INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

PRE-STUDY TEXT

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INTRODUCTION

This instrumentation and Control Training course provides the means for the U.S. Nuclear regulatory Commission Inspector to gain an appreciation of instrumentation and control technology and Applicable Documents (i.e, codes, standards, guides, etc.) as related to nuclear power plant construction. The Electrical Training Course provides the means for the NRC Inspector to gain an appreciation of Electrical Technology.

Project Objectives

The specific objectives of the training program are to provide Nuclear Regulatory Commission professional inspector personnel with the following:

a. An appreciation of the content and principal requirements, recommendations, and/or positions available in Applicable Documents (i.e., codes, standards, guides recommended practices, etc.) considered significant to nuclear power plant instrumentation and control technology.

b. An appreciation of the typical and generally acceptable design methods significant to instrumentation aspects of nuclear power plant construction.

c. An appreciation of the salient construction practices utilized with reference to nuclear safety related Applicable Documents. This provides the inspector with background information to be used in conjunction with U.S. Nuclear Regulatory Commission policies and positions to conduct nuclear power plant construction inspection activities.

Primarily, the inspector of a nuclear power plant construction project is confronted with documentation and the physical installation at the nuclear power plant construction site. Upon completion of this program, the inspector will have been presented with the system technological background to determine if an installation of a nuclear safety related instrumentation and control system warrants further in-depth review to determine acceptability. for example, Figure i-2-1 is a block diagram of an instrumentation and control loop. The inspector will be able to ident fy the documents (system and physical installation) associated with the nuclear safety related systems in this loop. The inspector will be able to identify the components used in the loop, determine the correct physical location of the components within the loop, and identify discrepancies between the actual installations and the applicable documents.

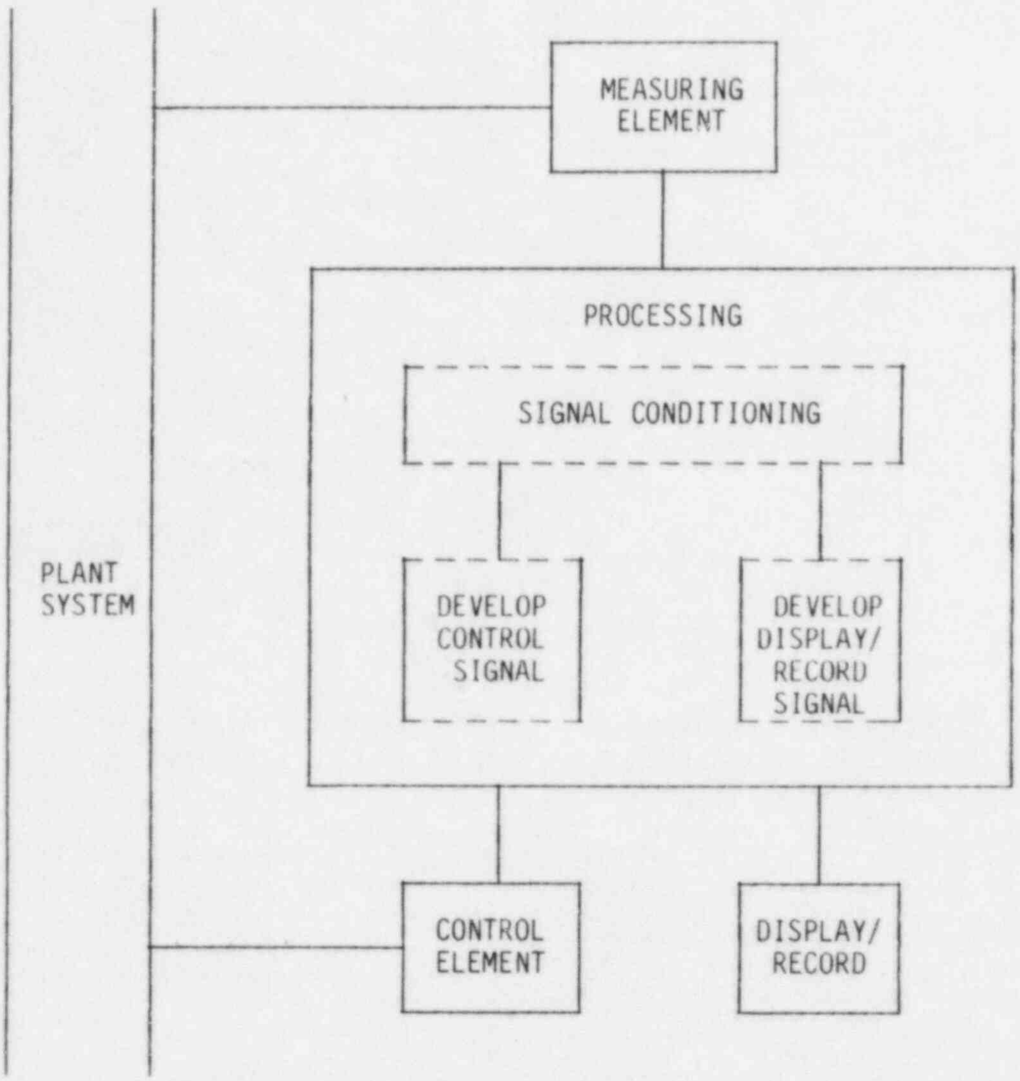

Figure i-2-1, Instrumentation and Control Block Diagram

General Design Criteria 13, "Instrumentation and Control", and General Design Criteria 19, "Control Room", of Appendix A, "General Design Criteria for Nuclear Power Plants", to Title 10 of the Code of Federal Regulation, Part 50 (10CFR50), "Licensing of Production and Utilization Facilities" contains the primary requirements (federal regulations) for instrumentation and control important to safety. Other general design criteria deal with specific safety related systems, such as reactor protection, emergency core cooling, cooling water, etc. this program describes how the Applicable Documents are used to interpret and expand on the federal regulations.

It is important to note that the evolution of Applicable Documents (i.e, codes, standards, guides, recommended practices, etc.) is an ongoing process. This training course addresses the salient applicable documents through November, 1978. The presentation of Applicable Documents attempts to provide their most up-to-date interpretation which has found industry-wide acceptance. However, the inspector can expect to encounter alternate interpretations or methods of compliance with interpretations which deviate from those presented in this course. The design of nuclear power plants has not reached the stage of static, pat solutions.

Rather, each new plant represents an evolutionary design development from previous plants. In addition, USNRC Regulatory Guides provide an NRC position on a specific issue of a standard; therefore, this project may address two issues of a standard (i.e., the one identified by the USNRC in a Regulatory Guide as well as later issues of a standard). Consequently, this course attempts to clarify basic principles to promote understanding without compromising the inspection function.

Selection of Industry Examples

Examples and data provided as the basis for training were selected from industry-wide sources (e.g., technical society presentaitons) and public documents (e.g., various safety analysis reports). This assures that the point of view provided does not reflect only the viewpoint of the preparer of this training program.

Program Organization

The program is organized into a multi-unit course text. These units are presented in a building block format.

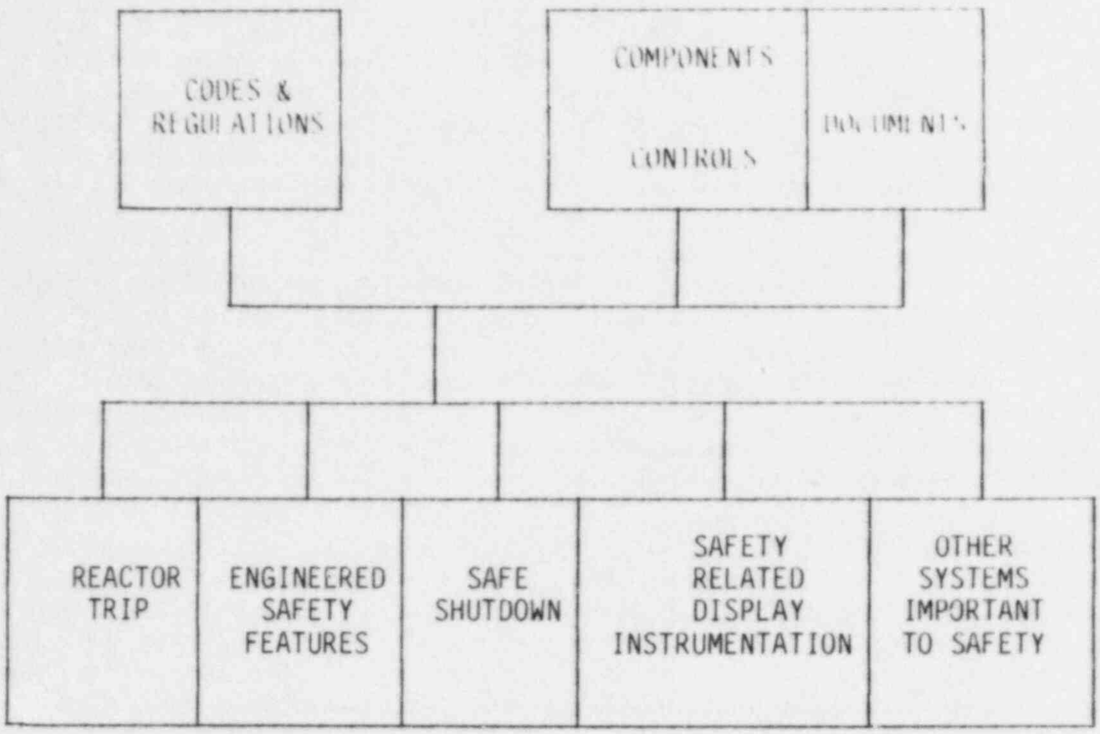

Figure i-2-2, Instrumentation and Control Course Outline

We will start out describing the types of components that are used in instrumentation and control. We will then organize these components into controls going from simple to complex along with the entire documentation used. After we have reviewed the codes and regulations applicable we will combine the components, controls, documents, and regulations as they apply to the reactor trip, engineered safety feature, safe shutdown, safety related display instrumentation and other systems important to safety.

Program Organization

Pre-study material is provided to the program participants prior to the lectures. This enables the inspector to preveiew the course test and lecture material, generally in less than 1-1/2 hours of time, and enhances assimilation of the basics of the course lectures to follow. tests are given to the inspectors on the pre-study material to enable the inspector and others to evaluate the successful achievement of the objectives indentified in the pre-study material.

Program Relationship to Inspection Function

Information provided in this training program shall not be the basis for a determination of the acceptability or unacceptability of practices seen during construction of a power plant. Rather, information gained from this program during lectures, and information provided in reference handouts shall be considered tools necessary for the inspector to understand the instrumentation and control technology he must use to perform his job.

Furthermore, this training program is not intended to serve as a training manual for other aspects of the necessary quality assurance progarms (e.g. design control, procurement control, organizing a quality assurance group, preoperational testing, qualification of equipment, etc.) for a nuclear power plant nor as a training program for I&C design engineers.

INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

UNIT I - Basic Instrumentation

PRE-STUDY TEXT

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UNIT I - Basic Instrumentation

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NUCLEAR REGULATORY COMMISSION INSPECTORS

UNIT I - Basic Instrumentation

PRE-STUDY MATERIAL

1.0 OBJECTIVE

The NRC inspector will learn the basic instrumentation hardware, symbology, and terminology and documents used in instruments and control for a nuclear power plant.

2.0 INTRODUCTION

2.1 This pre-study material will introduce the NRC inspector to the areas of instrumentation and control that will be covered in more detail in the subsequent units. The main topics for discussion for this unit include the following:

- Terminology and Symbology
- Sensing Instruments
- Signal Conversion and Transmission
- Display/Record
- Control
- Actuators
- Instrumentation Documents
- Installation Practices and Materials

2.2 The material to be presented will be of several levels of difficulty, ranging from relatively easy to assimilate to quite difficult to assimilate. This is because there are basically three areas which are necessary to cover for the inspector to have a complete background in instrumentation and controls. One area is the theory of operation of individual devices, such as measuring devices, controls, final drives, etc. This is straightforward material and will be relatively easy to assimilate.

The second area is terminology and definitions. Though attempts have been made to standardize terminology within the instrumentation and control industry as a whole, the attempts have not been successful. In fact, they have not been successful in any one given field, including the power industry. A classic example of this terminology confusion is the use of the words "transmitter" and "transducer". To some people these words are synonymous, to others, they have a specific meaning. In many cases, what one

vendor, utility, and/or architect/engineer (A/E) will call a transducer, the others will call a transmitter. As a result, we must consider what is meant by the term being used and apply a meaning to that term which is coincident with what is meant by the user.

The third area of coverage is the application and installation of various devices into a measuring and control, or indication loop. This material is not as straightforward as the first two areas since there are numerous ways to start out with a given measurement and end up with a desired indication, record, or control function. Presently, the only good personnel development tool is hands on experience.

2.3 Although a nuclear power plant is made up of many complex systems, only a few of these systems are nuclear safety related (NSR) and in general, they are of a "go" or "no go" mode of operation. To many people, this is known as digital control. The words digital or digital control should not be confused with the words direct digital control (DDC). DDC is used mainly in the process industry and to a lesser degree in fossil fuel power plants.

2.4 Nuclear Safety Related (NSR) is a general term applied to any system or component that performs a function that is required in a nuclear power generating facility to prevent undue risk to the health and safety of the public. This term has many interpretations and many attempts of clarification have been developed by A/E's, utilities, and nuclear steam system suppliers (NSSS). For the purpose of this course, nuclear safety related will be a term applied to any structure, system or component which identifies that structure, system or component as being required in a nuclear power generating facility to assure that the public's health and safety is not jeopardized during an accident condtion.

2.5 The objective of instrumentation or instruments and control (I&C) in a nuclear power generating facility is multi-faceted; basically I&C systems should maximize reactor power output efficiency while providing adequate plant control. Controlling the plant's operation effectively eliminates potential accident/incident conditions and thus assures that the health and safety of the public is being protected.

2.6 This unit presents specific types of measuring instruments used for various types of process control. Generally, the variables that are measured and/or controlled most frequently are temperature, pressure, level, rate of flow, and radiation. It is not the intent to identify and discuss every type of measuring device, but to cover those types most commonly used in nuclear power plants.

3.0 INSTRUMENTATION FUNDAMENTALS

3.1 Terminology and Symbology

The terminology and symbology used in I&C is unique and requires coverage in this pre-study. As indicated earlier, standardization of terminology and symbology unfortunately has not been reached in industry. Each NSSS, utility, and A/E have terms and symbols all their own. In general however, most of the aforementioned groups used the Instrument Society of America (ISA) standards as a basis for their individual terminology and symbology system. You should keep in mind that any NSSS, utility or A/E may vary from this reference list to suit their own immediate needs.

Symbology is generally very difficult to assimilate. It can be understood more rapidly and more completely when hands-on review of the drawing is being done. Hence, at this point we are reiterating that the various utilities, NSSS's, and A/E's generally use the ISA symbology as a basis for their own terminology system. Detailed discussions of the differences will be left to the units of instruction where the drawings are reviewed.

3.2 Sensing Instrumentation

3.2.1 Pressure

The five most common types of pressure sensors are:

- Bourdon Type

- Bellows Type

- Diaphragm Type

- Strain Gauge

- Piezoelectric Sensors

All of these five types employ the same basic principle. A pressure from a process will mechanically displace the sensor in a predetermined manner so that the degree of movement can be translated into a signal which is proportional to the process pressure.

3.2.2 Temperature

The most common types of temperature sensors are:

- Filled System (Liquid, Vapor, or Gas) This type of system relies on the thermal expansion or contraction

of the "fill" to correspond to the temperature in a predetermined manner.

- Bi-metallic Elements This type also relies on the thermal expansion or contraction of two metals in a predictable manner which causes a motion with a change in temperature. This motion can be translated to a signal that is proportional to the actual temperature.

- Thermocouples This type of temperature sensor relies on the principle of two dissimilar wires being joined (called the hot junction). The simple joining of these two dissimilar materials generates a millivoltage across the open leads of the two wires. Increasing the temperature of the hot junction will increase the millivolt output in proportion to the temperature change.

- Resistance Elements This type of temperature sensor relies on the principle of the change in resistance of a thin wire that is proportional to the temperature change. Throughout the industry this is known as a "RTD", standing for Resistance Temperature Detector.

3.2.3 Flow

The two most common categories of flow sensors are head and linear.

- Head or Differential Pressure Flow Sensor (Flowmeter) This type of flow sensor is made up of two devices: the primary element (orifice plate, flow nozzle or venturi tube), and the measuring device (differential diagram, opposing bellow, etc.).

  The primary element is mounted in the main process pipe, causing a differential pressure. This differential pressure is measured by the measuring device. The differential pressure is directly proportional to the square of the process fluid velocity. The velocity is then converted into a flow reading by either mechanical or electronic converting mechanisms, either internally or externally from the flow measuring device. This method of flow measurement is by far the most common utilized in nuclear power plants.

- Linear Flowmeters (including Area and Positive Displacement) Both of these types rely on the flow of fluid to mechanically displace various types of objects in the flow path in a predictable manner which are translated in a signal proportional to the flow.

3.2.4 Level

The two most common types of level elements are direct and hydrostatic head.

- Direct Type (Sight Glass, Float and Displacer Type) The sight glass or more commonly called "Gauge Glass", is a transparent plastic, glass, or treated glass tube attached or adjacent to the side or end of a liquid holding vessel (tank). The top of the tube is connected above the liquid level and the other end of the tube connected at or near the bottom of the vessel. The level in the vessel is indicated in the transparent tube. This device is basically used for local indication only.

  The float type is self-explanatory in that a float is installed in the tank or a stand pipe outside of the liquid holding tank. The float rises and lowers with the liquid level. This motion is transferred outside the tank via mechanisms through process seals. The most common use of the float type level sensor is for switch actuation for high or low level alarm, stopping and starting pump motors.

  The displacers (sometimes called buoyancy float type) relies on the buoyancy of a non-floatable device as it is immersed in liquid. To clarify the preceding statement, the weight of the device appears less to a mechanical linkage as the liquid comes up around the non-floating device.

  This change in weight via the mechanisms can be con verted into a signal that is proportional to the liquid level in the vessel. To draw a corollary on the buoyancy effect, your weight as measured by a scale is greater on dry land than when you are chest deep in water. This is one of the more commonly used methods to measure levels where relatively small level differentials exists and high accuracies are required.

- Hydrostatic Head Method This type of level measurement relies on the differential pressure between the unfilled area in the vessel and the pressure of the filled portion of the vessel. Essentially, the pressure in the unfilled area in the vessel is subtracted from the liquid level pressure to end up with the static head, which is the liquid level in the tank. Sometimes temperature compensation is required if the tank liquid temperature is considerably higher than the temperature

of the liquid at the sensor. This is only a concern when very accurate level detection is required. The main mechanism for measuring the differential is a diaphragm or bellows type of device, the movement of the diaphragm or bellows can be translated into a signal proportional to vessel liquid level.

3.2.5 Nuclear Radiation

There are many facets to and reasons for measuring nuclear radiation at a nuclear plant. The purpose of this section is to develop a basic background knowledge of the overall picture of nuclear radiation detection with specific emphasis on that which is installed during construction of the plant, that is NSR.

There are basically four types of radiation with which we are concerned: Alpha, Beta, Gamma, and Neutron. (In light water reactor plants, the two most important types are Gamma and Neutrons.) The following paragraphs identify and discuss the purpose of the various reasons for measuring radiation at a nuclear power plant.

3.2.5.1 Site Radiation

In general, when determining site radiation, we are looking for very low levels of radiation. There are many sources for this radiation, i.e., the sun, radioactive elements in the soil and water, bomb test fallout, etc. Prior to actual construction of a nuclear power plant measurements will be made to determine background radiation. The purpose of this is to establish a base for background radiation at the site. Site background radiation activity continues to be monitored throughout the construction phase and during the operation of the plant to determine if there are any changes in site background radiation levels.

3.2.5.2 Human Exposure

To protect the public and plant operating and maintenance personnel from being exposed to radiation, nuclear radiation detectors are required to be installed at various locations in the plant. There are definite guidelines established as to what dosages can be on a daily, weekly, monthly, quarterly, yearly, and lifetime basis. The dosage and exposure is recorded by the health physics group of the utility.

3.2.5.3 Materials Exposure

This is usually a long-term activity conducted to determine the effects of radiation on materials within the plant. We are now talking about relatively high radiation levels. The purpose of these measurements is to determine the physical degradation, if any, of the essential parts of the plant. Some primary targets of the study are the reactor vessel walls, reactor supports, etc.

3.2.5.4 Plant Radiation

Plant radiation measurement is comprised of area monitors. Area monitors are radiation monitors strategically located throughout the plant and measure radiation in the general area of the monitor. Examples of these locations are the new fuel handling and storage area, the reactor building, the spent fuel handling and storage area, the radioactive waste handling area (commonly called radwaste handling areas), and any other areas where radioactive materials are transferred via pipes, trucks, etc The locations of these devices are agreed upon by the utility and the NRC. Except for accident conditions, we are dealing with low radiation levels. Process monitors are radiation monitors which sense the activity of specific process systems in order to detect any leakage of radioactive materials from one system to another.

The other major plant radiation measurement that should be discussed is the portable measurement equipment. This activity is usually under the jurisdiction of the health physics group of the utility. There are several types of equipment used here that range from portable units on wheels to hand-carried devices.

3.2.5.5 Neutron Detectors

Because there is a direct relationship between flux distribution and thermal power distribution in the reactor, neutron detectors are used to measure power level. Power level is an input to the safety systems of the plant. The types of detectors used are gas-filled ion chambers and fission chambers. They both operate on the principle that radiation causes ionization and this ionization can be measured.

3.2.6 Seismic Instrumentation and Measurements

Seismic instrumentation and measurements vary as a result of the site location (seismic envelope), utility philosophy, the A/E's recommendation, and the NRC requirements for the site. However, the base system generally consists of a seismic measuring system located in the reactor building and another system located within the site boundary. The record of the seismic disturbance can be integral to the seismic measuring system, transmitted to a remote recording system(s) or a combination of both. The purpose of this instrumentation and record is to establish an intensity of the seismic disturbance to ascertain whether or not a design basis seismic disturbance has been approached.

3.3 Signal Conversion and Transmission

Signal transmission of the variable takes on many forms. Some of these are: human physical observation [i.e., the level sight glass, paddle flow switch (not discussed), and pointers on an indicator], the motion of diaphragm, etc., which is then translated into a signal proportional to the process variable being measured. This signal usually is a milliampere (ma) signal and most commonly a 4-20 milliampere signal. This signal is most commonly used because it does not tend to degradate over long wire runs, and is less susceptible to noise pickup. However, it can also be converted into a millivolt signal or a pneumatic signal. The most commonly used pneumatic signal is 3-15 psi. This conversion of the motion, thermocouple outputs, resistance changes of the resistance temperature detector (RTD's), is accomplished by a device called a transmitter (however, to many, as indicated earlier, the word "transducer" is equally used). The transmitter device can be an integral part of the measuring device. It can be remotely located on a convenient wall, on a structural beam, in a plant instrument rack, or in the control complex, which for the purpose of this specific discussion includes the electrical equipment room. In the next two units of instruction, we will identify this signal conversion device as a transmitter.

Another form of a measured variable is digital. The motion of the diaphragm, bellows, etc., is translated into the opening or closing of an electrical contact which can actuate an annunciator window, pilot light, directly start up or stop a small piece of electrical equipment and most commonly will pick up a relay coil. The majority of the nuclear safety-related (NSR) circuits are of the relay pickup type.

3.4 Display/Record

Display and/or record in a nuclear power plant most generally is

direct or indirect. Examples of the direct type are level sight glasses, local thermometers, local pressure glasses, direct connected recorders (thermocouples, RTD's, etc.). Indirect display or indicating types utilize a transducer between the measured variable and the indicator and/or recording device. The indication can be in the form of a pilot light, gauge (indicator), annunciator-window, etc. The record can be in the form of a computer printout, or an indicating recorder, with continuous trace multi pen recorders or multi point strip chart recorders.

3.5 Control

3.5.1 Control is a very complex and encompassing term. However, it can be simplified if it is broken down into categories or types, such as analog digital, digital, and analog/digital interface.

3.5.1.1 Analog control can be described as measuring a process variable, comparing that measurement to a set point, and adjusting controls to compensate for any discrepancies between the compared values. The adjustment of controls is done with a control signal. The control signal will increase or decrease to position the final drive unit to establish the process measured variable to the desired set point. As an example of this, assume the turbine bearing oil temperature return to the oil cooler is optimum at 125°F and its actual temperature is 130°F. This means that we have a 5°F above set point discrepancy. The controller will proportionately increase the flow of the cooling water to the heat exchanger by modulating the control valve. The increase in cooling water flow will increase the rate of heat removal from the oil, and decrease the temperature of the oil flowing from the oil cooler. This modulation of the control valve will continue until the set point is reached.

3.5.1.2 Digital control takes on many forms which range from the simple contact closure to the very complex interlock circuit. Simple contact closures are made with level switches, pressure switches, etc. Once the contact is made, or broken, a relay is activated, or deactivated, to start or stop a motor for a pump valve or other piece of control equipment. An example of a simple interlock circuit is the personnel access lock in the reactor building containment. The inside and outside doors of the lock cannot both be opened at the same time.

3.5.2 The majority of the NSR controls fall under the digital control area. Level switches, pressure switches, temperature switches, etc., either directly connected to the process or indirectly connected to the process (signal coming from the transducer) are involved. The method of obtaining the redundancy, backup, protection, etc., varies appreciably between the NSSS vendors. Each of these systems or approaches will be discussed in detail in the following units. However, the mounting, location, and physical operation of the devices; wiring, sensing lines, and taps, the materials used, and the quality of construction workmanship of these control loops/circuits, are the items of greatest importance to you.

3.6 Actuators

The device called an actuator takes on many shapes and forms; however, the end result is the same; something moves. Other words that are used by the industry are final drives, sound controlling element and drive units. Some of the actuator devices are: electric motor drive units, pneumatic or hydraulic pistons, pneumatic diaphragms, and power solenoids. All of these devices cause a movement to open or close a valve, damper, etc.; lock or unlock doors, etc.; and initiate an on/off activity. All actuators are capable of intermediate modulation of the device; i.e., control valve, damper, speed control, etc.; except the power solenoid.

4.0 INSTRUMENTATION DOCUMENTS

4.1 General

The various documents used by I&C for nuclear power plant design again vary considerably from one A/E to another. The end result is the same; a power plant that can be constructed, operated, and maintained. The variations between the drawing terminology and what is shown on which drawing is significant.

The documents available to help the inspector perform his work fall into three categories:

a. Written descriptions (SAR documents).

b. System documents that describe relationships, organization and operation (flow diagrams, P&ID's, loop diagrams, logic diagrams, elementary diagrams).

c. Physical documents that show the location and mounting of instrumentation and control equipment and the routing and connection of electrical cable between the instrumentation and control equipment (general arrangements, rack and panel

layouts, and mounting, installation details, instrument piping, connection diagrams).

4.2 Written Documents

The SAR documents contain a written description of the instrumentation and controls of a nuclear power plant. Chapter Seven of the SAR covers the I&C portion of the safety-related systems in the plant. The system descriptions are contained in other sections of the SAR, particularly Chapter 6, 8, 9, and 10. These chapters contain some of the system documents that will be used by the inspector.

4.3 System Documents

The term flow diagram is used by some A/E's, the term Piping and Instrument Diagrams (P&ID's) is used by others, and still others use the term Process Instrument Diagrams (also P&ID's). The one common vein to these three types of drawings is that the process is shown and the sequency of the equipment in the process is shown.

The detail and process information (pressure, flow rates, temperatures, etc.), pipe sizes, and the degree of instrumentation varies appreciably from plant to plant. The information is always shown somewhere. An example of this is that the flow diagrams and process instrument diagrams do not generally indicate pipe sizes. This information is available on the physical pipe drawings. The big difference that affects us is the degree of instrumentation and control detail shown on the various types of drawings. In general, the A/E's that use flow diagrams show considerable detail relative to analog measurement, indication/recording and control, but leave the interlock motor starters and electrical switch gear controls to be covered by other drawings. The two types of P&ID's go into other details, which will be discussed in this unit's text.

4.3.1 The term loop diagrams, instrument loop diagrams, instrument schematics, boolean logic diagrams, and logic diagrams, are used by the industry to label a multitude of information. The main purpose is to indicate the electrical control and instrument detail on a schematic basis that is not shown on the flow diagrams and P&ID's.

4.3.2 Elementary diagrams (sometimes called electrical elementary diagrams) is a term that probably has the most commonality among the A/E's. Basically, it is a drawing of electrical control (not necessarily including analog control) circuitry, with one side of the power supply at the top of the drawing and the other at the bottom. The

information shown in between the power supply lines are all the switches, relays, indicating lights, fuses, etc., which make up the circuit.

4.4 Physical Drawings

General arrangement drawings show the location of major equipment, including instrument racks and panels, process cabinets, and control boards in the control room areas. Rack and panel drawings contain detail information for the equipment mounted therein. Instrument installation details give specific direction for the installation of control elements and their connection to the operating system of the plant.

Connection diagrams, circuit wire diagrams (CWD's), wiring cable termination diagrams, computerized connection cards, termination listings, etc., are terms utilized by A/E's to present construction personnel with information required to make the necessary physical connections.

Probably one of the most common important challenges to each of you as an inspector is to ascertain what types of drawings are used in a particular plant. You will need to know what drawings to look for to obtain the location of the system of interest and to determine the extent of the system for your inspection.

4.5 Installation Practices and Materials

The course lessons will address the installation details, including mounting, location, slope of impulse (sensor) line, slope of signal lines (both transmitted signal and control signal), separation criteria, protection practices, piping and tubing selection (including the workmanship of the installation), wiring, terminal blocks, etc.

5.0 PRE-STUDY SUMMARY

This pre-study material has provided a basis for the inspector to understand the terminology associated with the various types of control systems terms that will be used in describing control loops. As the control loops are discussed in the actual text, the inspector will become familiar with the terms used in describing the complex array of instrumentation and control systems found in a typical nuclear power plant.

INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

Unit II - Codes and Regulations Applicable to Nuclear Power Plants

PRE-STUDY TEXT

INSTRUMENTATION AND CONTROL TRAINING COURSE

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NUCLEAR REGULATORY COMMISSION PLANT INSPECTORS

Unit II - Codes and Regulations Applicable to Nuclear Power Plants

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INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

Unit II - Codes and Regulations Applicable to Nuclear Power Plants

1.0 OBJECTIVE

Sufficient information is presented so that the NRC inspector may acquire a working knowledge of the guides, codes, and standards as applicable to instrumentation and control design, and construction practices in a nuclear power plant.

2.0 INTRODUCTION

Standardization can reduce the cost, inconvenience, and confusion that result from unnecessary and undesirable differences in equipment, systems, material, and procedures. Standards can also document accepted industry practice in areas such as safety, testing, and installation. Within an engineering organization, standardization is accomplished by company standards. Between organizations it is often accomplished through industry standards.

Industry standards are generally published by professional societies and standards organizations. Standards are written through an industry consensus as achieved by activities within these organizations.

To protect industrial employees and the general public, municipal, state and federal government incorporate codes into their laws and regulations. A code is, basically, a standard that has been written in enforceable language. It is written with the expectation that it will be adopted as law. The National Electrical Code and the ASME Boiler and Pressure Vessel Code are examples. The objective of each code is to ensure public and industrial safety in a particular technical activity. Codes are often developed by the same organizations that develop standards. For example, the American Society of Mechanical Engineers (ASME) has an active standards program and has also developed the ASME Boiler and Pressure Vessel Code.

In the nuclear power industry, the compelling need is for safety and this need has motivated the development of standards. Most of the approved nuclear reactor standards and those in preparation are related to reactor safety.

The public safety aspects of nuclear power reactors have been regulated by the federal government since the beginning of the industry. When few reactors were being built, it was practical to evaluate the safety

of each reactor on its own merits. As the activity in the nuclear industry increased, designers needed to know the basis on which their designs were to be evaluated. An industry consensus on various aspects of safety had to be developed so that any group proposing the construction of a reactor could then present to regulatory bodies a design conforming to that industry consensus. Although regulatory bodies have not been obligated to accept an industry consensus, many important safety questions have been resolved by its development.

In summary, the primary factors that have accelerated standardization in the nuclear power field are safety considerations in the design and operation of nuclear power plants and the fact that nuclear power is a government regulated industry.

3.0 APPLICABLE DOCUMENTS

3.1 Government Regulation and Regulatory Agency Guides and Positions

3.1.1 United States Nuclear Regulatory Commission

The Nuclear Regulatory Commission (NRC) is responsible for regulation over the production and utilization of nuclear energy and related facilities. Protection, both of the public health and safety and of the environment, is paramount to the NRC's regulatory activities.

3.1.2 Code of Federal Regulation

The NRC's rules and regulations are contained in Title 10 of the Code of Federal Regulations (10 CFR), Chapter 1. These regulations were put into force in accordance with the Atomic Energy Act of 1954 as amended, and in the Energy Reorganization Act of 1974. They have the force and effect of law, and compliance with them is mandatory.

Notice concerning rule changes or proposed new rules are published in the Federal Register and the time period is usually specified within which interested persons may offer comments. After consideration of comments, final rules are published in the Federal Register and become effective.

3.1.3 Safety Analysis Reports (SAR)

The NRC regulations requires that a Preliminary Safety Analysis Report (PSAR) be submitted as part of the application for a Construction Permit (CP), and a Final Safety Analysis Report (FSAR) be included as part of the application for an Operating License (OL).

In addition, the PSAR presents the design criteria and preliminary information for the proposed plant and comprehensive data for the proposed site. The report also discusses various hypothetical accident situations and safety features which will be provided to prevent accidents, or if they should occur, to mitigate their consequences.

The information presented in the FSAR includes plans for operation, detailed procedures for coping with emergency situations and pertinent details on the final design of the plant itself.

Prior to the submittal of the SAR, the applicant should have evaluated the plant in sufficient detail to conclude that it can be built and operated safely. Adequate information should be provided in the SAR to permit an in-depth review of the design, fabrication, construction, testing, and expected performance of the plant structures, systems and components important to safety. All such features should be in conformance with the NRC's regulations including the general design criteria. Details of the quality assurance program, design methods and calculational procedures should also be presented in the SAR.

The PSAR is reviewed by the NRC Regulatory Staff and by the Advisory Committee for Reactor Safeguards (ACRS). ACRS reports its findings to the NRC chairman. Regulatory Staff presents a Safety Evaluation Report (SER), and the applicant submits an Environmental Report (ER) to the Atomic Safety Licensing Board (ASLB) in the public hearing pertinent to the PSAR review. The ASLB determines whether the application and the record of proceedings contains sufficient information for assessment of radiological safety, environmental impact and whether review of the application by the Regulatory Staff has been adequate to support findings proposed by the Regulatory Staff and supports issuance of the Construction Permit.

The applicant receives an Operating Permit upon NRC review and ASLB approval of the Final Safety Analysis Report (FSAR). The final stage of the approval process of the FSAR is conducted by public hearing similar to the hearing leading to the approval of the PSAR. Guidelines on the preparation of both SAR's are contained in the Regulatory Guide 1.70 entitled "Standard Format and Content of

Safety Analysis Reports for Nuclear Power Plants", Chapter 7 of the SAR's deals with Instrumentation and Controls.

Section 7 of Regulatory Guide 1.70 addresses instrumentation and controls. It provides guidance for the inclusion of instrumentation and controls for the following in the SAR:

- Reactor trip system

- Engineered safety features systems

- Systems required for safe shutdown

- Safety-related display instrumentation

- All other instrumentation systems important for safety

- Control systems not required for safety

Table 7.1-1 contains the acceptance criteria from the standard review plans of Section 7. These acceptance criteria include the applicable general design criteria, IEEE Standards, Regulatory Guides, and branch technical position (BTP) of the Electrical, Instrumentation and Control Systems Branch (EICSB).

The applicability of these criteria to specific paragraphs of Chapter 7 of the SAR is indicated by an "X" in the matrix listing of criteria and SAR chapters

3.1.4 Environmental Reports (ER)

The NRC's procedures implementing the National Environmental Policy Act of 1969 require comprehensive evaluations and assessments of the full range of environmental effects. Consideration is given to both radiological and non-radiological effects of each proposed licensing action in order to arrive at a balance between the benefits to be derived and the environmental cost involved.

3.1.5 General Design Criteria

Appendix A to Part 50 of the Code of Federal Regulations provides a set of general design criteria for nuclear power plants. The scope of criteria include overall requirements for protection and control systems, reactor containment and fuel and radioactivity control.

3.1.6 USNRC Regulatory Guides

Regulatory Guides (formerly called Safety Guides) are issued to describe and make available to the public methods acceptable to the NRC staff for implementing specific parts of the Commission's regulations, to delineate techniques used by the NRC staff in evaluating specific problems for postulated accidents, or to provide guidance to applicants. Regulatory Guides are not substitutes for regulations and compliance with them is not mandatory. Methods and solutions other than those set out in the guides will be acceptable if they provide sufficient basis for findings requisite to the issuance of a construction permit or an operating license by the Commission.

3.1.7 USNRC Instrumentation and Control Safety Branch (ICSB) Positions

The ICSB Branch Technical Positions (BTP's) represent guidelines intended to supplement the acceptance criteria established in Commission regulations and regulatory guides and in applicable IEEE Standards.

3.1.8 Quality Assurance Criteria

Quality assurance criteria are contained in Appendix B, Part 50 in the Code of Federal Regulations. This Appendix establishes quality assurance requirements for the design, construction and operation of those structures, systems and components important to safety.

3.2 Government Standards Organizations

3.2.1 National Bureau of Standards (NBS), U.S. Department of Commerce

The National Bureau of Standards is generally concerned with standards promoting interchangeability of equipment and interchange of information. The most important of these is Handbook 84 entitled Radiation Quantities and Units.

3.2.2 Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (formerly U.S. Atomic Energy Commission, Directorate of Regulatory Standards)

In 1974, congress passed the energy reorganization act which abolished the Atomic Energy Commission and established the Nuclear Regulatory Commission (NRC) and the Energy Research and Development Administration (ERDA).

3.2.3 Division of Reactor Development and Technology (RDT)

RDT is a division of Reactor Development and Demonstration (RDD), Department of Energy (DOE), which was previously Energy Research and Development Administration. Standards developed by RDT were specifically for use in design of government owned reactors and at the present time are not applicable to commercial nuclear facilities. However, these standards will most probably, in the future, be applied in the private sector of the nuclear reactor industry.

3.3 Technical Societies

3.3.1 American National Standards Institute (ANSI)

In 1969, the American National Standards Institute (ANSI) replaced the United States of America Standards Institute (USASI). The latter had been created in 1966 as a successor to the American Standards Association in order to expand the standards program and to accelerate the output of voluntary national standards serving the entire economy.

3.3.2 American Society of Mechanical Engineers (ASME)

The American Society of Mechanical Engineers (ASME) sponsors many technical standards and recommended practices. All of those applicable to nuclear plants have been approved by the American National Standards Institute and now carry ANSI numbers with the notation "sponsored by American Society of Mechanical Engineers".

3.3.3 American Society for Testing and Materials (ASTM)

The American Society for Testing and Materials was founded in 1898 to promote the knowledge of materials of engineering and the standardization of specifications and testing methods.

3.3.4 Institute of Electrical and Electronics Engineers, Inc. (IEEE)

The Institute of Electrical and Electronics Engineers, Inc. (IEEE) was formed through the amalgamation, in 1963 of the American Institute of Electrical Engineers (AIEE) and the Institute of Radio Engineers (IRE).

The IEEE is composed of a number of groups and societies (Nuclear Science Group, Power Engineering Society, Reliability Group, etc.). The generation of reactor instrumentations standards is assigned to a Nuclear Power

Engineering Committee (NPEC). The NPEC has seven subcommittees, five of which are in the process of either developing a new standard or updating an existing standard; an example is subcommittee six which is in the process of updating IEEE 279 revision 1, titled "Criteria for Nuclear Power Generation Station Protection System".

3.3.5 National Fire Protection Association (NFPA)

The NFPA deals primarily in safety and protection regulations. The most important being standard number 70 titled "National Electrical Code".

3.3.6 Instrument Society of America

The Instrument Society of America (ISA) was initially a loose knit confederation of independent local chapters. In 1946, it was formally organized and its present name was adopted. The primary purpose of the ISA is to advance the arts and sciences related to the theory, design, manufacture and use of instrumentation. The society also takes part in the dissemination of information, stimulation of educational activities and development of standards with instrumentation technology. Its nuclear standards activities are now under review of American National Standards Institute Committee N-42.

3.3.7 American Nuclear Society (ANS)

The American Nuclear Society (ANS), established in 1954, is a nonprofit scientific and educational organization made up of some 4600 individual scientists and engineers active in nuclear science and technology. Its main objectives include the advance of science and engineering, the integration of the scientific disciplines, the encouragement of research, and the dissemination of information, all with respect to nuclear science and technology.

3.4 Applicable Documents (Codes, Standards, Guides, etc.) for Nuclear Power Plant Instrumentation and Control Systems

Appendix A to part 50 of Title 10 of CFR provides a set of General Design Criteria for nuclear power plants. The scope of criteria includes overall requirements for protection by multiple fission product barriers, protection and control systems, reactor containment, and fuel and radioactivity control.

Paragraph 50.34 of 10 CFR 50 establishes that the applicant for Construction Permits or Operating Licenses must demonstrate that the facility designed conforms to the requirements of Appendix A,

to part 50, "General Design Criteria for Nuclear Power Plants". These criteria are general and in many cases are expanded upon in certain Regulatory Guides.

The General Design Criteria for nuclear power plants are divided into two introductory sections and a main body as shown in the Table of Contents of 10 CFR 50, Appendix A. The introduction establishes that the General Design Criteria given in Appendix A are the minimum requirements for preliminary and final design. However, it also states that the development of the criteria is not yet complete as "...some of the specific design requirements for structures, systems and components important to safety have not as yet been defined..," and as such, warrants the applicant to also consider all those matters necessary to safety in the design of a specific facility. Safety matters include:

- Consideration of the need to design against single failures with passive components and fluid systems important to safety.

- Consideration of redundancy and diversity requirements for fluid systems important to safety.

- Consideration of the type, size, and orientation of possible breaks and components of the reactor coolant pressure boundary in determining design requirements to suitably protect against postulated loss of coolant accidents.

- Consideration of the possibility of systematic, non-random, concurrent failures of redundant elements in the design of protection systems and reactivity control systems.

The introduction also points out that there may be water cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified. In summary, the introduction establishes Appendix A as a necessary but not always sufficient basis for the design of nuclear power plants.

Following is a list of NRC General Design Criteria (GDC) and brief descriptions of what is contained in that design criteria, as applicable to instrumentation and control systems in a nuclear power plant:

General Criterion

GDC-1 - Quality Standards and Records

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be

performed. Where generally recognized codes and standards are used, they shall be identified. A quality assurance program shall be established. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by the licensee throughout the life of the unit.

GDC-2 - Design Basis for Protection Against Natural Phenomena

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design basis for these structures, systems, and components shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

GDC-3 - Fire Protection

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

GDC-4 - Environmental and Missile Design Bases

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss of coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

GDC-5 - Sharing of Structures, Systems, and Components

Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

Protection by Multiple Fission Product Barriers

GDC-10 - Reactor Design

The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

GDC-12 - Suppression of Reactor Power Oscillations

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

GDC-13 - Instrumentation and Control

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

GDC-15 - Reactor Coolant System Design

The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

GDC-19 - Control Room

A control room shall be provided from which actions can be taken

to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss of coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the Control Room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Protection and Reactivity Control Systems

GDC-20 - Protection System Functions

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences, and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

GDC-21 - Protection System Reliability and Testability

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function, and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

GDC-22 - Protection System Independence

The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in

component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

GDC-23 - Protection System Failure Modes

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

GDC-24 - Separation of Protection and Control Systems

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

GDC-25 - Protection System Requirements for Reactivity Control Malfunctions

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

GDC-26 - Reactivity Control System Redundancy and Capability

Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions or normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

GDC-27 - Combined Reactivity Control Systems Capability

The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods, the capablity to cool the core is maintained.

GDC-28 - Reactivity Limits

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures of other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

GDC-29 - Protection Against Anticipated Operational Occurrences

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

Fluid Systems

GDC-33 - Reactor Coolant Makeup

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for on-site electric power system operation (assuming off-site power is not available) and for off-site electric power system operation (assuming on-site power is not available), the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

GDC-34 - Residual Heat Removal

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat

and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for on-site electric power system operation (assuming off-site power is not available) and for off-site electric power system operation (assuming cn-site power is not available), the system safety function can be accomplished, assuming a single failure.

GDC-35 - Emergency Core Cooling

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented, and (2) clad metal water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for on-site electric power system operation (assuming on-site power is not available), the system safety function can be accomplished, assuming a single failure.

GDC-37 - Testing of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

GDC-38 - Containment Heat Removal

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss of coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment

capabilities shall be provided to assure that for on-site electric power system operation (assuming off-site power is not available) and for off-site electric power system operation (assuming on-site power is not available), the system safety function can be accomplished, assuming a single failure.

GDC-40 - Testing of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

GDC-41 - Containment Atmosphere Cleanup

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for on-site electric power system operation (assuming off-site power is not available) and for off-site electric power system operation (assuming on-site power is not available), its safety function can be accomplished, assuming a single failure.

GDC-43 - Testing of Containment Atmosphere Cleanup Systems

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system,

the transfer between normal and emergency power sources, and the operation of associated systems.

GDC-44 - Cooling Water

A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, lead detection, and isolation capabilities shall be provided to assure that for on-site electric power system operation (assuming off-site power is not available) and for off-site electric power system operation (assuming on-site power is not available), the system safety function can be accomplished, assuming a single failure.

GDC-46 - Testing of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss of coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

Reactor Containment

GDC-50 - Containment Design Basis

The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any loss of coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

GDC-54 - Piping Systems Penetrating Containment

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

GDC-55 - Reactor Coolant Pressure Boundary Penetrating Containment

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability of consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, of the site environs.

GDC-56 - Primary Containment Isolation

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

GDC-57 - Closed System Isolation Valves

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

Fuel and Radioactivity Control

GDC-61 - Fuel Storage and Handling, and Radioactivity Control

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components

important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

GDC-63 - Monitoring Fuel and Waste Storage

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels, and (2) to initiate appropriate safety actions.

GDC-64 - Monitoring Radioactivity Releases

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss of coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

3.5 USNRC Regulatory Guides

In order to amplify the regulation in Title 10 of the Code of Federal Regulations, the NRC has issued Regulatory Guides (formerly called Safety Guides). These guides provide detailed information and guidance for applicants on technical and administrative requirements for the nuclear power industry, and often reference American National Standards Institute (ANSI), Institute of Electrical and Electronic Engineers (IEEE) and American Society of Mechanical Engineers (ASME) standards as providing satisfactory methods of accomplishing objectives.

Regulatory Guides are issued to establish methods acceptable to the NRC staff for implementing specific parts of the Commission's regulations, to delineate techniques used by the staff in evaluating specific problems or postulated accidents or to provide guidance to applicants. Regulatory Guides are not regulations or codes, and compliance with them is not mandatory, but on the other hand, the NRC requires that if an alternate method is used, it be justified in light of the regulations. The ten divisions into which Regulatory Guides are divided are given here for convenience: 1) Power Reactors; 2) Research and Test Reactors; 3) Fuels and Materials Facilities; 4) Environmental and Siting; 5) Materials and Plant Protection; 6) Products; 7) Transportation; 8) Occupational Health; 9) Anti-trust Review; 10) General.

The two divisions that are important to our discussion of nuclear power plants are Power Reactors and Environmental and Siting. The guides published in these areas provide guidance for following the regulations established in 10 CFR 50.34 and 10 CFR 50 Appendix A, The General Design Criteria. Regulatory Guides are written in either three or four sections: A) Introduction; B) Discussion; C) Regulatory Position; D) Implementation (may not be included).

The introduction, the first section, states exactly which part of 10 CFR is being expanded upon by this Regulatory Guide. Often the requirement of several parts of 10 CFR are referenced. The introduction then states that this guide provides an acceptable way of meeting that requirement and that the guide has received concurrence of the Advisory Committee of Reactor Safeguards.

The discussion, the second section, either introduces a recurrent problem that has resulted from applicants attempts to satisfy the requirement referenced in the introduction, or it references a standard or standards that have been developed by the American National Standards Institute or other standards groups to deal with the specific area of nuclear power plant technology.

The third section, implementation, which may or may not be included, discusses the effective date of the guide and whether or not existing plants or components already ordered must comply with the new requirements.

Following is a list of NRC Regulatory Guides, with brief descriptions, that are applicable to instrumentation and control systems in a nuclear power plant:

- Guide No. 1.7 - Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident

  General Design Criterion 35, "Emergency Core Cooling", of Appendix A, "General Design Criteria for Nuclear Power Plants", to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", requires that a system be provided to provide abundant emergency core cooling. Criterion 50, "Containment Design Basis", as amended, requires that the reactor containment structure be designed to accommodate, without exceeding the design leakage rate, conditions that may result from degradation, but not total failure, of emergency core cooling functioning. Criterion 41, "Containment Atmosphere Cleanup", requires that systems to control hydrogen, oxygen, and other substances that may be released into the reactor containment be provided as necessary to control the concentrations of such substances following postulated accidents and ensure that containment integrity is maintained.

Following a loss-of-coolant accident (LOCA), hydrogen gas may accumulate within the containment as a result of

1. Metal-water reaction involving the zirconium fuel cladding and the reactor coolant.

2. Radiolytic decomposition of the postaccident emergency cooling solutions (oxygen will also evolve in this process).

3. Corrosion of metals by solutions used for emergency cooling or containment spray.

If a sufficient amount of hydrogen is generated, it may react with the oxygen present in the containment atmosphere or, in the case of inerted containments, with the oxygen generated following the accident. The reaction could take place at rates rapid enough to lead to high temperatures and significant overpressurization of the containment, which could result in a breaching of containment or a leakage rate above that specified as a limiting condition for operation in the Technical Specifications of the license. Damage to systems and components essential to the conditions could also occur.

In addition, the Commission has published amendments to Part 50 in which a new 50.44, "Standards for Combustible Gas Control Systems in Light-Water-Cooled Power Reactors", was added. This guide describes methods that would be acceptable to the NRC staff for implementing these regulations for light water reactor plants with cylindrical, zircaloy clad oxide fuel. Light water reactor plants with stainless steel cladding and those with non-cylindrical cladding will continue to be considered on an individual basis.

- Guide No. 1.11 - Instrument Lines Penetrating Primary Reactor Containment

General Design Criteria 55 and 56 require that each line that penetrates primary reactor containment and is part of the reactor coolant pressure boundary or that is connected directly to the containment atmosphere, have one automatic valve inside and one automatic valve outside containment, "unless it can be demonstrated that the design is acceptable on some other defined basis". This guide describes a suitable basis which may be used to implement General Design Criteria 55 and 56 for demonstrating the acceptability of a particular group of these lines, namely, instrument lines.

- Guide No. 1.12 - Instrumentation for Earthquakes

Paragraph C of 50.36 of 10 CFR Part 50 provides that the technical specifications will include surveillance requirements

to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met. Appendix A to 10 CFR, Part 100 requires, in paragraph 6, a suitable program for implementing this requirement with regard to seismic instrumentation needed to determine promptly the seismic response of nuclear power plant features important to safety to permit comparison of such response with that used as the design basis. Such a comparison is needed to decide whether the plant can continue to be operated safely. This guide describes seismic instrumentation acceptable to the NRC regulatory staff as satisfying the above stated requirements.

- Guide No. 1.21 - Measuring, Evaluating, and Reporting Radioactivity in Solid Waste and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Coolant Nuclear Power Plants

General Design Criterion 60 of Appendix A of 10 CFR, Part 50, requires that the nuclear power plant design include means to control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid waste produced during normal reactor operation including anticipated operational occurrences.

General Design Criterion 64 requires that nuclear power plant designs provide means for monitoring effluent discharge paths for radioactivity that may be released from normal operation and from postulated accidents.

This guide describes programs acceptable to the NRC staff for measuring, reporting and evaluating releases of radioactive materials in liquid and gaseous effluents and guidelines for classifying and reporting the categories and radiation level of solid waste.

- Guide No. 1.22 - Periodic Testing of Protection System Actuation Functions

General Design Criterion 20 of Appendix A to 10 CFR, Part 50, requires that the protection systems be designed to initiate the operation of systems and components important to safety. General Design Criterion 21 requires that the protection system be designed to permit periodic testing during reactor operation. In current designs, the ability of the protection systems to initiate the operation of safety systems depends on the proper performance of actuation devices. This safety guide describes acceptable methods of including the actuation devices in the periodic test of the protection system during reactor operation.

- Guide No. 1.29 - Seismic Design Classification

General Design Criterion 2 of 10 CFR, Part 50, requires that nuclear power plant structures, systems and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions.

Appendix B to 10 CFR, Part 50, establishes quality assurance requirements for the design, construction, and operation of nuclear power plants structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

Appendix A to 10 CFR, Part 100, requires that all nuclear power plants be designed so that if the Safe Shutdown Earthquake (SSE) occurs, certain structures, systems, and components remain functional.

This guide describes the method acceptable to the NRC staff for identifying and classifying those features of light-water-cooled nuclear power plants that should be designed to withstand the effects of the SSE.

- Guide No. 1.30 - Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment

Appendix B to 10 CFR, Part 50, establishes quality assurance requirements for the design, construction, and operation of nuclear power plant structures, systems, and components.

This safety guide describes an acceptable method of complying with the Commission's regulations with regard to the quality assurance requirements for the installation, inspection, and testing of nuclear power plant instrumentation and electric equipment.

- Guide No. 1.32 - Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants

General Design Criterion 17 requires that an on-site electric power system and an off-site electric power system be provided to permit functioning of structures, systems and components important to safety. In addition, Criterion 17 contains requirements concerning system capacity, independence, redundancy, availability, and reliability. General Design Criterion 18 contains requirements concerning periodic inspection, testing, and testability of electric power systems important to safety.

This guide describes a method acceptable to the NRC staff of complying with Criteria 17 and 18 with respect to the design,

operation, and testing of safety-related electric power systems in all types of nuclear power plants.

- Guide No. 1.38 - Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water Cooled Nuclear Power Plants

Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants", to 10 CFR, Part 50, "Licensing of Production and Utilization Facilities", establishes overall quality assurance requirements for the design, construction, and operation of safety-related structures, systems, and components of nuclear power plants. The American National Standards Institute (ANSI) Standards Committee N45, Reactor Plants and Their Maintenance, has prepared and approved standard N45.2.2-1972 that includes quality assurance requirements for the packaging, shipping, receiving, storage and handling of items for nuclear power plants.

The original issuance of this regulatory guide endorsed as acceptable the guidelines (indicated by the verb "should") as well as the requirements included in ANSI standard N45.2.2-1972. Some uncertainty arose with regard to the NRC staff's intent with this endorsement. As a result of this uncertainty, the staff reevaluated the guidelines contained in ANSI N452.2-1972 with respect to importance to safety. This guide has been revised to clarify NRC's position on the requirements and guidelines included in ANSI N45.2.2-1972. Where conformance to this regulatory guide is indicated in an application without further qualification, this means conformance with the requirements of ANSI N45.2.2-1972, as supplemented or modified by the regulatory position of this guide.

ANSI N45.2.2-1972 does not include the statement that is contained in other N45.2 series standards pertaining to its use for activities covered by the ASME Boiler and Pressure Vessel Code, Section III, Division 1 and 2, and Section XI. The NRC staff's review of the standard indicates that it should be applied to these Code-covered activities.

This guide describes a method acceptable to the NRC staff of complying with the Commission's regulations with regard to the quality assurance requirements for the packaging, shipping, receiving, storage and handling of items for water cooled nuclear power plants. The advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

- Guide No. 1.45 - Reactor Coolant Pressure Boundary Leakage Detection Systems

General Design Criterion 30 of Appendix A to 10 CFR, Part 50, requires that means be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. This guide describes acceptable methods of implementing this requirement with regard to the selection of leakage detection systems for the reactor coolant pressure boundary. This guide applies to light water cooled reactors.

- Guide No. 1.47 - Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems

Criterion 14 of Appendix A to 10 CFR, Part 50, requires that measures be established for indicating the operating status of structures, systems, and components of the nuclear power plant. Section 50.55A of 10 CFR, Part 50, requires that protection systems meet the requirement set forth in the Institute of Electrical and Electronic Engineers "Criteria for Nuclear Power Plant Protection Systems" (IEEE 279).

This guide describes an acceptable method of complying with the requirements of IEEE Standard 279-1971 and Appendix B to 10 CFR, Part 50, with regard to indicating the inoperable status of a portion of the protection system.

- Guide No. 1.53 - Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems

Section 50.55A of 10 CFR, Part 50, requires that protection systems meet the requirement set forth in IEEE 279. This guide describes an acceptable method of complying with the commission's requirements with respect to satisfying the single failure criterion.

- Guide No. 1.62 - Manual Initiation of Protection Actions

Paragraph 50.55A of 10 CFR, Part 50, requires that protection systems meet the requirements set forth in IEEE 279. Section 4.17 of IEEE 279 requires that protection systems include means for manual initiation of each protective action at the system level and that the single failure criterion as set forth in Section 4.2 of IEEE 279 be met. This guide describes a method acceptable to the NRC staff for complying with the requirements of Section 4.17, for including the means for manual initiation of protective actions. This guide applies to all types of nuclear power plants.

Section 4.17 of IEEE 279 includes, among its requirements, the following:

a Manual initiation of each protective action shall be provided at the system level.

b. No single failure shall prevent initiation of protective action.

c. Manual initiation shall depend upon the operation of a minimum of equipment.

- Guide No. 1.68 - Initial Test Programs for Water-Cooled Reactor Power Plants

This guide describes the general scope and depth of initial test programs acceptable to the NRC staff for light water cooled reactor power plants. Appendix A to this guide provides a representative listing of plant structures, systems, components, design features, and performance capability tests that should be demonstrated during the initial test program.

- Guide No. 1.68.1 - Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling-Water Reactor Power Plants

Tests for boiling water reactor (BWR) power conversion systems are described in Regulatory Guide 1.68 to provide assurance that these systems will perform as designed and to aid in minimizing the probability of system malfunctions during subsequent plant operations. This guide describes in more detail the type and nature of BWR feedwater and condensate system tests that are acceptable to the NRC staff.

- Guide No. 1.68.2 - Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants

Regulatory Guide 1.68 describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to preoperational and initial startup testing of nuclear power plant structures, systems, and components.

This guide describes an initial startup test program acceptable to the NRC staff for demonstrating hot standby capability and the potential for cold shutdown from outside the control room. This guide is applicable to water cooled nuclear power plants.

- Guide No. 1.70 - Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants

This guide is discussed in Section 3.1.3 of this text.

- Guide No. 1.73 - Qualification Test of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants

Section 3 of Appendix B to 10 CFR, Part 50, requires that, where a test program is used to verify the adequacy of a specific design feature, it includes suitable qualification testing of a prototype unit under the most adverse design conditions.

This regulatory guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to qualification testing of Class 1 electric valve operators for service within the containment of light water cooled and gas cooled nuclear power plants to assure that the valve operator design will meet its performance requirements.

- Guide No. 1.75 - Physical Independence of Electric System

This guide describes a method acceptable to the NRC staff of complying with IEEE 279-1971 and Criteria 3, 17, and 21 of Appendix A to 10 CFR, Part 50, with respect to the physical independence of the circuits and electrical equipment comprising or associated with the Class 1E power systems, the protection system, the systems actuated or controlled by the protection system, and an auxiliary of supporting systems that must be operable for the protection system and the systems it actuates to perform their safety-related functions. This guide applies to all types of nuclear power plants.

- Guide No. 1.78 - Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release

General Design Criterion 4, "Environmental and Missile Design Bases", of Appendix A "General Design Criteria for Nuclear Power Plants", to 10 CFR Part 50, "Licensing of Production and Utilization Facilities", requires, in part, that structures, systems and components important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Criterion 19, "Control Room", requires that a Control Room be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions. Release of hazardous chemicals can potentially result in the Control Room becoming uninhabitable.

At present, there is no one standard design evaluation method in use for evaluating the habitability of Control Rooms during the course of all postulated hazardous chemical releases. However,

the "Accidental Episode Manual" prepared for the Environmental Protection Agency (EPA) in April, 1972, presents a method for the evaluation and estimation of the area affected by the release of hazardous chemicals as a function of source strength type of chemical, distance from source, and meteorology. The "Accidental Episode Manual" rates accident potentials from both mobile and stationary sources and identifies some hazardous chemicals that may be released. Human tolerance for hazardous chemicals should be considered in the design of nuclear facilities.

For hazardous chemicals shipped on routes near the nuclear power plant, the shipment frequencies specified for consideration in this guide (Regulatory Position 2) reflect the relative accident probabilities for common modes of transportation.

The purpose of this guide is to identify those chemicals which, if present in sufficient quantities, could result in the Control Room becoming uninhabitable. The general design considerations that are used in assessing this guide describes assumptions acceptable to the Regulatory staff to be used in assessing the habitability of the Control Room during and after a postulated external release of hazardous chemicals and describes criteria that are generally acceptable to the Regulatory staff for the protection of the Control Room operators. This guide does not consider the explosion or flammability hazard of these chemicals, which also must be addressed. The advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

The Control Room of a nuclear power plant should be appropriately protected from hazardous chemicals that may be discharged as a result of equipment failures, operator errors, or events and conditions outside the control of the nuclear power plant.

- Guide No. 1.80 - Preoperational Testing of Instrument Air Systems

General Design Criterion 1 of Appendix A, 10 CFR, Part 50, requires that structures, systems, and components important to safety be tested to quality standards commensurate with the importance of the safety functions to be performed. Criterion 11 of Appendix B, to 10 CFR, Part 50, requires that a test program be established to assure that all testing, including preoperational testing required to demonstrate that structures, systems, and components will perform satisfactorily in service, is identified and performed. This guide describes the method acceptable to the NRC staff for complying with the Commission's regulations with respect to verifying the operability of safety-related instrument air systems before placing the systems into service.

- Guide No. 1.81 - Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants

General Design Criterion 5 prohibits structures, systems, and components important to safety from being shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions. These safety functions include the capability to perform an orderly shutdown and cool down of the remaining units in the event of an accident in one unit. This Regulatory Guide describes a method acceptable to the NRC staff for complying with NRC's requirements with respect to the sharing of on-site emergency and shutdown electric systems for multi-unit nuclear power plants.

- Guide No. 1.89 - Qualification of Class 1E Equipment for Nuclear Power Plants

Criterion 3 of Appendix B to 10 CFR, Part 50, requires that design control measures provide for verifying the adequacy of a specific design feature by design reviews, by calculational methods, or by suitable qualification testing of a prototype unit under the most adverse conditions. This Regulatory Guide describes the method acceptable to the NRC staff for complying with the Commission's regulations with regard to design verification of Class 1E equipment for service in light-water-cooled and gas cooled nuclear power plants.

- Guide No. 1.96 - Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants

General Design Criterion 54, "Piping Systems Penetrating Containment", of Appendix A, "General Design Criteria", to 10 CFR Part 50, "Licensing of Production and Utilization Facilities", requires in part, that piping systems penetrating primary containment be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities that reflect the importance to safety of isolating these piping systems.

Direct cycle boiling water nuclear power plants supply steam directly from the reactor vessel to the turbine via main steam lines. The main steam lines installed on current BWR plants are provided with dual quick closing isolation valves. These valves function to isolate the reactor system in the event of a break in a steam line outside the primary containment, a design basis LOCA, or other events requiring containment isolation. In the case of a steam line break, the isolation valves would terminate the blowdown of reactor coolant in sufficient time to prevent an

uncontrolled release of radioactivity from the reactor vessel to the environment. In the case of a LOCA, the valves would isolate the reactor from the environment and prevent the direct release of fission products from the containment.

The valves are part of the reactor coolant pressure boundary. As such, they are Quality Group A components and their integrity must be maintained by strict inservice inspection and testing requirements. However, operating experience has indicated that degradation has occasionally occurred in the leak tightness of main steam isolation valves, and the specified low leakage requirements have not always been maintained.

The staff has considered the need to provide additional features to ensure the low-leakage characteristics of the main steam isolation valves in the event of a postulated design basis loss-of-coolant accident. The use of a leakage control system would reduce direct untreated leakage from the isolation valves when isolation of the primary system and the containment is required.

The results of staff analyses have indicated calculated doses resulting from the maximum leakage.

This guide describes a basis acceptable to the NRC staff for implementing General Design Criterion 54 with regard to the design of a leakage control system for the main steam isolation valves of boiling water reactor (BWR) nuclear power plants to ensure that total site radiological effects do not exceed guidelines of 10 CFR Part 100, "Reactor Site Criteria", in the event of a postulated design-basis loss-of-coolant accident (LOCA).

- Guide No. 1.97 - Instrumentation for Light-Water-Cooled Nuclear Power Plants to Access Plant Conditions During and Following an Accident

General Design Criterion 13, "Instrumentation and Control", of Appendix A, "General Design Criteria for Nuclear Power Plants", to 10 CFR Part 50, "Licensing of Production and Utilization Facilities", includes a requirement that instrumentation be provided to monitor variables and systems for accident conditions as appropriate to ensure adequate safety.

General Design Criterion 19, "Control Room", of Appendix A to 10 CFR, Part 50, includes a requirement that a control room be provided from which actions can be taken to maintain the nuclear power unit in a safe condition under accident conditions, including loss of coolant accidents.

General Design Criterion 64, "Monitoring Radioactivity Releases", of Appendix A to 10 CFR, Part 50, includes a

requirement that means be provided for monitoring the reactor containment atmosphere spaces containing components for recirculation of loss-of-coolant accident fluid, effluent discharge paths, and the plant environs for radioactivity that may be released from postulated accidents.

Monitored variables and systems are used by the operator in accident surveillance to (1) assist in determining the nature of an accident; (2) determine whether the reactor trip and engineered safety feature systems are functioning properly; (3) determine whether the plant is responding properly to the safety measures in operation; (4) provide information to the operator that will enable him to determine the potential for breaching the barriers to radioactivity release; (5) furnish data for deciding on the need to take manual action if an engineered safety feature malfunctions or the plant is not responding effectively to the safety systems in operation; (6) allow for early indication of the need to initiate action necessary to protect the public and for an estimate of the magnitude of the impending threat; and (7) aid in determining the cause and consequence of the event for postaccident investigation.

At the start of an accident, the operator cannot always determine immediately what accident has occurred or is occuring and therefore cannot always determine the appropriate response. For this reason, the reactor trip and certain safety actions (e.g., emergency core cooling actuation, containment isolation, or depressurization) are designed to be performed automatically during the initial stages of an accident. Instrumentation is also provided to indicate information about plant parameters required to enable the operation of manually initiated safety-related systems and other appropriate operator actions.

Examples of serious events that threaten safety if conditions degrade beyond those assumed in the Final Safety Analysis Report are loss-of-coolant accidents (LOCA's), reactivity excursions, and radioactivity releases. Such events require that the operator understand, in a short time period, the state of readiness of engineered safety features and their potential for being challenged by an accident in progress.

This guide describes a method acceptable to the NRC staff for complying with the Commission's requirements to provide instrumentation to monitor plant variables and systems during and following an accident in a light water cooled nuclear power plant.

- Guide No. 1.100 - Seismic Qualifications of Electric Equipment for Nuclear Power Plants

Criterion 3 of Appendix B to 10 CFR, Part 50, requires among other things, that design control measures provide for

verifying the adequacy of design such as by the performance of a suitable testing program. Where a test program is used to verify the adequacy of a specific design feature, it is required to include a suitable qualification testing of a prototype unit under the most adverse design conditions. This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with respect to verifying the adequacy of the seismic design of electric equipment for all kinds of nuclear power plants.

- Guide No. 1.105 - Instrument Setpoints

  Criterion 13 of 10 CFR, Part 50, requires among other things, that instrumentation be provided to monitor variables and systems and that controls be provided to maintain these variables in systems within prescribed operating ranges.

  This guide describes a method acceptable to the NRC staff for complying to the Commission's regulations with regard to insuring that the instrument setpoints in systems important to safety initially are within and remain within the specified limits.

- Guide No. 1.106 - Thermal Overload and Protection for Electric Motors on Motor Operated Valves

  This regulatory guide describes the method acceptable to the NRC staff for complying with criteria in regard to the application of thermal overload protection devices that are integral with the motor starter for electric motors on motor operated valves. This method would insure that the thermal overload protection devices will not needlessly prevent the motor from performing its safety-related function. This satisfies Criteria 1, 4, and 13 of 10 CFR, Part 50.

- Guide No. 1.108 - Periodic Testing of Diesel Generator Units Used as On-Site Electric Power Systems at Nuclear Power Plants

  Criterion 11 of 10 CFR, Part 50, requires that a test program be established to ensure that systems and components perform satisfactorily and that the test program include operational tests during nuclear power plant operation.

  Criterion 17 of 10 CFR, Part 50, requires that on-site electric power systems have sufficient independence, capacity, redundancy, and testability to perform their safety functions, assuming a single failure.

  Criterion 18 of 10 CFR, Part 50, requires that electric power systems important to safety be designed to permit appropriate

periodic inspection and testing to assess the continuity of the systems and the conditions of their components.

This Regulatory Guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to periodic testing of diesel electric power units to ensure that the diesel electric power systems will meet their availability requirements.

- Guide No. 1.118 - Periodic Testing of Electric Power and Protection Systems

This guide describes a method acceptable to the NRC staff of complying with the Commission's regulations with respect to the periodic testing of the protection system and electric power system for systems important to safety. It also provides supplementary guidance to that included in Regulatory Guide 1.32 regarding the periodic testing of electric power systems.

- Guide No. 1.120 - Fire Protection Guidelines for Nuclear Power Plants

General Design Criterion 3 of 10 CFR 50, requires that structures, systems and components important to safety be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Non-combustible and heat resistant materials are required to be used wherever practical throughout the unit. Criterion 3 also requires that adequate fire detection and suppression systems be provided.

This guide presents guidelines acceptable to the NRC staff for implementing this criterion in the development of a fire protection program for nuclear power plants. The purpose of the fire protection program is to ensure the capability to shut down the reactor and maintain it in a safe shutdown condition and to minimize radioactive releases to the environment in the event of a fire. It implements the philosophy of defense in-depth protection agiarst the hazards of fire and its associated effects on safety-related equipment. This guide also supplements Regulatory Guide 1.75, entitled "Physical Independence of Electrical Systems", in determining the fire protection for redundant cable systems.

- Guide 1.139 - Guidance for Residual Heat Removal

General Design Criterion 19, "Control Room", of Appendix A, "General Design Criteria for Nuclear Power Plants", to 10 CFR Part 50, "Licensing of Production and Utilization Facilities", requires that it be possible to take actions from the control

room to maintain the power plant in a safe condition during normal operation or in the case of an accident. GDC-34, "Residual Heat Removal", requires that a system to remove residual heat be provided. GDC-34 defines the system's safety function as the transfer of fission product decay heat and other residual heat from the reactor core after the reactor is shut down so that acceptable design limits of the fuel and the reactor coolant pressure boundary are not exceeded. Furthermore, GDC-34 requires that the system safety function can be accomplished assuming the availability of only on-site or off-site power, coincident with a single failure.

Isolation of the suction side of the RHR system should be provided by at least two power operated valves in series, with valve positions indicated in the Control Room. Alarms in the Control Room should be provided to alert the operator if either valve is open when the RCS pressure exceeds the RHR system design pressure. The valves should have independent diverse interlocks to prevent the valves from being opened unless the RCS pressure is below the RHR system design pressure. Failure of a power supply should not cause any valve to change position. Independent diverse protective measures should be provided to close any open valve in the event of an increase in the RCS pressure above the RHR system design pressure.

One of the following should be provided on the discharge side of the RHR system to isolate it from the RCS:

1. The valves, position indicators, alarms, and interlocks in item a.

2. One or more check valves in series with a normally closed power operated valve with its position indicated in the Control Room. If the RHR system discharge line is used for an ECCS function, the power operated valve should be opened upon receipt of a safety injection signal once the reactor coolant pressure has decreased below the ECCS design pressure.

3. Three check valves in series, or

4. Two check valves in series, provided there are design provisions to permit periodic testing of the check valves for leak tightness and the testing is performed at least annually.

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to the removal of decay heat and sensible heat, after reactor shutdown.

- Guide No. 1.141 - Containment Isolation Provisions for Fluid Systems

General Design Criteria 54, 55, 56, and 57 of Appendix A, "General Design Criteria for Nuclear Power Plants", to 10 CFR Part 50, "Licensing of Production and Utilization Facilities", require that piping systems penetrating primary reactor containment be provided with isolation capabilities that reflect the importance to safety of isolating these piping systems. This guide describes a method acceptable to the NRC staff for complying with the Commission's requirements with respect to containment isolation of fluid systems.

The American Nuclear Society Standards Committee ANS-50, Nuclear Power Plant Systems Engineering, has prepared a standard which specifies the minimum design requirements for containment isolation of fluid systems that penetrate the primary containment boundary of light water cooled reactors. This standard was approved by the American National Standards Institute (ANSI) Committee N18, Design Criteria for Nuclear Power Plants, and designated ANSI N271-1976, "Containment Isolation Provisions for Fluid Systems".

The provisions of ANSI N271-1976 include minimum design, testing, and maintenance requirements for the isolation of fluid systems that penetrate the primary containment of light water cooled reactors. Requirements for the design and testing of power supplies, qualifying of Class 1E equipment, and the design and testing of protection systems are outside the scope of this standard. These areas are not completely covered by the references given in ANSI N271-1976.

- Guide No. 4.1 - Programs for Monitoring Radioactivity in the Environs of a Nuclear Power Plant

This guide describes the basis acceptable to the NRC staff for the design of programs for monitoring levels of radiation and radioactivity in the plant environs.

- Guide No. 4.2 - Preparation of Environmental Reports for Nuclear Power Stations

Prior to the issuance of a Construction Permit or an Operating License for a nuclear power plant, the NRC is required to assess the potential environmental effects of the plant in order to assure that issuance of the permit or license will be consistent with the National Environmental goals, as set forth by the National Environmental Policy Act of 1969. In order to obtain information essential to this assessment, the Commission requires each applicant for a permit or a license to submit a report on potential environmental impacts of the proposed plant associated facilities.

This guide contains information pertaining to natural environmental goals and details for applicants' reports in compliance with regulations. This guide also outlines in detail the purpose of the proposed facility, the site, the plant, environmental effects of site preparation, plant and transmission facilities, construction, environmental effects of plant operation, effluent and environmental measurements and monitoring programs, environmental effects of accidents, economic and social effects of plant construction and operation, alternative energy sources and sites, and plant design alternatives.

- Guide No. 8.2 - Guide for Administrative Practices in Radiation Monitoring

In 10 CFR, Part 20, it requires each licensee to make, or cause to be made, surveys to evaluate radiation hazards and to supply and require the use of appropriate personnel monitoring equipment. Part 20 also sets limits on the exposure of individuals to radiation and on release of radioactive effluents in nonrestricted areas and prescribed procedures for the disposal of radioactive wastes. Section 20.401 requires each licensee to maintain records of surveys, waste disposals, and the radiation exposures of all individuals for whom personnel monitoring was required under 20.202. This guide provides general information on radiation monitoring programs for administrative personnel.

3.5.1 USNRC Branch Technical Positions

The Instrumentation Control Safety Branch (ICSB), Branch Technical Positions (BTP's), represent guidelines intended to supplement the acceptance criteria established in Commission regulations and regulatory guides, and in applicable IEEE standards. The BTP's originate in technical problems or questions of interpretation that arise in the detailed reviews of plant designs. The staff must make a judgement in each such case in order to complete its review of the particular application. Where the same technical problem or question of interpretation arises in several cases, the staff's judgement on the point at issue is formalized in a BTP. The BTP is primarily an instruction to staff reviewers that outlines an acceptable approach to the particular issue and ensures a uniform treatment of the issue by staff reviewers. The approaches taken in the BTP's, like the recommendations of regulatory guides, are not mandatory, but do provide defined, acceptable, and immediate solutions to some of the technical problems and questions of interpretation that arise in the review process. Although the BTP is primarily an instruction to staff reviewers and not to NRC inspectors, all ICSB BTP's applicable to the Standard Review Plan (SRP) sections in

Chapter 7 have been collected in this Appendix for completeness and convenience:

Branch Technical Positions	Title
BTP ICSB 1	Backfitting of the Protection and Emergency Power Systems of Nuclear Reactors
BTP ICSB 3	Isolation of Low Pressure System from the High Pressure Reactor Coolant System
BTP ICSB 4	Requirements on Motor-Operated Valves in the ECCS Accumulator Lines
BTP ICSB 5	Scram Breaker Test Requirements - Technical Specifications
BTP ICSB 9	Definition and Use of "Channel Calibration" - Technical Specifications
BTP ICSB 10	Electrical and Mechanical Equipment Seismic Qualification Program
BTP ICSB 12	Protection System Trip Point Changes for Operation with Reactor Coolant Pumps Out of Service
BTP ICSB 13	Design Criteria for Auxiliary Feedwater Systems
BTP ICSB 14	Spurious Withdrawals of Single Control Rods in Pressurized Water Reactors
BTP ICSB 15	Reactor Coolant Pump Breaker Qualification
BTP ICSB 16	Control Element Assembly (CEA) Interlocks in Combustion Engineering Reactors
BTP ICSB 18	Application of the Single Failure Criterion to Manually Controlled Electrically-Operated Valves

BTP ICSB 19	Acceptability of Design Criteria for Hydrogen Mixing and Drywell Vacuum Relief Systems
BTP ICSB 20	Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recirculation Mode
BTP ICSB 21	Guidance for Application of Reg. Guide 1.47
BTP ICSB 22	Guidance for Application of Reg. Guide 1.22
BTP ICSB 23	Qualification of Safety-Related Display Instrumentation for Post-Accident Condition Monitoring and Safe Shutdown
BTP ICSB 24	Testing of Reactor Trip System and Engineered Safety Feature Actuation System Sensor Response Time
BTP ICSB 25	Guidance for the Interpretation of General Design Criterion 37 for Testing the Operability of the Emergency Core Cooling System as a Whole
BTP ICSB 26	Requirements for Reactor Protection System Anticipatory Trips
BTP ICSB 27	Design Criteria for Thermal Overload Protection for Motors or Motor-Operated Valves

3.6 Technical Society Codes and Standards

Generation of standards, particularly in reactor instrumentation, must be viable to meet the needs of the nuclear power industry. In the last few years, reactor instrumentation standards have undergone intense development. Since this activity will continue, the handbook or lesson plan providing only tabulation of today's standards would be incomplete. A knowledge of the identities of

organizations active in a reactor instrumentation standardization is necessary to obtain the latest information either from the publications of those organizations or by direct communication with them.

3.6.1 American National Standards Institute

The American National Standards Institute (ANSI) is a coordinating agency which compiles standards generated by various professional organizations throughout the United States. Those standards written by various organizations, are initially published as drafts for trial use and comment by the originating agency or committee. After all comments have been compiled and the proposed standard has been revised, it is published for industrial use by ANSI. Nuclear-related standards are published as ANSI N series standards, and are authored for the most part by the American Nuclear Society (ANS).

A set of nuclear-related standards approved by ANSI, originated by the Institute of Electrical and Electronics Engineers (IEEE), but retaining the IEEE serial numbers, are also listed n the ANSI catalog.

The American National Standards Institute has endorsed other nuclear-related standards by endorsing the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code - 1974 and subsequent addenda.

Approval of ANSI standards continues to be based on consensus of all parties concerned. In continuous liaison and cooperative work with hundreds of national trade associations, technical, professional and scientific societies, the Institute encourages the extension of this consensus principle. The Institute has a vital role in surveying nuclear standards needs, in initiating and scoping standards projects, in coordinating industry efforts, in eliminating duplication of efforts, and in setting priorities. The Institute is the focus of United States activities in the international standards division.

The Nuclear Standards Board of ANSI coordinates the work of the committees concerned with nuclear standards. The work of the following committees is of interest to reactor instrumentation engineers: Committee N-41, Controls, Instrumentation, and Electrical Systems for Nuclear Power Generating Stations; Committee N-42, Nuclear Instruments; Committee N-45, Nuclear Plants and Their Maintenance; Committee N-101, Atomic Industry Facility Design, Construction and Operation Criteria.

Following is a compilation, with brief descriptions, of the scope, purpose and function of ANSI approved standards that are applicable to instrumentation and control systems in nuclear power plants:

- ANSI - B31.1-77 - Power Piping (Sponsored by ASME)

  The general philosophy underlying this power piping code is to parallel those provisions of Section No. 1 - Power Boilers of the ASME Boiler and Pressure Vessel Code, as they can be applied to power piping systems. The allowable stress values for power piping are generally consistent with those designed for power boilers. This code is more conservative than some other piping codes reflecting the need for long service life and maximum reliability in power plant installations.

  The code for pressure piping (ANSI B31) consists of a number of sections, which collectively constitute the code. The code for pressure piping sets forth engineering requirements deemed necessary for safe design and construction of piping systems. While safety is the basic consideration of this code, this factor alone does not necessarily govern the final specifications for any pressure piping system. The code is not a design handbook. The code does not do away with the need for the engineer or competent engineering judgement.

  The code includes:

  1. Material specifications and component standards which have been accepted for code usage.

  2. The designation of proper dimensional standards for the elements comprising piping systems.

  3. Requirements for the design of component parts and assembled units, including necessary pipe supporting elements.

  4. Requirements for the evaluation and limitation of stresses, reactors and movements associated with pressure, temperature and external forces.

  5. Requirements for the fabrication, assembly and erection of piping systems.

  6. Requirements for testing and inspecting of elements before assembly or erection and of the completed systems after erection.

The sections of this code that are important to an instrumentation and control engineer are Chapter 1 and Chapter 2 which include Scope, Definitions, General Conditions and General Design Criteria. In addition, in Chapter 2 Design Part 6, Systems, Part 122.3 specifically addresses "Instrument, Control and Sampling Piping".

- ANSI - B31.7-69 - Nuclear Power Piping (Sponsored by ASME)

In 1963, Section 3, Nuclear Vessels (later revised to "Nuclear Power Plant Components"), of the ASME Boiler and Pressure Vessel Code was first issued. The Nuclear Vessels Code is intended to encompass the rigorous requirements for reliability necessary in the design and construction of nuclear pressure vessels. This Nuclear Power Piping section of the Piping Code parallels the requirements of the ASME Nuclear Vessels Code, again reflecting similarities in requirements for nuclear vessels and the associated piping.

This Nuclear Power Piping Code provides requirements for the design, materials, fabrication, assembly, erection, installation and examination of three classes of nuclear piping systems for nuclear power plant service. These three classes denote three levels of quality, and the designer shall choose the class that will provide the quality required for the intended service. The important points to cover in this section would be parts of Chapter 1 and 2 including Scope and Definitions and General Design Criteria, and also Chapter 1-722 entitled Design Requirements Pertaining to Specific Piping Systems (including instrument piping).

- ANSI - N1.1-76 - Glossary of Terms in Nuclear Science and Technology (Sponsored by American Nuclear Society)

- ANSI - N13.10-74 - Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity and Effluents (Sponsored by IEEE)

The release of radioactivity from nuclear facilities to the environment generally is monitored by installed instrumentation. The objective of such instrumentation is to measure the quantity or rate, or both, of release of radionuclides in the effluent streams and to provide documentation useful for scientific and legal purposes. This standard applies to continuous monitors that

measure normal releases, detect inadvertent releases, show general trends and annunciate radiation levels that have exceeded predetermined values.

- ANSI - N45.2.2-72 - Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plants (Construction Phase) (Sponsored by ASME)

This standard defines requirements for packaging, shipping, receiving, storage and handling of nuclear power plant items. These items include the parts of structures, systems and components whose satisfactory performance is required for the plant to operate reliably, to prevent accidents that could cause undue risk to the health and safety of the public, or to mitigate the consequences of such accidents if they were to occur. The requirements stated deal with the protection and control necessary to assure that the requisite quality of those important parts of the plant are preserved from the time items are fabricated until they are incorporated in the plant. This standard is intended to be used in conjunction with ANSI N45.2.

- ANSI - N45.2.3-73 - Housekeeping During Construction Phase of Nuclear Power Plants (Sponsored by ASME)

This standard defines the housekeeping requirements for the control work activities, conditions and environment that can affect the quality of important parts of a nuclear power plant during the construction phase. These parts include the structures, systems and components whose satisfactory performance is required for the plant to operate reliably to prevent accidents that cause undue risk to the health and safety of the public, or to mitigate the consequences of such accidents if they were to occur. Housekeeping encompasses all activities related to control of cleanness of facilities, cleanness of material and equipment, fire prevention and fire protection including disposal of combustible materials and debris, control of access and protection of equipment not denoted in other standards. The requirements may also be extended to other appropriate parts of nuclear power plants when specified in contract documents. This standard is intended to be used in conjunction with ANSI N45.2.

- ANSI - N45.2.6-73 - Qualifications of Inspectors, Examination and Testing Personnel for Construction Phase of Nuclear Power Plants (Sponsored by ASME)

This standard delineates the qualifications of personnel who perform inspection, examination and testing activities that assure the quality of important parts of a nuclear power plant during the construction phase. These parts include those structures, systems and components whose satisfactory performance is required for the plant to operate reliably, to prevent accidents that could cause undue risk to the health and safety of the public, or to mitigate the consequences of such accidents if they were to occur.

The requirements of this standard apply to personnel who perform inspections, tests, or nondestructive examinations; or who participate in the approval of procedures, the handling of data or test results, of the control of reports and records. This standard is intended to be used in conjunction with ANSI N45.2.

- ANSI - N45.2.11-74 - Quality Assurance Requirements for the Design of Nuclear Power Plants

This standard provides requirements and guidance for quality assurance programs for the design of nuclear power plant structures, systems and components whose satisfactory and reliable performance is required to prevent accidents that could cause undue risk to the health and safety of the public, or to mitigate the consequences of such accidents if they were to occur. The requirements of this standard may also be extended to other structures, systems and components in whole or in part as specified by the purchaser. This standard covers activities which affect the final design. This standard is intended to be used in conjunction with ANSI N45.2.

- ANSI - N45.2.13-76 - Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants

This standard describes requirements and provides guidelines for the control of activities to be exercised during procurement of items and services such as designing, purchasing, fabricating, handling, shipping, storing, cleaning, constructing, erecting, installing, inspecting, testing, maintaining, repairing, initial fueling,

refueling and modifying. This standard provides guidelines for application of quality assurance program requirements listed in ANSI N45.2 for various types of procurement.

- ANSI - N278.1-75 - Self-Operated and Power-Operated Safety-Related Valves Functional Specification Standard (Sponsored by ASME)

This standard established requirements for functional specifications for safety-related, and self-operated and power-operated valves or applications in nuclear power plants.

3.6.2 American Society of Mechanical Engineers (ASME)

Many applicable ASME standards have been approved by the American National Standards Institute (ANSI) and are listed above in Section 5.1.

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code has been endorsed by the American National Standards Institute. The ASME Boiler and Pressure Vessel Code is a tri-annual publication of the ASME's Boiler and Pressure Vessel Committee. The committee's function is "to establish rules of safety governing the design, the fabrication, the installation and the inspection during construction of boilers, pressure vessels and nuclear power plant components such as pressure vessels, piping, pumps and valves". The committee formulates these rules considering the needs of users, manufacturers and inspectors.

The code is not to be interpreted as an approval or recommendation of any specific design. The manufacturer has the freedom to use any design he desires as long as it conforms to the rules of the code.

The code is interpreted for particular questions by the Boiler and Pressure Vessel Committee and interpretations may be obtained by submitting written inquiries. An interpretation is effective only after confirmation by letter ballot of the committee and approval by the council of the ASME.

Interpretations which are of interest to the public are published in "Mechanical Engineering" (an ASME publication). In addition, "Mechanical Engineering" also contains proposed revisions to the code, published for

comment by the public. Semi-annually, these revisions and interpretations, if approved, are published in a Summer Addenda and Winter Addenda. Every three years, the addenda are incorporated into a new edition of the code.

Section 3 of the ASME Code is titled "Nuclear Power Plant Components". This section establishes "requirements for the construction of nuclear power plant components and appurtenances such as vessels, storage tanks, piping, pumps, valves and core support structures for use in, or containment of portions of, a nuclear power system of any power plant. A nuclear power system is that system which serves the purpose of producing and controlling an output of thermal energy from nuclear fuel and those associated systems essential to the functions and overall safety of the nuclear power system".

Nuclear power system components are defined as those components which are designed to provide a pressure containing barrier or to act as a pressure retaining member in the nuclear power system or to support the reactor core. Containment systems are defined as those components which are designed to provide a pressure containing barrier for the primary purpose of containing within leakage limits, or of channeling for containment of controlled disposal, of radioactive or hazardous effluents released from the nuclear power system components.

The Code is not applicable to valve operators, controllers, position indicators, pump impellers, pump drives or other accessories and devices, unless such items themselves are pressure retaining parts or act as core support structures. In addition, the Code is not applicable to "balance of plant" structures, systems or components.

Those components and systems that are covered by Section 3 of the Boiler and Pressure Vessel Code are classified in four separate categories to define the requirements for quality for the system. The basis for this establishment of categories is the relative importance to safety of the system or component considered. The quality requirements for the components and systems associated with each category are discussed in Section 3 of the Code. The four quality groups, as they are called, are listed as follows:

Class 1 - Components constructed in accordance with the rules of subsection NB of Section 3 of the Code.

Class MC - Metal containment vessels constructed in accordance with the rules of subsection NE of Section 3 of the Code.

Class 2 - Components constructed in accordance with the rules of subsection NC of Section 3 of the Code.

Class 3 - Components constructed in accordance with the rules of subsection ND of Section 3 of the Code.

The particulars of each group as discussed in Section 3 of the Code are too detailed to demonstrate the difference between groups. However, it is sufficient to say the Class 1 components are subject to extremely strict quality control requirements. Class 2 somewhat less strict requirements, and so forth. However, the code does not discuss the detailed basis for assigning classification to components or systems.

The basis for classifying structures, systems and components is partly established in 10 CFR 50.55A, Codes and Standards. This section requires, as discussed earlier, that components of the reactor coolant pressure boundary be designed, fabricated, erected and tested in accordance with the requirements for Class 1 components of Section 3 of the Code or equivalent quality standards. However, there are other systems which are important to safety which are not part of the reactor coolant pressure boundary. Thus, a further basis for a classification system is developed in Regulatory Guide 1.26, Quality Group Classifications and Standards. This guide describes a quality classification system related to specified industry codes, including the Boiler and Pressure Vessel Code, that may be used to determine quality standards that satisfy General Design Criteria 1 for other water and steam containing components important to safety of water cooled nuclear power plants.

Regulatory Guide 1.26 establishes quality Groups A, B, C and D which correspond to the classes of Section 3 of the Boiler and Pressure Vessel Code as follows:

Quality Group A - Class 1

Quality Group B - Class 2 and Class MC

Quality Group C - Class 3

Quality Group D - Other sections of the Boiler and Pressure Vessel Code and various other standards.

Another classification system has been established by the American Nuclear Society and ANSI N-18.2, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants. Structures and components are classified as Safety Class 1, Safety Class 2 or Safety Class 3, in accordance with their importance to nuclear safety. The three ANS safety classifications are based upon the consequence of failure, while the three classes of the ASME code are based on probability of failure. The ANS classification compares with the ASME as follows:

ASME	ANS
Class 1	Class 1
Class MC	Class 2A
Class 2	Class 2A
Class 3	Class 2B and Class 3

ANS Safety Class 1 applies to reactor coolant system components whose failure could cause a condition 3 or condition 4 loss of reactor coolant accidents are those from a small ruptured pipe or from a crack in a large pipe which could prevent orderly reactor shutdown and cooldown, assuming makeup is provided by normal makeup systems only. Condition 4 occurrences are the major rupture of a pipe containing reactor coolant. ANS Safety Class 2 applies to those structures and components of safety systems required to fulfill a system safety function. Safety Class 2 applies to those structures and components of safety systems required to fulfill a system safety function. Safety Class 2A applies to containment and to components of those safety systems, or portions thereof, through which reactor coolant water flows directly from the reactor coolant system or containment sump. Safety Class 2B applies to all other components of Safety Class 2. Safety Class 3 applies to components not in Safety Class 1 or 2, the failure of which would result in release to the environment of gaseous radioactivity normally held up.

As a result of the various requirements and guides, it becomes the responsibility of the owner of the nuclear power plant to classify the components of this plant. This classification system must appear in the FSAR as

suggested in Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. This results in the items being classified as "safety-related" or "non-safety-related". For the purposes of the NRC, all safety-related structures, systems and components are subject to the requirements of 10 CFR 50, Appendix B, Quality Assurance Requirements for Nuclear Power Plants, regardless of their relative importance. In relationship to the classification system given in Regulatory Guide 1.26, all items classified in quality groups A, B and C are considered safety-related.

3.6.3 American Society for Testing and Materials (ASTM)

The American Society for Testing and Materials (ASTM) was founded in 1898 to promote the knowledge of materials of engineering and the standardization of specifications in testing methods. To keep pace with technological advances, the ASTM sponsors more than 125 research projects and has 115 technical committees. The ASTM has developed over 4200 standard test methods, specifications and recommended practices now in use. Following is some of the applicable standards for instrumentation and control systems in a nuclear power plant:

- ASTM - B68-76 - Seamless Copper Tubing, Bright Annealed

  This specification covers annealed seamless copper tube suitable for use where tubing must be absolutely free from scale and dirt.

3.6.4 Institute of Electrical and Electronics Engineers, Inc., (IEEE)

In 1963, the Institute of Electrical and Electronics Engineers, Inc., (IEEE) was formed through the merger of two organizations, the American Institute of Electrical Engineers (AIEE) and the Institute of Radio Engineers (IRE). The present membership of the IEEE is over 156,000 and includes engineers and scientists in electrical engineering, electronics and allied fields, as well as about 30,000 students.

The IEEE, being composed of a number of groups and societies (i.e., Nuclear Science Groups, Power Engineering Societies, Reliability Groups, etc.), has assigned the generation of reactor instrumentation standards to the Nuclear Power Engineering Committee (NPEC). The NPEC has seven subcommittees, five of which are action in the process of either developing a new standard or updating

existing standards. Following is a list of IEEE standards applicable to instrumentation and control systems in a nuclear power plant:

- IEEE 91-73 - Graphic Symbols for Logic Diagrams Two-State Devices

  This standard established the graphic symbols for use in the preparation of logic diagrams representing logic functions implemented with two-state devices. The primary purpose of this standard is to enable the user to readily read and understand the function of the logic without requiring a specific knowledge of the constructional details of the device represented.

  This standard sets forth principles governing the formation of graphic symbols for logic diagrams in which the connections between symbols are generally shown with lines. Descriptions of logic functions, the graphic representation of these functions, and examples of their applications are given.

- IEEE 279-71 - Criteria for Protection Systems for Nuclear Power Generating Stations

  These criteria establish minimum requirements for safety-related functional performance and reliability protection systems for stationary, land bases nuclear reactors producing steam for electric power generation. Fulfillment of these requirements does not necessarily fully establish the adequacy of protective system functional performance and reliability. On the other hand, omission of any of these requirements will, in most instances, be an indication of system inadequacy. For purposes of these criteria, the nuclear power generating station protection system encompasses all electrical and mechanical devices and circuitry (from sensors to actuation device input terminals) involved in generating those signals associated with the protective function. These signals include those that actuate reactor trip and that, in the event of a serious reactor accident, actuate engineered safeguards such as containment isolation, core spray, safety injection, pressure reduction and air cleaning.

This standard is a generic type of the nuclear power plants related IEEE standard. Many IEEE standards were developed to address a specific section of the IEEE 279. The title of this standard suggests its applicability only to the protection systems. However, because of the generic nature of this standard, NRC frequently requests the

compliance with IEEE 279-1971 in whole or in part in the design of engineered safeguard systems and other safety-related systems.

To help the NRC inspector, the following is a cross referencing of some IEEE 279-1971 sections requirements with other IEEE standards, general design criteria of the 10 CFR 50 (Appendix A), NRC Regulatory Guides and NRC branch technical positions pertinent for the instrumentation and control.

- IEEE 279-1971 - Automatic Initiation (Section 4.1)

  IEEE 308
  RG 1.32
  GDC-20

  Single Failure Criterion (Section 4.2)

  IEEE 308
  RG 1.53
  GDC-21
  GDC-23
  EICSB 1
  EICSB 18

  Quality of Components and Module (Section 4.3)

  RG 1.30

  Equipment Qualification (Section 4.4)

  IEEE 323
  GDC-21

  Channel Integrity (Section 4.5)

  IEEE 344
  IEEE 308
  RG 1.29
  RG 1.100
  EICSB 10

  Channel Independence (Section 4.6)

  IEEE 308
  IEEE 384
  RG 1.6
  RG 1.75
  GDC-22

Control and Protection Interaction (Section 4.7)

GDC-24

Capability for Test and Calibration (Section 4.10)

IEEE 336
IEEE 338
RG 1.22
RG 1.118
GDC-21
GDC-40
EICSB 9
EICSB 22

Indication of Bypasses (Section 4.13)

RG 1.47
EICSB 21

Manual Initiation (Section 4.17)

IEEE 308
RG 1.32
RG 1.62

Access to Set Point Adjustments and Calibrations (Section 4.18)

RG 1.105

Identification (Section 4.22)

IEEE-494

- IEEE 308-1974 - Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations

This standard is applied to the electric systems in the stationary single-unit and multi-unit nuclear power generating stations that provide electric power to the Class 1E equipment. The electric systems included are comprised of the following interrelated systems:

1. Alternating current power systems.

2. Direct current power systems

3. Vital instrumentation and control power systems.

These systems consist of power supplies, i.e., standby generators and batteries, distribution equipment and components, i.e., tranformers, switchgear, bus cable, batt›ry chargers and inverters, and their associated instrumentation and controls, i.e., relays, meters, switches and control devices.

This standard does not apply to the unit generators and their buses, step-up, auxiliary and start-up transformers, connections to the station's switchyard, switchyard, transmission lines and transmission network.

The purpose of this standard is to provide:

1. The principle design criteria and the design features of the Class 1E power systems that enable the systems to their function requirement under the conditions produced by the design basis events.

2. The minimum operational conditions of the Class 1E power systems under which the station will be permitted to operate.

3. The surveillance requirements of the Class 1E power systems.

- IEEE 315-1975 - Graphic Symbols for Electric and Electronics Diagrams

  This standard provides a list of graphic symbols and class designation letters for use on electrical and electronics diagrams.

  Graphic symbols for electrical engineering are a shorthand used to show graphically the functioning or interconnections of the circuit. A graphic symbol represents the function of a part in the circuit. Graphic symbols are used on single line diagrams, on schematic or elementary diagrams, or as applicable on connection or wiring diagrams.

- IEEE 317-1976 - Electric Penetration Assemblies and Containment Structures for Nuclear Power Generating Stations

  This standard prescribes the requirements for the design, construction, test and installation of electric penetration assemblies and nuclear containment structures for stationary nuclear power generating stations. The requirements for external circuits which connect

to penetration assemblies are beyond the scope of this standard.

- IEEE 323-1974 - Qualifying Class 1E Equipment For Nuclear Power Generating Stations

  This document describes the basic requirements for qualifying Class 1E equipment and interfaces that are to be used in nuclear power generating stations. The requirements presented include the principles, procedures and methods of qualification. These qualification requirements, when met, will confirm the adequacy of the equipment design under normal, abnormal, design basis event, and containment test conditions for the performance of Class 1E functions.

- IEEE 336-1977 - Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations

  This standard sets forth the requirements for installation, inspection and testing of Class 1E electric power, instrumentation and control equipment and systems during the construction phase of a nuclear power generating station. These requirements are intended to assure that only materials and equipment of acceptable quality are incorporated into the plant, that quality is maintained and quality workmanship prevail throughout the construction process, and that complete installations conform to specified requirements, so as to promote public safety, prevent accidents, mitigate the consequences of accidents if they occur, and provide a high degree of plant reliability.

- IEEE 338-1977 - Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems

  This standard provides design and operational criteria for the performance of periodic testing of nuclear power generating station safety systems. The scope of testing covered consists of functional tests, checks, calibration verification and time response measurements. Criteria are also provided for determining system operational availability, status and necessary documentation; and for establishing test intervals and test procedures during operation. Routing preventive maintenance is not covered by this document.

- IEEE 344-1975 - Seismic Qualifications of Class 1E Electric Equipment for Nuclear Power Generation Stations

These recommended practices provide direction for establishing procedures that will yield data that will verify that the Class 1E equipment can meet its performance requirements during and following one Safe Shutdown Earthquake (SSE) preceded by a number of Operating Basis Earthquakes (OBE). This document may be used by equipment manufacturers to establish procedures that will yield data to substantiate performance claims or by equipment users to evaluate and verify performance of representative devices and assemblies as part of an overall qualification effort.

- IEEE 352-1975 - General Principles for Reliability Analysis of Nuclear Power Generating Station Protection Systems

This document was prepared to provide the designers and operators of nuclear power plant protection systems and the concerned regulatory groups with the essential methods and procedures of reliability engineering that are applicable to protection systems. By applying principles given, systems may be analyzed, acceptable test intervals may be established, results may be reconciled with reliability objectives, and the analysis may be suitably documented.

- IEEE 379-1977 - Application of the Single Failure Criterion to Nuclear Power Generating Station Class 1E Systems

The purpose of this document is to interpret the single failure criterion and to provide guidance in its application. It is intended that invoking system standards which utilize the single failure criterion will result in application of this standard. However, it is not the function of this standard to identify those standards in order to determine where the single failure criterion is to be applied, or to force compliance of the single failure criterion on a system. It is the specific function of this document to interpret how the single failure criterion is to be applied to Class 1E systems.

- IEEE 380-1975 - Definitions of Terms Used in IEEE Nuclear Power Generating Stations Standards

All of the terms and definitions in this document have

been taken from current IEEE standards on nuclear power generating stations. The source of each definition is indicated by a number of brackets immediately following the definition.

- IEEE 382-1972 - Guide for Type Tests for Class 1 Electric Valve Operators for Nuclear Power Generating Stations

This guide provides direction for establishing a type test that will yield data which verify that Class 1 electric valve operators for nuclear power generating stations can meet their design basis performance requirements.

- IEEE 384-1977 - Criteria for Independence of Class 1E Equipment and Circuits

The scope of this document is the independence requirements of the circuits and equipment comprising or associated with Class 1E systems. It sets forth criteria for the independence that can be achieved by physical separation and electrical isolation of circuits and equipment which are redundant, but does not address the determination of what is to be considered redundant.

The purpose of this document is to establish the criteria for implementation of the independence requirements for IEEE Standard 279-71, Criteria for Protection Systems for Nuclear Power Generating Stations, and IEEE Standard 308-74, Criteria for Class IE Power Systems for Nuclear Power Generating Stations.

- IEEE 415-1976 - Guide for Planning of Preoperational Testing Programs for Class 1E Power Systems for Nuclear Power Generating Station

This document provides guidance for preoperational testing of Class 1E power systems for nuclear power generating stations. The extent of the system shall be covered by IEEE 308-74, Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations.

The purpose of this document is to provide direction for establishing an acceptable preoperational testing program for Class 1E power systems in nuclear power generating stations. The preoperational tests are performed after the appropriate construction tests have been completed. The purpose of the preoperational tests is

to verify that the functioning of Class 1E power systems performs the required function as described in the Final Safety Analysis Report (FSAR) of the station.

- IEEE 420-1973 - Guide for Class 1E Control Switchboards for Nuclear Power Generating Stations

This document applies to all Class 1E control switchboards, regardless of their application or location. It does not apply to the non-Class 1E control switchboards, except as they affect the Class 1E control switchboards, that recognize that Class 1E control switchboards may be adjacent to, or joined with, other non-Class 1E control switchboards.

This document includes, but is not necessarily limited to, design, construction, wiring practices, shipping, handling, storage, installation, testing and quality assurance.

General design practices which are not unique to Class 1E control switchboards are not addressed in this document. This document applies to piping systems and Class 1E equipment only as they may affect the integrity of the Class 1E control switchboards.

The purpose of this document is to provide guidance for the design, manufacture and installation of Class 1E control switchboards and nuclear power generating stations to conform to IEEE Standard 279.

- IEEE - P567/D4G - Criteria for the Design of the Control Room Complex for a Nuclear Power Generating Station (Still Under Review)

This standard provides criteria for the design of the control room complex for a nuclear power generating station. This standard is directed toward the control room and the overall complex in which it is housed as a physical entity which protects personnel and safety-related equipment from the adverse effects of hazards which could prevent them from performing their intended function.

Each applicant for a construction permit or an operating license is required to provide assurance that the proposed nuclear generating station meets the General Design Criteria contained in 10 CFR 50, Appendix A. This standard is intended to assist the applicant in meeting those criteria which are directly applicable to the control room and its associated areas.

This standard addressed the normally attended central control room of a nuclear power generating station and the overall complex in which this room is housed. It is not intended to cover special or normally unattended control rooms such as those provided for radioactive waste handling or for emergency shutdown operations.

The nuclear power generating station control room complex provides a protective envelope for plant operating personnel and for instrument and control equipment vital to the operation of the plant during normal and abnormal conditions. In this capacity, the control room complex must be designed and constructed to meet the following criteria of 10 CFR 50, Appendix A, General Design Criteria for nuclear power plants:

Criterion 2 - Design Bases for Protection Against Natural Phenomena

Criterion 3 - Fire Protection

Criterion 4 - Environmental and Missile Design Bases

Criterion 5 - Sharing of Structures, Systems and Components (Multi-Unit Stations Only)

Criterion 19 - Control Room

The purpose of this standard is to provide guidance for the design of the nuclear power plant control room and control room complex, which must meet the applicable criteria. Requirements are established and recommendations are offered to aid the designer in meeting the applicable General Design Criteria.

- IEEE 494-1974 - Standard Method for Identification of Documents Related to Class 1E Equipment and Systems for Nuclear Power Generating Stations

This document establishes criteria for the uniform identification of documents significant to design, construction, testing, operation, and maintenance of Class 1E equipment and systems for nuclear power generating stations. It includes recommendations for identification of specific parts of these documents. Criteria are also established for identification on documents of redundant portions for Class 1E equipment and systems.

- IEEE 497-1977 - Trial-Use Criteria for Post Accident Monitoring Instrumentation for Nuclear Power Generating Stations

This standard applies to the design of instrumentation to monitor and display required post accident conditions within the nuclear power generating station.

Instrumentation addressed by this document includes that which enables the operator to: 1) identify the accident to the degree necessary for him to perform his role; 2) assess whether or not safety systems are accomplishing the required safety functions (for example, cooling the core, controlling containment pressure, etc.); 3) determine when conditions exist that require specified manual actions; and 4) follow the course of the accident to determine whether or not conditions are evolving within prescribed limits.

- IEEE 498-1975 - Supplementary Requirements for Calibration and Control of Measuring and Test Equipment Used in the Construction and Maintenance of Nuclear Power Generating Stations

This standard sets forth the requirements for a calibration program to control and verify the accuracy of M&TE (measuring and test equipment) which is used to assure that important parts of nuclear power generating stations are in conformance with prescribed technical requirements, and that data provided by testing, inspection, or maintenance are valid. These important parts include those structures, systems, and components whose satisfactory performance is required for the plant to operate safely, to prevent accidents that could cause undue risk to the health and safety of the public, or to mitigate the consequences of such accidents if they were to occur. This standard is intended to be used in conjunction with American National Standard Quality Assurance Program Requirements for Nuclear Power Plants, ANSI N45.2-1971.

- IEEE 501-1978 (ANSI C37.98) - Seismic Testing of Relays

This standard specifies the procedures to be used in the seismic testing of relays used in power system facilities. The standard is concerned with the determination of the seismic fragility level of relays, and also gives recommendations for proof testing.

- IEEE 566-1977 - Recommended Practice for the Design of Display and Control Facilities for Central Control Rooms of Nuclear Power Generating Stations

  This document establishes guidelines to be used by power plant system designers in selecting information and control devices to be made available in the central control room, and in determining how and where they shall be made available so that they can most reliably and quickly be used by the operator. The guide addresses the functional requirements of the information systems, controls, and displays, but not the selection of specific devices or equipment. It does not apply to the physical design of the control room enclosure or structures mounted therein.

  It provides uniform guidelines for the functional selection, coordination, and organization of control and information systems in a nuclear power plant central control room.

- IEEE 577-1976 - Requirements for Reliability Analyses in the Design and Operation of Safety Systems for Nuclear Power Generating Stations

  The purpose of this standard is to provide uniform, minimum acceptable requirements for the performance of reliability analyses for safety-related systems found in nuclear power generating stations, but not to define the need for an analysis. The need for reliablity analysis has been identified in other standards which expand the requirements of regulations (e.g., IEEE Standard 379-1972, ANSI N41.2-1972, "Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems", which describes the application of the single failure criterion).

- IEEE 353-1975 - Guide for General Principles of Reliability Analysis of Nuclear Power Generating Station Protection Systems

  This document provides guidance in the application and use of reliability techniques referred to in this standard.

- IEEE 603-1977 - Trial-Use Criteria for Safety System for Nuclear Power Generating Stations

  The criteria contained in this standard established

minimum functional and design requirements for safety systems for nuclear power generating stations. The safety system is the collection of systems required to minimize the probability and magnitude of release of radioactive material to the environment by maintaining plant conditions within the allowable limits established for each design basis event. Safety system functional and design criteria are also contained in other standards.

3.6.5 National Fire Protection Association (NFPA)

The National Fire Protection Association (NFPA) was organized in 1896 to promote the science and improve the methods of fire protection. The NFPA has acted as sponsor of the National Electrical Code since 1911. The original code document was developed in 1897 as a result of the united efforts of various insurance, electrical, architectural and allied interests.

- NFPA - 70 - National Electrical Code 1978 (NEC)

This National Electrical Code is sponsored by the National Fire Protection Association under the auspices of the American National Standards Institute. This code is purely advisory as far as the NFPA and ANSI are concerned, but is offered for use in law and regulatory purposes in the interest of life and property protection. The purpose of this code is the practical safeguarding of persons and property from hazards arising from the use of electricity.

This code contains provisions considered necessary for safety. Compliance therewith and proper maintenance will result in an installation essentially free from hazard, but not necessarily efficient, convenient or adequate for good service or future expansion of electrical use. This code is not intended as a design specification nor an instruction manual for untrained persons.

This code covers electrical conductors and equipment installed within or on public or private buildings or other structures, conductors that connect the installations to a supply of electricity, and other outside conductors on the premises.

This code is intended to be suitable for mandatory application by governmental bodies exercising legal jurisdiction over electrical installations and for use by insurance inspectors. The authority having jurisdiction

of enforcement of the code will have the responsibility for making interpretations of the rules, for deciding upon the approval of equipment and materials, and for granting the special permission contemplated in a number of the rules.

The authority having jurisdiction may waive the specific requirements in the code or permit alternate methods, where it is assured that equivalent objectives can be achieved by establishing and maintaining effective safety.

To promote uniformity of interpretation and application of the provisions of the code, the National Electrical Code Committee has established interpretation procedures.

- NFPA - 496 - Purged Enclosures for Electrical Equipment

The object of this standard is to provide information for the design or purged enclosures for the purpose of eliminating or reducing within the enclosure a Class 1 hazardous location classification, as defined in Article 500 of the National Electrical Code, NFPA No. 70. By this means, equipment which is not otherwise acceptable for the hazardous locations may be utilized in accordance with the National Electrical Code.

This standard applies to instruments, control rooms motors, motor controllers, switchgear, and similar equipment. This standard applies to locations where flammable gases or vapors may be present in the air in concentrations sufficient for the location to be classified as hazardous.

3.6.6 Instrument Society of America (ISA)

The Instrument Society of America (ISA) has been advancing the technology of instrumentation and control for the benefit of mankind since 1945. ISA members' expertise include all of the arts and sciences related to the theory, design, manufacture and use of instrumentation, computers and systems for measurement and control. The ISA works toward an improved standard of living for all mankind through voluntary efforts to generate and disseminate instrumentation and control knowledge, and to improve quality, productivity and safety.

Following is a compilation of ISA standards that are of particular interest to instrumentation and control engineers in the nuclear power field:

- ISA - S5-75 - Instrumentation Symbols and Identification

  Establishes a uniform means of designating instruments and instrumentation systems used for measurement and control. The differing established procedural needs of various organizations are recognized, where not inconsistent with the objectives of the standard, by providing alternative symbolism methods. A number of options are provided for adding information or simplifying the symbolism, if desired.

- ISA - S5-76 - Instrument Loop Diagrams

  Provides a method and practices for the preparation and use of instrument loop diagrams in the design, construction, checkout, start-up, operation, maintenance, and reconstruction of instrument systems and industrial plants.

- ISA - S7.3-75 - Quality Standard for Instrument Air

  This standard is to establish a maximum allowable moisture content at which the instruments will function satisfactorily; a maximum entrained particle size which will avoid plugging and wear/erosion of air passages and orifices; a maximum allowable oil content which will avoid malfunction due to clogging and wear of the components; and awareness of a possible source of corrosive or toxic contamination entering the air system through the compressor suction, plant air system cross connections, or instrument air connections directly connected to processes.

- ISA S7.4-70 - Air Pressures for Pneumatic Controllers and Transmission Systems

  Purpose is to establish standard operating pressure ranges for pneumatic intelligence transmission systems, and standard air supply pressures (with limit values) for operation and pneumatic controllers and pneumatic intelligence transmission systems.

- ISA - S37.1-75 - Electrical Transducer Nomenclature and Terminology

  Establishes uniform nomenclature for transducers and uniformed simplified terminology for transducer characteristics.

- ISA - S51.1-76 - Process Instrumentation Terminology

  Intended to include all specialized terms used to describe the use and performance of the instrumentation and instrument systems used for measurement, control, or both, in the process industries.

- ISA - RP3.1-60 - Flow Meter Installation, Seal and Condensate Chambers

  Intended to set rigid specifications for materials, fabrications, pressure rating and similar points, since these should ordinarily be in compliance with code requirements which are established for the particular industry concerned.

- ISA - RP3.2-60 - Flange Mounted Sharp Edged Orifice Plates for Flow Measurement

  Intended to cover only orifice plates designed for bolting between flanges, with flat or raised face, whose bolting dimensions are in accordance with ANSI/B16 Series Standards.

- ISA - RP4.2-56 - Standard Control Valve Manifold Designs

  Intended for use where maximum flexibility is needed. Only carbon steel piping and valves are considered and there is no intention of covering special designs or unusual materials.

- ISA - RP7.1-56 - Pneumatic Control Circuit Pressure Test

  Intended to provide a satisfactory procedure for the testing of pneumatic control circuits for leaks together with reasonable criteria for acceptance of work done and suitable aids for performance.

- ISA - RP12.1-60 - Electrical Instruments in Hazardous Atmospheres

  Provides general guidance for the safe installation of electrical instruments using appropriate means to prevent ignition of flammable gases and vapors.

- ISA - RP16.6-61 - Methods and Equipment for Calibration of Variable Area Meters (Rotameters)

  Describes the methods and equipment used for calibrating the glass and metal metering tube area meters (rotameters).

- ISA - RP18.1-65 - Specifications and Guides for the Use of General Purpose Annunciators

  This covers those annunciators which are intended to call attention to a change in operating conditions. Concerned only with that unit which receives the actuating signals and provides a visual display of the alarm; this unit in itself is commonly called the annunciator.

- ISA - RP25.1-57 - Materials for Instruments in Radiation Service

  Intended to serve as a guide to the selection of materials for use in intense radiation fields, such as those encountered in and around nuclear reactors.

- ISA - RP42.1-65 - Nomenclature for Instrument Tubing Fittings (Threaded)

  Defines nomenclature for tubing fittings most commonly used in instrumentation. It is not intended as a substitute for manufacturers catalog numbers nor does it apply to special fittings. It is intended to apply to a mechanical fitting rather than a sweat fitting.

The following standards are being developed and have not been issued as of yet:

- ISA - RP67.1 - Transducer and Transmitter Installation Practices

- ISA - SP67.2 - Instrument Piping and Tubing

- ISA - SP67.3 - Instrument Systems for Leak Detection

3.6.7 American Nuclear Society (ANS)

The main objectives of the American Nuclear Society (ANS) include the advancement of science and engineering, the integration of the scientific disciplines, the encouragement of research, and the dissemination of information all with respect to nuclear science and technology. The standards offered by the American Nuclear Society have been approved by the American National Standards Institute and are listed in that section of this lesson with the notation "sponsored by ANS".

In addition, see Section 3.6.2 of this text for a discussion of ANS safety classifications.

4.0 SUMMARY

Guides, codes, regulations, and standards as applicable to instrumentation and control design, and construction practices in a nuclear power plant by government agencies and technical societies have been presented in this unit. With this information, the NRC inspectors should be able to acquire a working knowledge of the pertinent codes and regulations.

INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

UNIT II ATTACHMENT 1

PRE-STUDY TEXT

APPLICABLE CODES AND STANDARDS TO I&C SYSTEMS

FROM THE STANDARD REVIEW PLAN, REV. 1

CRITERIA	TITLE	APPLICABILITY (SAR CHAPTER) 7.3	7.4	7.5	7.6	7.7	REMARKS
1. 10 CFR Part 50							
a. 10 CFR 50.34	Contents of Application: Technical Information	X	X	X	X	X	
b. 10 CFR 50.36	Technical Specifications	X	X	X	X	X	
c. 10 CFR 50.55a	Codes and Standards	X	X	X	X	X	
2. General Design Criteria (GDC). Appendix A to 10 CFR Part 50							See Section 3.1 for discussion of each design Criteria
a. GDC 1	Quality Standards and Records	X	X	X	X		
b. GDC 2	Design Bases for Protection Against Natural Phenomena	X	X	X	X		
c. GDC 3	Fire Protection	X	X	X	X		
d. GDC 4	Environmental and Missile Design Bases	X	X	X	X		
e. GDC 5	Sharing of Structures, Systems and Components	X	X	X	X		Not applicable
f. GDC 10	Reactor Design	X	X	X	X		See NSSS SSAR
g. GDC 12	Suppression of Reactor Power Oscillations			X			See NSSS SSAR

APPLICABLE CODES AND STANDARDS TO I&C SYSTEMS

FROM THE STANDARD REVIEW PLAN, REV. 1

CRITERIA	TITLE	APPLICABILITY (SAR CHAPTER) 7.3	7.4	7.5	7.6	7.7	REMARKS
h. GDC 13	Instrumentation and Control	X	X	X	X	X	
i. GDC 15	Reactor Coolant System Design			X	X	X	See NSSS SSAR
j. GDC 19	Control Room	X	X	X	X	X	
k. GDC 20	Protection System Functions	X	X	X	X		
l. GDC 21	Protection System Reliability and Testability	X	X	X	X		
m. GDC 22	Protection System Independence	X	X	X	X		
n. GDC 23	Protection System Failure Modes	X	X	X	X		
o. GDC 24	Separation of Protection and Control Systems	X	X	X	X	X	
p. GDC 25	Protection System Requirements for Reactivity Control Malfunctions			X			See NSSS SSAR
q. GDC 26	Reactivity Control System Redundancy and Capability		X	X		X	See NSSS SSAR
r. GDC 27	Combined Reactivity Control Systems Capability		X	X		X	See NSSS SSAR

APPLICABLE CODES AND STANDARDS TO I&C SYSTEMS

FROM THE STANDARD REVIEW PLAN, REV. 1

CRITERIA	TITLE	APPLICABILITY (SAR CHAPTER) 7.3	7.4	7.5	7.6	7.7	REMARKS
s. GDC 28	Reactivity Limits			X	X	X	See NSSS SSAR
t. GDC 29	Protection Against Anticipated Operational Occurrences	X	X	X	X	X	
u. GDC 33	Reactor Coolant Makeup		X	X			See NSSS SSAR
v. GDC 34	Residual Heat Removal	X	X	X	X		
w. GDC 35	Emergency Core Cooling	X		X	X		See NSSS SSAR
x. GDC 37	Testing of Emergency Core Cooling System	X		X	X		See NSSS SSAR
y. GDC 38	Containment Heat Removal	X		X	X		
z. GDC 40	Testing of Containment Heat Removal System	X		X	X		
aa. GDC 41	Containment Atmosphere Cleanup	X		X	X		
bb. GDC 43	Testing of Containment Atmosphere Cleanup Systems	X		X	X		
cc. GDC 44	Cooling Water	X		X	X		
dd. GDC 46	Testing of Cooling Water System	X		X	X		

APPLICABLE CODES AND STANDARDS TO I&C SYSTEMS

FROM THE STANDARD REVIEW PLAN, REV. 1

CRITERIA	TITLE	APPLICABILITY (SAR CHAPTER) 7.3	7.4	7.5	7.6	7.7	REMARKS
ee. GDC 50	Containment Design Basis	X		X	X		
ff. GDC 54	Piping Systems Penetrating Containment	X		X	X		
gg. GDC 55	Reactor Coolant Pressure Boundary Penetrating Containment	X		X	X		
hh. GDC 56	Primary Containment Isolation	X		X	X		
ii. GDC 57	Closed Systems Isolation Valves	X		X	X		
3. Institute of Electrical and Electronics Engineering (IEEE) Standards:							
a. IEEE Std. 279-1971 (ANSI N42.7-1972)	Criteria for Protection Systems for Nuclear Power Generating Stations	X	X	X	X	X	See 10 CFR 50.55a (h) and Reg. Guide 1.62
b. IEEE Std. 308-1974	Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations		X	X	X		See Reg. Guide 1.32

APPLICABLE CODES AND STANDARDS TO I&C SYSTEMS

FROM THE STANDARD REVIEW PLAN, REV. 1

CRITERIA	TITLE	APPLICABILITY (SAR CHAPTER) 7.3	7.4	7.5	7.6	7.7	REMARKS
c. IEEE Std. 317-1976	Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations	X	X	X	X	X	See Reg. Guide 1.63
d. IEEE 323-1974	Trial Use Standard Guide for Qualifying Class 1E Electric Equipment for Nuclear Power Generating Stations	X	X	X	X		See Reg. Guide 1.89
e. IEEE 334-1974	Type Test of Continuous Duty Class 1E Motors for Nuclear Power Generating Station	X	X		X		
f. IEEE Std. 336-1974 (ANSI N45.2.4-1972)	Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations	X	X	X	X	X	See Reg. Guide 1.30
g. IEEE Std. 338-1975	Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems	X	X	X	X		

<u>APPLICABLE CODES AND STANDARDS TO I&C SYSTEMS</u>

FROM THE STANDARD REVIEW PLAN, REV. 1

CRITERIA	TITLE	APPLICABILITY (SAR CHAPTER) 7.3	7.4	7.5	7.6	7.7	REMARKS
h. IEEE Std. 344-1975 (ANSI N41.7)	Guide for Seismic Qualification of Class 1 Electrical Equipment for Nuclear Power Generating Stations	X	X	X	X		
i. IEEE Std. 379-1972 (ANSI N41.2)	Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems	X	X	X	X	X	See Reg. Guide 1.53
j. IEEE 382-1977	Type Test of Class 1E Electric Valve Operators for Nuclear Power Generating Stations	X	X		X		
k. IEEE Std. 384-1974 (ANSI N41.14)	Criteria for Separation of Class 1E Equipment and Circuits	X	X	X	X	X	
l. IEEE 420-1973	Type-Use Guide for Class 1E Control Switch Boards for Nuclear Power Generating Stations	X	X	X	X	X	

APPLICABLE CODES AND STANDARDS TO I&C SYSTEMS

FROM THE STANDARD REVIEW PLAN, REV. 1

CRITERIA	TITLE	APPLICABILITY (SAR CHAPTER) 7.3	7.4	7.5	7.6	7.7	REMARKS
m. IEEE 494-1974	Standard Method for Identification of Documents Related to Class 1E Equipment and Systems for Nuclear Power Generating Stations	X	X	X	X		
n. IEEE 497-1977	Trial-Use Criteria for Post Accident Monitoring Instrument for Nuclear Power Generating Stations			X			See Reg. Guide 1.97
4. Regulatory Guides (RG)							See Appendix B for discussion of each Reg. Guide
a. RG 1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems		X	X	X		
b. RG 1.7	Control for Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident	X		X			

APPLICABLE CODES AND STANDARDS TO I&C SYSTEMS

FROM THE STANDARD REVIEW PLAN, REV. 1

CRITERIA	TITLE	APPLICABILITY (SAR CHAPTER) 7.3	7.4	7.5	7.6	7.7	REMARKS
c. RG 1.11	Instrument Lines Penetrating Primary Reactor Containment	X	X	X	X		
d. RG 1.22	Periodic Testing of Protection System Actuation Functions	X	X	X	X		
e. RG 1.29	Seismic Design Classification	X	X	X	X		
f. RG 1.30	Quality Assurance Requirements for the Installation, Inspection and Testing of Instrumentation and Electric Equipment	X	X	X	X	X	
g. RG 1.32	Use of IEEE Std. 308-1971, "Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations"		X	X	X		
h. RG 1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	X	X	X	X		Use in conjunction with Position 3, RG 1.17
i. RG 1.53	Application of the Single Failure Criterion at Nuclear Power Plant Protection Systems	X	X	X	X		

APPLICABLE CODES AND STANDARDS TO I&C SYSTEMS

FROM THE STANDARD REVIEW PLAN, REV. 1

CRITERIA	TITLE	APPLICABILITY (SAR CHAPTER) 7.3	7.4	7.5	7.6	7.7	REMARKS
j. RG 1.62	Manual Initiation of Protection Actions	X	X		X		
k. RG 1.63	Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants	X	X	X	X		
l. RG 1.68	Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors	X	X	X	X	X	
m. RG 1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Rev. 2	X	X	X	X	X	
n. RG 1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants	X					
o. RG 1.75	Physical Independence of Electric Systems	X	X	X	X		See Section 8.3.1.4 and Appendix B

APPLICABLE CODES AND STANDARDS TO I&C SYSTEMS

FROM THE STANDARD REVIEW PLAN, REV. 1

CRITERIA	TITLE	APPLICABILITY (SAR CHAPTER) 7.3	7.4	7.5	7.6	7.7	REMARKS
p. RG 1.78	Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	X					
q. RG 1.89	Qualification of Class 1E Equipment for Nuclear Power Plants	X	X	X	X	X	
r. RG 1.95	Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release				X		
s. RG 1.97	Instrumentation for Light-Water-Cooled Nuclear Power Plants to Access Plant Condition During and Following an Accident			X			See Section 7.5.2.7 and Appendix B
t. RG 1.100	Seismic Qualification of Electric Equipment for Nuclear Power Plants	X	X	X	X		
u. RG 1.105	Instrument Set Points	X	X	X	X		

APPLICABLE CODES AND STANDARDS TO I&C SYSTEMS

FROM THE STANDARD REVIEW PLAN, REV. 1

CRITERIA	TITLE	APPLICABILITY (SAR CHAPTER) 7.3	7.4	7.5	7.6	7.7	REMARKS
v. RG 1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves	X	X				
w. RG 1.118	Periodic Testing of Electric Power and Protection Systems	X	X	X	X		
x. RG 1.120	Fire Protection Guidelines for Nuclear Power Plants	X	X	X	X	X	See Section 9.5.1 and Appendix B
5. Branch Technical Positions (BTP) EICSB							
a. BTP EICSB 1	Backfitting of the Protection and Emergency Power Systems of Nuclear Reactors	X	X		X		Not Applicable
b. BTP EICSB 3	Isolation of Low Pressure Systems from the High Pressure Reactor Coolant System		X		X		See NSSS SSAR
c. BTP EICSB 4	Requirements on Motor-Operated Valves in the ECCS Accumulator Lines		X		X		See NSSS SSAR
d. BTP EICSB 5	Scram Breaker Test Requirements - Technical Specifications						See NSSS SSAR

APPLICABLE CODES AND STANDARDS TO I&C SYSTEMS

FROM THE STANDARD REVIEW PLAN, REV. 1

CRITERIA	TITLE	APPLICABILITY (SAR CHAPTER) 7.3	7.4	7.5	7.6	7.7	REMARKS
e. BTP EICSB 9	Definition and Use of "Channel-Calibration" - Technical Specifications		X	X	X		See Utility-Applicant's SAR
f. BTP EICSB 10	Electrical and Mechanical Equipment Seismic Qualification Program	X	X	X	X		
g. BTP EICSB 12	Protection System Trip Point Changes for Operation with Reactor Coolant Pumps Out of Service	X					See NSSS SSAR
h. BTP EICSB 13	Design Criteria for Auxiliary Feedwater Systems	X					
i. BTP EICSB 14	Spurious Withdrawals of Single Control Rods in Pressurized Water Reactors					X	See NSSS SSAR
j. BTP EICSB 15	Reactor Coolant Pump Breaker Qualification						See NSSS SSAR
k. BTP EICSB 16	Control Element Assembly (CEA) Interlocks in Combustion Engineering Reactors						See CESSAR

APPLICABLE CODES AND STANDARDS TO I&C SYSTEMS

FROM THE STANDARD REVIEW PLAN, REV. 1

CRITERIA	TITLE	APPLICABILITY (SAR CHAPTER) 7.3	7.4	7.5	7.6	7.7	REMARKS
l. BTP EICSB 18	Application of the Single Failure Criteria to Manually-Controlled Electrically-Operated Valves	X	X		X		
m. BTP EICSB 19	Acceptability of Design Criteria for Hydrogen Mixing and Drywell Vacuum Relief Systems	X			X		Not Applicable
n. BTP EICSB 20	Design of Instrumentation and Controls Provided to Accomplish Changeover from Injection to Recirculation Mode	X	X		X		
o. BTP EICSB 21	Guidance for Application of Reg. Guide 1.47	X	X	X	X		
p. BTP EICSB 22	Guidance for Application of Reg. Guide 1.22	X	X	X	X		
q. BTP EICSB 23	Qualification of Safety-Related Display Instrumentation for Post Accident Condition Monitoring and Safe Shutdown Condition			X			See Section 7.5.2.7

APPLICABLE CODES AND STANDARDS TO I&C SYSTEMS

FROM THE STANDARD REVIEW PLAN, REV. 1

CRITERIA	TITLE	APPLICABILITY (SAR CHAPTER) 7.3	7.4	7.5	7.6	7.7	REMARKS
r. BTP EICSB 24	Testing of Reactor Trip System and Engineered Safety Feature Actuation System Sensor Response Times	X	X		X		See NSSS SSAR
s. BTP EICSB 25	Guidance for the Interpretation of General Design Criterion 37 for Testing the Operability of the Emergency Core Cooling System as a Whole	X	X				
t. BTP EICSB 26	Requirements for Reactor Protection System Anticipatory Trips						See NSSS SSAR
u. BTP EICSB 27	Design Criteria for Thermal Overland Protection for Motors of Motor-Operated Valves	X	X		X		

INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

UNIT III - Documentation

PRE-STUDY TEXT

INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

UNIT III - Documentation

PRE-STUDY TEXT

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INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

UNIT III - Documentation

PRE-STUDY TEXT

1.0 OBJECTIVE

The objective of this unit is to introduce the Inspector to the various logic, wiring, and physical characteristic drawings necessary for the installation, checkout and operation of instruments and controls.

2.0 INTRODUCTION

The lesson will discuss the various forms of documentation, their application and how they interrelate to form a cohesive engineering communication package. The basic tools necessary to understand these documents can be obtained by reading through Appendices of this pre-study material - containing symbols and communication for two methods or types of diagrams. In the lesson the types of documents discussed will include the following:

- Flow Diagrams

  Which show basic piping, process and instrumentation information.

- Loop Diagrams

  Showing detailed instrumentation electrical connection information.

- Logic Diagrams

  Show operational information.

- Elementary Diagrams

  An electrical schematic diagram depicting the functional circuitry.

- Control Wiring Diagrams

  Show electrical connection information for both I&C and electrical equipment.

- Cabling Diagrams

  Similar to Control Wiring Diagrams but used in more complex systems.

- Instrument Location and Routing Drawings

  Show physical location and installation details of I&C devices.

- Instrument Specifications

  Show condensed engineering data on I&C devices.

3.0 INSTRUMENTATION DOCUMENTS

Many different engineering firms have developed their own system of I&C documentation implementation, however these are usually very similar or identical to standards discussed in the previous section. We shall discuss the many different types of I&C documents and how they are implemented in the engineering and construction of a nuclear power plant.

3.1 Types of Flow Diagrams

The term "Flow Diagram" is used loosely in various sectors of the process and control industries. In its simplest form the "Flow Diagram" is a basic block flow diagram showing only process information as related to major pieces of equipment. In a more complex form, the P&ID, most functions of piping, equipment and instrumentation are shown in detail.

In any case, the flow diagram is generally the "Bible" document of any control system. This document always shows the flow of the process to and from all pieces of process equipment. Types of flow diagrams vary in the degree of detail shown for the instrumentation that controls the process.

3.1.1 ISA Flow Sheets

As instruments controlling plant processes have grown more complex, so have the symbols used to represent them on I&C documentation. Hence about 25 years ago the Instrument Society of America issued Recommended Practice RP5.1 to establish some order to the foreseeable problem.

Since then the Recommended Practice has been revised twice and is now a Standard entitled "Instrumentation Symbols and Identification" and has the document number ISA - S5.1. (Also, it is ANSI approved and carries document number ANSI Y32.20).

This document serves as a guide for representating instruments and instrumentation systems, implementing both symbols and an identification code. This method, although not limited to, is found on flow diagrams.

When engineering firms develop their own standards for flow sheets, they are invariably very similar to this ISA standard with variations arising from individual needs. Generally, a chart or table showing symbology is given at the front of a drawing package. A knowledge of the ISA symbology will enable the user to more readily understand symbology variations.

3.1.2 P&ID's

This designation is given for flow sheets that usually follow the ISA method. The letters P&ID stand for either "Piping and Instrumentation Diagram" or "Process and Instrumentation Diagram."

INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

UNIT III - Documentation

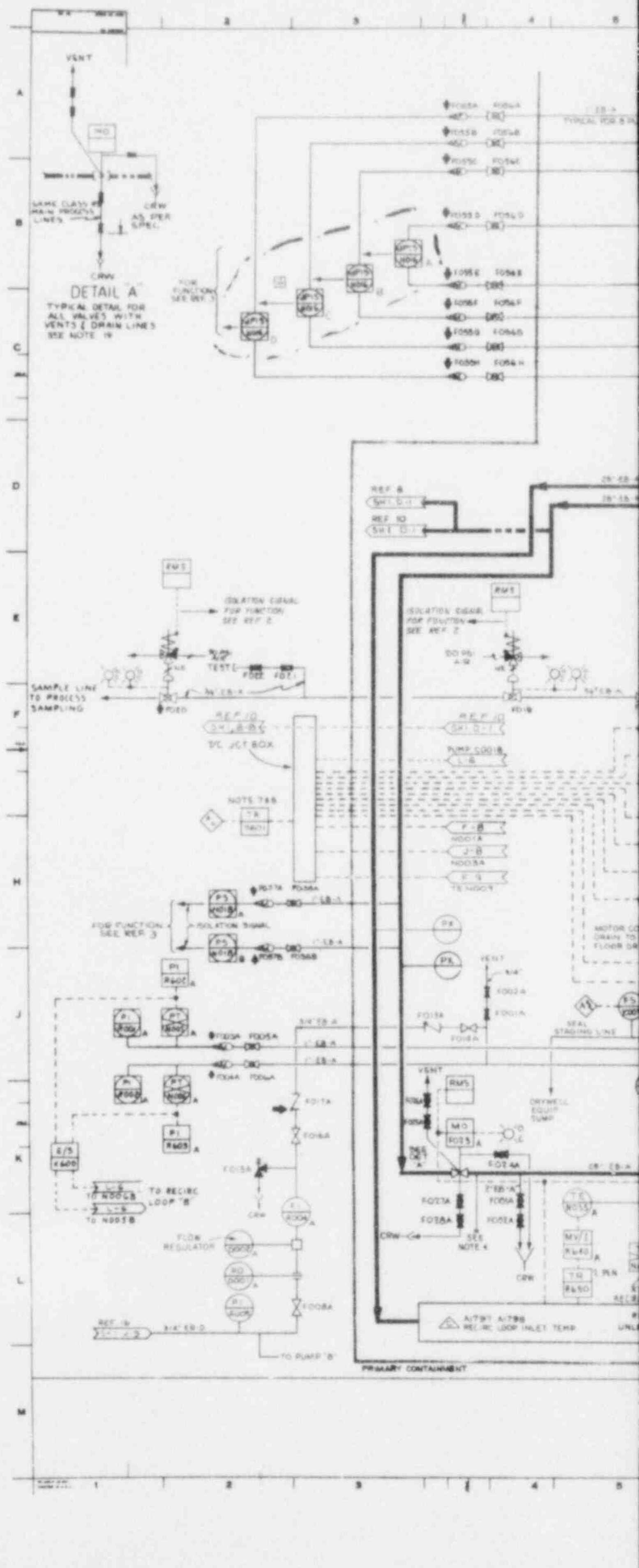
PRE-STUDY TEXT

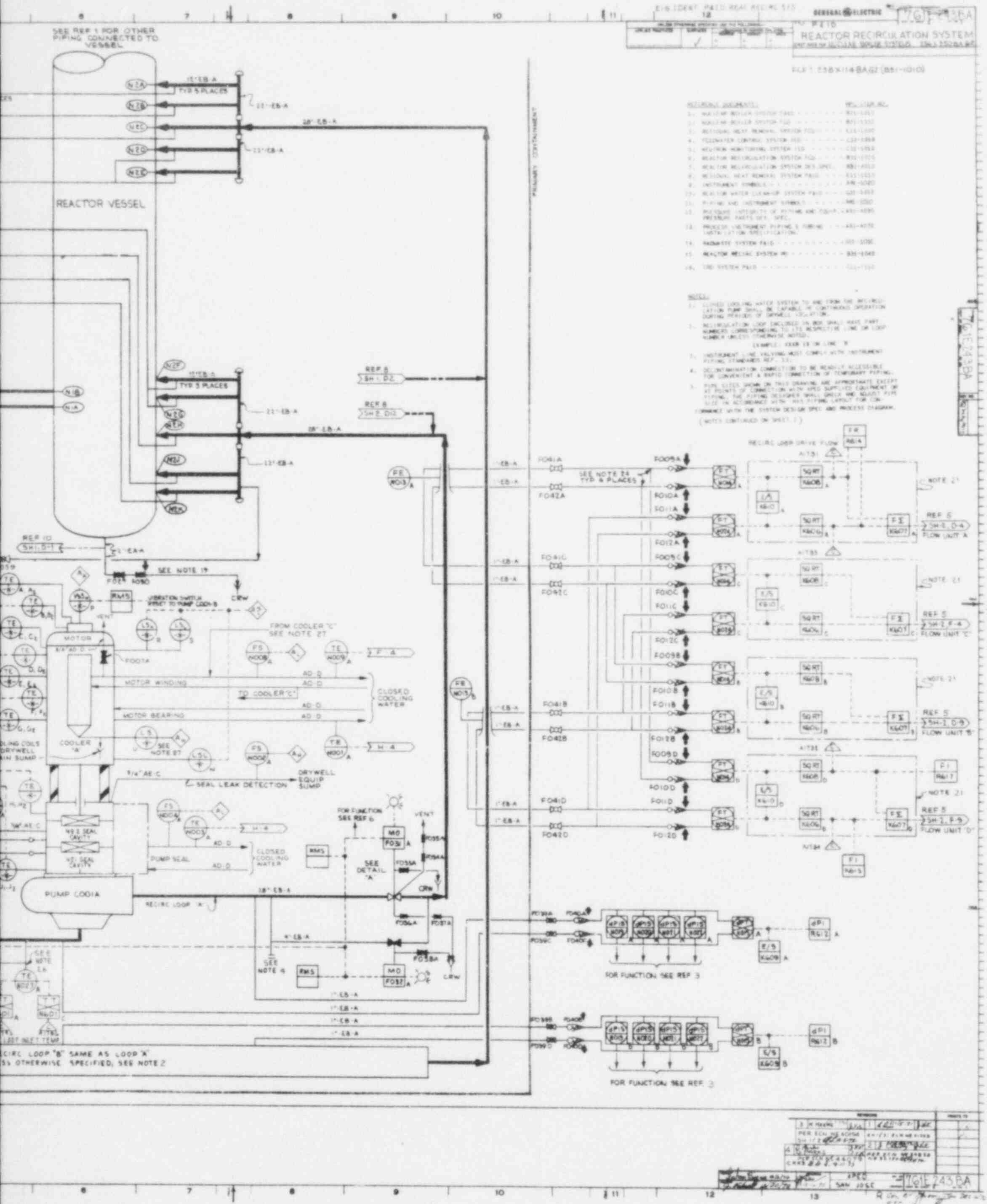
FIGURE III-1

"FLOW DIAGRAM"

TI
APERTURE
CARD

Also Available On
Aperture Card





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INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

UNIT III - Documentation

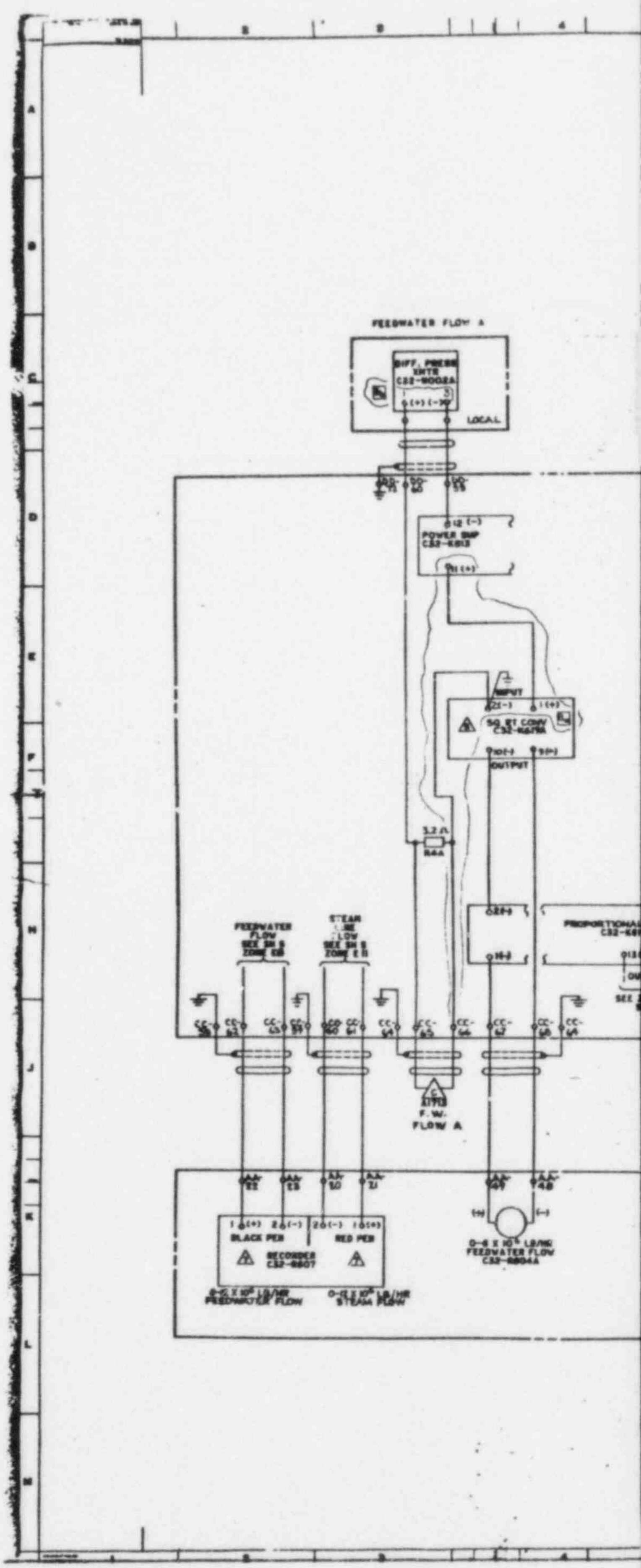
PRE-STUDY TEXT

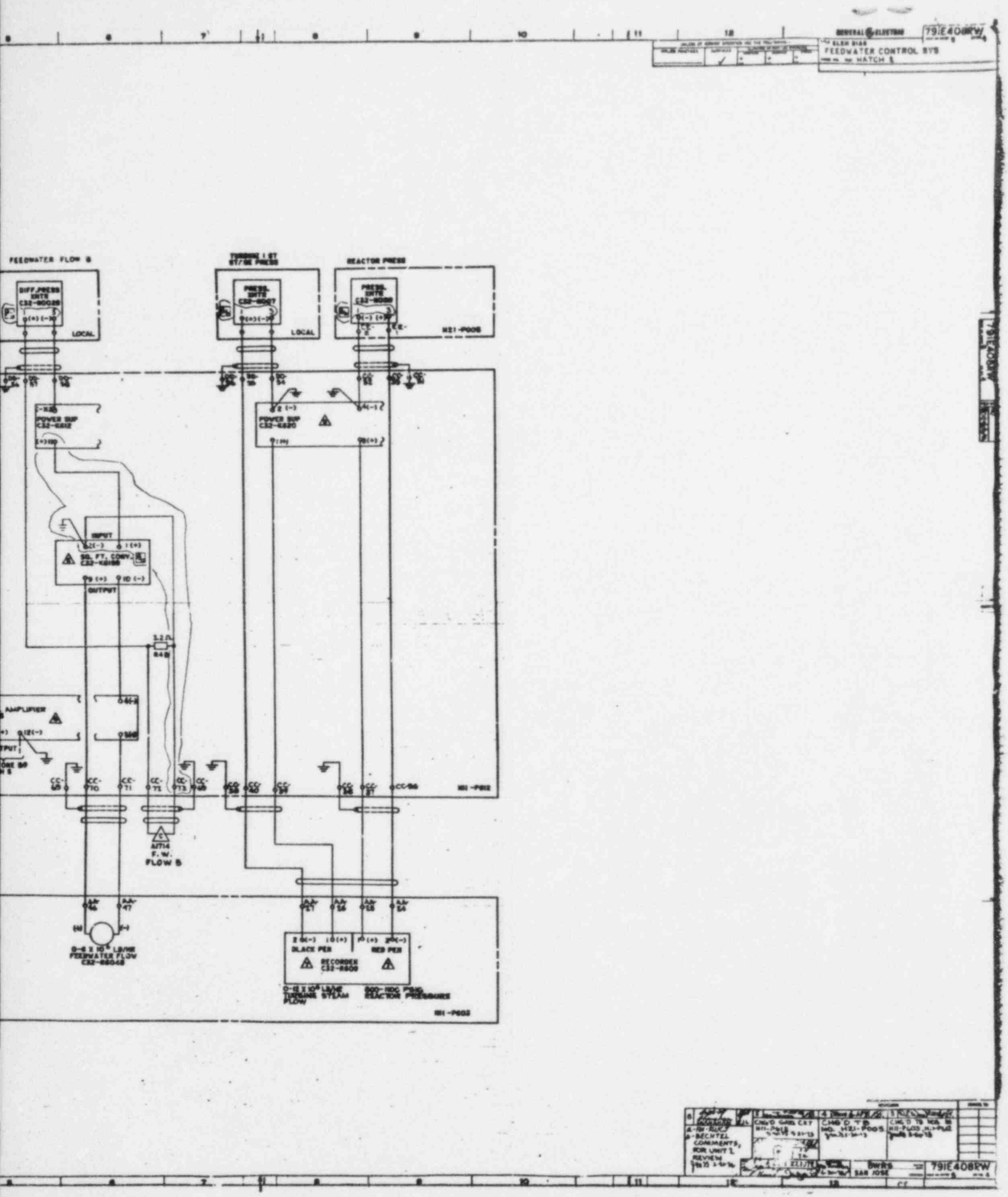
FIGURE III-2

"LOOP DIAGRAM"

TI
APERTURE
CARD

Also Available On
Aperture Card





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3.1.3 Instrument Schematics

This type of document is commonly used in the power industry and is a slightly different form of flow sheet. It is used in conjunction with a simple flow diagram that shows only the "root" instrument.

The Instrument Schematic breaks the simple flow diagram into sub-systems and represents one particular sub-system in a method similar or identifical to the ISA method. This document is usually accompanied by "Logic Diagrams" and Loop Diagrams."

3.2 Loop Diagrams

The Piping and Instrument Drawing (P&ID) or Flow Diagram applies to the whole process loop, while the loop diagram gives further information on the control loop of an individual parameter.

Once again, while different engineering firms may develop their own standards of documentation standards, by knowing and understanding the ISA method of loop diagramming, one is able to be proficient at understanding similar systems.

The Instrument Society of America has developed standard ISA - S5.4 "Instrument Loop Diagrams" to facilitate the undertstanding of loop diagrams and to improve communication between personnel concerned with instrument systems.

Loop diagrams graphically describe complete control loops. They show the operation of valves, pumps and other accessories and detailed connection data. Loop diagrams are used by the engineer as a basis for selecting the hardware needed for implementing the control design.

3.3 Logic Diagrams

Logic Diagrams may show the operation of equipment or pneumatic and electrical circuits and are usually on "B" (11" x 17") sized sheets to be compatible in size to the remainder of I&C documentation.

Logic diagrams are used to depict digital logic functions (two-state devices) for specified equipment operation and protection as simply as possible with minimum reference to physical implementations.

Logic Diagrams serve as a basis for developing Elementary Wiring Diagrams or Control Wiring Diagrams (CWDs). They can be used by the engineer to determine the specific hardware required and by "the field" to determine the proper operation of equipment.

In 1973, an American National Standard entitled "IEEE Standard Graphic Symbols for Logic Diagrams (Two-State Devices)" was approved by ANSI and IEEE, which carried the document numbers ANSI and Y32.14-1973 and IEEE Std 91-1973. This standard represents a major achievement in the continuing effort to reflect state-of-the-art accomplishments in standardized methods of communication.

However, the details of this document are at times cumbersome for use in I&C logic diagrams only because of the usual simplicity of I&C systems in relation to the document. This usually leads to engineering firms developing their own simplified version of this lengthy document to incorporate only their aspects of the standard that are applicable to their own method of I&C engineering.

3.4 Elementary Diagram

An Elementary Diagram is a type of electrical or electronic diagram that is also called a schematic. This is a diagram which shows, by means of graphic symbols, the electrical connections and functions of a specific circuit arrangement. The schematic or elementary diagram facilitates tracing the circuit and its functions without regard to the actual physical size, shape or location of the component device or parts.

These diagrams are a factor in international trade, the use of one common symbol language ensures a clear presentation and economical diagram preparation and economical diagram preparation for a variety of users. Hence, the development of the American National Standard entitled "Graphic Symbols for Electrical and Electronics Diagrams (Including Reference Designation Letters)." This document is approved by and carries the following standards organizations document numbers: ANSI Y32.2-1975, CSA Z99-1975 and IEEE Std 315-1975.

Since this is a National Standard, we are faced with a lengthy document that is applicable to many different phases of the electronic industries. We shall be concerned with only those sections that find their applicability in the Instrumentation and Control discipline of the Nuclear Power Industry. Therefore we will be studying only sections of this standard as follows:

- "Quick Reference to Symbols"
- "Section 1 Qualifying Symbols"
- "Section 2 Graphic Symbols for Fundamental Items"
- "Section 4 Graphic Symbols for Contacts, Switches, Contactors and Relays"

3.5 Control Wiring Diagram (CWD)

A CWD is exactly what the name implies; a diagram of actual control wiring. Drawing format may differ from one A-E firm to another, due to the lack of a national standard, but all CWDs perform the same function. That is, to show the details of wiring connections, device locations, circuit functions, separation, and in general to give enough control wiring information to facilitate design, equipment procurement, construction, testing, operation and maintenance of Instrumentation and Control systems and equipment.

One method of CWD implementation is that of using an Elementary Diagram or Schematic and adding information as to device locations and connection numbers.

Another method graphically shows physical location of devices in relation to each other and shows physically how wiring is to be routed and connected.

Still another method is a combination of both of these methods.

3.6 Cabling Diagrams

Cabling Diagrams are usually implemented in a documentation system that does not incorporate the use of a CWD, or they are used in a complex control system that is too cumbersome to show on a CWD. In any case, the Cabling Diagram is a drawing showing only wiring connection information and no circuit information.

3.7 Instrument Location and Routing Drawings

An Instrument Location and Routing Drawing, or simply Instrument Plan, is the actual location of all I&C devices imposed on the plant elevation and plan drawings; usually duplicated from the actual Architectural drawing package. This drawing shows every component including process connection, device location, rack locations, routing of all signals (tubing and wiring), location of air supplies, power supplies, control room complex, penetration details, separation, and in general, all details necessary for installation of systems that may not be shown on Installation Details.

Instrument Installation Details show the actual installation of each "field" mounted device including orientation, valving, air supply, mounting details, material used, elevation and tag numbers. This drawing serves the construction and installation portion of a Nuclear Power Plant.

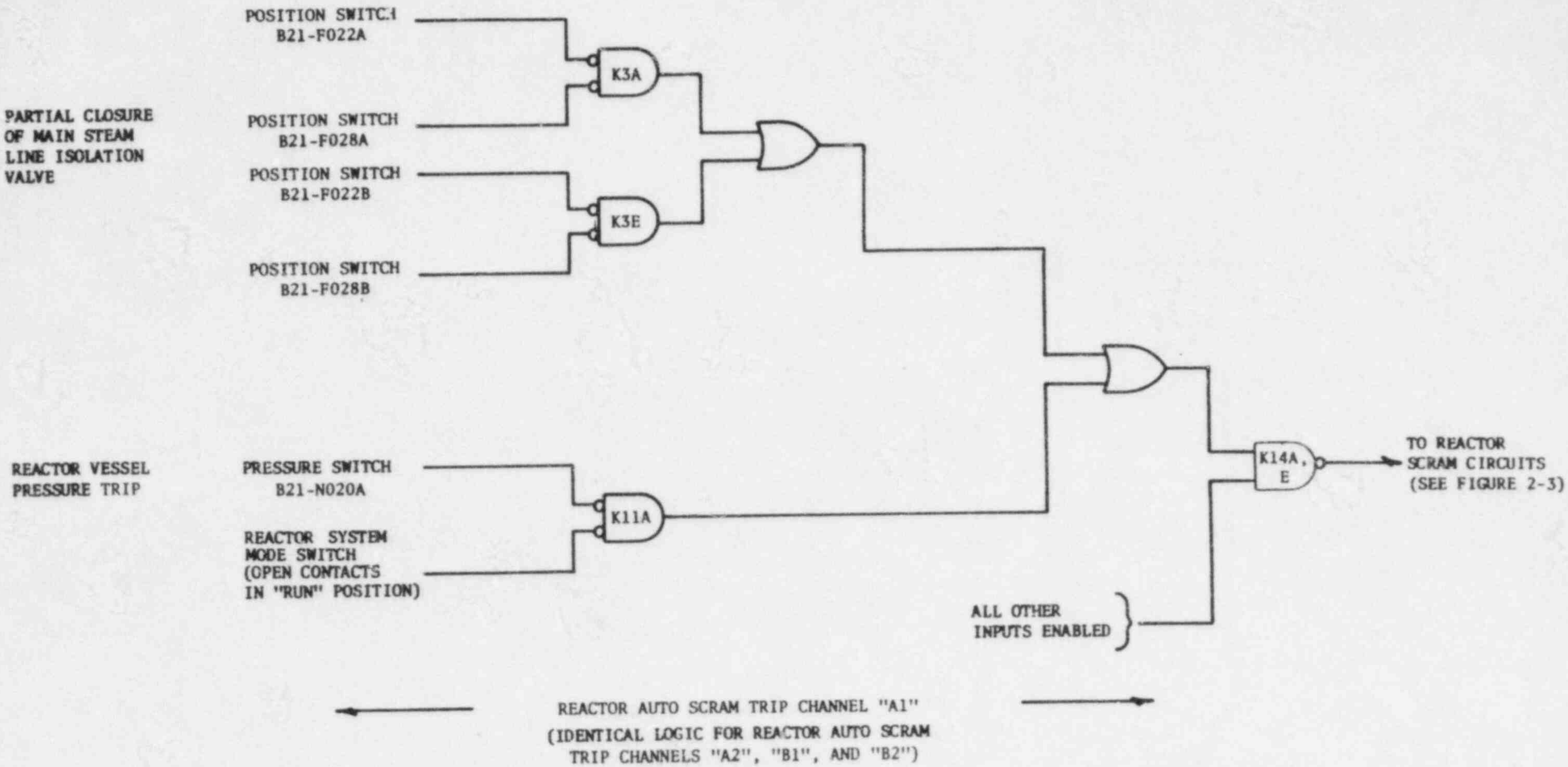


Figure III-3
Logic Diagram

INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

UNIT III - Documentation

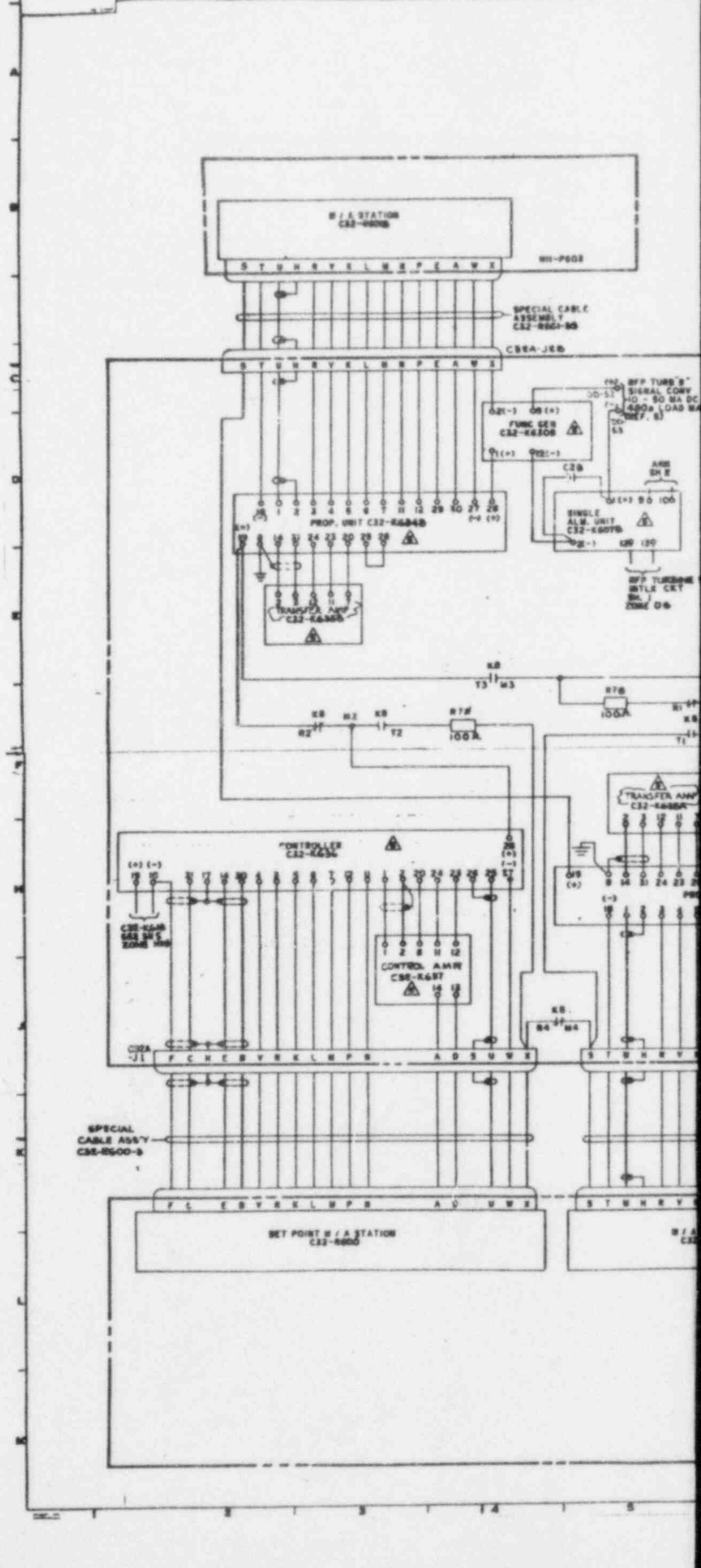
PRE-STUDY TEXT

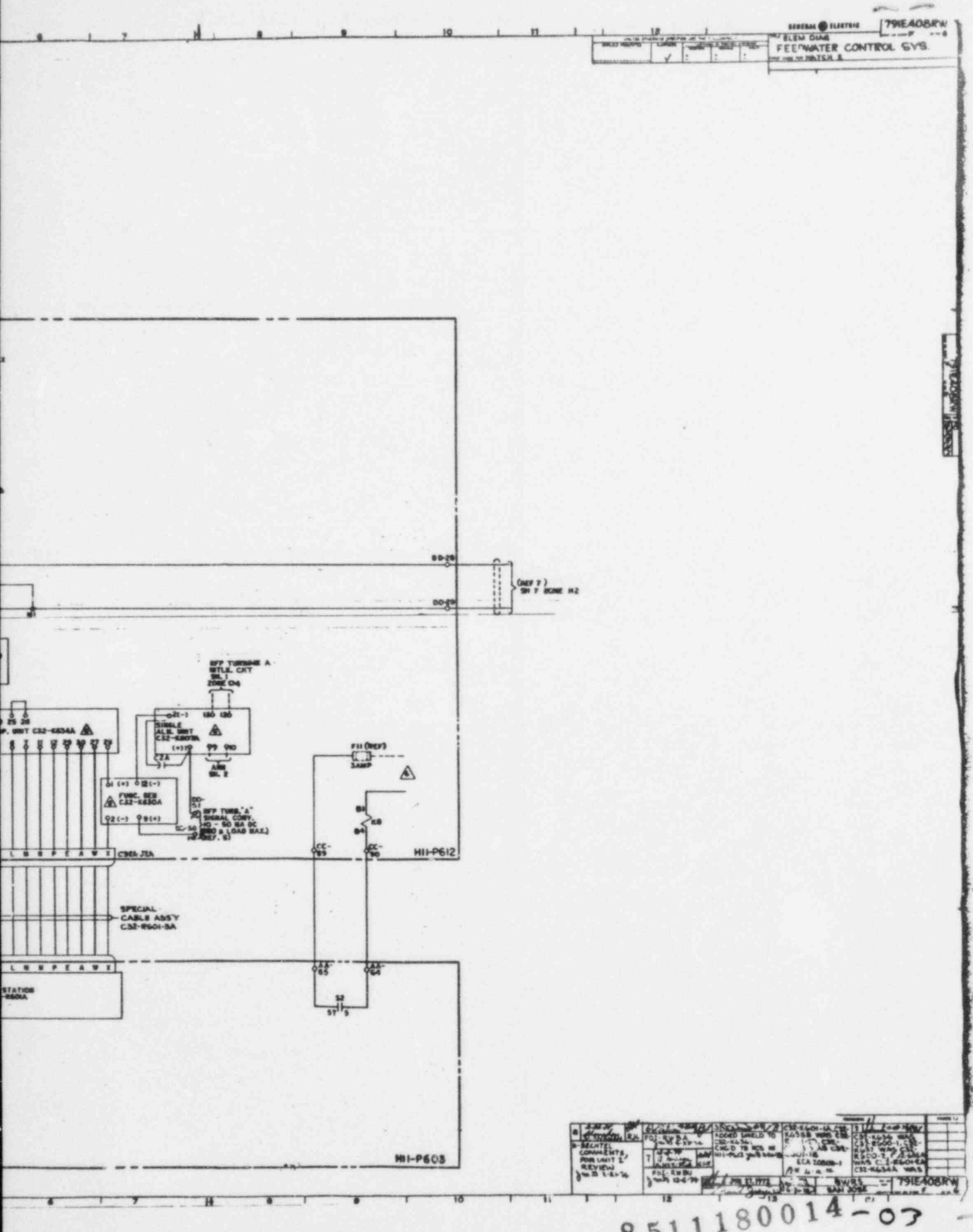
FIGURE III-4

"CONTROL WIRING DIAGRAM"

TI
APERTURE
CARD

Also Available On
Aperture Card





8511180014-07

3.8 Instrument Specifications

Because of the complexity of present day instruments and controls it is desirable to have some uniform method to list all pertinent data and details for all interested parties on one document. This can be accomplished by following Instrument Society of America standard number ISA-S20 entitled "Specification Forms for Process Measurement and Control Instruments, Primary Elements and Control Valves."

These forms are intended to assist the specification writer to present the basic information. In this sense they are "short form" specifications or "check sheets" and may not include all necessary engineering data or definitions of application requirements. However, these forms serve as a basis to present information of a minimum level and can be modified to accommodate specific applications.

This is another situation where most A-E firms have developed their own "short form" instrument specifications but are usually similar or identical to the ISA standard. Moreover, in the Nuclear Industry these "short form" specs usually serve only as an appendix to a lengthy detailed specification.

The main Instrument and Control Specifications in a Nuclear Power Plant are documents that cover in detail all pertinent parameters of any given type of instrument. (EXAMPLES OF INSTRUMENT SPECIFICATION CATEGORY TYPES FOR SPECIFICATIONS ARE: Electronic Instrumentation, Flow Elements, Thermocouple Assemblies, Pressure Gauges, and Temperature Switches). Some pertinent parameters covered in these specifications are:

a. Applicable standards

b. Environment

c. Exceptions to standard specifications

d. Instrument classifications

e. Quality assurance

f. Quality requirements

g. Quality compliance

h. Performance and guarantees

i. Testing

j. Special tools required

k. Technical data

l. Cleaning

m. Operation and maintenance manuals

n. Painting or coating

o. Packaging, shipping, handling and storage

p. Information and data required from seller

4.0 CALIBRATION STANDARD ACCURACY

The evolution of electrical measurement, and the measuring instruments to meet the requirements of electrical measurement, have changed rapidly over the past 40 years. From the early beginnings of the VOM (Volt-Ohm-Meter) whose accuracy was somewhere between 3% and 10%, the world of measurement has grown to the present-day instruments such as the Digital Voltmeter which can attain DC accuracies of 0.001%, or 10 ppm.

This factor of 10,000 to 1 in accuracy improvement has changed the approach and the attitude calibration throughout the world.

Possibly the greatest force in causing this rapid growth in accuracy has been the demand placed upon industry by the military over the years, and the aerospace programs of more recent years. The technology demand by these two factors alone has caused industry to develop more varied and more sophisticated instruments with which to carry out these programs. At the same time, it should not be overlooked that many of the advances in the area of measurement have been diligently carried forward by agencies such as National Bureau of Standards located throughout the world. In many instances, however, the breakthrough in the state-of-the-art came from engineering in industry, either through the development of a new component (such as the transistor), or from the application of that component in a new and unique way.

4.1 Active and Passive Instruments

In the world of calibration instrumentation there are active and passive instruments. A passive instrument is one that either (a) does not require or rely on operating power from an external source, or (b) does not contain elements within its circuitry that are responsible for amplification of the applied signal. An active instrument, on the other hand, does rely on powered circuitry which processes the applied signal in some manner. An example of a passive instrument is a resistive decade divider. An example of an active device is a Digital Voltmeter.

4.1.1 Calibration Levels

This categorization between passive and active devices generally forms the dividing line between the levels of calibration. Passive instruments are at the higher levels of accuracy and calibration. Active instruments are at lower levels of accuracy and calibration. This separation of accuracy levels helps to define also the dividing line between a primary standards laboratory and a secondary (or lower) laboratory.

4.2 Instrumentation Categories

Instrumentation falls into categories as a direct function of its accuracy capability, and we find that there are three major levels of accuracy which can be described. These are designated as Primary, Secondary, and Tertiary. The Primary level incorporates the passive elements such as the saturated standard cell and other standards such as the Thomas 1 ohm resistor, the 10k resistance standard, standard inductors and capacitors, and possibly a frequency standard. The Secondary level utilizes a much broader base of instruments, most of which are active (but still have a high accuracy) whose responsibility is to calibrate other higher level instruments used in research and development, or which are used to calibrate instrumentation used at a lower level of accuracy. The Tertiary level is considered to be the "working level" of instrumentation, and this is where the greatest quantity of instruments is located. But because this area of instrumentation is relied upon so greatly for day-to-day operations of manufacturing, testing and servicing, it is the area of instrumentation which should have the greatest attention when it comes to maintenance and calibration.

4.2.1 Levels of Accuracy

In order to better determine where different types of instruments should be used relative to their basic accuracies, the following table is given which displays the accuracies required for the Primary, Secondary, and Tertiary levels.

Primary
Resistance: 1-7 ppm to 100k; 10 ppm to 1M
Voltage (DC): 0-1200V, 5-10 ppm
Voltage (AC): 0-1200V, 10 Hz to 1.2 MHz; 0.01%
Current (DC): 0-10A, 0.02%
Current (AC): 2.5 mA-20A, 5 Hz-100 kHz; 0.03%
Frequency: 10 Hz-1,000 MHz; 1 x 10^{11}

Secondary
Resistance: 100 ohm - 12 M ohm; 0.02%
Voltage (DC): 0-1200V; 10-20 ppm
Voltage (AC): 1 mV-1200V, 10 Hz-1.2 MHz; 0.02-005%
Current (DC): 0-10A; 0.1%
Current (AC): 0-10A, 5 Hz-20 kHz; 0.05%
Frequency: 10 Hz-1,000 MHz; 2 x 10^{9}

Tertiary
Resistance: 100 ohm - 12 M ohm; 0.1%
Voltage (DC): 0-1200V, 0.02%
Voltage (AC): 1 mV-1200V, 10 Hz-100 kHz; 0.2%
Current (DC): 0-2A, 0.3%

Current (AC): 0-2A, 5 Hz-20 kHz; 0.5%
Frequency: 10 Hz-1,000 MHz; 5×10^{7}

As instrumentation becomes more advanced at any, or all, of the three levels depicted, these accuracies will undoubtedly improve in all respects. As a corollary this means that the demands of industry have risen, or that better instrumentation is available to industry or research at the same, or less, cost.

These events point out the continuing and growing requirement of people at all levels of the instrumentation business to become more aware of calibraiton requirements. Even though recently produced instruments are becoming more stable (along with their greater accuracy), they are still active devices which are subject to the normal deterioration due to time and environment. Therefore, it is always important to keep in mind that, in order to operate these instruments in a world that is demanding more and more in accuracy, it is necessary to consider the periodic calibration of these instruments much more seriously. It can be stated unequivocally that the technological capability of a nation can be directly related to the accuracy and stability of its instrumentation. A like comment could be made regarding the level of its technical personnel, and that is the theme of the material presented in this book. It is our endeavor to display information related to maintaining basic standards on down to the calibration level of bench instrumentation in the hope that this material will be used to further the technology of the electronic equipment user.

5.0 SUMMARY

For the NRC inspector's information, Attachment A and B have been added only to demonstrate the manner in which records are kept on calibration of NSSS instruments and instrument loops. While this is not important to the inspector during the construction phase, it does become imporatant to those involved with operating plants. These should be referred to during the lecture on calibration.

INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

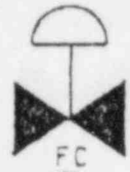
UNIT III - Attachment A

Pre-Study Text

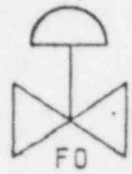
Sample Architect Engineer Instrument Symbols

TLD-PARTS

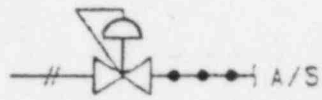
AIR OPERATED VALVE
FAIL CLOSE



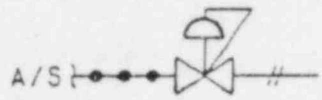
AIR OPERATED VALVE
FAIL OPEN



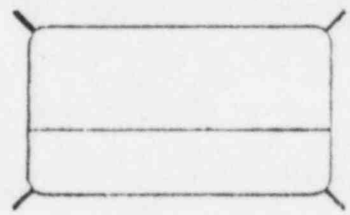
AIR SUPPLY REGULATOR



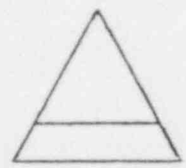
AIR SUPPLY REGULATOR



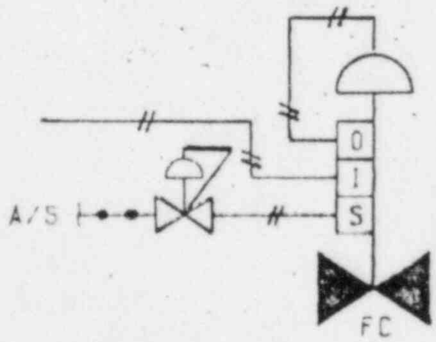
ALARM.ANNUNCIATOR



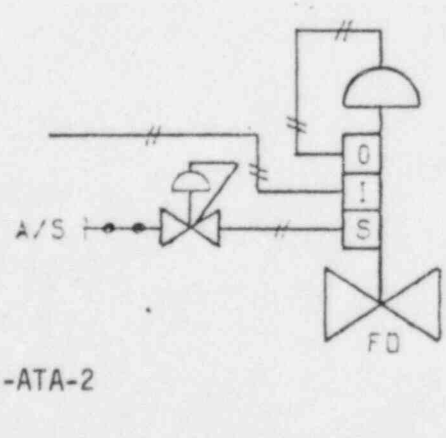
ALARM.COMPUTER



A.O. VALVE W/POSITIONER
FAIL CLOSE



A.O. VALVE W/POSITIONER
FAIL OPEN

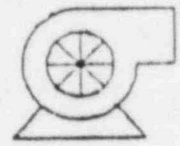


8-26-80

S&W CHOC
TLD-PARTS
INDEX
12210.TLD.A-

TLD-PARTS

BLOWER



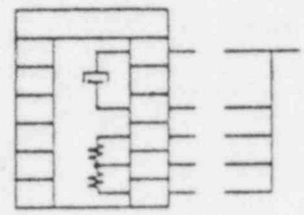
6-26-80

S&W CHOC
TLD-PARTS
INDEX
12210.TLD 3-

TLD-PARTS

CABLE DESIGNATION

CONDUCTIVITY CELL



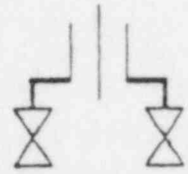
CONNECTION DOT

6-26-80

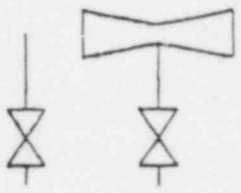
S&W CHOC
TLD-PARTS
INDEX
12210.TLD.C-

TLD-PARTS

FLOW ORIFICE



FLOW VENTURI



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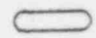
S&W CHOC
TLD-PARTS
INDEX
12210.TLD.F-

TLD-PARTS

GROUND.CHASSIS 

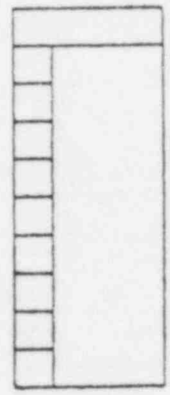
GROUND.SIGNAL 

GROUND.SHIELD 

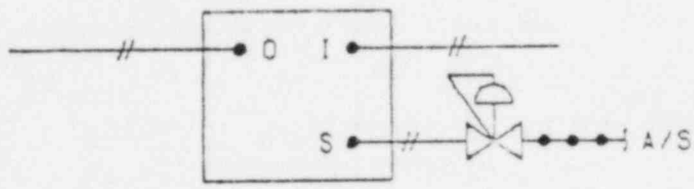
8-25-80

S&W CHOC
TLD-PARTS
INDEX
12210 TLD G-

INSTRUMENT,INDICATOR/RECORDER/HA STATION/
CONTROLLER



INSTRUMENT,PNEUMATIC CONTROLLER

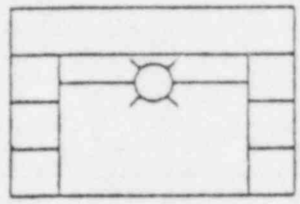


8-26-80

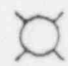
S&W CHOC
TLD-PARTS
INDEX

TLD-PARTS

LIGHT.INDICATING



LITE.GREEN/RED/AMBER/ETC...



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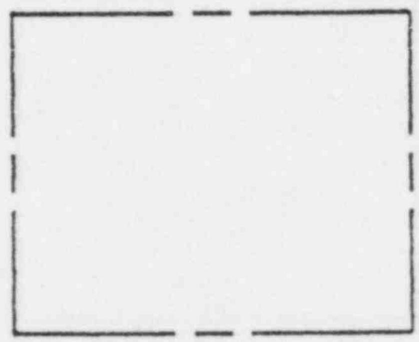
S&W CHOC
TLD-PARTS
INDEX

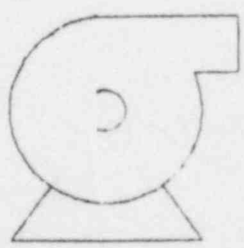
TLD-PARTS

PANEL/RACK

PANEL/RACK



PUMP



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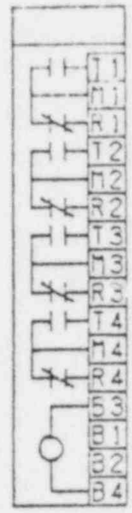
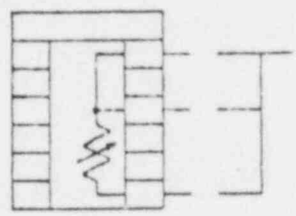
S&W CHOC
TLD-PARTS
INDEX
12210.TLD.P-

TLD-PARTS

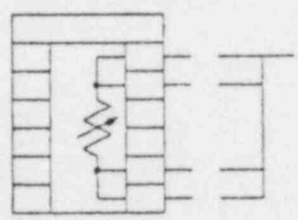
RELAY



RTD.3 WIRE



RTD.4 WIRE



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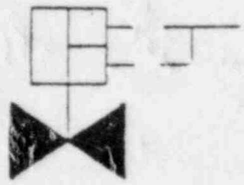
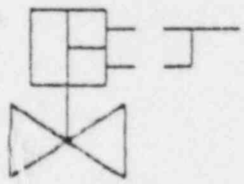
S&W CHOC
TLD-PARTS
INDEX
12210.TLD.R-

TLD-PARTS

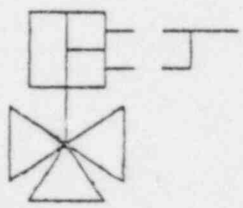
SOLENOID VALVE CLOSED.2 WAY



SOLENOID VALVE OPEN.2 WAY

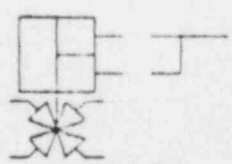


SOLENOID VALVE.3 WAY

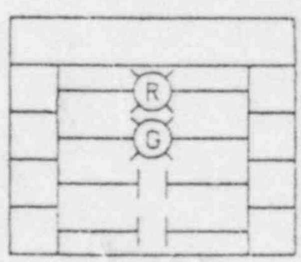


SOLENOID VALVE PORT.3 WAY

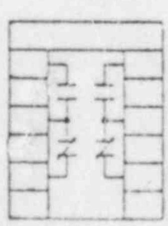
SOLENOID VALVE.4 WAY



SWITCH.CONTROL/
INDICATING

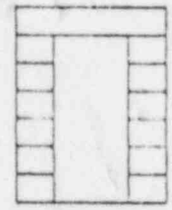


SWITCH.DUAL

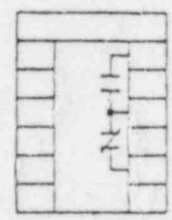


SWITCH.ELECTRICAL

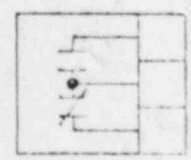
SWITCH.MISC



SWITCH.SINGLE



SWITCH.VALVE LIMIT



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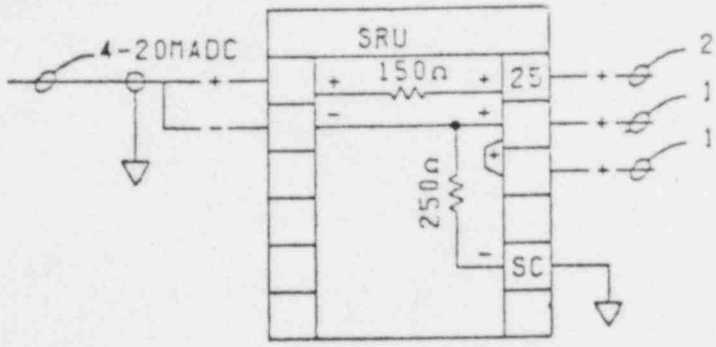
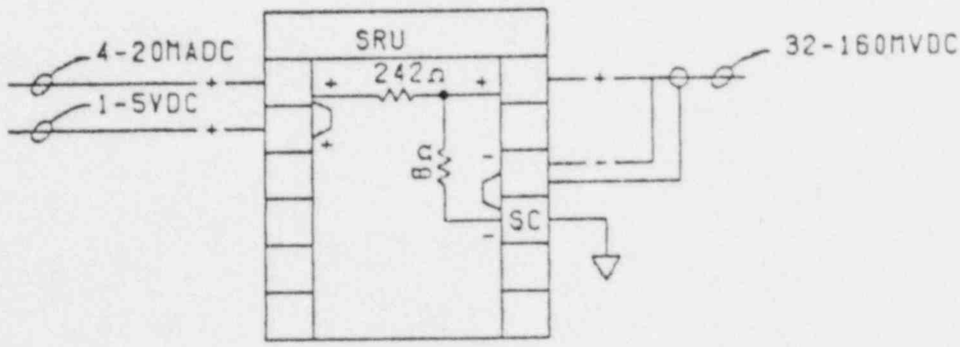
S&W CHOC
TLD-PARTS
INDEX
12210 TLD S1-

TLD-PARTS

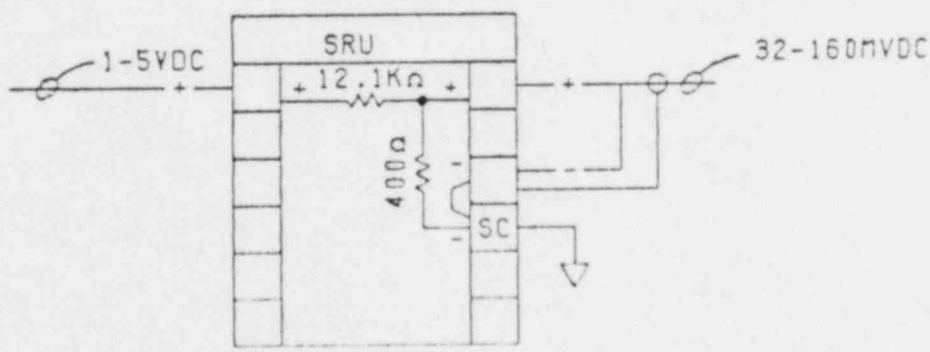
SRU. BAILEY 766100BAAA2



SRU. BAILEY 766010AAAA1



SRU. BAILEY 766012AAAA1



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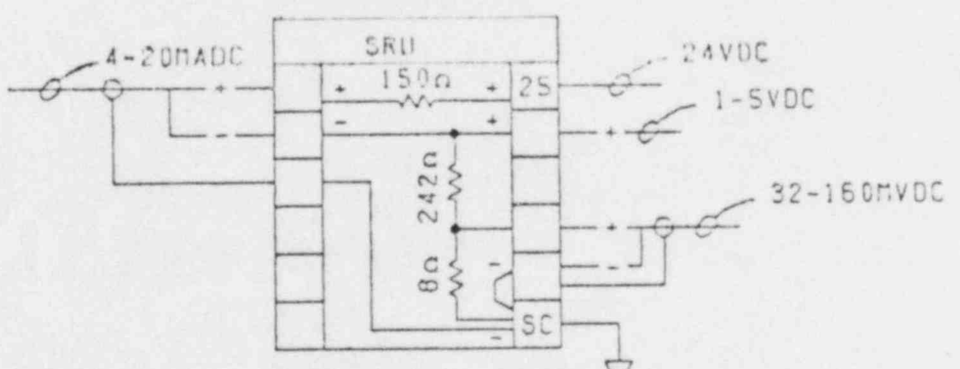

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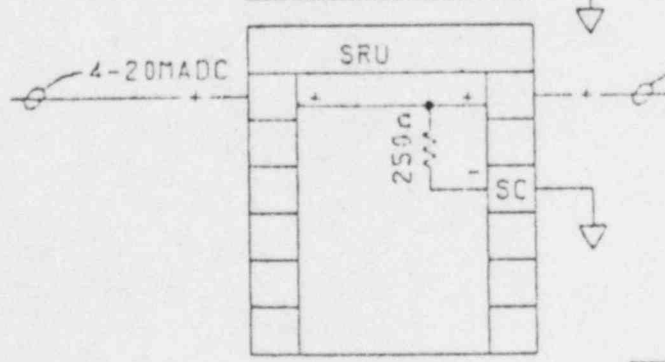
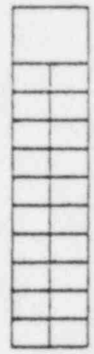


9-4-80

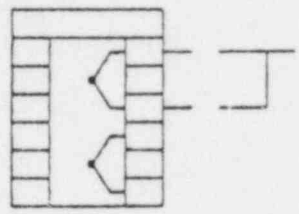
S&W CHOC
TLD-PARTS
INDEX
12210.TLD.S2-1

TLD-PARTS

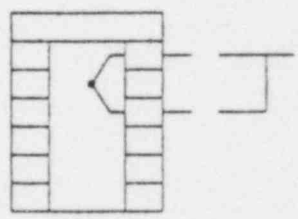
TERMINAL BOARD



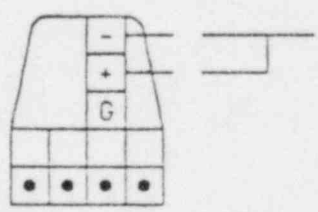
THERMOCOUPLE.DUAL



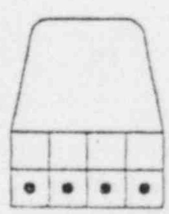
THERMOCOUPLE.SINGLE



TRANSMITTER.ELECTRONIC



TRANSMITTER.PNEU



8-26-80

S&W CHOC
TLD-PARTS
INDEX
12210.TLD.T-

TLD-PARTS

VALVE.CLOSED

VALVE.OPEN

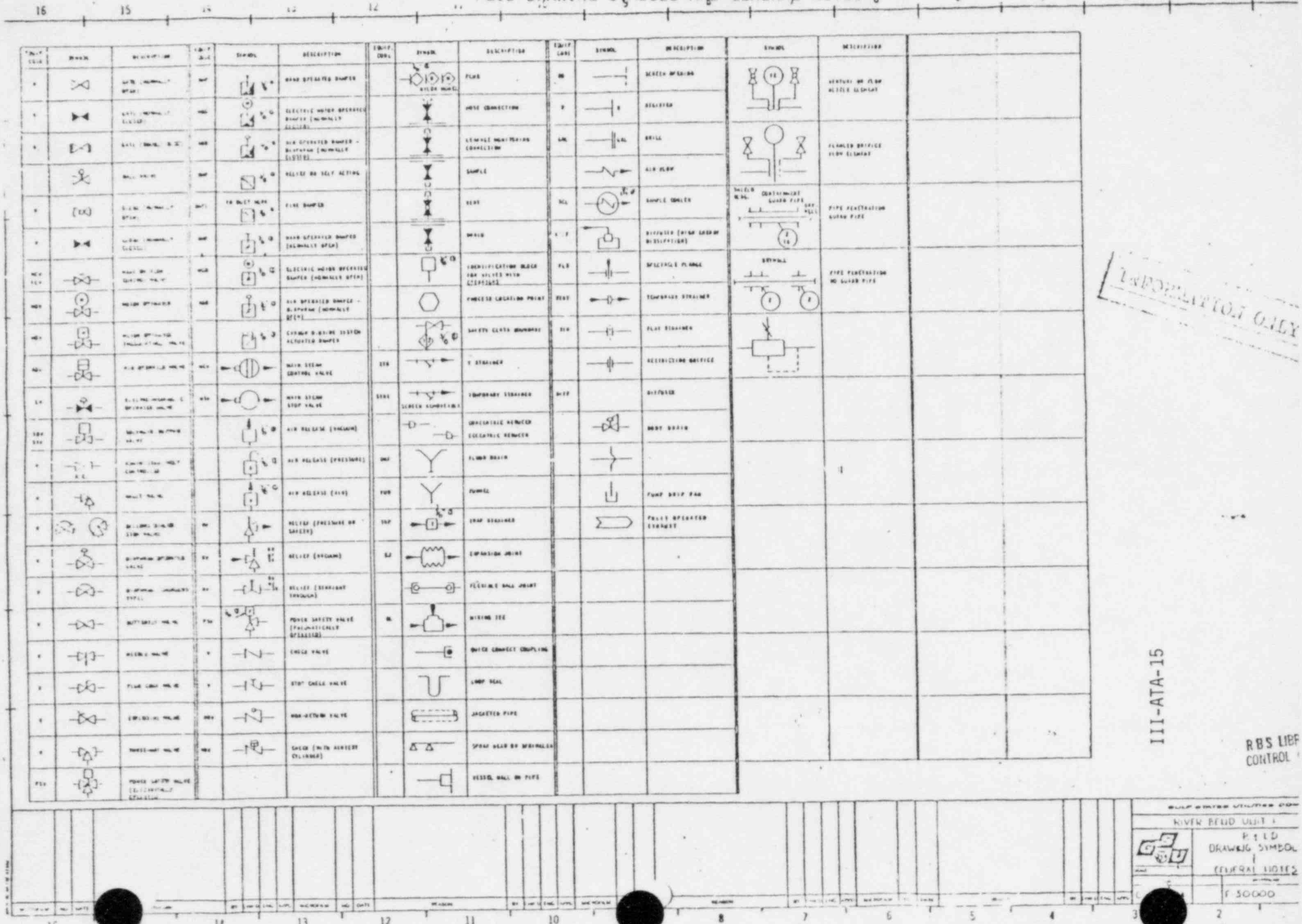
VALVE.PRESSCONTROL

8-26-80

S&W CHOC
TLD-PARTS
INDEX

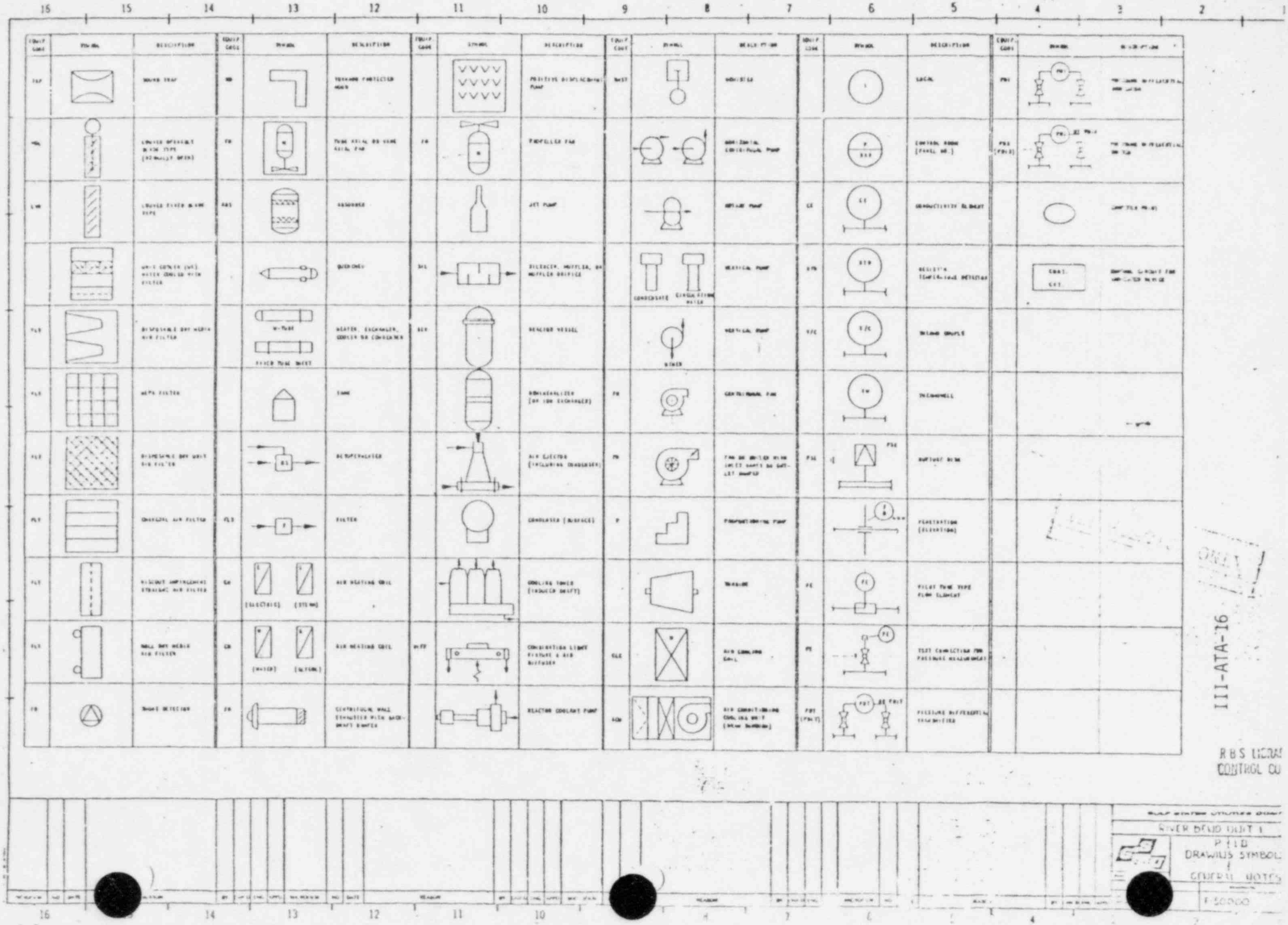
GULF STATES UTILITIES COMPANY

P&ID DRAWING SYMBOLS AND GENERAL NOTES



GULF STATES UTILITIES COMPANY

P&ID DRAWING SYMBOLS AND GENERAL NOTES



INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

UNIT III - Attachment B

Pre-Study Text

Sample Calibration Form from Various Nuclear Power Plants

III-ATB-2

INSTRUMENT CALIBRATION DATA CARD

ATTACHED TO:

ICR NO: 1.ILFPW.001
QA CAT: III
DATA CARD REV: 0

INSTR MARK NO: 1FPW-PI1A
TYPE: BOURDON TUBE
SYSTEM CODE: FPW
NOMENCLATURE: DIESEL FIRE PUMP P1A SUCTION
LOCATION: FPH
MANUFACTURE: MARSH
MODEL: J9024
RANGE IN/OUT: -30 to 300/-30 to 300
ACCURACY: ± 1% F.S.
SERIAL#
HEAD CORRECTION:
OUTPUT TOLERANCE: ± 12.0in.Hg/6.3PSIG
LOOP DIAG: 1FPW-PI
MFG. INST. BOOK:
UNITS IN: in. Hg/PSIG
UNITS OUT: in. Hg/PSIG
Scaling

REQUIRED		LIMITS		AS FOUND		AS LEFT	
IN	OUT	UPPER	LOWER	INC	DEC	INC	DEC

SWITCH		REQUIRED	LIMITS		AS FOUND	AS LEFT
No. 1	TRIP					
	RESET					
No. 2	TRIP					
	RESET					

SWITCH No.1 ACTION:
SWITCH No.2 ACTION:

M&TE ID#	CAL DUE DATE

M&TE ID#	CAL DUE DATE

COMMENTS:

ACCURACY IS EQUAL TO OR BETTER THAN THE ACCURACY OF THE DEVICE UNDER TEST. /
Initial

Nuclear Document Con

JUN 1 6 1981

0013

III-ATB-3

QA CAT: _

DATA CARD REV: _

PNEUMATIC VALVE DAMPER DATA CARD

VALVE MARK NO: ____________ TYPE: ____________ SYSTEM CODE: ____________

NOMENCLATURE: ____________ LOCATION: ____________

: ____________ SOV: ____________ LOOP DIAG: ____________

VALVE DATA

MANUFACTURE: ____________ MODEL NO: ____________ SERIAL NO: ____________

RANGE: ____________ SUPPLY PRESS: ____________ BENCH SET: ____________

ACTION AIR TO: ____________ STEM TRAVEL: ____________

AIR FAILURE POSITION: ____________ ELECT FAILURE POSITION: ____________

VALVE OR DAMPER				
	POSITION AS FOUND		POSITION AS LEFT	
INPUT	INC	DEC	INC	DEC

M&TE ID#	CAL DUE DATE

LIMIT SWITCH DATA		
	OPEN	CLOSE
MFG		
MODEL		

COMMENTS: ____________

LIMIT SWITCHES SET ______ / ______ INITIAL

______ / ______ TECHNICIAN DATE

______ / ______ APPROVED D

III-ATB-4

ECN

QA CAT

DATA CARD REV

PNEUMATIC VALVE/DAMPER AND POSITIONER DATA CARD

VALVE MARK NO. ______ TYPE: ______ SYSTEM CODE ______

NOMENCLATURE: ______ LOCATION ______

______ SOV: ______ LOOP DIAG. ______

VALVE DATA

MANUFACTURE: ______ MODEL NO: ______ SERIAL NO: ______

RANGE: ______ SUPPLY PRESS: ______ BENCH SET: ______

ACTION AIR TO: ______ STEM TRAVEL: ______

AIR FAILURE POSITION: ______ ELECT FAILURE POSITION: ______

POSITIONER DATA

MANUFACTURE: ______ MODEL NO: ______ SERIAL NO: ______

INPUT SIG: ______ SUPPLY PRESS: ______

POSITIONER					VALVE OR DAMPER POSITION			
	OUTPUT AS FOUND		OUTPUT AS LEFT		AS FOUND		AS LEFT	
INPUT	INC	DEC	INC	DEC	INC	DEC	INC	DEC

LIMIT SWITCH DATA		
	OPEN	CLOSE
MFG		
MODEL		

M&TE ID#	CAL DUE DATE

COMMENTS:

011

III-ATB-5

LCR NO:

QA CAT: ______

DATA CARD REV ______

SOLENOID VALVE DATA CARD

SOLENOID VALVE MARK NO: ______ LOOP DIAG: ______ SYSTEM CODE: ______

NOMENCLATURE: ______ LOCATION: ______

MANUFACTURE: ______ MODEL: ______ MFG INST BK ______

ORIFICE SIZE: ______ PIPE SIZE: ______ MEDIA : ______

VOLTAGE: ______ CYCLES: ______ WATTS: ______

ELEMENTARY: ______

VERIFICATION OF PROPER TUBING: ______

FUNCTIONAL LOOP CHECK: ______

LEAK CHECK OF S.O.V.: ______

DATE:	REMARKS:

TECHNICIAN: ______ DATE: ______ TEST ENGINEER: ______ DATE: ______

0012

III-ATB-6

LOOP CALIBRATION DATA CARD

LCR NO ______

QA CAT: ______

DATA CARD REV: ______

SYSTEM CODE: ______

NOMENCLATURE: ______

INJECTION POINT	MONITOR POINT		MONITOR POINT		MONITOR POINT		MONITOR POINT		MONITOR PO	
REQUIRED INPUT ( ) ± ( )	REQ OUT ± ( )	ACT OUT ( )	REQ OUT ± ( )	ACT OUT ( )	REQ OUT ± ( )	ACT OUT ( )	REQ OUT ± ( )	ACT OUT ( )	REQ OUT ± ( )	AC (
SIMULATOR I.D. NO	MONITOR I.D. NO		MONITOR I.D. NO		MONITOR I.D. NO		MONITOR I.D. NO		MONITOR I.D NO	

LOOP NOISE: ______

ANNUNCIATOR ALARM(S): ______

COMPUTER ALARM(S): ______

REMARKS :

CONTROL ACTION(S): ______

8014

III-ATB-7

CONTROLS EQUIPMENT REMARKS CARD

EQUIPMENT MARK NO: ______________________ LCR NO: __________

DATE	REMARK:

Unit	System	Instrument No.	Location
Instrument Name			

	Date	Time	Reset	Rate	Pb	Initial	Reason for Changing

TVA 6196A (PP-2-77) CONTROLLER DATA - WBNP

INST. NO.				CALIBRATION NO.				CALIBRATION NO.			
INST. NAME				DATE IM				DATE IM			
LOCATION				TEID				TEID			
CALIBRATION POINTS				DATE: OUT IN				DATE: OUT IN			
				TIME: OUT IN				TIME: OUT IN			
INPUT ( )		OUTPUT ( )		AS FOUND		AS LEFT		AS FOUND		AS LEFT	
				INC	DEC	INC	DEC	INC	DEC	INC	DEC
		±									
		±									
		±									
		±									
		±									
INST. NO.	SETPOINT	RESET	ACCURACY	SET	RESET	SET	RESET	SET	RESET	SET	RESET
			±								
			±								
REMARKS & PRECAUTIONS:				REMARKS:							

TVA 6192A (PP-2-77) CALIBRATION DATA - WBNP

UNIT NO.	SYSTEM	INSTRUMENT NO.	LOCATION	MFG.
VALVE NOMENCLATURE		VALVE AIR FAILURE POSITION		MODEL - TYPE
INPUT PRESSURE RANGE	☐ AIR TO OPEN ☐ AIR TO CLOSE	VALVE STROKE	TYPE CAM	REMARKS
SOLENOID VALVE(S) & FUNCTION(S)		RANGE		
		PROPORTIONAL BAND		
		RESET		

TVA 6195A (PP-2-77) VALVE CALIBRATION DATA - WATTS BAR NUCLEAR PLANT

NUCLEAR PLANT - LOOP CALIBRATION SHEET

LOOP NO. ______
DATE ______
PAGE ______ of ______

LOOP NAME ______

DATE: OUT ______ IN ______ I.M. ______

TIME: OUT ______ IN ______ COMMON TEST EQUIP. I.D. ______

INPUT (±)		LOOP COMPONENTS														
I.D. NO.																
T.E. I.D.																
HEAD		REQUIRED ( )	AS FOUND		AS LEFT		REQUIRED ( )	AS FOUND		AS LEFT		REQUIRED ( )	AS FOUND		AS LEFT	
			INC	DEC	INC	DEC		INC	DEC	INC	DEC		INC	DEC	INC	DEC
1		±					±					±				
2		±					±					±				
3		±					±					±				
4		±					±					±				
5		±					±					±				

	LOOP COMPONENTS														
I.D. NO.															
T.E. I.D.															
	REQUIRED ( )	AS FOUND		AS LEFT		REQUIRED ( )	AS FOUND		AS LEFT		REQUIRED ( )	AS FOUND		AS LEFT	
		INC	DEC	INC	DEC		INC	DEC	INC	DEC		INC	DEC	INC	DEC
1	±					±					±				
2	±					±					±				
3	±					±					±				
4	±					±					±				
5	±					±					±				

LOOP SETPOINT COMPONENTS								
I.D. NO.	SETPOINT ( )	RESET	ACCURACY	AS FOUND		AS LEFT		T.E. I.D.
				SET	RESET	SET	RESET	
			±					
			±					
			±					
			±					
			±					
			±					

OTHER LOOP COMPONENTS	

REMARKS: ______

NUCLEAR PLANT

INSTRUMENT INITIAL TEST SHEET

INSTRUMENT NO ______________ LOOP NO ______________ DATE ______________

MANUFACTURER ______________ MODEL NO ______________ S/N ______________

LOCATION OF TEST ______________ PHYSICAL APPEARANCE ______________

REMARKS ______________

		AUTHORIZED BY OR WHERE OBTAINED
TYPE OF TEST		
INPUT RANGE		
OUTPUT RANGE		
ACCURACY REQUIRED		
HEAD		
TEST PROCEDURE USED		
SETPOINTS	N/A	

INPUT ( )	OUTPUT ( )	AS FOUND		AS LEFT		T.E. ID
		INC	DEC	INC	DEC	
	±					INPUT
	±					
	±					OUTPUT
	±					-
	±					

INSTRUMENT NO.	SETPOINT ( )	RESET	ACCURACY	AS FOUND		AS LEFT		T.E. ID
				SET	RESET	SET	RESET	
			±					
			±					

DISPOSITION ______________

TEST PERFORMED BY ______________

REVIEWED BY ______________

CALIBRATION DATA RECORD

JOB ______________________________

INST. I.D. ______________ MANUF. ______________

SERVICE ______________________________

MOD. NO: ______________ SER. NO: ______________

INPUT: ______________ OUTPUT: ______________

CAL. DATA: ______________________________

NAMEPLATE DATA: ______________________________

TEST EQUIPMENT	MODEL	S/N

INPUTS ( )		OUTPUTS ( )		
%	VALUE	DESIRED	AS FOUND	AS LEFT

REMARKS: ______________________________

____

____

____

PERFORMED BY: ______________ DATE: ______________

Form Cal-A Rev 4/80

INST. NO.__________

EQUIPMENT HISTORY

MANUFACTURER______________________________ MODEL NO.____________

TYPE____________________________________ SERIAL NO.____________

INPUT RANGE________________ OUTPUT RANGE______________________

POWER/AIR SUPPLY______________ INSTRUCTION BOOK__________________

SERVICE/APPLICATION____

MAINTENANCE INTERVAL____

APPLICABLE PROCEDURES____

REMARKS____

MAINTENANCE RECORD

DATE	ACTIVITY	BY

Form EH Rev 7/80

VALVE CALIBRATION DATA SHEET

JOB: ____________________

INST. I.D.: ____________ MANUFACTURER: ____________

SERVICE: ____________________

MODEL NO: ____________ SERIAL NO: ____________

FAIL POSITION ____________ BENCH SET: ____________

LENGTH OF TRAVEL: ____________ LOCATION: ____________

TEST EQUIPMENT	MODEL	S/N

POSITIONER: ____________ YES: ________ NO: ________

MANUFACTURER: ____________ TYPE: ____________

POSITIONER	0% PSI	50% PSI	100% PSI
AS FOUND			
AS LEFT			

STEM TRAVEL	0%	50%	100%
AS FOUND			
AS LEFT			

LIMIT SWITCH TAG NO.: ____________ OPEN: ________ CLOSED: ________

REMARKS: ____________________

____

____

PERFORMED BY: ____________________ DATE: ____________

Cal-V 7/80

CALIBRATION DATA RECORD--SWITCH

JOB ______________________________

INST. I.D. ______________ MANUF. ______________

SERVICE ______________________________

MOD. NO: ______________ SER. NO: ______________

NAMEPLATE DATA: ______________________________

CONTACT LOGIC: #1 ________ #2 ________ #3 ________ #4 ________

TEST EQUIPMENT	MODEL	S/N

FUNCTION	DESIRED	AS FOUND	AS LEFT
SET #1			
RESET #1			
SET #2			
RESET #2			
SET #3			
RESET #3			
SET #4			
RESET #4			

If additional setpoints, use second sheet.

REMARKS: ______________________________

PERFORMED BY: ______________________ DATE: ______________

*CR=closes rising
CF=closes falling
OR=opens rising
OF=opens falling

Form CAL-S Rev 4/80

LOOP CALIBRATION DATA SHEET

INPUT TO		OUTPUT FROM				OUTPUT FROM				OUTPUT FROM			
		MEASURE AT				MEASURE AT				MEASURE AT			
STEP #													
% SPAN	UNITS	DESIRED	FOUND	ERROR	LEFT	DESIRED	FOUND	ERROR	LEFT	DESIRED	FOUND	ERROR	LEFT
PERMISSIBLE ERROR													
RECORDER INITIALS													

REMARKS: ______________________________

PERFORMED BY :______________________ DATE: ______________

III-ATB-15

INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

Unit IV - Seismic Design and Construction Practices

PRE-STUDY TEXT

INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION PLANT INSPECTORS

Unit IV - Seismic Design and Construction Practices

PRE-STUDY TEXT

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INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

Unit IV - Seismic Design and Construction Practices

PRE-STUDY TEXT

1.0 OBJECTIVE

To provide the USNRC inspector with some fundamental understanding and basic engineering knowledge of seismic theory and seismic qualification requirements for Class 1E equipment and specific non-Class 1E equipment requiring seismic withstand capability whose seismic induced failure might otherwise jeopardize a safety system function. This fundamental study effort will enhance the inspector's knowledge of seismic design and construction practices.

2.0 INTRODUCTION

The theories of seismic or earthquake activity, the resultant movement of the earth and its propagation and measurement, in general, is discussed. The susceptibility of instrumentation and control systems and components to vibration is reviewed. Spectral models and typical forces are presented.

Much of the information contained in this course is generally of a technical background nature only. Consequently, it provides a "feel" for seismology which is useful, but may not be directly applicable to construction inspectors. Therefore, the inspector is not expected to have an indepth knowledge of all the aspects of this program.

2.1 Seismic Engineering

The concepts and engineering criteria for the safe shutdown earthquake (SSE) and the operating basis earthquake (OBE) are reviewed as is response spectrum (e.g., required, test, and fragility) and other related concepts.

2.2 Seismic Qualification and Certification

Seismic qualification and certification requirements for Class 1E equipment are reviewed. Methods to demonstrate compliance by analysis, proof testing, or a combination of both analysis and proof testing requirements as they affect instrumentation and control devices are reviewed.

3.0 SEISMIC DESIGN AND CONSTRUCTION PRACTICES

10CFR50, Appendix A, General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena" requires that nuclear power plant

structures, systems, and components important to safety be designed to withstand the effects of earthquakes without loss of the capability to perform their safety function.

Furthermore, 10CFR100, "Reactor Site Criteria", Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants", provides the principle seismic and geologic considerations" relating to 10CFR50, Appendix A, GDC 2.

Contained within 10CFR50, Appendix A, Paragraph II, "Scope", is the following statement:

> "These criteria do not address investigations of volcanic phenomena required for sites located in areas of volcanic activity. Investigations of the volcanic aspects of such sites will be determined on a case-by-case basis".

Although it is credible that a nuclear power plant might be located within a volcanic activity area (e.g., Hawaii), such locations and associated requirements are sufficiently unique as to preclude coverage in this training program.

A possible result of seismic activity would be tsunami run-up or seismically induced floods and water waves as described in 10CFR100, Appendix A, Paragraph IV (c), "Required Investigation for Seismically Induced Floods and Water Waves". This type of flooding or water wave, if credible for a particular site, will typically be precluded from causing common mode failures of instrumentation and control (I&C) safety system equipment by the buildings, structures, and/or equipment location provided for the I&C equipment. Furthermore, the USNRC staff reviewers of the applicant's SAR will ascertain that the safety-related Class 1E equipment is reasonably protected from seismically induced floods and water waves. Consequently, the inspector's normal construction inspection routine to verify installation in accordance with "approved-for-construction drawings", which reflect SAR commitment, should provide reasonable assurance that the failure of safety-related I&C equipment from the cause is not credible. Therefore, this aspect of seismic effects is precluded from "in-depth" coverage in this training program.

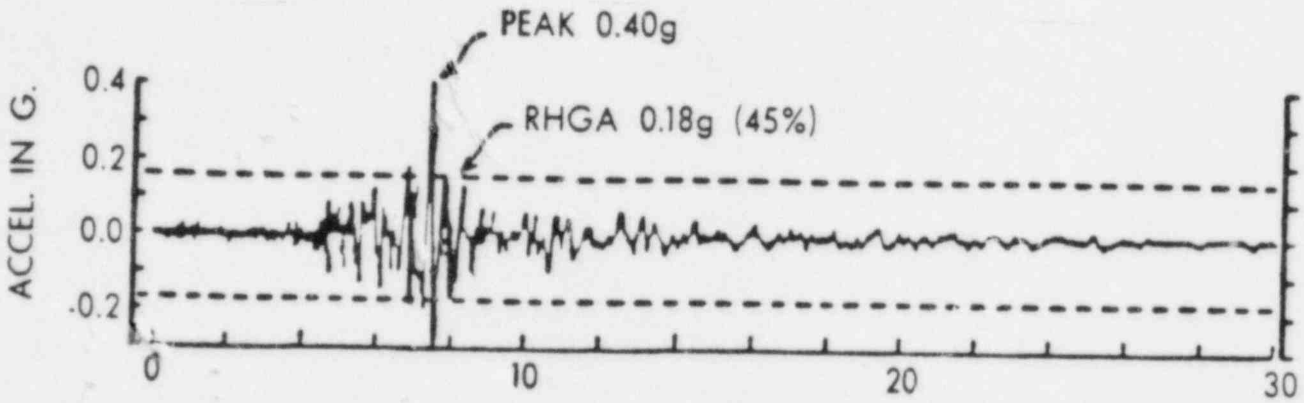
3.1 Seismic Activity Measurement

Vibrations produced by earthquakes are detected. measured, and recorded by instruments called seismographs. Seismographs record the varying amplitudes of the vibrations by responding to the motion of the ground beneath the instrument. The resultant zigzag trace recorded by a seismograph - called a seismogram - reflects the varying amplitude of the vibrations produced at that location by the earthquake.

Figure IV-6-1 is a typical accelerogram from a station close to an earthquake-causing fault. Such accelerograms typically have one

or two peaks or spikes which are considerably higher than the rest of the record. Below the peak are several high peaks which are repeated a significant number of times. The average of these secondary peaks or spikes is defined as the Repeatable High Ground (or Bedrock) Acceleration (RHGA).

Figure IV-6-1
A Typical Accelerogram



Time in Seconds

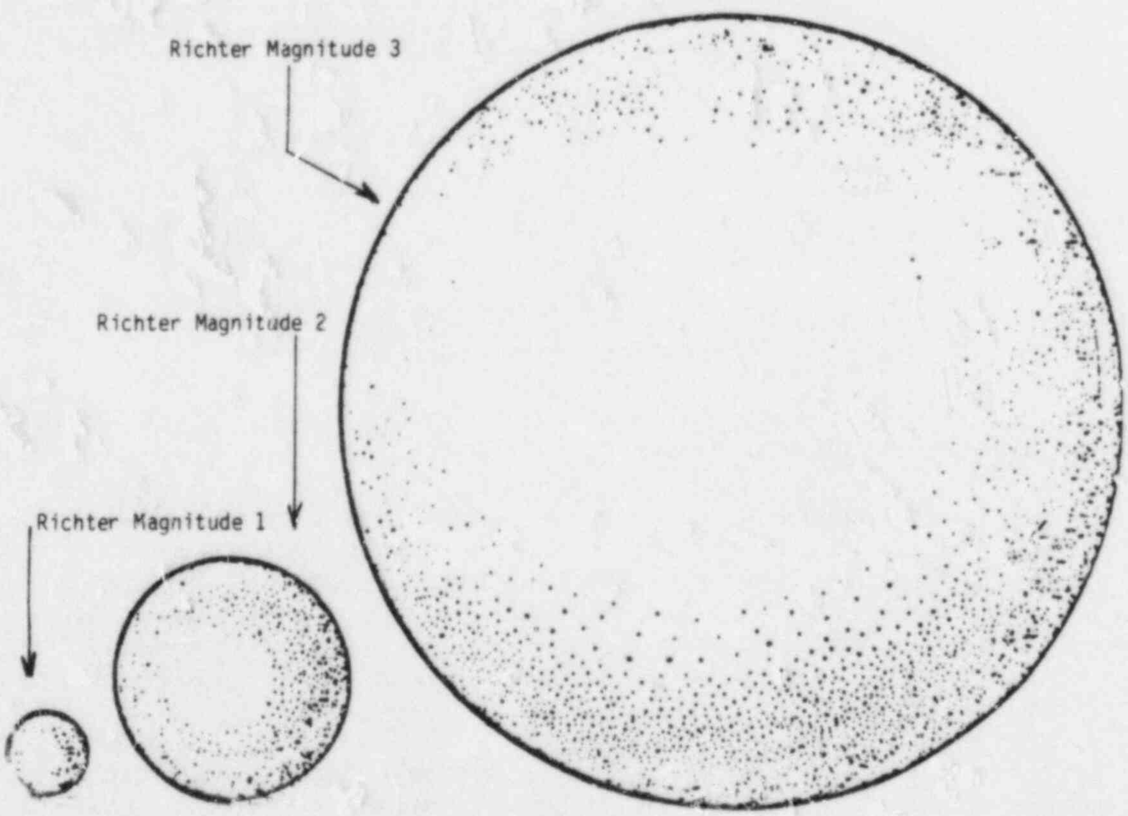
3.2 Seismic Magnitude and Intensity

As most earthquakes do not originate directly at or below the surface of the site or location of interest, an understanding of seismic propagation mechanisms is beneficial for an understanding of earthquake effects. From the data expressed in seismograms, estimates can be made of the amount of energy released. The epicenter and the rate of seismic vibration can also be determined from the seismogram.

Figure IV-6-2 illustrates the relationship between earthquake magnitude and energy. The magnitude is given using the Richter scale.

Figure IV-6-2
Relationship Between Earthquake Magnitude and Energy

The volumes of the spheres are roughly proportional to the amount of energy released by earthquakes of the magnitude given and illustrate the exponential relationship between magnitude and energy. At the same scale, the energy released by the San Francisco earthquake of 1906 (Richter magnitude 8.3) would be represented by a sphere with a radius of 110 feet.



The severity of an earthquake can be expressed in several ways. The magnitude of an earthquake, as expressed by the Richter magnitude scale, is a measure of the amplitude of the seismic waves. The amplitude is measured on seismograph recordings. When the earth quakes, the amplitude of the wave recorded on the seismograph is measured and is then corrected mathematically to what the amplitude would have been if it had been recorded at a distance of 100 kilometers from the epicenter. The Richter magnitude derived from these corrected seismograph recordings indicates the amount of energy released as if it had been recorded at this standard 100-kilometer distance from the epicenter of the quake. The intensity, as expressed by the Modified Mercalli intensity scale, is a partly subjective measure which depends on the effects of a quake such as damage at a particular location.

The Richter magnitude scale, named after Dr. Charles F. Richter, Professor Emeritus of the California Institute of Technology, measures the energy of an earthquake, and is the scale most commonly used, but often misunderstood. On this scale, the earthquake's magnitude is expressed in whole numbers and decimals. However, Richter magnitudes can be confusing and misleading unless the mathematical basis for the scale is understood. It is important to recognize that magnitude varies logarithmically with the wave amplitude. The amplitude of an 8.0 magnitude earthquake is not twice as large as a shock of magnitude 4.0, but 10,000 times as large. Correspondingly, a magnitude 8.0 earthquake releases almost one million times more energy than one of magnitude 4.0.

$$\frac{\text{Magnitude 8.0}}{\text{Magnitude 4.0}} \quad \frac{10^8}{10^4} = 10^{(8-4)} = 10^4 = 10{,}000$$

A quake of magnitude 2 on the Richter scale is the smallest quake normally felt by humans. Earthquakes with a Richter magnitude of 7 or more are commonly considered to be major. The Richter magnitude scale has no fixed maximum or minimum - observations have placed the largest recorded earthquakes in the world at about 8.9, and the smallest at about .3. Earthquakes with magnitudes smaller than 2 are called "micro-earthquakes". Richter magnitudes are not used to estimate damage. An earthquake in a densely populated area, which results in many deaths and considerable damage, may have the same magnitude as an earthquake that occurs in a barren remote area, that may do nothing more than frighten the wildlife.

Modified Mercalli Intensity Scale of 1931

The first scale to reflect earthquake intensities was developed by de Rossi of Italy, and Forel of Switzerland, in the 1880's. This scale, with values from I to X, was used for about two decades. A need for a more refined scale increased with the advancement of

the science of seismology, and in 1902, the Italian seismologist, Mercalli, devised a new scale on a I to XII range. The Mercalli Scale was modified in 1931 by American Seismologists Harry O. Wood and Frank Neumann to take into account modern structural features and is provided below:

I. Not felt except by a very few under especially favorable circumstances.

II. Felt only by a few persons at best, especially on upper floors of buildings. Delicately suspended object may swing.

III. Felt quite noticeably indoors, especially on upper floors of buildings, but many people do not recognize it as an earthquake. Standing motor cars may rock slightly. Vibration like passing of truck. Duration estimated.

IV. During the day felt indoors by many, outdoors by few. At night some awakened. Dishes, windows, doors disturbed; walls make cracking sound. Sensation like heavy truck striking building. Standing motor cars rocked noticeably.

V. Felt by nearly everyone, many awakened. Some dishes, windows, etc., broken; a few instances of cracked plaster; unstable objects overturned. Disturbances of trees, poles, and other tall objects sometimes noticed. Pendulum clocks may stop.

VI. Felt by all, many frightened and run outdoors. Some heavy furniture moved; a few instances of fallen plaster or damaged chimneys. Damage slight.

VII. Everybody runs outdoors. Damage negligible in buildings of good design and construction; slight to moderate in well-built ordinary structures; considerable in poorly built or badly designed structures; some chimneys broken. Noticed by persons driving motor cars.

VIII. Damage slight in specially designed structures; considerable in ordinary substantial building, with partial collapse; great in poorly built structures. Panel walls thrown out of structures. Fall of chimneys, factory stacks, columns, monuments, walls. Heavy furniture overturned. Sand and mud ejected in small amounts. Changes in well water. Persons driving motor cars disturbed.

IX. Damage considerable in specially designed structures; well-designed frame structures thrown out of plumb, great in substantial buildings, with partial collapse.

Buildings shifted off foundations. Ground cracked conspicuously. Underground pipes broken.

X. Some well-built wooden structures destroyed; most masonry and frame structures destroyed with foundations; ground badly cracked. Rails bent. Landslides considerable from river banks and steep slopes. Shifted sand and mud. Water splashed (slopped) over banks.

XI. Few, if any, (masonry) structures remain standing. Bridges destroyed. Broad fissues in ground. Underground pipelines completely out of service. Earth slumps and land slips in soft ground. Rails bent greatly.

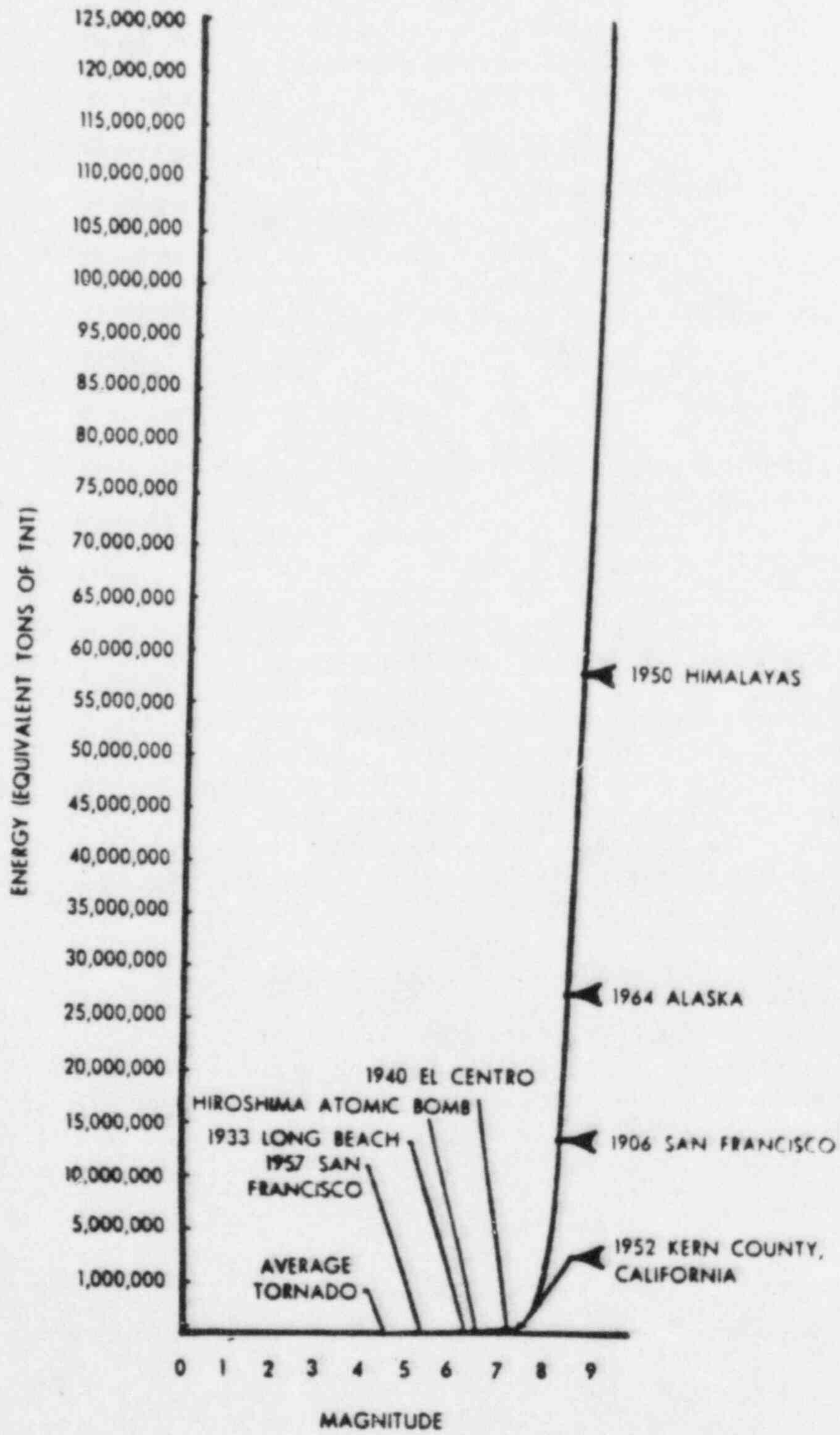
XII. Damage total. Practically all works of construction are greatly damaged or destroyed. Waves seen on ground surface. Lines of sight and level are distorted. Objects are thrown upward into the air.

The Modified Mercalli Intensity Scale measures the intensity of an earthquake's effects in a given locality, and is perhaps much more meaningful to the layman because it is based on actual observations of earthquake effects at specific places. It should be noted that because the data used for assigning intensities can be obtained only from direct first hand reports, considerable time - weeks or months - are sometimes needed before an intensity map can be assembled for a particular earthquake. On the Modified Mercalli Intensity Scale, values range from I to XII. The most commonly used adaptation covers the range of intensity from the conditions of "I - not felt except by very few, favorably situated", to "XII - damage total, lines of sight disturbed, objects thrown into the air". While an earthquake has only one magnitude, it can have many intensities, which decrease with distance from the epicenter.

Comparison of Magnitude and Intensity

It is difficult to compare magnitude and intensity because intensity is linked with the particular ground and structural conditions of a given area, as well as distance from the earthquake epicenter, while magnitude is a measure of the energy released at the focus of the earthquake. (See Figure IV-6-3.)

Figure IV-6-3
Earthquake Magnitude Versus Intensity



A chart helps to visualize the vast differences in energy represented by the various earthquake magnitudes. Energy is shown on the vertical scale in terms of equivalent tons of TNT. For example, a 6.35 magnitude earthquake is approximately equivalent to the energy of the Hiroshima atomic bomb (20,000 tons of TNT). The San Francisco earthquake of 1906 (magnitude 8.25) released about the same energy as a 15 megaton hydrogen bomb, and the 1964 Anchorage earthquake (magnitude 8.5) approximately represents the energy of a 32 megaton hydrogen bomb.

The most vigorous earthquakes recorded by seismographs were the 1906 earthquake off the coast of South America, and the 1933 earthquake of Japan, both of which had magnitude ratings of 8.9. Neither earthquake was a catastrophe, as it struck where there were no cities to be leveled or people to be injured. Neither earthquake had an intensity, since there was no one to be hurt by it, or even to feel it.

Two of the world's greatest historical shakes, the New Madrid (Missouri) earthquakes of 1811-12, and the Lisbon (Portugal) disaster of 1755, are not plotted here because they occurred before there were seismographs to measure them. Figure IV-6-4 provides a good comparison bewteen Richter magnitude "M", Mercalli intensity "I", total energy "E" and acceleration "A".

3.3 Seismic Propagation Characteristics

Studies of information on actual ground motion, provided by specially designed seismographs called accelerographs, which record damage level intensities of ground shaking produced by strong earthquakes, have substantiated the following characteristics for seismic waves.

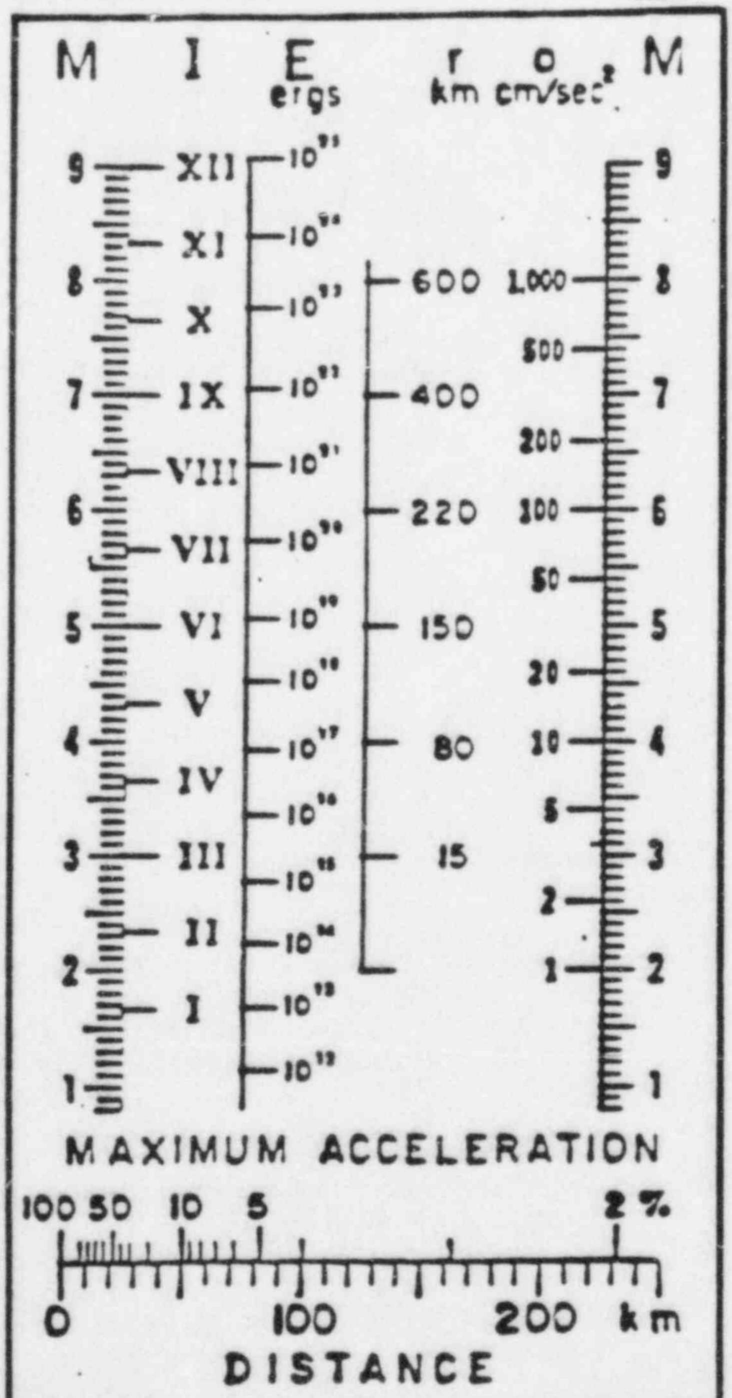
Any seismic disturbance on or beneath the earth's surface, whether an earthquake or a blow from a sledge hammer, will produce a variety of seismic waves.

There are two main types of seismic waves; (1) body waves which travel through the earth, and (2) surface waves which are confined to a zone near the earth's surface. Engineering seismologists believe body waves are the main contributors to strong ground motion near an earthquake.

Body waves consist of two types: compressional (P) and shear (S) waves. Compressional waves produce a back-and-forth or a "push-pull" traveling through air. P waves travel faster than S waves and have velocities in surface earth materials ranging from about 1,000 to 20,000 feet per second (fps). Shear waves are characterized by motion transverse to the direction of wave travel, much like the wave that travels down a taut rope when one end is suddenly jerked sideways. These waves travel slower than P waves and

have velocities in earth materials ranging from 400 to about 12,000 fps. Shear waves are believed to be the primary cause of damage to man-made structures. The shear waves arrive after the compressional waves and are greater in amplitude but shorter in duration.

Figure IV-6-4



3.4 Seismic Engineering

3.4.1 OBE and SSE

Two earthquake environments are identified and specified for the vibratory ground motion of a typical nuclear site:

OBE (Operating Basis Earthquake) is that earthquake which could reasonably be expected to affect the plant site during the operating life of the plant. It is that earthquake which produces the vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional.

SSE (Safe Shutdown Earthquake) is that earthquake which produces the maximum vibratory ground motion for which certain structures, systems and components are designed to remain functional. These structures, systems and components are those necessary to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capacity to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposure comparable to the guideline exposure of the Code of Federal Regulations, Title 10, Part 100 (December 5, 1973).

3.4.2 Structure and Component Classes

The method of analysis used to determine seismic loadings on structures is a function of their relative importance to safe reactor operation and shutdown. It is for this reason that structures and components are normally divided into two categories.

- Seismic Category 1

  Those structures, systems, and components that should be designed to remain functional if an SSE occurs.

  USNRC Regulatory Guide 1.29 (Rev. 3, 09/78), "Seismic Design Classification", provides seismic-related aspects pertinent to this electrical effort and are paraphrased as follows:

  -- USNRC Regulatory Guide Position C.1r-Class 1E systems necessary for a safety function to be performed is to be classified as Seismic Category 1. Seismic Category 1 equipment is required to remain functional after a safe shutdown earthquake (SSE).

-- USNRC Regulatory Guide Position C.2 - Equipment whose continued function is not required but whose seismically induced failure can reduce the ability of a safety system from functioning, should be designed and constructed such that an SSE cannot cause such a failure. Practical applications of this requirement are described in Paragraph 3.7 below:

-- USNRC Regulatory Guide Position C.4 - Pertinent quality assurance requirements should be applied to all activities affecting the safety-related functions of the items covered in Position C.2 above.

- Non-Seismic Category 1

All other structures, systems and components.

3.4.3 Definitions

Certain selected definitions are included herein:

Period - The time it takes for a motion to repeat itself.

Frequency - The number of complete cycles of motion in a unit of time. Frequency is the reciprocal of period.

Response - The response of a vibrating body is the resultant displacement, velocity, or acceleration to an excitation.

Single Degree of Freedom System - A system in which only one coordinate is necessary to completely describe the motion of the system. (See Figure IV-6-5.)

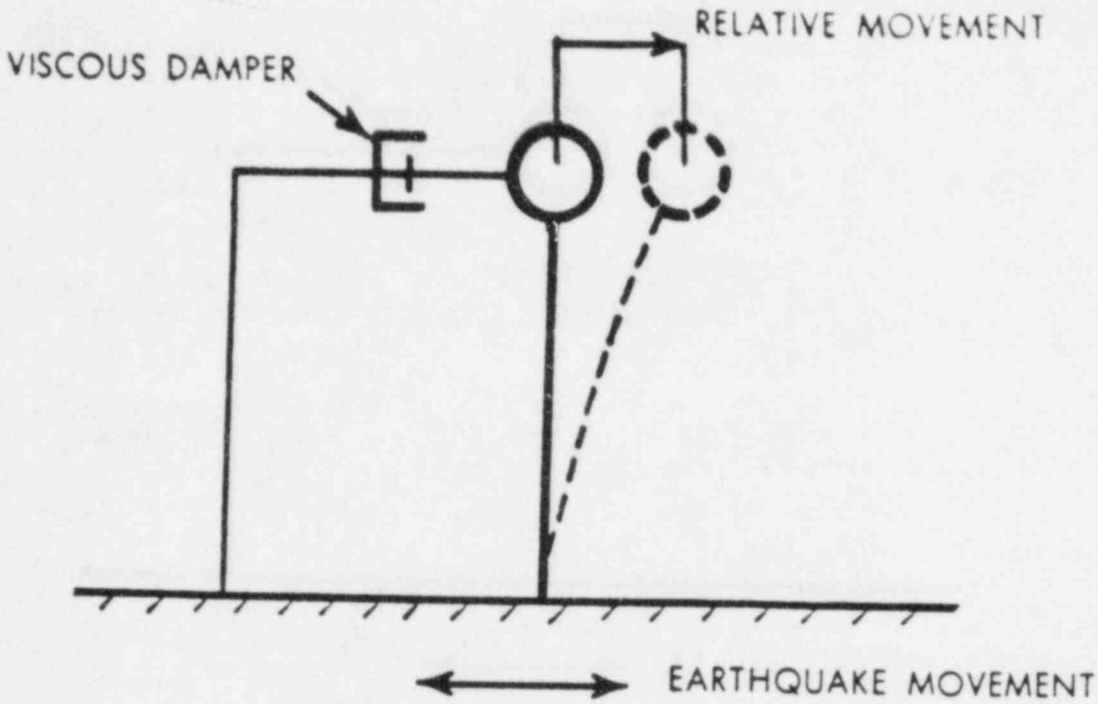


Figure IV-6-5
Single-Degree-of-Freedom Model for
Studying Earthquake Response of Equipment

Natural Frequency - The frequency or frequencies at which a body vibrates due to its own physical characteristics (mass, shape) and elastic restoring forces when the body is distorted in a specific direction and then released.

Design Response Spectra - Logarithmic plots of the simulated seismic motion used to reproduce the postulated earthquake environment for a specific site and in a conservative manner. (See Figures IV-6-6 and IV-6-7.)

Response Spectrum - Means a plot of the maximum response (acceleration, velocity, displacement) of a family of idealized single-degree-of-freedom damper oscillators as a function of the natural frequencies (or periods) of the oscillators to a specified vibratory motion input at their supports.

Floor Response Spectrum - A response spectrum of the oscillators when their support is at the pertinent floor. (See Figure IV-6-8.)

Broadband Response Spectrum - A response spectrum that describes the motion indicating that multiple frequency excitation predominates.

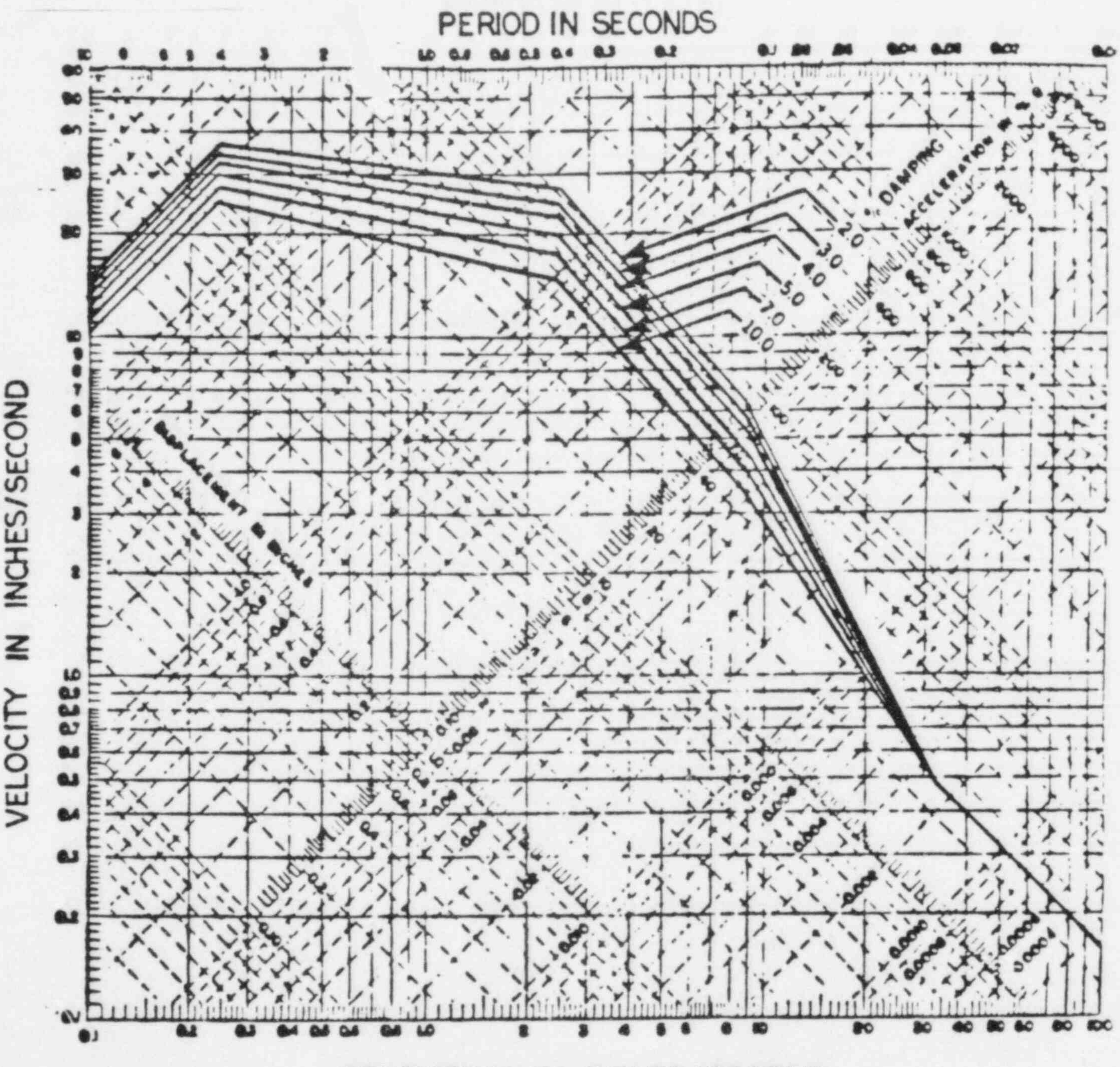
Floor Acceleration - The acceleration of a particular building floor (or equipment mounting) resulting from a given earthquake's motion. The maximum floor acceleration can be obtained from the floor response spectrum as the acceleration at high frequencies (in excess of 33 Hz) and is sometimes referred to as the ZPA (zero period acceleration).

Fragility - Susceptibility of equipment to malfunction as a result of structural or operational limitations, or both.

Fragility Level - The highest level of input excitation, expressed as a function of input frequency, that an equipment can withstand and still perform the required Class 1E functions.

Fragility Response Spectrum (FRS) - A TRS (test response spectrum) obtained from tests to determine the fragility level of equipment. (See Test Response Spectrum.)

Figure IV-6-6



PERIOD IN SECONDS
VELOCITY IN INCHES/SECOND
FREQUENCY IN CYCLES/SECOND
DAMPING
ACCELERATION
DISPLACEMENT IN INCHES
2.0
3.0
4.0
5.0
7.0
10.0

Figure IV-6-7

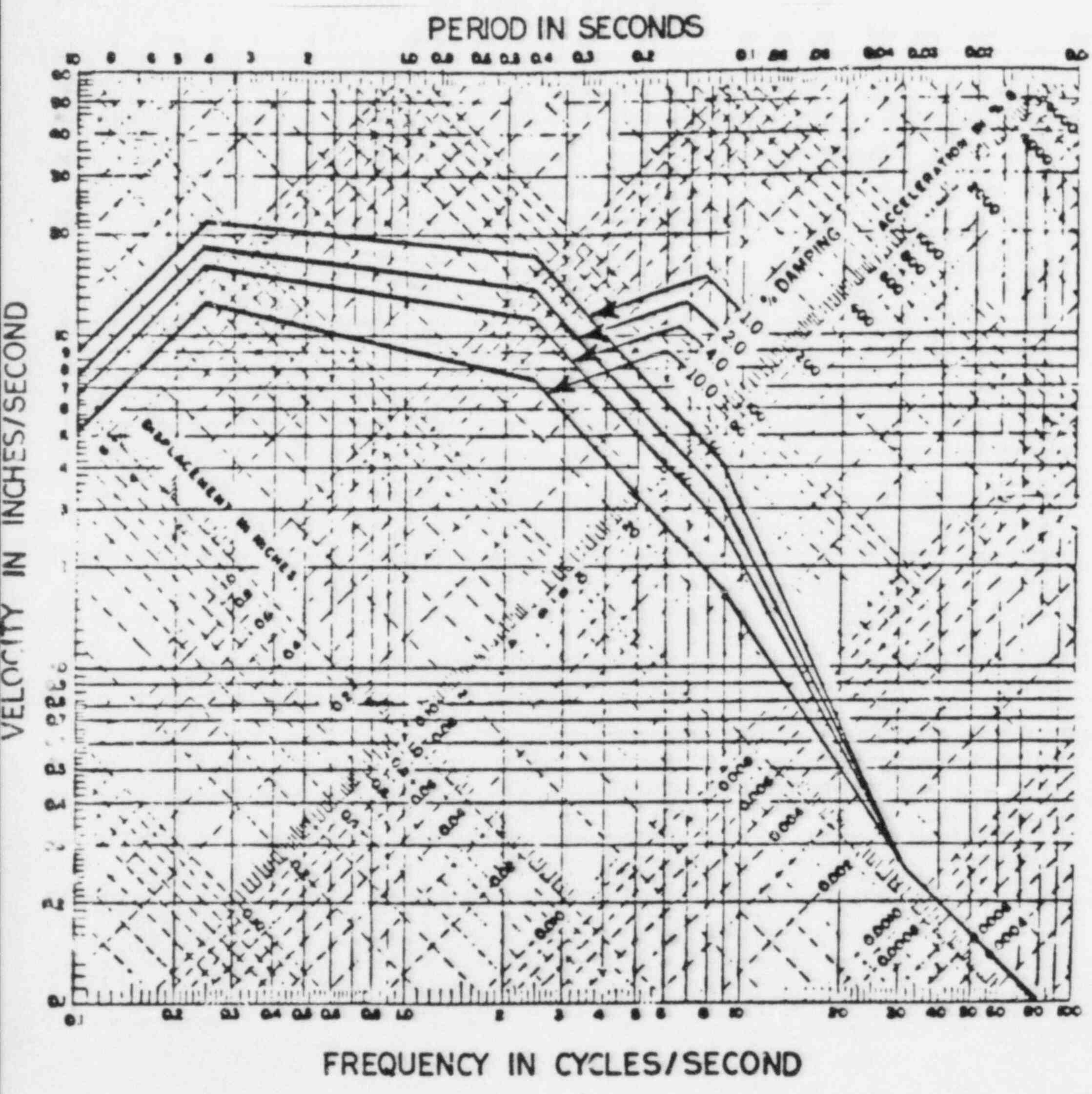
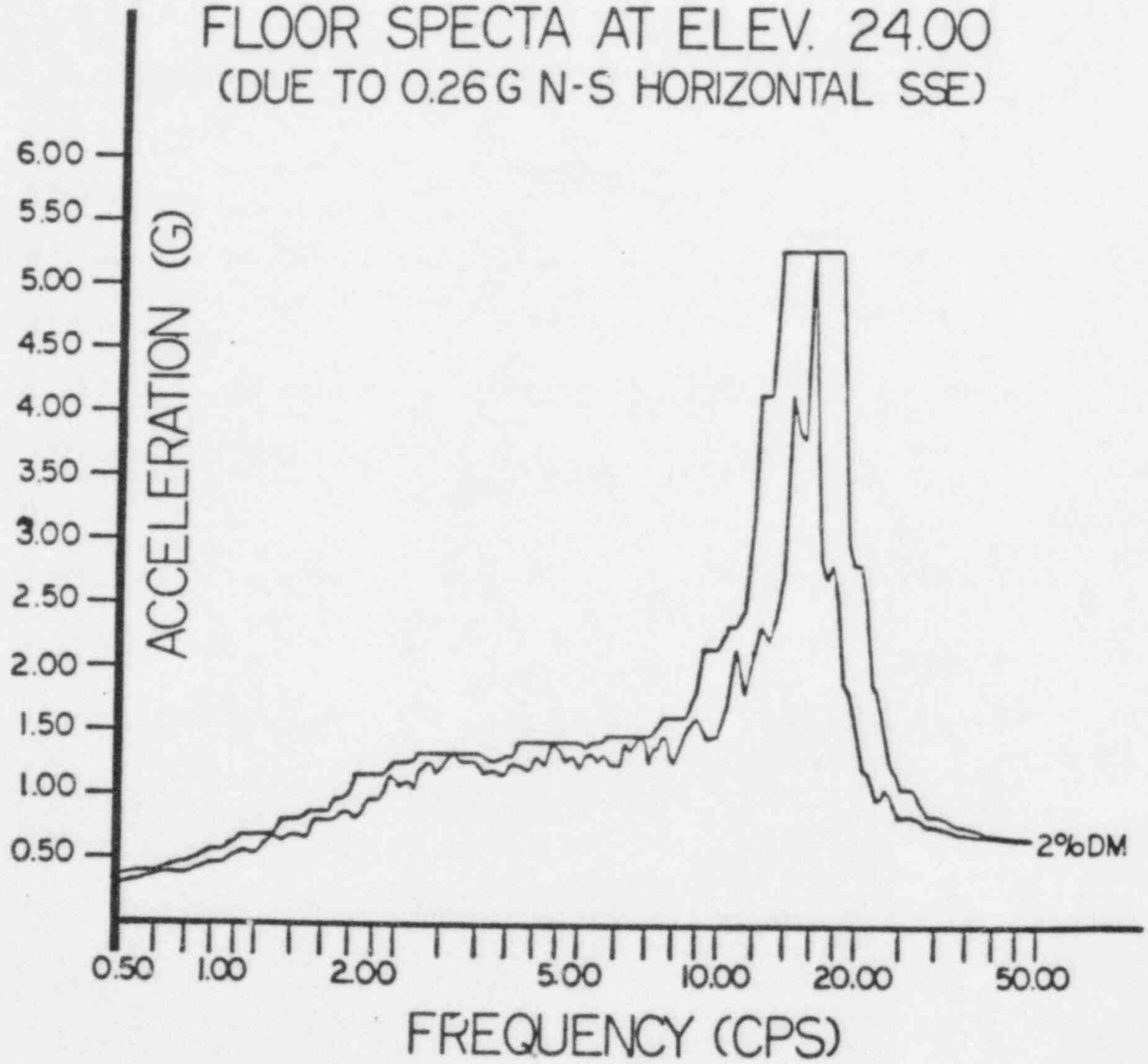

PERIOD IN SECONDS
VELOCITY IN INCHES/SECOND
FREQUENCY IN CYCLES/SECOND
% DAMPING
ACCELERATION
DISPLACEMENT IN INCHES
1.0
2.0
4.0
10.0

Figure IV-6-8
Diesel Generator Building-Floor Response Spectra



Ground Acceleration - The acceleration of the ground resulting from a given earthquakes' motion. The maximum ground acceleration can be obtained from the ground response spectrum as the acceleration at high frequencies (in excess of 33 Hz).

Low-cycle Fatigue - A progressive fracture or cumulative fatigue of the material which may occur for less than 1000 cycles because of localized stress concentration at high strains under fluctuating loads.

Narrow Band Response Spectrum - A response spectrum that describes the motion indicating that a single frequency excitation predominates.

Octave - The interval between two frequencies which have a frequency ratio of two.

Power Spectral Density (PSD) - The mean squared acceleration density of a random waveform. PSD is usually expressed in square gauss per hertz.

Required Response Spectrum (RRS) - The response spectrum issued by the user or his agent of his specifications for proof testing, or articifially created to cover future applications. The RRS constitutes a requirement to be met.

Sine Beats - A continuous sinusoid of one frequency, whose amplitude is modulated by a sinusoid of a lower frequency.

NOTES:

1. As used in this document, the amplitudes of the sinusoids represent acceleration and the modulated frequency represents the frequency of the applied seismic stimulus.

2. Beats are usually considered to be the result of the summation of two sinusoids of slightly different frequencies with the frequencies within the beats as the average of the two, and the best frequency as one-half the difference between the two. However, as used here, the sine beats may be an amplitude modulated sinusoid with pauses between the beats.

Test Response Spectrum (TRS) - The response spectrum that is constructed using analysis or derived using spectrum analysis equipment based on the actual motion of the shake table.

NOTE: When qualifying equipment by utilizing the response spectrum, the TRS is to be compared to the RRS, using the methods described in the body of this document.

Shock Excitation - The displacement or force causing a body to vibrate when it is applied to a system.

Mode of Vibration - A pattern of motion of the individual particles due to (1) stresses applied to the body, (2) its properties, and (3) the boundary conditions.

3.4.4 Modeling of Structural Dynamics

The seismic loading applicable to the particular site on which the nuclear power station is located is specified in the form of a design response spectra which is defined in ANS 2.1. Figures IV-6-6 and IV-6-7 illustrate typical SSE and OBE design response spectra for a site where the SSE and OBE maximum horizontal ground accelerations are 0.25G and 0.13G respectively. Since the structures at the site which house and support the Class 1E systems are elastic in nature, they tend to modify or filter the ground motion. This must be taken into account by developing an analytical or dynamic model of the building structure like the one shown in Figure IV-6-9. This lumped mass model is then subjected to a particular vibratory motion which simulates the motion of the ground during an actual earthquake. The response of each floor is then determined over a range of frequencies and dampling ratios. The result, a response spectra, is plotted as shown in Figure IV-6-8.

The response spectra represents the focus of points corresponding to the maximum reponse of a series of damped single degree of freedom systems subjected to a specific ground motion.

The model shown in Figure IV-6-9 consists of lumped masses each of which is considered a single degree of freedom system, but combined, they constitute a multi-degree of freedom system. The dynamic analysis of a multi-degree of freedom system can be simplified to the solution of an equation for each mode and the total response obtained by superimposing individual modal effects. (See Figure IV-6-10.)

Figure IV-6-9
Typical Lumped Mass Model Used in Seismic Analysis

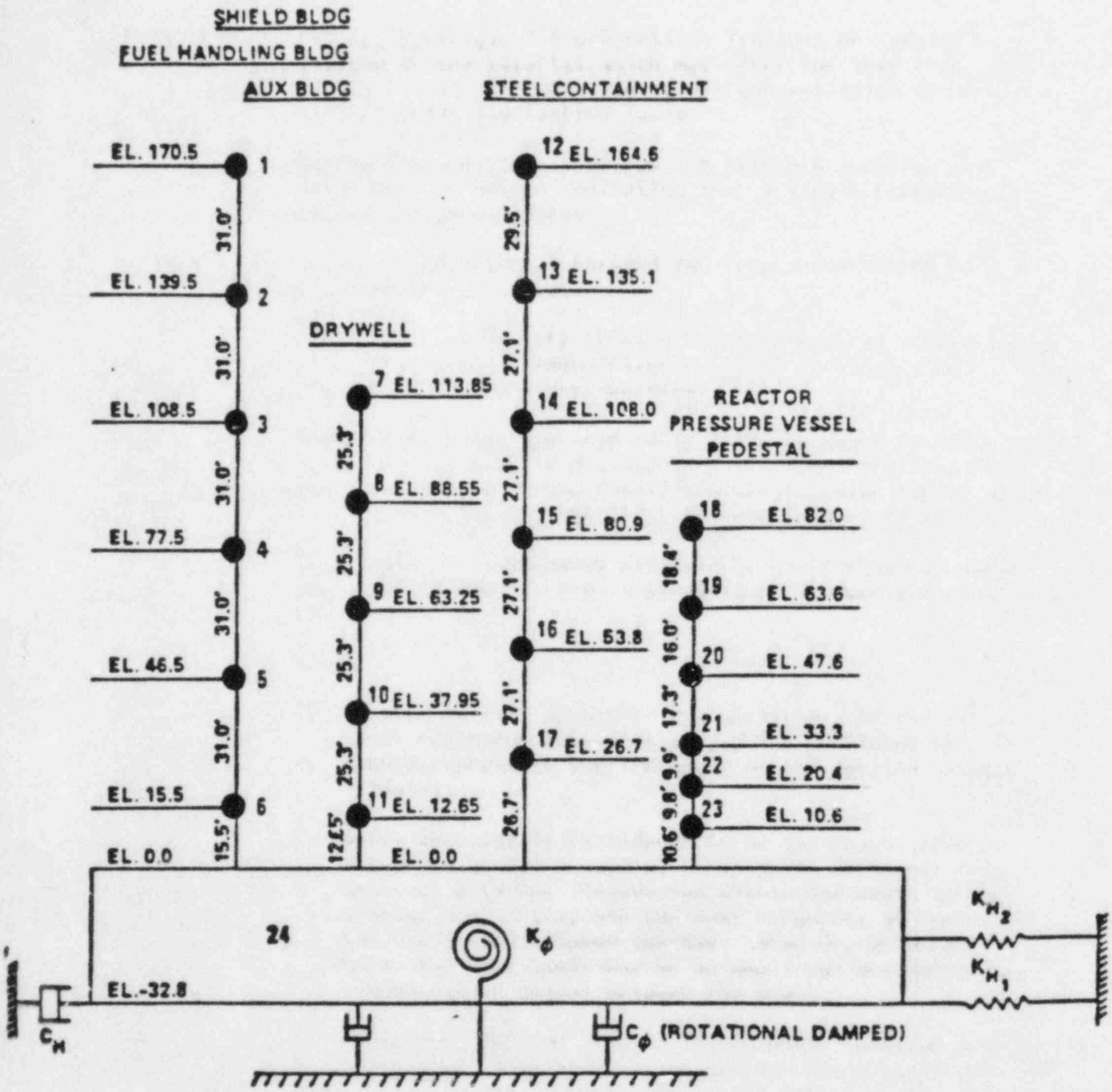
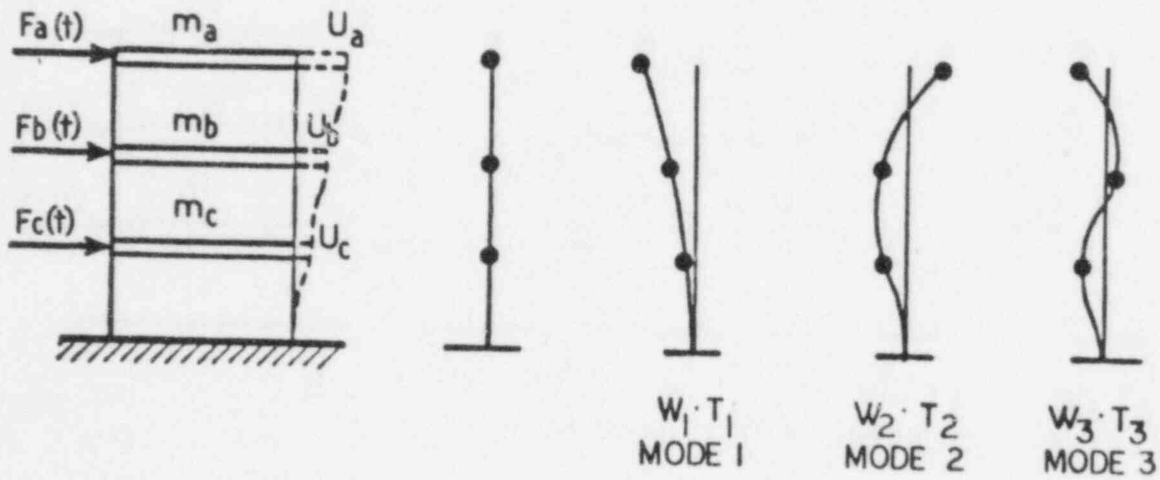


Figure IV-6-10
Multi-Degree-of-Freedom System Subjected to Dynamic Loading



An important simplification can be made in the equations of motion due to the fact that each mode has an independent equation of exactly equivalent form to that for a single degree of freedom system.

3.5 Seismic Qualification and Certification

3.5.1 Codes and Standards

The design requirements for nuclear facilities are to a considerable degree, unique. Few other systems are designed for such extreme earthquake effects which are typically in excess of any recorded historical data at a site. However, these considerations are warranted by the potential consequences of such occurrences at a nuclear facility.

As a result, certain federal regulations, and industry codes and standards have been written in order to promote sound nuclear power plant design. The following is a list of applicable nuclear regulations, codes and standards:

3.5.1.1 Federal Regulations

The most important of the NRC sponsored documents are the Federal Regulations, 10CFR100, Appendix A and 10CFR50, Appendices A and B. These documents, in addition to being design criteria developed by the NRC, also have the weight of federal law to require compliance.

- 10CFR100, Appendix A

  This federal regulation defines the operational basis and safe shutdown earthquakes and the manner in which their magnitude and impact intensities are determined. It sets the locations where the earthquake motion is defined and sets the width of fault zones as a function of fault length. It also defines capable and non-capable faults.

- 10CFR50, Appendix A

  In the document are set forth the general design criteria to be used in the design of safety class nuclear plant facilities. The following specific criteria are of particular interest for seismic design:

  Criterion 1 - Quality Standards and Records

Criterion 2 - Design Bases for Protection Against Natural Phenomena

- <u>10CFR50, Appendix B</u>

  This federal regulation presents the quality assurance criteria for nuclear power plants and fuel reprocessing plants.

3.5.1.2 <u>USNRC Regulatory Guides</u>

USNRC Regulatory Guides are prepared by the Division of Regulatory Standards and are issued to describe and make available to the public, methods acceptable to the USNRC Regulatory staff of implementing specific parts of the Commission's regulations, to delineate techniques used by the staff in evaluating specific problems or postulated accidents, or to provide guidance to applicants. USNRC Regulatory Guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions different from those set out in the guides and this program's interpretation of those guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.

The following describes the most significant USNRC Regulatory Guides:

- USNRC R.G. 1.12 - <u>Instrumentation for Earthquakes</u>

  This regulatory guide sets for the USNRC Regulatory staff position on the acceptability of ANSI N18.5, "Earthquake Instrumentation Criteria for Nuclear Power Plants", as a means of compliance with 10CFR100, Appendix A, "Reactor Site Criteria", regarding seismic instrumentation.

- USNRC R.G. 1.28 - <u>Quality Assurance Program Requirements (Design Construction</u>

  This guide references ANSI N45.2, "Quality Assurance Program Requirements for Nuclear Power Plants", as being generally acceptable and provides an adequate basis for complying with the program requirements of 10CFR50, Appendix B.

- USNRC R.G. 1.29 - Seismic Design Classifications

This guide defines those systems and components which require verification of seismic design adequacy. Such seismic design adequacy is required to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capacity to shut down the reactor and maintain it in a safe shutdown condition, and (3) prevent or mitigate consequences of accidents which could result in a radiation release in excess of 10CFR100 guideline exposures. (See Paragraph 3.4.2.)

- USNRC R.G. 1.60 - Design Response Spectra for Seismic Design of Nuclear Power Plants

This guide establishes the ground response spectra for various levels of assumed damping, which will be used to seismically qualify the structure located at ground, and to qualify time history input motions which are used to develop amplified spectra for use in the design of components located within buildng structures. Horizontal as well as vertical spectra are specified.

- USNRC R.G. 1.61 - Damping Values for Seismic Design of Nuclear Power Plants

This guide presents interim modal damping values which are currently acceptable to the NRC Regulatory staff. Values are tabulated for both the OBE and SSE seismic events.

- USNRC R.G. 1.92 - Combination of Modes and Spatial Components in Seismic Response Analysis

This guide provides a method acceptable to the NRC Regulatory staff for combining seismic response effects between modes (including closely spaced modes), as well as spatial combinations considering the independence of the directional components of the earthquake.

- USNRC R.G. 1.100 - Seismic Qualification of Electrical Equipment for (Rev 1, 8/77) Nuclear Power Plants

  This regulatory guide sets forth the NRC Regulatory staff position on the acceptability of IEEE 344, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations" as a means of compliance with 10CFR50, Appendix B, Criterion III, "Design Control", in regards to safety system electrical equipment seismic adequacy.

- USNRC R.G. 1.122 - Development of Floor Design Response Spectra for Seismic Design of Floor Support Equipment or Components

  This guide describes methods acceptable to the NRC staff for developing two horizontal and one vertical floor design response spectra at various floors or other equipment support locations of interest from the time-history motions resulting from the dynamic analysis of the supporting structure.

- USNRC R.G. 1.132 - Site Investigations for Foundations of Nuclear Power Plants

  This regulatory guide sets forth the regulatory staff position on programs of site investigations that would normally meet the needs for evaluating the safety of the site from the standpoint of the performance of foundations and earthworks under most anticipated loading conditions, including earthquakes.

3.5.1.3 IEEE Standards

- IEEE-344 - Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations

  This document describes the basic requirements for seismic qualification of Class 1E equipment and interfaces utilized as part of a safety system. Included in the document are definitions, review of earthquake environments and

associated equipment and response, seismic qualification (analysis, testing, and combined analysis and testing), and documentation as applicable to seismic testing. IEEE 344 provides generic requirements for seismic qualification as a supplement to IEEE 323.

- ANSI N18.5 1974 /ANS 2.2 1977 - Earthquake Instrumentation Criteria for Nuclear Power Plants

  This standard describes the seismic instrumentation required for surveillance requirements to determine if the design criteria of the Category 1 structures and components have been exceeded.

3.5.2 Qualification

Seismic qualification is a demonstration of functional capability (or operability) with the postulated seismic environmental loads superimposed upon the applicable operational loads. Generally, the operational loads will be those associated with the plant condition that represents the design basis for the equipment being qualified.

As IEEE 344 has not been provided to the inspector as part of the program, the following material was taken directly or paraphrased from IEEE 344-1975 - "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations".

The seismic qualification of Class 1E equipment should demonstrate an equipment's ability to perform its required function during and after the time it is subjected to the forces resulting from one SSE. In addition, the equipment must withstand the effects of a number of OBE's prior to the application of an SSE.

The most commonly used methods for seismic qualification are contained in this document; other methods, if justified, may also be used. The methods are grouped into the following three general categories:

a Predict the equipment's performance by analysis

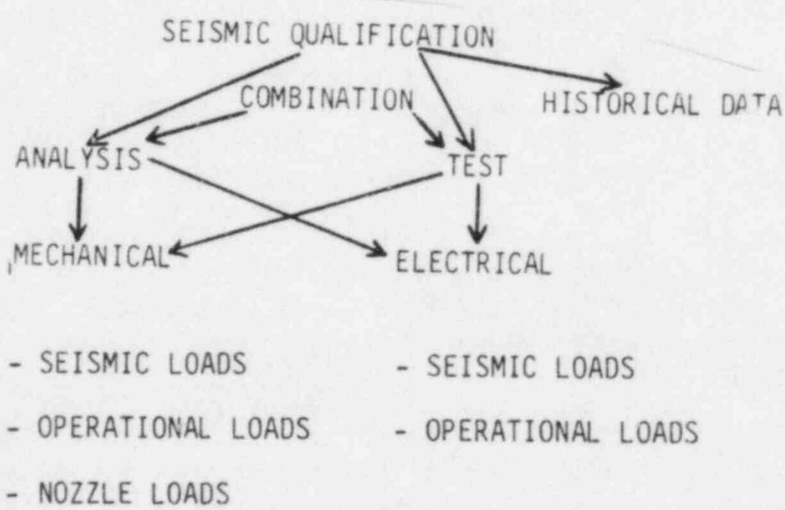
b. Test the equipment under simulated seismic conditions

c Qualify by combined test and analysis

The options for performing seismic qualification and the criteria for selecting the most practical option are

shown in Figure IV-6-11. Although the NRC has gone on record as stating a preference for the test approach to seismic qualification (see USNRC RG 1.100), one should always take into account the capability of test facilities that are available to him or to his supplier, and not specify or commit to a test approach a piece of equipment that exceeds these capabilities. In cases like this, it will generally be more practical to use an analytical approach, or a combination of test and analysis, even though more justification and explanation may be required during audit.

The second criteria is the equipment operating mode. Generally, a piece of equipment that is idle or passive during the seismic event presents a problem that more practically lends itself to an analytical solution. On the other hand, a piece of equipment that will be operating during the seismic event and must perform an active function presents a problem that more practically lends itself to a test solution.



CRITERIA

1. CAPABILITY OF AVAILABLE TEST FACILITY
2. EQUIPMENT OPERATING MODES
3. DYNAMIC CHARACTERISTICS OF FAILURE MECHANISM

Figure IV-6-11

The third criteria deals with the failure mechanism which would cause loss of the safety function. If this failure mechanism is structural in nature (e.g., maximum allowable stresses exceeded, maximum permissible deflections exceeded), an analytical approach is generally the most practical. On the other hand, if the failure mechanism is operational in nature (e.g., loss of power, interruption of signal, loss of monitoring functions, loss of control function) and there are no established procedures for modeling and analyzing these failure mechanisms, then a test approach is generally the most practical.

Each of the preceding methods, or other effective methods. may be adequate to verify the ability of the equipment to meet the seismic qualification requirements. The choice should be based on the practicality of the method for the type, size, shape, and complexity of the equipment and the reliability of the conclusions. Qualification methods may be standardized for a particular form of equipment.

Margins specified in Section 6.3.1.5 of IEEE Std. 323--1974, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations", shall be employed where applicable

3.5.2.1 Analysis

The analysis method is not recommended for complex equipment that cannot be modeled to correctly predict its response. Analysis without testing may be acceptable only if structural integrity alone can assure the design-intended function.

3 5.2.1.1 Introduction

Generally there are two approaches to seismic analysis. One approach is based on equivalent static analysis. the other on dynamic analysis. The methods described are those most commonly used, but other methods may be used if they are justifiable. Figure IV-6-12 is a flow chart of the recommended analytical process. The general procedure is to first study the equipment to assess the dynamic characteristics; second, to determine the response using one or more of several methods described in the following sections; third, to analyze the stresses which result from the response; and finally, to determine if the design is adequate.

The study stage should take into account the complexity of the equipment and the adequacy of analytical techniques to properly predict the equipment's Class 1E functions while under seismic excitation. The study should determine which method will most accurately represent the equipment's performance under seismic conditions.

The response determination phase of the analysis can take several paths, the first of which is determined by the choice between the static coefficient method or the dynamic analysis method. In general, the choice is based on the expected margin of strength of the equipment since the static coefficient method, although relatively crude, is the easier to perform of the two but is far more conservative and yields higher stresses.

The dynamic analysis or tests may indicate that the equipment is either rigid or flexible. Rigid equipment is analyzed using static analysis and the seismic acceleration associated with the mounting location. Flexible equipment, on the other hand, requires computation of the dynamic response using either response spectrum, time history, or another method.

Using the calculated dynamic response, an evaluation of the effects of the calculated stresses on mechanical strength, alignment (if critical to proper operation), electrical performance (microphonics, contact bounce, etc.) and noninterruption of function as related to the functional requirements of the equipment during an SSE should be performed. Maximum displacements should be computed and interference effects determined. The mathematical models used for analysis can be based on structural parameters which are calculated, or on parameters established by test, or by a combination of these.

The structural damping which should be used in analysis should have a reference basis. For example, it should be specified in the safety analysis report, or specification, or established by testing. If no damping value has been defined, one can be established by any means as long as the damping value used is justifiable.

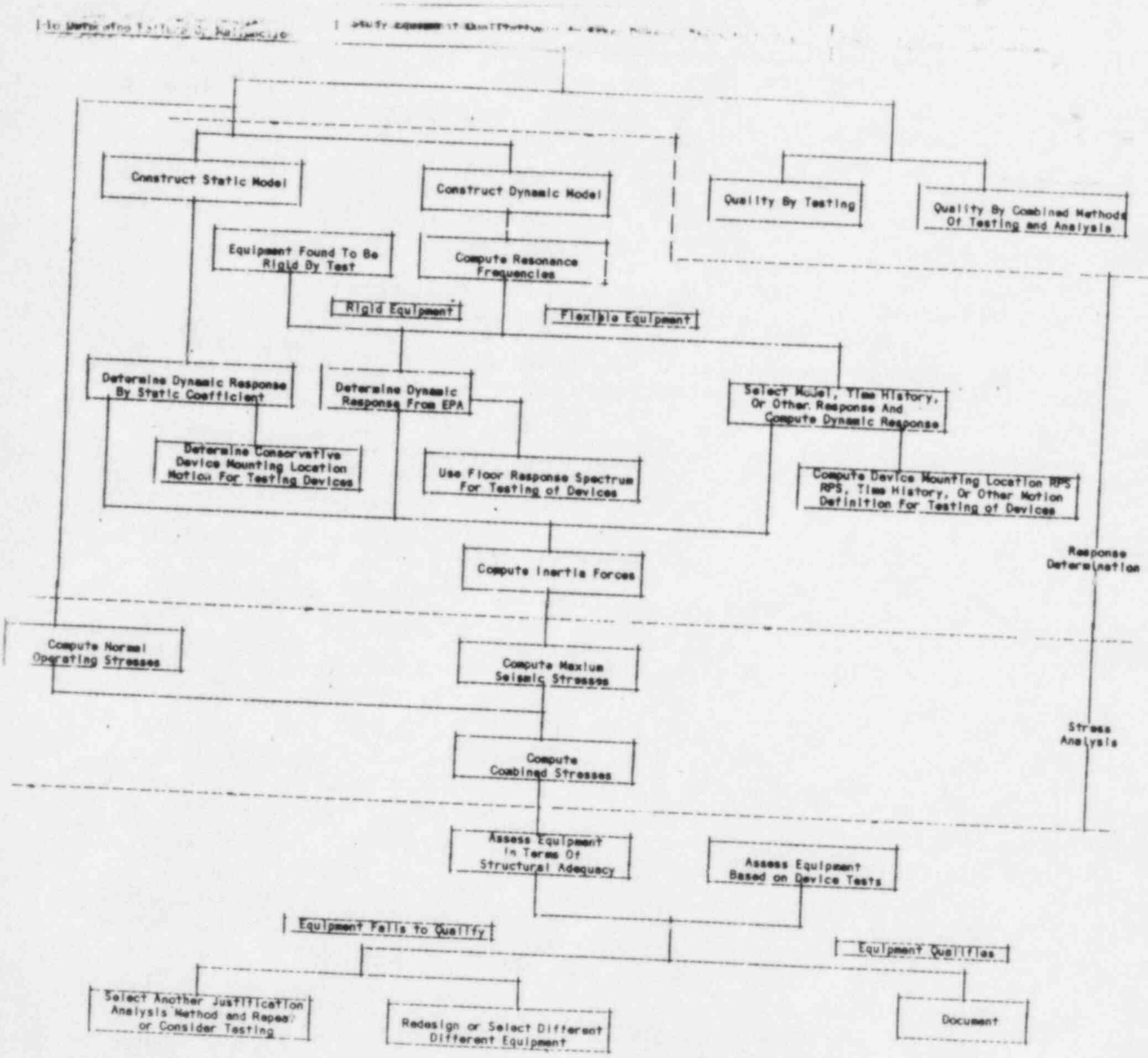


Figure IV-6-12
Seismic Analysis Flowchart

3.5.2.1.2 Damping

Damping is a generic name ascribed to the numerous complex energy dissipation mechanisms in a system. In practice, damping depends on many parameters, such as the structural system, mode of vibration, strain, normal force, velocity, materials, joint slippage, etc. In linear vibration theory, the simplifying assumption is made that damping is purely viscous, or dependent on the relative velocity of moving parts. Therefore, when a value of damping is associated with a practical system, it is usually assumed to be equivalent viscous or linear damping. This is a convenient simplifying way of relating realworld hardware behavior, which may be nonlinear to some degree, with theoretical concepts that normally utilize linear methods of analysis

For equipment composed of an assembly of components, there is usually no single value of damping. Damping is associated with every part of the equipment, ranging from bolted construction to uniform material. The value of damping may vary from place to place depending on the numerous other factors previously mentioned. Therefore, when a value of damping is ascribed to equipment, it is common to give a range of typical values. A useful practice is to associate a value of damping to each mode of vibration of the equipment which is in the frequency range of interest.

3.5.2.1.3 Measurement of Damping

Linear vibration theory indicates that there are numerous methods of measuring damping. Considerable care must be exercised in making the transition from an idealized model to a practical system. For example, it is rarely possible to locate any point in a system which corresponds accurately with

a lumped mass element of the model. Therefore, methods of calculating damping which require the motion of a particular mass point in the model to be measured and compared to the input excitation are not generally acceptable. The following methods for evaluating the damping coefficient are commonly used, but other justifiable methods may be used.

- Damping by Measuring the Decay Rate

  This method assumes that a pure mode of vibration can be excited in the equipment and that motion transducers are mounted at positions other than at a point of zero motion. The equivalent linear damping can be calculated by recording the decay rate of the particular mode of vibration.

- Damping by Measuring Resonance Peaks

  The equipment is excited in a single frequency test such as a slow sine sweep test. The response of any desired location in the equipment is measured and plotted as a function of frequency. The damping associated with that equipment location can be calculated by measurements on the width of the resonance peaks obtained for the different modes of vibration. This procedure is often referred to as the bandwidth method.

3 5.2.1.4 The Application of Damping in Testing

Ranges of damping are valuable data for the equipment designer and are frequently useful in the selection of detailed requirements during proof tests for seismic qualification. In any test, for a given peak amplitude of shake table input, the response of the actual equipment depends on two factors. The first is the dynamic amplification property of the input

wave shaped employed. This property is characterized by response spectra. The second factor combines the mass and stiffness distribution together with the actual damping inherent in the equipment. Since the first factor can be computed, the prediction of the response of the equipment depends entirely on the second factor. For proof tests and particularly single frequency tests, the required peak amplitude of the shake table input is calculated based on measured equipment damping.

3.5.2.1.5 Damping Values

Typical damping values for structures or components are shown in USNRC RG 1.61. The damping values are given as a function of the earthquake, OBE, and SSE. However, to allow for different effects produced by design response spectra for various sites (where the OBE ground response for one site may correspond to that of the SSE for another site), the damping values should also be related to some measure of stress level. This may be achieved by associating the OBE and SSE with stress limits as given by the ASME categories of "upset and faulted conditions", respectively. The OBE damping values would then be used with stress levels within the upset condition and the SSE damping values would be applicable for higher stresses. This would also correct the unrealistic condition of higher seismic response for the OBE than for the SSE which could result if the SSE damping values were used indiscriminately. The lower damping values should be used if the maximum combined stresses due to static, seismic, and other dynamic loading are significantly lower than the yield stress and one-half yield stress for the SSE and OBE, respectively. Damping values higher than those given in USNRC RG 1.61 may be used if justified by documented

test data. If equipment damping is not known, a value of 5 percent is recommended.

3.5.2.1.6 Dynamic Analysis

The equipment is generally modeled to best represent its mass distribution and stiffness characteristics. This model is used to determine if the equipment is rigid or flexible, and upon this determination depends the subsequent analytical steps.

For rigid equipment, the seismic forces on each component of the equipment are obtained by concentrating its mass at its center of gravity and multiplying the values of the mass and the appropriate maximum floor acceleration. The seismic stress shall be added to the equipment's operating stresses, and a determination made of the adequacy of the strength of the equipment.

If flexible, the model can be analyzed using the response spectrum model analysis technique or a time history analysis. Response spectrum analysis is a statistical method which assumes the statistical independence of each mode. Therefore, using this technique allows the response of interest, be it deflection, stress, or acceleration, to be determined by combining each modal response considering all significant modes by the square root of the sum of the square. The absolute sum of similar effects should be considered for closely spaced in-phase modes. Closely spaced modes are those with frequencies differing by 10 percent or less.

In the analysis, the effects of each of the two major horizontal directions and the vertical direction should be considered.

3.5.2.1.7 Static Coefficient Analysis

This is an alternate method of analysis which allows a simpler technique in return for added conservatism. No determination of natural frequencies is made by, rather, the response of the equipment is assumed to be the peak of the required response spectra at a conservative and justifiable value of damping. This response is then multiplied by a static coefficient of 1.5 which has been established from experience to take into account the effects of both multi-frequency excitation and multi-mode response. However, USNRC RG 1.100 Position C.1 requires justification by analysis for the 1.5 factor. A lower static coefficient may be used if it can be shown to yield conservative results. In a static coefficient analysis, the seismic forces on each component of the equipment are obtained by multiplying the values of the mass and the acceleration. The resulting force should be distributed over the component in a manner proportional to its mass distribution. The stress analysis may then be performed in a normal manner.

3.5.2.1.8 OBE and SSE Analysis

An analysis should be performed using one of the previously described methods but assuming a number of OBE events (the number chosen shall be justified for each site or 5 OBE's shall be used). Each OBE shall contain ten cycles of maximum stress as determined from the required acceleration spectrum at the mounting surface. The analysis should determine that the structural integrity of the equipment is maintained in combination with normal operating load during the OBE's.

The analysis must show that OBE events followed by an SSE will not result in failure of the equipment to perform

its Class 1E function. This may be particularly difficult to show for complex electrical equipment.

3.5.2.1.9 Documentation of Analysis

The demonstration of qualification shall be documented and shall include the requirements of the application of specifications of the equipment, the results of the qualification, and the justification that the methods used are capable of demonstrating that the equipment will not malfunction.

3.5.2.2 Testing

3.5.2.2.1 Introduction

Generally, if the seismic qualification is accomplished by a test approach, the seismic loads and operational loads would be applied simultaneously, and the equipment performance evaluated during and after the test. Figure IV-6-13 presents a seismic test approach.

For seismic loading, existing test systems can accommodate up to approximately 50,000 pounds in specimen weight sizes. All the waveforms called out in IEEE 344-1975 are commercially available.

Power is generally available at test facilities up to approximately 15,000 volts; 2,000 KVA is generally available for either driving electrical equipment or for providing electrical operational loads.

SEISMIC TEST APPROACH

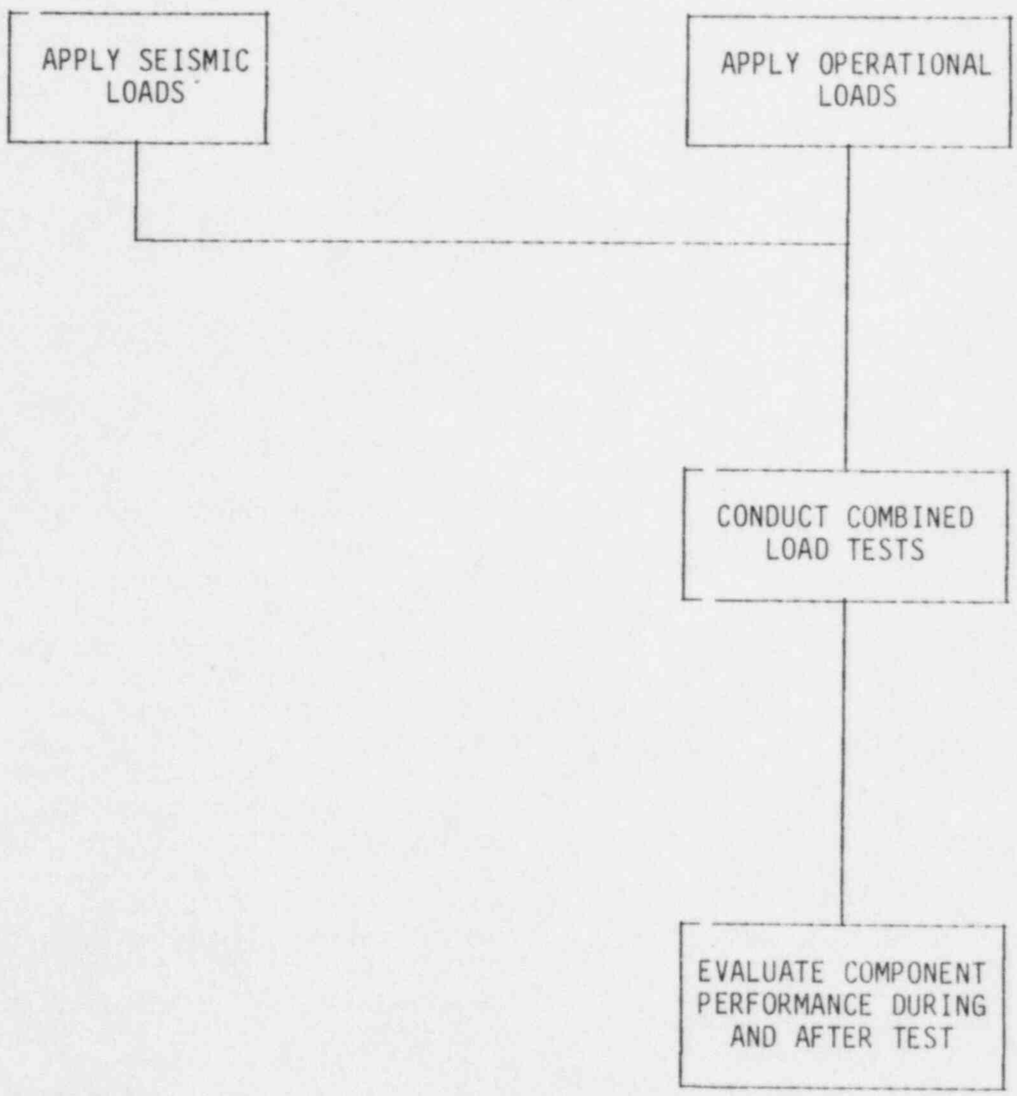


Figure IV-6-13

Table IV-6-1
Typical Test Facilities Capabilities

Seismic Loading Capability	
- Maximum Specimen Weight:	50,000 pounds
- Maximum Specimen Size:	11 ft x 17 ft x 40 ft high
- Biaxial, multi-frequency per IEEE 344-1975	
Operational Loading Capability	
- Power Available:	15,000 volts Three Phase 60 HZ 2000 KVA

Normally, the seismic load is applied using a shake table or some other loading device, such as a servo-controlled hydraulic actuator, in conjunction with either a proof test or fragility test approach. A proof test is simply a simulation of the specified seismic environment; the equipment either passes or fails, i.e., a go/no-go test. A fragility test is actually a series of tests in which the intensity of the seismic environment is increased by step functions until a level is reached at which there is a functional failure, or until a level clearly higher than any level that the equipment would have to be qualified to in the near future is reached. The fragility test approach is often used by manufacturers who have been unable to establish generic requirements. Future qualification would be accomplished by analytically comparing the fragility level to specified levels. The determination of the number of axes to test in and the waveform to be utilized is a function of the physical characteristics of

the equipment to be tested, and the size and shape of the required response spectrum. Figure IV-6-14 shows seismic loading techniques.

Seismic tests should be performed by subjecting equipment to vibratory motion which conservatively simulates that postulated at the equipment mounting during an SSE. The details of the test procedures given herein constitute the more common ones currently in use but do not preclude other justifiable methods. It is outside the scope of this document to give the theoretical bases for the test procedures.

SEISMIC LOADING TECHNIQUES

SEISMIC LOADING

PROOF TEST

AXES

SINGLE AXIS

MULTI-AXIS

FRAGILITY TEST

WAVEFORM

SINGLE FREQUENCY

MULTI-FREQUENCY

Figure IV-6-14

One practical problem that arises when attempting to establish the tests to be used to qualify equipment is the choice of the earthquake environment. As was discussed earlier, many factors must be considered. These involve the location of the equipment, the nature of the equipment, the nature of expected earthquakes, and other factors. A consideration not mentioned previously has to do with whether the

equipment is to be used on one application or many. In the first case, it is more likely that the seismic motion can be specified and the qualification test chosen to meet that specification, while in the latter case, the test should be designed to qualify the equipment for future, and therefore, undefined applications. This document will treat the two differently and will call the first proof testing (testing to a specific seismic motion) and the second fragility testing (testing to determine the equipment's capabilities).

Another factor to be considered is the multi-directional nature of the earthquake. Equipment should be tested to conservatively account for multi-directional effects of the earthquake.

Another practical problem arises in attempting to describe tests for devices (relays, motors, sensors, etc.), as well as for complex assemblies such as control panels. In the first case, it is reasonable to assume that the device can be subjected to seismic tests while simulating the operating condition and monitoring its performance during the test; however, in the case of complex equipment such as control panels, this is no longer true. Such panels usually contain many devices which are part of several systems extending over many other panels located in various parts of the power generating station. To test such panels while in an operating condition may be impractical and in such cases the following alternate approaches are recommended. The individual devices are tested separately while simulating an operating condition in order to establish the acceleration level for qualification. The panel, with the actual devices installed but inoperative, or with the devices' dynamic properties simulated, is vibration tested to determine if the

accelerations at the device locations are less than the levels at which the devices were qualified. A second approach is to apply the appropriate vibration input to the panel with the actual devices installed but inoperative or with the devices' dynamic properties simulated. The acceleration levels at the devices' locations are measured and are used as input accelerations to qualify the devices separately in an operating condition. The purpose of installing the non-operating devices is to assure that the panel has the dynamic characteristics it will have when in use. Where practical, control panels may be treated as devices and operationally tested.

Regardless of whether devices or assemblies are to be proof tested or fragility tested, there are certain common requirements.

3.5.2.2.2 Testing Methods

The choice of the type of motion to best simulate the postulated seismic environment is difficult, but the methods available fall into two categories, single frequency and multiple frequency. The choice of method will depend upon the nature of the equipment and the expected vibration environment. The various technical requirements appropriate to each test method may provide extra benefits for specific applications. However, such considerations should not preclude the legitimate use of any one of the methods. all of which can be justified as meeting some basic seismic criterion.

In general, the proof test seismic simulation wave forms should:

a. Produce a TRS which closely envelops the RRS or the applicable portions thereof using single or multiple frequency as required, to provide a conservative (but not

overly conservative) test table motion. See Figure IV-6-15.

b. Have a peak amplitude equal to or greater than the ZPA, except at low frequencies where the value of the RRS decreases below and stays the ZPA.

c Not include frequencies above the ZPA asymptote.

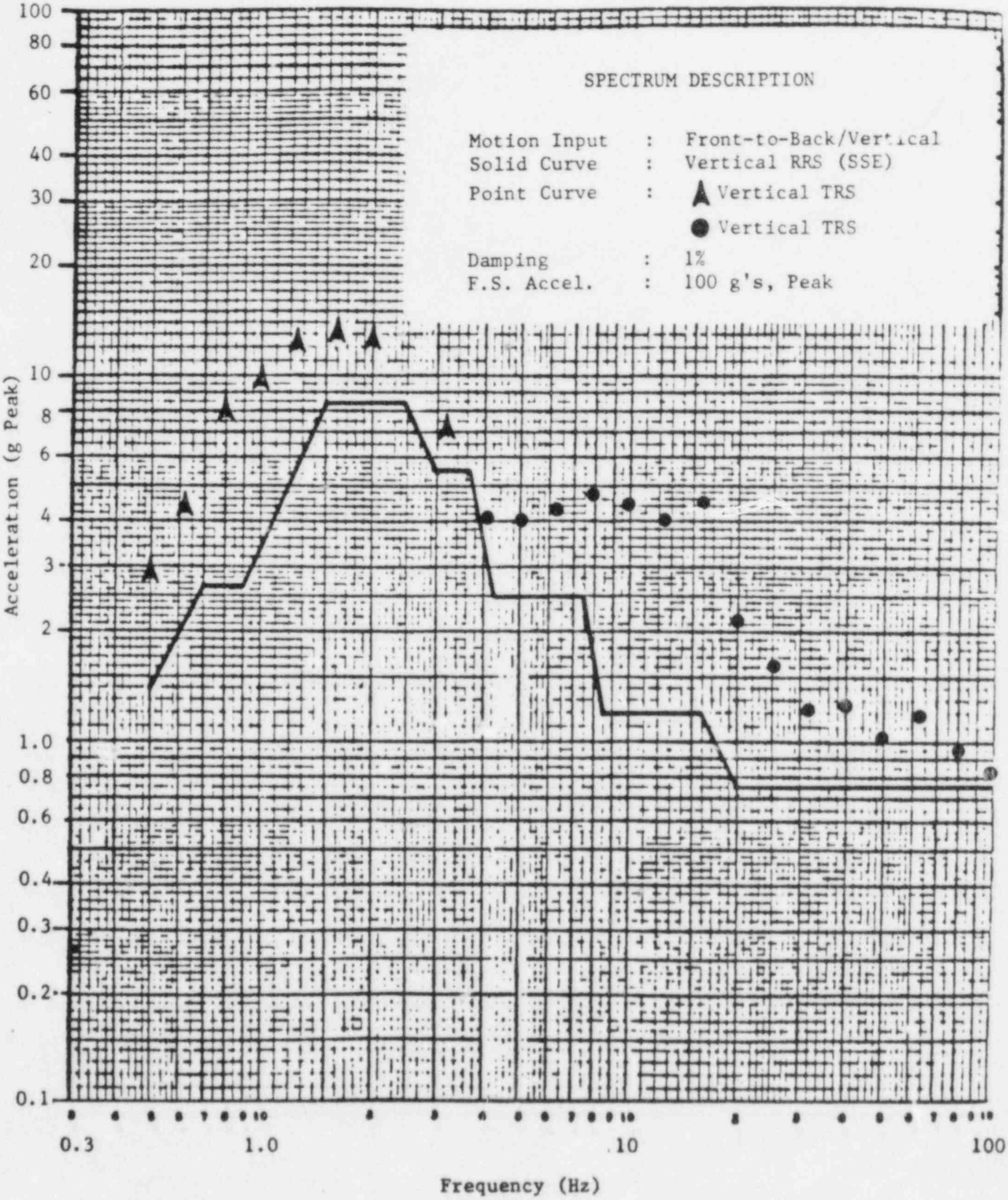
As a further complication, consideration must be given to the choice of single-axis or multiple-axis testing.

- Single-Frequency Tests

When the seismic ground motion has been filtered due to one predominant structural mode, the resulting floor motion may consist of one predominant frequency. In this case, a short duration steady-state vibration can be a conservative input excitation to the shake table. Further, single-frequency testing may be used to determine (or verify) the resonant modes and damping of equipment. If it can be shown that the equipment has no resonances, or only one resonance, or resonances are widely spaced and do not interact to reduce the fragility level, or if otherwise justified, single-frequency tests may be used to fully test the equipment.

The TRS from single-frequency testing results from each individual frequency and cannot be generated on the composite of several non-simultaneous single-frequency tests.

Figure IV-6-15
Comparison of Customer's Vertical RRS to the Vertical TRS



For any waveform employed, the shake-table motion should produce a TRS acceleration at the test frequency at least equal to that given by the RRS. The maximum acceleration of the shake-table motion should be at least equal to the zero period acceleration of the RRS.

For equipment with more than one predominant frequency, the shake-table motion should produce a TRS acceleration at the test frequencies of 1.5 times that given by the specified RRS or less if justified. This conservatively allows for a combined multimode response. The choice of the preceding factor depends on the shape of the RRS with the largest value (1.5) applicable to broadband RRS. As a consequence, the TRS need not envelop the RRS provided proper justification is given in accordance with USNRC RG 1.100 Position C.2. In addition, testing must be performed at all equipment resonances and at frequencies spaced no farther apart than 1/2 octave intervals, unless otherwise justified. Alternatively, when all the resonances of the equipment can be definitely established by an actual test, it will be sufficient for the single frequency TRS to envelop the RRS only at the equipment resonances with one single-frequency input.

- Continuous Sine Test

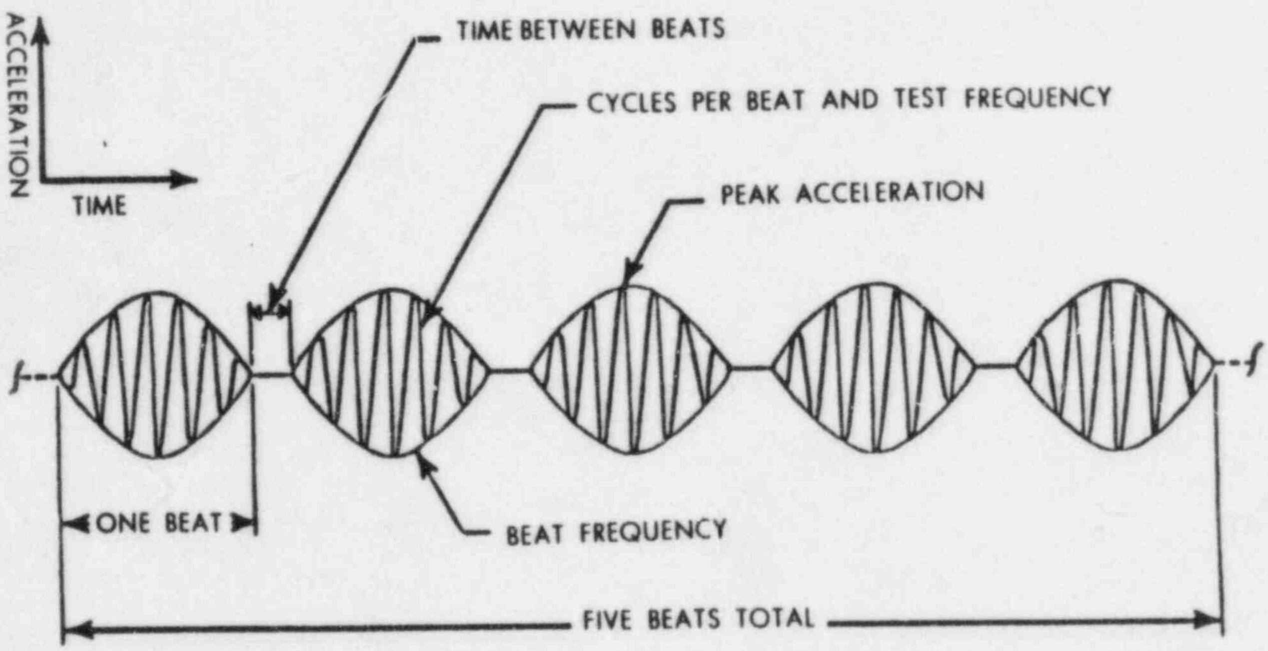
A test at any frequency should consist of the application of a continuous sinusoidal motion corresponding to the maximum acceleration at which the equipment is to be qualified and for an appropriate length of time.

The selection of the peak input acceleration and the length of time the test is to be run becomes a matter of the degree of conservatism desired. The peak input acceleration must be at least equal to the ZPA of the RRS (except at low frequencies where the RRS goes below and stays below the ZPA for which the value of the RRS must be met), unless the equipment does not meet the criteria in which case the input must be adjusted to produce a TRS which envelopes the RRS.

- Sine Beat Test

  A test at any frequency should consist of the application of sine beats of peak acceleration corresponding to that for which the equipment is to be qualified. The sine beats consist of a sinusoid at the frequency and amplitude of interest as shown in Figure IV-6-16. The peak amplitude of the beat should at least be equal to the ZPA of the RRS except at low frequencies where the RRS goes and stays below the ZPA for which the value of the RRS must be met.

Figure IV-6-16
Sine Beat Input for Testing



For a test at any frequency, a series of beats are used to represent low-cycle fatigue effects, with a sufficient pause between the beats such that there results no signficant superposition of equipment response motion. For a given peak amplitude of the beat, the degree of conservatism in the test will increase as the number of cycles per beat.

- Decaying Sine Test

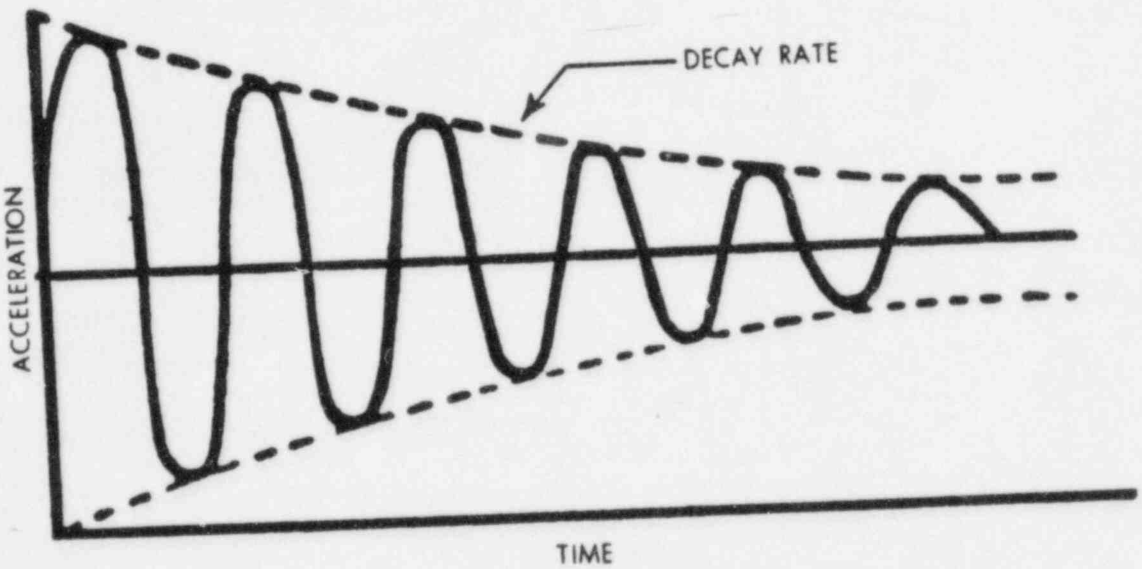
  A test at any frequency should consist of the application of decaying sinusoids of peak acceleration corresponding to that for which the equipment is to be qualified. The decaying sinusoids consist of a single frequency of exponentially decaying amplitude and a decay rate is chosen to most nearly produce a TRS from the shake-table motion which envelops the RRS. The peak amplitude of the sinusoid should at least be equal to the ZPA of the RRS except at low frequencies where the RRS goes and stays below the ZPA for which the value of the RRS must be met. For a test at any frequency, a series of decaying sinusoids are used to represent low-cycle fatigue effects, with a sufficient pause between the sinusoids such that there results no significant superposition of equipment response motion. For a given peak amplitude, the degree of conservatism will increase as the decay rate decreases. The frequencies of interest are the natural frequencies of the equipment being tested. See Figure IV-6-17.

- Sine Sweep Test

In this test, a sinusoidal input with continuously varying frequency is applied to the equipment. The frequency band should cover the range for which the equipment is to be qualified. This test closely approaches the conservatism of the continuous sine test in terms of producing maximum response. The percentage of steady state resonant response obtained depends on the sweep rate and the damping of the equipment. For sweep rates of 2 octaves per minute or less, and for typical equipment damping, this percentage exceeds 90 percent. Maximum response is obtained separately at every frequency in the test range. Consequently, this test produces the most thorough search for all natural frequencies and it is customarily used for this purpose as an exploratory test, with a low level of input such as 0.2g.

To qualify an equipment using the sine sweep test, the input amplitude must be at least equal to the ZPA of the RRS, except at low frequencies where the RRS goes and stays below the ZPA for which the value of the RRS must be met. The TRS may not be a composite of the entire frequency sweep. It must be the response spectrum centered around any instantaneous frequency. The TRS must envelop the RRS.

Figure IV-6-17
Decaying Sine



- Multiple-Frequency Tests

When the seismic ground motion has not been strongly filtered, the floor motion retains the broadband characteristics. In this case, multi-frequency testing is applicable for qualification. It is applicable as a general qualification method as long as the TRS envelops the RRS. Specific input excitation to the shake-table includes time history and random and complex wave shapes.

Multi-frequency testing provides a broadband test motion which is particularly apt for producing a simultaneous response from all modes of multi-degree of freedom systems. Multiple-frequency testing provides a closer simulation to a typical seismic ground motion without the requirements to introduce a higher degree of conservatism. Fragility data can thus be obtained by testing equipment under a realistic simulation of the environment.

The shake-table input excitation waveforms described in the following sections can be employed to test an RRS. The degree of conservatism varies from one method to the next. Other inputs which are not specifically referenced here can also be employed, provided they excite the equipment being tested.

For any waveform employed, the shake-table motion must be adjusted so that the TRS envelops the RRS over the frequency range for which the particular test is designed, and as a minimum, the shake-table acceleration must equal the ZPA of

the RRS. This comparison must be made using comparable values of damping. The adjustment of the table motion to produce enveloping should be made considering the following three factors:

1. The RRS may have motion amplification over a broad or narrow band of frequencies

2. The input excitation waveform may be one of several multiple-frequency types

3 The equipment being tested may have one of many possible dynamic characteristics

For assemblies or devices where the dynamic responses results from numerous interacting modes the shake-table input excitation must be adjusted such that the TRS envelops the RRS over a frequency range which includes all natural frequencies of the equipment up to 33 Hz. In all cases, the TRS must be derived using either justifiable analytical techniques or spectrum analysis equipment.

- Time History Test

A test may be performed by applying to the equipment a specified time history which has been synthesized to simulate the probable input to the equipment. It must be demonstrated that the actual test machine motion was equal to or greater than the required motion.

A time history record can be synthesized to match the RRS using simulation techniques or the required time history can be used. The duration of the input

excitation must be sufficient to simulate the effects of a seismic event.

- Random Motion Test (Response Spectrum)

  A test may be performed by applying to the equipment a random excitation, the amplitude of which is controlled in 1/3 octave, or narrower, frequency bandwidth filters with individual output gain controls. The excitation must be controlled to provide a TRS which meets or exceeds the RRS. The peak value of the input excitation shall equal or exceed the ZPA of the RRS.

  The duration of the random excitation should be a minimum of 15 seconds to allow a reasonable probability of occurrence of the expected excitation.

- Random Motion with Sine Beat Test

  To meet an RRS which includes a moderately high peak random excitation may require an unreasonably high peak value of the input. It is acceptable to adjust the random input to equal or exceed as much of the RRS as possible without using a peak input acceleration substantially greater than the ZPA. A sine beat or beats should be superimposed with random input motion to provide a composite excitation so that the TRS equals or exceeds the entire RRS over a frequency range which includes the natural frequencies of the equipment up to 33 Hz. (See Figure IV-6-18.) The optimum number of oscillations per beat may be determined from a plot

showing the ratio between the 1/2 and 5 percent spectrum damping values and the oscillation per beat as shown in Figure IV-6-19.

When more than one frequency of sine beats is required to meet the bandwidth of a spectrum, the beats should be initiated simultaneously. However, if the bandwidth of the peak value of the RRS has been widened to account for building analysis frequency uncertainty, the beats may be applied in sequence. The technique specified must be justified.

Figure IV-6-18
Random Spectrum with Superimposed Sine Beats

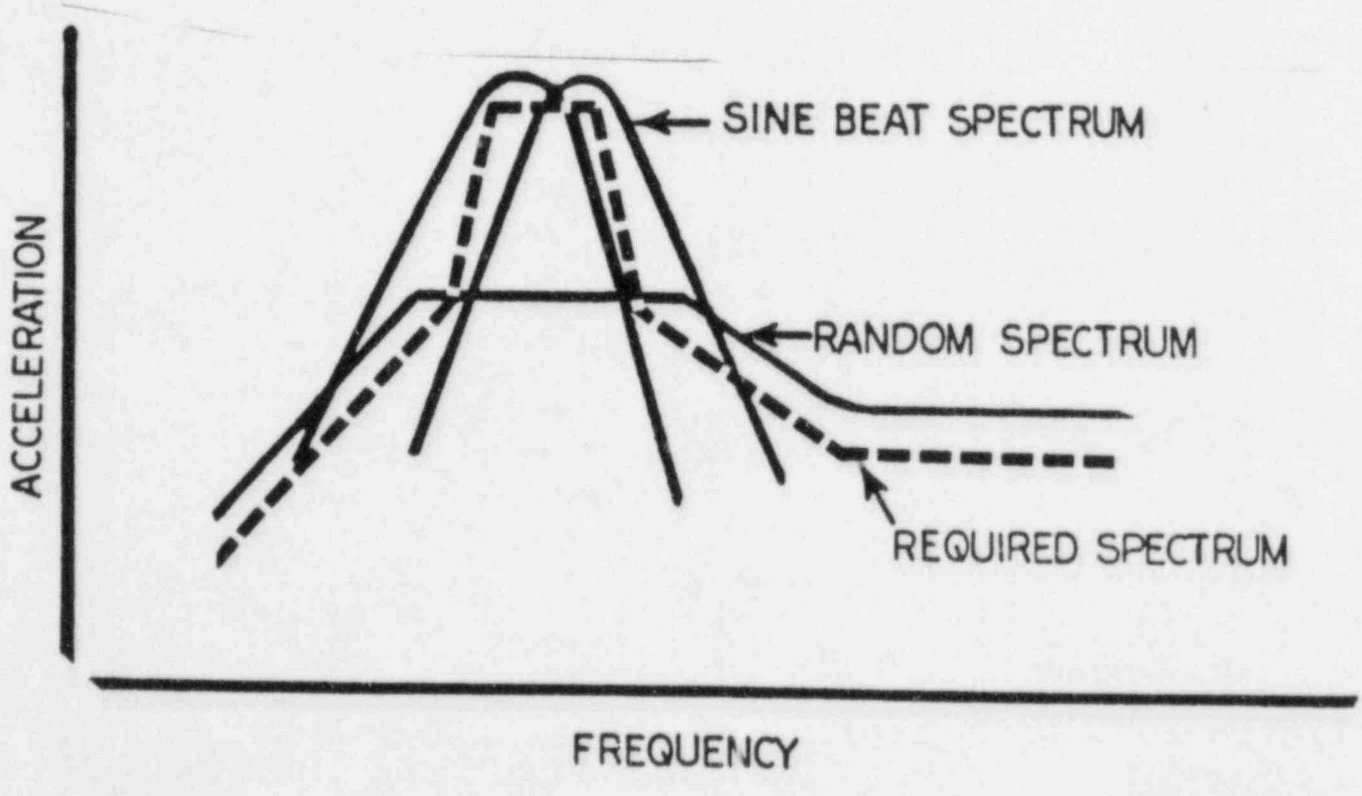
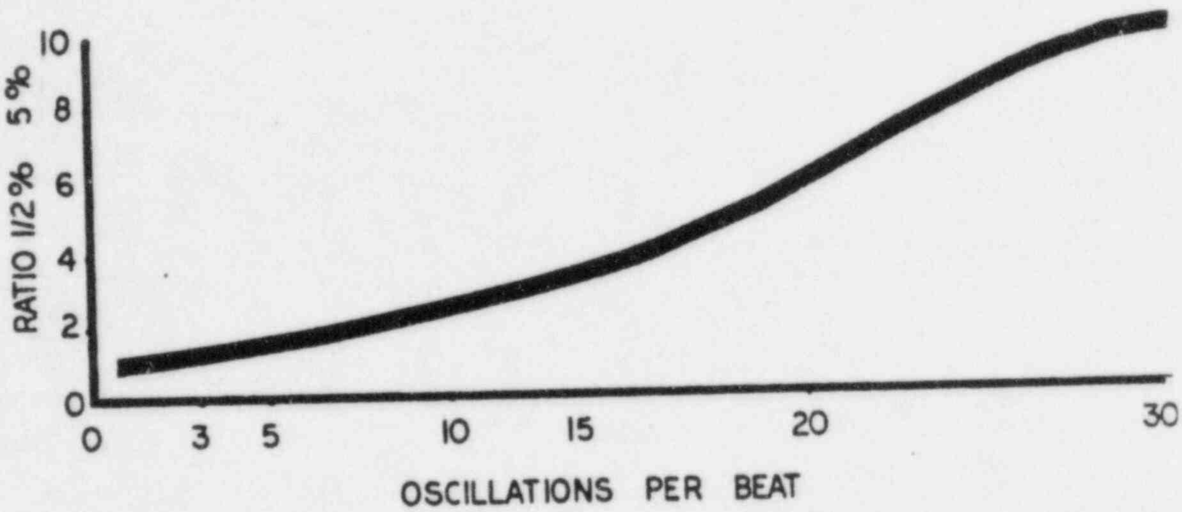


Figure IV-6-19
Spectrum Ratio Versus Oscillations Per Beat



- Complex Wave Test

In those cases where a random motion TRS cannot be adjusted to fit the RRS within a reasonable tolerance or without excessive conservatism, a complex wave test may be used. The test may be performed by subjecting the equipment to a motion which has been generated by summing a group of decaying sinusoids. The frequencies of the component signals should be spaced at 1/3 octave or narrower frequency intervals to cover the range required by the RRS. The decaying sinusoids should have individual decay rate controls covering the range of 1/2 to 10 percent. Each frequency must have individual amplitude and phase controls. All frequencies must be initiated simultaneously with the phase controls set to shape the peak amplitude of the composite waveform. It is desirable to vary the decay rate and the amplitude of each frequency to optimize the fit of the TRS to the RRS. The peak acceleration of the test table during composite waveform excitation must be greater than or equal to the ZPA of the RRS. The test should consist of several applications of the motion spaced apart in time such that no significant superposition of response motion occurs. The number of applications of the motion must be justified as being representative of the strong motion portion of the SSE; for example, the number of applications may be such that the total duration of the middle-frequency components equals the duration of the SSE.

- Other Tests

If there are vibration tests that conservatively simulate the expected seismic environment other than those described and that are equally justifiable, they may be used instead.

The following factors must be considered to justify the test method employed to qualify equipment:

1. Bandwidth of the RRS compared to that of the TRS and equipment characteristics and responses.

2. Duration of the test compared to the defined seismic event.

3 Peak acceleration of the test input and the magnification observed.

4. Natural modes and frequencies of vibration of the equipment.

5 Typical damping of the equipment.

6 Fragility levels.

7. Total number of cycles and the simulation of fatigue failure.

- Time Duration

Each site seismic evaluation should contain historical data from which the number of OBE's can be estimated for the service life of the facility (usually 40 years) if a number of OBE's less than 5 is to be justified.

This same number of tests, run as proof tests, shall be run at the appropriate g level. The OBE tests shall be followed by at least one SSE test. The duration of each test shall at least equal the strong motion portion of the original time history used to obtain the RRS for the SSE. Credit may be taken for any test preceding the SSE test, if shown to be greater than or equal in severity to the required OBE's fragility test durations, which by their nature, will normally far exceed the preceding duration requirement. However, each test duration simulating OBE's or SSE's shall at least equal the time duration strong motion portion of the SSE in order to properly account for vibration buildup.

When the strain range in structural elements is doubled, the fatigue life is reduced exponentially, such as by a factor of four. Therefore, a conservative seismic test for SSE's and OBE's requires that peak equipment response be reproduced reliably and for a sufficient number of cycles, and that the duration of each test should at least equal the strong motion portion of the SSE.

- Single and Multi-Axis Tests

  Seismic ground motion occurs simultaneously in all directions in a random fashion. The direction of test input motion should, therefore, be in all three principle axes simultaneously. At the present time, however, two-axis test facilities are limited and three-axis facilities are non-existent; therefore, several satisfactory

alternatives are allowed.

Single-axis tests are allowed if the tests are designed to conservatively reflect the seismic event at the equipment mounting locations or if the equipment being tested can be shown to respond independently in each of the three orthogonal axes or otherwise withstand the seismic event at its mounting location. This is the case if the coupling is zero or very low or if other justification can be provided. For example, if a device is normally mounted on a panel that amplifies motion in one direction, or if a device is restrained to motion in one direction, single-axis testing of the device may be adequate.

If the preceding considerations do not apply, multi-axis testing should be used. The minimum is biaxial testing with simultaneous inputs in a principle horizontal and the vertical axis. Independent random inputs are preferred and, if used, the test shall be performed in two steps with the equipment rotated 90° in the horizontal plane for the second step. If independent random inputs are not used (such as with single-frequency tests), four tests should be run. First, with the inputs in phase; second, with one input 180° out of phase; third, with the equipment rotated 90° horizontally and the inputs in phase; and finally, with the same equipment orientation as in the third step, but with one input 180° out of phase.

3.5.2.3 Combined Analysis and Testing

3.5.2.3.1 Introduction

Some types of Class 1E equipment cannot be practically qualified by analysis or testing alone. This may be because of the size of the equipment, its complexity, or the large number of similar configurations. This section specifically treats these types of equipment.

3.5.2.3.2 Low Impedance Excitation

Large equipment such as motors, generators, and prime movers, may be impractical to test due to limitations in vibration equipment loading capability. A useful method which combines testing and analysis utilizes the application of the excitation at points in the equipment with low impedance for response determination. This data may be used in subsequent analysis.

- Test Method

  With the equipment mounted to simulate the recommended service mounting (or on-site), a number of portable exciters are attached at points on the equipment which will best excite the various modes of vibration of the equipment. The data obtained from vibration sensors placed on the equipment can be used to analyze the equipment's seismic performance.

- Analysis

  The amplification of resonant motion can be used to determine the modal frequencies and damping which may be used in a dynamic analysis for the equipment. This method provides a greater degree of certainty in analysis, since the analytical model can be refined to reflect the measured natural frequencies.

- Qualification

  This method can adequately qualify Class 1E equipment in either of two

ways, namely, (1) the equipment can be excited to levels at least equal to the expected response from an SSE, using analysis to justify the excitation, or (2) the test data on modal frequencies can be used in a mathematical model to verify performance.

3.5.2.3.3 Extrapolation for Similar Equipment

As discussed earlier, the qualification of complex equipment by analysis is not recommended because of the great difficulty in developing an accurate model of the equipment and in obtaining numbers describing the physical parameters of the model. There are, however, many instances of equipment similar to a type which was qualified but with changes in size or in specific qualified devices in a fixed assembly or structure. In such cases, it is neither practical nor necessary to test every variation on the basic qualified version. Qualification by combined test and analysis applies in these situations. This is not intended to cover modifications following failure of a proof test.

- Test Method

  A full test program is conducted on a typical piece of equipment. Data on modal frequencies, damping, and responses throughout the equipment must be taken and recorded. Note that a time history test or a random test alone is not adequate for this purpose, since they do not reveal sufficient information about the equipment. A single frequency test must be used in addition to any multi-frequency test used.

- Analysis

  As mentioned earlier, if no resonant frequencies are found, the equipment may be analyzed as a rigid body.

Assurance should be obtained that changes from the originally tested equipment did not result in the formation of previously non-existent resonances. This can be done by simple testing or analysis.

If the equipment is not rigid, the effects of the changes shall be analyzed using previously mentioned techniques. For very complex equipment. this requires sufficient knowledge of the equipment to include the significant mass points to enable the responses at all points of interest to be calculated.

The test results, combined with the preceding analysis, allow the model of the similar equipment to be adjusted to produce a revised stiffness matrix and analysis for the modal frequencies of the similar equipment to be tuned. The result is a verified analytical model that can be used to qualify the similar equipment.

3.5.2.3.4 Extrapolition for Similar Seismic Conditions

Test results can be extrapolated for seismic conditions in excess of or different from those of recorded tests on a given piece of equipment if the test results are in sufficient detail to allow an adequate dynamic model of the equipment to be generated. The model should provide the analyst with the capability of predicting failure under the increased or different seismic excitation.

3.5.2.3.5 Shock Testing

Laboratory shock testing performed in conformance with various military standards (for example, MIL-S-901C, "Requirements for (Navy) Shock Tests, H.I. (High-Impact), Shipboard Machinery, Equipment and Systems") consists

of subjecting the component to high-impulse, shock-type loads (accelerations). Unless these accelerations are of sufficiently high magnitude (far higher than the earthquake levels) and of sufficient duration, shock testing without additional vibration testing is not considered adequate seismic simulation.

Since the primary objective of testing is to verify the seismic adequacy of components, the use of shock data can provide only an approximation of the adequacy of the equipment tested. This is the result of the difficulty encountered in matching the frequency content and duration of the simulated shock with that of the shock experienced in seismic service.

3.5.2.3.6 Other Test Analyses

In addition to the above, analysis may be used for the following:

- To explain unexpected behavior during a test.
- To obtain a better understanding of the dynamic behavior of the equipment so that the proper test can be defined.
- To obtain a measure of expected response before a test.

3.5.2.4 Documentation

3.5.2.4.1 General

The inspector should verify that the documentation for each equipment type demonstrates that the equipment meets its performance requirements when subjected to the seismic accelerations for which it is to be qualified.

3.5.2.4.2 Specifications

The following are to give direction

in the inspection of specification information required for either analysis or test of the specified equipment:

- RRS: The RRS for the surface on which equipment will be mounted should contain the data for the principal horizontal axis and the vertical axis as a minimum. The RRS should include critical damping values at which the RRS was made and should indicate artificially broadened areas. Equipment location relative to the location for the RRS should be included as should any unusual mounting plans.

- Floor Motion: If an RRS is not furnished, either surface (floor or structure) motion maximum accelerations at all significant frequencies or a time history shall be provided.

- Functional Requirements: A listing or description of the functional requirements and malfunction criteria shall be provided.

- Operation Settings: Typical operational settings (or ranges) for adjustable devices shall be provided.

- Class 1E: Identification of Class 1E devices and circuitry, and the functions of related Class 1E equipment, shall be provided.

- Duration: The earthquake's strong motion time duration shall be specified.

- OBE: The required number of OBE tests shall be specified.

- Loading: Loading requirements to be applied during the test, if any, shall be specified.

- Acceptance: Special acceptance criteria, if any, shall be specified.

- Test Versus Analysis: Special requirements for tests or analysis on specific equipment shall be provided.

- Power Spectral Density Function: Vibratory motion in terms of power spectral density as a function of frequency shall be specified.

- Margin: All margins included in this document shall be identified (See IEEE 323-1974).

3.5.2.4.3 Analytical Data

If proof of performance is obtained by analytical means, it should be presented in a step-by-step form which is readily auditable by the inspector and should include a listing of the potential failure modes considered in the analysis.

3.5.2.4.4 Test Data

If proof of performance is obtained by testing, the test data should be inspected to verify that it contains the following information:

- Equipment identification

- Equipment specification

- Test facility

  a. Location

  b Test equipment and calibration

- Test method and procedure

- Test data (including proof of performance)

- Results and conclusions (particularly natural frequencies and maximum accelerations such as a comparison of TRS and RRS. NOTE: Fragility data may be proprietary.

- Approved signature and date

3.5.2.4.5 Extrapolation Data

If proof of performance is obtained by extrapolation from similar equipment, the inspector should verify that the data contains:

- A description of pieces of equipment

- Test data on the original equipment

- A detailed description of the differences between the two pieces of equipment

- Justification that the differences do not degrade the seismicity below acceptable limits (may require some additional analyses or testing) including any additional supporting data.

3.5.2.5 Exceptions

When a single-frequency test is performed, the use of a factor of 1.5 and the concept that the test response spectra (TRS) need not envelop the required response spectra (RRS) as described in Section 6.6.2.1 of IEEE 344-1975, are not acceptable unless justified, per RG 1.100.

"Sine Sweep Test" input with the TRS enveloping the RRS according to the criteria described in Section 6.6.2 and 6.6.2.1 of IEEE 344-1975 is not acceptable unless justified per RG. 1.100.

Analysis without testing is acceptable only if the equipment functional operability is assured by its structural integrity alone. The procedures described in Section 5.2 through 5.4 of IEEE 344-1975 are utilized, except the use of a static coefficient of 1.5 for the equipment static analysis is not considered acceptable unless fully justified by the equipment supplier. In addition, component fatigue shall be checked to account for five OBE's and one SSE when the analysis is made. (See RG 1.100 in Appendix VII for additional details.)

Finally, it is important for the inspector to remember that functional requirements may allow degraded operation, failure in a certain direction, momentary contact bounce, indication device failure, etc., provided the equipment can still perform its safety function.

3.5.2.6 Installation Considerations for Safety-Related Instrumentation and Control Hardware

Seismic Category 1 instrumentation and control equipment is generally designed, tested, and fabricated to perform as follows during and subsequent to SSE:

- Exhibit no undue deflection which would prevent any component from performing normal uninterrupted operation. For example, if a breaker is in the test or withdrawn position, no deflection shall occur which may result in flashover from the live part to breaker.

- Have no components dislocated, which would prevent uninterrupted normal operation (i.e., such as, but not limited to, fuse thrown out of fuse holder, bolt used to mount control transformer sheared, etc.).

- Maintain all components in the same operating position during and after the disturbance as they were prior to it.

- Permit continued operation of all components during and after the disturbance, including interruption of rated short circuit currents (i.e., if the breaker is closed or tripped during the disturbance, it will react accordingly). It also shall permit undiminished operation after the disturbance (i.e., an open breaker shall be closeable and vice versa).

- All devices which are operated by electromechanical forces against mechanical restraining forces (spring tension) shall be tested with minimum restraining force and the test shall verify requirements contained in foregoing paragraphs.

Verify that complete documentation has been provided in accordance with the standards, codes, and regulatory authorities, covering test data,

analysis, and/or a combination of both. This documentation will demonstrate the adequacy of the equipment and component parts to withstand the specified seismic forces.

Floor response spectra curves representing earthquake input functions will be part of this data. These spectra curves are developed at the equipment base elevations and represents the variation of acceleration in multiples of "g" with the natural period of vibration of the supporting structure.

3.6 Seismic Monitoring Instrumentation

3.6.1 Introduction

When an earthquake occurs in the vicinity of the nuclear power plant, its effects may not be immediately detected. The seismic instrumentation system is used for measuring the acceleration magnitudes and frequencies generated by the earthquake on the plant structures. The measured values are compared by the plant operator with the calculated values used in the design criteria, in order to determine how severe the plant structures, equipment, and piping were affected by the earthquake, and to decide whether or not the plant can continue to operate safely. The system is designed to satisfy the requirements of the NRC Regulatory Guide 1.12 Rev. 1 of April 1, 1974.

No automatic initiation signal to shut the plant down is included in this system. Annunciation is provided in the control room for peak accelerations reaching OBE (Operating Basis Earthquake) and/or SSE (Safe Shutdown Earthquake) values. The operator then takes the appropriate action.

The primary functions of the seismic instrumentation system are to determine:

a. Whether or not the design response spectra have been exceeded.

b. Whether or not the calculated vibratory responses used in the design of the representative Category 1 structures and equipment have been exceeded at instrumented locations.

c. The degree of applicability of the mathematical models used in the seismic analysis of the building and equipment.

During an earthquake, the instrumentation shall be capable of measuring the motions at the ground input, the containment structure and equipment, and at other Category 1 structures, equipment, and piping.

When an earthquake occurs, the system must provide a permanent record of the resultant motions at the Category 1 structures and equipment. The data recorded shall be a three-axis time-history so that the plant operator may subsequently visually obtain a plot of actual accelerations and frequency of the response spectrum.

The locations for the installation of the instruments must be selected:

a. to aid the design verifications during the time when actual earthquake recorded data will be analyzed, and

b. to measure responses with high amplification rather than locations of slight amplification. The number of instruments to be installed and their location requirements are given in ANSI N18.5-1974.

3.6.2 Seismic Instrumentation Hardware

3.6.2.1 Time History Accelerograph

An instrument capable of measuring and permanently recording the absolute acceleration versus time. The components of the time-history accelerograph (acceleration sensor, recorder trigger) may be assembled in a self-contained unit or may be separately located.

3.6.2.2 Acceleration Sensor

An instrument capable of sensing absolute acceleration and transmitting the data to the recorder.

The acceleration sensors normally have a sensitivity range of 100:1 zero to peak (such as 0.01g to 1.0g), frequency range flat from 0.1 Hz to 30 Hz, proportional velocity damping adjusted between 55 percent and 70 percent of critical, no spurious reasonance within the frequency of interest and cross axis sensitivity of .03 g/g. (See Figure IV-6-20 and IV-6-21.)

3.6.2.3 Centralized Recorder Units

The multi-channel centralized recorder is

designed to provide a time-history record of the acceleration of each remotely located sensor. It is activated by a triaxial seismic trigger typically located on the containment foundation. The schematic diagram of a recording section of a typical centralized recorder unit is shown on Figure IV-6-22.

3.6.2.4 Response Spectrum Recorder

The response spectrum recorder is a passive device with 16 single degree-of-freedom oscillators tuned to specific frequencies in the band of interest to structural engineers. A diamond tipped stylus is attached to the moving end of each sensor to inscribe a permanent record of its deflection on one of sixteen record plates. Regulatory Guide 1.12 requires that the records on the containment foundation provide signals for control room notification. For this application, the RSR is made active with the addition of contacts set at the OBE displacement for each frequency. The contact switch output is carried by cable to the central location and connected to a response spectrum annunciator. If the OBE limit is exceeded at one or more frequencies, a red light is illuminated for those frequencies. Thus, the operator has notification when the OBE design response spectra have been exceeded. The peak accelerations are determined from the scribed record plates by the use of a microscope, scale, and sensitivity data.

RSR units are mounted directly on the foundation or floor. One vertical and two horizontal units are required. The horizontal units are mounted at 90° to one another to obtain the North/South and East/West recordings. The response spectrum annunciator panel is rack mounted (an optional desk-top mount is available) in a central location, typically in the cabinet rack. The power is supplied from 115VAC directly or from a rechargeable battery.

The response spectrum recorders should have the following specifications:

a. Dynamic Range-50:1 zero to peak (such as 0.02g to 1.0g).

b. Frequency Range-Minimum coverage from 1 Hz to 30.0 Hz.

c. Damping-not less than nominal 2% nor more than nominal 5% of critical damping, controlled to +0.15 of nominal. The actual amount of damping is to be consistent with the OBE based design damping for the supported structure or equipment.

3.6.2.5 Seismic Trigger

A device having the function of starting the time-history accelerograph.

A seismic trigger is typically installed in the free field or on the basement slab, and it causes the recording system to activate at a preset acceleration threshold value. A value of about 0.01g is recommended because it is:

- Exceeded during the first part of the p-wave arrival.

- Lower than accelerations of structural importance.

- Higher than most local noise, such as that caused by vehicles, people, operating equipment, etc.

The early arriving p-waves from an earthquake are largest in the vertical direction. This is because the energy path from the hypo-center to the site is faster along the deeper, high velocity layers and up to the surface than horizontally along the lower velocity surface layers (Figure IV-6-23). Thus, a vertically sensitive starter response is filtered with non-earthquake sources (Figure IV-6-24). Nature has provided a convenient separation for us - earthquake frequencies predominate in the 1 to 10 Hz band while local, cultural, and construction based energy predominates at higher frequencies.

Shear wave energy from distant earthquakes may be stronger in the horizontal than the vertical direction; therefore, typical starter units include both vertically and horizontally sensitive triggers. Separate horizontal and vertical devices also add to system reliability. The horizontal trigger can be a pendulum, or biaxial acceleration triggers can be used.

Figure IV-6-20
Accelerometer Response Curve

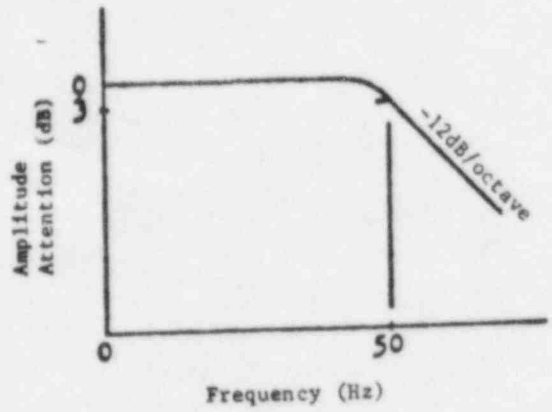


Figure IV-6-21
Centering of the Mass for a
Horizontal Sensor

Centering of the Mass For A
Horizontal Sensor

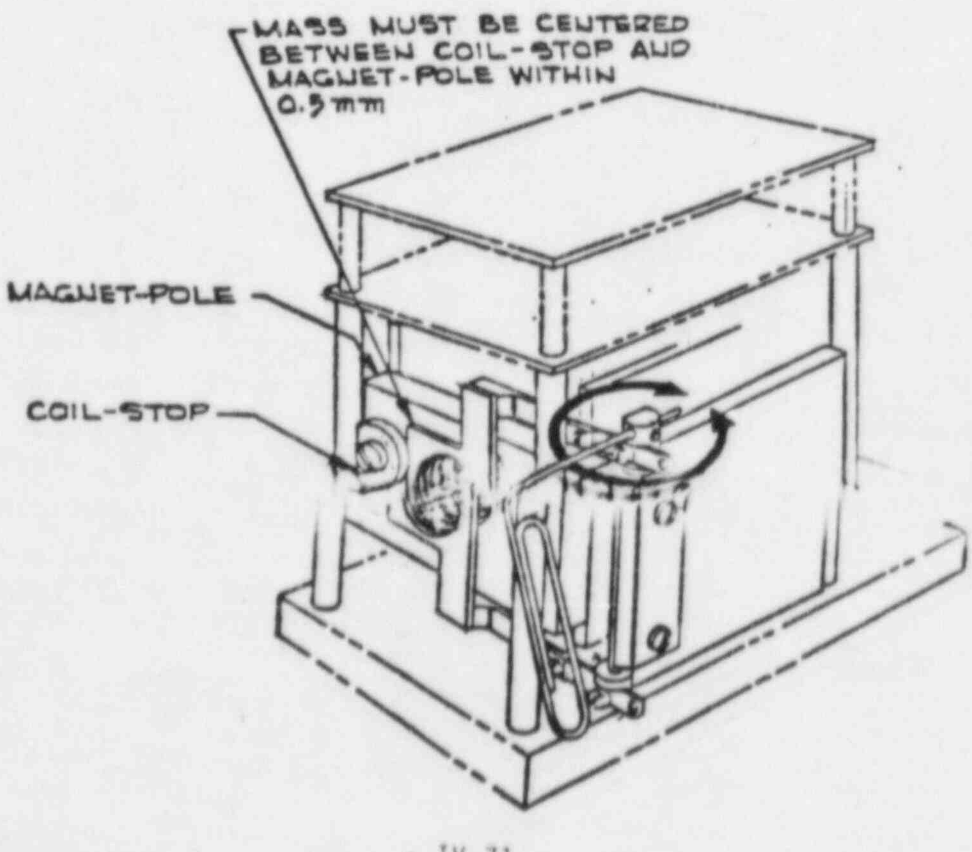


Figure IV-6-22
Recording Section
(Courtesy of Kinemetrics, Inc.)

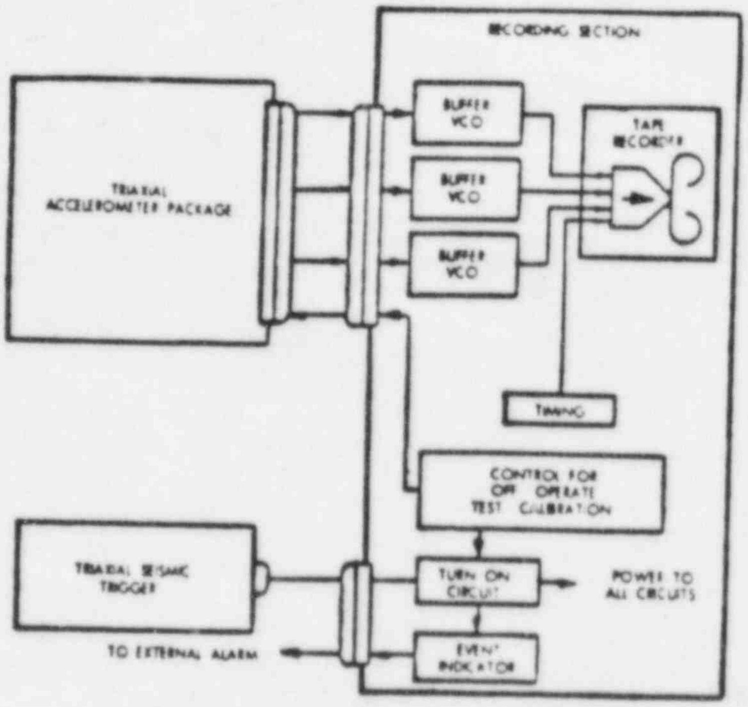


Figure IV-6-23
Possible Transmission Paths for
Seismic Waves of Early Ground Motion

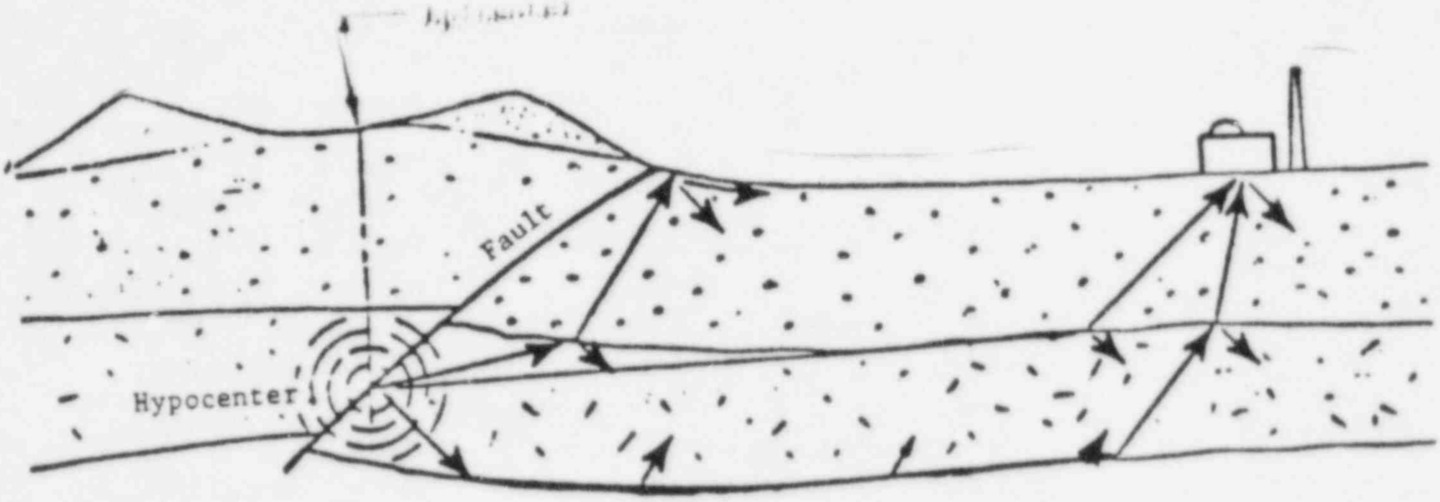
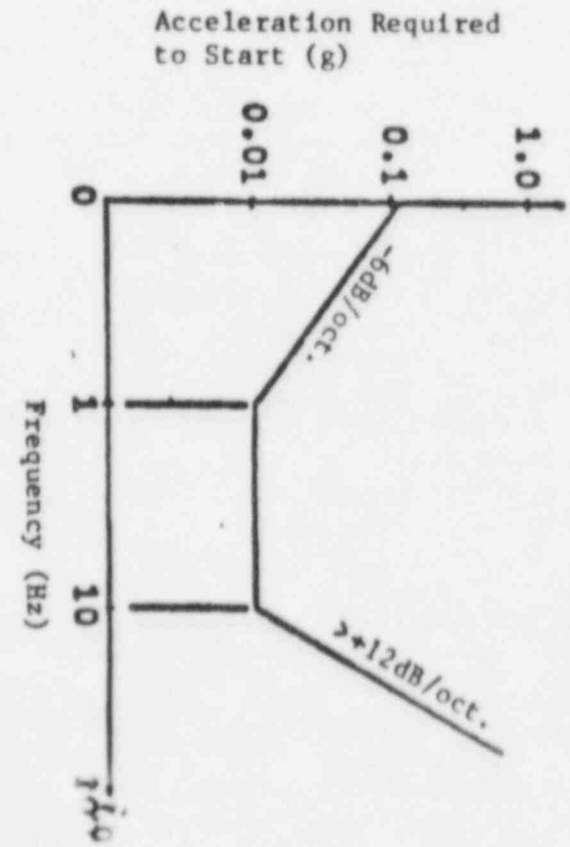


Figure IV-6-24
Acceleration Starter Response Curve



3.6.2.6 Seismic Switch

A seismic switch is a triaxial trigger unit which provides a relay closure whenever OBE zero period acceleration on at least one of the axes has been exceeded. The zero period acceleration is the maximum ground acceleration specified for a given site. At zero period, the design response spectra acceleration is identical for all damping values and is equal to the maximum ground acceleration. Regulatory Guide 1.60 defines these parameters. Usually the vertical set point is 2/3 of the horizontal valve.

To be acceptable, the seismic switch located at the foundation of the containment should be connected to event indicators that are located in the control room, so that a signal is given when earthquake is exceeded. Also, both audio and visual signals should be provided to the control room operators in the event of an earthquake.

In addition, the triaxial time-history accelerograph is located in the containment foundation or in the free field. This signal should be available to the control room operator. The response spectrum recorder in the reactor containment foundation or in the free field is also connected to the control room to indicate if the design response spectra values for discrete frequencies are exceeded during an earthquake.

3.6.2.7 Peak Recording Accelerograph

The PRA is an unpowered triaxial sensor/recorder device suitable for installation on slabs, floors and brackets. With the properly designed insulation, it can be used on high temperature equipment. The record is retrieved from the remote location and when developed shows the maximum positive and negative acceleration values along each, orthogonal axis. No frequency data is recorded.

3.6.2.8 Seismic Instruments Installation Considerations

In many of the nuclear power plants, calculated floor response spectra are used to design Seismic Category 1 systems and components supported on these levels. It is, therefore, important to install triaxial response-spectrum recorders at

the selected support (floor) locations to determine if the calculated floor response spectra have been exceeded. This information will be needed to determine (1) the conservatism in the modeling and design assumptions made for the structure and design input motion to the support systems and components and (2) the advisability of continuing the operation of the plant without a safety analysis following an earthquake.

The magnitude of the response of the systems and components supported on the containment structure is required in order to verify if the actual response of these parts has exceeded the design basis. This can be monitored by installing triaxial peak accelerographs over selected locations on these parts. In addition, peak response of these parts will be necessary to determine the conservatism in the modeling and design assumptions made for these systems and components. (See Figure IV-6-25.)

3.6.2.9 Starter, Sensor, and Switch Units

The remote sensor and starter units should be bolted to a concrete slab or another part of the structure with at least 20 ft. - lb. torque. A single anchor is sufficient and permits easy orientation of the unit during installation. The units should be placed directly on concrete slabs, as this provides the most rigid anchor. A typical shelf bracket is shown in Figure IV-6-26, however, shelf brackets should be avoided if at all possible. A temperature insulating mount should be used when the sensor units are to be mounted on pipes, tanks or systems which have a surface temperature of over 160°F. (See Figure IV-6-27.) For pipes or tanks where flanges are not available and where welding is not desirable, a saddle mount can be used. (See Figure IV-6-28.)

The selection of the free field sensor unit location offers a number of possibilities. If a surface location is desired, a small concrete slab, approximately three feet square, can be poured. The objective is to produce a foundation which behaves the same as the soil or rock. To avoid the construction of a separate foundation, the free field unit can be mounted on the slab of a reasonably small single-story structure, such as

a guard house, storage shed, or water treatment building. This slab, if not larger than about 15 or 20 feet in plan, will behave sufficiently similar to the smaller slab. A downhole version of the remote sensor can be used if it is desired to measure the free field response at rock below soil overburden. In this case, the unit is anchored in its hole with grout, sand, or a hole-lock device. The free field unit should be located at least one structure-plan dimension away from the nearest significant structure to avoid recording the feedback of structural response into the ground. For example, this unit should be placed at least 200 feet away from a 200-foot wide reactor building. At the base of a stack, an area with radius equal to about one-half the stack height should also be avoided. A protective shelter is advisable if the unit is not inside a building. If the temperature will be below 0°F, a heater should be provided. Interconnecting cable from the free field unit should be run in waterproof conduit.

Common practice is for the starter unit and one of the sensor units to be located side-by-side on the containment foundation. In this way, the two cables can be run through the same conduit and cable trays to the central recording location. Period maintenance is simplified as well with a single location for the two instruments. Each sensor unit should be aligned with the principal axis of the structure to facilitate subsequent use of the recorded data with the mathematical mode. The axis of the cable and connector parallels the longitudinal accelerometer axis. The interconnecting cable may be laid in instrumentation or thermocouple trays, but not in high voltage trays. If it is necessary to carry the cable through a penetration of the containment, the shields should be carried through individually. Junction boxes near the remote units are permissible if they are of high quality.

3.6.2.10 Summary of Seismic Instrumentation Considerations

Specific installation requirements for seismic instrumentation vary with equipment type and manufacturer. However, some general guidelines are applicable for most installations. Inspection should include, but not necessarily be limited to the following areas:

Figure IV-6-25

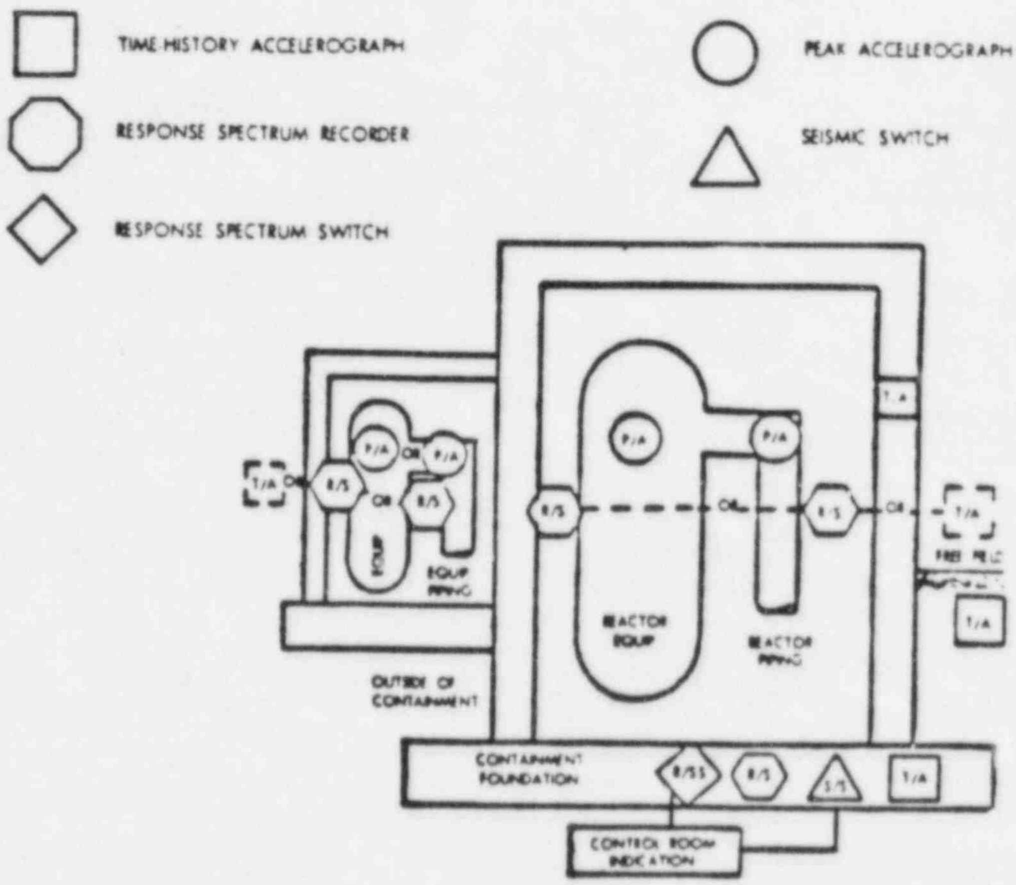


Figure IV-6-26
Typical Shelf Bracket

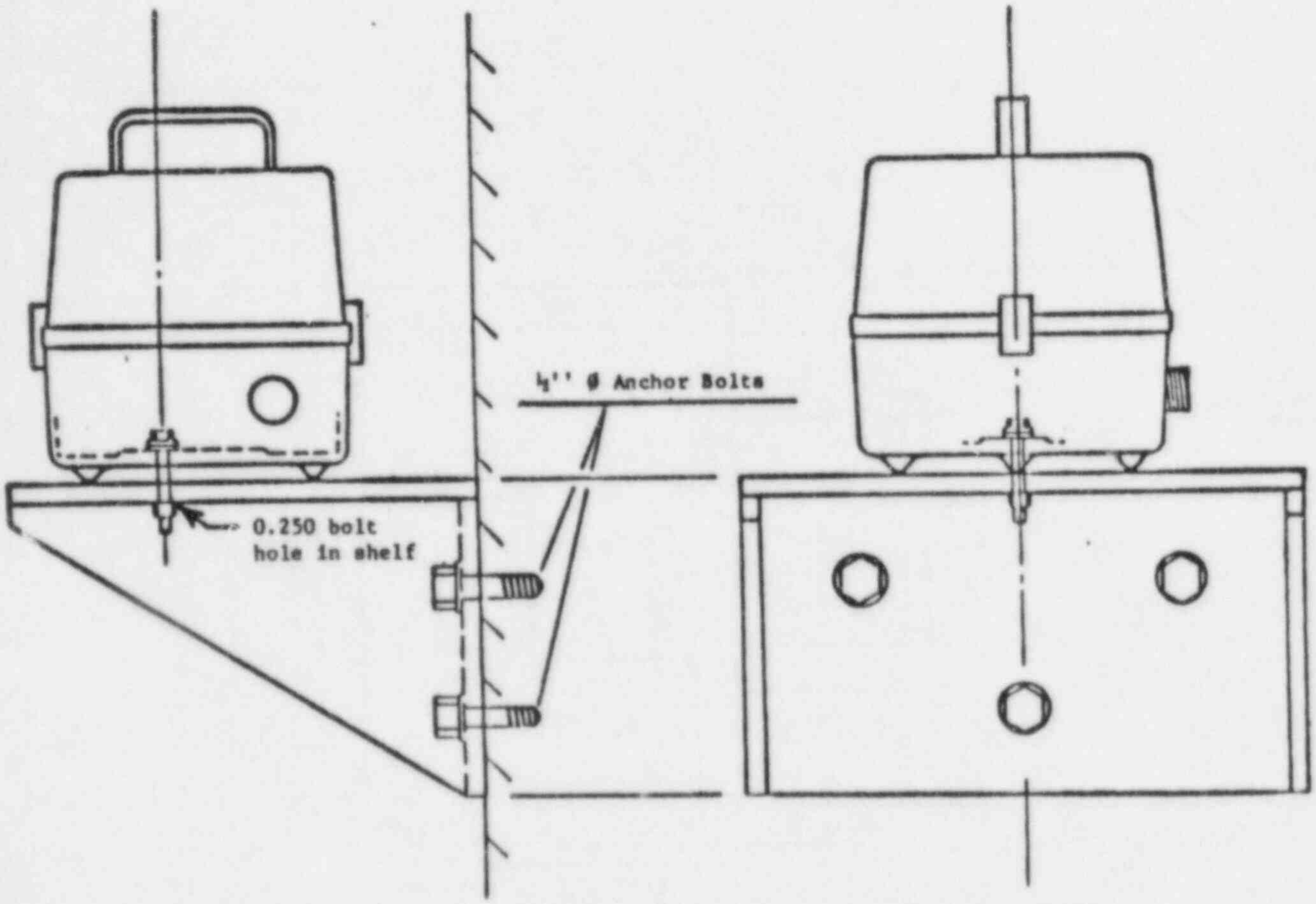
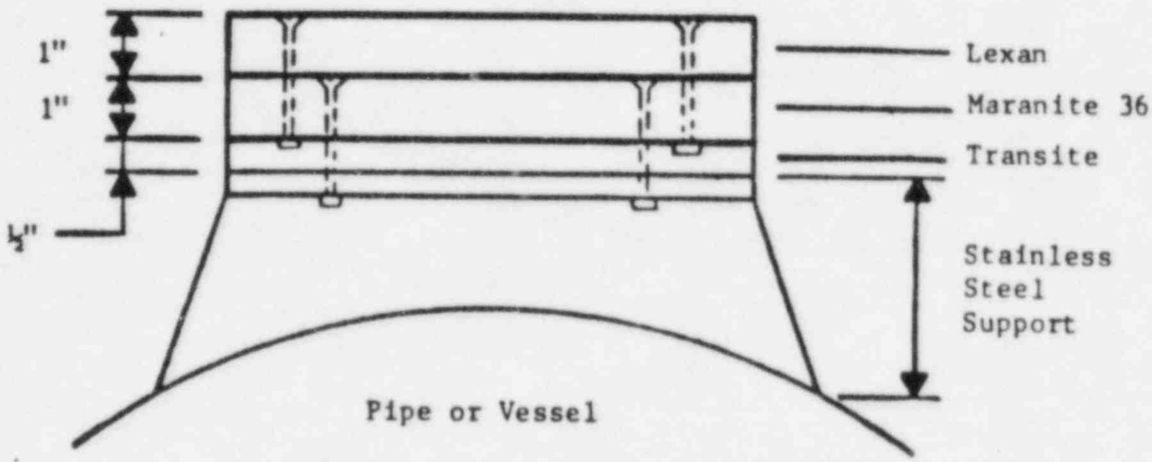


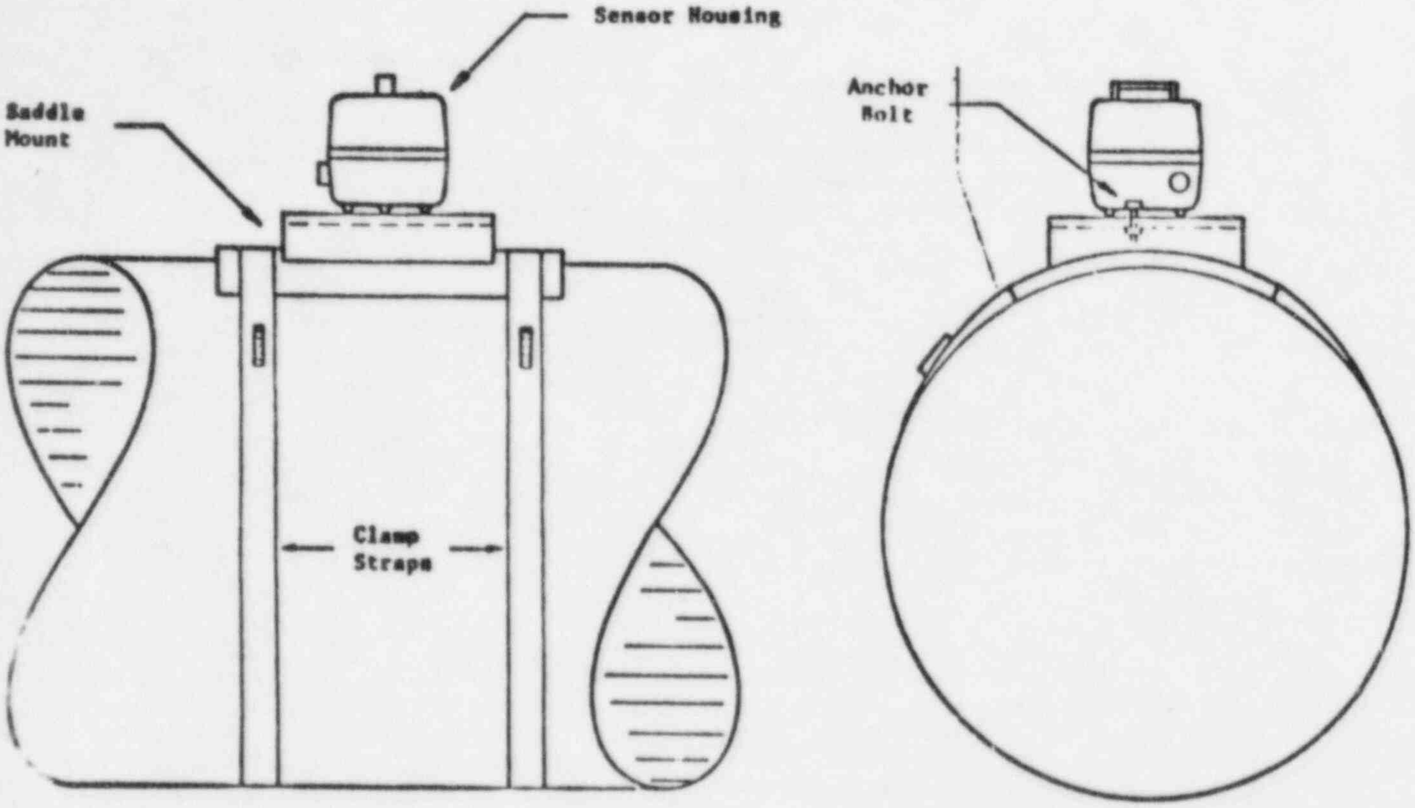
Figure IV-6-27
High Temperature Support for Accelerometer Unit



Notes

1. Mount designed to insulate from 600°F
2. Connections made with ¼" x 28 stainless steel flat head machine screws

Figure IV-6-28
Saddle Mount for Sensor Unit
(Courtesy of Kinemetrics, Inc.)



- adherence to ANSI N18.5-1974 instrument location standards

- structural integrity of sensor mounting

- adherence to sensor environmental constraints

- sensor directional alignment and whether or not the sensor is level

- installation and test documentation

4.0 SUMMARY

The fundamental considerations behind the theory of earthquake activity, criteria applicable to seismic Category 1 instrumentation and control equipment and seismic monitoring instrumentation have been discussed in this unit. It is expected that NRC inspectors have acquired sufficient knowledge to review the documentation pertinent to the service qualifications and testing of the safety-related instrumentation and control components.

INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

UNIT IV ATTACHMENT 1

Pre-Study Text

EXAMPLE OF REQUIREMENTS FOR SEISMIC CATEGORY 1 INSTRUMENT TUBING SUPPORTS UTILIZED BY ONE ARCHITECT ENGINEER (SAMPLE REQUIREMENTS ONLY)

INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

UNIT IV ATTACHMENT 1

Pre-Study Text

EXAMPLE OF REQUIREMENTS FOR SEISMIC
CATEGORY 1 INSTRUMENT TUBING SUPPORTS
UTILIZED BY ONE ARCHITECT ENGINEER
(SAMPLE REQUIREMENTS ONLY)

PROTECTION OF INSTRUMENTS, INSTRUMENT RACKS, AND SENSING LINES
FROM THE EFFECTS OF PIPE RUPTURE AND SEISMIC EVENTS

Example of Requirements for Seismic
Category 1 Instrument Tubing Supports

The following is a sample set of seismic requirements for supporting instrument tubing.

Criteria for Installation of Tube Supports in
Seismic Category 1 Safety Class 2 or 3 Installations

1.0 Material

Tube track or equivalent shall be used to support tubes in Safety Class 2 or 3 installations. Tube track and other tube support appurtenances, including structural steel, require manufacturers' material certificates of compliance.

1.1 Tube Track - (channels, angles, flat stock) - 12 gauge hot dipped galvanized steel, ASTM A569-72.

1.2 Clamps - (yoke, space bundle lock) - 302 or 304 stainless steel, ASTM A240-75a or ASTM A493-74.

1.3 Tube spacer - 302 or 304 stainless steel, ASTM A240-75a.

1.4 Bolts - Hex nuts (truss head with hex nut, tubing bolts, all threaded rods) - zinc plated carbon steel, ASTM A307-74.

2.0 Installation of Tube Track

2.1 Structural members, such as angles, channels, etc., shall be used to support tube tracks. Tube track shapes are not acceptable

members to be used for support of tube-track tubing supports.

2.2 Tube track shall be placed close to walls wherever practical, but shall be approximately 2 inches minimum from wall.

2.3 Tube track located close to the wall shall be bolted to structural steel angle members which are in turn welded to embedded plates in the wall. (See Figure IV-6-AT8.)

2.4 If embedded plates are not available for attaching the tube tracks to walls, the following methods are considered acceptable alternatives:

2.4.1 For concrete walls, use Phillips Red Heads. (See Figure IV-6-AT9 for typical detail.)

2.4.2 For brick walls, use bolting through walls and through support plate or other side of wall. (See Figure IV-6-AT10 for typical detail.)

2.5 Tube track may be suspended by structural steel from the ceiling above. The structural steel shall be welded to embedded plate in ceiling. Likewise, similar installations can be made by supporting from the floor. (See Figure IV-6-AT4 and IV-6-AT6.)

2.6 Tube track is to be supported at maximum spans of 6 feet for horizontal or vertical runs.

2.7 Tube track shall be covered to enclose tubes in areas that may prove hazardous to uncovered tubes such as potential missile areas. The covers shall be installed after all tests and inspection of the tubing have been completed. Site engineering shall report areas in which tubing must be enclosed within the tube track.

3.0 Placement of Tubing

3.1 Tubing shall be placed in single tiers in either the horizontal or vertical plane with preference given to the vertical plane. Two or more tiers shall not be used since inspection will be made difficult.

3.2 Tubing shall be placed in accordance with separation requirements, if applicable.

3.3 Tubing that may become thermally hot shall follow the criteria for supporting thermally hot tubing issued by mechanical site engineering. Care should be taken to place the tube clamps with spacers to allow for free thermal expansion where required and also to restrain the tube from movement where required.

3.4 Tubing joints shall be staggered, one from the other, inside the tube track. (See Figure IV-6-AT11.)

3.5 Tubing joints, either compression type or socket welded unions, will require inspection and tests of the joints. To provide inspection space, the tubing shall be mounted with a spacer between the tube and the tube track to enable viewing the entire joint with a mirror.

3.6 Tubing shall be supported not more than every 4 feet in the tube track.

4.0 Exceptions

It is recognized that all installations situations cannot readily be defined and that in certain instances variations from the criteria will be necessary.

Those installations which must vary from the criteria must be approved by site engineering.

5.0 Use of this Criteria for Non-Seismic Class 1E Installation

The criteria set forth herein shall also be used for Non-Seismic Class 1 installations wherever practical to keep the entire installation as uniform as possible.

PROTECTION OF INSTRUMENTS, INSTRUMENTS RACKS
AND SENSING LINES FROM THE EFFECTS OF PIPE
RUPTURE AND SEISMIC EVENTS

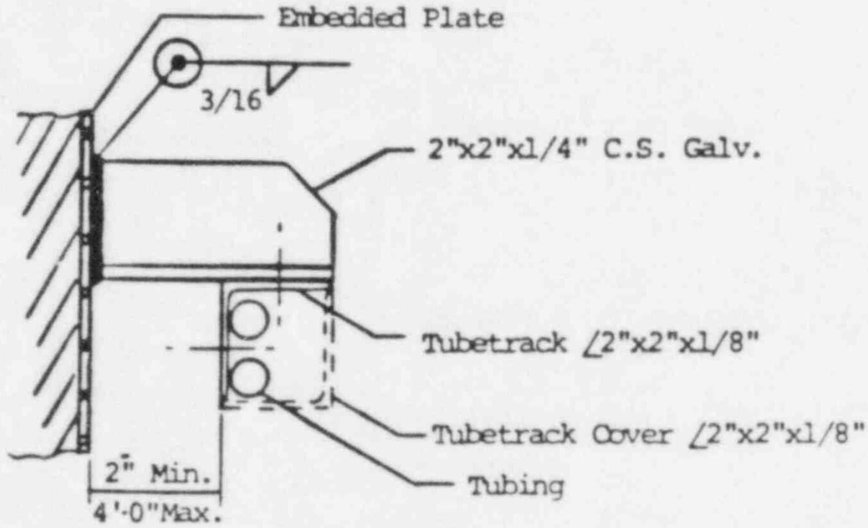
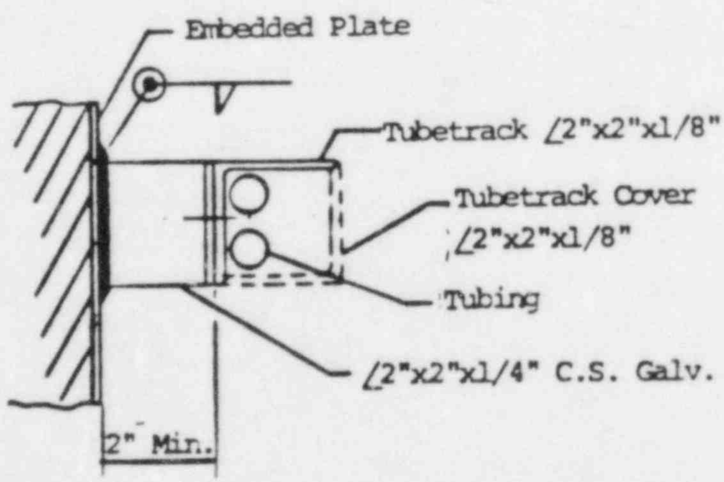


Figure IV-6-AT1
2 Tube Support From Wall



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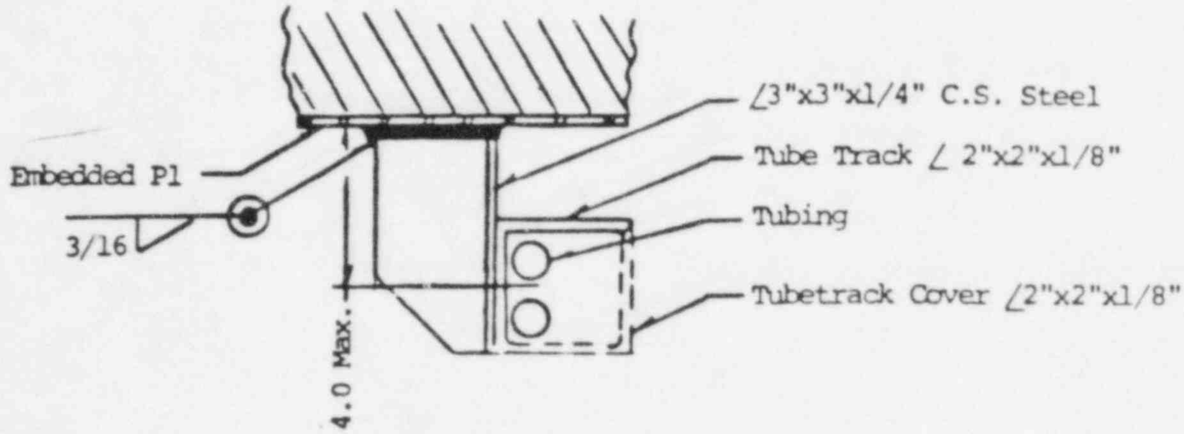

Figure IV-6-AT3
2 Tube Support From Ceiling

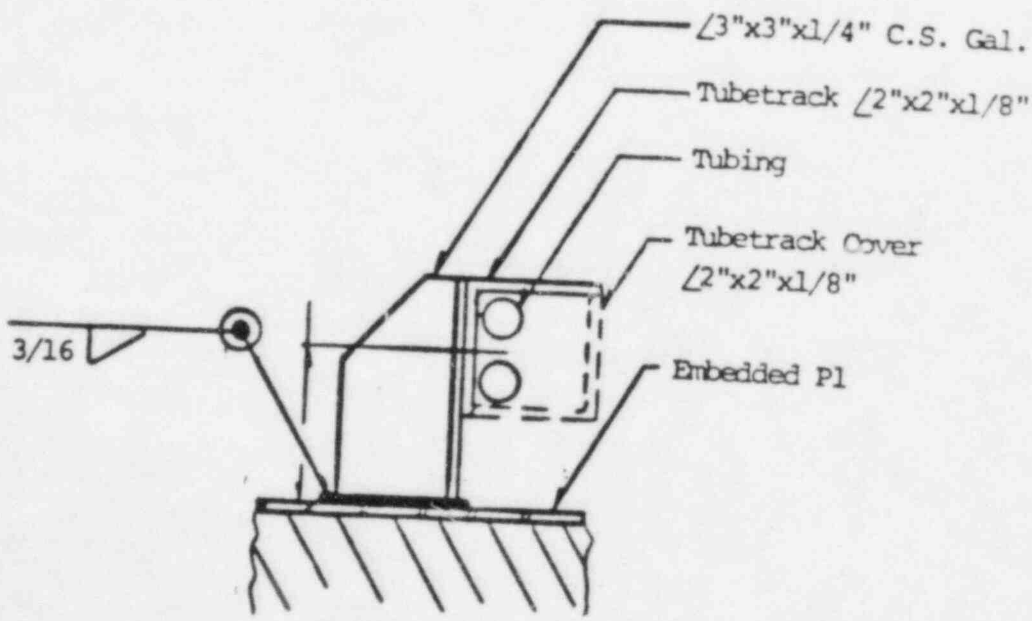

Figure IV-6-AT4
2 Tube Support From Floor

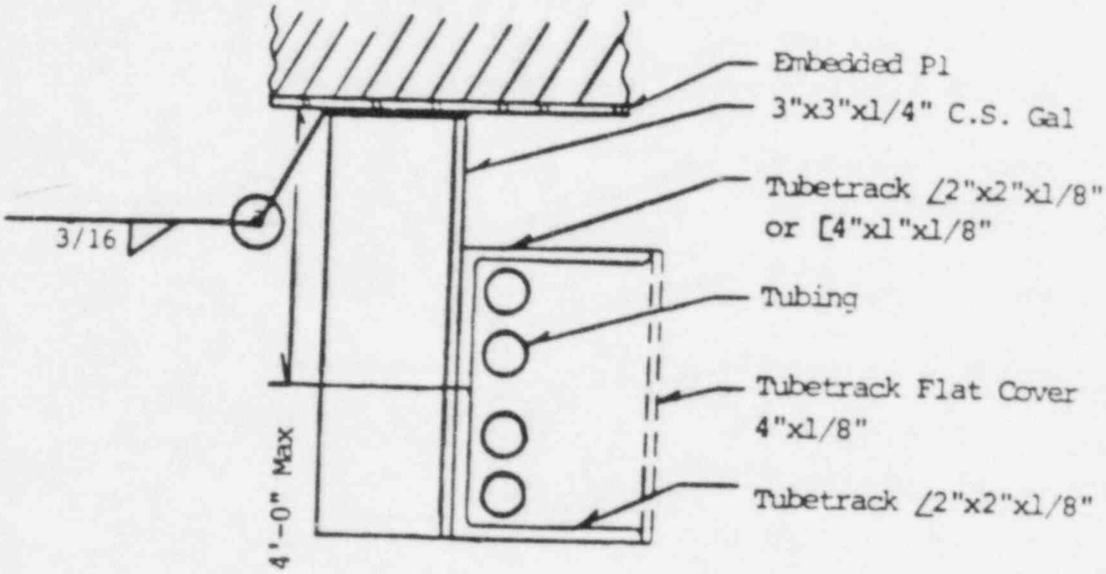

Figure IV-6-AT5
4 Tube Support From Ceiling

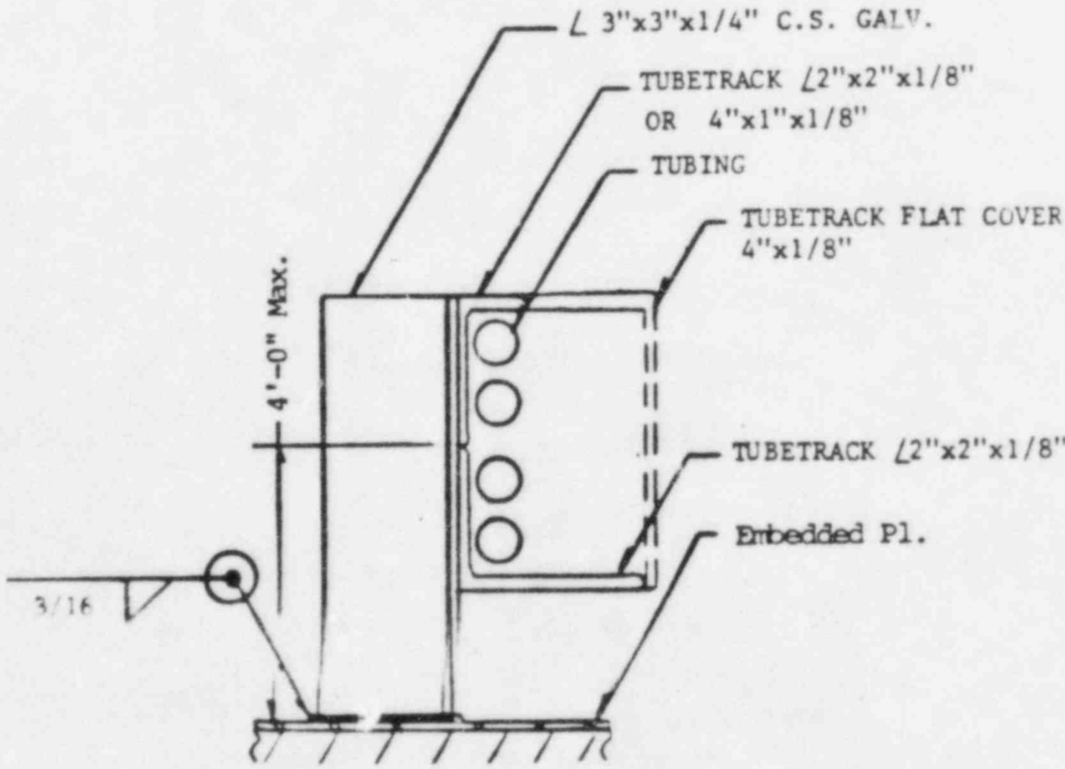

Figure IV-6-AT6
4 Tube Support From Floor

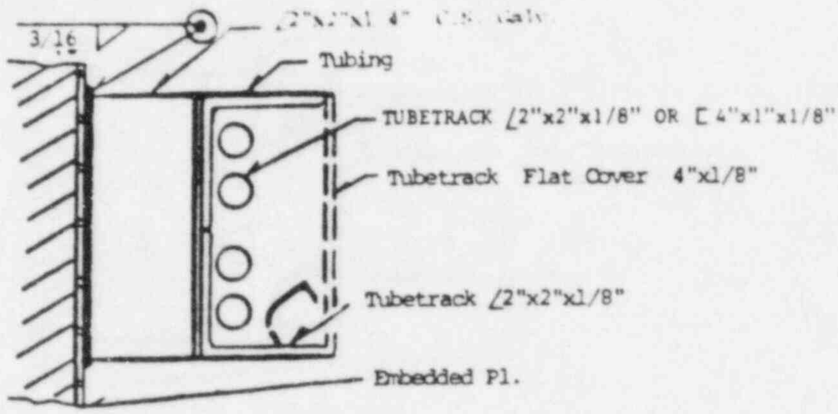

Figure IV-6-AT7
4 Tube Support From Wall

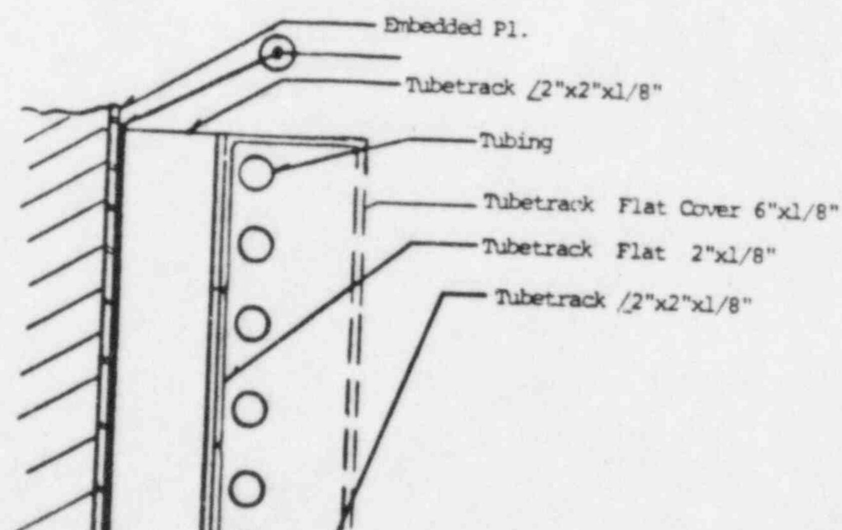

Figure IV-6-AT8
6 Tube Support From Wall

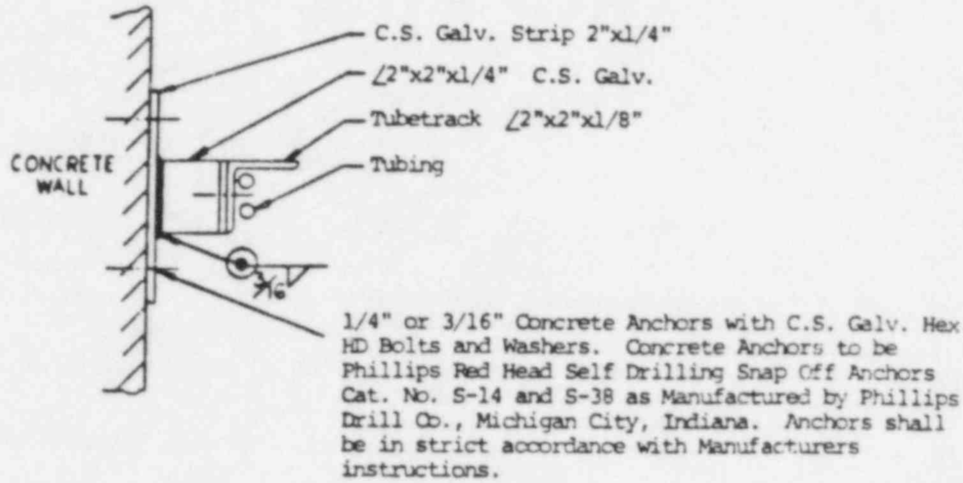

Figure IV-6-AT9
Concrete Wall Support Without Embedded IL

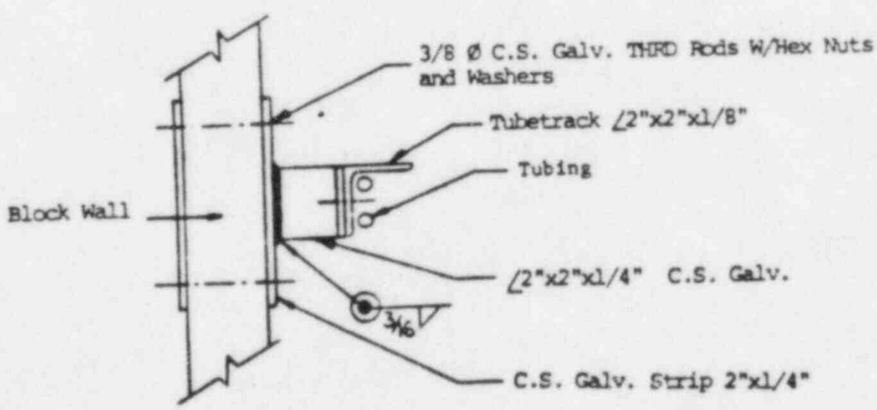

Figure IV-6-AT10
Block Wall Support Without Embedded IL

PROTECTION OF INSTRUMENTS, INSTRUMENTS RACKS
AND SENSING LINES FROM THE EFFECTS OF PIPE
RUPTURE AND SEISMIC EVENTS

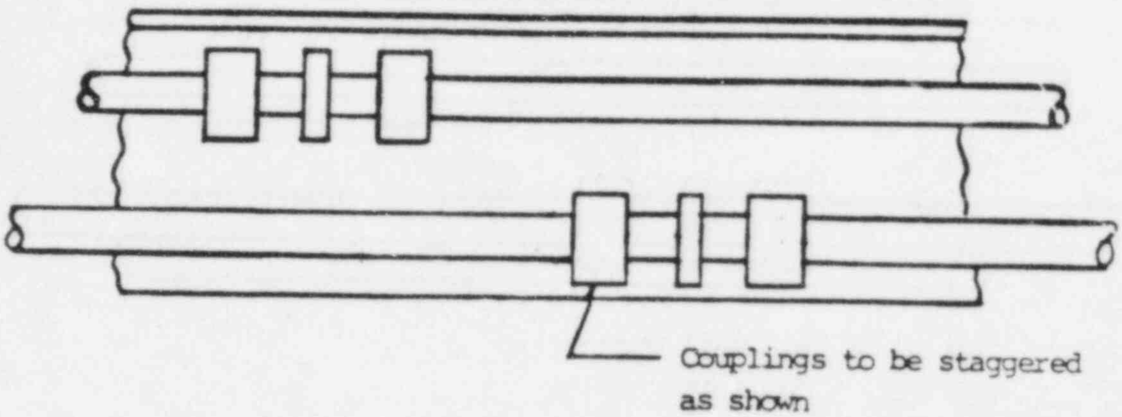

Figure IV-6-AT11
Typical Couplings

INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

Unit V - Missile Design and Construction Practices

PRE-STUDY TEXT

INSTRUMENTATION AND CONTROL TRAINING COURSE

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INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

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Unit V - Missile Design and Construction Practices

PRE-STUDY TEXT

1.0 OBJECTIVE

The USNRC inspector will be provided with background information on design philosophy and installation practices employed to protect safety-related instrumentation and control equipment from the effects of missiles and pipe rupture. This information will enable the inspector to recognize deficiencies at the construction site with regards to missile protection.

2.0 INTRODUCTION

The missile design and construction practices, which we will study in this unit, are simply an elaboration of the Single Failure Criterion as it applies to a specific cause of failure - namely, missiles. The Single Failure Criterion requires that the safety-related system be designed in such a manner that no single failure may cause a loss of power to safety-related equipment for a period long enough to permit a dangerous release of radioactivity as a result of a design basis event. One way to meet the Single Failure Criterion is to provide redundant systems and to prevent their simultaneous failure. Missiles are a potential cause of this "common mode" failure.

What is a missile? For our purposes, we may define a missile as an airborne projectile, traveling with enough kinetic energy to damage equipment which it impacts. There are a variety of types and causes of missiles, as we shall see.

While not itself a missile, a category of phenomenon closely related to missiles is pipe rupture. The types of damage it can cause, and the means of protecting against it, are similar to those for missiles. Thus, we have included pipe rupture in this unit.

The dangers of pipe rupture are those associated with jet impingement, high pressure fluid streaming out of the break, and pipe whip, the swinging of broken pipe end. Besides these physical effects of pipe rupture, consideration has to be given to the environmental effects of the fluid release on safety-related equipment and its ability to function. Of course, interactions between the safety-related instrumentation and control systems and the rest of the plant must be considered. We shall first investigate the causes and types of missiles; we will

then discuss the methods to protect the equipment against missiles.

This unit deals with material which is considered to be more pertinent to design than construction aspects. For example, the designer will include specific information on this subject in the SAR for USNRC review. It is unlikely that the construction inspector will be required to interpret the adequacy of a missile withstand design. Rather, his function will be to assure that the installation will be as per the approved-for-construction drawings which reflect the SAR's intent.

It should be noted that this subject matter is most appropriate to mechanical, nuclear, and structural engineering disciplines, not primarily to the instrumentation and control (I&C) engineering discipline. Inclusion of this unit in the I&C Program is primarily for overview treatment of this subject. In practice, the mechanical, nuclear, and structural engineering disciplines would actually perform the detailed engineering analyses to verify the adequacy of the I&C equipment to perform its function, or to provide for protective missile barriers for the I&C equipment.

2.1 Applicable Documents

This section lists the Code of Federal Regulations Criteria, USNRC Regulatory Staff positions, and draft national consensus documents applicable to missile design and construction practices.

2.1.1 10CFR50, Appendix A, General Design Criteria for Nuclear Power Plants

- Criterion 2, "Design Bases for Protection Against Natural Phenomena"
- Criterion 4, "Environmental and Missile Design Bases"
- Criterion 22, "Protection System Independence"
- Criterion 24, "Separation of Protection and Control Systems"

General Design Criterion 2 requires that nuclear power plant structures, systems, and components important to safety, be designed to withstand the effects of natural phenomena (e.g., earthquake, tornado, hurricanes, etc.). These natural phenomena may produce missiles which may be harmful to safety systems unless adequate design and construction means are provided to preclude damage.

General Design Criterion 4 requires that nuclear power plant structures, systems, and components important to safety, be designed to withstand the effects of environmental conditions and missiles which may include the

dynamic effects of a missile during postulated events, pipe whipping, and discharging of fluids.

General Design Criterion 22 requires that the protection system shall be designed to assure that the effects of natural phenomena, and of normal operation, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

General Design Criterion 24 requires that the protection system be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel, that is common to the control and protection systems, leave intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

2.1.2 USNRC Regulatory Guides

- RG 1.46 (Rev 0, 05/73), "Protection Against Pipe Whip Inside Containment"
- RG 1.53 (Rev 0, 06/73), "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems" (Addresses IEEE 379-1972)
- RG 1.75 (Rev 2, 09/78), "Physical Independence of Electric Systems" (Addresses IEEE 384-1974)
- RG 1.91 (Rev 1, 02/78), "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants"
- RG 1.115, "Protection Against Low-Trajectory Turbine Missiles"
- RG 1.117, "Tornado Design Classification"

2.1.3 USNRC Standard Review Plan NUREG-75/087

- Section 3.5.1.1, "Internally Generated Missiles (Outside Containment)"

- Section 3.5.1.2, "Internally Generated Missiles (Inside Containment)"

- Section 3.5.1.3, "Turbine Missiles"

- Section 3.5.1.4, "Missiles Generated by Natural Phenomena"

- Section 3.5.1.5, "Site Proximity Missiles (Except Aircraft)"

- Section 3.5.1.6, "Aircraft Hazards"

- Section 3.5.2, "Structures, Systems, and Components to be Protected from Externally Generated Missiles"

- Section 3.5.3, "Barrier Design Procedures"

- Section 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment"

- Section 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping"

It should be noted that the Standard Review Plan (SRP) is a document used by the Office of Nuclear Reactor Regulation staff responsible for the review of applications for construction permits and operating licenses. The SRP presently available to the public is keyed to RG 1.70, Revision 2, in lieu of presently available RG 1.70, Revision 3 (November, 1978), "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition". The SRP does not require compliance by the applicant nor does it provide guidance as available from USNRC Regulatory Guides. However, in a practical sense, the SRP provides the most thorough treatment of the requirements relating to missile criteria and is the document used as a checklist (not mandatory) for completeness of design.

2.1.4 <u>USNRC Branch Technical Positions</u>

- AAB 3-2, "Tornado Design Classification"

- APCSE 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment" with appendices

- MEE 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment"

The Branch Technical Positions (BTP) serve as specific guidance to the regulatory staff by a technical branch such as the Instrumentation and Control System Branch (ICSE), the Power System Branch (PSB), the Mechanical Engineering Branch (MEE), etc. These BTP's are an adjunct to the Standard Review Plans; therefore, the inspector is referred to the commentary of Paragraph 2.1.3 above.

2.1.5 IEEE Standards

- IEEE 379-1977, "Application of the Single-Failure Criterion to Nuclear Power Generating Station Class 1E Systems"

- IEEE 308, "Class 1E Systems for Nuclear Power Generating Stations"

- IEEE 279, "Criteria for Protection Systems for Nuclear Power Generating Stations"

- IEEE 384-1977, "Criteria for Independence of Class 1E Equipment and Circuits"

2.1.6 ANSI Draft Standards

- ANSI N176, June, 1974 Draft, "Design Basis for Protection of Nuclear Power Plants Against Effects of Postulated Pipe Rupture" (Superseded by ANSI/ANS 58.2)

- ANSI N177, April, 1977 Draft, "Plant Design Against Missiles"

- ANSI/ANS 58.1, "Plant Design Against Missiles"

- ANSI/ANS 58.2 Draft, November, 1978, "Design Basis for Protection of Nuclear Power Plants Against Effects of Postulated Pipe Rupture"

No consensus industry standard on the unit subject matter exists. The above listed draft standards were an industry attempt at consensus; however, the degree of information contained in these standards are less comprehensive than that contained in the documents described in Paragraphs 2.1.2 through 2.1.4.

2.2 Definitions of Missile Hazard-Related Terms

Certain terms used in this unit must be defined in order to clarify the intent of their use. For example, the unit describes typical designs in a missile hazard area, therefore, the definition of what a missile hazard area is must be understood. The following

are the significant missile-related terms:

Missile - An airborne projectile traveling with a high enough kinetic energy to damage equipment into which it impacts. For example, this projectile may be a standard roadway lighting pole, an automobile which has become a missile due to a tornado, a failed part of a large rotating machine, or a jet of fluid from a ruptured pipe.

High Energy Piping Systems - Those piping systems that, during normal plant conditions, are either in operation or maintained pressurized under conditions where either:

- The maximum operating temperature exceeds 200°F, or

- The maximum operating pressure exceeds 275 psig.

Moderate Energy Piping Systems - Those piping systems that, during normal plant conditions, are either in operation or maintained pressurized under conditions where both:

- The maximum operating temperature is 200°F or less, and

- The maximum operating pressure is 275 psig or less.

Missile Hazard Areas - Areas where high energy rotating equipment is operated, areas containing an operating crane, or wherever there is potential for locally generated missiles.

Pipe Failure Hazard Areas - Areas containing high and moderate energy piping which has a potential for cracking or rupture, and which would subject adjacent systems and equipment to jet impingement, pipe whip forces, or an adverse environment. Pipe rupture is the loss of the pressure integrity of the pipe from circumferential breaks, longitudinal breaks, and through-wall cracks.

3.0 MISSILE DESIGN AND CONSTRUCTION PRACTICES

The causes and types of missiles can be grouped into two broad and general categories: (1) natural phenomena-induced or other events encompassing a large plant area (e.g., redundant divisional areas) and (2) postulated phenomena-induced or other events encompassing a limited area. These categories, in general, are consistent with GDC-2 and GDC-4 coverage as natural phenomena and postulated phenomena.

3.1 Natural Phenomena or Extensive Area Events

Natural phenomena can produce a large number of missiles, directed randomly throughout the plant. Because these missiles can simultaneously bombard many systems and equipment, the phenomena can cause a common mode failure of the redundant safety system, unless plant missile protective features are provided.

This category includes several kinds of natural phenomena, namely earthquakes, tornadoes, and hurricanes.

Certain types of man-induced extensive area events can also be included in this category. These are events that can produce either a large number of missiles randomly directed throughout the plant or a single missile so powerful that it can damage a large collection of systems and equipment.

3.1.1 Earthquake

Earthquakes can cause structures to crumble, producing missiles in the form of falling debris. Earthquakes can vibrate cranes to the point where they drop their loads - potentially massive missiles. Seismic stresses can break equipment loose from its mountings and send it flying about the plant. This is yet another form of missile. Finally, seismic vibrations can cause pipes to rupture. Any or all of these phenomena can occur when there is an earthquake. They can occur simultaneously in a variety of locations throughout the plant.

3.1.2 Tornado

The most severe form of wind disturbance that a nuclear power plant could face is a tornado. A tornado can pick up anything from a small wood plank to an automobile and hurl it through the air with tremendous momentum. Unprotected, any piece of plant equipment could be severely damaged by a tornado-propelled missile. The number and direction of these missiles are completely unpredictable.

3.1.3 Hurricane

While hurricane winds are not as severe as tornado winds, they are still capable of generating potentially damaging missiles. As with tornadoes, the number and direction of missiles are unpredictable, so that the entire plant is subject to their effects. In addition, hurricanes last longer than tornadoes. Thus, in the course of a hurricane, there may be a large cumulative number of missile strikes which, though not occurring exactly simultaneously, will have the same effect on unprotected equipment as if they did.

3.1.4 Man-Made Phenomena

Certain types of man-made phenomena can produce either a large number of missiles randomly directed throughout the plant or a single missile so powerful that it can damage a large collection of systems and equipment. Thus, these

phenomena can be classified as extensive area events. They include collisions by aircraft, vehicles, and ships, and explosions occurring in proximity to the plant site. An example of the latter would be the explosion of a truck carrying explosives on a road near the plant.

3.2 Postulated Phenomena or Limited Area Events

Single event-type missiles are confined to a limited area of the plant. The major types of events within this category include:

- Rotating equipment failure (burst blade on turbine-generator, pump, motor, etc.)

- Accidental release of crane load

- Ejection of a pressurized component of a high energy piping system (valve stems and bonnets, temperature and pressure sensor assemblies, etc.)

- Localized explosion

- Pipe rupture

Pipe rupture may be included in this category of single event missiles. The dangers of pipe rupture are those associated with jet impingement, high pressure fluid streaming out of the break, and pipe whip - the swinging of a broken pipe end. Besides these physical effects of pipe rupture, consideration has to be given to the environmental effects of the fluid release on safety-related equipment and its ability to function.

The following list covers the major causes of limited area event missiles; however, it is not meant to be all inclusive.

3.2.1 Equipment Failure

- Rotating Equipment

  Rotating equipment is a potential source of missiles. Such missiles could be generated if a rotating part burst. The turbine-generator is of special concern in this area because of its large size and ability to go into overspeed should the overspeed trip fail. The same applies to any equipment which could, through malfunction, go into excessive overspeed. This primarily includes pumps, motors, and diesel generators. It is essential that the maximum turbine generated missile (most likely, turbine blade) be established as to size, energy, and flight pattern.

- Crane Loads

The loads attached to cranes are potential missiles. The crane may be anything from the large main gantry crane to a small temporary hoist. Failure of the crane itself or an operator error can turn the crane load into a damaging missile. The cable can snap or unravel uncontrollably (due to failure of the cable brake mechanism), causing the load to fall. If the crane is in motion to begin with, the resulting missile may have a horizontal as well as vertical trajectory. An improperly attached load can likewise fall off. Finally, crane operator error may cause a load to fall on, or crash sideways into, a piece of equipment.

- Pressurized Components on High Energy Piping Systems

High energy piping systems (further defined below in Paragraph 3.2.3) contain certain elements held in place under high pressures. These include valve stems and bonnets, and temperature and pressure sensor "pipe well" assemblies. Failure of the means by which these items are held in place could eject them into the plant as missiles.

Temperature or other detectors, installed on piping or in wells, are evaluated as potential missiles if failure of a single circumferential weld would cause their ejection.

Valve stems are not considered potential missiles if at least one feature in addition to the stem threads is included in their design to prevent ejection. Valves with backseats are prevented from becoming missiles by this feature. In addition, air or motor operated valve stems are effectively restrained by the valve operators.

Nuts, bolts, nut and bolt combinations, and nut and stud combinations have only a small amount of stored energy, and thus are generally of little concern as potential missiles.

3.2.2 Explosions

Any explosion in the plant may send out a shower of missiles in the form of debris in its immediate vicinity. Examples of items that can explode are: diesel-generators, the turbine-generator, batteries, high-voltage power circuit breakers, pressure vessels, and vessels containing oil or other combustible fluids. The latter could explode during a fire.

3.2.3 Pipe Rupture

Two categories of piping must be considered: high energy lines and moderate energy lines. High energy piping systems are defined as those systems that, during normal plant conditions, are either in operation or maintained pressurized under conditions where either:

- The maximum operating temperature exceeds 200°F, or

- The maximum operating pressure exceeds 275 psig.

Moderate energy fluid systems are defined as those systems that, during normal plant conditions, are either in operation or maintained pressurized under conditions where both:

- The maximum operating temperature is 200°F or less, and

- The maximum operating pressure is 275 psig or less.

The difference in energy between the two categories produces different effects when a pipe rupture occurs.

The energy of the fluid in high energy lines is sufficient to propel the broken pipe into pipe whip. The whipping pipe can damage equipment that it contacts. In addition, the force of the jet stream pouring out of the break may be high enough to damage equipment that it impinges on. For example, the impingement force which will cause a cable tray to buckle is perhaps several hundred pounds (the specific value being dependent on tray type and margin used by the designer) applied on the projected impingement area of the tray. Such a force is well within the capability of some piping lines found in nuclear plants. Finally, the environmental effects of the fluid release must be considered. These include wetting, flooding and compartment pressurization, and depending on the fluid's characteristics, corrosive action, steam atmosphere, and high temperature effects.

The effects of a moderate energy pipe rupture are less drastic than for high energy lines because of the lower energy of the fluid. Rather than an outright break, we would usually expect a crack and an ensuing through-wall leakage of the fluid. Therefore, for moderate energy piping, pipe whip and jet impingement need not be considered: however, the environmental effects must be. These include wetting, flooding, and possibly corrosive action.

3.3 Protection from Missiles and Pipe Rupture

The primary objective in protecting the safety-related instrumentation and control system from missiles and pipe rupture is to prevent common mode failure of redundant systems which would result in a loss of protective function. This can be achieved by locating the redundant systems such that both systems would not be susceptible to the same damage.

3.3.1 Protection from Natural Phenomena or Extensive Area Events

As stated above, the primary objective in protecting the safety-related instrumentation and control system from missiles and pipe rupture is to prevent common mode failure of redundant systems which would result in a loss of protective function. How is this accomplished for natural phenomena or extensive area events? The design generally provides for the protection of all I&C system components from the direct effects of extensive area events, and in addition, from the effects of expected failures of equipment or structures which are not designed to withstand the event. (Note: The following discussion applies only to the on-site power system. Upon occurrence of the severest natural phenomena or extensive area event, the off-site power system can be sacrificed without loss of the protective function.)

3.3.1.1 Earthquakes

To protect the safety-related instrumentation and control system from earthquake generated missiles, the designer typically reduces the number of potential missile sources and separates safety-related instrumentation and control equipment and cables from the remaining sources.

The number of potential missile sources is reduced first by locating safety-related instrumentation and control components within Seismic Category 1 structures. This prevents missiles, in the form of debris, from crumbling structures. In addition, all safety-related instrumentation and control equipment and their mountings are seismically qualified. This prevents the safety-related instrumentation and control equipment itself from being flung about as missiles. The restraints are used on safety system pipes to prevent their rupture. All cranes in the area of safety-related instrumentation and control equipment are generally seismically qualified. We shall learn more about protection from pipe

rupture and missiles due to cranes in Paragraph 3.3.2.1, which deals with them as single events.

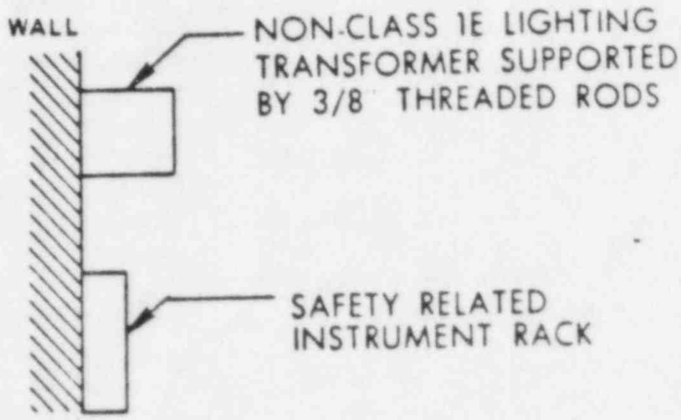
These methods greatly reduce the number of potential missile sources. What sources remain? Non-safety-related instrumentation and control, and non-safety system equipment in general, are not usually seismically qualified. Their failure may produce missiles and pipe rupture against which the safety-related instrumentation and control system must be protected. This is accomplished by physically separating them from the safety-related instrumentation and control system by means of distance, barriers, and safety class structures. In some instances, instead of this separation approach, non-safety system equipment is installed or mounted as Seismic Category 1 to preclude its becoming a missile. Figure V-6-1 shows a situation with a potential for a seismically induced missile to cause failure of Class 1E equipment.

3.3.1.2 Tornadoes and Hurricanes

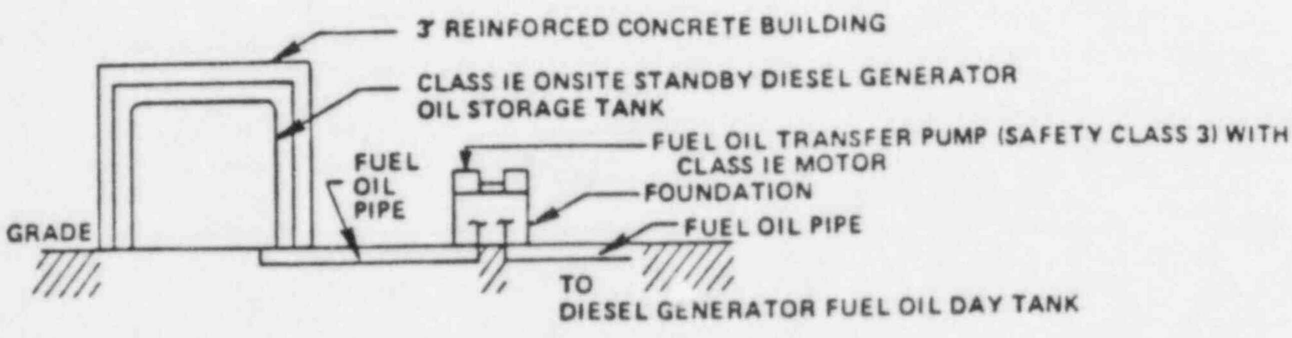
The safety-related instrumentation and control system is normally protected from missiles generated by tornadoes and hurricanes by locating all safety-related instrumentation and control components in buildings designed to withstand the missiles, as well as the storm itself. Where necessary, additional missile barriers may be used to protect specific safety-related instrumentation and control equipment. The idea is to prevent any safety-related instrumentation and control element from being struck by a missile.

It is interesting to note that the reason separation of redundant divisions is insufficient to protect against tornado and hurricane generated missiles may be regarded as either: a) multiple missiles may destroy both divisions, or b) a single missile may destroy one division concurrent with a single equipment failure in the other division.

An example of a potential installation is provided on Figure V-6-2 which should be suspect due to a tornado or hurricane induced missile.



Potential for Seismically Induced
Missile Caused Failure of Class 1E Equipment
Figure V-6-1



Potential for Tornado/Hurricane Induced Missile
Caused Failure of Class 1E Equipment
Figure V-6-2

3.3.1.3 Man-Made Phenomena

To protect the safety-related instrumentation and control system from man-made extensive missiles, it is typical to either minimize the probability of occurrence of the phenomena or provide barriers against any missiles that may occur. Probability of occurrence is minimized, for example, by locating the plant site away from airports, air corridors, and highways. As in the case of protection against tornadoes and hurricanes, the barriers usually consist of buildings designed to withstand the impact.

It is not generally considered within the inspector's responsibility to determine the adequacy of the probability analysis for the plant. Significant work relating to this aspect is performed and/or reviewed by the regulatory staff (e.g., USNRC Standard Review Plan NUREG-75/087, Section 3.5.1.6, "Aircraft Hazards").

3.3.2 Protection from Postulated Phenomena or Limited Area Events

The primary means of protecting safety-related instrumentation and control systems located in missile hazard areas from common mode failure due to single event missiles is to separate redundant system components by an adequate distance or by a barrier. The barrier should have an energy absorbing capacity consistent with the design basis missile for the area. This way, the missile is allowed to damage only one (at most) of the redundant components.

Alternative methods of protecting against single event missiles include:

- Spatial Arrangement

  Missiles generated by rotating equipment usually have a well defined trajectory. By locating potential targets outside the range of this trajectory, the designer can achieve adequate protection without resorting to barriers or large separation distances.

- Barrier Around Missile Source Itself

- Appropriate Layout to Avoid Locating Safety-Related Instrumentation and Control Equipment in Missile Hazard Areas

As we shall see, there are some modifications required to this approach, such as in the case of pipe rupture. However, it is a good general rule to follow for protection against limited area events.

Let us see how this design philosophy is implemented in actual practice.

3.3.2.1 Missile and Pipe Failure Hazard Areas

The first step is for the designer or inspector to identify the areas of the plant subject to single event missiles and pipe rupture. The same area may have both missile and pipe failure hazard and be known as a simple hazard area.

Missile hazard areas (mechanical damage areas) are areas where high energy rotating equipment is operated, or areas containing an operating crane, or wherever there is a potential for locally generated missiles. Material movement routes or the areas in the vicinity of hatches pose special hazards. Heavy equipment maintenance areas are also classified as mechanical damage areas.

Note that the missile generated must have high enough kinetic energy to damage the equipment exposed to it in order to identify the area as a missile hazard area. Thus, for example, if the only potential missile source was a one hp motor, the area probably would not qualify as a hazard area. The designer is expected to perform an analysis of the equipment to be protected and the missile source to make this determination.

Pipe failure hazard areas are areas containing high and moderate energy piping which has potential for cracking or rupture, and which would subject adjacent systems and equipment to jet impingement, pipe whip forces, or an adverse environment.

The following criteria are then applied by the designer and documented in the design drawings subject to inspection for the safety-related instrumentation and control systems (and safety systems in general) situated in these areas.

3.3.2.2 Separation in Missile Hazard Areas

In general, redundant system components in

missile hazard areas are separated by adequate distance or by a barrier having an energy absorbing capacity consistent with the design basis missile for the area. This involves separation between redundant safety-related instrumentation and control divisions, as well as separation between each division of the safety-related instrumentation and control system and mechanical, control, and other parts of the opposite safety division.

Maximum use is generally made of inherent barriers such as floors, walls, and massive pieces of equipment.

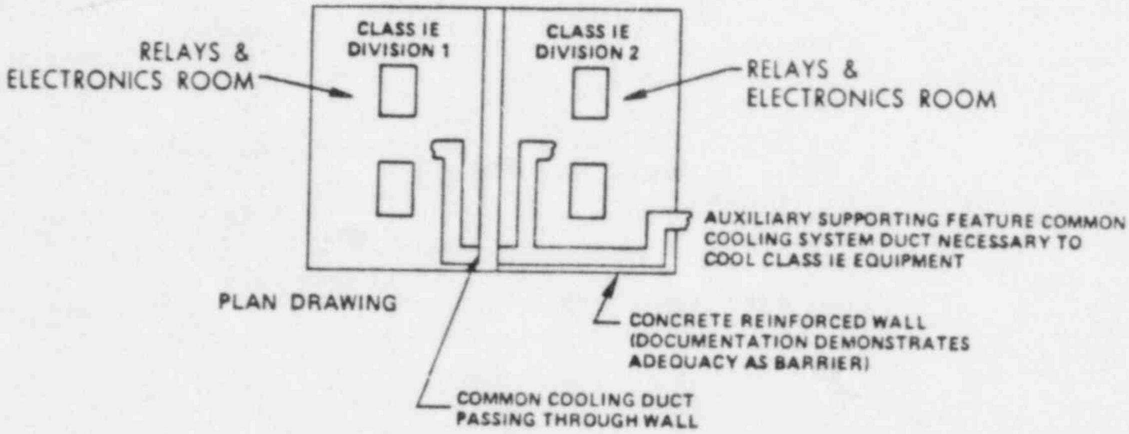
Where inherent barriers are not available and wide separation distances are impractical, barriers can be erected specifically to provide the necessary separation. These may be single barriers or multiple barriers. A single barrier is one, usually heavy-duty, partition intended to completely stop the design basis missile. A multiple barrier is a set of lighter duty partitions, usually in a parallel arrangement. The first few partitions are not strong enough to stop the design basis missile; however, they slow it down enough so that it is stopped before penetrating through the final partition. If a multiple barrier is counted on to protect an essential system, the protection is deemed adequate if the first partition will stop the missile without generating secondary missiles that could damage the system. Likewise, this "backface scabbing" is also unacceptable for single barriers.

The amount of separation, strength of the barrier, etc., depends on the source trajectory, and kinetic energy of the anticipated missile. This must be evaluated on an individual basis for each missile hazard area.

In any area containing an operating crane, raceways and equipment associated with redundant systems are generally separated by a physical barrier and/or distance so that the largest dropped load cannot result in the loss of more than one division.

Figure V-6-3 illustrates an installation of redundant Class 1E I&C equipment with a barrier wall between the redundant units. This

particular example is suspect, as a single HVAC system duct was used to cool Division 1 and Division 2 Class 1E equipment. However, the barrier wall between the redundant divisions reflects a design concept which provides a barrier to preclude common mode failure.



Separation of Divisions by a Barrier Wall
Figure V-6-3

3.3.2.3 Spatial Arrangement in Missile Hazard Areas - Main Turbine-Generator Failure

In the case of missiles generated by rotating machinery, the designer can take advantage of his knowledge of the missile's trajectory to achieve adequate protection without resorting to barriers or large separation distances. The method used requires an appropriate spatial arrangement of equipment.

Let us first consider the unit or main turbine-generator. Missiles from a turbine failure can be divided into two groups: "high trajectory" missiles, which are ejected upward through the turbine casing and may cause damage if the falling missile strikes an essential system, and "low trajectory" or "direct" missiles, which are ejected from the turbine missiles are characterized by their nearly vertical trajectories.

Missiles ejected more than a few degrees from the vertical, either have sufficient speed such that they land off-site, or their speeds are low enough so that their impact on most plant structures is not a significant hazard. Low trajectory missiles, on the other hand, can be damaging within a fairly well defined strike zone (see Figure V-6-4).

Evidence currently available indicates that low trajectory turbine missile strikes will be concentrated within an area bounded by lines inclined at 25 degrees to the turbine wheel planes and passing through the end wheels of the low pressure stages. This applies to the low pressure stage shrunk-on wheels of the 1800 rpm turbines generally used with light water-cooled reactors. Essential systems within this area and close to the turbine axis are the most vulnerable; those further removed from the turbine axis are less likely to be hit by a missile. Systems outside this area are not endangered by high energy low trajectory missiles. Thus, the preferred method trajectory turbine missiles is to locate them outside the strike zone pursuant to USNRC Regulatory Guide 1.115 (Rev 1, 07/77), "Protection Against Low Trajectory Turbine Missiles".

Where safety-related instrumentation and control systems must be located in the strike zone, redundant equipment must be separated by distance and barriers so as not to be damaged simultaneously. As an alternative, buildings which lie in the strike zone and which contain essential systems can be constructed to withstand the impact of the missile. As a matter of fact, this is generally done in some cases, such as the control room. A missile barrier could also be placed around the turbine-generator itself. Of course, reliable overspeed protection will also greatly reduce the probability of occurrence of turbine missiles to begin with.

Missiles generated by two smaller redundant pieces of high energy rotating equipment, located side-by-side in close proximity with their axes of rotation in parallel to each other, should be taken into consideration. This assumes that any missile that would be generated would leave the equipment normal or at right angles to the axis

of rotation. If such an arrangement is unavoidable, then the design will generally verify that the pieces of equipment have an adequate distance or a physical barrier between them.

3.3.2.4 Alternate Methods of Missile Protection

An obvious alternative to the philosophy of separating redundant divisions as a means of missile protection is to put a barrier around the missile source itself. For example, the inspector may expect to find large pumps and motors (usually above 200 hp) and pressure vessels (e.g., reactor coolant holdup tanks) installed in isolated cubicles (see Figure V-6-5). This way if the barriered equipment fails, the missiles and other effects of the failure are limited to the equipment itself and any of its auxiliaries confined with it. Nothing outside the cubicle is damaged, and so common mode failure is prevented.

A final method of protection against single event missiles is simply not to locate safety-related instrumentation and control components in missile hazard areas if at all avoidable. Separation of hazard areas from non-hazardous areas by means of distance, barriers, and safety class structures is a means of achieving this.

3.3.2.5 Design to Minimize the Effects of Pipe Rupture

The methods of protection in missile hazard areas outlined in Paragraph 3.2.2. - namely, separation - are broadly applicable to pipe failure hazard areas as well. However, these areas require more attention to separation than the others because of the large distance over which a whipping pipe or discharge from a ruptured pipe can cause physical or environmental damage to other equipment.

Two or more divisions of the safety-related I&C system must be adequately separated to prevent a pipe rupture from disabling more than one division. Spatial separation or distance is one approach; barriers are the alternative. These barriers are somewhat more complicated than missile barriers: physical protection from the whipping pipe and jet impingement shields must be provided if the piping is high energy; and unless the equipment being protected is qualified to withstand the environmental effects of the pipe

Figure V-6-4*
Low-Trajectory Turbine Missile Strike Zone
* Extracted from USNRC RG 1.115 (Rev 1, 07/77)

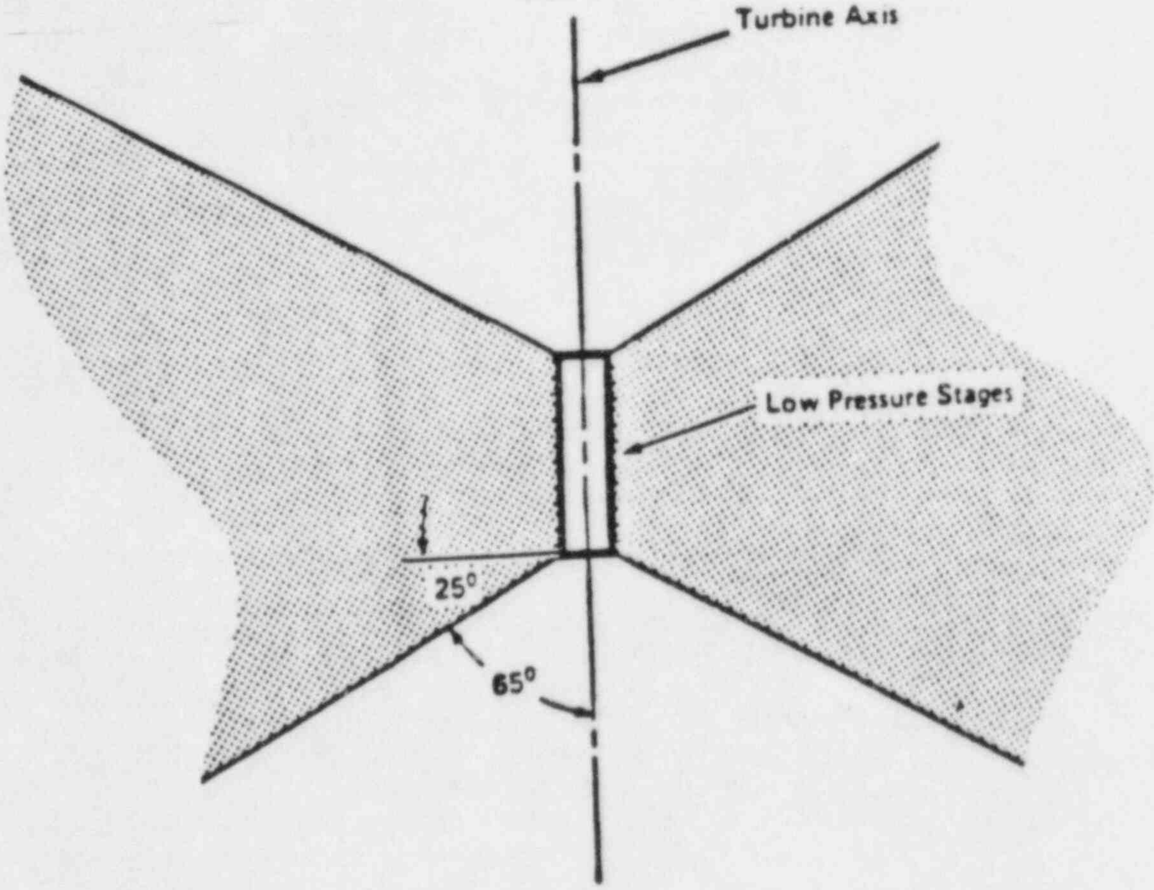
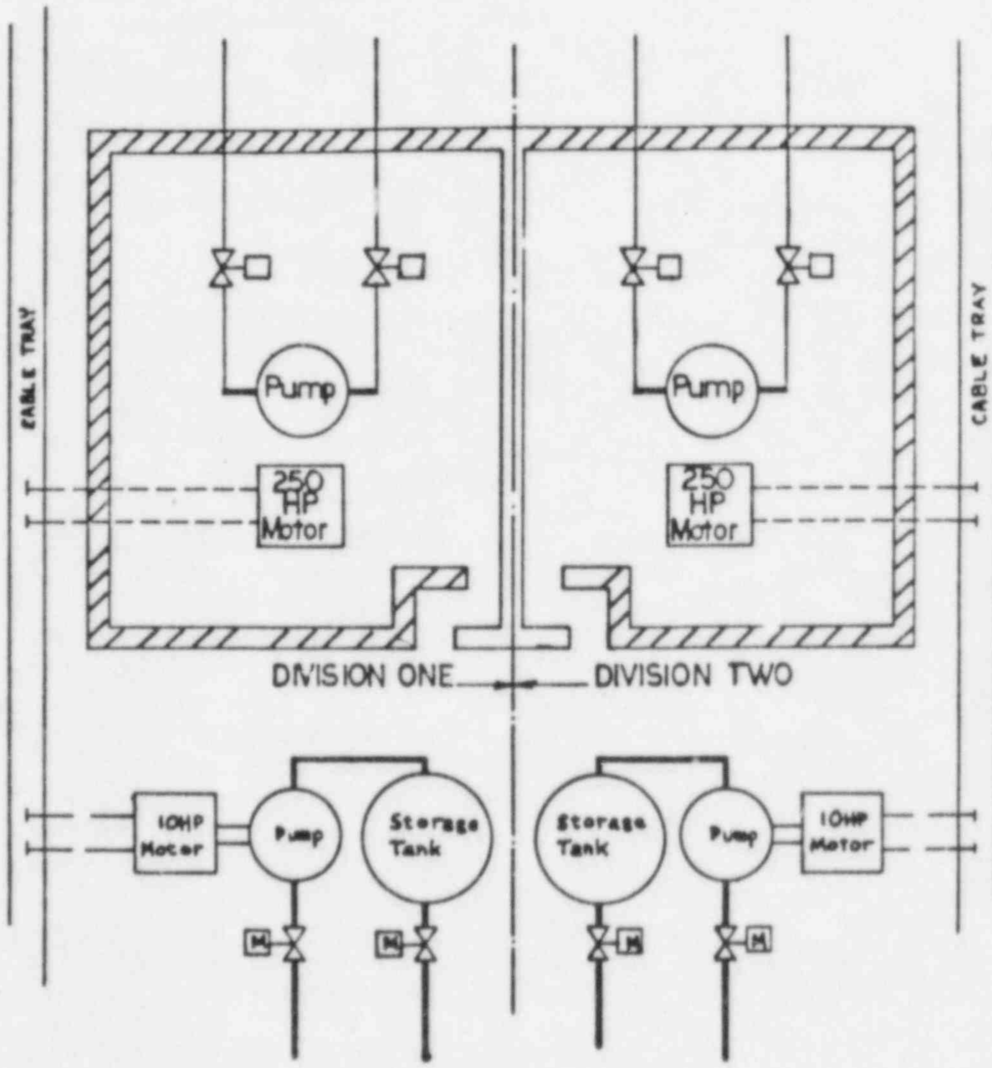


Figure V-6-5
Barriers Around Missile Sources



rupture, the barriers must attenuate or exclude the adverse environment from the protected equipment to a degree necessary to support the safety function.

The degree of separation, and the type, strength, and location of barriers is determined by the designer, based on an analysis of potential pipe failures. This analysis includes a determination of the most probable locations for breaks, and for each case, the type of rupture, ensuing forces, and environmental effects. (Regulatory Guide 1.46; Standard Review Plan, Section 3.6.2; and Branch Technical Position MEB 33-1 provide guidance for performing the analysis.) The inspector is not in a position to repeat this analysis. Rather, he might check to see that the protection provided is in general accordance with the SAR and SER. If an installation appears unreasonable to him, he may ask for justification. A fair amount of forethought must go into the design of pipe rupture protection. As an example of possible oversights to be aware of, consider the case of a diesel-generator that had been "completely" isolated from the effects of pipe rupture by enclosure in its own safety class structure. The inspector exited the diesel-generator building only to find a high-pressure non-ASME code class, non-Seismic Category 1 steam pipe passing a few feet from the diesel's air intake on the outside of the building. Its rupture would flood the intake with steam making it difficult, if not impossible, to start the diesel.

Sometimes it is not enough to prevent common mode failure by divisional separation. There is another criterion crucial to adequate pipe rupture protection, which can be stated as follows: for pipe breaks which require a protective function, any system which is required to place the reactor in a safe shutdown condition after the pipe break may not lose redundancy. In other words, if the rupture is itself a design basis event requiring a function, the Single-Failure Criterion requires that, in addition to all failures caused by the design basis event (e.g., loss of Division 1 switchgear), a single other failure (e.g., loss of Division 2 switchgear) must not prevent completion of the protective function. To satisfy this, the rupture must be prevented from

affecting equipment in either division of the safety-related I&C system - provided the equipment is needed to complete the protective function. This is accomplished by isolating the pipe in question by the following methods:

- Pipe restraints: prevent whipping of high energy pipes; will not attribute to environmental effects.

- Distance

- Enclosure of the piping; e.g., in a concrete pipe chase. Pipes within a pipe are sometimes used, the other pipe being known as a "guard pipe".

- Enclosure of the protected equipment (both divisions); e.g., cable trays in piping penetration compartments as they pass through the pipe failure hazard areas, or safety-related I&C equipment in watertight rooms in moderate energy pipe areas.

The methods are often combined. For example, the piping enclosure need not be made as strong if the pipe is also restrained within the enclosure. As with pipe failure protection via divisional separation, an analysis must be performed to determine an appropriate method. Piping system ruptures requiring a protective function cannot always be isolated from all safety-related I&C systems. In these instances, the analysis must verify that sufficient systems and components will still be available to carry out the protective function following the rupture.

Although a postulated pipe rupture may not require missile protection, it is sometimes more attractive for economic and practical reasons to protect against the rupture by isolating the pipe rather than by separating divisions. For instance, the spatial separation required to prevent pipe whip from simultaneously damaging two redundant cable trays might be prohibitively large, whereas simple use of pipe restraints would prevent damage to either tray. In fact, the inspector can expect to find pipe restraints on almost all high energy piping as the principal means of preventing pipe whip.

Examples of piping systems whose rupture requires a protective function, and thus must be isolated from safety-related I&C systems which are designed to mitigate the results of that rupture, if only two divisions exist, are the:

- Reactor coolant system

- Main steam line in containment

In summary, to ensure that the installation adequately protects against pipe rupture, the design must distinguish between piping whose failure requires a protective function or is a design basis event, and piping whose failure does not. In the case of the former, the design would typically ascertain that the effects of a rupture cannot cause a loss redundancy in elements of the safety-related I&C system needed for the protective function. In the case of the latter, the design will typically ascertain that the effects of a rupture cannot cause common mode failure of redundant safety-related I&C systems.

(NOTE: The USNRC's guidelines in this area have changed recently. Stated here are the latest guidelines. However, the inspector may encounter installations conforming to the old guidelines, which would be acceptable; otherwise, the construction permit would not have been granted. This, of course, does not preclude backfit pursuant to 10 CFR 50.199 restrictions for the health and safety of the public. For more specific guidance in this difficult area, the inspector should consult SRP Section 3.6.1 and BTP APCSB 3-1, including appendices.)

3 3.2.6 Installation of Equipment

Protection against missiles and pipe rupture is more an art than a science. There are few really specific rules. Rather, each situation is treated individually as it arises in accordance with the general criteria we have discussed. In this section, we shall consider some acceptable methods of installing certain common types of safety-related I&C equipment. The inspector must bear in mind that alternative installations are acceptable as long as they meet the general criteria.

3.3.2.6.1 Missile Hazard Areas

Utilizing an analytical approach, a design valve would be provided to separate redundant components (field tubing, I&C racks and cabinets) from missile producing sources in missile hazard areas by a minimum distance consistent with the identified hazards.

Two important rooms in which heavy machinery and high energy piping should not be placed are the main control room and the cable spreading room, pursuant to IEEE 384-1974, as modified and endorsed by USNRC Regulatory Guide 1.75 (Rev 2, 9/78).

3.3.2.6.2 Pipe Failure Hazard Areas

Safety-related instrumentation and control equipment located in the pipe failure hazard areas must be qualified to the worst expected environmental (temperature, pressure, humidity, etc.) conditions expected to be produced due to the pipe failure. The environmental qualification shall meet the requirements of the IEEE Standard 323-1974.

The phenomena of pipe rupture can cause environmental hazards to instrumentation and/or physical damage to instruments, sensing lines, electrical wiring, and instrument racks.

- Environmental Hazards

In the case of a rupture of a high or medium energy pipeline, flashing fluids or steam ejection from the break will cause an increase in temperature, pressure, and relative humidity of the space in which the pipe is located. Flooding and chemical sprays may also ensue. Safety-related instruments located within the space must be capable of withstanding for varying lengths of time

(depending on the instrument's function), the environmental changes caused by the rupture. Wherever practical, the safety-related instruments, with instrument taps located in spaces which will be pressurized in the event of a pipe rupture, should be located outside of the shield wall surrounding this space. Taking this precaution could either eliminate or significantly decrease the environmental hazard to the instrument itself.

- Physical Damage

  Pipe whip is generally prevented through the use of pipe restraints.

  The impingement force which causes a cable tray to buckle is calculated by the designer for the projected impingement area of the tray. Based on this and on a pipe failure analysis, the utility determines the minimum separation distance, distance "X" (without barriers) that will provide adequate protection against jet impingement. This valve is incorporated into the design as shown in Figure V-6-6. Note that in Figure V-6-6(a), rupture of either pipe would not damage either the Division A or B raceway. Hence, no loss of redundancy would occur. When a loss of redundancy can be tolerated, the minimum separation distance would be required only between the pipe of one division and the raceway of the other division, as in Figure V-6-6(b).

3.3.2.6.3 Missile and Pipe Failure Hazard Protection of Safety-Related Instrumentation and Control Equipment Found in the Design of Nuclear Power Plants

- Separation of Redundant Instrument Channels

  The impingement zone of the jet

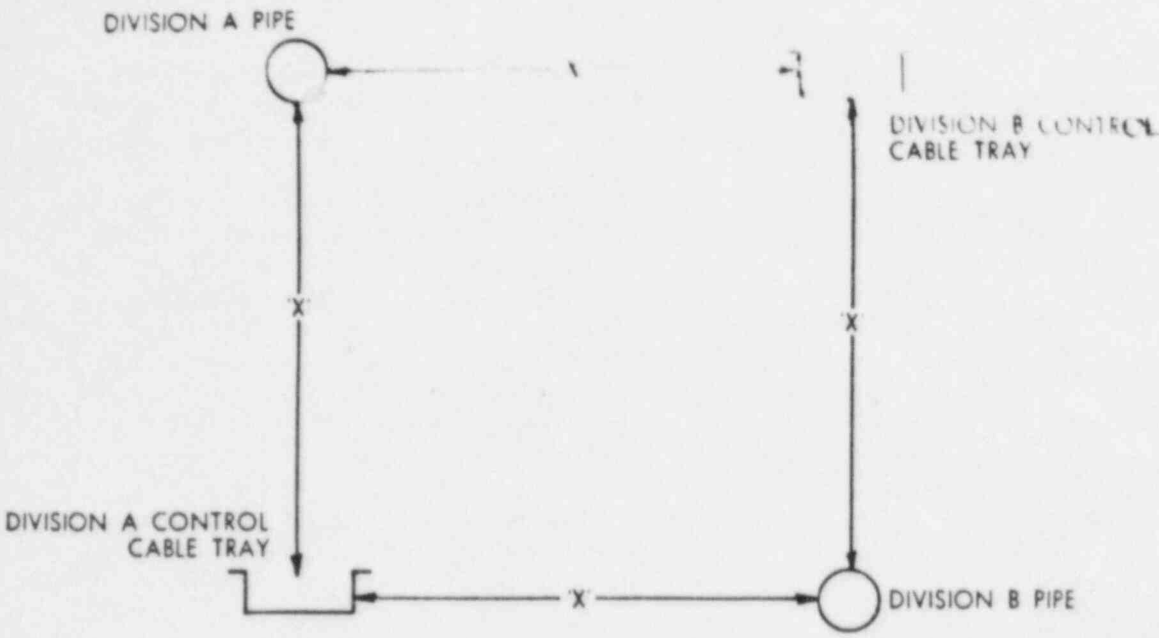
caused by a major, high energy line break will probably be quite large. Therefore, separation criteria (of a few feet) used for prevention of fire spreading from one channel to another may be quite inadequate to prevent physical damage to more than one redundant instrument channel. To best use separation as a means of preventing multi-channel failure due to pipe rupture, the postulated rupture points of the pipes in question should be obtained together with the predicted dimensions of the spray and pipe whip outlines. With this information, it will be possible in most cases to avoid having safety-related instrument channels located in the predicted damage zone of a single pipe rupture.

- Barriers Between Redundant Channels

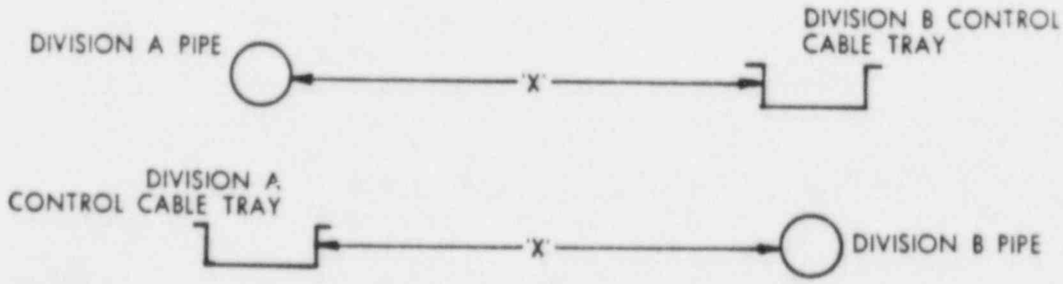
  At times, barriers such as reinforced concrete walls, floors, shield wall curvatures, or other type structures, can be used to keep redundant instrument channels from being damaged by a single pipe rupture. Advantage should be taken of these barriers wherever practical. The forces of jet impingement or pipe whip are so large that simple and lightweight barriers, such as instrument cabinet doors, lightweight sheet metal, etc., offer no effective protection.

- Instrumentation for Diverse Parameters

  There are certain cases, e.g., taps for redundant channels originating from the same pipeline or piece of equipment, in which neither adequate separation nor the use of a barrier is viable as means of ensuring that damage will not occur to more than one channel. In these cases, instruments using diverse parameters performing the same safety function



(a) No Loss of Redundancy



(b) May Lose Redundancy

FIGURE V-6-6

Protection of Cable Trays Against Jet Impingement.

(a) Pipe failure requires protective function - no loss of redundancy allowed.

(b) Pipe failure does not require protective function - loss of redundancy allowed.

are acceptable as a means of permitting a safe shutdown of the plant and/or providing alternate means of obtaining post-accident information.

It should be noted, however, that although some instrumentation may be designed as "fail-safe", the effects of pipe rupture, e.g., the crimping of instrument lines or the damaging of an instrument such that it fails to produce the proper signal output, will render the instrument as "non-fail-safe".

4.0 SUMMARY

This unit was intended to familiarize the USNRC inspector with the basic principles usually employed in the design of systems to protect the instrumentation and control equipment against the effects of pipe rupture and missile hazard. Protection against the effects of natural phenomena or extensive area events and postulated phenomena or limited area events was discussed in this unit.

INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

UNIT VI - Physical Isolation Requirements Between Redundant Safety Systems

PRE-STUDY TEXT

INSTRUMENTATION AND CONTROL TRAINING COURSE

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PRE-STUDY TEXT

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INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

UNIT VI - Physical Isolation Requirements Between Redundant Safety Systems

PRE-STUDY TEXT

1.0 OBJECTIVE

Design philosophy and construction practices employed to comply with the separation criteria for redundant Class 1E circuits and equipment in control boards, panels, and racks, are presented to the USNRC inspectors. Based on this information, the USNRC inspector will be able to determine how physical isolation is established between redundant safety systems.

2.0 INTRODUCTION

The NRC inspector should be aware that in the design and installation of instrumentation and control systems for a nuclear power plant, there are safety considerations over and above the normal safety considerations for a conventional fossil plant. In essence, the arrangement of the plant, indeed the entire design, has to be such that no single credible event, equipment failure, or natural phenomenon will result in the release of radioactive material which could unduly endanger the public.

It is recognized that some equipment failures are inevitable. In order that the design may meet its intent, certain restrictions are essential to assure safety. Redundant systems or channels have to be independent so that either system or channel may suffer any postulated failure without impairing the ability of the remaining system or channel to perform the required function.

Certain features of the plant that are essential to nuclear safety cannot be made redundant. The reactor containment structure is an example of this.

3.0 PHYSICAL ISOLATION REQUIREMENT BETWEEN REDUNDANT SAFETY SYSTEMS

Where there are at least two or more systems or pieces of equipment, the layout of the plant should be such that no two redundant systems or channels could be disabled by a common failure mode.

In all safety-related systems, redundant channels consisting of sensors, sensing lines, instrument racks or cabinets, wires and cables, logic components, and electrical control and power supply equipment are provided. Redundant sensor circuits in each trip system (sensors, transmitter, wiring, amplifiers, etc.) are electrically, mechanically and physically independent to satisfy the single-failure criterion.

3.1 Applicable Documents

3.1.1 NRC Regulations and Regulatory Guides

a. General Design Criterion 3, "Fire Protection", of Appendix A "General Design Criteria for Nuclear Power Plants", to 10 CFR Part 50 requires, in part, that structures, systems, and components important to safety be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires. General Design Criterion 17, "Effective Power Systems", requires, in part, that the on-site electric power supplies, including the distribution system, have sufficient independence to perform their safety functions assuming a single failure. General Design Criterion 21, "Protection System Reliability and Testability", requires, in part, that independence designed into protection systems be sufficient to assure that no single failure results in loss of the protective function.

b. Regulatory Guide 1.11 (Safety Guide 11) describes a suitable basis which may be used to implement General Design Criteria 55 and 56 for demonstrating the acceptability of instrument lines that penetrate or are connected to the primary reactor containment.

c. Regulatory Guide 1.75 describes an acceptable method of complying with IEEE 279-1971 and General Design Criteria 3, 17, and 21, with respect to the physical independence of the circuits and electrical equipment comprising or associated with the Class 1E power systems, the protection system, systems actuated or controlled by the protection system, and auxiliary or support systems that must be operable for the protection system and the systems it actuates to perform their safety-related functions.

3.1.2 IEEE Standards

- IEEE 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations"

  4.6 Channel Independence. Channels that provide signals for the same protective functions shall be independent and physically separated to accomplish decoupling of the effects of unsafe environment factors, electric transients, and physical accident consequences documented in the design basis, and to reduce the likelihood of interactions between channels during maintenance, operation, or in the event of channel malfunction.

- IEEE 308-1974, "IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations"

  5.4 Vital Instrumentation and Control Power Systems.

  5.4.1 General. Dependable power supplies are required for the vital instrumentation and control systems of the unit(s) including:

  - Instrumentation and control for the reactor protection system.

  - The engineered safety features instrumentation and control systems.

  5.4.2 Design Requirements. The diverse arrangements, special requirements, and complexity of these systems preclude a detailed delineation of their power supply requirements. However, power must be supplied to these systems in such a manner as to preserve their reliability, independence, and redundance. Typically, one or more of the following may be required:

  - Two or more independent direct current power supplies for control and instrumentation.

  - Two or more independent direct current power supplies for instrumentation that has special requirements regarding stability and freedom from noise.

  - Two or more independent alternating current power supplies having a degree of reliability and availability, compatible with the systems they serve.

  To accomplish the above requirements, special supplies may be required that are isolated from the alternating current and direct current power supplies used for the normal instrumentation and control of the unit(s).

- IEEE 384-1977, "Criteria for Independence Requirements of Class 1E Equipment and Circuits

IEEE 384-1977 gives the independence requirements of the circuits and equipment comprising or associated with Class 1E systems. Further, it gives the methods of achieving independence - "The physical separation of circuits and equipment shall be achieved by the use of safety class structures, distances, or barriers, or any combination thereof. Electrical isolation shall be achieved by the use of physical separation, isolation devices, or shielding and wiring techniques".

In the design and installation of instrumentation and controls (sensors, sensing lines, instrument racks, wires and cables, amplifiers, etc.), "channel independence" is achieved by locating the sensors and the points at which the sensing lines are connected to the process loop to provide physical separation of channels or divisions. This precludes a situation in which a single event could remove or negate a protective function. The NRC inspector should check that the routing of cables from a redundant division of sensors are arranged so that the cables are separated from each other and from power cabling to comply with Paragraph 4.6 of IEEE 279-1971. This includes separation at the containment penetration areas. In the control room, the redundant system trip channels are physically separated. Outputs from the control boards are routed in a separated cable system.

In the following sections, the design and installation practices are further detailed. As a refresher, the following definitions as per IEEE 384-1977 are reviewed:

Acceptable - Demonstrated to be adequate by the safety analysis of the station.

Associated Circuits - Non-Class 1E circuits that share power supplies, signal sources, enclosures, or raceways with Class 1E circuits or are not physically separated or electrically isolated from Class 1E circuits by acceptable separation distance, barriers, or isolation devices.

Barrier - A device or structure interposed between Class 1E equipment or circuits and a potential source of damage to limit damage to Class 1E systems to an acceptable level.

Class 1E - The safety classification of the electric equipment and systems that are essential to emergency reactor shutdown, containment and reactor heat removal, or are otherwise essential in preventing a significant release of radioactive material to the environment.

Flame Retardant - Capable of limiting the propagation of a fire beyond the area of influence of the energy source that initiated the fire.

Isolation Device - A device in a circuit which prevents malfunctions in one section of a circuit from causing unacceptable influences in other sections of the circuit or other circuits.

Redundant Equipment or System - An equipment or system that duplicates the essential function of another equipment or system to the extent that either may perform the required function regardless of the state of operation or failure of the other.

3.2 Implementation of Applicable Documents

3.2.1 Internal Separation of Redundant Class 1E Circuits and Instruments in Separate Cabinets or Compartments

Separation of redundant Class 1E equipment and circuits may be achieved by locating them in separate cabinets, physically separated by the use of safety class structures, distances, or barriers, or any combination thereof.

3.2.2 Internal Separation of Redundant Class 1E Circuits and Instruments in Separate Compartments of a Single Cabinet

Where redundant Class 1E circuits are located in separate compartments of a single cabinet to satisfy the requirements of IEEE 384-1974, circuit separation is achieved by providing minimum separating distances of six inches between Class 1E circuits and non-Class 1E circuits, and between redundant Class 1E circuits. The NRC inspector should check for these separations. The six inch minimum separation distance is based on all materials, devices, and wiring, internal to a control panel or instrument cabinet, connecting Class 1E equipment or circuits, being flame retardant. Where such materials, devices, or wiring are not flame retardant, an analysis based on tests to determine their flame retardant characteristics is performed to establish the minimum separation distance.

The inspector should be aware that where the minimum

distances as described previously are not practical, a physical barrier may be installed between circuits requiring separation. The barrier is designed such that the damage to the Class 1E circuits being protected is no worse than the damage that would occur with a physical separation of six (6) inches.

Non-Class 1E circuits that share power supplies, enclosures, or raceways with Class 1E circuits, or that are not physically separated from Class 1E circuits by an acceptable separation distance or by barriers, are called "associated circuits". Associated circuits shall comply with one of the following:

- They are uniquely identified as associated circuits and remain with, or are separated the same as, those Class 1E circuits with which they are associated.

- Associated circuits shall be in accordance with the above requirement from Class 1E equipment or wiring up to and including an isolation device. Beyond the isolation device, a circuit is not subject to the requirements of Class 1E circuits, if it does not again become associated with Class 1E circuits. Examples of isolation devices are optical isolation, isolation amplifier, or relays devices. In every case, the minimum separation distance of six (6) inches must be maintained or a barrier installed.

- They shall be analyzed or tested to demonstrate that Class 1E circuits are not degraded below an acceptable level.

3.2.3 Separation Requirements for Associated Circuits

The USNRC Regulatory Guide 1.75, "Physical Independence of Electric Systems", requires the separation of Class 1E associated circuits from non-Class 1E circuits. An associated circuit is one which serves a non-essential function, but which is not sufficiently separated from essential circuits. Sufficient separation is obtained by six inches of air space or by an adequate fire barrier.

Examples of associated circuits include wiring from safety system equipment to annunciators, computer inputs, indicators and recorders, and interlocks to non-essential systems or functions. Associated circuits may be handled by one of two methods:

- Once a non-Class 1E circuit is identified to be an associated circuit, it shall be treated exactly as a

Class 1E circuit. An alternate method of treating associated circuits is to analyze or test them to demonstrate that although they are not physically separated or electrically isolated, as described above, they will not degrade Class 1E circuits below an acceptable level.

- The associated circuit may be connected to a qualified circuit isolation device. The output of the isolation device may be treated as a non-Class 1E circuit.

Several practical methods to implement the above-listed requirements are:

- Eliminate as far as practicable, all associated wiring. This should be done by separating all Class 1E wiring and devices from all non-Class 1E wiring and devices.

- Where associated circuits cannot be eliminated, they should be treated as Class 1E circuits or be isolated from Class 1E circuits by a qualified isolation device, and then "declassified" as non-Class 1E circuits and treated accordingly.

- If the plant computer and/or annunciator is not a Class 1E hardware, then a qualified isolation module is usually provided to isolate all inputs that originate in Class 1E circuits. This situation is shown in Figure VI-6-1.

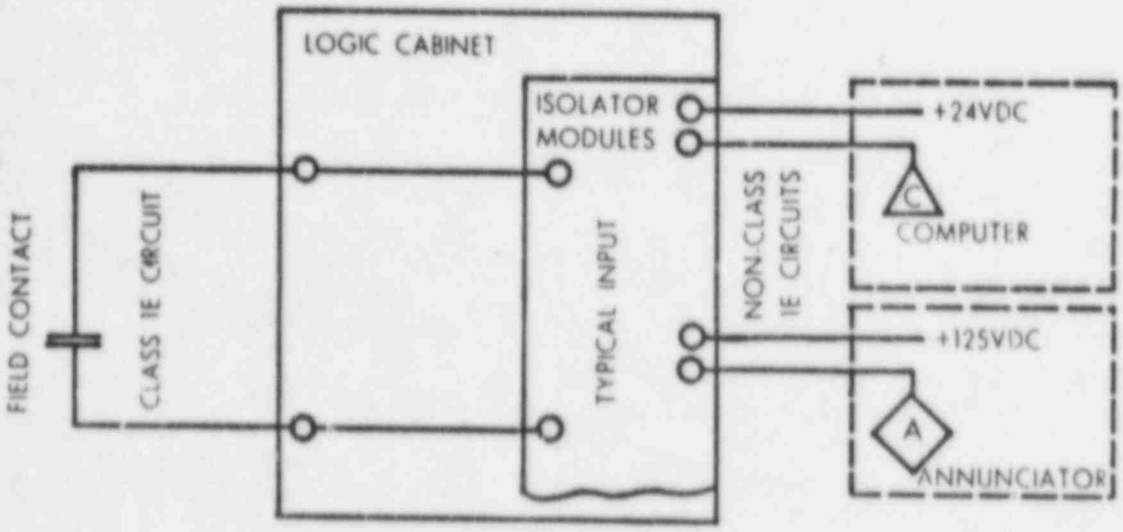
  Where the above methods cannot be employed, the circuits must be analyzed to show that circuit failures in the associated circuits will not propogate to Class 1E electrical circuits. These analyses are required for the FSAR and are documented and filed with verification data.

3.3 Isolation Devices

The following devices are used to separate channels in nuclear power plants:

a. Fuses - Fuses will isolate a circuit upon the application of a current that exceeds the current rating of the fuse. They can be used in digital or analog circuits as well as power circuits. The use of fuses is not permitted by RG 1.75, but it is felt that they can be used in certain applications; the NRC inspector, therefore, should refer to the project SAR for specific commitments.

Figure VI-1
Example - Digital Signal Isolator Module



b. High Impedance Networks - These circuits can be used to limit maximum current and damage potential in analog circuits. (See Figure VI-2.)

c. Optical Isolation - Optical isolators are devices using light-pipes to transmit the digital (on-off) signal and may be used in digital circuits. Some designs may utilize a fire retardant barrier between the input and output.

d. Isolation Amplifiers - These are very useful devices in isolating analog circuits if they can be located so that input and output wiring can be separated by six inches or a fire barrier, or an analysis can be performed as stated below. (See Figure VI-2.)

e. Relays - Relays are adequate isolators, but only where fire barriers or separation can be achieved between the circuits. (See Figure VI-3.)

3.4 Design and Construction Techniques Employed to Achieve Separation

3.4.1 Separation

Redundant sensory equipment is identified and segregated so that no single credible event is capable of disabling sufficient equipment to prevent a safety system from performing its function. Separation requirements also apply to control power and motive power.

Typical protection systems have four independent input channels for each monitored variable. Separation between redundant channels includes the taps, the sensor, its input, the wiring between the sensors and the actuation logic, separation within the protection system cabinets, wiring between the protection system cabinets, and control and safety feature equipment. Logic matrix wiring and other interconnecting wiring between the four (4) cabinets may be made by running wires through rigid metallic conduits penetrating the mechanical barrier between each cabinet bay. Two adjacent panels containing circuits of different divisions shall be physically separated. All barrier penetrations are sealed with a fireproof material.

Penetration of separation barriers within a subdivided panel is permitted, provided that such penetrations are sealed or otherwise treated so that an electrical fire could not propagate from one section to the other and disable a protective function.

Separation within the plant protection system (PPS) cabinets is provided by barriers. These barriers run the

full depth and full vertical dimensions of the cabinets.

All cabling entering each section of the panel is separated as it enters the cabinet.

Figure VI-2
Low Level Analog Signal Circuit Isolation

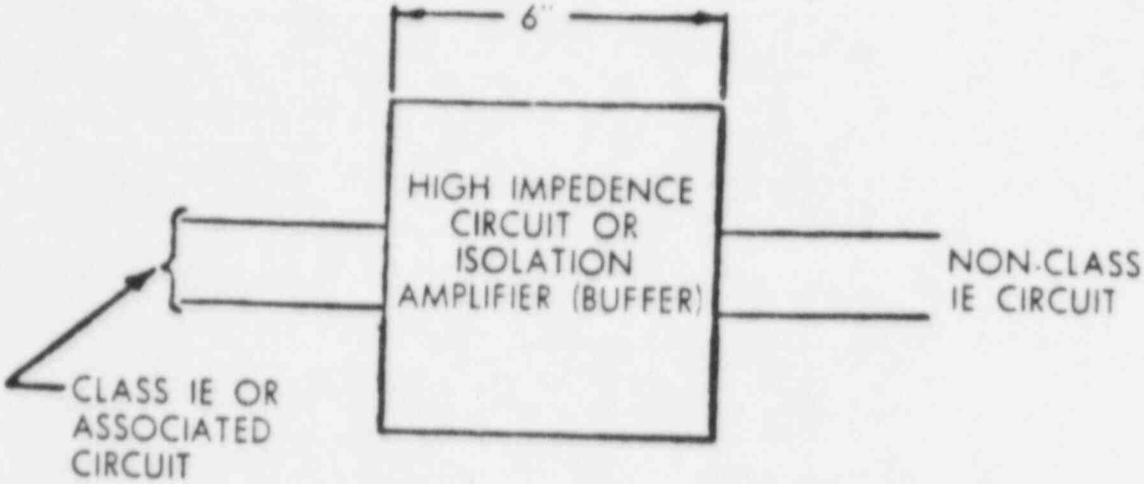
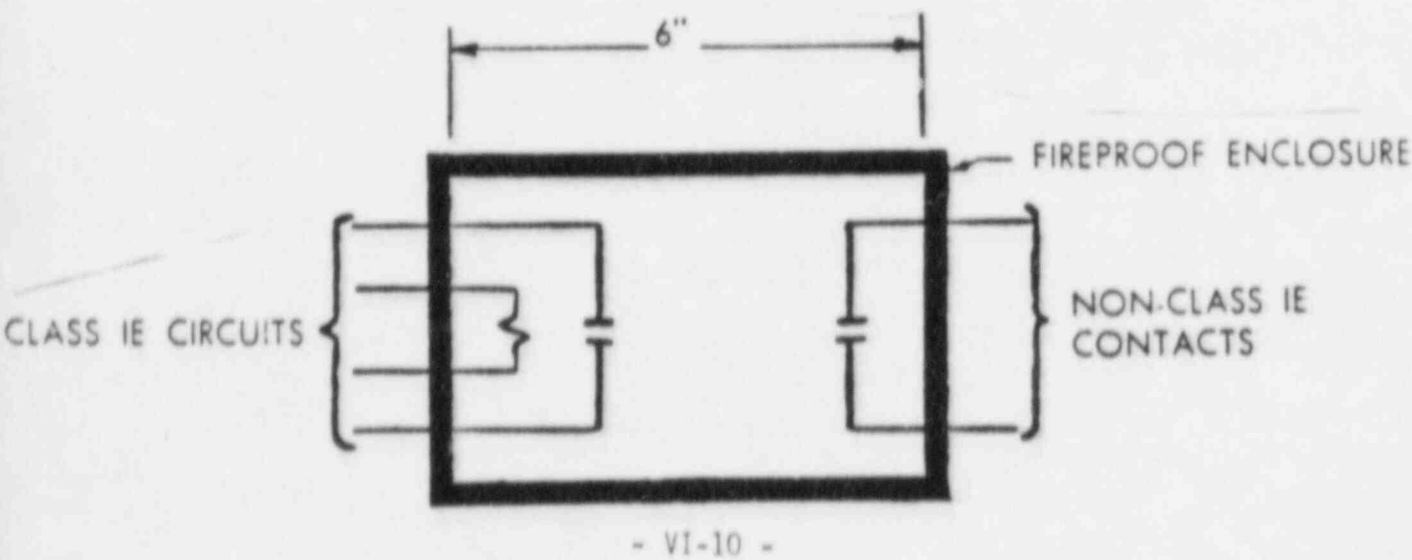


Figure VI-3
Relay Isolation



As shown in Figure VI-4, all four PPS cabinet bays are interconnected through internal cable wireways. These metal wireways are totally enclosed and surrounded by an insulated fireproof material. In addition, the wireways are packed with a fireproof material to preclude the transmission of fire within the interconnecting wireway ducts.

The reactor trip switchgear, which interrupts power to the coils of the control element drive mechanisms, is a single cabinet that houses the trip breakers. It is also separated into four sections to assure separation of protection signals.

The ESFAS is provided with two independent and separate relay cabinets that maintain separation between the two actuation trains.

Safety-related system analog signals that go to both the plant computer and the control element assembly (CEA) position display CRT are isolated from the non-safety-related systems by means of active electronic isolation devices. Signals from relays in safety-related systems that actuate the plant annunciator, plant computer, and control element drive mechanism control system are isolated from the originating system by the inherent coil-to-contact isolation of the relays. Redundant channels of Class 1E indicators are separated in accordance with IEEE 279-1971. All of these techniques assure that no credible failures will affect the function of safety-related systems, and that the independence of safety-related systems is not jeopardized.

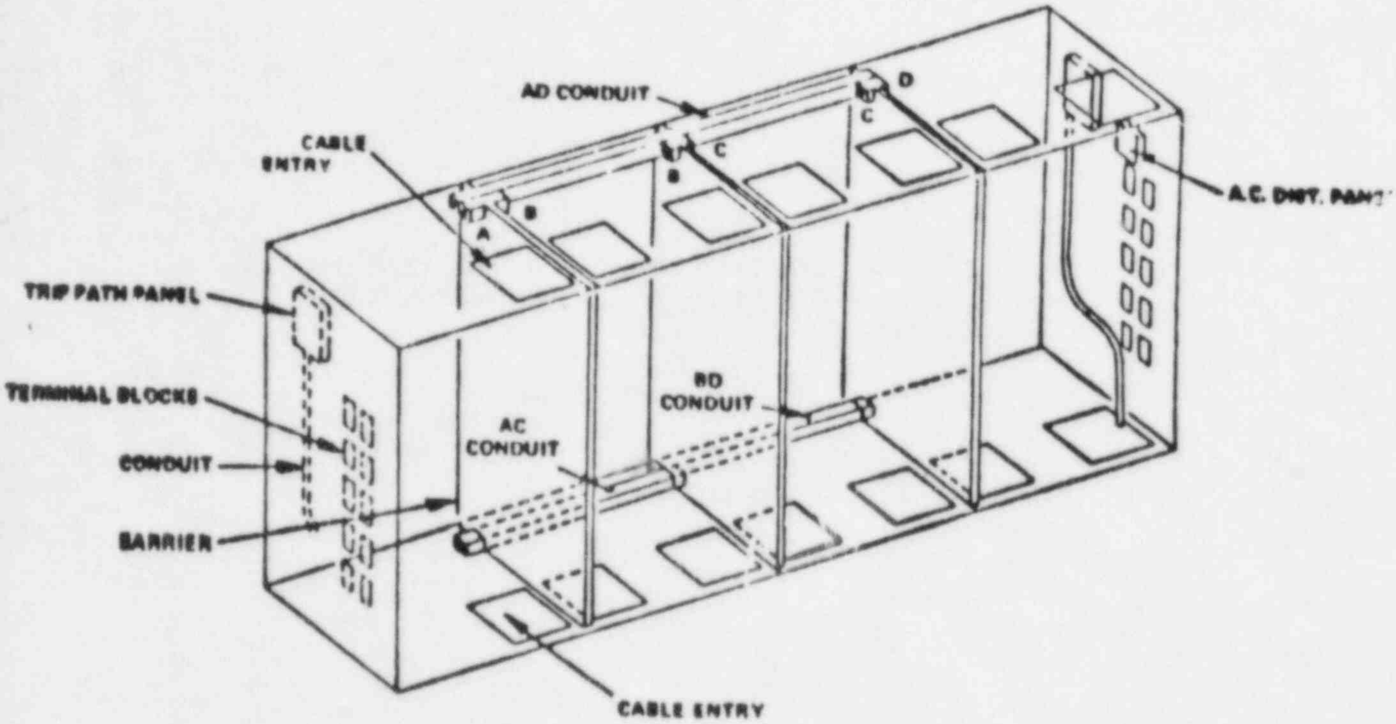
The degree of separation required varies with the potential hazards in a particular area of the station. These areas may be classified as follows:

General - Nonhazardous Areas - These areas which do not contain high pressure piping, flammable liquids, or gases, and are not subject to flooding or damage from missiles.

Mechanical Damage - Missile Areas - These are areas containing an operating crane or wherever there is a potential for locally generated or external missiles.

Fire Hazard Areas - These are areas where there is the potential for accumulation of large quantities (more than one gallon) of oil, other combustible fluids, gases, a heavy concentration of cables, oil filled transformers, or other combustible materials.

Figure VI-4
Simplified PPS Cabinet Layout (Rear View)



Flood Hazard Areas- These are areas where there is potential for accumulation of flood water from either outside the station or from ruptured pipes, pumps, or storage tanks.

High Pressure Piping Areas - These are areas containing high pressure piping which has potential for cracking or rupture and would subject adjacent systems and equipment to jet impingement, pipe whip forces and/or high temperature fluids or a steam atmosphere.

Special Areas - These are areas for specific functions such as containment penetration access, cable vault or cable spreading room, switchgear rooms, diesel-generator rooms, and battery rooms.

Instrumentation and instrumentation tubing channels which provide redundant safety-related signals for the same instrumentation protective function shall be physically separated to comply with IEEE 279-1971. Where physical separation is not practical, suitable fire retardant and missile barriers should be used. Mechanical instrumentation channels consist of the process couplings that detect the process variables, the sensors that detect the variable conditions and produce a signal, and the sensing line connecting them. For the redundant safety-related instrumentation and instrumentation tubing, the following general rules are followed:

Safety-related instrumentation belonging to one division may be grouped together.

The preferred method of separation for redundant safety-related instrumentation tubing is to route it through different rooms of a building or on opposite sides of a room.

3.4.2 Physical Identification

Regulatory Guide 1.75 requires that all safety-related electrical components be readily identifiable in the field. The preferred method is color coding. To avoid compromising the independence of these redundant systems, a usual system of color coding is employed on all components. This may vary due to the NSSS vendors standards. The following is an example:

Measurement Channel A - Red	Post Accident
Measurement Channel B - White	Monitoring Channel 1-Red/White

Measurement	C - Blue	Post Accident
Measurement	D - Yellow	Monitoring Channel 2-Yellow/White
ESF	A - Orange	
Associated	A - Orange/Brown	
ESF	B - Green	
Associated	B - Green/Brown	
Non-nuclear	Safety - Black	

Equipment may be marked using colored nameplates.

Exposed electrical raceway systems are marked in a permanent manner at least every fifteen feet and at points of entry and exit from enclosed areas.

Cables installed in open raceways are marked at a sufficient number of points to facilitate initial verification that the installation is in conformance with the separation criteria.

Class 1E cables are identified by a permanent color coded marker at each end.

Safety-related sensing lines are identified by the channel color coding.

Safety-related instrument mounts and racks are identified by the appropriate channel color code.

All sensors are identified with a metal tag containing sensor number as shown on the engineer's drawings.

3.4.3 Nonhazardous Areas

3.4.3.1 Process Coupling

Where more than one process coupling is required on the same piece of safety-related equipment, they shall be separated as much as practical and shall also assure that:

- The value being measured by both couplings is the same, and

- A single event does not damage a redundant process coupling

The same process coupling may be used for safety-related instrumentation assigned to the same channel.

3.4.3.2 Sensing Lines

Sensing lines for a corresponding safety-related process coupling may run together or use a common sensing line to both process couplings.

At the present, there is no standard requiring the minimum separating distance between redundant safety-related sensing lines. As a logical extension of the minimum separation distance requirements between redundant Class 1E electrical hardware, a typical A/E would use separations for sensing lines and other components of mechanical instrumentation or channels as required for Class 1E redundant electrical circuits according to IEEE 384-1977.

A minimum air separation between redundant safety-related sensing lines within instrumentation cabinets may be 6 inches.

Minimum air separation between a safety-related sensing line and a sensor or open cable tray from another redundant safety-related channel shall be 3 feet measured horizontally and 5 feet measured vertically.

The horizontal and vertical distance between sensors or racks is measured between the closest points.

3.4.3.3 Cable Routing

Cables of redundant systems shall be routed in separate tray or conduit systems and separate duct bank systems. Separation shall be in accordance with Regulatory Guide 1.75.

On multi-unit stations, cables associated with different units shall be run in separate cable trays, duct banks, manholes and conduits. The number of cable trays, duct banks, and conduits to be routed through a unit, other than the one with which they are associated, shall be kept to a minimum. The separation criteria described above shall be utilized to minimize the possibility of a single credible event from causing the damage of cables or buses of more than one unit.

Separate tray and conduit systems shall be provided for all classes of cables.

Separate tray and conduit systems shall be provided for non-Class 1E and Class 1E systems. Separation of these two systems shall be in accordance with Regulatory Guide 1.75.

The engineered safety features actuation signals are arranged with two output trains. The same separation must be maintained as is described above for channel separation.

Nuclear instrumentation system cables have four (4) channel redundancy. They are especially sensitive to outside noise pickup and have special shielding and routing requirements. These shall be run in separate conduits or low level cable trays.

Associated circuits (as defined by Regulatory Guide 1.75) are to be avoided wherever practical.

It is recognized that it may be necessary to make some cross-connections between mutually redundant channels. In these cases, the physical provisions for making these connections shall be designed to prevent the propagation of fire or any fault between channels.

3.4.4 Hazardous Areas

3.4.4.1 Fire Hazard Areas

Fire hazard areas are not usually of concern for mechanical equipment except where lines or tanks contain flammable liquids or gases. In this case, only one system of redundant components shall be located in the area, and fire rated walls or partitions shall be installed.

The placement of redundant instrumentation racks in the same area is not acceptable. HVAC systems shall be designed such that they do not cause a fire to migrate to another area containing redundant equipment. This can be achieved with fire rated enclosures and dampers.

The routing of cables for redundant safety-related systems through a fire hazard area shall be avoided whenever possible.

Where such routing is unavoidable, only one system of redundant cables shall be allowed in the area and these cables shall be protected by conduits or solid trays with covers.

Safety-related switchgear, power centers, and motor control centers shall not be located in fire hazard areas.

3.4.4.2 Flood Hazard Areas

Protection of redundant portions of safety-related systems against common failure due to compartmental flooding can be achieved by either of two ways. The redundant counterparts can be located in separate compartments which cannot experience common flooding, or where the redundant counterparts are in the same compartment, they can be located or supported in such a way as to preclude flooding damage.

3.4.5 Special Areas

3.4.5.1 Penetration Areas

Sensing lines for redundant safety-related measurement channels shall not be run through the same penetration.

The minimum separation for penetrations containing redundant safety-related sensing lines shall be equivalent to that for raceways in general areas.

For electric penetrations, in addition to the requirements of paragraph 5.5 of Regulatory Guide 1.75, redundant cables penetrating the containment building shall be routed through separate penetration areas. Each of these penetration areas shall be separated from each other.

3.4.5.2 Cable Vault

Redundant cable vaults are required in accordance with Regulatory Guide 1.75.

3.4.5.3 Diesel Generator Units

Redundant units are placed in separate enclosures which are physically separated. If this is not practicable, the separating shall be designed as

a four (4) hour NFPA fire barrier and shall provide missile protection in the event of explosion or failure of the rotating equipment. Each unit shall have an independent air supply and exhaust.

3.4.5.4 Battery Rooms

Redundant batteries are placed in separate fire rated enclosures, each having an independent air conditioning and/or ventilation system to prevent hydrogen accumulation.

3.4.5.5 Switchgear Rooms

Redundant emergency medium voltage switchgear and 480 volt power centers shall be located in separate fire rated rooms. In areas where it is not practical for separate rooms, redundant low energy electrical equipment, such as motor control centers, volt power panels, lighting panels, etc., are separated by a distance greater than the height of the adjacent cabinets.

Physical separation and/or barriers are provided between the normal medium voltage switchgear and the emergency medium voltage switchgear such that in the event of fire, explosion, or loss of structural integrity of the normal switchgear, the emergency switchgear function is not jeopardized.

3.4.5.6 Main Control Room

The main control boards are located in a control room within a safety class structure. The control room is protected from (and should not contain) high-energy equipment such as switchgear, transformers, rotating equipment, high pressure or temperature process lines, or any other potential sources of missiles or pipe whip. Internal separation of redundant safety-related circuits and instruments is discussed in Section 3.2.

4.0 SUMMARY

Physical isolation requirements between redundant safety systems have been presented in this unit. Applicable documents that address instrumentation and control isolation requirements, and their methods of implementation have been discussed. It is expected that the NRC inspector has acquired sufficient knowledge from this unit to enable him to become familiar with the requirements, design philosophy, and construction techniques related to the physical isolation between redundant safety systems.

INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

UNIT VII - Single Failure Criteria

PRE-STUDY TEXT

INSTRUMENTATION AND CONTROL TRAINING COURSE

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INSTRUMENTATION AND CONTROL TRAINING COURSE

FOR

NUCLEAR REGULATORY COMMISSION INSPECTORS

UNIT VII - Single Failure Criteria

PRE-STUDY TEXT

1.0 OBJECTIVE

The objective of the unit is to present to the NRC inspectors the evolution of the single failure criterion and what his role is in its use.

2.0 INTRODUCTION

The general design criteria of 10CFR50, Appendix A emphasizes the need to design protection systems and safety-related systems against single failure. Single failure is defined in 10CFR50, Appendix A as:

> "A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electrical systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly), or (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions."

"Active" and "passive" failure terms are used in the mechanical and fluid process portion of a safety system including instrumentation sensing lines. These terms are defined as follows:

a. An active failure is a malfunction of a component which relies on mechanical movement to complete its intended function upon demand. Examples of active failures include the failure of a powered valve or a check valve to move to its correct position, or the failure of a pump, fan, or diesel generator to start.

b. A passive failure is a breach of a fluid pressure boundary or blockage of a process flow path.

This unit discusses the documents related to single failure criteria and their application of the single failure criteria. These documents include general design criteria from 10CFR50 - Appendix A, IEEE 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," and IEEE 379, "Application of the Single Failure Criterion to Nuclear Power Generating Station Class 1E Systems."

3.0 SINGLE FAILURE CRITERIA

The General Design Criteria 21, 34, 35, 38, 41, and 44 require that the plant protection, residual heat removal, emergency core cooling, containment heat removal, containment atmosphere cleanup and cooling water systems be designed to meet the single failure criteria.

3.1 Documents Related to Single Failure Criteria

3.1.1 10CFR50 Appendix A General Design Criteria

The single failure criterion for safety-related systems is contained in the following general design criteria:

GDC-21 Plant Protection

GDC-34 Residual Heat Removal

GDC-35 Emergency Core Cooling

GDC-38 Containment Heat Removal

GDC-41 Containment Atmospheric Cleanup

GDC-44 Cooling Water

The following excerpt from Criterion 35 is typical of the requirements for each safety-related system:

"Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for on-site electric power system operation (assuming off-site power is not available) and for off-site electric power system operation (assuming on-site power is not available) the system safety function can be accomplished, assuming a single failure."

The following excerpt from GDC-21 is typical for protection system requirements:

"The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that:

- No single failure results in loss of the protection function; and

- Removal from service of any component or channel does not result in loss of the required minimum redundancy

unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

3.1.2 IEEE 279 Criteria

The IEEE developed Standard IEEE 279, "Criteria for Protection Systems for Nuclear Power Generating Stations", to assist in the design of the protection system. Section 4.2 of IEEE 279 defines single failure as:

"Single failure: Any single failure within the protection system shall not prevent proper protective action at the system level when required.

NOTE: "Single failure" includes such events as the shorting or open-circuiting of interconnecting signal or power cables. It also includes single credible malfunctions or events that cause a number of consequential component, module, or channel failures. For example, the overheating of an amplifier module is a "single failure" even though several transistor failures result. Mechanical damage to a mode switch would be a "single failure" although several channels might become involved."

The definition of single failure in IEEE 279 places emphasis on action being performed at the system level.

Single failure should not be confused with the capability for completing a system protective function. A single failure results in the inability of a component, module, or channel to carry out its part in the implementation of a protective function; however, unless it violates the single failure criteria, it does not, of itself, prevent its redundant counterpart from implementing the protective function.

3.1.3 IEEE 379 Criteria

As single failure was applied using the GDC and IEEE 279 (1971), a number of questions came up. To aid in the implementation of single failure, IEEE 379, "Application of the Single Failure Criterion to Nuclear Power Generating Station Class 1E Systems", was issued. It includes the following amplification on single failure:

3 1.3.1 Multiple Faults - Whenever analysis shows that multiple failures in a system will result from

a single event, these multiple failures, collectively, will be considered to be one single failure.

3 1.3.2 Design Basis Events and Single Failures - A design basis event that results in the need for protective action may cause failure of protection system components, modules, or channels. When analysis indicates that such would be the case, the protective system failures are considered to be a starting point (that is, they are a condition of the event). For the protection system to meet the single failure criterion, it must be shown that, in addition to these event-caused failures, it can tolerate one single failure without defeating the protection system's ability to implement any of its assigned protective functions.

3.1.3.3 Single Failures and Operational Reliability - The single failure criterion is strictly applicable only to situations involving the capability for system implementation of protective functions. These same principles, however, are appropriate to an assessment of system operational reliability (for example, freedom from spurious trips).

3.1.3.4 Classification of Single Failures - To clarify the single failure concept and to provide a basis for common understanding among the users of this guide, single failures have been grouped into three types. Single failures within the meaning of the single failure criterion, regardless of whether or not they violate the single failure criterion, are classified as follows:

Type 1: A detectable failure resulting from one failed component or module or from a circuit fault.

NOTE: Examples of circuit faults include short circuits, open circuits, grounds, and the application of the maximum credible ac or dc potential.

Type 2: Multiple detectable failures resulting from a single cause external to the protection system.

Type 3: Multiple detectable failures resulting from a single cause within the protection system.

Detectable failures are those that can be identified through periodic testing or are revealed by alarm, anomalous indication, etc.

3.1.3.5 Undetectable Failures - Failures that would be undetectable are significant to the single failure analysis. The detectability of failures is fundamental to the above types of single failures and to the single failure criterion itself. Undetectable failures differ from detectable failures in that they may exist in a protection system for years. In fact, they could be built into the system by the manufacturer or wired into the system by the installer.

In the single failure analysis, all potential undetectable failures should be identified. This requires that the means for detecting failures be analyzed with the protection system. When undetectable failures are identified, the following courses of action are available.

- The preferred course is to redesign the protection system or the test scheme to eliminate potential undetectable failures, or

- In the analysis of the effect of each single failure, all potential undetectable failures must be assumed to be in their failed mode.

3.1.3.6 Common Mode Failures - The multiple failures of Types 2 and 3 are examples of common mode failures. Consideration of common mode failure in this guide is limited to those common mode failures that result from events that are identified in the design basis (Section 3 of IEEE 279-1971). Other common mode failures resulting from manufacturing and maintenance errors, certain unanticipated design shortcomings, and factors not considered in the design basis, should be considered but are not discussed in IEEE 379. The potential for common mode failures is more difficult to identify than for random failures. Because of the difficulty of assuring that all credible causes of common mode failure have been identified in the design basis, other defensive provisions beyond the scope of IEEE 379 may be appropriate. These may include the use of alternative channels that either sense a set of plant variables different from the primary channels, or use equipment different from the primary channels, or utilize a combination of these methods.

3.2 Application of Single Failure Criterion in Safety Systems

3.2.1 Explanation of Single Failure Criteria

The safety system consists of a protection system that senses and develops the signal necessary to energize the appropriate safety-related systems and the protective action system that completes the protective function, such as cooling the reactor core.

The safety system is usually made up of four protection channels (protection system) and two safety-related trains of the safety-related system as shown on Figure VII-1.

We will go through the process of meeting single failure criteria for a protective function. We will start with the assumption that the equipment is qualified to withstand the natural phenomena and normal and accident environmental conditions. To meet basic single failure, we would provide two protection system channels and two protective action system trains.

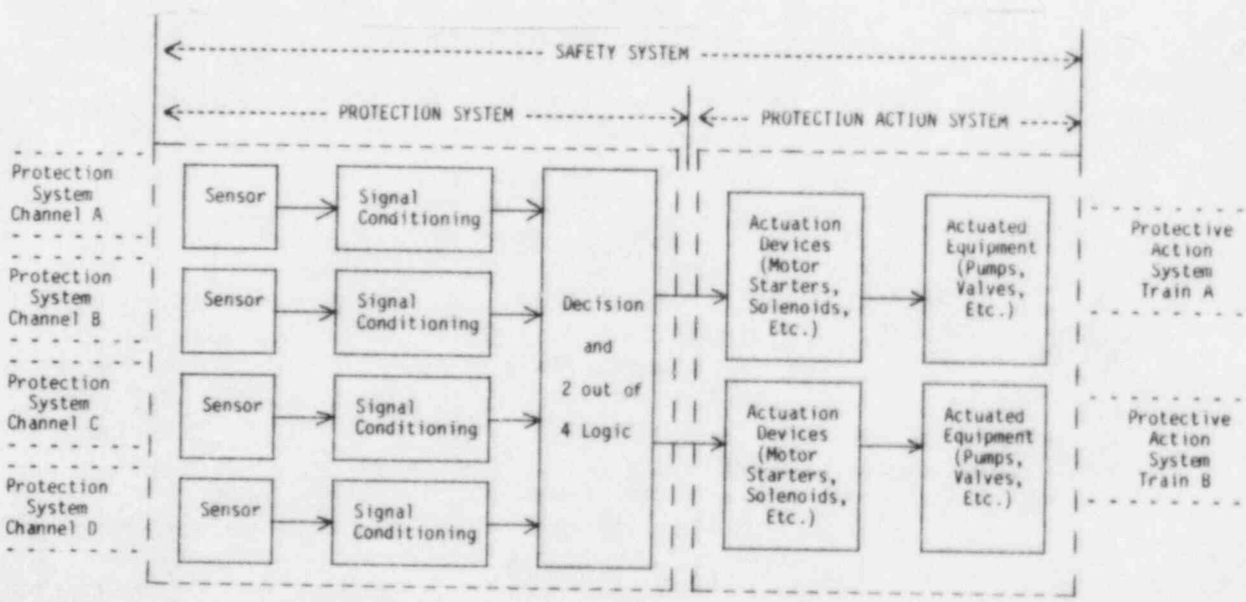

Figure VII-1

The arrangement shown on Figure VII-2 does not meet the single failure criteria requirement. To meet this we will add a third protection channel with a two out of three logic as shown on Figure VII-3.

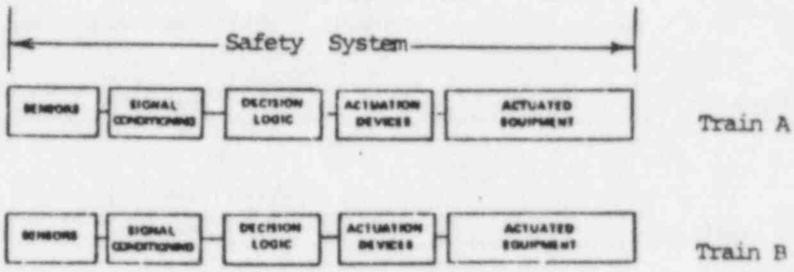

Figure VII-2

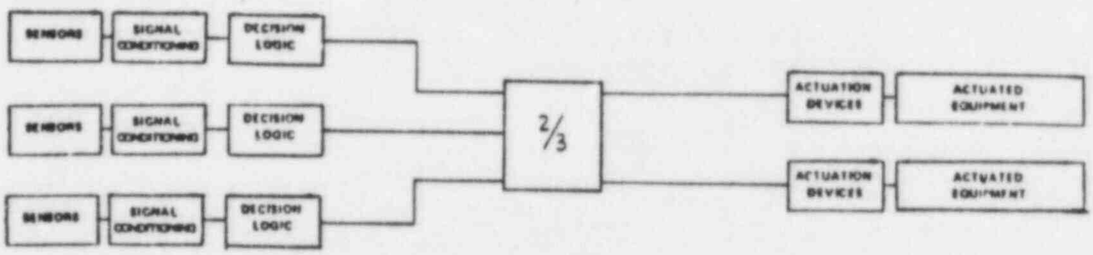

Figure VII-3

This will permit testing, but does not meet single failure because a failure in the 2 out of 3 logic will wipe out the safety system. However, according to Section 4.10 of IEEE 279 (1971), the protection system has to remain testable on-line and meet the single failure criterion while one protection channel is taken out of service. Redundant logic as shown below on Figure VII-4 will solve this problem. This arrangement requires attention to isolation of the logics:

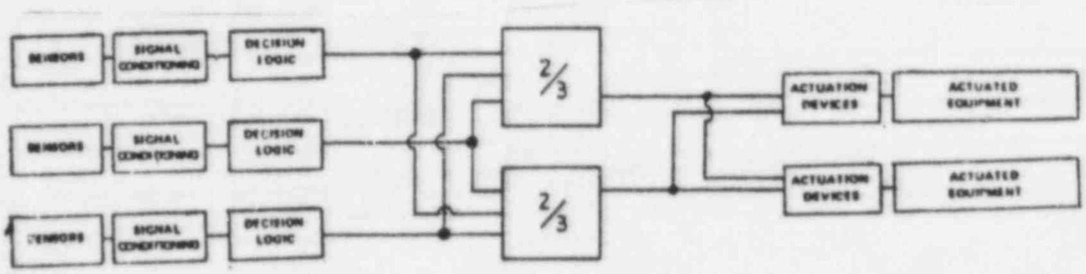

Figure VII-4

The addition of one redundant 2 out of 4 logic (2/4) will allow either 2/4 matrix to energize both safety-related systems trains.

Two out of four has become the generally accepted logic for the protection system as shown on Figure VII-5.

Now the 2/4 logic can be checked without affecting the safety-related system.

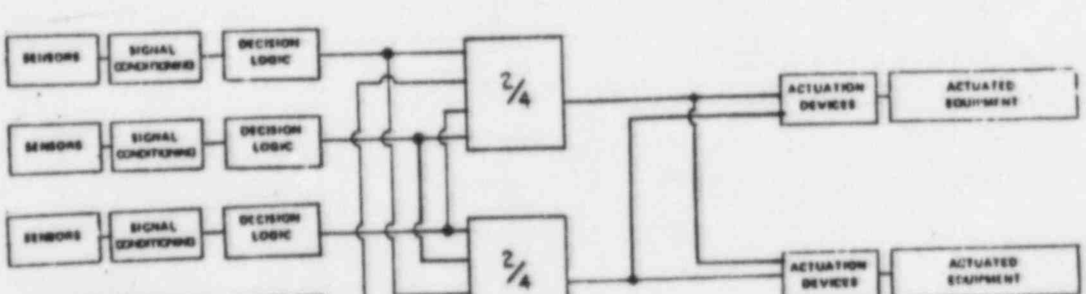

Figure VII-5

3.2.2 Discussion Related to the Compliance with the Single Failure Criteria in the Design of Safety System

A single protection system channel could meet single failure if the quality of the system was so high that it would never fail even during the design basis environmental conditions, fire or missile hazard.

The cost of approaching the above is so high that single failure is usually met by providing redundant channels within the protection system.

3.2.2.1 Single failure of the protection system for conditions of natural phenomena, fire and environmental or missile hazard is met as follows:

Protection against environmental conditions is met by specifying equipment that will function during and after the specified environmental conditions.

Protection against fire and missile hazards is met by redundancy sufficient to cover on-line testing.

3.2.2.2 Single failure of the protective action system for conditions of natural phenomena, fire and environmental or missile hazard is met as described in paragraph 3.2.2.1 above. Although the general design criteria requires periodic testing, it does not require meeting single failure criteria while one train of the protective action system is being tested.

3.2.2.3 The inspector role in the implementation of single failure criterion on safety systems is as follows:

a. Verify that the specified equipment is used.

b. Verify that the installation is in accordance with the design documents as shown on FSAR (Final Safety Analysis Report).

c. Review the location of all equipment with respect to sources of fire and missiles. Fire or missile barriers should be installed in identified areas.

4.0 SUMMARY

After the development of a protection system and safety-related system that meet single failure criteria, it can be lost if the equipment is not adequately separated in the plant and if the interconnections are not proper and separated. The NRC inspector's job is to assure that single failure criterion is not violated in the plant installation.

This text and the classroom instruction offered with it are designed to acquaint students with accepted good practice for the operation and maintenance of equipment and/or systems.

They do not purport to be complete nor are they intended to be specific for the products of any manufacturer, including those of the General Electric Company; and the Company will not accept any liability whatsoever for work undertaken on the basis of the text or classroom instruction. The manufacturer's operating and maintenance specifications are the only reliable guide in any specific instance; and where they are not complete, the manufacturer should be consulted.

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