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DENTON,H.R. Document Control Branch (Document Control Desk)

SUBJECT: Forwards response to RA Gilbert 870113 request for addl info re emergency response capability,per open items identified in interim rept on conformance to Rev 2 of Reg Guide 1.97. Selected pages from updated FSAR also encl.

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Iowa Electric Light and Power Company

March 31, 1987

NG-87-1032

Mr. Harold Denton
Director, Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Washington, D.C. 20555

Re: Duane Arnold Energy Center, Docket No. 50-331
Subject: Emergency Response Capability - Conformance to R.G. 1.97, Rev. 2: Response to Request for Additional Information
Reference: R.A. Gilbert letter to L. Liu dated January 13, 1987
File: A-370

Dear Mr. Denton:

Attached is our response to the open items identified in the interim report on conformance to Regulatory Guide 1.97 by the Duane Arnold Energy Center.

The referenced letter requested that this response be submitted within 60 days of receipt of that letter. However, the interim report was not enclosed with the original transmittal of that letter. We subsequently received a copy of the interim report on January 30, 1987. Therefore, this response is being provided to you within 60 days from that latter date.

Please contact this office if you require further information on this matter.

Very truly yours,

William C Rothert for

Richard W. McGaughy
Manager, Nuclear Division

RWM/TLF/pc

Attachment: 1. Response to Open Items: Conformance to Regulatory Guide 1.97
2. FSAR references.

cc: T. Forker
L. Liu
L. Root
R. Gilbert
NRC Resident Office
Commitment Control No. 870017

A003
1/1

Attachment 1
NG-87-1032

IOWA ELECTRIC LIGHT AND POWER COMPANY

DUANE ARNOLD ENERGY CENTER

RESPONSE TO OPEN ITEMS:

CONFORMANCE TO REGULATORY GUIDE 1.97

1. <u>Neutron Flux</u>

Background:

Regulatory Guide 1.97 (reference 1) recommends that Category 1 instrumentation be provided to monitor neutron flux for reactivity control (variable B-1). Although our existing neutron monitoring instrumentation does not meet Category 1 requirements, we maintain (reference 2, Issue 1) that the existing instrumentation does meet the intent of the guide.

Open Item:

Section 3.3.1 of the interim report (reference 4) states,

> *"Based on our review, we conclude that the existing instrumentation is acceptable for interim operation. The licensee should follow industry development of this equipment, evaluate newly developed equipment, and install Category 1 instrumentation to cover the recommended range when it becomes available."*

Response:

We are participating in the BWR Owner's Group (BWROG) activities on this subject. We will evaluate the conclusions and recommendations of that group and take appropriate actions. We will inform you of our specific plans and schedules through the semi-annual updates of our Integrated Plan.

2. Radiation Exposure Rate

Background:

Regulatory Guide 1.97 recommends that the radiation exposure rate inside buildings or areas, e.g., auxiliary building, fuel handling building, secondary containment, which are in direct contact with primary containment where penetrations and hatches are located (variable C-14) should be monitored over the range of 10^{-1} to 10^{4} R/hr for an indication of a breach. In addition, it recommends that the radiation exposure rate inside buildings where access is required to service equipment important to safety (variable E-3) should be monitored over the range of 10^{-1} to 10^{4} R/hr for the detection of significant releases, for release assessment, and for long-term surveillance.

We maintain that even though our existing area radiation monitors have ranges which are lower than specified by Regulatory Guide 1.97, they are adequate because the detection of a breach in the containment, the detection of significant releases, release assessment, and long-term surveillance are performed by the extended range airborne effluent radiation monitoring system. In addition, access is not required to any area of the secondary containment in order to service safety-related equipment in a post-accident situation, and the safety of any reentry would be established by post-accident sampling of the containment atmosphere and by portable radiation survey instruments.

Open Item:

Section 3.3.6 of the interim report states,

> *"The licensee has not shown analysis of radiation levels expected for the monitor locations. The licensee should show that the existing radiation exposure rate monitors have ranges that encompass the expected radiation levels in their locations."*

Response:

Radiation levels expected in various areas of the plant during normal operation are outlined in Section 12.3.1.1 of our FSAR (reference 5 - copy attached). Detailed maps of these areas are provided in Figures 12.3-1 through 12.3-5 of our FSAR. Our area radiation and airborne radioactivity monitoring instrumentation is described in Section 12.3.3 of our FSAR. The location and range of each instrument in our area radiation monitoring system is described in Figure 12.3-7 of our FSAR. The purpose of our area radiation monitoring system is to warn of abnormal gamma radiation levels within the plant. Each instrument is selected to cover the full range of radiation levels expected during normal operation plus at least two decades above the maximum rate expected during normal operation. They are not intended to cover the full range of post-accident radiation levels. If the radiation level exceeds the range of a particular instrument under post-accident conditions, that instrument would still provide a warning of abnormal radiation levels at the monitor location by reading upscale.

The extended range airborne effluent radiation monitoring system described in Section 11.5.9 of our FSAR is a more effective system for the detection of a breach in the containment, the detection of significant releases, release assessment, and long-term surveillance. This system covers the range of 10^{-7} to 10^{5} uCi/cc for noble gases. This is three decades greater than the 10^{-6} to 10^{3} uCi/cc range recommended in Regulatory Guide 1.97 for noble gas effluent monitors.

The postaccident sampling system described in Section 12.3.4 of our FSAR is capable of collecting and analyzing gas samples from the secondary containment (reactor building) atmosphere.

Access is not required to any area of the secondary containment to service safety-related equipment in the post accident situation. Our procedures do not permit reliance upon area radiation monitors to establish the safety of an area for reentry. The safety of such reentry would be established by sampling the secondary containment atmosphere and using appropriate portable radiation survey instruments.

Therefore, we believe that our existing area radiation monitoring instruments, when used in conjunction with our extended range airborne effluent radiation monitoring system, the postaccident sampling system, and portable radiation survey instruments are adequate to meet the intent of Regulatory Guide 1.97 concerning the detection of a breach in the containment, the detection of significant releases, release assessment, and long-term surveillance.

3. Drywell Atmosphere Temperature

Background:

Regulatory Guide 1.97 recommends Category 2 instrumentation to monitor the drywell atmosphere temperature over a range from 40 to 440°F (variable D-7). Our existing instrumentation covers the range of 0 to 350°F. The interim report found this range to be acceptable.

Open Item:

Section 3.3.10 of the interim report states:

> *On page 43 of [reference 2], the licensee identifies this as Class 1E instrumentation that meets Category 1 recommendations. In Table 1 of [reference 2], variable D-7, this instrumentation is identified as Category 3, but meeting the recommendations of Category 2 instrumentation except for environmental qualification.*
>
> *The licensee should clarify the characteristics of the instrumentation for this variable. If there is a deviation from environmental qualification, it should be addressed in accordance with 10 CFR 50.49."*

Response:

The description of this variable in Table 1 of reference 2 is correct. The statement on page 43 of reference 2 which identifies the instrumentation as Class 1E is premature.

Although the temperature elements and the cables used to connect them have been environmentally qualified, the cabling used in this system does not meet the divisional separation requirements of Class 1E. We have committed (reference 3) to reroute the cables to achieve divisional separation during our cycle 9/10 refueling outage which is currently scheduled for the fall of 1988. When that work is complete, the system will meet the requirements for Class 1E.

4. SLCS Storage Tank Level

Background:

Regulatory Guide 1.97 recommends Category 2 instrumentation to monitor the level in the Standby Liquid Control System (SLCS) Storage Tank (variable D-18).

Open Item:

Section 3.3.13 of the interim report states:

> *Regulatory Guide 1.97 recommends Category 2 instrumentation for this variable. The licensee's instrumentation meets the Category 2 recommendations except in the area of environmental qualification. The licensee states that Category 3 instrumentation is sufficient for this variable because it does not serve a primary safety function, it is not a key variable (but it is the key variable to show that SLCS flow is occurring), it is not needed to ensure design basis behavior, and it does not indicate the need for contingency actions. This justification is not acceptable.*
>
> *Environmental qualification has been clarified by the Environmental Qualification Rule, 10 CFR 50.49. The licensee should therefore provide the required justification for this deviation from Regulatory Guide 1.97 or provide instrumentation that is environmentally qualified in accordance with the provisions of 10 CFR 50.49 and Regulatory Guide 1.97."*

Response:

The SLCS level instrumentation is located in a mild environment as defined by 10 CFR 50.49. 10 CFR 50.49 states, "(c) Requirements for ... (3) environmental qualification of electrical equipment important to safety located in a mild environment are not included within the scope of this section."

5. Residual Heat Removal Heat Exchanger Outlet Temperature

Background:

Regulatory Guide 1.97 recommends Category 2 instrumentation to monitor the residual heat removal (RHR) heat exchanger outlet temperature (variable D-20).

Open Item:

Section 3.3.14 of the interim report states:

> *The licensee states that Category 3 instrumentation is sufficient for this variable because it does not serve a primary safety function, it is not a key variable, it is not needed to ensure design basis behavior, and it does not indicate the need for contingency actions. However, this instrumentation is needed to determine quantitatively, the heat removed from containment. Therefore, this justification is not acceptable.*
>
> *Environmental qualification has been clarified by the Environmental Qualification Rule, 10 CFR 50.49. The licensee should therefore provide the required justification for this deviation from Regulatory Guide 1.97 or provide instrumentation that is environmentally qualified in accordance with the provisions of 10 CFR 50.49 and Regulatory Guide 1.97."*

Response:

RHR heat exchanger outlet temperature is defined as a Type D variable by Regulatory Guide 1.97. Type D variables are defined as "those variables that provide information to indicate the operation of individual safety systems and other systems important to safety. These variables are to help the operator make appropriate decisions in using the individual systems important to safety in mitigating the consequences of an accident."

In addition to RHR heat exchanger outlet temperature, there are numerous other parameters that indicate the operation of the RHR system. These include RHR pump motor current, RHR pump discharge pressure, RHR system flow, and valve position indications.

There is no great need to determine quantitatively the amount of heat removed from the containment. However, the temperature and the amount of heat left in the containment is important, and there are numerous variables available to indicate that.

The RHR temperature elements and cables in the RHR heat exchanger outlet temperature instrumentation system are environmentally qualified.

For the reasons stated above, we believe that our existing RHR heat exchanger outlet temperature instrumentation is adequate to meet the intent of Regulatory Guide 1.97.

6. Cooling Water Flow to ESF System Components

Background:

Regulatory Guide 1.97 recommends Category 2 instrumentation for monitoring cooling water flow to engineered safety feature (ESF) system components (variable D-22).

Open Item:

Section 3.3.16 of the interim report states:

> *The licensee states that Category 3 instrumentation is sufficient for this variable because it does not serve a primary safety function, it is not a key variable, it is not needed to ensure design basis behavior, and it does not indicate the need for contingency actions. We find this justification inadequate. This instrumentation does provide a leading indication of failure of safety-related equipment.*
>
> *Environmental qualification has been clarified by the Environmental Qualification Rule, 10 CFR 50.49. The licensee should therefore provide the required justification for this deviation from Regulatory Guide 1.97 or provide instrumentation that is environmentally qualified in accordance with the provisions of 10 CFR 50.49 and Regulatory Guide 1.97."*

Response:

The instrumentation for this system is located in a mild environment as defined by 10 CFR 50.49. 10 CFR 50.49 states, "(c) Requirements for ... (3) environmental qualification of electrical equipment important to safety located in a mild environment are not included within the scope of this section."

7. Reactor Building or Secondary Containment Area Radiation

Background:

Regulatory Guide 1.97 recommends that reactor building or secondary containment area radiation (variable E-2) should be monitored over the range of 10^{-1} to 10^4 R/hr for detection of significant releases, for release assessment, and for long-term surveillance. We maintain that even though our existing area radiation monitors have ranges which are lower than specified by Regulatory Guide 1.97, they are adequate to warn of abnormal gamma radiation levels, and because other systems are available which are better suited to the detection of significant releases, for release assessment, and for long-term surveillance.

Open Item:

Section 3.3.18 of the interim report states,

> *"The licensee states that the instrumentation for this variable is not needed, as the noble gas effluent monitors are more useful and practical in detecting or assessing primary containment leakage. This is due to the radioactivity in the fluids flowing in the emergency core cooling systems piping, and the large number of piping and electrical penetrations and hatches between the primary containment and the reactor building. For the Mark I containment, the recommended range is 10^{-1} to 10^4 R/h. The licensee has not shown how the recommended range is met by the noble gas effluent monitors."*

Response:

The adequacy of our area radiation monitor system for detection of significant releases, for release assessment, and for long-term surveillance was addressed in our response to Open Item 2.

The extended range airborne effluent radiation monitoring system described in Section 11.5.9 of our FSAR is a more effective system for the detection of significant releases, release assessment, and long-term surveillance than the area radiation monitor system. The airborne effluent radiation monitoring system covers the range of 10^{-7} to 10^5 uCi/cc for noble gases. This is three decades greater than the 10^{-6} to 10^3 uCi/cc range recommended in Regulatory Guide 1.97 for noble gas effluent monitors.

Therefore, we believe that our existing reactor building or secondary containment area radiation monitoring instruments, when used in conjunction with our extended range airborne effluent radiation monitoring system, the postaccident sampling system, and portable radiation survey instruments are adequate to meet the intent of Regulatory Guide 1.97 concerning the detection of significant releases, release assessment, and long-term surveillance.

REFERENCES

1. Instrumentation for Light-Water-Cooled Nuclear Power Plants to assess Plant and Environs Conditions During and Following an Accident, Regulatory Guide 1.97, Revision 2, NRC, Office of Standards Development, December 1980.

2. Iowa Electric Light and Power Company letter, R.W. McGaughy to H.R. Denton, NRC, "Regulatory Guide 1.97," July 3, 1985, NG-85-2423, File: A-370.

3. Iowa Electric Light and Power Company letter, R.W. McGaughy to H.R. Denton, NRC, "Plans and Schedules for Implementation of Plant Instrumentation Upgrades for Regulatory Guide 1.97 and Generic Letter 84-23," October 16, 1985, NG-85-4481, File: A-107d, A-370.

4. NRC letter, R.A. Gilbert to L. Liu, "Emergency Response Capability -Conformance to R.G. 1.97, Rev. 2: Request for Additional Information," January 13, 1987.

5. Updated Final Safety Analysis Report, Duane Arnold Energy Center, Iowa Electric Light and Power Company, 1986.

Attachment 2
NG-87-1032

SELECTED PAGES FROM

UPDATED FINAL SAFETY ANALYSIS REPORT

DUANE ARNOLD ENERGY CENTER

IOWA ELECTRIC LIGHT AND POWER COMPANY

gamma-spectrum analyzed for gross beta. Air samples with a gross beta activity in excess of 10 times the yearly mean of control samples are analyzed by gamma spectrometry. The activated charcoal filters are also analyzed by gamma spectrometry for I-131. 4

Air Radiation Dosimeters

The normal airborne emissions from the plant consist predominantly of noble gases that are not sampled by the air samplers previously discussed. 4

Integral beta and gamma radiation measurements are made with TLDs placed at 44 locations. Each TLD used contains one Teflon wafer impregnated with 25% $CaSO_4$:Dy Phosphor. Each wafer has four sensitive areas. The locations are specified in the Offsite Dose Assessment Manual. 4

Duplicate sets of TLDs are placed at each sampling point around the plant. One TLD from each location is processed quarterly. The second (emergency) dosimeter at a location will be processed when requested by the DAEC. If the regular dosimeter is lost or damaged, the corresponding emergency dosimeter will be processed and reported. 4

11.5.8 POSTACCIDENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

The postaccident liquid and gas sampling systems are discussed in Section 12.3.4. 2

Containment postaccident radiation monitors are discussed in Section 12.3.3.3.4.

11.5.9 EXTENDED RANGE AIRBORNE EFFLUENT RADIATION MONITORING SYSTEM

11.5.9.1 System Description

The extended range airborne radiation monitor system was installed to satisfy the requirements of NUREG-0737, Item II.F.1, Attachments 1 and 2, and Regulatory Guide 1.21. The system is a Kaman Sciences Corporation digital radiation monitoring system (DRMS) which acquires and displays radiation data from the turbine building and reactor building vents and the offgas stack. This system is in addition to the offgas stack radiation monitoring system described in Section 11.5.3. The DRMS consists of two major systems, a data acquisition system and a display and control system. See Figure 11.5-5. The data acquisition system consists of a serial chain of 10 radiation monitor units. Each monitor unit consists of a multichannel radiation monitor and an Intel microprocessor acting as an interface control between the monitors and the display and control system. The chain of 10 monitor units or data loop terminates at a minicomputer. Asynchronous communications multiplexers interface the ends of the data loop to the minicomputer. The display and control system interrogates all 3

monitors from one end of the data loop and then alternately from
the other end to ensure the integrity of the data links between
monitor units. Alternating the direction of interrogation also
ensures that all monitors are still in communication with the
display and control system even after a single failure of the
communications data loop. The interrogation message is sent to
each monitor at least once every 2 sec to obtain the current
radiation level and at time intervals to acquire 1-min, 10-min,
1-hr, and 1-day averages of the radiation level. The operator has
the ability to review the status of all monitors through color- 3
coded displays generated on the CRT. A set of keyboard commands
allows the operator to call up displays, acknowledge alarms,
initiate certain monitor activities, and enter monitor parameters
and other data base changes. Critical conditions reported by the
monitors cause an audible alarm at the operator's station. Alarm
setpoints are derived in the manner described in the DAEC Offsite
Dose Assessment Manual (Reference 1). All significant events are
recorded on a printed log by the hard copy unit in the technical
support center as the events occur. Display and control units are
located in the control room and the radiochemistry laboratory for
interactive data retrieval.

There are 10 monitor units, a normal and an accident range monitor in each of the following: turbine building vent, reactor building vent numbers 1, 2, and 3, and the offgas stack.

The DRMS is supplied by the 120-V ac distribution system. If
primary power is lost, a battery supplies 10 to 12 V to the micro- 4
processors so that memory and calibration data are not lost while
primary power is off.

11.5.9.2 Monitor Characteristics

The radiation monitors have up to four types of channels for detecting and measuring radiation as follows:

1. Gas detector - measures the gross beta activity level of radioisotopes in gaseous form present in air streams.

2. Particulate detector - measures the gross beta or gamma activity level of radioisotopes in solid particulate form present in air streams.

3. Iodine detector - measures gamma activity of iodine radioisotopes in air streams.

4. Combined particulate-iodine sampler - collects particulate or iodine radioactive substance from an air stream for laboratory analysis.

The characteristics of the particulate, iodine, and noble gas detectors are given in Table 11.5-3 for each of the four effluent vents and the offgas stack.

In addition to the turbine building vent radiation monitor units, the turbine building effluent radiation monitoring system includes 2 two-pen recorders located in the turbine building. One recorder records flow and the second recorder records normal and accident noble gas activity of the turbine building vent.

Table 11.5-3

CHARACTERISTICS OF EXTENDED RANGE AIRBORNE EFFLUENT MONITOR SYSTEM

Characteristics	Particulate	Iodine	Noble Gas	
REACTOR BUILDING VENTS 1, 2, AND 3				
Measurement type	Gross beta	Iodine gamma	Gross beta	
Detector type	Beta scintillator	Gamma scintillator	Beta scintillator Geiger-Mueller counter (2 required)	
Dynamic range, $\mu Ci/cm^3$	10^{-10} to 10^{-4}	10^{-10} to 10^{-4}	10^{-7} to 10^{-5}	
Minimum detectable concentration (MDC), $\mu Ci/cm^3$	1×10^{-11}	1×10^{-10}	2×10^{-7}	
Calibration isotope	Sr^{90}	Ba^{133}	Mixed noble gases[a] Xe^{133}	
Maximum response time at MDC	10 min	10 min	1 min	
Minimum filter efficiency	99% > 0.3 μ	95%	Not applicable	3
Filter type	HV-LB 5211	SAI	Not applicable	
Analog outputs	--	--	2 required	
Remote indication	--	--	1 required	
TURBINE BUILDING VENT				
Measurement type	Collection only	Collection only	Gross beta	
Detector type	Not applicable	Not applicable	Beta scintillator Geiger-Mueller counter (2 required)	
Dynamic range, $\mu Ci/cm^3$	Not applicable	Not applicable	10^{-7} to 10^5	
MDC, $\mu Ci/cm^3$	Not applicable	Not applicable	2×10^{-7}	
Calibration isotope	Not applicable	Not applicable	Mixed noble gases[a] Xe^{133}	
Maximum response time at MDC	Not applicable	Not applicable	1 min	
Minimum filter efficiency	99% > 0.3 μ	95%	Not applicable	
Filter type	HV-LB 5211	SAI	Not applicable	
Analog outputs	--	--	2 required	
Remote indication	--	--	1 required	

[a]Mixed noble gases for beta scintillator and Xe^{133} for Geiger-Mueller counter.

Table 11.5-3 Sheet 2 of 2

CHARACTERISTICS OF EXTENDED RANGE AIRBORNE EFFLUENT MONITOR SYSTEM

Characteristics	Particulate	Iodine	Noble Gas	
OFFGAS STACK				
Measurement type	Collection only	Collection only	Gross beta	
Detector type	Not applicable	Not applicable	Beta scintillator Geiger-Mueller counter (2 required)	
Dynamic range, $\mu Ci/cm^3$	Not applicable	Not applicable	10^{-7} to 10^5	3
MDC, $\mu Ci/cm^3$	Not applicable	Not applicable	2×10^{-7}	
Calibration isotope	Not applicable	Not applicable	Mixed noble gases[a] Xe^{133}	
Maximum response time at MDC	Not applicable	Not applicable	1 min	
Minimum filter efficiency	99% > 0.3 μ	95%	Not applicable	
Filter type	HV-LB 5211	SAI	Not applicable	
Analog outputs	--	--	2 required	
Remote indication	--	--	1 required	

[a]Mixed noble gases for beta scintillator and Xe^{133} for Geiger-Mueller counter.

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 FACILITY DESIGN FEATURES

12.3.1.1 Radiation Zones

All plant areas are identified by radiation zones according to their expected length of occupancy by personnel. Each zone is assigned a maximum dose rate that ensures that the exposure of personnel to radiation will be less than guidelines contained in 10 CFR 20, based on 50 weeks per year. Radiation zones are shown in Figures 12.3-1 through 12.3-5.

The shielding is designed such that, during operation at design power, the dose rate in each area is as follows:

Zone	Design Dose Rate (mrem/hr)	Description
A	≤0.5	Continuous occupancy outside controlled areas.
B	≤1.0	40 hr/week occupancy in controlled areas.
C	≤6.0	Controlled occupancy to 10 hr/week.
D	≤12.0	Controlled occupancy to 5 hr/week.
E	≤100	Controlled limited occupancy for short periods of less than 1 hr/week.
F	>100	Normally inaccessible. Occupancy for short periods during emergencies.

Specific examples of the various zones are as follows:

Zone	Location
A	General plant yard areas.
	General administrative areas.
	Employee service areas outside controlled areas.
	Visitor's area.
	Main plant control room.
	Cable spreading room.

Zone	Location
B	Motor-generator rooms.
	Radwaste control room.
	General plant full-time-occupancy corridor and work areas.
C	Passageways and valve areas outside turbine plant auxiliary equipment cubicles.
	Passageways and drumming-baling operational areas in radwaste building.
	Area directly above drywell shielding plugs.
	Demineralizer mixing and charging areas.
D	Turbine operating floor outside the high-pressure turbine exclusion area.
	Radwaste sample and surge tank room.
	General pump and manifold areas.
E	Turbine-generator exclusion area.
	Valve area adjacent to condensate demineralizer.
F	Drywell, demineralizer, and filter compartments.

12.3.1.2 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 wall 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. |2

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 10 CFR 20 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 whole body.

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 correponding 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. The most limiting of these is the loss-of-coolant accident that releases fission products as described in TID-14844.

12.3.2.4 Shield Design Calculations

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 Shielding Material

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. |1

12.3.2.6 Description of Plant Shielding

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. The sacrificial shield is approximately 2 ft 3 in. thick.

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. More than 8 ft of concrete thickness 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 concrete walls that are up to 3 ft 9 in. thick, 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. A maximum of 6 ft of concrete thickness is used for radiation protection at the sides and bottom of the storage pool. A minimum cover of 10 ft of water above the fuel assemblies is maintained to protect plant personnel during fuel storage and transfer operations.

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 3
is pulled into the drive mechanism. The shield walls consist of high-density grout-filled masonry and the roof slab consists of reinforced concrete and structural steel. Access control is provided by a lockable door.

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. The shield wall is constructed of concrete
blocks with all cells filled with grout. Viewing windows have 1
been installed to allow viewing the sample sink and gauge panel. An access gate has been provided to control access to this area. 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.

A 3-in. steel shield at the high-pressure end of the turbine provides protection for limited access for instrument checks. A 6-in.-thick steel shield between the turbine and generator 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 Other Plant Areas

12.3.2.6.5.1 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.

12.3.2.6.5.2 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 to the whole body.

12.3.2.6.5.3 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.

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

12.3.2.7 Design Review of Plant Shielding for Postaccident Operations

| 1

A postaccident (DBA) shielding evaluation has been performed for the DAEC. The dose rates calculated for the main control room, technical support center, operational support center, and turbine building are shown in Table 12.3-1.

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. 2

12.3.2.8 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 2

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 2

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

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 2

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 | 2

Monitors are located in appropriate areas within the reactor
building, turbine building, control building, radwaste building,
and administration building. Annunciation and indication are | 2
provided in the main control room. 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 at the air intake 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 | 2
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. | 2

12.3.3.3.5 Improved In-Plant Iodine Instrumentation Under Accident Conditions

In response to NUREG-0578, Section 2.1.8.c, Iowa Electric 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. 2

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 below. 2

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. | 4

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. | 4

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. | 4

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. | 4

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. | 4

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 4
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 JP-5 and JP-13 outside of the 4
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. 4

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 Nonseismic 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 estimates 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. 4

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 2
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.

Gas sample lines are heat-traced to prevent precipitation of moisture and resultant loss of iodine in the sample lines. Redundant, non-Class 1E heat tracing has been provided which is sized to hold a line temperature of 250°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 RE-8771 and monitor RI-8771 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 RE-8708 and monitor RI-8708 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 RE-8742 and monitor RI-8742 are provided to monitor the deposition of radioactive material on the filter cartridges during gas sampling operations. 2

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 2
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. 2

12.3.4.2.5 Chemical Analysis

A postaccident sampling chemical analysis laboratory is located in the DAEC administration building at elevation 786 ft-0 in. 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. 2

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. 2

12.3.4.2.6 Radiological Analysis

A postaccident sampling radiological analysis laboratory (counting room) is located in the DAEC administration building at elevation 757 ft-6 in. 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. | 2

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.

4

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. 4

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

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 SGTS inside secondary containment. This will prevent the leakage of gaseous radioactive material outside the secondary containment. 2

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.

12.3.4.4 Inspection and Testing

Operability and surveillance requirements for the postaccident sampling system are contained in the Technical Specifications.

REFERENCES FOR SECTION 12.3

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

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. U.S. Atomic Energy Commission, Calculation of Distance Factors for Power and Test Reactor Sites, TID-14844, 1962.

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.

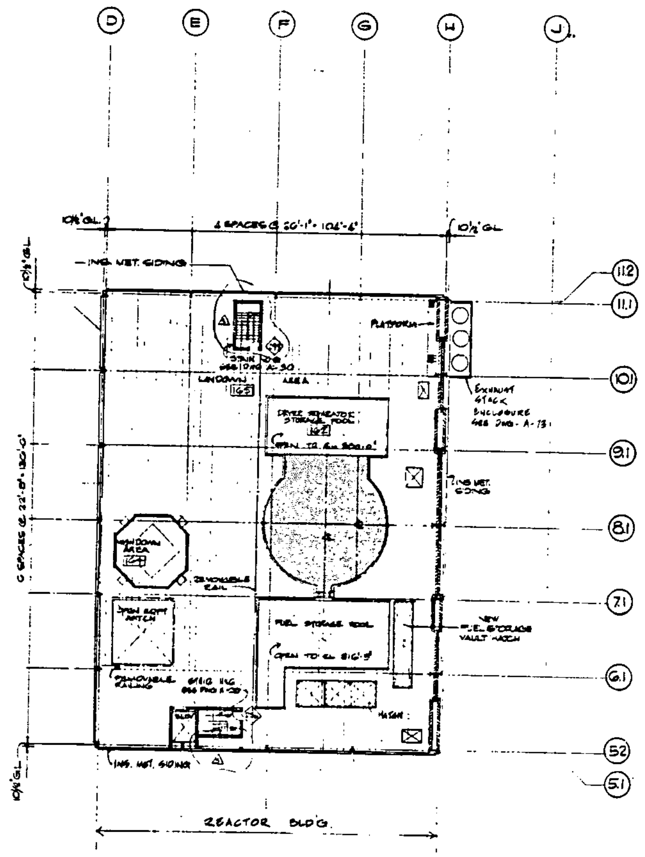
8. J. K. Kircher and R. E. Bowman, Effects of Radiation on Materials and Components, Reinhold Publishing Corporation, 1964.

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. 2



PLAN EL. 855'-0"

RADIATION ZONES
(DURING NORMAL OPERATION)

ZONE	DESIGNATION	DOSE RATE	mrem/hr	DESCRIPTION
	A	≤	0.5	CONTINUOUS OCCUPANCY OUTSIDE CONTROLLED AREAS
	B	≤	1.0	CONTINUOUS OCCUPANCY IN CONTROLLED AREAS
	C	≤	6	CONTROLLED OCCUPANCY TO 10 HRS/WEEK
	D	≤	12	CONTROLLED OCCUPANCY TO 5 HRS/WEEK
	E	≤	100	CONTROLLED LIMITED OCCUPANCY FOR SHORT PERIODS OF LESS THAN 1 HR/WEEK
	F	>	100	NORMALLY INACCESSIBLE OCCUPANCY LIMITED AND CONTROLLED BY DIRECT RADIATION READING

● - LOCKED DOOR

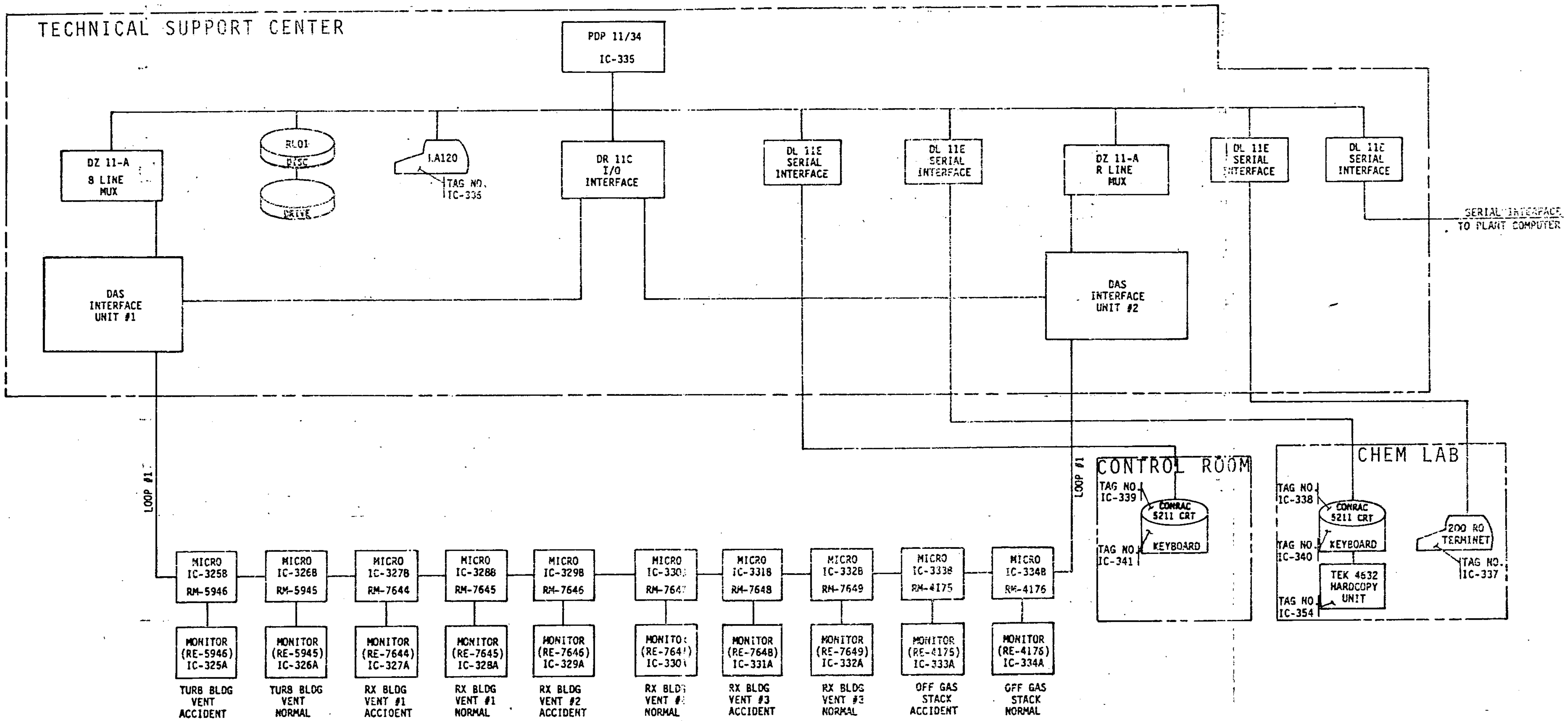
NOTE: SEE DWG A-57 FOR NOTES

A-40 Rev 3

DUANE ARNOLD ENERGY CENTER
IOWA ELECTRIC LIGHT & POWER COMPANY
UPDATED FINAL SAFETY ANALYSIS REPORT

Radiation Zones - Elevation 855 Ft

Figure 12.3-5



8704060487-01

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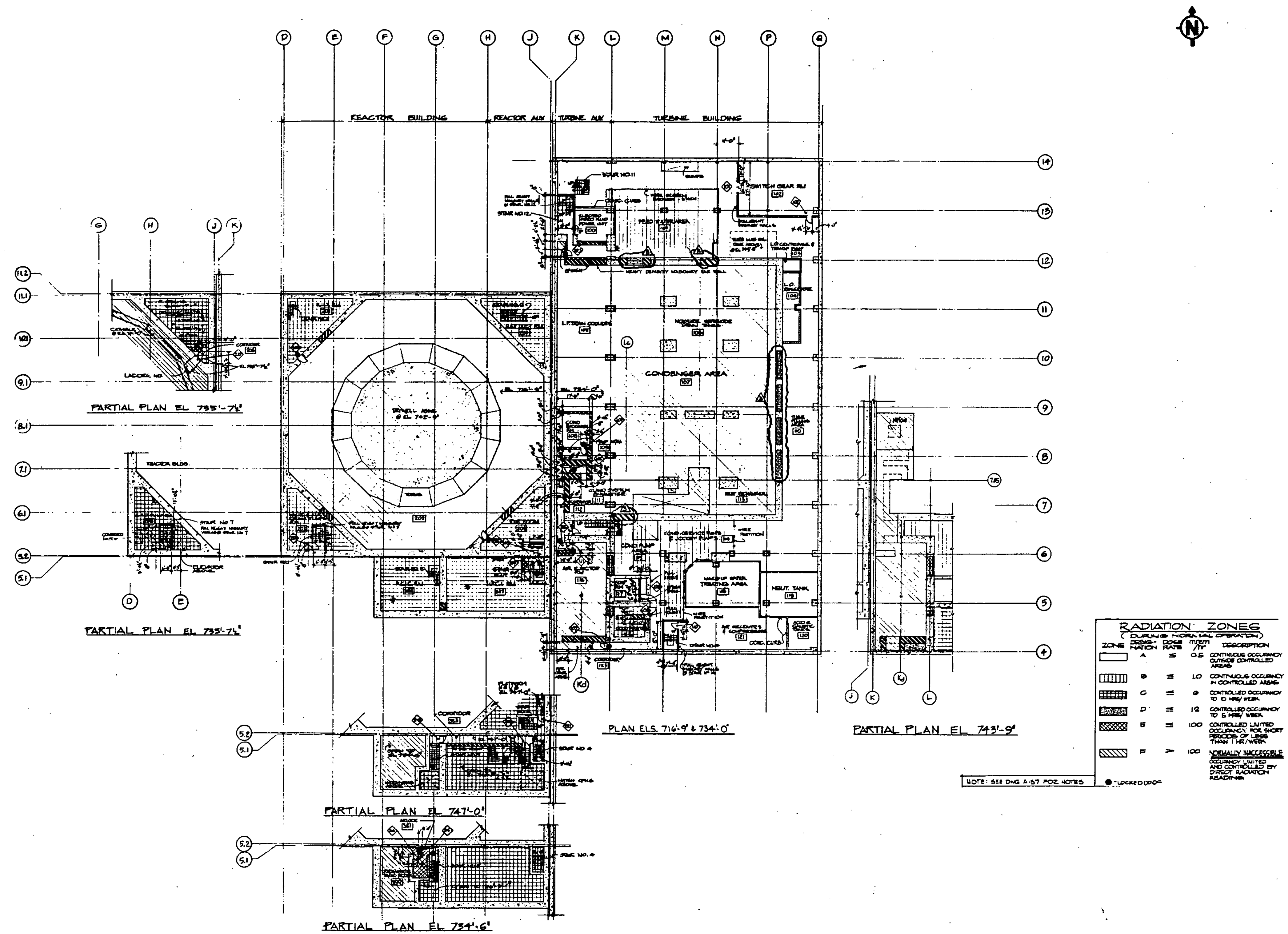
Extended Range Airborne Effluent Radiation Monitoring System

Figure 11.5-5



Pending Update to Include:

DCP 1207 Appendix R Modification - Fire Barriers (Doors, Walls, Seals, and Fire Dampers)

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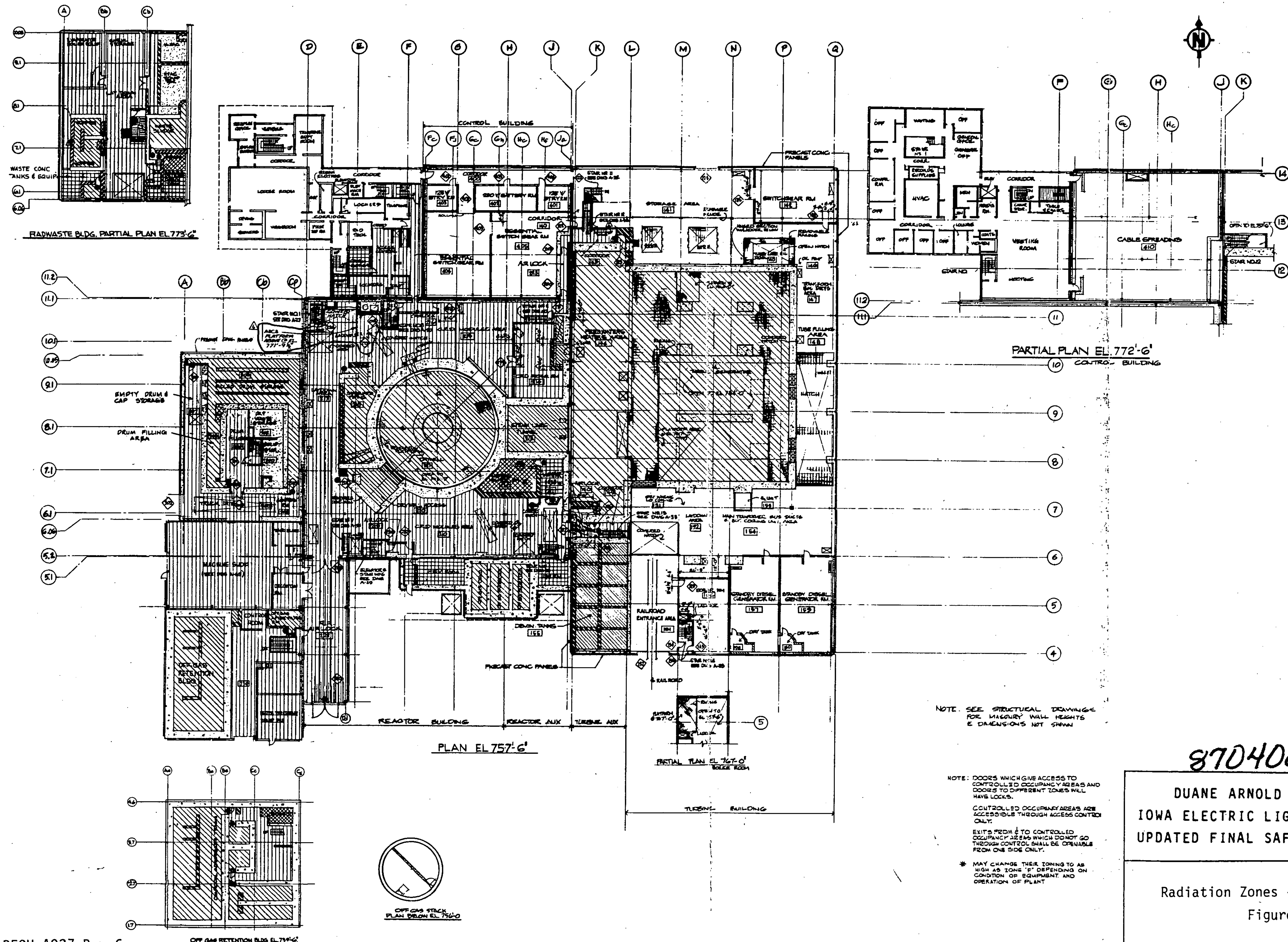
DUANE ARNOLD ENERGY CENTER
IOWA ELECTRIC LIGHT & POWER COMPANY
UPDATED FINAL SAFETY ANALYSIS REPORT

Radiation Zones - Elevation 735 Ft
Figure 12.3-1

BECH-A036 Rev 3



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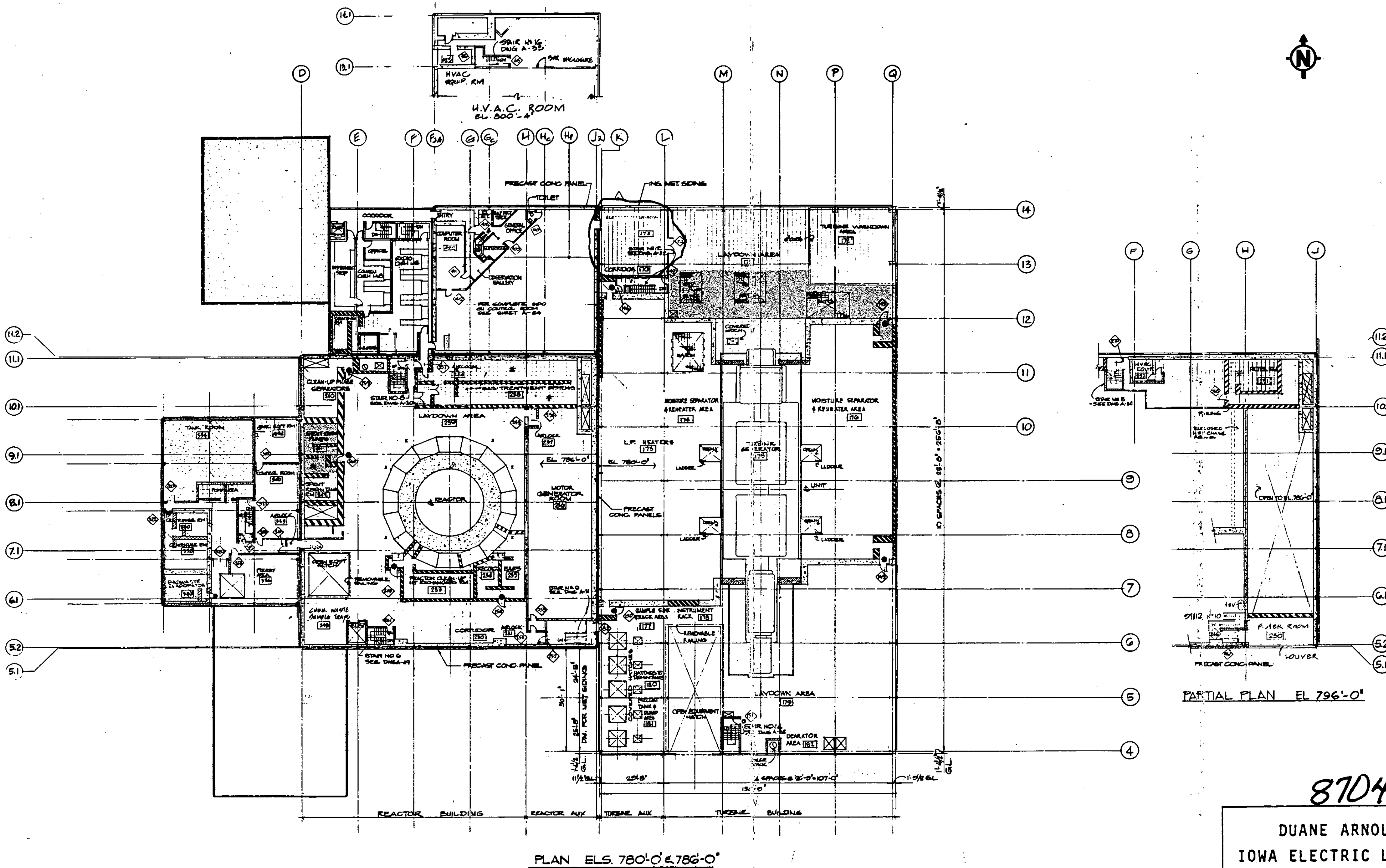
DUANE ARNOLD ENERGY CENTER
IOWA ELECTRIC LIGHT & POWER COMPANY
UPDATED FINAL SAFETY ANALYSIS REPORT

Radiation Zones - Elevation 757 Ft

Figure 12.3-2

BECH-A037 Rev 6



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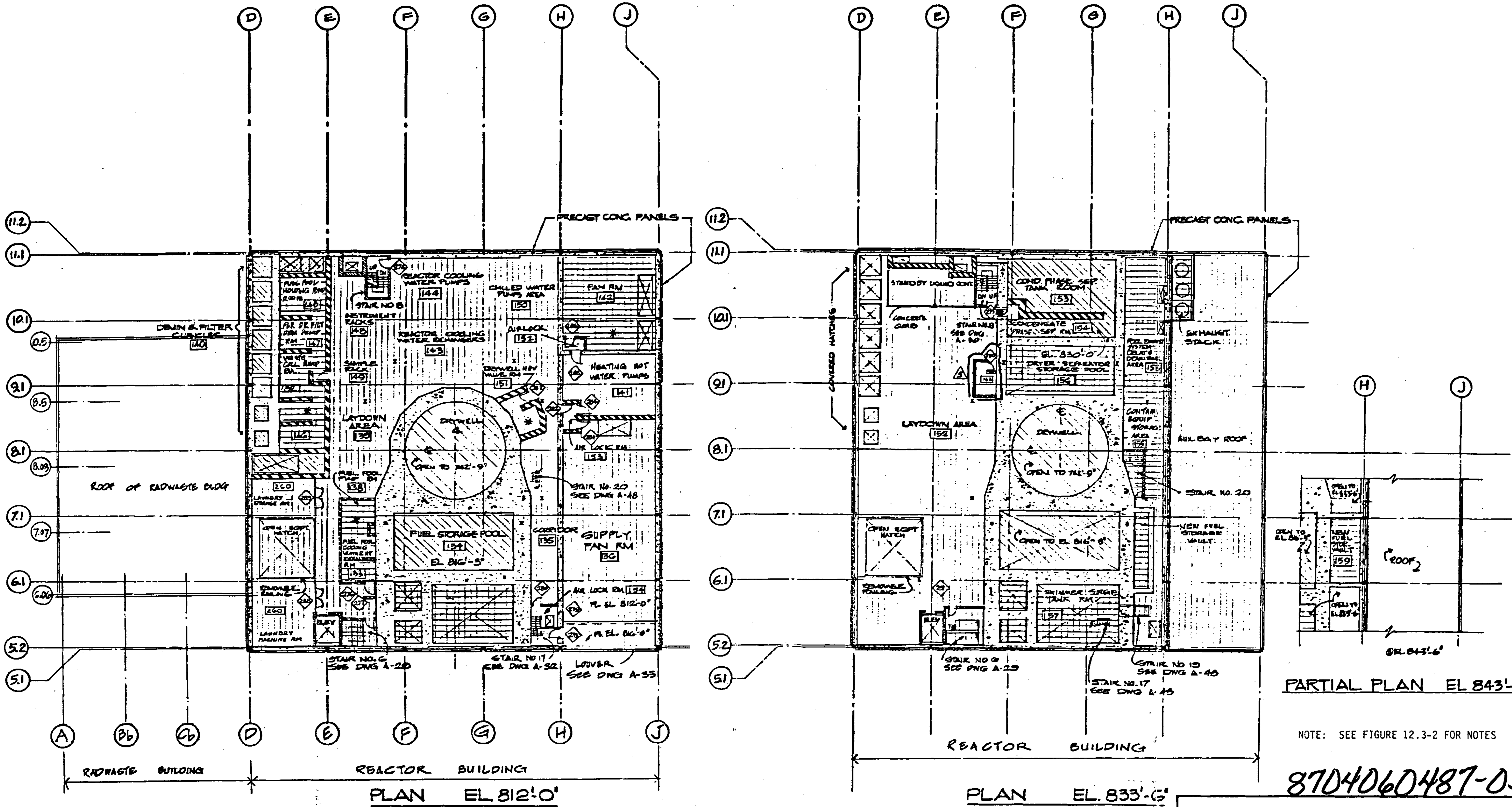
DUANE ARNOLD ENERGY CENTER
IOWA ELECTRIC LIGHT & POWER COMPANY
UPDATED FINAL SAFETY ANALYSIS REPORT

Radiation Zones - Elevation 780 Ft

Figure 12.3-3

A-38 Rev 6



NOTE: SEE FIGURE 12.3-2 FOR NOTES

8704060487-05

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IOWA ELECTRIC LIGHT & POWER COMPANY
UPDATED FINAL SAFETY ANALYSIS REPORT

Radiation Zones -
Elevations 812 and 833 Ft

Figure 12.3-4

TI APERTURE CARD

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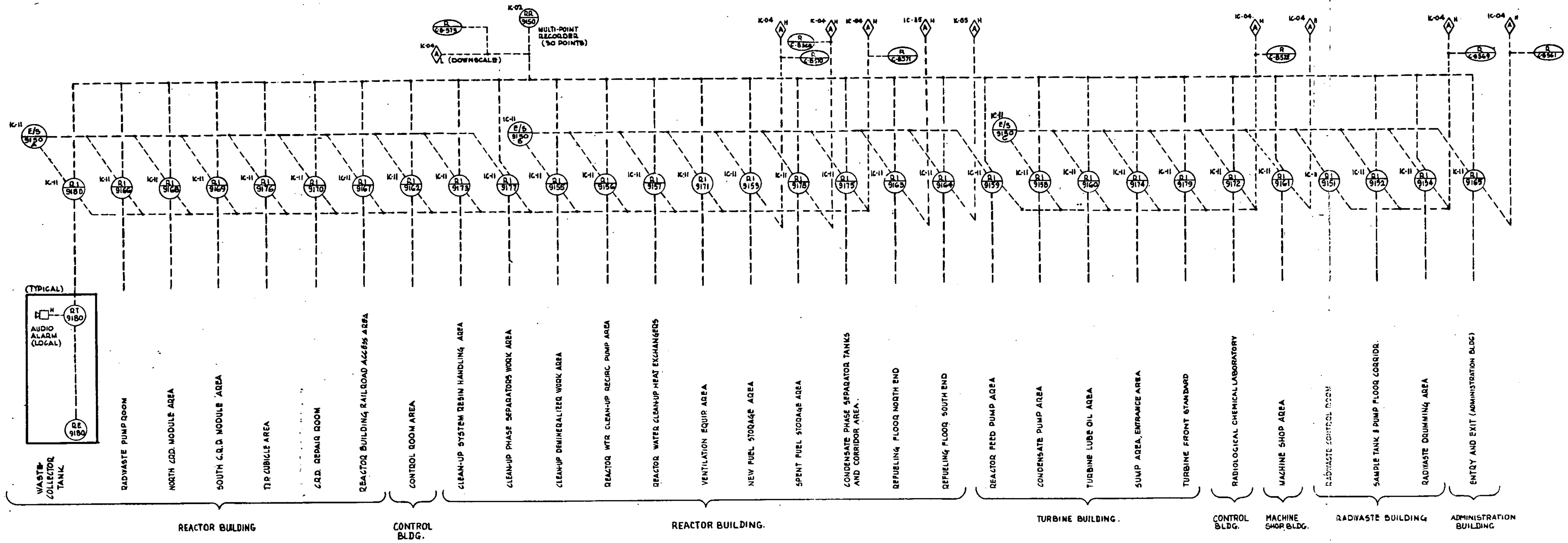
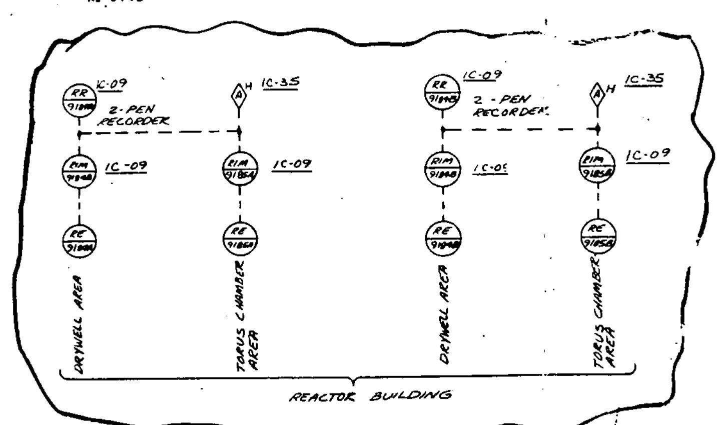
A-39 Rev 6



	INSTRUMENT IDENTIFICATION NO.	SERVICE	LOCATION	AREA NO.	PRIM. INST. LOCATION DWG NO. M-	COORD'S
1	RE-9151	RADWASTE CONTROL ROOM	RADWASTE BLDG.	RW BLDG	405-8	F-5
2	RE-9152	SAMPLE TANK & PUMP FLOOR CORRIDOR	RADWASTE BLDG	SAME	405-8	D-5
3	RE-9153	NEW FUEL STORAGE AREA	REACTOR BLDG.	5	405-5	D-6
4	RE-9154	RADWASTE DRUMMING AREA	RADWASTE BLDG.	RW BLDG	405-6	D-4
5	RE-9155	CLEAN-UP DEMINERALIZER WORK-AREA	REACTOR BLDG.	4	405-4	F-7
6	RE-9156	REACTOR WTR CLEAN-UP RECIRC. PUMP AREA	REACTOR BLDG	5	405-3	D-6
7	RE-9157	REACTOR WTR CLEAN-UP HEAT EXCHANGERS	REACTOR BLDG.	5	405-3	D-7
8	RE-9158	CONDENSATE PUMPS AREA	TURBINE BLDG.	3	405-1	C-4
9	RE-9159	REACTOR FEED PUMPS AREA	TURBINE BLDG.	1	405-1	F-3
10	RE-9160	TURBINE LUBE OIL AREA	TURBINE BLDG.	1	405-1	F-2
11	RE-9161	MACHINE SHOP AREA	MACHINE SHOP	-	405-11	F-5
12	RE-9162	CONTROL ROOM AREA	CONTROL BLDG	-	405-3	G-6
13	RE-9163	REFUELING FLOOR - NORTH END	REACTOR BLDG	4	405-5	F-7
14	RE-9164	REFUELING FLOOR - SOUTH END	REACTOR BLDG.	5	405-5	C-7
15	RE-9165	ENTRY AND EXIT (ADMINISTRATION BUILDING)	ADMINIST. BLDG	-	405-2	H-8
16	RE-9166	RADWASTE PUMP ROOM	REACTOR BLDG.	5	405-1	C-7
17	RE-9167	REACTOR BLDG. RAILROAD ACCESS AREA	REACTOR BLDG	5	405-2	C-7
18	RE-9168	NORTH C.R.D. MODULE AREA	REACTOR BLDG	4	405-2	F-6
19	RE-9169	SOUTH C.R.D. MODULE AREA	REACTOR BLDG.	5	405-2	C-6
20	RE-9170	C.R.D. REPAIR ROOM	REACTOR BLDG.	4	405-2	E-5
21	RE-9171	VENTILATION EQUIPMENT AREA	REACTOR BLDG	4	405-4	F-5
22	RE-9172	RADIOLOGICAL CHEMICAL LAB.	CONTROL BLDG.	-	405-3	G-6
23	RE-9173	CLEAN-UP SYS. RESIN HANDLING AREA	REACTOR BLDG.	4	405-3	E-7
24	RE-9174	SUMP AREA ENTRANCE	TURBINE BLDG.	3	405-1	B-2
25	RE-9175	CONDENSATE PHASE SEPARATOR TANKS & CORRIDOR AREA.	REACTOR BLDG	4	405-4	E-3
26	RE-9176	T.I.P. CUBICLE AREA	REACTOR BLDG.	5	405-2	D-5
27	RE-9177	CLEAN-UP PHASE SEPARATORS WORK AREA	REACTOR BLDG.	4	405-3	F-7
28	RE-9178	SPENT FUEL STORAGE AREA	REACTOR BLDG.	5	405-5	D-6
29	RE-9179	TURBINE STANDARD FLOOR AREA	TURBINE BLDG.	1	405-3	F-3
30	RE-9180	WASTE COLLECTOR TANK AREA	REACTOR BLDG.	5	405-1	C-6
31	RE-9184A	DRY WELL	REACTOR BLDG	4	405-2	F-9.1
32	RE-9184B	DRY WELL	REACTOR BLDG	5	405-2	G-7.1
33	RE-9185A	TORUS CHAMBER	REACTOR BLDG	4	405-1	F-11.1
34	RE-9185B	TORUS CHAMBER	REACTOR BLDG	5	405-1	J-7.1

6



6

SENSITIVITY TABLE

RANGE		GE APED Nº
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
10^2 - 10^6 MR/HR 5 DECADES	9155, 9156, 9157	G 17
1 - 10^7 R/HR 6 DECADES	9184A, 9184B	021

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Pending Update to Include:

DCP 1261 Safety Parameters Display System Electrical Installation

8704060487-06

DUANE ARNOLD ENERGY CENTER
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Area Radiation Monitoring System
Figure 12.3-7

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Revision 4 - 6/86