	Process radiation monitors
W03	Cooling water
W03.A	Reactor building cooling water
W03.B	Turbine building cooling water
W04	Service water
W04.A	Demineralized makeup water
W04.B	Station service water
	Essential service water
	Nonessential service water
W04.C	Chilled water
W05	Refueling
W06	Spent fuel storage
W06.A	Fuel pool cooling and cleanup
W07	Compressed air
W07.A	Service air
W07.B	Instrument air
W08	Process sampling
W09	Plant gas
W09.A	Nitrogen
W09.B	Hydrogen

[a]The process auxiliary systems are essentially the same for both BWRs and PWRs.

or stored. There are three subsystems to handle each of the three states in which waste is produced: gaseous, liquid, and solid. Each of these subsystems are discussed below.

W01.A Gaseous radwaste system. The gaseous radwaste system is usually made up of two portions: the process gas (hydrogenated) and low-activity process vent (aerated) streams. The process gas (hydrogenated portion of the system is designed to remove fission product gases from the reactor coolant letdown stream and from the liquids collected in the reactor coolant drain tank in PWRs. In BWRs, fission product gases are removed from the reactor coolant through the offgas subsystem discussed in Sect. W01.A1. The process gas portion of the gaseous radwaste system is a Safety Class 3 system. Radioactive gases collected from reactor building vents, from the steam jet air ejectors (in PWRs), and from the process gas adsorption bed gas drain are collected by the low-activity process vent portion of the system. These gases are filtered, monitored, and, if the concentration of radioactive nuclides is low enough, discharged to the environment. The low-activity process vent portion of the gaseous radwaste system is a nonsafety system. Both portions may penetrate containment.

Other names used to identify the gaseous radwaste system are

- radioactive gaseous waste
- gaseous waste disposal
- gaseous radioactive waste

There are numerous designs of this system. Almost all are unique in some manner; however, a large number of them involve gas compressors along with a filtration and monitoring section. Those designs that remove dissolved gases from liquids use a degasifier for this purpose. Some designs feature hydrogen control using a recombiner. Most all of the major components have installed spares. After collecting the waste gases, most systems run the gases through a system of filters, dryers, and air compressors. When hydrogen is a concern, a recombiner is normally used early in the process. After processing, the gases are monitored, and if their level of radioactivity is low enough, they are released to the atmosphere. If the level is too high, the gases may be either recycled or sent to a holdup storage tank until the level of radioactivity falls to a predetermined level. The specific arrangement of the equipment in the gaseous radwaste system is highly dependent upon the architect-engineer and the utility involved with the design of the plant.

Systems interfacing with the gaseous radwaste system are

- plant ac distribution
- chemical and volume control (PWRs)
- combustible gas control
- boron recovery subsystem (of CVCS)
- various vents and drains
- containment isolation (BWRs) or engineered safety features actuation (PWRs)
- condenser evacuation subsystem
- liquid waste subsystem

- refueling
- instrument air
- offgas subsystem (BWRs)

W01.A1 Offgas subsystem. The offgas subsystem is used in BWRs to collect the fission product gases from the reactor coolant. These gases are separated from the coolant in the steam jet air ejectors of the condenser evacuation subsystem as noncondensable gases. The discharge of the steam jet air ejectors is directed to a series of filters and compressors. The output of these components can be either released to the atmosphere (if radioactivity levels are low) or routed to the gaseous radwaste system (Sect. W01.A). The offgas subsystem is a Safety Class 3 system.

System equipment typically includes filters, fans (or compressors), radiation monitors, and valving. The arrangement is left largely up to the architect-engineer and operating utility.

Systems interfacing with the offgas subsystem are

- plant ac distribution
- condenser evacuation subsystem
- instrument air
- gaseous radwaste

W01.B Liquid radwaste system. The liquid radwaste system collects potentially radioactive liquids from the plant and treats them to reduce their concentration of radioactive nuclides. The liquid radwaste system penetrates containment. The system may be designated a Safety Class 3 system or a nonsafety system, depending on design.

Other names that may be used for the liquid radwaste system are

- radioactive liquid waste
- liquid radioactive waste
- liquid waste disposal

System design usually includes two piping systems that connect at several points. One section of piping is used for high-level radioactive liquids; the other is for low-level radioactive liquids. Liquids from the process gas section of the gaseous radwaste system, from the boron recovery subsystem (PWRs), the steam generator blowdown system (Westinghouse and CE units), the offgas subsystem (BWRs), or from the solid radwaste system enter the high-level section. The low-level section collects from the low-level process vent section of the gaseous radwaste system, the laundry and showers, and from the boron recovery subsystem (PWRs). Drain tanks are the first components in either section of piping. One of two paralleled pumps draws suction from its associated set of tanks. The high-level pumps discharge into a waste evaporator and then condenser. This combination removes impurities from the liquid. Periodically, the sludge in the bottom of the evaporator is drained, cooled, and sent to the solid radwaste system. The condensed liquid is pumped through a cooler into a series of demineralizers and filters. Upon exiting, the liquid is transferred to a storage tank for reuse in the plant if it has had its radioactivity level reduced enough. If the level is still high, it is transferred to the low-level section. From the initial tanks, the low-level pumps discharge through a set of filters into a waste evaporator where laundry and shower waste liquid is added. Again, suspended solids are separated and sent to the solid radwaste subsection. The low-level distillate is released to the environment if the level of radioactivity is low enough. If the level is above the release value, it is recycled or stored in a tank until the amount of radioactivity has been reduced.

The following systems interface with the liquid radwaste system:

- plant ac distribution
- gaseous radwaste
- solid radwaste
- containment isolation (BWRs) or engineered safety features actuation (PWRs)
- chemical and volume control (PWRs)
- steam generator blowdown (Westinghouse and CE units)
- laundry and showers
- offgas subsystem (BWRs)

W01.C Solid radwaste subsystem. The solid radwaste system is used to collect, holdup, solidify, package, and store radioactive materials prior to their shipment off-site to a disposal facility. This system may be classified as nonsafety or Safety Class 3, depending on design. It does not penetrate containment. Wastes such as sludge, spent resins from demineralizers, spent filter cartridges, and miscellaneous solid materials that have become contaminated with use are treated in the solid radwaste system.

Other names used to identify this system are

- radioactive solid waste
- solid waste disposal
- solid radioactive waste

The sludge and resins entering the system go to a holdup tank. When it is sufficiently filled for processing, it is transferred to an evaporation cycle that further solidifies the wastes. Next, a solidifying material is added, and the combination is placed into a container. After solidification is complete, the container is capped, labeled, and stored awaiting removal from the site. Compressible solid wastes are compressed into a container before a mixture of sludge and solidifier are added to immobilize them. Noncompressible materials are handled in the same manner except that they are not compressed. After solidifying, these containers are capped, labeled, and stored also.

Systems interfacing with the solid radwaste system are

- plant ac distribution
- liquid radwaste
- chemical and volume control (PWRs)
- boron recovery subsystem (of CVCS)
- reactor water cleanup (BWRs)
- condensate cleanup/polishing subsystem

- steam generator blowdown (Westinghouse and CE units)
- demineralized makeup water
- filters from many systems
- contaminated tools, clothing, cloths, etc.

W02 Radiation monitoring system

The radiation monitoring system is provided to ensure that any substantial abnormal radioactivity that is released can be detected within a reasonable amount of time. A level of protection philosophy is employed in that various points in the process systems are monitored, the plant itself is monitored, and the environment surrounding the plant is monitored. Limits are established by 10CFR20.

All points where potentially radioactive materials may enter the environment are monitored and alarmed. Usually the monitors' output is tied into control circuits to shut off equipment to stop the release. Selected areas inside the plant are monitored and alarmed. The plant is divided into zones with increasingly stringent levels used to generate alarms as potential contact with the environment is increased. Areas where plant personnel are expected to spend large amounts of time are also monitored. All monitors have independent power supplies to prevent losing a number of monitors because of a single failure. Areas outside the plant are monitored to ensure that no radioactivity exceeding established levels is released without detection. These environmental monitors may be either fixed or mobile.

W03 Cooling water systems

Numerous water systems are used in a nuclear plant. Some that are directly involved in the power generation process and in safeguards protection have already been discussed. Other auxiliary systems that support these systems' operation are listed below. The design of these systems varies greatly from plant to plant as well as from NSSS vendor to vendor; however, they all provide the same basic functions. Among these are cooling water systems, demineralized makeup water systems, service water systems, and chilled water systems.

W03.A Reactor building cooling water system

The reactor building cooling water system provides an intermediate cooling loop for removing heat from the engineered safety systems and transferring it to the essential service water system. The reactor building cooling water system is designated as Safety Class 3 and is typically subdivided into two or three distinct trains. The number of trains is dependent on the number of engineered safety systems provided by each reactor vendor. Electrical power for each train is provided through separate emergency buses. The reactor building cooling water system penetrates containment in PWRs and secondary containment in BWRs.

The reactor building cooling water system may be called by several names. Among these are

- component cooling water
- reactor plant component cooling water
- reactor building closed cooling water

A typical train of the reactor building cooling water system will consist of one or more (redundant) pumps and their associated motors connected in series with a heat exchanger for transferring heat to the essential service water system and several parallel piping legs that connect to the various engineered safety systems.

Systems that typically interface with the reactor building cooling water system are

- essential ac power
- essential service water
- containment spray
- residual heat removal/low-pressure safety injection
- high-pressure safety injection (PWRs)
- containment cooling/heat removal
- spent fuel cooling and cleanup

Other interfacing systems are

- reactor coolant/recirculation
- chemical and volume control (PWRs)
- spent fuel storage and cooling
- liquid and gaseous radwaste
- containment isolation (BWRs) or the engineered safety features actuation (PWRs)

W03.B Turbine building cooling water system. The turbine building cooling water system provides an intermediate cooling loop for removing heat from components located inside the turbine and auxiliary buildings and transferring it to the nonessential service water system. Coolers in such systems as the chemical and volume control system, the chilled water system, the reactor water cleanup system, and the steam generator blowdown system are cooled by the turbine building cooling water system. This system does not penetrate containment and is designated as a non-safety system. Electrical power is supplied by the nonessential ac distribution system.

The turbine building cooling water system may be called by other names. Among these are

- recirculating water
- turbine plant component cooling water
- turbine building closed cooling water

System design typically involves a single train with redundant pumps. Pump discharge flows through the tube side of a turbine building cooling water heat exchanger before entering a common header. Several parallel piping legs leave the header and pass through the shell side of various heat exchangers.

Systems that interface with the turbine building cooling water system are

- nonessential ac distribution subsystem
- chemical and volume control (PWRs)
- reactor water cleanup (BWRs)
- chilled water
- steam generator blowdown (Westinghouse and CE units)

W04 Service water systems

W04.A Demineralized makeup water system. The demineralized makeup water system is the source of high-purity water for use as primary grade water in the primary auxiliary systems, such as condensate in the secondary auxiliary systems, and for general use wherever demineralized water is needed inside the plant. The demineralized water system is a nonsafety system that penetrates containment. Electrical power is provided by the nonessential ac distribution subsystem.

The demineralized water system is normally divided into two sections: a water treatment section and a storage and transfer section. The water treatment section is supplied water by a tap on the circulating water system. This water enters a settling tank, and then it is passed through a series of filters and demineralizers until it reaches a high level of purity. It is then sent to the demineralized water storage tank for use in the various water systems in the plant. One of two redundant pumps is used to transfer the water. The storage tank, transfer pumps, and associated piping make up the transfer section of the system.

The following systems interface with the demineralized makeup water system:

- nonessential ac distribution
- standby liquid control (BWRs)
- reactor water cleanup (BWRs)
- chemical and volume control (PWRs)
- auxiliary feedwater (PWRs)
- condenser and condensate
- circulating water
- auxiliary steam
- solid radwaste
- potable and sanitary water
- chilled water
- spent fuel cooling and cleanup
- reactor and turbine building cooling water
- service water

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W04.B Station service water system. The station service water system is used to transfer heat from the cooling water systems and various other plant components to the ultimate heat sink (usually the circulating water system). In most designs, there is one service water system, but it is broken into essential and nonessential subsystems. The essential service water subsystem is designated a Safety Class 3 system and is powered by the essential ac distribution subsystem. The nonessential service water system is a nonsafety system and is usually isolated from the essential subsystem by a safeguards signal.

The station service water system is a two-train system. Redundant pumps draw suction from taps on the circulating water system piping. Some designs have a third pump as an installed spare. The pumps usually discharge into a common header (for nonessential use) and two parallel, redundant headers (for essential use). The common header can be isolated by a motor operated valve.

From the common header, legs branch off to provide cooling water flow for the turbine building cooling water cooler, for cooling jackets and large motors in the turbine and auxiliary buildings, and for the nonsafety Heating, Ventilating, and Air Conditioning (HVAC) water chillers. Each leg of the essential portion branches into parallel legs that connect to redundant components such as the reactor building cooling water heat exchanger, the fuel pool makeup (emergency), safety HVAC water chillers, cooling jackets of large motors inside containment, and the diesel generator cooling water heat exchangers. Flow from all the paralleled legs join together at a common discharge header and flow back into the circulating water.

Systems interfacing with the station service water system are

- plant ac distribution (both subsystems)
- circulating water
- reactor and turbine building cooling water
- chilled water
- compressed air
- reactor coolant and reactor recirculation (motor cooling)
- diesel-generator cooling water

W04.C Chilled water system. The chilled water system provides cold water to various HVAC air cooling coils and generator leads coolers. The chilled water system is normally divided into two parts. Typically, a Safety Class 3 portion of the system provides cooling water to the safety related air cooling systems inside containment. A nonsafety portion supplies cooling water to generator-lead coolers and to HVAC cooling units throughout the plant but outside containment. The nonsafety portion can be isolated on receipt of a safeguards signal.

The chilled water system is a closed loop system. Two or three pumps, depending on design, draw water from mechanical refrigeration units that are serviced by the station service water system. The pumps discharge into a common header that branches to go to each of the associated coolers. Isolation valves are provided to ensure flow to the safety related coolers during accident conditions. Flow from the various coolers returns to the refrigeration units. A surge tank is usually provided to control system water inventory.

Systems interfacing with the chilled water system are

- essential ac distribution subsystem
- containment cooling (PWRs)
- HVAC
- turbine-generator
- station service water
- containment isolation (BWRs) or the engineered safety features actuation (PWRs)

W05 Refueling system

The refueling system is used to exchange new fuel assemblies for used ones at the end of each fuel cycle. The refueling system is designed to meet Seismic Category I requirements. The system is also designed so that all refueling activities take place under water to take advantage of water's shielding and cooling capabilities.

A typical system consists of equipment in three places: in the reactor building, in the spent fuel building, and in the fuel transfer tube connecting the reactor building and spent fuel building. Equipment in the reactor building typically consists of one or two fuel handling bridges (depending on design), a control rod assembly handling tool (in PWRs), various cranes and hoists, and, to aid in control, an underwater closed circuit television system. In the spent fuel building, there is typically a fuel handling bridge, fuel storage racks, and a new fuel handling tool. The transfer tube is, itself, a part of the system. It may actually be two tubes: one for fuel entering the reactor building and one for fuel leaving the reactor building. The transfer tube, or tubes, can be isolated through the use of valves at the entrance to the reactor building. A carriage or a boom, depending on design, is used to move fuel assemblies through the transfer tube. The transfer tube and the containment in PWRs are flooded with borated water from the refueling, or borated, water storage tank. In BWRs, water for flooding is drawn from the condensate storage tank.

The refueling system interfaces with the plant ac distribution system, the spent fuel storage and cooling system, the condensate system (PWRs), the reactor core, and the control rod drive system (PWRs).

W06 Spent fuel storage system

The spent fuel storage system is used to store spent fuel assemblies that have been taken from the reactor core. The spent fuel storage system is designed for Seismic Category I loads and for tornados and external missiles. The atmosphere in the spent fuel building is confined and filtered.

The spent fuel pool is the major component of the spent fuel storage system. It is normally filled with primary grade demineralized water. In PWRs, borated water is added from the refueling or borated water storage tank. Storage racks in the bottom of the pool hold the spent fuel assemblies and maintain an adequate separation to prevent the fuel from reaching criticality.

The spent fuel storage system interfaces with the refueling system, the demineralized water system, and the fuel pool cooling and cleanup system.

W06.A <u>Fuel pool cooling and cleanup system</u>. The fuel pool cooling and cleanup system removes decay heat from the spent fuel and provides clarification and purification for the water in the fuel pool and refueling, or borated, water storage tank. The fuel pool cooling and cleanup system is a Safety Class 3 system. Electrical power is supplied from the essential ac distribution subsystems. The cooling and purification sections of the system may be completely separate or the purification section may divert a portion of the cooling stream through the filters and demineralizers. The cooling portion typically uses two independent, redundant piping trains, each with a pump and cooler. If the cooling and purification sections are combined, taps are provided in each line to divert flow through the filters and demineralizers. The purification section is also a two-train system with taps on the cooling piping lines, or with pumps and suction points in the spent fuel pool and refueling cavity. Pump discharge passes through a filter and into a demineralizer. The number of filters and demineralizers depends upon the system's design. From the demineralizer, the flow returns to the spent fuel pool. A loop is also provided for recirculating water in the refueling, or borated, water storage tank through the filtration system.

The fuel pool cooling and cleanup system interfaces with the essential ac distribution system, the spent fuel storage system, the demineralized water system, and the reactor building cooling water system (for secondary coolant flow).

W07 <u>Compressed air system</u>

The compressed air system supplies compressed air throughout the plant for pneumatic valve operation, instrumentation (where pneumatic systems are used), and for use with pneumatic tools. The system has two major subsystems: the instrument air subsystem and the service air subsystem. The instrument air subsystem is frequently divided into a Safety Class 3 portion that supplies loads inside containment and a nonsafety portion for the remainder of the plant.

The service and nonsafety instrument air portion are sometimes combined. This system normally has two 100% capacity or three 50% capacity air compressors. Each compressor has an intake filter and aftercoolers into a common header. Air receivers are provided to store compressed air. A service air leg branches from the common header in addition to the instrument air legs. Each instrument air leg contains an air filter and a 100% capacity air dryer. In designs featuring a separate containment instrument air system, air is drawn from and discharged to the containment, creating a situation of no pressure increase inside containment. As with the nonsafety portion, two compressors with filters, aftercoolers, and air dryers make up two redundant trains. Each train also has air receivers to store the compressed air.

Systems interfacing with the compressed air system are

- plant ac distribution
- station service water (for aftercooler secondary flow)
- various systems having pneumatically operated valves and instruments

W08 Process sampling system

The process sampling system provides a means of drawing samples from various process systems and the facilities for analyzing the samples. Samples drawn from inside the containment have remotely operated valves for isolation purposes. Typical of the analysis performed on the samples are boron concentration (PWRs), fission product radioactivity level, hydrogen and oxygen gas content, pH, corrosion product concentration, and conductivity. The system is designed to be manually operated on an intermittent or continuous basis under all plant conditions.

The process sampling system features numerous lines connecting process piping to sample sinks. Hot lines have sample coolers with secondary flow provided by the station service water system. Those samples on which conductivity analysis are performed have a constant temperature bath to minimize errors in measurement caused by temperature differences. Demineralized water is provided at the sample sinks for rinsing purposes. All sample sinks drain into the liquid radwaste system.

The process sampling system interfaces with the following systems:

- various process systems requiring chemical analysis
- station service water
- demineralized water
- liquid radwaste
- containment isolation (BWRs) or the engineered safety features actuation (PWRs)

W09 Plant gas system

The plant gas system is used to supply compressed gases to the various places it is needed in the plant. It is a nonsafety system. The plant gas system normally features a nitrogen and a hydrogen subsystem.

Both the compressed nitrogen and hydrogen are provided in gas bottles. Each of the bottles discharges through a pressure control valve into a header with parallel piping legs to distribute the gas.

Nitrogen bottles connect to

- safety injection accumulators
- volume control tank (in some designs)
- refrigerant dryers
- charcoal bed gas adsorbers
- gas compressors
- gaseous vents and drains (for purging)

Hydrogen bottles connect to the volume control tank (in some designs) and the generator stator cooling subsystem.

A.7 Plant Auxiliary System - X

Plant auxiliary systems are provided to support plant activities and personnel. They are typically nonsafety systems. Design of these systems varies greatly because almost all are plant specific. The systems and subsystems of the plant auxiliary system grouping are discussed below and listed in Table A.7.

Table A.7. Plant auxiliary systems[a]

X01	Potable and sanitary water
X02	Fire protection
X02.A	Water
X02.B	Carbon dioxide
X03	Communications
X04	Security
X05	HVAC
X05.A	Control room habitability
X05.B	Turbine building ventilation
X05.C	Diesel building ventilation
X05.D	Auxiliary building ventilation
X05.E	Fuel building ventilation
X06	Nonradioactive waste
X06.A	Gaseous waste
X06.B	Liquid waste
X06.C	Solid waste

[a]Plant auxiliary systems for both BWRs and PWRs are essentially identical.

X01 <u>Potable and sanitary water system</u>

The potable and sanitary water system provides water throughout the plant (except in containment) for drinking water, showers, laundries, and restroom facilities. It is a nonsafety system. The source of water may be a local water system or, in remote locations, the circulating water system. If circulating water is used, stations are provided for filtration and chemical addition to insure the water is safe for consumption. A common header is used to distribute the water, and another collects discharge. Discharge returns to a local sewage system when available, or through a sewage processing station before release to the plant water supply (downstream of intakes).

X02 <u>Fire protection system</u>

The fire protection system is designed to detect, annunciate, and extinguish any fires that may occur and to provide some mitigation for the effects of the fires. The fire protection is generally a nonsafety system. It is typically divided into two subsystems: a water system and a carbon dioxide system.

The water system is used in all areas of the plant except those that contain electrical equipment. Two water pumps are usually provided. One of these may be steam turbine driven. The pumps discharge into a header that supplies various fire hoses and overhead sprinklers throughout applicable areas of the plant. The water supply is taken from the circulating water system in most cases. Heat actuated detection devices activate the systems and annunciate its operation.

The carbon dioxide system is used to protect areas such as the cable spreading room, switchgear rooms, and electrical penetrations and tunnels. This system is supplied CO_2 from a storage tank. Piping connects the storage tank with valved distribution points in electrical areas. Heat actuated sensors operate the system. Rooms that require occupancy during normal operation, such as the control room, use portable CO_2 extinguishers.

X03 Communications system

The communications system is designed to provide reliable communications between essential areas of the station and to essential locations remote from the plant during all plant conditions. To ensure this capability, a diverse communications system is provided. Intraplant communications are provided by:

- a page/party public address and evacuation alarm system
- a sound powered telephone system
- a hand-held portable radio system
- a private branch telephone exchange (PBX)

Off-site communications systems include:

- local telephone service
- a microwave link through the utility's system-wide communications system

X04 Security system

The security system is designed to protect vital areas of the plant from intrusion by nonauthorized personnel. The system includes many levels of protection from the fence around the exclusion area to security procedures to microwave detection systems to card-keyed locks on doors within the plant. Because these designs are considered proprietary, little information is available about them.

X05 HVAC system

The HVAC system removes heat, ventilates, and maintains a comfortable environment for personnel in the various buildings and rooms of the power plant. (Containment cooling and ventilating systems are excluded from this group.) The HVAC system is made up of a number of individual systems. Each of the major equipment and personnel areas normally has its own system. Some designs, however, have a central system. The HVAC systems are nonsafety systems with the exception of the control room HVAC. The HVAC systems for safety related equipment are safety related if they are required for equipment operability.

All HVAC systems feature one or two trains, each with a fan, a filtration system, and an air cooler. Most are closed systems with return and discharge registers in appropriate positions to maximize air flow through the rooms.

Typical HVAC systems are provided for:

- control room
- switchgear rooms
- cable spreading room
- electrical tunnels
- diesel generator room (usually ventilation only)
- battery rooms
- solid waste and decontamination building
- turbine building (usually ventilation only)
- fuel building
- administrative building

X06 Nonradioactive waste system

The nonradioactive waste system is used to process all gaseous, liquid, and solid wastes that do not contain radionuclides and that are not handled by the radioative waste system (W01) that is discussed earlier in this document. It is not a safety system. This system is divided into three subsystems by waste category, similarly to W01: gaseous, liquid, and solid. Each of these subsystems are discussed below.

X06.A Gaseous waste subsystem. This category is included for consistency, but there are very few nonradioactive or uncontaminated gases generated, and those that are present are usually handled by the radioactive waste system.

X06.B Liquid waste subsystem. The liquid waste system collects liquid wastes that are not contaminated with radionuclides and may, therefore, be handled more simply and much less expensively. There is a large amount of this type of waste generated in both BWR and PWR power plants.

X06.C Solid waste subsystem. The solid waste system handles routine nonradioactive wastes similar to any large chemical plant and is separated from the radioactive waste system for economic reasons. In most power plants, the solid wastes generated are combustible and are, therefore, incinerated for volume reduction; the resultant ash is disposed of in sanitary landfills.

The nonradioactive waste system interfaces with virtually every other system in the plant.

Appendix B

GLOSSARY OF IN-PLANT RELIABILITY DATA TERMINOLOGY

B.1 General Terms

AE	Architect-engineer.
ANSI	American National Standards Institute.
BOP system	Balance-of-plant system: system that supports the nuclear steam supply system.
BWR	Boiling-water reactor.
Component	An off-the-shelf item procured as a basic building block for a system (e.g., pump or valve). It is to be distinguished from nuts, bolts, seals, and other smaller piece parts.
Corrective maintenance	Operationally initiated maintenance (i.e., maintenance motivated by a problem detected in operations or test).
Demand failure rate	Probability of failure to function per demand, where functioning may be starting, changing state, etc.
Failure	Termination of the ability of a part or component to perform its design function.
FSAR	Final safety analysis report. This document is issued by the utility as construction nears completion and the design of the plant is finalized.
Generic system	System that is common across many plants, regardless of nuclear steam supply system vendor and light-water reactor type.
IEEE	Institute of Electrical and Electronics Engineers.
LER	Licensee event report. Data collected since 1969 on safety-related component malfunctions, failures, human errors, and procedural deficiencies.
LWR	Light-water reactor. Subsets are BWR and PWR (see separate definitions).
NPRDS	Nuclear Plant Reliability Data System.
NSSS	Nuclear steam supply system. Consists of the reactor, hardware directly connected with the reactor vessel, and equipment directly associated with the reactor coolant and the principal subsystems.

P&ID	Piping and instrument diagram. This is an engineering drawing that shows the functional location of components such as pumps, valves, and instruments in a system.
Plant	A facility that contains one or more nuclear power generating units.
Preventive maintenance	Prescheduled or preplanned maintenance (i.e., maintenance motivated by a desire to prevent operating problems).
PSAR	Preliminary safety analysis report.
PWR	Pressurized-water reactor.
SAR	Safety analysis report. Subsets are PSAR and FSAR.
System	A collection of components arranged to provide a desired function (e.g., containment spray system or residual heat removal system).
Time failure rate	Probability of failure to function per unit time.

B.2 Failure Types

Catastrophic failure	Failure where component is completely unable to perform its design function.
Common cause failures	Failures of two or more components or systems due to a single cause.
Degraded failure	Failure where component operates at less than its specified performance level.
Demand-related failure	Failure that is most probably associated with one or more demands for a component to start, change state, etc.
Generic failure	Failure that is common across many components, systems, or plants.
Incipient failure	Failure where component performs within its design envelope but exhibits characteristics which, if left unattended, could develop into a degraded or catastrophic failure.
Time-related failure	Failure that is most probably associated with the normal activity and exposure of a component over a reasonable span of time.

B.3 Definitions of IPRD Variables

Annual operations	Number of starts or demands of a component during a typical one-year period.

Availability	Proportion of nonoutage hours in a given unit's operations over a specified period of time, per the NRC's Licensed Operating Reactors: Status Summary Report ("The Gray Book"), NUREG-0020.
Component ID	Plant-specific code name for component.
Component type code	Six-letter code which identifies a component.
Component vendor code	One-letter/three-digit code for the component's manufacturer. Same codes are used in NPRD.
Crew size	Number of people in the maintenance crew that performed the repair of the failed component.
Duty cycle	Percentage of time that the component is active in a particular plant operating mode during a typical one-year period if always in good repair.
Failure cause code	Two-digit code keyed to causes such as design error, seal/packing failure, normal wear, personnel error, or leaky fittings (may be multiple failure causes).
Failure cause description	Failure record text description of failure event.
Failure date	Month, day, and year (mmddyy) associated with a failure. Depending on plant reporting procedures, this date could be the date of failure occurrence, the date of failure detection, or the date of the written failure report.
Failure mode code	A one-letter code keyed to characterizing the effect by which a failure is observed.
Failure severity code	One-letter code keyed to "catastrophic, degraded, incipient" (see separate definitions).
Functional name	Population record text description of component's function.
Hierarchical number	Summarizes a component's physical and operating characteristics (e.g., for a pump, positive displacement vs centrifugal type, head, horsepower, and operating temperature).
Operating mode	The component's dominant mode of operation. For pumps, a one-letter code keyed to running, alternating, or standby.
Plant code	Two-digit coded plant number.
Repair action description	Repair record text description of repair action as filed on the plant's maintenance work request.
Repair category	Indicates the degree of repair performed on the component.

Repair date	Month, day, and year (mmddyy) associated with the repair of a failed component. Depending on plant reporting procedures, this date could be the date of component repair initiation or completion.
Repair time	Elapsed clock time to the nearest half-hour required to repair a failed component, reported as a real number with a decimal point.
Report number	Plant-specific report number or log page number where failure or repair is recorded.
System code	One-letter/two-digit code for the generic system to which a component is assigned. Letter code keyed to "Nuclear (N), Engineered Safety (S), Containment (C), Electrical (E), Power Conversion (P), Process Auxiliary (W), Plant Auxiliary (X)."

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