CHAPTER 12: RADIATION PROTECTION

TABLE OF CONTENTS

CHAPTER 12: RADIATION PROTECTION

TABLE OF CONTENTS

CHAPTER 12: RADIATION PROTECTION

LIST OF TABLES

CHAPTER 12: RADIATION PROTECTION

LIST OF FIGURES

Section

Chapter 12
RADIATION PROTECTION

12.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS IS REASONABLY ACHIEVABLE (ALARA)

12.1.1 POLICY CONSIDERATION

NextEra Energy Duane Arnold, LLC is committed to implementing a Radiation Protection program that supports maintaining occupational radiation exposures at the Duane Arnold Energy Center (DAEC) to as-low-as-reasonably-achievable levels within the guidelines to 10 CFR 20.

A radiation protection program is in effect at the DAEC to keep radiation exposures as low as reasonably achievable. To implement this program, the following policy has been promulgated:

1. Each individual working within radiation controlled areas shall receive radiation safety training so as to understand the reasons for radiation safety and the methods of maintaining exposures as low as reasonably achievable.

2. The DAEC shall be operated and maintained in a manner to reduce to a level as low as reasonably achievable both occupational radiation exposures and the spread of radioactive contamination. Activities involving radiation exposure shall be planned to prevent unnecessary exposure to radiation or spread of radioactive contamination.

3. Modifications or changes to the DAEC shall be designed and constructed giving consideration to maintaining exposures and contamination as low as reasonably achievable during installation as well as during operation and maintenance throughout the life of the plant.

Existing resources are expected to be utilized in a manner which enhances the objective of maintaining exposures as low as reasonably achievable. All levels of management are required to recognize the need for and to initiate actions to support this objective.

12.1.2 DESIGN CONSIDERATIONS

Required at PSAR stage only. See Section 12.3 for radiation protection design features.

12.1.3 OPERATIONAL CONSIDERATIONS

12.1.3.1 Leakage Reduction Program

The DAEC has implemented a program in response to NUREG-0578, Section 2.1.6.a, to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious accident to as-low-as-reasonably-achievable levels.

12.1.3.1.1 Immediate Leakage Reduction Program

The immediate leakage reduction program is designed to provide an assessment of the leakage rates for the systems outside the containment at the DAEC that could contain highly radioactive fluids during a serious transient. This includes a walkdown for visual inspection and identification of leakage sources for these systems, and where possible, immediate correction of these leakages.

Program Description

The immediate leakage reduction program is being accomplished as follows:

1. Systems subject to postaccident leakage have been selected and are listed below.

2. Leakage test procedures have been developed.

3. Testing hardware and personnel have been provided.

4. Testing is being conducted and leakage test data are being collected.

5. Collected leakage test data are being evaluated and reviewed.

6. Corrective actions required for leakage reduction are being taken.

7. Corrective actions and retests, as appropriate, are being implemented.

8. The preventative maintenance program for the plant to reduce the leakage to as-low-as-reasonably-achievable levels is being implemented.

System Selection

Systems outside the primary containment that could contain reactor water, reactor steam, containment atmosphere, or suppression pool water after a postulated accident resulting in gross fuel failure have been selected for this program.

The following systems have been selected:

1. Residual heat removal (RHR) system.

2. Core spray system.

3. Reactor core isolation cooling (RCIC) system.

4. High-pressure coolant injection (HPIC) system.

5. Reactor water cleanup (RWCU) system (only to second isolation valve).

6. Control rod drive (CRD) system (scram discharge headers only).

7. Liquid radwaste system (up to radwaste and floor drain collection tank).

8. Containment atmosphere monitoring system.

9. Postaccident sampling system

<u>Basis for the Selection of Systems</u>

The RHR system may be used in all its operating modes following a postulated major accident. It will carry reactor coolant and suppression pool water. The Core Spray system, carrying suppression pool water, is also likely to be required to operate.

Although it is likely that the reactor will be rapidly depressurized following a serious accident, it is possible that the RCIC and HPCI systems will operate early in the accident.

The RWCU system will be automatically isolated as soon as the accident is detected. This system has been included only to the second isolation valves. Radiation levels dictate that this system be inspected during an outage in order to minimize personnel exposure to as-low-as-reasonably-achievable levels.

The scram discharge headers will not contain postaccident reactor water following the postulated accident. However, should the reactor remain pressurized for an extended period, the scram discharge headers will collect reactor water leaking into the CRD system.

The liquid radwaste system is not expected to process waste. It is likely, however, that some time during the accident recovery it will be necessary to pump highly contaminated fluids to the radwaste and floor drain collector tanks in the HPIC and RCIC rooms. The liquid radwaste system has been included only to these tanks.

The containment atmosphere monitoring system will circulate postaccident containment atmosphere at about the containment pressure through a 1-in. piping system while monitoring the hydrogen and oxygen content of the drywell and the suppression pool. Therefore, this system has been included.

The postaccident reactor coolant sampling system will be used to obtain samples of the potentially highly radioactive coolant following an accident. On installation of the sampling system with this capability, it will be included in the leak reduction program.

The postaccident containment atmosphere sampling system will be used to obtain samples of the containment atmosphere following an accident and is included in the program.

Systems Excluded from the Leakage Reduction Program

Systems that are isolated at the containment following postulated accidents are excluded from the leakage reduction program. Containment isolation valves are tested in accordance with the Primary Containment Leakage Rate Testing Program.

It is unlikely that the primary containment will be purged via the standby gas treatment system following a postulated accident. Therefore, the standby gas treatment system is excluded since it is located inside the reactor building and contains reactor building atmosphere.

The gaseous radwaste system (offgas) is excluded since the steam jet air ejectors, steam packing exhausters, and mechanical vacuum pumps will not operate after the accident.

The RWCU system beyond the second isolation valve will be isolated by the accident. It is unlikely that the filter-deminieralizer units will be useful for the accident recovery.

12.1.3.1.2 Leakage Measurement Program

The intent of the leakage measurement program is to provide the assurance of integrity for systems outside the primary containment that could contain highly radioactive fluids after an accident.

Purpose

The purpose of the leakage measurement program is to obtain a quantitative measure of leakage within selected systems by identifying and measuring the leakage from each component or potential source of leakage in the system. After leaking components have been repaired and reinspected, total system leakage can be determined. Total leakage from all system will allow the estimation of airborne radioactivity.

Leakage measurements from individual components will provide an input to the preventative maintenance program.

Scope

The leakage measurement program includes the systems and portions of systems listed above the "System Selection." The inspection and subsequent actions are concentrated on areas where leakage is likely to occur. This includes valve stems, vents and drains, pump seals, pump seal leakoffs, pump case joints, valve body-to-bonnet joints, flanged pipe joints, body drain plugs, and relief valve discharges.

Only visually detectable leakage is reported; insulation will not be removed for this inspection.

Test Methods

The systems containing water and steam are visually inspected for leakage while the system operating conditions are duplicated.

The preferred method of testing systems containing water is in the test mode with recirculation to the source of water. Pump seal leakage is measured with the pump operating. In most cases, the system lineup provides postaccident flows and pressures.

In some cases, it is not practical to inspect systems operating in the test mode. In such cases, the system is inspected under static conditions. Portions of the system that operate at high pressure are pressurized, with a hydrostatic pressure test pump, to the operating pressure. (Note that piping between pump discharges and the first block valve may not be pressurized.)

The steam side of the RCIC and HPCI systems are tested with nuclear steam, unless this is precluded by high-radiation levels. In that case, auxiliary steam is used during shutdown.

Detailed test procedures for water pressurized systems and for steam pressurized systems have been implemented.

Corrective Action

The results of these inspections are noted for evaluation and corrective actions where required. It should be noted that in some cases that reported leakage will be as low as reasonably achievable (e.g., pump seal leakoffs). In cases where corrective action is performed, the leakage will be measured and recorded after the corrective action is complete.

Cases where leakage is observed, but is too low to be quantified or corrected, are monitored during the preventative maintenance program.

Documentation

Attached to each inspection procedure is a form listing each component to be inspected for leakage; the inspection of each system component is documented on this form.

The report forms are retained for future use and documentation of the work.

If the inspection reveals additional components not listed on the forms, such components shall be added to the form by the inspector.

12.1.3.1.3 Preventative Maintenance Program

The data collected during the inspections are evaluated for acceptability and capability for further leak reduction.

Additional walkdown inspections (without the component checklists attached to the system procedures) are performed quarterly. This may be done in conjunction with the system operation test if the area is accessible. A general inspection for visible leakage is performed.

The scram discharge volume headers will not have a quarterly walkdown inspection performed as the headers are vented and drained during normal plant operation. The Containment Atmosphere monitoring system and post accident containment sampling system will not have a quarterly walkdown inspection performed as these are gaseous systems and general inspections for visible leakage are not practical.

Once per year, or no less frequently than once per refueling cycle, a detailed walkdown inspection with the component checklists is made. Whenever practicable the inspection is performed in conjunction with the system pressure test required by Section XI of the ASME B&PV Code (Article IWA5000). Gaseous systems will be tested by measuring the leakage and walkdowns will be performed as necessary to eliminate the sources of excess leakage.

In both of the above cases, special attention is given to leakages reported during previous inspections.

Any detected leaks are evaluated and repaired as required. The effectiveness of repairs is verified after completion.

The DAEC is taking prompt action to reduce all leakages detected to as-low-as-reasonably-achievable levels. Corrective action is being documented and retests are being made. The preventative maintenance program requires regular reinspection of these systems and prompt corrective action where required.

The initial leak-testing and reduction program has been completed and corrective action accomplished.

12.1.3.2 Access Control

A system specifying the use of cautions signs, areas posting, labelling, and personnel access controls is in use at DAEC in accordance with regulations contained in 10CFR20.

Plant procedures are used to directly implement 10CFR20 regulations and to meet the 10CFR19 requirement regarding providing information to workers concerning exposure to radiation. These plant procedures provide detailed instructions for posting and controlling access to Radiation Areas, High Radiation Areas, Locked High Radiation Areas, Very High Radiation Areas, and Airborne Radioactivity Areas.

Access controls are specified in plant procedures to limit personnel access as necessary to positvely control the radiation dose received by personnel. A graded system of access controls is implemented, based on the severity of the radiological conditions and the potential magnitude of exposure resulting from area entry. The DAEC system of access controls utlizes a combination of physical barriers and adminstrative controls, and meets the follwing objectives:

a. Ensure that personnel accessing areas with a potential for radiation exposure have received appropriate training and are monitored appropriately with personnel monitoring device(s).

b. Prevent unauthorized access to areas.

c. Meet the 10CFR19 requirements relative to providing information and instructions to workers regarding exposure to radiation.

d. Meet 10CFR20 access control and dose limit requirements for both occupational radiation workers and members of the public.

e. Ensure that doses resulting from all entries are maintained ALARA.

12.2 RADIATION SOURCES

12.2.1. Contained Sources

Specific activities for various radioisotopes in the primary coolant are given in Tables 12.2-1, 12.2-2, 12.2-3, and 12.2-4. See Figure 11.2-1 for potential sources in the liquid radwaste system.

Some daughter isotopes (for example yttrium and lanthanum) have not been listed in reactor water. Their independent release from fuel is expected to be negligible; however, the isotopes may be observed in some samples in equilibrium or approaching equilibrium with the parent isotope.

12.2.2 Airborne Radioactive Material Sources

See Section 11.1.

Table 12.2-1

COOLANT ACTIVATION PRODUCTS

Isotope	Concentration (μCi/g)
Reactor Steam	
N-13	6.5×10^{-3}
N-16	1.0×10^{2}
N-17	1.6×10^{-2}
O-19	8.0×10^{-1}
F-18	4.0×10^{-3}
Reactor Water	
(Equilibrum values - entering recirculation lines)	
N-13	4.3×10^{-2}
N-16	6.0×10^{1}
N-17	5.8×10^{-3}
O-19	$7.6\ 10^{-1}$
F-18	4.8×10^{-3}

Table 12.2-2

REACTOR WATER NONCOOLANT ACTIVATION PRODUCTS[a]

Isotope	Concentration (μCi/g)
Na-24	2.4×10^{-3}
P-32	2.4×10^{-5}
Cr-51	6×10^{-5}
Mn-54	4.8×10^{-5}
Mn-56	6×10^{-2}
Co-58	6×10^{-3}
Fe-59	9.6×10^{-5}
Ni-65	3.6×10^{-4}
Zn-65	2.4×10^{-6}
Zn-69m	3.6×10^{-5}
Ag-110m	7.2×10^{-5}
W-187	3.6×10^{-3}

[a] Products formed by activation of impurities in the coolant or by corrosion of irradiated system materials.

Table 12.2-3

REACTOR WATER FISSION PRODUCTS - HALOGENS

Isotope	Concentration (μCi/g)
Br-83	3.4×10^{-2}
Br-84	6.31×10^{-2}
Br-85	3.84×10^{-2}
I-131	3.11×10^{-2}
I-132	2.82×10^{-1}
I-133	2.08×10^{-1}
I-134	5.57×10^{-1}
I-135	3.02×10^{-1}

Table 12.2-4

REACTOR WATER FISSION PRODUCTS - OTHER ISOTOPES

Isotope	Concentration (μCi/g)
Sr-89	6.96×10^{-3}
Sr-90	5.28×10^{-4}
Sr-91	1.56×10^{-1}
Sr-92	2.52×10^{-1}
Zr-95	9.12×10^{-5}
Zr-97	7.32×10^{-5}
Nb-95	9.48×10^{-5}
Mo-99	5.16×10^{-2}
Tc-99m	6.48×10^{-1}
Tc-101	3.12×10^{-1}
Ru-103	4.44×10^{-5}
Ru-106	5.88×10^{-6}
Te-129m	9.00×10^{-5}
Te-132	1.12×10^{-2}
Cs-134	3.72×10^{-4}
Cs-136	2.40×10^{-4}
Cs-137	5.52×10^{-4}
Cs-138	4.32×10^{-1}
Ba-139	3.72×10^{-1}
Ba-140	2.04×10^{-2}
Ba-141	4.08×10^{-1}
Ba-142	3.84×10^{-1}
Ce-141	9.00×10^{-5}
Ce-143	7.92×10^{-5}
Ce-144	8.04×10^{-5}
Pr-143	8.64×10^{-5}
Nd-147	3.24×10^{-5}
Np-239	5.52×10^{-1}

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 FACILITY DESIGN FEATURES

12.3.1.1 Radiation Zones

A system in which all plant areas were identified by radiation zones according to expected operational dose rates and anticipated area occupancy was used in initial plant design to determine shielding requirements to minimize personnel dose for operation and maintenance of the plant.

Radiation zones are shown in Figures 12.3-1 through 12.3-5, and are based on expected dose rates during operation at design power operation as represented in the following key:

Zone	Design Dose Rate (mrem/hr)
A	≤ 0.5
B	≤ 1.0
C	≤ 6.0
D	≤ 12.0
E	≤ 100
F	>100

Personnel access to plant areas is subject to an access controls and posting system. Access controls and postings are based on periodically updated field measurements of dose rates; the radiation zone system used in initial plant design may or may not be consistent with current plant dose rates, and has no direct impact on the operational Radiation Protection program.

Techniques may be implemented in accordance with plant procedures to reduce the magnitude of station radiation fields in order to maintain personnel dose ALARA. Techniques available to achieve this result include system flushing, mechanical or chemical decontamination, and installation of permanent or temporary shielding.

12.3.2 SHIELDING

12.3.2.1 Safety Objectives

1. The primary objective of radiation shielding is to restrict the exposure of operating personnel and the general public to radiation emanating from the reactor, turbine, and auxiliary systems.

2. The secondary objective of radiation shielding is to reduce radiation effects on materials to acceptable levels. Of principal concern are organic materials used as insulation, tank linings, gaskets, etc.

12.3.2.2 Safety Design Bases

1. Shielding design for normal plant operations ensures that radiation exposures are in accordance with 10CFR20 requirements.

2. For design-basis accidents, shielding is provided for individuals occupying the plant control room sufficiently to limit their exposures for a 30-day period to not more than 5 rem Total Effective Dose (TEDE).

3. All areas of the plant are zoned according to their design radiation level and expected length of occupancy by personnel under normal operating conditions.

4. No regulations similar to those established for the protection of individuals exist for materials and components. However, materials and components are selected on the basis that radiation exposure as a result of the shielding design will not cause significant changes in their physical properties that could adversely affect operation of equipment during the design life of the plant. Materials for equipment required to operate under accident conditions are selected on the basis of the additional exposure received in the event of the limiting design-basis accident.

12.3.2.3 General Shielding Design Criteria

The shielding design considers the following three conditions:

1. Operation at Design Power

   This includes the shielding requirements associated with operating the plant with defective fuel elements corresponding to an offgas rate of 100,000 μCi/sec after a 30-min decay.

2. Shutdown

   This condition deals mainly with the radioactivity from the subcritical reactor core, with radiation from spent-fuel assemblies during onsite transfer, with the residual fission product activity in the reactor coolant, and with neutron-activated materials.

3. Design-Basis Accidents

   These are the hypothetical accidents for which design-basis fission product source terms are described in Chapter 15.

12.3.2.4 <u>Shield Design Calculations</u>

A list of publications and computer programs that have been used in the design of the radiation shielding is provided in References 2 through 11.

12.3.2.5 <u>Shielding Material</u>

The material used for most of the plant shielding is ordinary concrete with a bulk density of 147 lb/ft^3 . Wherever cast-in-place concrete has been replaced by concrete blocks, the design ensures protection on an equivalent shielding basis. Only in a very few instances has steel or water been used as primary shielding materials.

12.3.2.6 <u>Description of Plant Shielding</u>

The different areas of radiation protection are described as listed by specific location or building for convenience.

12.3.2.6.1 Main Control Room

The shielding of the main control room has been designed to limit the dose rate to operating personnel within the control room to less than 0.5 mrem/hr during normal plant operations.

In addition to normal operations, the radiation conditions resulting from the design-basis accidents have been evaluated. Adequate shielding has been provided to permit access and occupancy of the control room for a 30-day period without personnel receiving radiation exposures in excess of 5 rem whole body.

12.3.2.6.2 Reactor Building

The reactor building contains four major shielding structures: the reactor sacrificial shield, the drywell biological shield, the main steam pipe chase, and the spent-fuel pool.

The sacrificial shield has several shielding functions. It protects certain major portions of the drywell space from excessive nuclear radiation exposures during operation. After shutdown, it provides protection from reactor vessel radiation for plant personnel engaged in inservice inspection, maintenance, and repair of drywell equipment and components. Also, together with the drywell biological shield, it protects the general reactor building work areas. [redacted]

The drywell biological shield concrete together with the reactor sacrificial shield provide the main protection for the areas surrounding the reactor vessel, the primary coolant, and recirculation systems. [redacted] is used to keep the radiation dose rates in the fully accessible reactor building work areas to less than 1.0 mrem/hr.

The main steam line pipe chase, with [redacted] is the connecting shield structure between the reactor and turbine buildings. The pipe chase shielding protects against the N-16 gamma radiation that is contained in the passing steam.

The spent-fuel pool contains the highly radioactive spent-fuel assemblies. [redacted] When the 360 degree work platform is in use, and fuel is being transferred, [redacted] This configuration has been evaluated and DAEC will mitigate the radiation exposure to personnel using shielding, as needed.

The RWCU system, the incore flux monitoring equipment, the radwaste equipment, and the reactor internals during storage are housed in numerous concrete-shielded rooms surrounding the drywell concrete structure. Enclosing these secondary sources of radiation in shielded rooms permits the adjacent areas to be accessible to personnel on a 40-hr/week basis. The entrances into the drywell space are well shielded (equipment lock and personnel access lock).

A permanent radiation shield has been built around the traversing incore probe drive mechanisms to reduce the radiation field in the vicinity of the mechanisms that exists when a probe is pulled into the drive mechanism. [redacted]

A permanent radiation shield has been built around the reactor building sample hood area to reduce the radiation field around the normal access door to the radwaste facilities and the surrounding area. [redacted] There is no safety-related equipment close to the shield walls that could be damaged should the wall fail during a seismic event.

12.3.2.6.3 Turbine Building

Fission and activation products are transported with the steam and some enter the turbine and turbine condenser. Approximately 80% of the activity is discharged via the air ejector to the offgas system while the other 20% remains in the condensate and is treated by the condensate filter-demineralizers.

Radiation shielding is provided around the following areas:

1. Main steam lines.
2. Primary and extraction steam piping.
3. High-pressure and low-pressure turbines.
4. Feedwater pumps.
5. Moisture separators.
6. Reactor feedwater system heaters.
7. Main condenser and hotwell.
8. Air ejectors and steam packing exhauster.
9. Condensate demineralizer.
10. Offgas lines.

Some of the equipment, such as the air ejectors, feedwater pumps, and heaters, are in individual rooms or areas enabling the shutdown of part of the system without interrupting plant operation.

[redacted] allows 40 hr/week access to the generator and central laydown area.

12.3.2.6.4 Radwaste Building

The design basis for the shielding of the radwaste facility assumes the quantity of radioactivity in the reactor coolant is that which results in an offgas release of 100,000 μCi/sec, after 30-min decay. All areas for preparing, handling, or storing the radwaste are shielded to meet these conditions.

The individual radwaste systems have been separated from each other and shielded as much as practicable in order to minimize personnel exposure during maintenance and repair of any of the equipment. The fully accessible areas surrounding the radwaste building are adequately shielded.

12.3.2.6.5 Process Piping and Valve Stations

Plant design and field construction personnel routed process pipe containing radioactive fluids in such a manner that the radioactive shine hazard to plant personnel through shield was penetrations is minimized. Radiation levels at valve stations for process equipment containing radioactive fluids are in accordance with the criteria discussed in Section 12.3.1.1 and shown in Figures 12.3-1 through 12.3-5.

12.3.2.6.6 Other Plant Areas

12.3.2.6.6.1 Administration Building and Shop and Warehouse. All areas of the administration building and the adjacent shop and warehouse areas are fully accessible at all times. The shop building has an area that will handle radioactive material and is controlled accordingly.

12.3.2.6.6.2 Technical Support Center. The occupied areas of the technical support center (TSC) have been provided with adequate shielding to permit access and occupancy for a 30-day period following the design-basis accidents without personnel receiving radiation exposures in excess of 5 rem TEDE.

12.3.2.6.6.3 Stack. The shielding design for the stack is based on a gaseous fission product release rate of 100,000 μCi/sec after 30-min decay, as well as the accompanying radioactive particulates removed by the offgas filters. Shielding is provided for controlled access at ground level to maintain the filters and instrumentation.

12.3.2.6.6.4 General Plant Yard Area. Plant yard areas that are frequently occupied by plant personnel generally receive a dose rate of less than 0.5 mrem/hr.

12.3.2.6.7 Low-Level Radwaste Processing and Storage Facility (LLRPSF)

[redacted] The LLRPSF design includes fixed shielding of the resins and DAW storage areas to limit radiation levels in accessible areas inside the building and in areas outside the building. The design of the shielding configuration is based on the following criteria.

1. The dose rate at the site boundary resulting from LLRPSF sources shall be less than or equal to 5 mrem/year; and
2. The dose rate on LLRSPF external walls shall be less than or equal to 0.5 mrem/hr.

12.3.2.8 Design Review of Plant Shielding for Postaccident Operations

A postaccident (DBA) shielding evaluation has been performed for the DAEC. The dose rates calculated for the operational support center, and turbine building are shown in Table 12.3-1. Control Room and TSC dose consequences are described in Chapter 15.2

As a result of the shielding evaluation, two general design modifications were needed to permit access to vital areas under postaccident conditions. The first modification involved construction of a backup sample station for reactor building ventilation exhaust stack effluents and is discussed in Section 11.5.5.3. The second modification involved installation of a new postaccident sampling system and is discussed in Section 12.3.4. As documented in Reference 12, the NRC verified completion of the modifications and concluded that DAEC plant shielding meets the guidance of NUREG-0737.

12.3.2.9 Inspection and Performance Analysis

The normal construction quality control program ensured that there were no major defects in the shielding. The quality control program included onsite inspection surveys by the shielding designer before plant startup.

Since plant startup, the adequacy of the shielding is checked by radiation surveys.

12.3.3 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

12.3.3.1 Area Radiation Monitoring System Power Generation Objective

The power generation objective of the area radiation monitoring system is to warn of abnormal gamma radiation levels in areas where radioactive material may be present, stored, handled, or inadvertently introduced and to provide information regarding radiation levels at selected locations within the plant.

12.3.3.2 Area Radiation Monitoring System Power Generation Design Bases

1. The area radiation monitoring system provides operating personnel in the main control room with indication of gamma radiation levels at selected locations within the various plant buildings.

2. The area radiation monitoring system provides local indication and alarms where it is necessary to warn personnel of substantial immediate changes in radiation levels. High radiation levels in any area of the plant will activate an annunciator in the main control room.

12.3.3.3 System Description

12.3.3.3.1 Area Radiation Monitoring System

The area radiation monitoring system is shown as a functional block diagram in Figure 12.3-6. Each channel consists of a combined sensor and converter unit, a combined indicator, audible alarm and trip unit, a shared power supply, and a shared multipoint recorder. Figure 12.3-7 shows the locations of the area radiation monitors.

Each monitor has an upscale trip that indicates high radiation and a downscale trip that may indicate instrument trouble. These trips sound alarms but cause no control action. The system is powered from the 120-V ac instrument bus. The trip circuits are designed so that a loss of power causes an alarm. The environmental and power supply design conditions are given in Table 12.3-2.

12.3.3.3.2 Area Radiation Monitor Locations

Monitors are located in appropriate areas within the reactor building, turbine building, control building, radwaste building, LLRPSF, and administration building. Annunciation and indication are provided in the main control room, except for the LLRPSF. Some monitors also provide local indication and all provide an alarm at the detector.

12.3.3.3.3 Technical Support Center Radiation Monitoring System

A radiation monitoring system has been installed in the technical support center to meet the requirements of NUREG-0578, Section 2.2.2.b.

An Eberline monitoring system consisting of three detector assemblies and three readout channels with remote indicators provides the capability of monitoring radiation at both the TSC air intake and within the technical support center itself. Two detector assemblies are located inside the air intake duct to monitor incoming air. One detector is located in the engineering support staff area to monitor ambient air. Readout channels are located in the communications room.

The radiation monitoring system for the technical support center is not safety related. It has no interface with any other radiation monitoring system associated with the plant.

12.3.3.3.4 Containment High-Range Monitors

High-range containment radiation monitors have been installed in response to NUREG-0737, Section II.F.1.3. They consist of four (two in the torus and two in the drywell) physically separated monitors designed and qualified to function in an accident environment and with a maximum range of 10^7 rad/hr.

12.3.3.3.5 Improved In-Plant Iodine Instrumentation Under Accident Conditions

In response to NUREG-0578, Section 2.1.8.c, the DAEC has installed a multiple channel analyzer system to measure I-131 concentrations that may be present in various region(s) of interest following an accident. This system, along with the training of appropriate personnel in its calibration and use under special procedures, provides reasonable assurance that the DAEC has the capability to accurately detect and thereby obtain an initial estimate of the presence of I-131 to determine if the use of respiratory protection equipment by plant personnel is warranted or required.

12.3.3.3.6 Airborne Radioactivity Monitoring System

Continuous air monitors with fixed particulate and iodine collectors are located in the turbine building, reactor building, and offgas stack. They provide indication of changing airborne activity conditions. In addition, low volume air samplers run continuously in various areas of the plant for evaluating general airborne activity levels.

12.3.3.3.7 Low-Level Radwaste Processing and Storage Facility (LLRSPF) Monitoring System

An area radiation monitoring system has been installed in the LLRPSF. The system is furnished by Eberline and consists of three monitors located as follows: one in the truck bay, one in the radwaste access area, and one in the LLRPSF control room. Local displays are provided for all the monitors, with remote indication provided in the LLRPSF control room.

The radiation monitoring system for the LLRPSF is not safety related nor does it interface with any other radiation monitoring system associated with the plant.

12.3.3.4 Area Radiation Monitoring System Inspection and Testing

An internal trip test circuit, adjustable over the full range of the trip circuit, is provided. The test signal is fed into the indicator and trip unit input so that a meter reading is provided in addition to a real trip. All trip circuits are of the latching type and must be manually reset at the front panel. A portable calibration unit is also provided. This is a test unit designed for use in the adjustment procedure for the area radiation monitor sensor and converter unit. A cavity in the calibration unit is designed to receive the sensor and converter unit. Located on the back wall of the cylindrical lower half of the cavity is a window through which radiation from the source emanates. A chart on each unit indicates the radiation levels available from the unit for the various control settings.

12.3.4 POSTACCIDENT SAMPLING SYSTEM

The function of the postaccident sampling system (Figure 12.3-8) is to obtain representative liquid samples from the primary containment and the reactor coolant system and gas samples from primary and secondary containments for radiological and chemical analysis in association with a postulated loss-of-coolant accident. Generic design requirements are given in Reference 1. A detailed system description is given in Section 12.3.4.2 below.

Per NRC letter, dated 8/8/03, (TS Amendment 252), the DAEC Post-Accident Sampling System capability to sample and analyze for the following may be eliminated:

1. Reactor Coolant dissolved gases
2. Reactor Coolant hydrogen
3. Reactor Coolant oxygen
4. Reactor Coolant chloride
5. Reactor Coolant pH
6. Reactor Coolant boron
7. Reactor Coolant conductivity
8. Reactor Coolant radioisotopes
9. Containment atmosphere hydrogen
10. Containment atmosphere oxygen
11. Containment atmosphere radioisotopes
12. Suppression Pool (Torus) pH
13. Suppression Pool (Torus) chloride
14. Suppression Pool (Torus) boron
15. Suppression Pool (Torus) radioisotopes

However, the NRC required certain contingency plans for sampling and analyses of highly radioactive samples. The following is a description of those sampling and analyses requirements.

<u>Contingency Sampling/Analyses:</u>

Radioisotope analyses are not required to support emergency decision-making during initial phases of an accident. Therefore, dedicated equipment for prompt sampling is not required. However, the NRC concluded that radioisotope analyses could provide useful information in the longer term. The NRC requires that the plant have contingency plans for sampling and analyses. Contingency plans include sampling of reactor coolant, torus water, and containment atmosphere. Contingency analyses include: radioisotopes in reactor coolant, torus water, and containment atmosphere; containment atmosphere hydrogen; and torus water pH. Contingency sampling and analyses are done using plant procedures.

Containment hydrogen monitors continues to be required. The NRC also requires that the plant maintain the capability to obtain grab samples to complement the hydrogen monitors in the long term. Note: During normal operation, the continuous indication of hydrogen concentration is not required.

Because contingency samples and analyses are not required for short-term or prompt decisions during the early phases of an accident, contingency actions include assessments of ALARA and other requirements to determine any requirements of limits of such actions.

Classification of Fuel Damage Events at the Alert Level

The NRC requires that the plant maintain capability to classify fuel damage events at the Alert level threshold (commensurate with 300 µCi/ml dose equivalent iodine (DEI)) using normal sampling systems and/or correlations of radiation readings to radioisotope concentrations in reactor coolant. This commitment is met through the use of containment radiation monitors to classify EALs in the DAEC Emergency Plan. The capability to sample highly radioactive samples and analyze for DEI for the purpose of EAL classification will not be maintained.

Iodine-131 Survey Capabilities

The NRC requires that the station verify and commit to maintaining an Iodine-131 site survey capability to assess radioactive iodines released to the offsite environs using effluent monitoring systems of portable sampling equipment. This capability currently exists.

The remainder of Section 12.3.4 is maintained for historical purposes.

12.3.4.1 Design Bases

12.3.4.1.1 Safety Design Bases

Although not safety related itself, the postaccident sampling system has the following safety-related interfaces.

1. Sample lines and components that are connected to the containment atmosphere monitoring system are classified as safety related, Quality Group D, and Seismic Category I up to and-including the sample line isolation valve.

2. Sample lines and components that are connected to the jet pump flow-sensing instrument lines are classified as safety-related, Quality Group A, and Seismic Category I up to and including the outboard containment isolation valve.

3. The liquid sample return line from the torus penetration to the outboard containment isolation valve is classified as safety-related, Quality Group B, and Seismic Category I

4. Reactor liquid samples drawn from the jet pump flow-sensing instrument lines shall be capable of being taken with the containment isolated.

5. Reactor or suppression pool liquid samples drawn from the residual heat removal system shall be capable of being taken with a residual heat removal (RHR) isolation signal actuated.

6. Ventilation exhaust piping from the secondary containment penetration to the outboard isolation damper is classified as safety-related, Quality Group D, and Seismic Category I.

7. Sample station ventilation shall be capable of being established with the secondary containment isolated.

8. The system is designed to permit samples to be taken with or without offsite power available.

9. Isolation valves in each RHR sample line and each jet pump flow-sensing instrument line shall be powered from diverse electrical power divisions to provide the capability to open at least one sample line in each system following a loss of power in one division. Diverse isolation signals shall be provided for the valves in each sample line.

12.3.4.1.2 Power Generation Design Bases

1. The system is designed to meet NUREG-0737, Section II.B.3, requirements regarding sampling capability.

2. The system is designed to permit sampling at any time during normal plant operation or during abnormal conditions.

3. The system is designed such that it does not interfere with normal operations.

4. The system is designed to enable personnel to obtain and analyze a sample without radiation exposures to any individual exceeding the criterion of GDC 19.

5. The sampling system is designed in accordance with applicable codes and standards.

6. The cooling water supply for the sample coolers is designed to be operable with or without offsite power available.

12.3.4.2 System Description

12.3.4.2.1 General Description

The postaccident sampling system is designed to enable an operator to obtain representative grab samples of reactor coolant, suppression pool liquid, and containment atmosphere for radiological and chemical analyses in association with a postulated loss-of-coolant accident. The system consists of a sample station, sample control panels, a sample piping station, a sample station exhaust fan, a cyclone separator rack, a refrigeration unit, and demineralized water, nitrogen, and tracer gas supplies.

The sampling system equipment is located in two areas of the plant. The sample piping station, cyclone separator rack, and the sample station exhaust fan are located in the northwest corner room inside the reactor building. The sample station, sample control panels, refrigeration unit, and demineralized water, nitrogen, and tracer gas supplies are located in the administration building access control area. Isolation valves for liquid and gas sample lines, sample return lines, and the sample station exhaust duct isolation dampers are operated from the control room. The sample station and components located inside the reactor building but not operated from the control room are remotely operated from the sample control panels in the access control area.

The sample station consists of a liquid sampling unit, gas sampling unit, sampler mounting frame, and associated lead brick shielding. The liquid and gas sampling units each contain a compact, removable equipment tray designed to provide easy access to individual components for maintenance. Special sample handling tools are provided for installing and removing sample bottles from the sample station. Shielded sample casks are provided for transporting samples from the sample station to the laboratory areas.

12.3.4.2.2 Liquid Samples

The liquid sampling unit is used to obtain small- and large- volume samples of reactor coolant and suppression pool liquid. The small-volume sample is a 0.1-ml liquid sample intended for onsite analysis when activity levels are high. The large-volume sample is approximately a 10-ml liquid sample provided for onsite analysis when activity levels are low and for shipment of samples to an offsite laboratory for independent analysis. The liquid sampling unit is also used to obtain a sample of dissolved gas collected from a specific volume of liquid sample.

Reactor liquid samples are obtained from two jet pump flow-sensing instrument lines when the reactor is pressurized. This sample location was selected as an optimum sample point during accident conditions because the pressure taps for the instruments are protected from damage and debris. With the recirculation pumps secured, natural circulation is expected to provide a continuous flow of reactor coolant past these taps. The taps are located at an elevation which permits sampling with reactor water level below the lower core support plate. The reactor coolant sample lines connect to the jet pump flow-sensing instrument lines (high-pressure tap) from jet pump [redacted] and [redacted] outside of the drywell. The sample lines are classified as Quality Group A and Seismic Category I up to and including the sample line containment isolation valves. Downstream of the isolation valves, the sample lines are classified as Quality Group D and Nonseismic.

Liquid samples can also be taken from both loops of the RHR system. A reactor coolant sample can be obtained when the reactor is depressurized and one loop of the RHR system is operating in the shutdown cooling mode. Similarly, a suppression pool liquid sample can be obtained from the RHR loop operating in the suppression pool cooling mode. The sample lines connect to the RHR heat exchanger discharge sample lines downstream of the discharge sample line isolation valves in each RHR loop. The postaccident sampling system lines are classified as Quality Group D and Non-seismic downstream of the sample isolation valves.

Purge flow for all liquid samples is directed to the suppression pool through the liquid sample return line. Containment isolation valves are provided on the sample return line upstream of the torus penetration. The liquid sample return line is classified as Quality Group B and Seismic Category I from the torus penetration to the outboard containment isolation valve. Upstream of the isolation valves, the sample return line is classified as Quality Group D and Nonseismic.

Isolation valves for the jet pump and RHR sample lines and for the liquid sample return line are operated from the control room. To enable sampling with the primary containment isolated after an accident, each of the isolation valves has been provided with an override for the isolation signal. During sampling operations, liquid sample lines are initially purged at a flow rate of 1 gpm, which is sufficient to maintain turbulent flow in the sample line. Liquid samples are cooled before entering the sample station by sample coolers located on the sample piping station. Cooling water is supplied from the reactor building cooling water system.

Hydrocyclone separators are installed on each of the RHR sample lines upstream of the sample piping station to remove excessive insoluble impurities from RHR system/suppression pool liquid samples. The separators are designed to remove 95% of the particles greater than 10 μ in size based on specific gravity of a particle, approximately the same as that of ferric oxide. The separators are used to minimize the potential for plugging in the system when sampling RHR system or suppression pool liquid after an accident. The use of cyclone separators will not affect the accuracy of postaccident core damage estimates because such estimate's are based on the sample I-131 concentration, which is expected to be present in stable ionic solution.

The reactor recirculation system process sample line (not part of the postaccident sampling system) also has postaccident liquid sample capabilities that could be used as a backup. See Section 9.3.2 for a discussion of the process sampling system.

12.3.4.2.3 Gas Samples

The gas sampling unit is used to obtain gas samples from the drywell and suppression pool atmosphere and from the secondary containment (reactor building) atmosphere. The gas sample system is designed to operate at pressures ranging from subatmospheric to the pressure inside primary containment 1 hr after a LOCA. Gas samples can be passed through a particulate filter and silver zeolite cartridge for determination of particulate activity and total iodine activity by counting the samples on a gamma spectrometer. Alternatively, the sample flow can bypass the iodine sampler, be chilled to remove moisture, and a 15 ml grab sample taken for determination of gaseous activity and for gas composition by gas chromatography. This size sample vial has been adopted for all gas samples to be consistent with offgas sample vial counting factors.

Primary containment atmosphere samples are obtained from the containment atmosphere monitoring (CAM) system. Redundant sample connections for drywell and suppression pool atmosphere samples are provided. The sample lines connect to CAM system sample lines downstream of the containment isolation valves. The sample lines are classified as Quality Group D and Seismic Category I up to and including the sample isolation valves. Downstream of the isolation valves, the sample lines are classified as Quality Group D and Nonseismic.

Gas sample purge flow is directed to the suppression pool atmosphere through a CAM system gas sample return line. The sample return line connects to the CAM system return line upstream of the containment isolation valves. The sample return line is classified as Quality Group D and Seismic Category I up to and including the isolation valve. Upstream of the isolation valve, the sample return line is classified as Quality Group D and Nonseismic.

Containment isolation valves for the CAM system sample lines are operated from the control room. Each group of isolation valves has been provided with a key-lock override for the isolation signal to enable sampling with the containment isolated. The isolation valves for the gas sample lines and sample return line which connect to the CAM system are operated from the sample control panel. These isolation valves are normally closed valves which are not actuated by a containment isolation signal.

The PASS gas sample lines are heat-traced to minimize precipitation of moisture. Redundant, non-Class 1E heat tracing has been provided which is sized to hold a line temperature of approximately 200-215°F and is capable of continuous operation. All heat-traced sample lines are thermally insulated.

Positive displacement vacuum pumps are provided to draw gas samples through the sampling unit at a minimum flow rate of 0.3 scfm. The gas sampler is equipped with a flow-indicating device and direct-reading pressure gauge for sample bottle air evacuation status. Grab samples taken into sample vials are cooled using chilled water supplied from a refrigeration unit to remove entrained moisture.

Local area radiation detector [redacted] and monitor [redacted] are provided to inform the operator of the ambient radiation level near the sample station. In addition, radiation detectors with individual channel monitors are provided for the liquid and gas sampling units. Radiation detector [redacted] and monitor [redacted] provide an immediate assessment of liquid sample activity and also provide indication of the effectiveness of the demineralized water flush of the system following the sampling operation. Radiation detector [redacted] and monitor [redacted] are provided to monitor the deposition of radioactive material on the filter cartridges during gas sampling operations.

To minimize radiation exposure to personnel, the liquid and gas sampling units are equipped with external lead brick shielding. The liquid sampler is surrounded by 6 in. of lead shielding; the gas sampler is provided with 2 in. of lead shielding. Additionally, a demineralized water purge is employed in the liquid sampler to displace radioactive sample liquid with demineralized water. The motive force for purge water flow is provided by pressurizing the demineralized water tank with nitrogen. Nitrogen is also used to purge radioactive sample gas from sample lines and filter cartridges in the gas sampler unit.

12.3.4.2.4 Sample Handling

Liquid and gas grab samples are taken into septum type sample bottles mounted on sampling needles. Sample bottles are installed and removed from the sample station using special sample handling tools. Gas sample vials are installed and removed using a vial positioner through the front of the gas sampler. After removal, the gas vial is manually transferred into a gas vial cask for transport to onsite laboratory facilities.

The particulate filters and iodine cartridges are removed from the gas sampler using a special drawer arrangement. The quantity of activity which is accumulated on the cartridges is controlled by a combination of flow orificing and time sequence control of the flow valve opening. In addition, the deposition of iodine is monitored during sampling using the radiation detector installed adjacent to the cartridge. These samples will be limited to activity levels which will minimize the size of the shielded sample carriers needed to transport the samples to onsite laboratory facilities.

The small-volume liquid sample is obtained remotely through the bottom of the sample station using a small-volume cask and cask positioner. The cask positioner holds the cask and positions the cask directly under the liquid sampler. The sampler vial is manually raised from the cask to engage the hypodermic needles. When the sample bottle has been filled, the bottle is manually withdrawn into the cask. The cask is then lowered and sealed for transport to onsite laboratory facilities.

A large-volume cask and cask positioner is used for handling large-volume liquid samples. The cask is transported into position under the sample station using a four-wheel dolly cask positioner. When in position, this cask is hydraulically raised by a small hand pump to contact the sample station shielding under the liquid sampler. The sample bottle is raised from the cask to engage the hypodermic needles using a simple push/pull cable. When filled, the sample bottle is withdrawn into the cask, and the cask is sealed with a threaded top plug. The large-volume sample can then be transported for onsite analysis, or be prepared for shipment to an offsite laboratory for independent analysis. The sample bottle is shielded by 5 to 6 in. of lead when in position under the sample station and during the raise, fill, and withdraw operations to minimize radiation exposure to the operator.

The sample station exhaust system is used to control the leakage of gaseous radioactivity from the sample station. The system consists of an enclosure which surrounds the liquid and gas sampling units; an air inlet at the top of the enclosure; an exhaust fan; and exhaust piping, isolation dampers, and ductwork. A sample station exhaust fan, located inside the reactor building, is used to maintain the sample station under negative pressure. Ventilation exhaust from the sample station is drawn by the exhaust fan through piping and ductwork into the reactor building and discharged through ductwork into a reactor building exhaust duct. From there, the sample station exhaust exits through the reactor building exhaust stacks during normal plant operation. The sample station exhaust is directed through the reactor building exhaust duct to the standby gas treatment system (SGTS) whenever secondary containment has been isolated and the SGTS is in operation.

Two secondary containment isolation dampers are provided on the sample station exhaust piping downstream of the secondary containment penetration. The isolation dampers are operated from the control room. To allow ventilation to be established with the secondary containment isolated, the isolation dampers have been provided with a key-lock override for the isolation signal. The sample station exhaust piping is classified as Quality Group D and Seismic Category I from the secondary containment penetration through the second isolation damper. Nonseismic stainless steel duct is used from the second isolation damper to the connection at the reactor building exhaust duct.

12.3.4.2.5 Chemical Analysis

A postaccident sampling chemical analysis laboratory is located in the DAEC administration building at elevation [redacted] in the vicinity of the normal plant hot chemistry laboratory. The design of the postaccident sampling chemical laboratory is based upon the General Electric generic design requirements provided in Reference 1.

The postaccident sampling chemical laboratory is equipped to perform the following sample, analyses:

1. Quantify hydrogen, oxygen, and nitrogen levels in containment atmosphere gas samples.

2. Reactor coolant chloride scoping analyses. In order to satisfy the chloride-measurement requirements of NUREG- 0737, Item II.B.3, provisions have been made with an offsite laboratory for analysis of postaccident samples.

The results of this analysis will be available within the time period required by NUREG-0737.

3. Measure the pH of the postaccident liquid samples.

4. Determine boron concentration in postaccident liquid samples.

The chemical laboratory is equipped to determine the total dissolved gas concentration in liquid samples by sampling the gas phase over a specific liquid volume and applying Henry's Law. Sample dilution capability, in addition to the partial dilution capability at the sample panel, is also provided in the chemical laboratory.

The Postaccident Sampling Procedures describe the analysis equipment available and the techniques used.

After analysis, the samples will be stored in a shield until they can be disposed of properly.

12.3.4.2.6 Radiological Analysis

A postaccident sampling radiological analysis laboratory (counting room) is located in the DAEC administration building at elevation [redacted] and is adjacent to the postaccident sampling station. The design of the postaccident sampling radiological analysis laboratory is based upon the General Electric design requirements given in Reference 1.

The counting room is equipped with a hyperpure germanium detector and computerized multichannel analyzer. This equipment provides the capability to identify and quantify the isotopes of nuclides in reactor coolant and containment atmosphere samples as described in the NUREG-0737, Item II.B.3. The equipment is based upon the recommendations contained in Reference 1 and has accuracy, range, and sensitivity adequate to provide pertinent data to the operator to describe the radiological status of the reactor coolant system.

Primary coolant samples obtained from the station are diluted by a factor of 100 (0.1 ml diluted to 10 ml). By using dilution, extended shelf geometry, or absorbers as needed, isotopic analysis can be performed on samples with activity concentrations up to 10 Ci/ml.

Direct counting of the initial 100:1 diluted (10 ml) sample allows analysis at coolant activity levels down to approximately 10 μCi/ml. In addition, the degassed, undiluted 10 ml sample can be used for analysis of samples in the 10^{-2} to 10^{-3} μCi/ml range.

Analysis that will be performed on postaccident samples includes isotopic analyses of all samples and analyses of liquid samples for boron, chlorides, and dissolved gas (total dissolved gas or hydrogen and oxygen). Containment atmosphere samples may also be analyzed for hydrogen, oxygen, and iodine content. Boron analyses of reactor liquid samples are performed using the carminic acid method that is capable of measuring boron concentrations in coolant down to 10 ppm. Chloride analysis is a scoping analysis of a 0.1-ml reactor liquid sample using the turbidimetric method.

Analysis of dissolved gases from reactor coolant samples is accomplished by measuring total dissolved gases using the pressure differential method. If a decision is made to obtain a grab sample of dissolved gases, analysis of the sample would be performed using a gas chromatograph.

Measurement of pH of reactor coolant samples will be performed using a semi-micro combination pH electrode that is capable of accurately measuring the pH of a small volume of liquid sample.

Isotopic analysis is expected to be accurate within at least a factor of two over a coolant activity range of approximately 10 μCi/ml to 10 Ci/ml. Background radiation levels will have no significant effect on the accuracy of the isotopic analysis. The DAEC counting facilities have shielding designed to maintain background radiation levels (due to contained and external airborne sources) below 2 mrem/hr following a postulated release of fission products equivalent to that described in Item II.B.2 of NUREG-0737. "Hot" samples are removed from the counting room so as not to affect background levels.

A postaccident sampling procedure is available for estimating the degree of core damage based on radionuclide data and taking into consideration the containment hydrogen levels and radiation levels as indicators of core damage.

12.3.4.3 Safety Evaluation

With the worst-case fission product release assumptions required by NUREG-0737, Item II.B.2, the reactor building must be considered inaccessible after a postulated loss-of-coolant accident. Therefore the postaccident sampling facilities are designed to enable personnel to obtain and analyze, under postaccident conditions, representative grab samples of reactor coolant and containment atmosphere gas without radiation exposure to any individual exceeding the guidelines of General Design Criterion 19 (i.e., 5-rem whole-body, 75-rem extremities).

The postaccident sampling facilities are designed to enable personnel to obtain samples and perform the required chemical and radiological analyses within 3 hr from the time a decision is made to sample.

The postaccident sampling facilities are powered from reliable ac power supplies which can be tied to the diesel generators in the event that offsite power is lost.

The probability of a leak of radioactive fluid from the postaccident sampling system is very small for the following reasons:

1. The system is included in the overall plant leak reduction program (Section 12.1.3.1). This will ensure that the system is periodically leak-tested and maintained to keep leakage at a minimum over the life of the plant.

2. The system is manually initiated and is not required to mitigate the consequences of a design-basis accident. Therefore, should there be any reason to suspect actual or potential leakage, the system can be leak-tested and/or maintenance can be performed prior to initiating system operation in a postaccident situation.

3. When the system is being used, it is not in continuous operation. Taking a sample requires operation of the system with radioactive fluid in the lines for only a short period of time. After the sample is taken, the lines are flushed with demineralized water.

4. It is not possible to align sample valves in the system such that an open flow path exists to a sample collection point with or without a sample vial in place. This is a basic design feature of the sample system.

The postaccident sampling system includes the following features to contain leakage outside the secondary containment:

1. The sample panel has provisions for collecting liquid leakage from within the panel. Fluid collecting in the sump can be isolated and discharged to the suppression pool.

2. The sample panel enclosure is maintained under negative pressure and continuously vented by an exhaust fan to the SGT inside secondary containment. This will prevent the leakage of gaseous radioactive material outside the secondary containment.

The postaccident sampling system can be operated only by direct operator action at the control panel located near the sample station. This area is equipped with an area radiation monitor to alert the local operator if there is a high radiation level caused by uncontained leakage from the sample station. In the event of a high radiation alarm, the operator can immediately isolate the system using local controls and then evacuate the area. If the area is evacuated without isolating the system, it can be isolated using controls in the control room.

REFERENCES FOR SECTION 12.3

1. General Electric Document C5474-SP-1, Revision 2, BWR Generic Post-Accident Sample System Design Requirements.

2. C. H. Lederer, J. M. Hollander, and I. Perlman, Table of Isotopes, John Wiley and Sons, Inc., 1967.

3. Office of Federal Register, Code of Federal Regulations, Title 10, as of January 1, 1970.

4. International Commission on Radiological Protection, Report of Committee II on Permissible Dose for Internal Radiation, Pergamon Press, 1960.

5. Not Used.

6. R. L. Walker and M. Grotenhuis, A Summary of Shielding Constants for Concrete, ANL-6443, Argonne National Laboratory, 1961.

7. U.S. Atomic Energy Commission, Mass Absorption Coefficients, ANL-5800, 1963.

8. J. K. Kircher and R. E. Bowman, Effects of Radiation on Materials and Components, Reinhold Publishing Corporation, 1964.

9. R. G. Jaeger et al., Engineering Compendium on Radiation Shielding.

10. Atomics International, GRACE II - For computing gamma-ray attenuation and heat generation (FORTRAN code).

11. Bechtel Corporation, FAIM - For computing neutron attenuation and heat generation (modified FORTRAN code).

12. Letter from D. B. Vassallo, NRC, to Lee Liu, Iowa Electric, Subject: NUREG-0737, Item II.B.2, Plant Shielding Modifications, dated August 2, 1983.

Table 12.3-1

POSTACCIDENT SHIELDING ANALYSIS DOSE RATES (mR/hr)

Time After Accident (hr)	Operational Support Center	Turbine Building
1	532	5370
8	574	6790
168 (1 wk)	29	840
720 (30 days)	4	96
Highest Rate	920	6790
Time, hr	0.0-0.5	8.0

Table 12.3-2

AREA RADIATION MONITORING SYSTEM
ENVIROMENTAL AND POWER SUPPLY DESIGN CONDITIONS

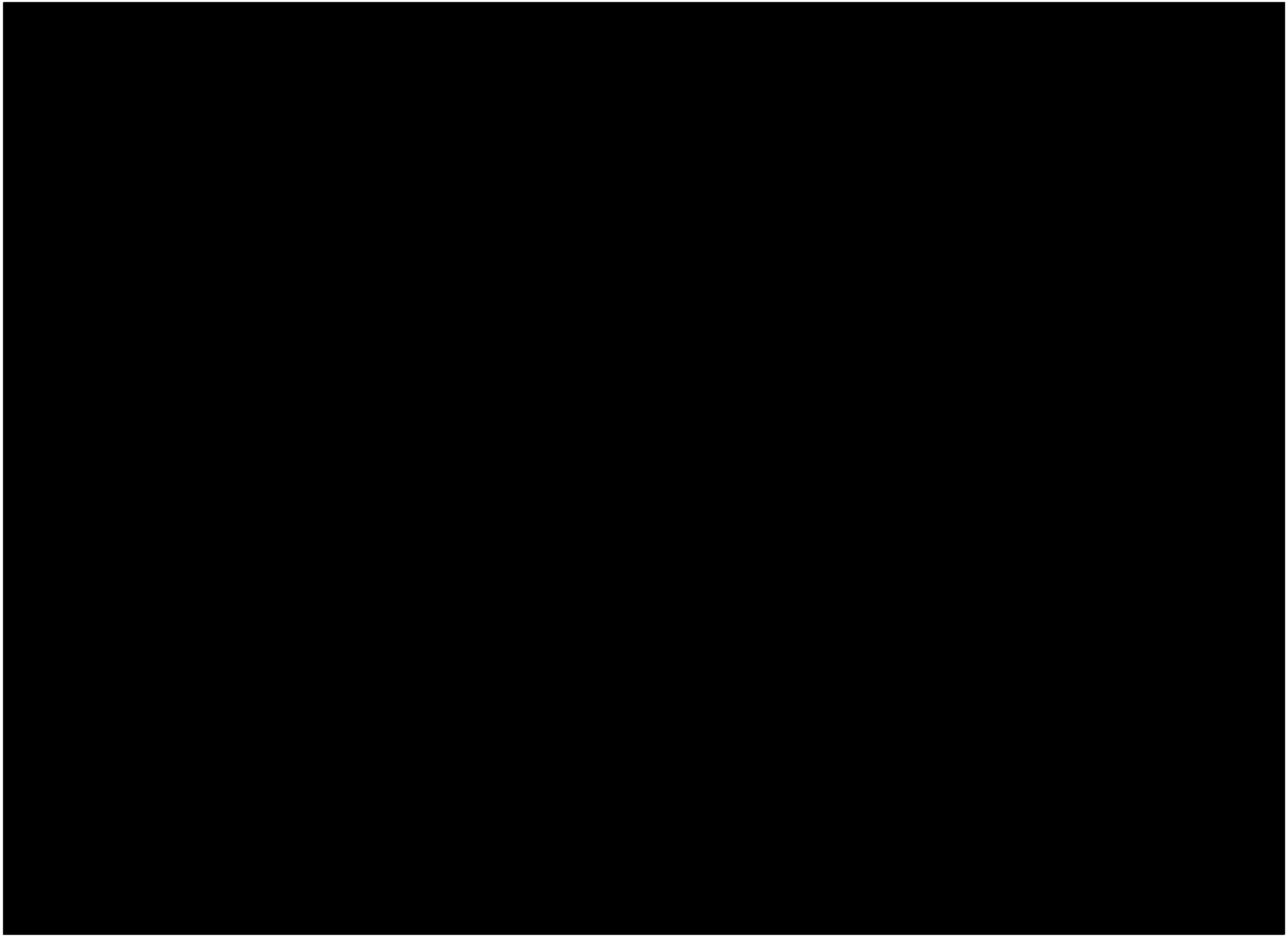
(Not applicable to LLRPSF)

	Sensor Location		Control Room	
Parameter	Design Center	Range	Design Center	Range
Temperature, °C	25	0 to 60	25	5 to +50
Relative humidity, %	50	20 to 100	50	20 to 90
Power	115V, 50/60 Hz (local alarm only)	±10% ±5%	115V 50/60 Hz	±10% ±5%

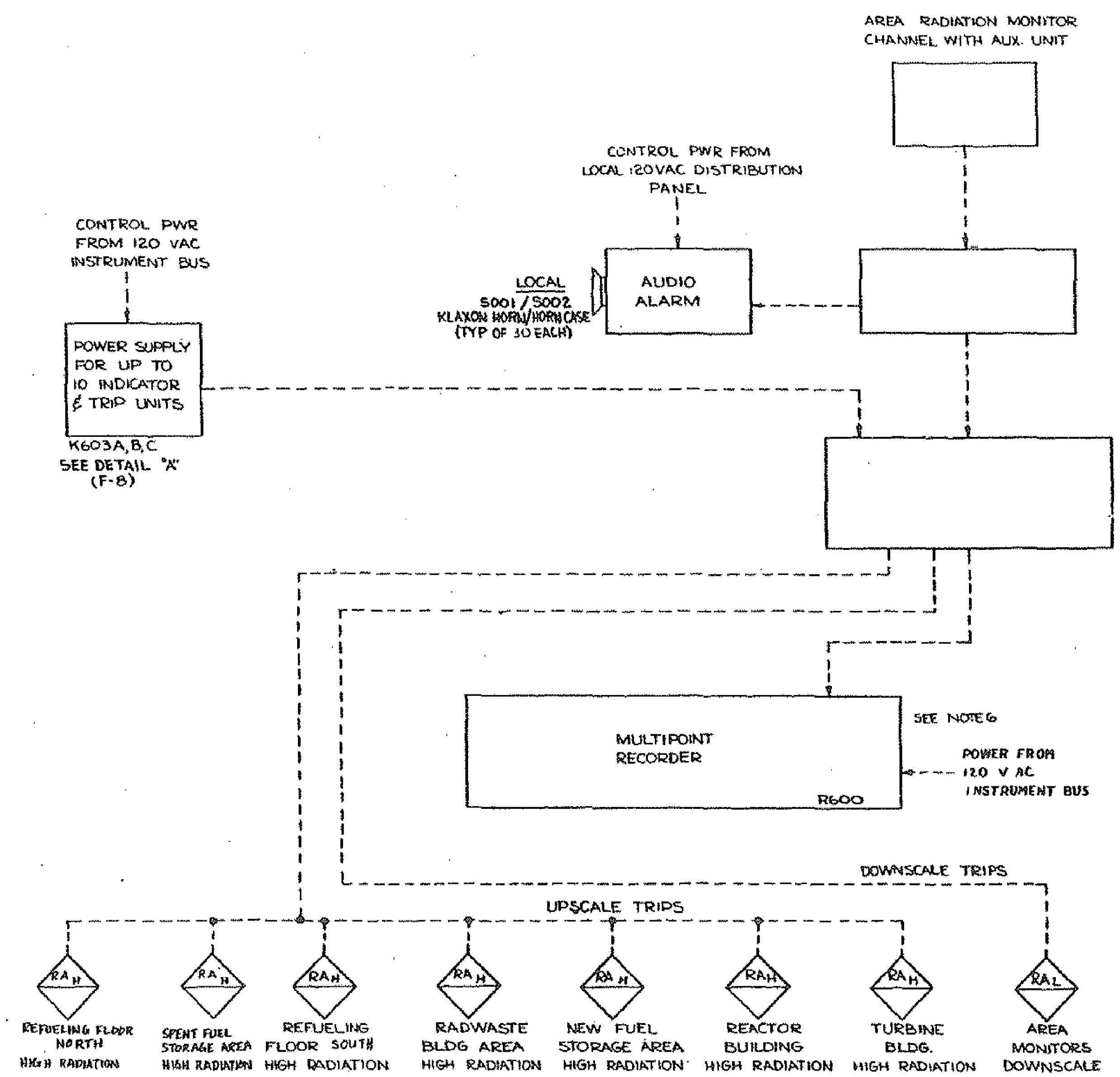
[redacted]

[redacted]

[redacted]

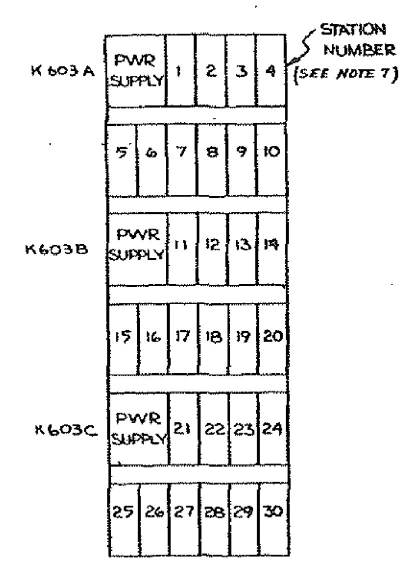


[redacted]



RANGE	QUANTITY (NOTE 1)			
MR/HR (NOTE 3)	SENSOR & CONVERTER	AUXILIARY UNIT	INDICATOR & TRIP UNIT	AUDIO ALARM
.01-100				
0.1-1000				
$1-10^4$				
$1.0-10^6$				

TABLE 1



INDICATOR & TRIP UNITS FOR EACH STATION
PANEL ~~D21-P600~~ 1H1-P605

DETAIL "A"

NOTES

1. RANGES TO BE CHOSEN BY PURCHASER WITHIN LIMITATIONS OF SCOPE OF SUPPLY.
2. FOR SUGGESTED SENSOR & CONVERTER LOCATIONS REFER TO DESIGN SPECIFICATION.
4. CHANNEL CALIBRATED BY USE OF CALIBRATION UNIT X003. (NOT SHOWN)
5. FOR LOCATION AND IDENTIFICATION OF INSTRUMENTS SEE INSTRUMENT DATA SHEET, LISTED IN MPL FOR EACH INSTRUMENT.
6. IN PROCESS RADIATION RECORDER VERTICAL BOARD. (1H1-P600)
7. STATION NUMBER REFERS TO PHYSICAL LOCATION IN PANEL. FOR CHANNEL NUMBER AND EXACT LOCATION OF INDICATOR AND TRIP UNITS IN 1C11 SEE BECH-E070.

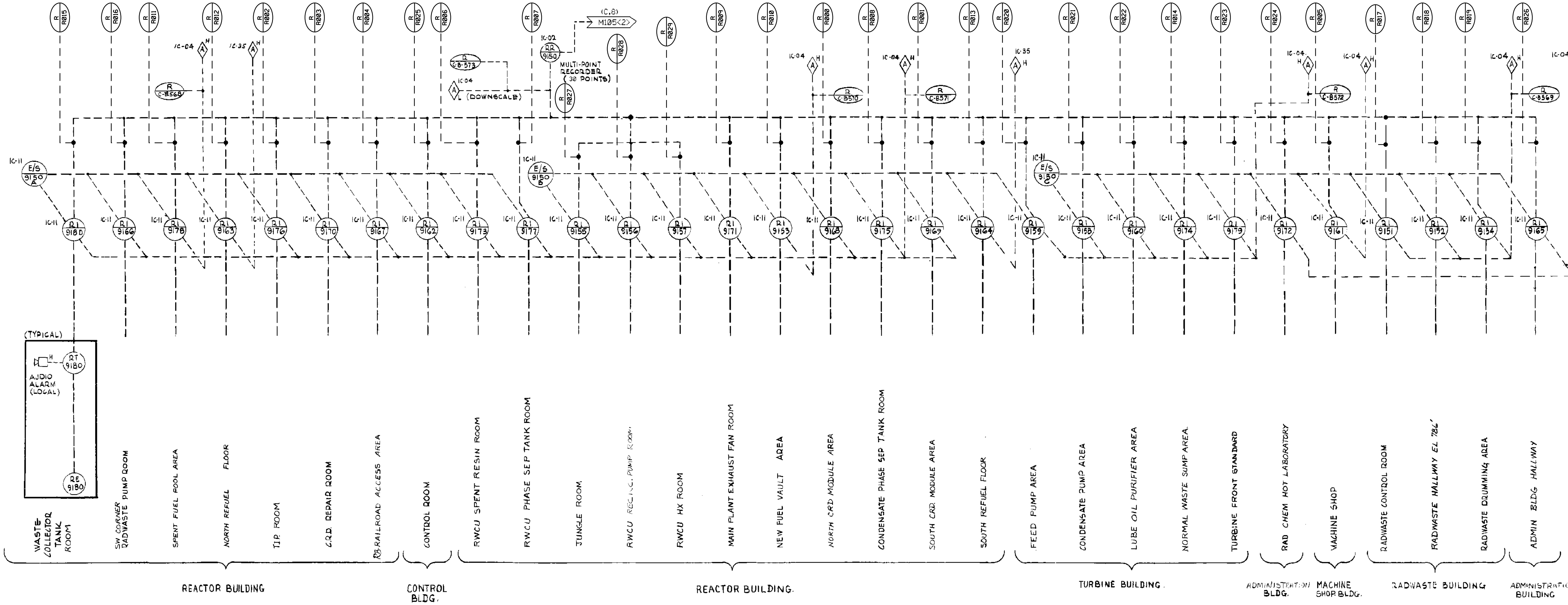
DUANE ARNOLD ENERGY CENTER
IES UTILITIES
UPDATED FINAL SAFETY ANALYSIS REPORT

IED,
AREA RADIATION MONITORING SYSTEM

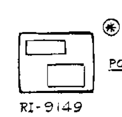
FIGURE 12.3-6

REVISION 17 - 10/03

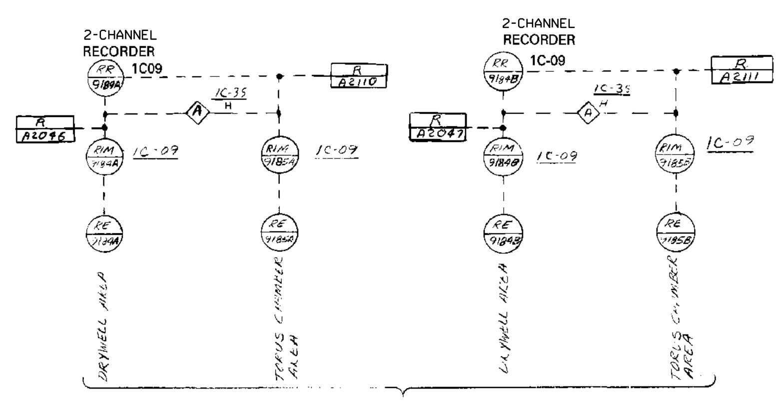
APED-D21-002 REV. 6



INSTRUMENT IDENTIFICATION NO.	SERVICE	COMPUTER POINT	LOCATION	AREA NO.	PRIM INST. LOCATION DWG NO M-	COORD'S
RE-9151	RADWASTE CONTROL ROOM	R017	RADWASTE BLDG.	RW BLDG	405-8	F-5
RE-9152	RADWASTE HALLWAY ELEV. 786'	R018	RADWASTE BLDG	SAME	405-8	D-5
RE-9153	NEW FUEL VAULT AREA	R010	REACTOR BLDG	5	405-5	D-6
RE-9154	RADWASTE DRUMMING AREA	R019	RADWASTE BLDG	RW BLDG	405-6	D-4
RE-9155	JUNGLE ROOM	R027	REACTOR BLDG.	4	405-4	F-7
RE-9156	RWCU RECIRC. PUMP ROOM	R028	REACTOR BLDG	5	405-3	D-6
RE-9157	RWCU HX ROOM	R029	REACTOR BLDG.	5	405-3	D-7
RE-9158	CONDENSATE PUMP AREA	R021	TURBINE BLDG.	3	405-1	C-4
RE-9159	FEED PUMP AREA	R020	TURBINE BLDG.	1	405-1	F-3
RE-9160	LUBE OIL PURIFIER AREA	R022	TURBINE BLDG.	1	405-1	F-2
RE-9161	MACHINE SHOP	R005	MACHINE SHOP	-	405-11	F-5
RE-9162	CONTROL ROOM	R025	CONTROL BLDG.	-	405-3	G-6
RE-9163	NORTH REFUEL FLOOR	R012	REACTOR BLDG	4	405-5	F-7
RE-9164	SOUTH REFUEL FLOOR	R013	REACTOR BLDG.	5	405-5	C-7
RE-9165	ADMIN BUILDING HALLWAY	R026	ADMINIST. BLDG	-	405-2	H-8
RE-9166	SW CORNER RADWASTE PUMP ROOM	R016	REACTOR BLDG.	5	405-1	C-7
RE-9167	RB RAILROAD ACCESS AREA	R004	REACTOR BLDG	5	405-2	C-7
RE-9168	NORTH C.R.D. MODULE AREA	R000	REACTOR BLDG	4	405-2	F-6
RE-9169	SOUTH C.R.D. MODULE AREA	R001	REACTOR BLDG	5	405-2	C-6
RE-9170	C.R.D. REPAIR ROOM	R003	REACTOR BLDG.	4	405-2	E-5
RE-9171	MAIN PLANT EXHAUST FAN ROOM	R009	REACTOR BLDG	4	405-4	F-5
RE-9172	RAD CHEM HOT LAB	R024	ADMINIST. BLDG	-	405-3	G-6
RE-9173	RWCU SPENT RESIN ROOM	R006	REACTOR BLDG.	4	405-3	E-7
RE-9174	NORMAL WASTE SUMP AREA	R014	TURBINE BLDG	3	405-1	B-2
RE-9175	CONDENSATE PHASE SEP TANK ROOM	R008	REACTOR BLDG	4	405-4	E-3
RE-9176	T.I.P. ROOM	R002	REACTOR BLDG.	5	405-2	D-5
RE-9177	RWCU PHASE TANK ROOM	R007	REACTOR BLDG.	4	405-3	F-7
RE-9178	SPENT FUEL POOL AREA	R011	REACTOR BLDG	5	405-5	D-6
RE-9179	TURBINE FRONT STANDARD	R023	TURBINE BLDG.	1	405-3	F-3
RE-9180	WASTE COLLECTOR TANK ROOM	R015	REACTOR BLDG.	5	405-1	C-6
RE-9184A	DRY WELL		REACTOR BLDG.	4	405-2	F-9.1
RE-9184B	DRY WELL		REACTOR BLDG.	5	405-2	G-7.1
RE-9185A	TORUS CHAMBER		REACTOR BLDG.	4	405-1	F 11.1
RE-9185B	TORUS CHAMBER		REACTOR BLDG.	5	405-1	J-7.1
RE-9186	TRUCK LOADING BAY ROOM 805		LOW LEVEL	LL	405-22	Aaei - 6.06
RE-9187	ACCESS AREA ROOM 800		LOW LEVEL	LL	405-22	Aa - 6.8
RE-9188	CRANE CONTROL ROOM 845		LOW LEVEL	LL	405-22	Aag - 5.9

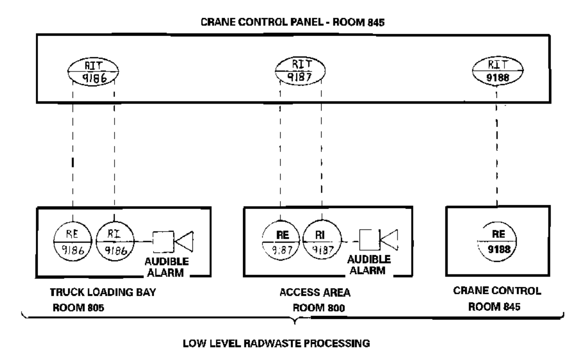


PORTABLE RADIATION CALIBRATION UNIT



SENSITIVITY TABLE

RANGE	INSTRUMENT #RE-XXXX	GE APED №
0.01 - 10^2 MR/HR, 4 DECADES	9161, 9162, 9163, 9164, 9165	G 11
0.1 - 10^3 MR/HR, 4 DECADES	9151, 9152, 9153, 9154 9158, 9159, 9160 9166, 9167, 9168, 9169, 9170 9171, 9172, 9173, 9174, 9175	G 12
1.0 - 10^4 MR/HR 4 DECADES	9176, 9177, 9178, 9179, 9180	G 13
1.0 - 10^6 MR/HR 6 DECADES	9155, 9156, 9157	G 16
1 - 10^7 R/HR 6 DECADES	9184A, 9184B, 9185A, 9185B	D21



DUANE ARNOLD ENERGY CENTER
NEXTERA ENERGY DUANE ARNOLD, LLC
UPDATED FINAL SAFETY ANALYSIS REPORT

AREA RADIATION MONITORING SYSTEM

FIGURE 12.3-7

[redacted]

[redacted]

12.4 DOSE ASSESSMENT

The (Original Licensing Basis) estimated yearly onsite exposure was 800 man-rem for the DAEC based on the following assumptions:

1. Approximately 150 permanently assigned plant personnel.
2. Two hundred to 300 contract maintenance personnel onsite for 2 months.
3. One maintenance outage of 2-months duration, which includes a refueling and a turbine overhaul.
4. One maintenance outage of 1-week duration.
5. Normal preventative maintenance accomplished.

The estimated (1997) yearly occupational exposure for operation and maintenance of DAEC is approximately 300 man-rem, based on the following assumptions:

1. Approximately 600 permanently assigned plant personnel.
2. 300 to 400 contract maintenance personnel onsite for 2 months to support scheduled outages.
3. One maintenance outage of 45 days duration, which includes a refueling and a turbine overhaul.
4. One maintenance outage of 1-week duration.
5. Normal preventative maintenance and plant operation accomplished.

12.5 HEALTH PHYSICS PROGRAM

12.5.1 ORGANIZATION

The Radiation Protection Manager is head of the DAEC Radiation Protection Department, which includes the health physics organization. The qualifications, responsibility, and authority of the Radiation Protection Manager are set forth in Section 13.1.2.2.11.

The qualifications of the Radiation Protection Manager meet or exceed the requirements of Regulatory Guide 1.8, September 1975.

The organization of the Radiation Protection Department is discussed in Section 13.1.2.1. Radiation protection training is discussed in Section 13.2.

Supervisors within the Radiation Protection Department provide supervisory oversight of technical and field health physics activities.

12.5.2 EQUIPMENT, INSTRUMENTATION, AND FACILITIES

12.5.2.1 Health Physics and Laboratory Analysis Radiation Monitors

12.5.2.1.1 Power Generation Design Basis

Portable radiation survey instruments are available for the measurement of the alpha, beta, gamma, and neutron radiations expected in normal operations and emergencies. Appropriate instruments and auxiliary equipment are available to detect and measure radioactive contamination on surfaces, in air, and in liquids.

12.5.2.1.2 System Description

Various survey meters, particulate sample counters, and associated analytical equipment are furnished in order for health physicists to monitor radiological conditions in the plant. Contamination monitoring of personnel, tools and equipment is provided at exits from potentially contaminated areas.

Primary dosimeters for record dose (thermoluminescent dosimeters or equivalent) and a secondary dosimeter (self-reading dosimeter or electronic dosimeter) pre provided to and worn by persons in situations as required by 10CFR20. Reading of the secondary dosimeters are entered into a computerized system to track and control current personnel exposure on real-time basis.

Alarming dosimeters are available for use by personnel in special high dose situations.

Measured data from the primary dosimeters are used to replace the accumulated self-reading dosimeter data periodically.

Laboratory radiation measuring instruments are provided for alpha, beta, and gamma radiations and for gaseous, liquid, and solid samples.

Secondary calibration sources and check-test sources for the various instruments are provided.

12.5.2.1.3 Inspection and Testing

Proper operation of radiation safety equipment is typically checked with built-in testing circuits and/or radiation sources. All measuring instruments are periodically calibrated with radioactive or electronic calibration sources by qualified personnel in accordance with approved procedures. Standards for the calibration of equipment are traceable to the National Institute of Standards Technology.

12.5.2.2. Respiratory Protection Equipment

The use of respiratory equipment is discussed in the DAEC Health Physics Procedures and Instructions.

The types of respiratory protection equipment available at the DAEC include the following:

1. Self-contained breathing apparatus

   These are the pieces of equipment typically stored at emergency response locations because of their qualification for use in IDLH (Immediately Dangerous to Life and Health) environments. The fresh air supply is contained in the tank and provided to the wearer through a regulator, hose, and face piece.

2. Full-Face Respirators

   These are the full-face mask using air-line supplied atmosphere, or using attached canisters to filter and/or absorb contaminants from the air being inhaled by the wearer. The DAEC does not use air-line supplied-air respirators that operate in the demand mode.

12.5.3 PROCEDURES

Procedures for personnel radiation protection consistent with the requirements of 10 CFR 20 have been prepared and approved and are adhered to for all operations involving personnel radiation exposure.

The DAEC Health Physics Procedures define the policies and procedures of the DAEC for operations involving work with ionizing radiation and radioactive materials. The primary objectives of these policies and procedures are as follows:

1. Controlling the radiation exposure (both internal and external) of individuals to a level of "as low as is reasonably achievable" below the established permissible limits.

2. Containment of radioactive materials, as necessary, to maintain releases to unrestricted areas "as low as is reasonably achievable," below the established limits.

The DAEC program for controlling internal exposure includes the use of administrative controls, surface and airborne survey data, protective clothing, application of engineering controls, respiratory equipment, and direct surveillance of work activities. A bioassay device (whole body counter) is available to evaluate any internal exposure of personnel.

Special bioassays and/or whole-body counting may be performed for the following cases:

1. Any time periodic whole-body counting checks indicate a significant internal exposure.

2. Possible exposure of personnel following an abnormal release of radioactivity.

3. Any other cases of suspected inhalation or ingestion of radioactive materials.

4. Medical uptake of radioactive isotopes for diagnostic purposes.

Included in the DAEC program is a program to ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions, including training of personnel, procedures for monitoring and provisions for maintenance of sampling and analysis equipment.

The control of exposure of personnel to radiation is achieved by shielding design and by limiting the amount of time individuals are subjected to measured levels of radiation. Access to radiologically controlled areas is controlled by a number of physical and administrative controls.

It is the task of the plant health physics staff to support plant personnel in their daily work routines in order to control exposures. Area survey schedules based on the radiological hazards associated with particular jobs are developed prior to performing work. Surveys are performed by health physics technicians.

A minimum of one member from the Radiation Protection Department is assigned to each operating shift and is responsible for implementing radiation protection procedures.

The DAEC has procedures for decontamination/repair activities. Decontamination of areas or equipment may be accomplished by using a variety of equipment or techniques, such as the following:

1. Vacuuming.
2. Electropolishing.
3. Ultrasonic cleaning.
4. Scraping/sanding.
5. Water/detergent wash.
6. Solvent, alcohol, or acid cleaning.

Decontamination activities are conducted in accordance with a Radiation Work Permit and are supervised by qualified health physics repair personnel.