SECTION 10
ACCIDENT ASSESSMENT

1.0 General

1.1 Emergency Action Level Determination

The plant parameter and instrument values used to identify an emergency class are provided in Plan Attachment 5-1 and 5-2.

2.0 Accident Assessment and Instrumentation - Salem Generating Station

There are several monitoring systems used to support emergency planning activities at Salem Generating Station. The primary systems utilized are listed below.

- Radiation Monitoring System (See Radiation Monitoring System Manuals) and CBD DE-CB. RM - 0064(Q)
- Safety Parameters Display System (SPDS)
- Reactor Coolant Sampling System

2.1 Radiological Monitoring Instrumentation - Salem Generating Station

The radiological monitors consist of process radiation monitors, effluent radiation monitors and area radiation monitors, (see Table 10-1). The system continuously displays and/or records the radiation levels in key areas. The Unit 1 Radiation Monitoring System (RMS) is a predominately analog system while the Unit 2 Radiation Monitoring System (RMS) is predominately a digital system. Both systems have been modified to comply with the recommendations of NUREG 0578. The Unit 2 RMS consists of individual sensors with independent micro processors (MPs), which are able to perform a variety of tasks. The MPs can convert count rate pulses from the individual channels into various engineering units or factor in flow rate information to provide information to the operator to aid accident assessment. The information is fed to two mini computers, which are in a master-slave configuration for redundancy. The operator, by keying in certain commands, is able to display selected radiation monitoring channels in a particular elevation on his display screen. The information may be trended for pre-selected time periods, as required. The listing of the radiation monitoring channels, which may be used to assess an accident, is provided in Table 10-1.

Permanent monitor channels are not always available at a location of interest and the use of portable area monitors may be required during an accident. Unit 2 RMS uses a "communication loop" in which the radiation monitors in the "field" communicate to the computer via a loop of wire.

2.2 Process and Area Monitors

In order to provide the operators with essential information on plant conditions during an emergency, various plant processes are continuously monitored. Many of these processes involve Limiting Conditions for Operations (LCO) and are controlled by the Technical Specifications. If an LCO parameter "goes out of specification" it requires the operators to implement the action required by the associated action statement. The intent of this action is to take corrective measures under abnormal conditions before a situation becomes more serious. These parameters would be monitored closely during an accident for assessment purposes. These process indications that are monitored are also listed in Table 10-1.

2.3 Gaseous Release Path Monitoring

In addition to the main plant vent, a monitored vent, the other potential major release points from the plant during an accident are the main steam dump valves, pilot operated relief valves and the turbine driven auxiliary feed water pump. Procedures have been developed to monitor these potential release pathways and perform the necessary dose assessment.

2.4 Reactor Coolant and Containment Air Sampling - Salem Generating Station

Reactor coolant and containment gaseous activity sampling (normal and high activity/emergency samples) will be performed using station procedures and the normal day-to-day sampling systems. The plant vent, which is the final release point, is continuously monitored by the RMS for noble gases. The iodine cartridge can be physically removed and taken into a laboratory for analysis by a multi-channel analyzer available in the station (Hope Creek will provide backup analysis). There are also provisions provided in the plant vent for extracting a grab sample.

Analysis of reactor coolant and containment air samples provides detailed information on the status of the reactor core. These samples can be used to provide confirmation of a loss of the fission product barriers.

3.0 Accident Assessment Instrumentation - Hope Creek Generating Station

There are several monitoring systems used to support emergency planning activities for the Hope Creek Generating Station.

The plant computer systems and their functions are described in Final Safety Analysis Report (UFSAR). Specifically, the primary systems utilized to support emergency planning activities are:

- Control Room Integrated Display Systems (CRIDS)(UFSAR Section 7.5.1.3.3.1)
- Radiation Monitoring System (RMS) (UFSAR Sections 11.5 and 12.3.4), Hope Creek CBD DE-CB.SD - 0044(Z) and CBD DE-CB- SP.0044 (Q)
- Safety Parameters Display System (SPDS) (UFSAR Section 7.5.1.3.3.4)

3.1 Radiological Monitoring Instrumentation - Hope Creek Generating Station

The radiological monitors consist of process radiation monitors, effluent radiation monitors and area radiation monitors. The system continuously displays and/or records the radiation levels in key areas. The listing of the radiation monitoring channels, which may be used to assess an accident, are provided in Table 10-2. A complete description of the radiation monitor program is provided in UFSAR Sections 11.5 and 12.3.4. PSEG NUCLEAR – Radiation Protection (RP) also has portable hand held instruments, which can be used, if thought necessary.

3.2 Process and Area Monitors

In order to provide the operators with essential information on plant conditions during an emergency, various plant processes are continuously monitored. Many of these processes will involve Limiting Conditions for Operations (LCO) and are controlled by the Technical Specifications. If an LCO parameter "goes out of specification," it requires the operators to implement the action statement. The intent of this action is to take corrective measures under abnormal conditions before a situation becomes more serious. These parameters would be monitored closely during an accident for assessment purposes.

3.3 Gaseous Release Path Monitoring

There are four designed gaseous release pathways. These include the North Plant Vent, the South Plant Vent, the Filtration Recirculation Ventilation System (FRVS) and the Hardened Torus Vent (HTV). The North Plant Vent serves the off-gas system, the solid radwaste exhaust system, and the chemistry lab exhaust system.

The South Plant Vent serves the following systems:

a. Reactor Building Ventilation System
b. Radwaste Area Exhaust System
c. Service Area Exhaust System
d. Turbine Building Exhaust System
e. Turbine Building Compartment Exhaust System
f. Turbine Building Oil Storage Room Exhaust System
g. Gland Seal Exhaust
h. Mechanical Vacuum Pump Discharge
i. Radwaste Decontamination Evaporator Exhaust

The locations of the North and South Plant Vents are shown on general arrangement drawings in the Hope Creek UFSAR (Figures 1.2-8 and 1.2-9).

In the Reactor Building Ventilation System exhaust ductwork, radiation monitors isolate the normal heating, ventilation and air-conditioning (HVAC) flow path and initiate FRVS upon sensing high radiation. With the reactor building isolated, FRVS recirculates the reactor building air through high-efficiency particulate absorbers (HEPA) and charcoal filters.

A small amount of effluent is then filtered and released via one of two vent fans. These fans discharge through a vent atop the reactor building to maintain the building at a negative pressure of approximately 0.25-inch water gauge.

The Hardened Torus Vent is a flow path designed to mitigate the effects of a loss of decay heat removal capability. This piping provides a direct venting of the primary containment to the environment, taking advantage of the scrubbing properties of the Torus water. The 12 inch diameter Torus vent pipe runs from the Torus, through the Reactor Building square roof and up the outside of the reactor building cylinder wall.

Continuous monitoring or sampling is provided for all expected radioactive release pathways, with main control room annunciation to indicate when levels are higher than allowed limits.

In addition to the systems mentioned above, a list of portable sampling and survey instrumentation has been provided in the Hope Creek UFSAR. Multi-channel analyzers for isotopic analysis are also available within the Hope Creek station with backup support available from Salem Station.

3.4 Reactor Coolant and Containment Air Sampling - Hope Creek Generating Station

Analysis of reactor coolant and containment air samples provides detailed information on the status of the reactor core. These samples can be used to provide confirmation of a loss of a fission product barrier.

Reactor coolant and containment gaseous activity sampling (normal and high activity/emergency samples) will be performed using station procedures and the normal day-to-day sampling systems. The final release point will be continuously monitored by the RMS for noble gases and continuously sampled for particulates and iodines.

4.0 Dose Assessment From Plant Effluent Monitors for Artificial Island

Plume dose calculation procedures use plant effluent monitor data to project offsite doses due to noble gases and iodines. The primary purposes of the offsite dose calculation are to determine the axial location of highest expected dose at selected distances from the release point, to project dose rates and time integrated doses for downwind portions of the Emergency Planning Zone, and to determine if protective actions are to be recommended. These procedures and calculation capabilities are to be available at the Hope Creek Control Point, Salem Control Room, Hope Creek TSC, Salem TSC, and EOF. The procedures will use the meteorological dispersion factor (X/Q), dose rate or commitment conversion factors, and plant effluent monitor readings to project an offsite dose. The X/Q are selected according to the existing temperature differentials, wind speed, and distance from the plant vent. The dose calculation is based on expected isotopic mixtures or specific mixtures if an isotopic mix has been determined. The plant effluent monitor readings are used in the calculations. The actual isotopic mix of the releases can be used if the releases have been sampled and analyzed. Calculated offsite doses are then compared to Protective Action Guides developed using EPA-400-R-92-001.

The Hope Creek Radiation Monitoring System computer and Salem Safety Parameter Display Systems provide early indication of abnormal radiological conditions from both process and area monitors. The computer systems provide monitoring capability for the radiological parameters identified in Regulatory Guide 1.97, including high range monitoring capability for effluent release paths. This data will be automatically provided to the MIDAS computers at Hope Creek and Salem Stations.

The Hope Creek Digital Radiation Monitoring System provides radiological release rate information. The Salem Computer Systems provide meteorological data acquisition for both Salem Units. MIDAS software provided by ABS Consulting (formerly PLG EQE International) has been installed in computer systems in each station to provide redundant emergency dose assessment modeling capability in manual mode and all modes at the EOF.

The MIDAS System for emergency response is operational on microprocessor based computers. Each system receives meteorological (MET) and Radiation Monitor System (RMS) data automatically from other plant computers via dedicated phone lines or manually via user entry. The user interface is made using graphics screen prompts where selections are made using a system mouse. Source term information is available using several release options including (1) RMS, (2) user entry of monitor data, (3) default accident release, (4) event trees, (5) release rate by isotope, and (6) back calculation. Dispersion is computed using either a straight line or variable trajectory dispersion model. Both models are time dependent and provide integrated doses as well as dose rates using EPA 400 dose factors. Ingestion pathway calculations including (1) airborne concentrations, (2) ground level contamination, (3) foodstuff contamination, (4) ground shine committed dose, and (5) population doses are performed in accordance with the intermediate phase objectives provided in EPA 400.

Several choices are available to the user for determining the source term. If a Design Basis Accident is assumed, but the release rate is unknown, preset release scenarios can be used for up to ten accident scenarios. Otherwise, real time data from effluent monitors will be used.

Upon declaration of a General Emergency (which is done by evaluating specific system parameters), a predetermined Protective Action Recommendation (PAR) is provided to the State governments in New Jersey and Delaware.

The predetermined PARs are developed as outlined in NUREG-0654, Rev. 1, Appendix 1, and Inspection and Enforcement Information Notice 83-28. These PARs are incorporated into both the Event Classification Guide and Emergency Plan Implementing Procedures for Protective Action Recommendations. The use of predetermined PARs allows the transmission and consideration of protective actions in a manner, which affords timely notification of the Emergency Planning Zone (EPZ) municipalities/counties.

The dose calculations use the best information available from the plant effluent monitoring and sample system and the field monitoring team surveys. The doses are integrated over the appropriate sectors and distances around the station.

Transient population is not expected to affect person-rem dose calculations significantly within 10 miles of the plant.

5.0 Dose Assessment From Containment Radiation Monitoring

Dose assessment, utilizing containment high range dose rate monitors, can be obtained with the use of dose assessment computer programs.

6.0 Dose Estimates When Instruments Are Off-Scale or Out of Service

6.1 Defaults for Salem Generating Station

Emergency Plan Procedures describe in detail how projected dose calculations will be made if radiation monitors normally used for monitoring the Containment or Plant Vent are inoperable or off-scale. The procedures call for determining the type of accident, which is occurring and classifying it according to five (5) classes, which are described below.

Class	Conditions Associated with Accident Class
Default 1 (LOCA)	Severe core damage is postulated. Fuel melting is evident by thermocouple readings indicating that the melting point of uranium oxide has been reached. It is also assumed that one hundred percent of the noble gases become airborne in the Containment along with 25% of the iodines. The Containment is assumed to be leaking at the maximum design leakage rate.
Default 2 (LOCA)	Reactor coolant is postulated to be leaking at a rate fast enough to increase the temperature of the fuel cladding to the point where there is threshold damage to the fuel rods. In this case it is assumed that all of the gap activity (the gases contained between the fuel itself and the zircalloy cladding) is released into the coolant and then into the Containment. The Containment is then assumed to be leaking at the maximum design leak rate. In Default 2, LOCA, it is up to the Operations Superintendent or the Emergency Duty Officer to determine that there has been no fuel melting. If there is any uncertainty about fuel melting, Default 1 is assumed.
Default 3 (Gas Decay Tank Rupture)	If local area monitors in the vicinity of (Gas Decay the gas decay tanks indicate a Gas Decay Tank Rupture) Tank has ruptured, a Default 3 incident is assumed to have occurred.
Default 4 (Fuel Handling)	In this accident it is assumed that radioactivity is drawn into the Fuel Handling Building Ventilation System and subsequently released through the Plant Vent.
Default 5 (Steam Generator Tube Rupture)	Available instrumentation indicates that a steam generator tube has ruptured. Normally, the activity discharge can be determined from the Plant Vent monitors. However, in this case it is assumed that the vent monitors and steam generator blow down monitors are inoperable or are not capable of quantifying the releases.

Once a determination of the type of accident has been made, Total Effective Dose Equivalent (TEDE) and thyroid committed doses are projected in accordance with Emergency Plan Implementing Procedures.

6.2 Defaults for Hope Creek Generating Station

Emergency Plan Procedures are developed to describe in detail how projected dose calculations are made if radiation monitors normally used for monitoring plant conditions are inoperable or off-scale. The procedures are developed for determining the type of accident, which is occurring and classifying it. These procedures contain methodology and guidance including defaults that are derived from the HCGS UFSAR, Section 15. The associated radiological assumptions are described in HCGS UFSAR Appendix 15 A.

Class	Conditions Associated with Accident Class
Default (1) LOCA - 1 hour Duration Default (2) LOCA - 24 Hour Duration	These events involve the postulation of a spectrum of piping breaks inside primary containment varying in size, type, and location. The break type includes steam and/or liquid process system lines (HCGS, UFSAR, Section 15.6.5).
Default (3) Steam Line Break	It is postulated that a main steam line breaks downstream of the isolation valve. The plant is designed to immediately detect such an occurrence, initiate isolation of the broken line, and actuate the necessary protective features (HCGS, UFSAR, Section 15.6.4).
Default (4) Feed water Line Break	The postulated break of the feed water line representing the largest liquid line outside the Primary Containment provides the design basis for this event. The break is assumed to be complete and already past the outermost isolation valve. (HCGS, UFSAR, Section 15.6.6).
Default (5) Offgas Treatment	A failure of an active component of the gaseous radwaste treatment system is assumed to occur. This event results in System Failure the activity normally processed by the off gas system being released to the Turbine Building, and subsequently released through the ventilation system to the environment without treatment. (HCGS, UFSAR, Section 15.7.1).

Class	Conditions Associated with Accident Class
Default (6) Control Rod Drop	The radiological consequences of a control rod drop accident (a design basis accident) are Postulated in the HCGS UFSAR Appendix 15 A, Section 15A. This postulated accident assumes cladding failure of several hundred fuel rods, fuel melting localized failure, and subsequent circumstances resulting in radiological releases. (HCGS, UFSAR, Section 15.4.9).
Default (7) Fuel Handling Accident	The fuel handling accident is assumed to occur as a consequence of a failure of the fuel assembly lifting mechanisms resulting in the dropping of a raised fuel assembly onto other fuel assemblies. A variety of events, which qualify for the class of accidents termed “fuel handling accidents”, were investigated (HCGS, UFSAR, Section 15.7.4).
Default (8) Instrument Line failure	This event involves a postulated small break in a steam or liquid line inside or outside the Primary Containment but within the Reactor Building structure. It is assumed that a small instrument line breaks at a location where the break may not be able to be isolated and where immediate detection is not automatic or apparent (HCGS, UFSAR, 15.6.2).

7.0 Dose Assessment From Field Monitoring – Salem–Hope Creek

The Salem and Hope Creek Offsite Dose Calculation Manuals (ODCM's) summarize Environmental Radiological Monitoring. Field monitoring within the plume exposure EPZ takes place whenever the radiological emergency response organization is fully activated. Field teams take direction from the radiological support personnel in the TSC and/or EOF. Data is obtained and updated quarter hourly and hourly on the meteorological variables of wind direction, speed and vertical temperature change (Delta T). This data is used to direct the onsite and offsite survey teams. Each field monitoring team is capable of performing the necessary functions required to obtain reliable data. Communications are accomplished by the use of emergency radios and cellular phones by each team. Deployment times range from 30 to 90 minutes for the onsite and offsite emergency radiation survey team(s). Field monitoring is performed in accordance with Emergency Plan Implementing Procedures. Procedures have been prepared which allow personnel to determine release rates from field data and then calculate doses at other locations.

Station survey instruments are able to detect radioiodine concentrations as low as I.0E-07 uCi/cc provided that noble gases and background radiation (which can adversely affect the Minimum Detectable Activity (MDA) are minimized. In order to achieve this, silver zeolite cartridges, which can be placed in portable field samplers, are used. The silver zeolite cartridges have better iodine to noble gas adsorption ratio than standard charcoal cartridges. Since high background can also adversely affect readings, survey team personnel are pre-directed to count the cartridges in low background areas.

Emergency Plan Implementing Procedures list equipment required for a field survey team. This equipment provides the means for directly measuring or relating measured field contamination levels to dose rates. The dose rate due to contamination and the plume are obtained directly from the dose rate meter.

8.0 Dose Assessment from Liquid Sample Activity Concentration

Since the Delaware River is not a source of potable water in the vicinity of the Station, the major critical pathways by which a population would receive a radiation exposure from liquid effluent releases are swimming and boating activities.

The radiation dose received by such activities is dependent upon three factors:

a. The isotopic mixture of the release;

b. The concentration of the nuclides at the point of interest;

and

c. The time period of exposure.

All three factors are highly variable, but certain assumptions can be made to calculate a conservative dose conversion factor. The isotopic mixture varies according to the operating history of the plant and on the status of the radwaste system at the time of the incident. The concentration of the nuclides is also dependent upon plant conditions but of equal importance is that this factor varies according to the hydrological mixing and dilution during transport of the liquid release to the site of interest. Based on predicted surface temperature profile data, a dilution factor of 10 can be assumed for swimming and boating activities near Artificial Island.

In the event of a radioactive release to the Delaware River, water samples will be taken and counted. The total counts per minute determined would then be converted to a gross gamma concentration.

8.1 Water Immersion (Swimming)

The radiation dose from water immersion (swimming) depends upon the concentration of the nuclides present at the location of the immersion and the period of exposure. Dose rate conversion factors have been calculated on the assumption that the swimmer is completely submerged and surrounded on all sides by a large volume of water. This physical arrangement approximates 4 π geometry for gamma radiation and 2 π for beta radiation.

8.2 Normalized Conversion Factors for Water Immersion and Boating

Based on a typical isotopic mixture, general dose equations can be formulated which incorporate a weighted average dose rate conversion factor, a gross isotopic concentration value, and the time period of exposure.

Based on sample analysis, exposure time, and the normalized conversion factors, dose can be calculated for any swimming or boating activities in the vicinity of Artificial Island. A comparison would then be made of these calculated doses with State Action Levels as indicated in the State Radiological Emergency Response Plan for Nuclear Power Plants.

9.0 Other Onsite Emergency Equipment- Assessment

Onsite instrumentation, which can be used to initiate emergency measures, is described in the implementing procedures of this plan.

9.1 Meteorological Monitoring

A meteorological program in accordance with the recommendation of NRC Regulatory Guide 1.23 "Onsite Meteorological Programs" and Section 2.3.3 of NUREG 75/087 (Rev. 1) has been established.

The primary meteorological monitoring system measures wind speed and direction at three elevations (300 ft., 150 ft., and 33 ft.). Temperature difference is measured between 300 ft. and 33 ft. and between 150 ft. and 33 ft., in order to provide vertical lapse rates for air stability estimates. Calculated sigma theta values of the wind direction at the three elevations are also provided.

Backup meteorological data is provided by a backup tower located onsite approximately 500 ft. south of the primary meteorological tower. Backup meteorological data is provided through wind speed and wind direction sensors mounted on a ten-meter pole. In addition to the 15-minute averaged wind speed and wind direction, a computed sigma theta value is provided. The primary as well as the backup meteorological information is available in the Control Rooms, Salem and Hope Creek TSC, and the EOF.

The meteorological monitoring system is provided with a dedicated battery backup power supply. The system is calibrated quarterly using equipment traceable to an NBS Standard. The Meteorological Monitoring Program is reviewed biennually in accordance with the Hope Creek and Salem UFSAR. (A detailed description of the onsite meteorological measurements program is provided in Section 2.3.3 of the Hope Creek and Salem UFSARs).

A system to provide alternate remote interrogation of the meteorological system is available by way of direct telephone dial-up capability.

The Emergency Plan Implementing Procedures provide for meteorological support from the closest NOAA Weather Station (National Weather Service-NWS). Information, including synoptic weather conditions, forecast, regional precipitation and severe weather alerts from this NWS station is available on a 24-hour-per-day basis. Monthly communication checks with this NWS station are made in accordance with Section 15.0 of this Plan. It has been determined that the data from this nearby NOAA weather station is representative of the combination of local and regional meteorology. Backup communication with this weather station uses the Delaware NAWAS.

9.2 Seismic Monitoring

A Control Room alarm is provided in the event of seismic activity associated with the Operating Basis Earthquake (OBE). Seismic monitoring is performed using triaxial accelographs (with a range of +1g and sensitivity of 0.0lg) multi-channel recorders and response spectrum analyzers. Time history accelographs are placed throughout the plant site. (A complete discussion of seismic instrumentation is provided in Hope Creek UFSAR, Section 3.7.4. and Salem UFSAR, Section 3.7).

9.3 River Level Monitoring

River water levels at each service water pump sump, upstream of the intake structure, are indicated at the Control Room. This system includes two (2) level sensing elements, two (2) transmitting/recording channels, and a signal conditioner.

The geophysical instrumentation monitors the parameters required for evaluating action levels contained in the Event Classification Guide (ECG) and Emergency Plan Implementing Procedures.

9.4 Fire Detection

Both station Fire Protection Systems are designed in general accordance with the National Fire Protection Association's standards. Any fire initiates fire alarms and the protection systems as appropriate. An alarm is initiated by automatic sprinkler actuation, smoke detector actuation, heat sensor actuation or by manual action.

TABLE 10-1
SALEM GENERATING STATION RADIATION MONITORS UNIT ONE

CHANNEL	DESCRIPTION OF MONITOR
R1A	CONTROL ROOM GENERAL AREA
R1B	CONTROL ROOM INTAKE DUCT
R2	LOW RANGE GENERAL AREA CTMT 130'
R3	CHEMISTRY LAB
R4	CHARGING PUMP GENERAL AREA
R5	SPENT FUEL POOL GEN AREA FUEL HANDLING BLDG
R6A	PRIMARY SAMPLE LAB
R7	INCORE SEAL TABLE CTMT 100'
R9	NEW FUEL STORAGE FUEL HANDLING BLDG
R10A	PERSON HATCH GEN AREA CONTAINMENT 100'
R10B	PERSON HATCH GEN AREA CONTAINMENT 130'
R11A	CONTAINMENT PARTICULATE
R12A	CONTAINMENT NOBLE GAS
R12B	CONTAINMENT IODINE
R13A & B	CFCU SW LOCAL MONITOR
R15	CONDENSER AIR EJECTOR
R17A & B	COMPONENT COOLING
R18	LIQUID RAD WASTE
R19A, B, C, & D	STEAM GENERATOR BLOWDOWN
R20B	CHEMISTRY COUNT ROOM
R23	P.S. CONTROL POINT
R26	REACTOR COOLANT FILTER
R31A	LETDOWN HX FAILED FUEL (GROSS)
R32A	FUEL HANDLING CRANE
R34A	MECHANICAL PENETRATION GENERAL AREA
R36	EVAP & FDWTER PREHEAT CONDENSATE
R40	CONDENSATE FILTER
R41A	PLANT VENT LOW RANGE NOBLE GAS
R41B	PLANT VENT MID RANGE NOBLE GAS
R41C	PLANT VENT HIGH RANGE NOBLE GAS
R44A	CONTAINMENT GENERAL AREA HIGH RANGE 130'
R44B	CONTAINMENT GENERAL AREA HIGH RANGE 100'
R45A	PLANT VENT NOBLE GAS BKGD SUBTRACT
R45B	PLANT VENT NOBLE GAS MEDIUM RANGE
R45C	PLANT VENT NOBLE GAS HIGH RANGE
R45D	PLANT VENT FILTER MONITOR
R46A, B, C, & D	MAIN STEAM LINES
R47	ELECTRICAL PEN. GENERAL AREA HIGH RANGE
R51	TECHNICAL SUPPORT CENTER INDUCT
R53A, B, C, & D	N16 MAIN STEAMLINE MONITOR

TABLE 10-1 (cont.)

SALEM GENERATING STATION RADIATION MONITORS UNIT TWO

CHANNEL	DESCRIPTION OF MONITOR
R1A	CONTROL ROOM GENERAL AREA
R1B	CONTROL ROOM INTAKE DUCT
R2	LOW RANGE GENERAL AREA CTMT 130'
R4	CHARGING PUMP GENERAL AREA
R5	SPENT FUEL POOL GENERAL AREA
R7	INCORE SEAL TABLE CTMT 100'
R9	NEW FUEL STORAGE FUEL
R10A	PERSONNEL HATCH GENERAL AREA CTMT 100'
R10B	PERSONNEL HATCH GENERAL AREA CTMT 130'
R11A	CONTAINMENT PARTICULATE
R12A	CONTAINMENT NOBLE GAS
R12B	CONTAINMENT IODINE
R13A & B	CFCU SW LOCAL MONITOR
R15	CONDENSER AIR EJECTOR
R17A & B	#21 & 22 COMPONENT COOLING
R18	LIQUID RAD WASTE
R19A, B, C, & D	STEAM GENERATOR BLOWDOWN
R26	REACTOR COOLANT FILTER
R31	LETDOWN HX FAILED FUEL
R32A	FUEL HANDLING CRANE
R34	MECHANICAL PENETRATION GENERAL AREA
R37	NON-RADIOACTIVE LIQUID WASTE BASIN
R40	CONDENSATE FILTER
R41A	PLANT VENT LOW RANGE NOBLE GAS
R41B	PLANT VENT MID RANGE NOBLE GAS
R41C	PLANT VENT HIGH RANGE NOBLE GAS
R44A	CONTAINMENT GENERAL AREA HIGH RANGE 130'
R44B	CONTAINMENT GENERAL AREA HIGH RANGE 100'
R45A	PLANT VENT NOBLE GAS BKGD SUBTRACT
R45B	PLANT VENT NOBLE GAS MEDIUM RANGE
R45C	PLANT VENT NOBLE GAS HIGH RANGE
R45D	PLANT VENT FILTER MONITOR
R46A, B, C, & D	MAIN STEAM LINES
R47	ELECTRICAL PEN. GENERAL AREA HIGH RANGE
R52	PASS RM (LOCAL)
R53A, B, C, & D	N16 MAIN STEAMLINE MONITOR

TABLE 10-2
HOPE CREEK GENERATING STATION RADIATION MONITORS

CHANNEL	DESCRIPTION OF MONITOR
9RX500	RACS
9RX501	SACS A
9RX503	SACS B
9RX505	TBCW
9RX506	CTB MEAS. CONC.
9RX507	DLD RMS
9RX508	LIQ. RADWASTE
9RX509	MSL A
9RX510	MSL B
9RX511	MSL C
9RX512	MSL D
9RX516	HTV N/G LOW
9RX517	HTV N/G HIGH
9RX518	HTV EFF
9RX580	SPV EFF
9RX581	SPV N/G HIGH
9RX590	NPV EFF
9RX591	NPV N/G HIGH
9RX598	CTB CALC CONC
9RX599	CTB EFF
9RX602	NPV N/G LOW
9RX603	NPV N/G MID
9RX606	SPV N/G LOW
9RX607	SPV N/G MID
9RX610	FRVS N/G MID
9RX611	FRVS N/G HIGH
9RX612	NFS A
9RX613	NFS B
9RX614	RX. BLDG. VENT. EXH.
9RX615	TB. BLDG. EXH.
9RX616	RADW. EXH. SYS
9RX617	GAS. RADW. AREA EXH.
9RX618	TB. BLDG. COMP. EXH.
9RX619	RADW. AREA EXH.
9RX620	TECH. SUP. CTR.
9RX621	OFFGAS A
9RX622	OFFGAS B
9RX625	OFFGAS TREATED A
9RX626	OFFGAS TREATED B
9RX627	RFE A
9RX628	RFE B
9RX629	RFE C
9RX630	CRV C
9RX631	CRV C1
9RX632	RBE A
9RX633	RBE B
9RX634	RBE C

TABLE 10-2 (cont.)
HOPE CREEK GENERATING STATION RADIATION MONITORS

CHANNEL	DESCRIPTION OF MONITOR
9RX635	DAPA A
9RX636	DAPA B
9RX637	CRV D
9RX638	CRV D1
9RX640	FRVS N/G LOW
9RX680	FRVS EFF
9RX698	OUTER TIP RM ARM
9RX699	INNER TIP RM ARM
9RX700	PERS. AIRL.
9RX701	MDT. EQPT. HATCH
9RX702	OPEN EQPT. HATCH 145'
9RX703	OPEN EQPT. HATCH 162'
9RX704	SAFEG. INST. RM.
9RX705	EQPT. AIRLOCK
9RX706	RCDSE
9RX707	SPENT FUEL SP
9RX708	RBSS
9RX709	RADW. DRUM SHIP AREA
9RX710	MAIN CR
9RX711	CHEM. LAB. SPL.
9RX712	AUX. HATCHWAY
9RX713	RESTR. MS. 1
9RX714	RESTR. MS. 2
9RX715	TECH. SUP. CTR. INLET.
9RX716	OG VIAL SAMP. STA.
9RX717	RADW SAMP. STA.
9RX720	FRVS LRP
9RX721	FRVS SKID
9RX722	LIQ. RADW. CR
9RX723	ORBSS
9RX724	OFFGAS CR