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8 ELECTRICAL SYSTEMS

8.1 DESIGN BASES

The design of the electrical systems for this three unit nuclear station is based on providing the required electrical equipment and power sources to assure continuous operation of the essential station equipment under all conditions.

8.2 ELECTRICAL SYSTEM DESIGN

8.2.1 NETWORK INTERCONNECTIONS

Each unit will generate electric power at 19 kV which is fed through an isolated phase bus to a unit stepup transformer where it is stepped up to the transmission voltage, 230 kV for Units 1 and 2 and 500 kV for Unit 3. From Unit 1 and Unit 2, overhead transmission lines feed the power to the station 230 kV switching station which is connected to the existing Duke Power Company 230 kV transmission network by eight circuits. Six circuits are installed with
35. Unit 1, two east-northeast to North Greenville and four east-southeast to Central. Two additional circuits north-northwest to Jocassee are installed with Unit 2.

From Unit 3, an overhead transmission line feeds power to the 500 kV switching station which is connected to Duke's 500 kV transmission network by two cir-
35. cuits. One 500 kV circuit is installed east-northeast to McGuire Station and one east to Newport Station. In addition, with Unit 3, a 230/500 kV autotransformer connects the 500 kV switching station to the 230 kV switching
35. station.

In addition to the power available from the nuclear units, the 230 kV and 500 kV transmission networks, the entire output of two 87,500 kVA units of the onsite Keowee Hydro Station is connected to the station's 230 kV switching station through a single circuit overhead transmission line. Also, one of the hydroelectric units is connected continuously through a 13.8 kV underground feeder to a 4160 volt transformer which feeds the standby power buses at Oconee.

The 230 kV buses, disconnect switches, and circuit breakers are arranged into
35. a breaker and a half configuration. The 500 kV switching station is also arranged into a breaker and a half configuration.

A 100 kV substation is also provided and installed at Oconee on the opposite
2. side of the station from the 230 kV facilities. A 100 kV transmission line runs from Lee Steam Station through the Central substation to the Oconee substation providing an additional emergency power supply.

See Figure 8-1 and 8-2 for arrangement of lines in the Oconee Station and on the site.

8.2.1.1 Reliability Considerations

Reliability considerations to minimize the probability of power failure due to faults in the network interconnections and the associated switching are as follows:

(a) Redundancy is designed into the network interconnections by installing two full capacity transmission circuits for each connection to the 230 kV grid.

(b) The two single 230 kV transmission circuits are installed on the same line of double circuit towers. Each line of double circuit towers is separated a safe distance from the others and in most cases installed over a different route.

(c) One of the circuits on a line of 230 kV transmission towers is insulated at a higher insulation level than the other, thus minimizing the probability of double outages due to flashovers.

(d) Each circuit is protected from lightning and switching surges by an overhead electrostatic shield wire and in addition, lightning arresters are installed at both terminals.

35. (e) The breaker and a half switching arrangement in the 230 kV and 500 kV switching stations includes two full capacity main buses which feed each circuit through a circuit breaker connected to each bus. Completely redundant primary and backup relaying is provided for each circuit along with circuit breaker failure backup protection. These provisions permit the following:

1. Any circuit can be switched under normal or fault switching without affecting another circuit.

2. Any single circuit breaker can be isolated for maintenance without interrupting the power or protection to any circuit.

3. Short circuits of a single main bus will be isolated without interrupting service to any circuit.

4. Short circuit failure of the tie breaker will result in the loss of its two adjacent circuits until it is isolated by disconnect switches.

5. Short circuit failure of a bus side breaker will result in the loss of the associated bus until it is isolated.

6. Failure of either the primary protective relaying or the backup protective relaying will not result in the loss of circuit protection.

35.

With the above protection features, the probability of loss of more than one source of 230 kV or 500 kV power from credible faults is low; however, in the event of an occurrence causing loss of all the 230 kV and 500 kV connections, the station is supplied from the two hydro units or the
36. 100 kV line supplied from the Lee combustion turbines.

8.2.2 STATION DISTRIBUTION SYSTEM

The station distribution system consists of the various electrical systems designed to provide reliable electrical power during all modes of station operation and shutdown conditions. The systems are designed with sufficient power sources, redundant buses, and required switching to accomplish this. Engineered safeguard equipment for each unit is arranged onto three load group buses such that the loss of a single bus section for any reason results in only the loss of equipment fed from that bus leaving redundant equipment to perform the same function. In general, the equipment related to unit operation is connected to its respective unit auxiliary electrical buses, whereas equipment common to and serving all units is distributed between unit auxiliary electrical buses. The control of power sources and switching for Units 1 and 2 is accomplished from the Units 1 and 2 control
36. room while control of power sources and switching for Unit 3 is from the Unit 3 control room.

8.2.2.1 <u>System Diagrams</u>

Figure 8-3 is a single line diagram of a typical unit's 6900 volt and 4160 volt auxiliaries systems. Figure 8-4 is a single line diagram of a typical unit's engineered safeguards 600 volt and 208 volt auxiliaries systems. Figure 8-5 is a diagram of the 125/250 volt dc system and 120 volt ac vital power system.

8.2.2.2 <u>Auxiliary Transformers</u>

Each unit is provided with one full size unit auxiliary transformer and one full size startup transformer. The unit auxiliary transformer is connected to its respective generator bus and provides a source of normal power to the unit. Each startup transformer is connected to the 230 kv switching station and provides a source of power for startup, shutdown, and after shutdown requirements. The startup transformer also serves as a complete standby source to serve auxiliaries in the event of failure of the unit auxiliary transformer. In addition, each unit's startup transformer can backup another unit's startup transformer through emergency startup buses and dual isolating disconnect switches. Two standby transformers CT4 and CT5 are installed to provide additional sources of standby power to all three units through the standby power buses.

Transformer CT4 is supplied by a 13.8 kV underground feeder from the on-site Keowee Hydro Station while transformer CT5 is supplied from the 100 kV network by a 100 kV transmission line which can be supplied from three combustion turbines located at our Lee Steam Station.

Both the unit auxiliary transformer and the startup transformer are rated at 45/60 MVA and have two isolated secondary windings rated 6900 volts and 4160 volts respectively. Standby transformers CT4 and CT5 are rated at 12/16/20 MVA and have single secondary windings rated 4160 volts each. The individual
24. unit auxiliary transformer can also supply emergency power to the auxiliary buses by disconnecting the generator from the 19 kV main bus and backfeeding through the main step-up transformer.

8.2.2.3 6900 Volt Auxiliary System

The 6900 volt auxiliary system for each unit is designed to supply electric power to the 9000 horsepower reactor coolant pump motors. This system is arranged into two bus sections. Both bus sections feed into two switchgear bus sections, each feeding two motors. Each switchgear bus supplies one motor for each of the two reactor coolant piping loops. Either transformer is capable of feeding both switchgear buses. During startup, shutdown, and after shutdown, the switchgear buses are supplied from the startup transformer.
36. During normal operation the switchgear buses are supplied from the unit auxiliary transformer. Normal bus transfers between the two sources are initiated at the discretion of the operator from the control room, while emergency transfer from the unit auxiliary to the startup transformer is initiated automatically by protective relay action. Normal bus transfers used on startup or shutdown of a unit are "live bus" transfers, i.e., the incoming source feeder circuit breaker is closed onto the energized bus section and its interlocks will trip the outgoing source feeder circuit breaker which results in transfers without power interruption. Emergency bus transfers used on the loss of the normal unit source are rapid bus transfers, i.e., the outgoing source feeder circuit breaker is tripped and its interlocks close the incoming source feeder circuit breaker which
9. results in a transfer within a maximum of nine cycles.

The 6900 volt auxiliary systems as shown in Figures 8-2 and 8-3 is similar in arrangement for each of Units 1, 2, and 3.

8.2.2.4 4160 Volt Auxiliary System

The 4160 volt auxiliary system for each unit is arranged into a double bus - double circuit breaker switching arrangement. The three power sources, (1) the unit's auxiliary transformer, (2) the startup transformer and (3) the standby power buses, feed each of the main feeder buses by this double circuit breaker arrangement. Each of the two redundant main feeder buses provide power to each of the three redundant engineered safeguards switchgear bus sections. With no in-station emergency, manual and emergency transfers are as described above for the 6900 volt system. In the event of an accident within a unit requiring engineered safeguards and if power is unavailable from the unit auxiliary transformer and the startup transformer, an automatic
6. transfer is made as described in 8.2.3.3.c. Power will be available to the standby power buses within 23 seconds from the on-site Keowee hydroelectric generators. These two 87,500 kVA generating units start automatically upon

an engineered safeguards actuation or loss of the external grid. One hydro unit is connected at all times through a 13.8 kV underground feeder to standby transformer CT4. Transformer CT4 supplies the two redundant 4160 volt standby power buses when required. In addition, a 4160 volt power feeder from the 100 kV transmission system is available to supply power to the standby power buses whenever required. Standby emergency power is also available to the startup transformer from the two hydroelectric generators through the 230 kV switching station.

The 4160 volt auxiliary system as shown in Figures 8-2 and 8-3 is similar in arrangement for all three units.

8.2.2.5 600 Volt Auxiliary System

Each unit's 600 volt auxiliary system is similar and arranged into multiple bus sections. Each bus section is fed from a separate load center transformer which is connected to one of the three 4160 volt switchgear bus sections. Various 600 volt motor control centers are located throughout the station to supply power to equipment within the related area. The three engineered safeguards load centers and associated motor control centers as shown in Figure 8-4 are redundant and are supplied independently from the three 4160 volt engineered
9. safeguards load buses. Each motor control center has an alternate feeder with manual transfer to be utilized only for maintenance. No common failure mode exists for this system.

8.2.2.6 208 Volt Auxiliary System

For each unit a system is provided to supply instrumentation, control, and power
2. loads requiring unregulated 208Y/120 volt ac power. It consists of motor control centers, distribution panels, and transformers fed from 600 volt motor control centers.

The redundant engineered safeguards 208 volt motor control centers for a unit are shown in Figure 8-4. Each of these motor control centers have redundant supply feeders from separate transformers and redundant 600 volt motor control
9. centers. The feeder breakers have mechanical interlocks and manual transfers.

The 208 volt auxiliary system is similar in arrangement for each of the three units.

8.2.2.7 DC Power Systems

For each nuclear unit, two separate dc power systems are provided; namely, a 125 volt dc system provides a source of reliable continuous power for control and instrumentation for normal operation and orderly shutdown for each unit, and a
2. separate 125/250 volt dc system is provided to supply large power loads for each unit. In addition, a separate 125 volt dc system is provided for the switching
16. station protection and control. These systems are shown in Figure 8-5. For each Keowee hydro unit, separate and independent dc power systems are provided to assure a source of reliable continuous power for normal and emergency operation. These systems are shown in Figure 8-6.

8.2.2.7.1 125 Volt DC Instrumentation and Control Power System

2. For each unit, two independent and physically separated 125 volt dc batteries
and dc buses are provided for the vital instrumentation and control power sys-
2. tem. The dc buses are two conductor metalclad distribution center assemblies.
Three battery chargers are also supplied with two serving as normal supplies
to the bus sections with the associated 125 volt dc battery floating on the
bus. The batteries supply the load without interruption should the battery
2. chargers or the ac source fail. Each of the three battery chargers are sup-
plied from the redundant 600 volt ac engineered safeguards motor control cen-
ters of each unit. One of these three battery chargers serves as a standby
2. battery charger and is provided for servicing and to backup the normal power
supply chargers. A bus tie with normally open breakers is provided between
each pair of dc bus sections to "backup" a battery when it is removed for
servicing.

Four separate 125 volt dc instrumentation and control panelboards are also provided for each unit. Each panelboard receives its dc power through an auctioneering network of two isolating diode assemblies. One assembly is connected to the unit's 125 volt distribution system and the other assembly is connected to another unit's 125 volt distribution system. The functions of the diode assemblies are to discriminate between the voltage level of the two dc distribution systems, pass current from the dc system of higher potential to the instrumentation and control panelboard connected on the output of the diode assemblies, and block the flow of current from one dc distribution system to the other.

Each isolating diode assembly is composed of a series-parallel network of
four diodes in each polarity leg of the dc supply to the panelboard it serves.
With this series-parallel arrangement of diodes, either an open circuited or
short circuited diode can be tolerated without affecting the operability of
the diode assembly. The individual diodes are sized for a continuous current
6. of 500 amperes with the maximum panelboard load current being 304 amps. Each
diode is also rated for continuous operation with a peak inverse voltage of
800 volts.

Continuous monitoring of each diode is provided in the design of each isolat-
ing diode assembly to detect a shorted or open circuited diode. Since each
individual monitor is connected across the diode it monitors, a complete fail-
9. ure analysis was conducted to assure that a failed component in the monitor
does not prevent the detection of diode trouble. Factory tests are conducted
on the monitor to check their operability at their design ambient of 137 F
which is the worst-case environmental condition they will see in service
(loss of air conditioning for prolonged periods of time). Factory tests also
are conducted to check monitor operability under varying voltage levels. The
monitors are designed to operate continuously without component failure with
a back voltage of 50 volts continuous or 800 volts for 10 seconds. Since
the battery undervoltage relay alarms are 123 vdc, only a voltage difference
between battery voltages (back voltage) less than 9 volts can occur undetected,
and, with one battery in the network completely discharged, the back voltage
seen by an isolating diode assembly would be 25 volts. With a back voltage of
25 volts on a monitor assembly, the current flow from battery system to battery
system would be less than 0.5 amps.

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An alarm relay, connected to an individual control room annunciator point, is provided in each isolating diode assembly to advise the operator of diode trouble in the particular assembly in difficulty. The alarm system is designed to be void of nuisance tripping.

6. Test provisions are included in each isolating diode assembly to allow the in service checking of the operability of individual diode monitors, and, in addition, to allow the out of service periodic checking of the peak inverse voltage capability of each individual diode. The latter test can be conducted on one isolating diode assembly with the other diode assembly in the network in operation. Breakers on the input and output of each isolating diode assembly are provided for complete isolation during maintenance and testing of an assembly.

8.2.2.7.2 125/250 Volt DC Station Power System

For each unit a separate 125/250 volt dc power system is supplied. Each system consists of three 125/250 volt dc power supply battery chargers, a three conduc-
2. tor, metalclad distribution center assembly, and two 125 volt dc batteries. This arrangement provides 125 volts from "P" bus to "PN" bus, 125 volts dc from "PN" bus to "N" bus, and 250 volt dc from "P" bus to "N" bus. Loads on this system are basically the 250 volt dc power loads of units. Each 125 volt dc half of a bus section normally is supplied from one of the 125 volt dc power supply battery chargers with the associated 125 volt dc battery floating on the bus. The batteries supply the load without interruption should the battery charger or the ac source fail. A bus tie with normally open double breakers is provided between the three
2. units' distribution center bus sections to backup a battery when it is removed for servicing. One standby 125 volt dc power supply battery charger is provided between each pair of the 125 volt dc batteries for servicing and to "backup" the normal power supply battery chargers.

8.2.2.7.3 125 Volt DC Switching Station Power System

The switching station dc system consists of two 125 volt dc, two conductor,
2. metalclad distribution center assemblies; three battery chargers; and two 125 volt dc batteries. A bus tie with breakers is provided between the switchgear bus sections to "backup" a battery when it is removed for servicing. One standby 125 volt dc battery charger is also provided between the two 125 volt dc batteries for servicing. One battery supplies power through panelboards for primary control and protective relaying, and the second battery supplies power through panelboards for backup control and protective relaying. Dual feeds
2. from the redundant panelboards are provided to each Power Circuit Breaker (PCB) for closing and tripping control. Separate dual trip coils are provided
2. for each PCB. Isolating diodes are provided for the redundant power feeds to the common closing coil circuit.

8.2.2.7 4 125 Volt DC Keowee Station Power System

For each Keowee hydro unit a separate 125 volt dc power system is supplied. Each system consists of one 125 volt dc power supply battery charger, one 125 volt dc, two conductor, metalclad distribution center assembly and one 125 volt dc battery. A bus tie with normally open double circuit breakers is
16. provided between the two switchgear bus sections to "back up" a battery when it is removed for servicing. One standby 125 volt dc battery charger is also provided between the two 125 volt dc batteries for servicing. The batteries, battery charger and distribution center associated with one unit are physically separated in separate enclosures from those associated with the other unit.

16. 8.2.2.7.5 125 Volt DC Batteries

Each battery is sized to carry the continuous emergency load for a period of one hour in addition to supplying power for the operation of momentary loads during the one hour period.

In normal operation the batteries are floated on the buses, and assume load without interruption on loss of a battery charger or ac power source.

The lead-acid batteries will be tested to prove their ampere-hour capacity. Inservice periodic checks of the status of each cell are made through battery hydrometer log readings.

16. 8.2.2.7.6 Single Failure Analysis of the 125 Volt DC Instrumentation and Control Power System

As shown in Table 8-1, the 125 volt dc instrumentation and control power system is arranged such that a single fault within either system does not preclude the reactor protective system, engineered safeguards protective system, and the engineered safeguards equipment from performing their safety functions.

16. 8.2.2.7.7 Single Failure Analysis of the 125 Volt DC Switching Station Power System

As shown in Table 8-2, the 125 volt dc switching station power system is arranged such that a single fault within the system does not preclude the protective relaying and control in the switching station from performing its intended functions.

8.2.2.7.8 Single Failure Analyses of the 125 Volt DC Keowee Station Power System

16. The 125 Volt DC Keowee Station Power System is arranged such that a single fault within either unit's system does not preclude the other unit from performing its intended function of supplying emergency power.

8.2.2.8 120 Volt AC Vital Power Buses

Figure 8-5 shows the arrangement of the 120 volt ac vital power buses. For
2. each unit, four redundant 120 volt ac vital instrument power buses are pro-
vided to supply power in a predetermined arrangement to vital power, instru-
mentation, and control loads under all operating conditions. Each bus is
2. supplied separately from a static inverter connected to one of the four 125
volt dc control power panelboards described in 8.2.2.7.1. Upon loss of power
2. from 125 volt dc bus DCA or DCB, the affected inverter is supplied power
from the remaining bus through its respective dc control power panelboards
and transfer diodes. A tie with breakers is provided to each of the 120
volt vital ac buses from the alternate 120 volt ac regulated bus to
provide backup for each vital bus and to permit servicing of the inverters.
Each inverter has the synchronizing capability to permit synchronization with
the regulated buses.

For each unit, each of the four redundant channels of the nuclear instrumentation
and reactor protective system equipment described in 7.1.2 is supplied from a
2. separate bus of the four redundant buses. Also for each unit, each of the three
redundant channels of the engineered safeguards protective system described in
2. 7.1.3 is supplied from a separate bus of the four redundant buses. The two engi-
neered safeguards actuation power buses are supplied from separate vital power
buses.

8.2.2.8.1 Single Failure Analysis of the 120 Volt Vital Power Buses

The 120 volt vital power system is arranged such that any type of single failure
or fault will not preclude the reactor protective system, engineered safeguards
protective system, and engineered safeguards equipment from performing their
2. safety functions. There are four independent buses available to each unit, and a
single failure within the system can involve only one bus. A single failure
analysis is presented in Table 8-3.

8.2.2.9 240/120 Volt AC Essential Power System

For each unit, three essential power systems are provided to supply power. They are:

1. The unit's integrated control system (ICS) power system, which is 120 volt ac, single phase.

2. The unit's auxiliary power system (APS), which is 120 volt ac, single phase.

2. 3. The unit's computer power system (CPS), which is 240/120 volt ac, single phase.

Each of these three systems consist of a static inverter, with redundant 125 volt dc supplies from separate 125 volt dc buses, circuit breakers, and distribution panelboard. Also, a static transfer switch is provided in each system as a means for automatic transfer of system loads to the alternate ac regulated power system should the inverter become unavailable. The output of each inverter is synchronized with the ac regulated power system through the static switch in order to minimize transfer time from inverter to the regulated supply.

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8.2.2.10 <u>240/120 Volt AC Regulated Power System</u>

For each unit, a system is provided to supply instrumentation, control, and power loads requiring regulated ac power. It also serves as an alternate power source
2. to both the vital power panelboards and to the essential power panelboards. The system consists of two distribution panels, two regulators, and two transformers fed from separate motor control centers. These systems are shown in Figure 8-5.

8.2.2.11 <u>DC and AC Vital Power System Monitoring</u>

Failure and/or misoperation of all DC and AC vital power system equipment is being monitored on two local alarm annunciators located in the equipment room near most of the vital equipment. Several variables within each piece or re-
6. dundant group of equipment are being monitored on one of the local panels, with one alarm from each group being taken to alarm panels in the control room. The control room alarms alert the operator if an alarm condition occurs on any piece or group of equipment or if power is lost to the local alarm monitoring equipment.

The DC bus tie breakers, battery breakers and standby charger breaker position indication contacts; the standby charger trouble contact; and the computer, ICS and auxiliary inverter isolating diode trouble contact are monitored directly in the control room.

The other vital alarms are divided into two separate and independent monitoring systems. Alarms for equipment which have battery ICA for their primary source of power are maintained physically and electrically separate from battery ICB powered equipment. For example, the distribution center, isolating diodes, breakers, panelboards, inverters and transfer switches associated with battery ICA are alarmed on local and remote annunciators which are physically and electrically separated from the annunciators being used for monitoring battery ICB associated systems.

Specifically, the variables being monitored locally with a composite alarm from each of the 17 groups being taken to the control room are as follows:

Group 1 and 11 for each of the two DC buses
- System Ground
- Charger trouble
- Charger output breaker tripped
- Bus voltage low (123 V DC)

Group 2, 3, 4, 5, 12, 13, 14, 15 for each of eight isolating diodes
6.
- Fuse blown
- Diode failure
- Input breaker open
- Output breaker open
- Feeder breaker open

Group 6, 7, 16, 17 for each of four vital inverters and panelboards

- Fuse blown
- Inverter output voltage low
- Bypass voltage low
- Inverter bypassed
- Panelboard voltage low (60%)

6. Group 8, 18, 19 for computer, ICS and auxiliary inverters and panelboards

- Fuse blown
- Inverter output voltage low
- Bypass voltage low
- Inverter bypassed
- Panelboard voltage low (60%)

6. 8.2.2.12 Interactions

The 125 volt dc instrumentation and control power system and the 120 volt ac vital power system are designed such that upon loss of power supplies no interactions exist between reactor protection systems, engineered safeguards protection systems, and control systems that would preclude these systems from performing their respective functions. Also, any interactions between units as a result of the loss of power supplies does not preclude the safety and control systems of any unit from fulfilling their function. This is verified by safety analyses and is shown in Tables 8-1, 8-3, and 8-4.

6. 8.2.2.13 Evaluation of the Physical Layout, Electrical Distribution System Equipment

The physical locations of electrical distribution system equipment shown in Figure 8-2, 8-3 and 8-4 are arranged to minimize vulnerability of vital circuits to physical damage as a result of accidents. The locations are as follows:

(a) Auxiliary transformers, startup transformers, and the 100 kV transformer are located out of doors and physically separated from each other. The 13.8 kV transformer CT4, fed from the on-site Keowee Hydro Station is physically separated from the other transformers and located in a Class I enclosure. Lightning arresters are used where applicable for lightning protection. All transformers are covered by automatic water spray systems. Transformers are well spaced to minimize their exposure to fire, water, and mechanical damage.

(b) The 6900 volt switchgear, 4160 volt switchgear, and 600 volt load centers are located in areas to minimize exposure to mechanical, fire, and water damage. This equipment is coordinated electrically to permit safe operation of the equipment under normal and short circuit conditions. Metal-clad construction is used throughout for personnel and equipment protection.

(c) The 4160 volt main feeder buses, switchgear sections, and standby power buses switchgear sections are located in a Class I enclosure. The redundant engineered safeguards 4160 volt switchgear bus sections and their associated 600 volt switchgear bus sections, motor control centers, etc. are located within the turbine building and auxiliary building below the operating floor level. They are located in areas with separation and protection to minimize exposure to mechanical, fire, and water damage. This equipment is coordinated electrically to permit safe operation under normal and short circuit conditions. The engineered safeguards system is of Class I seismic design.

(d) The 600 volt motor control centers are located in the areas of electrical load concentration. Those associated with the turbine-generator auxiliary system in general are located below the turbine-generator operating floor level. Those associated with the nuclear steam supply system are located in the auxiliary building. Motor control centers are located in areas with separation and protection to minimize their exposure to mechanical, fire, and water damage.

9. (e) The 125 volt dc instrumentation and control power system batteries are physically separated in separate enclosures to minimize their exposure to any damage. The battery chargers and associated dc bus sections and switchgear are located in three separate rooms in the auxiliary building and physical separation is maintained between redundant equipment.

(f) Metal-enclosed buses are used for all major bus runs where large blocks of current are to be carried. They are also routed to minimize exposure to mechanical, fire, and water damage.

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(g) Control, instrumentation, and power cables are applied and routed to minimize their vulnerability to damage from any source. All cables are selected using conservative margins with respect to their current carrying capacities, insulation properties, and mechanical construction. Cable insulations in the Reactor Building are selected to minimize the effects of radiation, heat, and humidity. Appropriate instrumentation cables are shielded to minimize induced voltage and magnetic interference. Wire and cables related to engineered safeguards and reactor protective systems are routed and installed to maintain the integrity of their respective redundant channels and protect them from physical damage. Power and control cables for redundant auxiliaries or services are run by different routes to reduce any probability of an accident disabling more than one piece of redundant equipment. **Floor sleeves will be filled with**
18. **a fire retardant material.**

(h) It is our intent wherever physically possible to utilize metallically armored and protected cable systems. By this we mean the use of rigid and thin wall metal conduit, aluminum sheath cables, bronze armored control cables, steel interlocked armor power and control cables and either interlocked armor or served wire armored instrumentation cables. With this type construction fire stops as such are not required.

Overload protection for cables is very closely related to the basic power and control systems designs. 4KV power systems are protected by Electro-mechanical overcurrent relays and solid state type ground relays. The relays are selected for the loads protected and the cables are sized based on the maximum currents which these relays should allow without tripping for the loads they are protecting. 600 Volt load centers are used to feed individual motor control centers. The feeder breakers used are furnished
4. with long-time and instantaneous electro-mechanical short-time trip elements.
Cables to each breaker are sized in coordination with the trip elements selected for that particular breaker. Small motor loads at the 600 volt and 208 volt levels are generally handled through combination motor starters located in our motor control centers. Short circuit protection for the load is provided by molded case circuit breakers with magnetic trip devices while overcurrent protection is provided by standard starter overload elements sized for the application. On small ES motor loads two of the three overload elements are oversized for cable protection rather than motor protection and are wired in the contactor trip circuit. The third element is sized for motor protection but is wired to alarm only. This is based on the premise that the motor should operate even if motor damage does occur. Cable sizing is based on maximum service factor loading of the motor.

Power cable trays are loaded with a single layer of cable. Each cable is clamped in place with a spacing being maintained between cables equal to 1/4 the diameter of the cable. Power cables are derated based on IPCEA recommendations for Interlocked Armor Power Cables when installed with one-quarter cable diameter spacing in cable trays. The maximum fill in control and instrumentation cable trays is such that trays will be filled to the
16. top of the tray rails except in some Unit 1 locations. Units 2 and 3 cable trays will not be filled above the side rails. A minimum of five
13. inches rail to rail separation will be maintained between all vertical trays on Units 2 and 3.

Early cable tray requirements were based on types of cable which had been used in the past which were primarily not armored. Armored cable was used at Oconee to achieve better mechanical protection and fire retardance. This caused the trays to fill faster than anticipated and in several locations the fill became excessive. Steps have also been taken to insure that no additional
18. cables are routed through trays which are already overfilled.

Where overfill situations exist in Unit 1 between vertically adjacent cable trays to the extent that the top cable in the lower tray is within three (3) inches of the bottom cable in the tray immediately above, a 1/8" fire retardant
18. fiberglass reinforced polyester barrier will be placed between the trays. These barriers will be attached to the bottom of the upper tray and fitted around cables which may pass through the barrier.

Studies of heating due to I^2R loss in the cables were made. It was determined that the worse case was tray section 1ME8 which contained 322 cables. Cables were classed in three groups: control, control power and instrumentation. Losses were determined by conservative means and were found to be a total of 1.3 watts per lineal foot of tray. Assuming that one cable dissipates 36% of the total heat and that this cable is in the center of a nine inch pile of cable, its maximum temperature would be only 14°C above the ambient cable spreading room temperature, even though the insulation qualities of the cable pile were assumed to be almost perfect. No air flow
18. was assumed through the cables; therefore, the addition of barriers does not alter the heating calculations. Due to the small amount of heat generated and since all cable used in this area is rated 90°C, these temperatures will have no detrimental affect on adjacent cables or on cables in other trays.

Temperature measurements will be made periodically at ten selected locations
18. for the first year of operation. These locations will be where the tray overfill is the most severe.

Our criteria for routing cables requires that mutually redundant safety related cables be run in separate trays. Trays are spaced vertically in the cable room a minimum of 10 inches apart and in some cases redundant cables are in vertically adjacent trays. It should be pointed out that the cable armors used provide excellent mechanical and fire protection which would not be provided with conventional, unarmored cable systems. An early warning fire detection system has also been provided in this area.

Loadings and stresses in the cable tray and hangers were examined under both the steady state and seismic conditions. Hanger type HC-18, which is one of the most heavily laden hangers, was checked. It supports eight trays vertically, some of which are overfilled.

Original hanger calculations were based on the assumption that all hangers would be loaded at 200 pounds per tray. Under those conditions maximum stresses reached in any hanger member are 382 psi during steady state conditions and 11,100 psi under seismic loadings. This stress occurred in the angle brace which was added due to lateral seismic forces. Since the material for hangers and braces used is rated at 25,000 psi allowable,[1] hangers are stressed at less than 50% of their allowable under the worst conditions.

[1]Unistrut Corporation General Engineering Catalog No. 6, 1966, Page 11.

Recent calculations were made using existing loadings on the heavist loaded HC-18 hanger. Detailed inventory lists of all cables in each tray section have been maintained, and from this list it was determined that the hanger was loaded as follows:

Level H (top)	112.6 pounds
G	155.0 pounds
F	282.0 pounds
E	364.0 pounds
D	194.5 pounds
C	149.5 pounds
B	97.0 pounds
A (bottom)	196.0 pounds

Calculations with these loads show that stresses reached were 373 psi under steady state conditions and 11,000 psi during an earthquake. These stresses are actually slightly lower than the original calculation. This is due to several factors. First, although some trays are loaded heavier than the assumed 200 pounds per foot, some of the trays are considerably under the 200 pounds per foot. Secondly, many of the tray sections which are volume-wise overfilled are not overloaded from a weight standpoint because Oconee control cables have generally been random placed in the tray which has caused many voids to exist.

Overfilled trays were examined and it was determined that Section 1ME8 contains 120.4 pounds of cable per lineal foot. The tray manufacturers' safe load chart[2] states that 24" tray with 9" rung spacing will support a load of 215 pounds per foot with a 2.2 safety factor. The tray used has an ultimate strength of 2.2 x 215 = 473 pounds. With an existing load of 120.4 pounds the minimum safety factor is 3.8. Therefore, the present tray system is capable of supporting the weight of the cable even with the existing over-
18. filled conditions and the additional fire retardant barriers.

[2]Unistrut Corporation Catalog KUR4P-2, Page 16

In addition to detailed connection diagrams, detailed cable lists are developed for all cables. These cable lists identify each cable by cable type, specific cable routing by tray section number, and termination points. Protective system cables are identified as such on the cable lists. These lists are issued and are used by the field for cable installation. Each cable tray section is identified by tags bearing the tray section number assigned to it. Cables required for protective systems are identified as follows:

a. Power and control cables are color coded to identify their use and/or channel association. The color code is as follows:

Gray	Swgr 1TC Ld Ctr 1X8 MCC 1XS1 ESG channel 1, 3, 5, & 7 DC Pnlbd 1DIA Vital Pwr Pnlbd 1KVIA RPS Ch A
Yellow	Swgr 1TD Ld Ctr 1X9 MCC 1XS2 ESG channel 2, 4, 6, & 8 DC Pnlbd 1DIB Vital Pwr Pnlbd 1KVIB RPS Ch B
Blue	Swgr 1TE MCC 1XS3 DC Pnlbd 1DIC Vital Pwr Pnlbd 1KVIC RPS Ch C ESG Channel Even-Odd
Orange	DC Pnlbd 1DID Vital Pwr Pnlbd 1KVID RPS Ch D

b. All cables have their identifying number permanently affixed to both ends.

(i) Sensitive combustion gas detectors are located in areas with major cable concentrations as follows:

Cable rooms - four each
Equipment rooms - four each
Switchgear rooms - two each
Penetration rooms - two each
Equipment bays- two each
Reactor buildings - four each

The presence of combustion gases at a detector is annunciated in the control room which is associated with the area being monitored. The detectors are of the dual ion chamber type using radium sources. The system is self monitoring against circuit faults. Power for the systems is from a regulated source, the loss of which is monitored.

8.2.3 EMERGENCY POWER

8.3.2.1 Description of Power Sources to Each Unit

Each unit is provided with two physically independent circuits from the off-site power sources. One is the circuit from the switchyard through the startup transformer which is designed to be available within a few seconds following a loss of coolant accident. The second circuit is the path from the switchyard through the main step-up transformer, the main generator bus and the unit's auxiliary transformer with the generator disconnected from the main bus. This second circuit is designed to be available in time following a loss of coolant accident to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded and are consistent with the safety analysis of the unit as described in Section 14.1.2.8.3.

Each of the Oconee Nuclear Station units have six available sources of power as shown in Figure 8-2. These are:

36.

1. The 230 kV transmission system and/or the 500 kV system
2. Two Keowee hydro units
3. The 100 kV transmission system
4. The two other nuclear units

All of these power sources are available to supply emergency power to a unit, except the unit subjected to an accident. The normal arrangement is for three of these to serve any or all units and to be switched in the preferential order as follows: (1) the 230 kV transmission network through the unit startup transformers, (2) one Keowee hydro unit through an underground circuit, and (3) the other Keowee hydro unit through the overhead 230 kV circuit.

36.

Whenever the underground circuit from Keowee is unavailable, a circuit from the 100 kV transmission network can be connected to the standby buses and serve as an emergency power source.

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The 230 kV/500 kV system serves as the second off-site circuit to the individual units by disconnecting the generator from the main 19 kV bus and energizing the unit auxiliary transformer by backfeeding through the units main step-up transformer.

Each source has several degrees of redundance and reliability as outlined below.

(a) Generating Units:

As described in 8.2.2.2, the normal power supply to a unit's auxiliary
loads is provided through the unit auxiliary transformer connected to
36. the generator bus. Each unit's auxiliary transformer is sized to carry
its full load auxiliaries plus the engineered safeguards equipment of
another unit. This source of power is available except when:

1. The generating unit is in a normal shutdown condition, or

2. There is an in-station malfunction or failure of equipment preventing continued operation of the reactor-turbine-generator-auxiliary transformer combination.

36. 3. A 230 kV system blackout occurs resulting in a turbine-generator trip.

(b) 230 kV Transmission Network:

If power is not available from the unit's generator through the unit's
auxiliary transformer, power is supplied to the unit through its startup
transformer fed from either or both of the buses in the 230 kV switching
station. Provisions have been made to also allow disconnecting the unit's
generator from its main buses to allow backfeeding the units auxiliary
transformer from the 230 kV/500 kv switching station through the main
step-up transformer. Power to the startup transformer can flow through
the 230 kV switching station from any one of twelve supplies. These
include eight 230 kV transmission circuits (six installed with Unit 1;
36. two added with Unit 2; the nuclear generating unit if operating, the
other two nuclear generating units if operating and two hydroelectric
units. Each unit's startup transformer is sized to carry full load
auxiliaries for one nuclear generating unit plus the engineered safeguards equipment of another unit. This source of power is available
except when:

1. Both of the 230 kV buses in the switching station are unavailable, or

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2. There is a 230 kV system blackout, no nuclear generating unit is running, and neither hydro unit is capable of supplying power through its 230 kV connection; or

3. The startup transformer fails or their connection to the 230 kV switching station fails and the unit's auxiliary transformers or their backfeeding circuitry are not available.

(c) Keowee Hydro Units:

Upon loss of all sources of power described in (a) and (b) above, power is supplied from both of the 87,500 kVA units of the Keowee Hydro Station through two separate and independent routes. One route is a 4000 ft. underground 13.8 kV cable feeder to transformer CT4 at Oconee Nuclear Station which supplies the redundant 4160 volt standby power buses. The second route is a 230 kV transmission line to the 230 kV switching station at Oconee which supplies each unit's startup transformer. The 13.8 kV underground feeder is connected at all times to one of the two hydro units on a predetermined basis. The 230 kV feeder is automatically synchronized to the 230 kV switching station during normal system conditions. If the external system is in an abnormal condition, the automatic connection to the 230 kV switching station is delayed until the switching station is isolated automatically from the system.

Both units are started automatically by independent and redundant signals. These units can be started without any external source of power. They are available to carry load within 23 seconds upon occurrence of any of the following abnormal incidents:

1. Engineered safeguards required.

2. System network undervoltage.

3. System network underfrequency.

In addition both can be started by the operator in either Oconee Nuclear Station control room.

26. During all periods when the Keowee units are available for emergency power service, the Keowee Hydro Headgate will be rigidly fastened to assure that failure of the hoist system will not permit the gate to move into the closed position.

The Keowee Hydro Station contains two units rated 87,500 kVA each which generate at 13.8 kV. Each generator is connected to a common 230 kV stepup transformer through a 13.8 kV metal-enclosed bus and synchronizing air circuit breaker. The 13.8 kV underground emergency power feeder for the nuclear station is arranged with double air circuit breakers so that it can be connected to either 13.8 kV generator bus. The connection to the generator bus is made with metal-enclosed bus. At the nuclear station a transformer located in a Class I enclosure will transform the voltage to 4160 volts. If one hydro unit is out for maintenance, the other unit is available for service. The two units are served by a common tunnel-penstock, and unwatering for tunnel or scroll case maintenance will make both units unavailable. Based upon Duke's experience since 1919 with a hydro station similarly arranged, it is expected that unwatering frequency will be about one day per year plus four days every tenth year.

The 13.8 kV underground feeder and associated transformer are sized to carry full engineered safeguards auxiliaries of one unit plus auxiliaries for safe shutdown of the other two units. Power from the hydro units is available except when:

1. Both units are out of service, or

2. There is a coincident failure of the 13.8 kV underground cable feeder and a complete outage of the 230 kV feed through the switching station.

(d) 100 kV Transmission Network:

Whenever there is inadequate power from sources described in (a) through (c) above, power is available from a 100 kV transmission line connected to a separate transformer located on the opposite side of the station from the 230 kV facilities. This single 100 kV circuit is connected to the 100 kV transmission system through the substation at Central located eight miles from Oconee. Central substation is connected to Lee Steam Station twenty two miles away through a similar 100 kV line. Located at Lee Steam Station are
32. two 44.1 MVA combustion turbines. One of these two combustion turbines can be started in 15 minutes and connected to the 100 kV line. This 100 kV line can be isolated from the rest of the system and supply emergency power to Oconee. The transformer is sized to carry all the engineered safeguards auxiliaries of one unit plus the shutdown loads of the other two units. This source of power is available except:

1. When the 100 kV line or transformer is out of service, or

2. Temporarily after a complete system blackout of all transmission facilities.

(e) The tabulation of loads to be supplied power from the emergency power sources is shown in Table 8-5.

8.2.3.2 <u>Power to Vital Loads</u>

The power sources supply power through transformers to the redundant 4160 volt main feeder buses and circuit breakers which supply power to the three redundant 4160 volt switchgear bus sections that serve the engineered safeguards auxiliaries and reactor protective systems. The engineered safeguards auxiliaries and reactor protective systems are arranged so that a failure of any single bus section does not prevent the respective systems from fulfilling their protective functions. On loss of their normal sources of power, the engineered safeguards auxiliary buses are transferred as described in 8.2.2.4 to alternate sources of power in the following preferential sequence:

a. Transfer from unit auxiliary transformer to startup transformer where:

1. Power is supplied from the 230 kV transmission system, or

2. Power is supplied from the two Keowee hydro units.

b. Transfer to 4160 volt standby power buses where:

    1. Standby power is supplied from one Keowee hydro unit, or

    2. Standby power is supplied from the 100 kV transmission line.

The control system is designed to prevent the paralleling of two sources during the switching operation and is similar to the transfer systems Duke has used for many years in their fossil fired plants.

Upon loss of the unit auxiliary transformer source and startup transformer source, and in the absence of an engineered safeguards (ESG) signal, the following occurs:

1. The turbine-generator and reactor are tripped and the main feeder buses become deenergized. Control power is still available from the dc and vital power systems.

2. Both of the Keowee hydro units are started and either is connected to the standby power buses from which power can be supplied to the shutdown auxiliaries.

3. The non-essential loads are shed.

4. The equipment required to bring the reactor to a hot shutdown is energized.

Logic and control circuits will be fed without interruption from dc sources and vital power buses.

8.2.3.3 Reliability Considerations

The emergency electric power system provided for each nuclear generating unit possesses certain inherent design features which improve its reliability over limited capacity split-bus arrangements usually provided in nuclear power plants.

The basic design criteria of the entire emergency electric power system of a nuclear unit, including the generating sources, distribution system and controls, is that a single failure of any component passive or active will not preclude the system from supplying emergency power when required. Special provisions have been employed to accomplish this which include a double bus - double breaker distribution system, redundant circuit breaker trip coils and circuits, diverse protective relaying for each circuit breaker, redundant load shedding and transfer logic equipment, physical separation and other features.

The reliability afforded by the split bus concept is included in the design of the double bus - double breaker system employed here. Consideration has been given to the capacity of the emergency power sources, the method of switching, redundancy utilized and the protective features. For example, the electrical system together with the sources of electric power which are installed to supply emergency power to a nuclear unit possesses the following design features:

1. Each electric power source is extremely large for the requirements. For example, each of the redundant on-site Keowee hydro electric units is rated 87,500 kVA while the maximum combined load demand on one nuclear unit with a LOCA and the other two nuclear units in a hot shutdown condition is 16,730 kVA as shown in Table 8-5. The smallest of the emergency power sources is the connection to the 100 kV transmission system through transformer CT-5 which has a maximum continuous rating of 22,500 kVA. The significant effect of these large sources of emergency power is to improve the overall reliability of the electric system for several reasons; namely, the transient effects of both fault clearing and switching required loads to any of the sources is negligible. In addition, a greater number of plant auxiliaries may be run and used to help cope with an incident as well as shut down and maintain the other nuclear units in safe shutdown conditions as compared to nuclear plants which have limited capacity emergency power sources such as diesel generators.

2. The Keowee hydroelectric units, as described in (B) below are inherently reliable sources of power as proven by years of operating experience with similar generating units. Since they are stored energy type machines, their ability to start is far more reliable than the diesel generator which is an internal combustion type machine.

3. Each electric power distribution system is designed with redundant full capacity buses to match the capacity of the large emergency power source. This thereby provides two continuous sources of supply from the two full capacity main feeder buses to each of the three engineered safeguards load buses.

4. Reliability of the engineered safeguards load buses is assured by the following protective features:

   a. Engineered safeguards load bus faults or bus overload conditions are protected for by both ground fault overcurrent relays and phase overcurrent relays. These relays are provided on each load bus feeder breaker and function to open the associated breaker to isolate the load bus from the main feeder buses, thereby maintaining the integrity of the main feeder buses.

   b. Each load bus feeder breaker is also included in the zone of protection afforded by the main feeder bus differential current relays which would function to isolate a faulted breaker from any source of supply.

   c. Each load bus feeder breaker is provided with breaker failure protective relaying. This feature will initiate action to isolate the breaker from any source of supply if the breaker fails to open upon a protective relay trip. The maximum equipment this would remove from service is one load bus and one main feeder bus, leaving two load buses and the other main feeder bus to supply the required loads which are sufficient to perform the intended safety functions.

d. Each load bus feeder breaker is provided with redundant trip coils, supplied from separate dc supplies, assuring positive trip action.

With the above protective features plus their metalclad construction and the physical separation maintained, failure of any one of the three redundant load buses or components will not affect the ability of the other two load buses to supply their engineered safeguards loads.

5. Reliability of the main feeder buses and the standby buses are assured by the following protective features:

   a. Each main feeder bus and each standby bus is protected independently by differential current relays. These relays will sense any fault condition in the zone between the high side of the incoming bus feeder breakers to the low side of the outgoing feeder breakers. The outgoing feeder breakers on the standby bus would be the breakers connecting to the main feeder buses and they would have overlapping differential protection from both buses. The outgoing feeder breaker of the main feeder buses would be the feeder breakers to the engineered safeguards load buses. If a fault condition occurs, the relays will function to isolate the affected bus from all sources of supply by opening all circuit breakers associated with that bus. The other redundant bus will still provide the required power to all three engineered safeguards load buses.

   b. Each feeder breaker to each of the buses is protected with phase overcurrent and ground fault overcurrent protective relaying. These relays function to open the breaker and isolate the main feeder bus from the power source upon occurrence of these overcurrent conditions. This thereby maintains the integrity of the power source and allows the continued supply of power of the other bus and all three engineered safeguards load buses. The comparable condition on a split bus concept would cause the loss of one engineered safeguards bus.

   c. Each feeder breaker is also provided with breaker failure protective relaying. This feature will initiate action to isolate the breaker from any source of supply if the breaker fails to open on a protective relays trip. The maximum loss on this condition would be the connected source of supply and the associated bus. The other bus would transfer by the redundant transfer logic to the alternate source of supply and continue supplying power to all three engineered safeguards load buses. The maximum loss under the split bus concept would not only be the source of supply, but also the associated engineered safeguards load bus.

   d. Each feeder breaker is provided with redundant trip coils supplied from separate dc supplies, assuring positive trip action.

With the above protective features, their metal-enclosed construction and their physical separation, failure of any one of the redundant bus sections or components will not affect the ability of the other buses to supply the engineered safeguards loads.

9. The emergency power sources are independent of each other and switched on to the main feeder buses such that this independency is maintained. Paralleling of emergency power sources is prevented by redundancy in transfer logic equipment and interlocking. For example, each hydro generating unit and transformer is an independent entity in itself as discussed in 8.2.3.1c and (B) below.

In the event of an accident requiring engineered safeguards, and the simultaneous loss of the complete external transmission network; power is provided from either or both of the two on-site Keowee generating units. The independency and redundancy of these power sources are maintained by electrically
36. separate switchgear, feeders, transformers, and motor controls from the source to the engineered safeguards auxiliaries. Redundant instrumentation, control equipment, and vital power supplies are provided to assure reliability in selection and switching of the emergency power. The following is a more detailed description of these provisions made to assure emergency power availability to engineered safeguards.

A. Engineered Safeguards Auxiliary Buses:

The engineered safeguards are provided with redundancy as described in Section 6. To maintain this redundancy, the engineered safeguards equipment is connected to three redundant switchgear buses such that engineered safeguards equipment performing the same function is connected to different switchgear buses. Each of these switchgear buses is supplied from both of the redundant 4160 volt main feeder buses which are supplied from the redundant sources described in 8.2.2.4. In the event of an accident and the simultaneous loss of the external transmission network, the switchgear buses are supplied emergency power through both 4160 volt main feeder buses from either the 4160 volt startup transformers through their respective feeder breakers or from both of the redundant standby power buses. The standby power buses receive emergency power from either the Keowee Hydro Station or the 100 kV transmission line described in 8.2.3.4. In the event of a Loss of Coolant Accident (LOCA) any breakers supplying the engineered safeguards loads are closed automatically. In the event of a LOCA and the simultaneous loss of both the normal auxiliary source and the startup source, the non-essential load breakers are tripped. Redundant engineered safeguards load-shedding logic equipment assures positive shedding of non-essential equipment by energizing separate trip coils provided in their circuit breakers. Redundant engineered safeguards actuation channels initiate closing of the essential equipment feeder breakers.

B. Keowee Hydro Station:

Two 87,500 kVA generating units are installed and generate power at 13.8 kV. Except for the penstock, each unit is entirely independent of the other, consisting of its own turbine, governor system, generator, exciter, voltage regulator, generator circuit breaker, synchronizing equipment, protective relaying, automatic startup control equipment, manual controls, unit dc control battery, etc. Each unit is provided with its own automatic startup equipment located in separate cubicles within the Keowee control room. The initiation of startup is accomplished by control signals from the Oconee control area. Normal startup of either unit is by operator action while emergency startup is automatic. Both units are started automatically and simultaneously and run on standby on either of two conditions; namely, if the external transmission system is lost or if engineered safeguards action is required. If the units are already operating when either of the above conditions occur, they are separated from the network and continue to run on standby until needed. Each unit's voltage regulator is equipped with a volts-per-cycle limiting feature which permits it to accept full emergency power load as it accelerates from zero to full speed within 23 seconds from receipt of the emergency startup initiation signal.

On normal automatic startup, each unit is automatically connected and supplies power to the Oconee 230 kV switching station through the stepup transformer by its respective 13.8 kV generator circuit breaker. This is accomplished by the automatic synchronizing equipment of each unit. On emergency automatic startup, both units are started; the unit with the 13.8 kV underground feeder connected to it supplies that feeder and the other unit is available to supply the 230 kV Oconee switching station. If there is a system disturbance, this unit is connected automatically to the Oconee 230 kV switching station through interlocks
6. indicating the 230 kV switching station is isolated from the system and the bus voltage has decayed. The redundant External

Grid Trouble Protective Systems are provided to isolate the 230 kV switching station on failure of the external transmission network. Therefore, on loss of the external transmission network, both of the Keowee hydro units can provide emergency power to any of the Oconee units through either the 230 kV switching station to the unit's respective startup transformer or the 13.8 kV underground feeder and its 4160 volt transformer at Oconee. This 13.8 kV underground feeder is connected to the terminals of one hydro electric generator by air circuit breakers and is energized whenever that generator is in service in either emergency or normal mode.

Hydro units are extremely reliable sources of electric power, as proven by years of operating experience. The independent Keowee units, along with the alternate circuits, provide the required redundance to assure reliable emergency power. Storage capacity of the Keowee reservoir and naturally occurring minimum streamflow are such that the generating units can provide continuous emergency power following an accident. The Keowee reservoir, between its normal elevation and maximum planned drawdown, has sufficient storage which, when combined with minimum recorded streamflow on the Keowee River will permit a hydro unit to carry continuously one nuclear unit's emergency auxiliary loads for 126 days.

Additional redundant equipment provided to assure reliable control and operation of the units includes:

24. 1. Two separate and redundant groups of underground multiconductor control cables between the Keowee and the Oconee control room, each of which carries separate startup and control signals.

2. Redundant control initiating signal devices in the Oconee station to initiate manual and automatic startup of each of the Keowee generating units.

3. Redundant control receiving devices in the Keowee station to receive redundant signals to initiate manual and automatic startup of each of the Keowee generating units.

The failure analysis covering the Keowee Hydro Station is outlined in Table 8-6.

C. Emergency Power Switching:

Emergency power from the Keowee generating station is available through two circuits; the overhead 230 kV circuit through the startup transformers, and the underground 13.8 kV circuit through a transformer to the 4160 volt standby power buses and feeder circuit breakers. The power from the startup transformer circuits is available to each unit's redundant 4160 volt main feeder buses through redundant startup circuit breakers. The power from the 4160 volt standby power buses is available to each unit's main feeder buses through redundant circuit breakers. This arrangement assures that sufficient emergency power for engineered safeguards of any unit can be supplied from the 13.8 kV underground feeder from Keowee.

Redundant systems of emergency power switching equipment are provided to switch the emergency power to the unit's 4160 volt redundant main feeder buses. The redundant transfer logic will seek the most available source of power and when it becomes available close into it. If this source is then subsequently lost, the switching logic and equip-
6. ment will transfer to the other source automatically. Without off-site power, the power sources are as follows: (1) power from Keowee through the startup transformer and (2) power from Keowee through the 13.8 kV underground feeder to the standby buses.

In the event of a loss of coolant accident requiring engineered safeguards action, the following action takes place:

1. Both Keowee hydro units are started immediately. The unit not
18. connected to the 13.8 kV underground feeder is run on standby and connected to the 230 kV switching station when the switchyard is isolated.

2. The 13.8 kV underground circuit from Keowee becomes energized as the hydro unit to which it is connected is started.

3. The 4160 volt redundant main feeder buses of the unit with the accident are switched to the emergency power sources in the preferential order as described in 8.2.3.2.

4. The engineered safeguards of the unit with the accident are started and the non-essential loads are shed when power is unavailable from the normal or startup sources.

In the event the external transmission network is lost, the following action takes place:

1. Both Keowee hydro units are started immediately and the unit not connected to the 13.8 kV underground feeder is connected automatically to the 230 kV switching station by closing its respective generator circuit breaker and the 230 kV Power Circuit Breaker (PCB)-9 when the 230 kV switching station is isolated from the system network.

2. The 230 kV switching station is isolated automatically by energizing the dual trip coils of the 230 kV PCB's 8, 12, 15, 17, 20, 21, 23,
2. 24, 26, 29, 31, and 33.

3. The startup transformers No. CT1, CT2, and CT3 remain connected to the 230 kV switching station.

4. The 13.8 kV underground circuit from Keowee becomes energized as the hydro unit to which it is connected is started.

The failure analysis covering the emergency electrical systems is outlined in Table 8-7.

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8.2.3.4 Backup Power from Gas Turbines

In the unlikely circumstance that power is unavailable to any of the 4160 volt main feeder buses of Units 1, 2, and 3 from all the sources outlined in 8.2.3, a source of power will be made available to supply the shutdown loads of all three units through the standby power buses. This source consists of
32. one of two available 44.1 MVA gas turbine generating units located 30 miles away at the Lee Steam Station and arranged to supply power over the 100 kV transmission line from Lee to Oconee via Central shown in Figure 8-2. Under this circumstance, the 100 kV transmission line is isolated from the 100 kV transmission system to supply power solely to Oconee. The 100 kV transmission line is located above the level of any flood that is postulated on the Keowee River. On the Duke system, as on the West Coast, wind and ice loadings are more severe than seismic loadings and govern the structural design of transmission lines, including this 100 kV line. Should maintenance requirements make both Keowee hydro units unavailable, a Lee gas turbine can be brought to speed-no-load and directly connected to Oconee through the 100 kV line, and if an emergency occurs this 100 kV line can be separated from the balance of the transmission system.

8.2.3.5 Seismic Considerations

The seismic design basis for the emergency power system and its controls is based on the design or maximum hypothetical earthquake. The equipment is designed to assure it will not lose its capability to perform its intended function during and following the design bases event. If a seismic disturbance occurred after a major accident the emergency power system will perform its intended function.

4. The instrumentation and control devices are being evaluated to show their ability to withstand the seismic conditions in accordance with the design objective by either predicting the equipment's performance by analysis, testing the equipment to determine its resonance frequency and applying the applicable building response curve or testing the equipment under simulated seismic conditions. To assure that the seismic design is met on the equipment, the equipment or components comprising the equipment assemblies will be analyzed or tested by the vendors or private testing laboratories or Duke Power Company personnel using seismic loads as obtained from the building response calculations to show that stresses are within allowable limits and will not result in loss of function. Testing of this equipment will be done and reports compiled to verify the seismic design objectives.

A summary of the seismic design bases which apply to the Emergency Power System
6. is given in Table 8-8.

8.2.4 EMERGENCY LIGHTING SYSTEM

For each unit, two separate emergency lighting systems are provided; namely,
4. an emergency 250 volt dc lighting system and a separate engineered safeguards 208Y/120 volt ac lighting system. These two systems are separate and distinct.

8.2.4.1 Emergency 250 Volt DC Lighting System

The 250 volt dc lighting system, which is normally de-energized, provides operating level lighting in the control room and lighting at selected stairs and corridors in the Auxiliary, Turbine, and Reactor Buildings. The emergency lighting is energized automatically by an undervoltage sensing relay mounted on the individual panelboards located in their associated areas. Control power for the undervoltage transfer circuit is provided from the 250 volt dc station batteries. A test button is also provided at each panelboard to test the operability of the system without affecting normal lighting. All associated lighting units are incandescent.

8.2.4.2 Engineered Safeguards AC Lighting System

4. The engineered safeguards lighting system, which is normally de-energized, provides lighting in the following parts of the auxiliary building: control room, cable room, equipment room, stairs, exits, corridors, hot machine shop, spent fuel pool room, fuel unloading area, decontamination rooms, pump and tank room areas, fan and ventilation rooms of roof elevation, penetration rooms, and purge rooms. The stairs and platforms in the reactor building are also provided lighting to enable personnel to leave or enter the entire building. Power is provided from two engineered safeguards 600 volt ac control centers through two 600/208Y/120 volt ac dry type transformers which in turn feed each of two panelboards located in the equipment room area. The engineered safeguard lighting is energized automatically by undervoltage sensing relays monitoring the normal 600 volt ac feeder voltage.

8.3 TESTS AND INSPECTIONS

Remote startup of the Keowee generators is provided in each of the control rooms of the nuclear station. Provisions are made in the control rooms to
6. manually initiate a fast start of both of the two hydroelectric generators connecting the generator to the nuclear station's 4160 volt buses. Testing of this system may be scheduled any time the Keowee hydro units are not running.

The 100 kV, 230 kV and 500 kV circuit breakers are inspected, maintained and tested as follows:

(a) 100 kV transmission line circuit breakers are tested on a routine basis.

(b) 230 kV and 500 kV transmission line circuit breakers are tested on a routine basis. This is accomplished without removing the transmission line from service.

(c) 230 kV and 500 kV generator circuit breakers are tested with the generator in service.

Transmission line protective relaying is tested on a routine basis.

Generator protective relaying is tested when the generator is off-line.

The 4160 volt circuit breakers and associated equipment are tested in service by opening and closing the circuit breakers in a manner that does not interfere with the operation of the station. The circuit breakers are "jacked out" to a test position and operated without energizing the circuits, if necessary.

The 600 volt circuit breakers, motor contactors, and associated equipment are tested in service by opening and closing the circuit breakers or contactors so as not to interfere with operation of the station. The power circuits can be opened and the circuit breakers and motor contactors operated without energizing the circuits.

Emergency transfers to the various emergency power sources are tested on a routine basis to prove the operational ability of these systems. Associated normal, startup, and standby circuit breakers on one bus can be "jacked out" into test position and initiated manually for an emergency transfer test.

2. The ungrounded dc system has detectors to indicate when there is a ground existing on any leg of the system. A ground on one leg of the dc system will not cause any equipment to malfunction. Simultaneous grounds on two legs of the system may cause all energized equipment to drop out if the grounds are of sufficiently low resistance. This may be momentary if the grounded circuit is cleared by its circuit breaker or sustained if not. Grounds are located by a logical isolation of individual circuits connected to the faulted system, while taking the necessary precautions to maintain the integrity of the vital bus supplies.

TABLE 8-1

Single Failure Analysis for 125 Volt DC Instrumentation and Control Power System

	Component	Malfunction	Comments & Consequences
1.	600V AC Power Supply to Charger	Loss of power to one	No consequence - power from battery is available to supply power without interruption until standby charger is switched in.
2.	Battery Charger	Loss of power from one	(a) The 125 volt dc bus would continue to receive power from its respective battery without interruption except as in (c) below.
			(b) Standby battery charger is available.
			(c) Several internal faults may cause high short circuit currents to flow with the resulting voltage reduction on the 125 volt dc bus until the fault is cleared by the isolating circuit breakers. Complete loss of voltage on the 125 volt dc bus may result if the battery circuit breakers open. However, power to reactor protection systems and engineered safeguards instrumentation and control would be unaffected since they are supplied from redundant panelboards, each supplied from redundant feeders.
3.	125V DC Battery	Loss of power from one	(a) Those 125 volt dc control panelboards supplied from the affected bus will continue to receive uninterrupted power from their alternate power supplies through isolating diodes.

TABLE 8-1 (Cont'd)

		Component	Malfunction	Comments & Consequences
				(b) All power would be lost to the other loads supplied from the faulted bus; however, they are not associated with reactor instrumentation, protective systems, or engineered safeguards.
				(c) Isolating circuit breakers clear the battery from the bus on a fault condition thereby allowing the battery charger to continue supplying power to the connected loads.
	4.	125V DC Switchgear Buses P-N	Bus shorted	Same comment as 3a and 3b.
2.	5.	125V DC Switchgear Bus 1DCA, 1DCB	Grounding a single bus	(a) The 125 volt dc system is an ungrounded electrical system. Ground detector equipment monitors and alarms a ground anywhere on the 125 volt dc system. A single ground will not cause any malfunction or prevent operation of any safety feature.
2.	6.	125V DC Switchgear Bus 1DCA, 1DCB	Gradual decay of voltage on one bus	(a) Each 125 volt bus will be monitored to detect the voltage decay on the bus and initiate an alarm at a setting above a voltage where the battery can deliver power for safe and orderly shutdown of the station. Upon detection, power will be restored either by correcting the deficiency by switching to a redundant source or by employing one of the redundant circuits.
	7.	DC Switchgear Load Feeder Cables	Cables shorted	(a) Same comments as 3a and 3b.

TABLE 8-1 (Cont'd)

	Component	Malfunction	Comments & Consequences
			(b) All feeder cables are provided with isolating circuit breakers that would isolate the "shorted" cable on a sustained fault condition.
	8. Isolating Diodes	Failure of one	(a) If the diode fails "shorted" then the other series diodes will still provide adequate isolation and power will be uninterrupted.
			(b) If the diode fails "open" then the other redundant supply through its isolating diodes will continue to supply power without interruption.
2.	9. 125V DC Control Power Panelboard 1DIA, 1DIB, 1DIC or 1DID	Bus shorted	(a) Voltage on two of the 125 volt dc bus systems will decay until isolated by the isolating circuit breakers causing consequences same as comments 3a and 3b.
			(b) For one unit, one-half of control and instrumentation power will be lost; however, they are not associated with reactor instrumentation and protective systems or engineered safeguards.
			(c) For one unit, one-half of 6900 volt switchgear closing control power would be lost but dual trip coils and redundant tripping power supplies are provided.
2.			(d) For one unit, one-third of the 4160 volt switchgear closing control power would be lost. Dual trip coils and redundant tripping control power are provided. The remaining redundant switchgear is adequate and is supplied control power from the other dc panels.

TABLE 8-1 (Cont'd)

Component	Malfunction	Comments & Consequences
		(e) For one unit, the 4160 volt main feeder bus circuit breakers on only one of the two buses would lose closing control. All 4160 volt circuit breakers have redundant trip coils and power supplies. The remaining main feeder bus and circuit breakers are supplied control power from the other dc panels, permitting the switching of 4160 volt emergency power to any unit.
		(f) For one unit, one-half of the 600 volt load centers will lose dc control power; however, each load connected to the load centers is dual fed from the redundant 600 volt load centers.
		(g) One static inverter would be lost and power to one instrument bus would be lost temporarily until a manual transfer could be made to a regulated instrument bus. The temporary loss of one vital instrument bus would result in the temporary loss of one channel of reactor protection and instrument systems and engineered safeguards systems. Other remaining channels will receive vital instrument control power from the other panelboards.

TABLE 8-2

Single Failure Analysis for 125 Volt DC Switching Station Power System

	Component	Malfunction	Comments & Consequences
1.	480V AC Power Supply to Charger	Loss of power to one	No consequence - power from battery is available to supply power without interruption.
2.	Battery Charger	Loss of power from one	(a) The 125 volt dc bus would continue to receive power from its respective battery without interruption except as in (c) below. (b) Standby battery charger is available. (c) Several internal faults may cause high short circuit currents to flow with the resulting voltage reduction on the 125 volt dc bus until the fault is cleared by the isolating circuit breakers. Complete loss of voltage on the 125 volt dc bus may result if the battery circuit breakers open. However, redundant protective relaying and panelboards are provided and are supplied from the other redundant 125 volt dc bus.
3.	125V DC Battery	Loss of power from one	Only those 125 volt dc control panelboards supplied from the affected bus will be lost. However, the redundant panelboards supplied from the other 125 volt dc bus would be unaffected and continue to provide power for protection and control.
4.	DC Switchgear Buses P-N	Bus shorted	Same comment as 3.

TABLE 8-2 (Cont'd)

		Component	Malfunction	Comments & Consequences
2.	5.	125V DC Bus SY-1, SY-2	Grounding a single bus	(a) The 125 volt dc system is an ungrounded electrical system. Ground detector equipment monitors and alarms a ground anywhere on the 125 volt dc system. A single ground will not cause any malfunction or prevent operation of any safety feature.
2.	6.	125V DC Bus SY-1, SY-2	Gradual decay of voltage on one bus	(a) Each 125 volt bus is monitored to detect the voltage decay on the bus and initiate an alarm at a setting above a voltage where the battery can deliver power for safe and orderly shutdown of the station. Upon detection power will be restored by correcting the deficiency.
	7.	DC Switchgear Load Feeder Cables	Cables shorted	Same comments as 3.
	8.	125V DC Primary or Backup Panelboards	Bus shorted in one	(a) Voltage on associated 125 volt dc bus will decay until isolated by isolating circuit breakers. (b) Protective relaying connected to the affected panelboards would be lost; however, redundant protective relaying supplied from the other 125 volt dc bus would provide protection. (c) One source of control power would be lost to the switching station power circuit breakers; however, a redundant source of control power is provided from the other 125 volt dc bus.

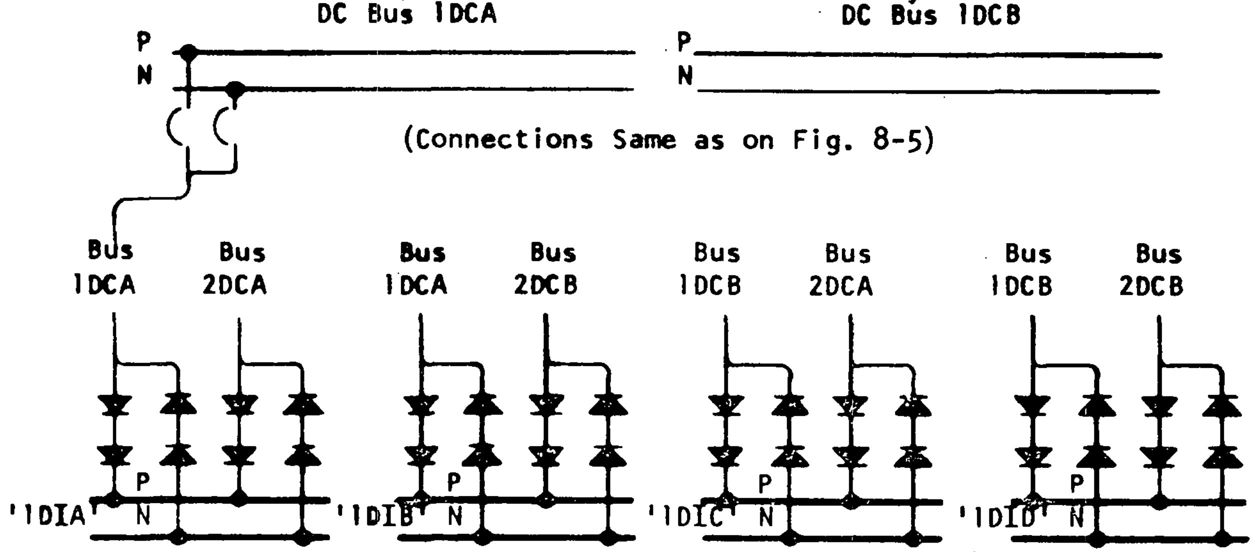
TABLE 8-3

Single Failure Analysis for the 120 Volt AC Vital Power System

		Component	Malfunction		Comments & Consequences
2.	1.	125V DC Control Power Panel Board 1DIA, 1DIB, 1DIC, or 1DID	Bus shorted		One static inverter would be lost and power to one instrument bus would be lost temporarily until a manual transfer could be made to a regulated instrument bus. The temporary loss of one vital instrument bus would result in the temporary loss of one channel of reactor protection and instrument systems and engineered safeguards systems. Other remaining channels will receive vital instrument control power from the other panelboards.
	2.	Static Inverter Feeder Cable	Failure		Same as comment 1.
	3.	Static Inverter	Failure		Same as comment 1.
2.	4.	Vital Instrument Power Panelboard 1KVIA, 1KVIB, 1KVIC or 1KVID	Failure of one	(a)	For any one bus failure only one channel of any system associated with reactor instrumentation and protective systems or engineered safeguards would be lost. Sufficient redundant channels supplied from other vital instrument buses would provide adequate protection.

TABLE 8-4

125 Volt DC Panelboard Fault Analysis



FAULT	EQUIPMENT LOSS	PNLB'S AFFECTED	COMMENTS
Pnlbd 1DIA BUS	Pnlbd 1DIA & (No. 1DCA D.C.Bus) (No. 2DCA D.C.Bus)	1DIB,3DIA,3DIB 1DIC,2DIA,2DIB	All would receive power from redundant DC buses in Unit 1,Unit 2 or Unit 3(see below)
Pnlbd 1DIB Bus	Pnlbd 1DIB & (No. 1DCA D.C.Bus) (No. 2DCB D.C.Bus)	1DIA,3DIA,3DIB 1DID,2DIC,2DID	All would receive power from redundant DC buses in Unit 1,Unit 2 or Unit 3(see below)
Pnlbd 1DIC Bus	Pnlbd 1DIC & (No. 1DCB D.C.Bus) (No. 2DCA D.C.Bus)	1DID,3DIC,3DID 1DIA,2DIA,2DIB	All would receive power from redundant DC buses in Unit 1,Unit 2 or Unit 3(see below)
Pnlbd 1DID Bus	Pnlbd 1DID & (No. 1DCB D.C.Bus) (No. 2DCB D.C.Bus)	1DIC,3DIC,3DID 1DIB,2DIC,2DID	All would receive power from redundant DC buses in Unit 1,Unit 2 or Unit 3(see below)

Panelboard No.	Unit #1				Unit #2				Unit #3			
	1DIA	1DIB	1DIC	1DID	2DIA	2DIB	2DIC	2DID	3DIA	3DIB	3DIC	3DID
Supplied from bus	1DCA 2DCA	1DCA 2DCB	1DCB 2DCA	1DCB 2DCB	2DCA 3DCA	2DCA 3DCB	2DCB 3DCA	2DCB 3DCB	3 DCA 1 DCA	3 DCA 1 DCB	3 DCB 1 DCA	3 DCB 1 DCB

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TABLE 8-5

Loads to be Supplied from the Emergency Power Sources

I. Auxiliaries for a Single Unit Actuated by Engineered Safeguards Signals

Auxiliary	Number and Size	Actuated on ESG	Required Following LOCA	Required During and Following Hot Shutdown
H.P. Injection Pump	3 @ 600 hp	(3) 1800 hp	(1) 600 hp	(1) 600 hp
L.P. Injection Pump	3 @ 400 hp	(2) 800 hp	(1) 400 hp	(1) 400 hp
L.P. Injection Valves	2 @ 20 hp	(2) 40 hp	0	0
L.P. Service Pump	2 @ 600 hp	(2) 1200 hp	(1) 600 hp	(1) 600 hp
R.B. Spray Pump	2 @ 250 hp	(2) 500 hp	(1) 250 hp	0
R.B. Cooling Fans	3 @ 150 hp	(3) 450 hp	(2) 300 hp	(2) 300 hp
Penetration Rm. Vent Fans	2 @ 5 hp	(2) 10 hp	(1) 5 hp	0
Motor Operated Valves	(39)100 hp	(39) 100 hp	0	0
		4900 hp	2155 hp	1900 hp
		(4778 kVA)	(2101 kVA)	(1853 kVA)

TABLE 8-5 (Cont'd)

II. Auxiliaries for a Single Unit Actuated at the Option of the Station Operator

Auxiliary	Number and Size	Actuated on ESG	Operated Following LOCA	Operated During and Following Hot Shutdown
Cond. Circ. Pump	4 @ 1750 hp	0	(1) 1750 hp	(1) 1750 hp
Spent Fuel Cool. Pump	2 @ 40 hp	0	(2) 80 hp	(2) 80 hp
Recirc. Cool. Wtr. Pump	4 @ 150 hp	0	0	(1) 150 hp
Moisture Separator Pump	2 @ 15 hp	0	0	(1) 15 hp
Reheater Drain Pump	2 @ 10 hp	0	0	(1) 10 hp
Comp. Cooling Pump	2 @ 60 hp	0	0	(1) 60 hp
Reac. Coolant Oil Pump	4 @ 10 hp	0	(4) 40 hp	(4) 40 hp
Borated Wtr. Stor. Tank Htr.	2 @ 50 kw	0	0	(2) 100 kw
R.C. Bleed Transf. Pump	2 @ 20 hp	0	0	(2) 40 hp
Low Act. Aux. Bldg. Sump P.	2 @ 5 hp	0	(2) 10 hp	(2) 10 hp
High Act. Aux. Bldg. Sump P.	2 @ 5 hp	0	(2) 10 hp	0
Waste Gas Compressor	2 @ 25 hp	0	(2) 50 hp	0
Turning Gear Motor	1 @ 60 hp	0	(1) 60 hp	(1) 60 hp
Turning Gear Oil Pump	1 @ 50 hp	0	(1) 50 hp	(1) 50 hp
Stm. Packing Exh. Mtr.	1 @ 20 hp	0	(1) 20 hp	(1) 20 hp
Air Compressor	3 @ 125 hp	0	(2) 250 hp	0

TABLE 8-5 (Cont'd)

II. (Cont'd)

Auxiliary	Number and Size	Actuated on ESG	Operated Following LOCA	Operated During and Following Hot Shutdown
Air Cond. Sys. Chiller Comp.	2 @ 350 hp	0	(1) 350 hp	(1) 350 hp
Air Cond. Sys. Chill. Wtr. Pump	2 @ 30 hp	0	(1) 30 hp	(1) 30 hp
Air Cond. Sys. Fan	3 @ 20 hp	0	(1) 20 hp	(1) 20 hp
Stator Cool. Wtr. Pumps	2 @ 75 hp	0	(2) 150 hp	(2) 150 hp
Main Seal Oil Pump	1 @ 10 hp	0	(1) 10 hp	(1) 10 hp
Recirc. Seal Oil Pump	1 @ 5 hp	0	(1) 5 hp	(1) 5 hp
Polish. Demin. Holding Pump	5 @ 3 hp	0	(5) 15 hp	(5) 15 hp
Elevators (RB and TB)	1 @ 75 kVA	0	(1) 75 kVA	(1) 75 kVA
Lighting	700 kVA	0	700 kVA	700 kVA
H.P. Serv. Water Pump	2 @ 500 hp	0	(2) 1000 hp	0
Control & Cable Rm. Lgt.	75 kW	0	75 kW	75 kW
Battery & Vital Pwr. Sys.	100 kVA	0	100 kVA	100 kVA
Regulated Inst. Power Sys.	50 kW	0	50 kW	50 kW
Boric Acid Mix Tank Htr.	2 @ 25 kW	0	0	(2) 50 kW
		0	(4937 kVA)	(4123 kVA)

TABLE 8-5 (Cont'd)

III. Combined Load Demand for Station

Unit with LOCA (Actuated on ESG)				4778 kVA
2nd Unit in Hot Shutdown:				
a) Required during & following Hot Shutdown		1853		
b) Operated during & following Hot Shutdown		4123		
	Subtotal	5976		5976 kVA
3rd Unit in Hot Shutdown:				
a) Same as 2nd Unit		1853		
b) Same as 2nd Unit		4123		
	Subtotal	5976		5976 kVA
			Total	16,730 kVA

IV. Source Size

Two Keowee Hydro Units 2 @ 87.5 MVA = 175 MVA

Startup Transformers (CT1, CT2, CT3) = 45/60 MVA Each

Standby Transformers (CT4, CT5) = 12/16/20 MVA Each

TABLE 8-6

Single Failure Analysis for the Keowee Hydro Station

	Component	Malfunction	Comments and Consequences
1.	Keowee Hydro Units	Loss of one	(a) One emergency power source would be lost; however, the other unit would supply 100% of emergency power load.
			(b) If the 13.8 kV underground feeder were connected to the unit which was lost, it would also be lost; however, the other unit would supply power through the stepup transformer and the 230 kV switching station to the startup transformers and the underground feeder could be transferred by the Oconee operator to the running unit.
2.	Generator Circuit Breakers and Buses	Loss of one	Same as 1 above.
3.	Stepup Transformer, Low Side Buses, 230 kV Overhead Line and PCB-9	Loss of one	Both hydro units would be separated from the 230 kV switching station; however, one hydro unit would supply emergency power through the 13.8 kV underground circuit.
4.	13.8 kV Underground Feeder Circuit Breaker, Cables, or Transformer	Loss of one	One circuit of emergency power would be lost; however, both hydro units could supply emergency power over the 230 kV overhead line.
5.	Keowee Hydro Unit Automatic Startup and Unit Control Systems	Loss of one unit's system	Same as 1 above.

TABLE 8-6 (Cont'd)

	Component	Malfunction	Comments and Consequences
6.	Keowee Hydro Unit 125V DC Control Battery, Panelboard, Feeders, etc.	Loss of one	Same as 1(a) above.
7.	Keowee Hydro Unit Emergency Startup and Switching Logic	Loss of one	Same as 1(a) above.
8.	Keowee Hydro Unit Emergency Startup and Switching Cir cuits from Oconee	Loss of one	No consequence, since independent and redundant underground signal cables are provided.

TABLE 8-7

Single Failure Analysis for the Emergency Electrical Power Systems

	Component	Malfunction	Comments and Consequences
1.	Any 230 kV Bus, PCB-18,27 and PCB-30, 230 kV Circuit to Startup Transformers, CT1, CT2, CT3, and Associated Buses	Loss of one	On loss of the yellow bus, the 230 kV emergency power circuit would be lost until the Oconee operator could reroute this supply in the switching station. However, emergency power would be available through the 13.8 kV underground circuit from one of the Keowee units. Other singular losses would have no consequence.
2.	230 kV Power Circuit Breaker Trip Coils or 125V DC Trip Coil Power Supply	Loss of one	No consequence as two trip coils are provided for each circuit breaker and each trip coil is provided with a separate 125 volt dc control circuit.
3.	13.8 kV Underground Circuit from Keowee Hydro or Transformer No. CT4	Loss of one	One circuit of emergency power would be lost; however, both hydro units could supply emergency power over the 230 kV overhead line.
4.	4160V Main Feeder Buses, 4160V Standby Power Buses and Feeder Circuit Breakers	(a) One circuit breaker fails to close when required to supply emergency power.	No consequence, as sufficient redundant circuit breakers and buses are provided with redundant switching logic.
		(b) One bus section faults	No consequence as sufficient redundant circuit breakers and buses are provided with redundant switching logic.
5.	4160 kV Auxiliary Switchgear Bus Sections	Loss of one	Same as 4(a) above.

TABLE 8-7 (Cont'd)

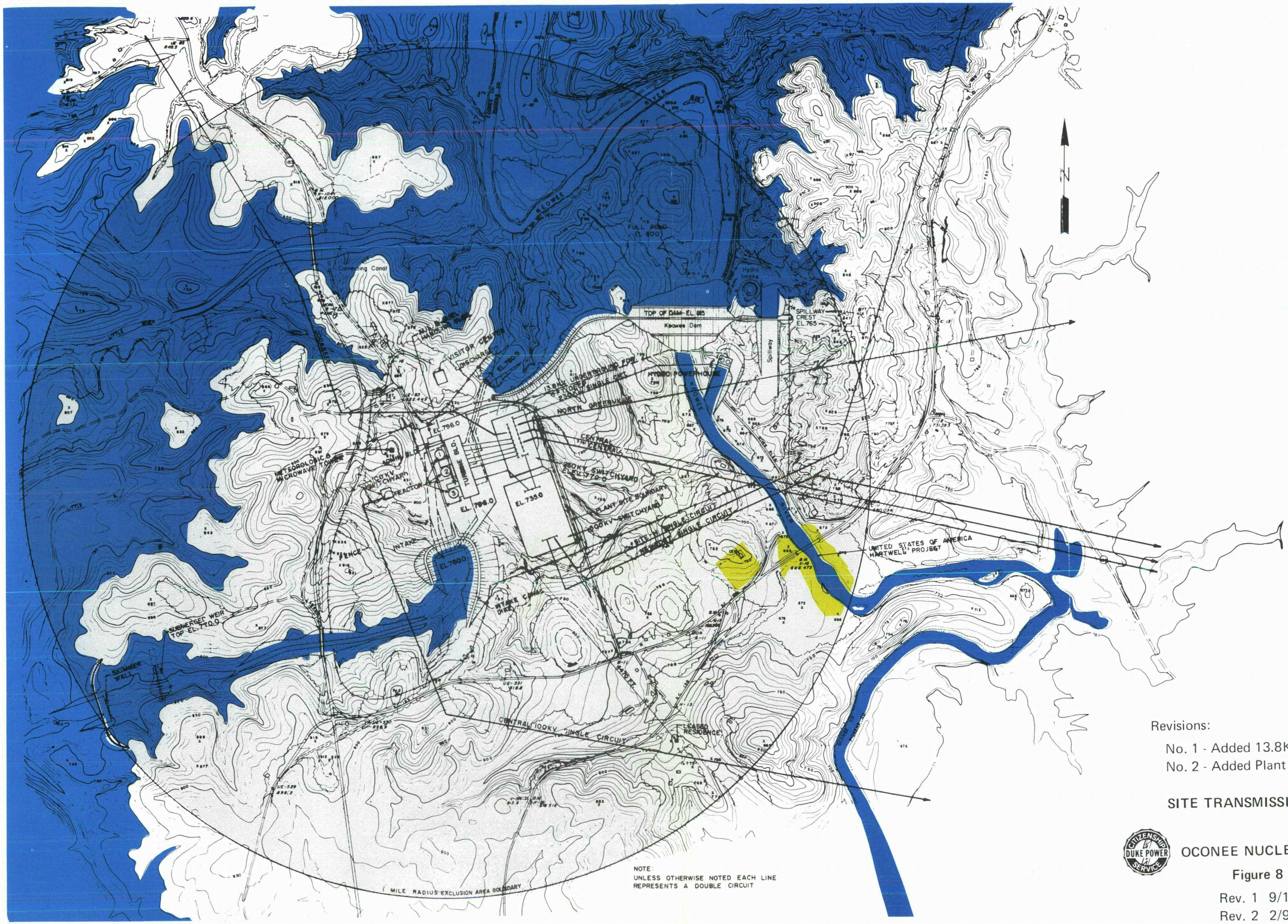
	Component	Malfunction	Comments and Consequences
6.	600V Auxiliary Switchgear Bus Sections	Loss of one	One 600 volt bus section containing engineered safeguards would fail to receive emergency power; however, sufficient redundant engineered safeguards will be supplied from the remaining redundant buses to perform the engineered safeguards function.
7.	125V DC System	Single failures	See 8.2.2.7.5, 8.2.2.7.6 and Table 8-1 and 8-2 for single failure analysis.
8.	120V AC Vital Power Buses	Single failures	See8.2.2.8.1 and Table 8-3 for single failure analysis.

Table 8.8
Summary of Seismic Considerations Applied to the Emergency Power System

EQUIPMENT	LOCATION	SEISMIC DESIGN BASES	TESTS AND ANALYSES
4.16 KV Station Auxiliary Switchgear	Turbine Building Mezzanine Floor (Elevation 796'6")	Maximum floor acceleration[1] at the base of equipment is 0.25g with the critical frequencies[2] between 1.5 and 13.2 cycles per second.	Representative switchgear frames were tested at frequencies of 4, and 10 cycles per second with accelerations up to and including 3.1g with no problems.
600 Volt Load Centers	Auxiliary Building Equipment Room (Elevation 796'6"	Maximum floor acceleration at the base of equipment is 0.15g with the critical frequencies between 0.7 and 18 cycles per second	Representative load center assemblies were tested at frequencies of 4, 6 and 11 cycles per second with accelerations up to and including 3.3g. The natural frequency survey of the equipment showed that there was no appreciable resonant conditions of the structure frame from 5 through 33 cycles per second (test machine limits); however, one panel on the equipment has a resonant frequency of 19 cycles per second which yielded 4g acceleration with 3.3g on the switchgear base without problems in the panel.
Motor Control Centers and D. C. Distribution Centers	Auxiliary Building Equipment Room (Elevation 796'6")	Maximum floor acceleration at the base of equipment is 0.15g with the critical frequencies between 0.7 and 18 cycles per second.	Representative motor control centers and D.C. distribution center sections are tested between 9.68 to 12.16 cycles per second horizontal and 10 to 16.6 cycles per second vertical with acceleration up to and including 3g.
A.C. & D.C. Panelboards	Auxiliary Building Equipment Room (Elevation 796'6")	Maximum floor acceleration at the base of equipment is 0.15g with the critical frequencies between 0.7 and 18 cycles per second.	Representative equipment was tested at frequencies of 4, 6, 8.5 an 10 cycles per second with accelerations up to and including 1.8g with no problems.
Control Batteries and Racks	Auxiliary Building Battery Room Elevation 809'3"	Maximum floor acceleration at the base of equipment is 0.25g with the critical frequencies between 1.5 and 13.2 cycles per second.	Representative cells are tested between 4 and 11 cycles per second with accelerations in excess of 0.25g. The adequacy of the rack design is verified by calculation.
Battery Chargers, Inverters & Isolating Diode Assemblies	Auxiliary Building Equipment Room (Elevation 796'6")	Maximum floor acceleration at the base of the equipment is 0.15g with the critical frequencies between 0.7 and 18 cycles per second.	Representative equipment is tested at frequencies between 4 and 11 cycles per second with accelerations in excess of 0.15g.

NOTES:

1. Maximum floor acceleration shall be defined as the peak acceleration[3] of that floor as a result of the maximum hypothetical earthquake at the base of the supporting structure.
2. Critical frequency shall be defined as that range of frequencies on the respective response spectrum in which there exists an acceleration greater than the maximum floor acceleration.
3. Peak floor acceleration has been determined by an acceleration time history of the supporting structure.



Revisions:

No. 1 - Added 13.8KV Underground Feeder

No. 2 - Added Plant Site Boundary Fence

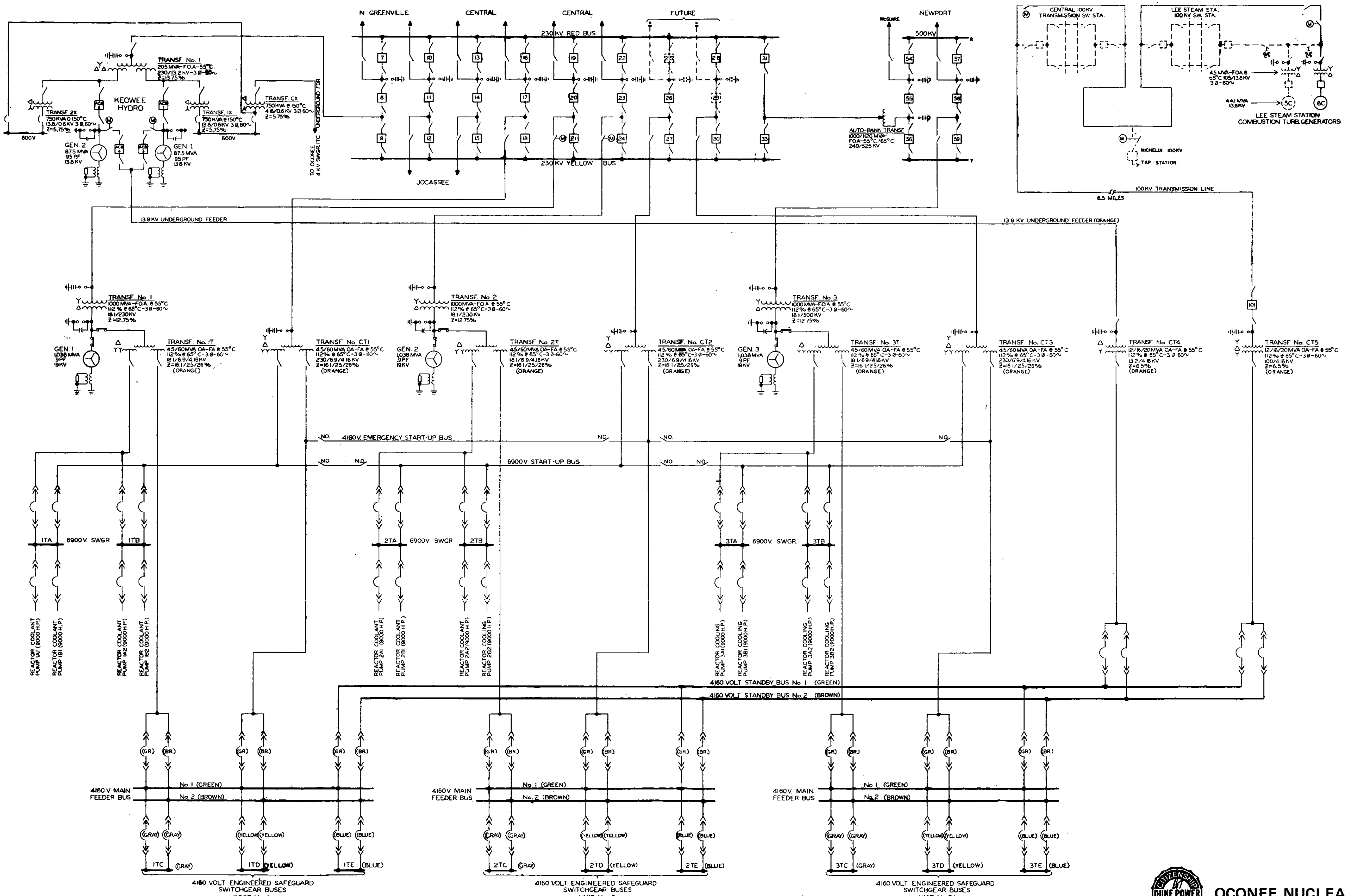
SITE TRANSMISSION MAP

OCONEE NUCLEAR STATION

Figure 8 - 1

Rev. 1 9/15/69

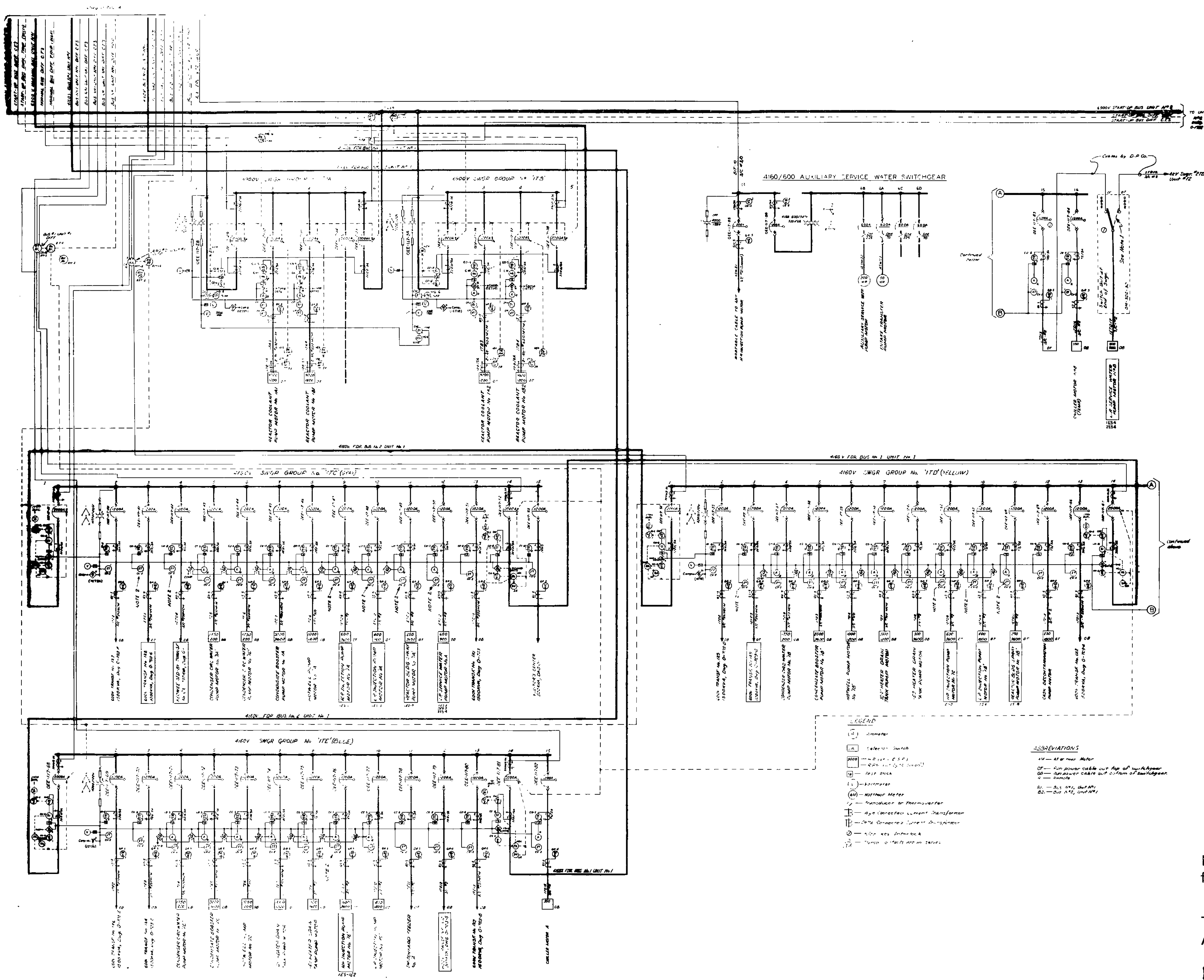
Rev. 2 2/9/70



OCONEE NUCLEAR STATION

FIGURE 8-2

Rev. 35. 9/30/74



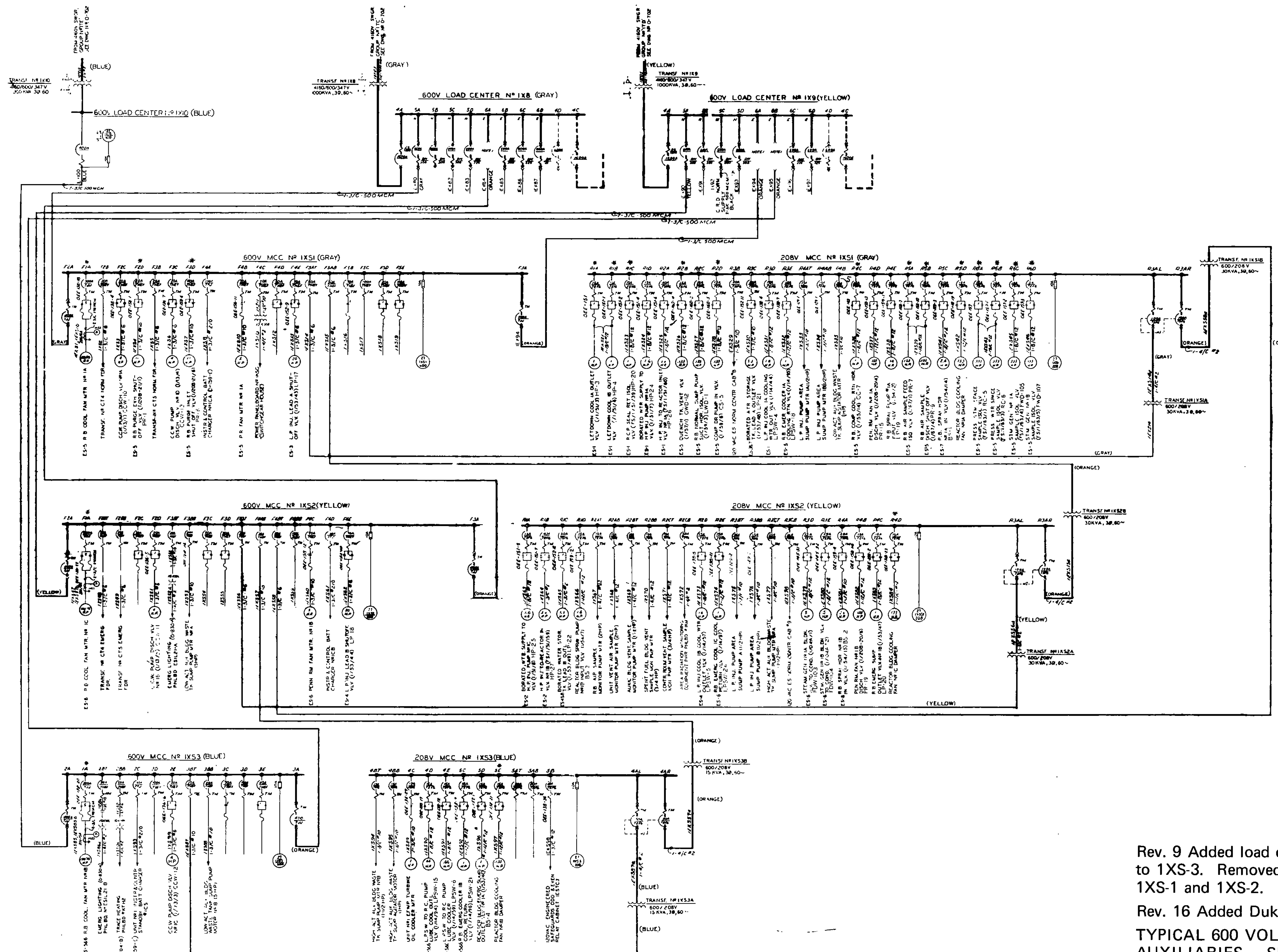
Rev. 9 Added feeder to 1XS-10 load center from switchgear 1T-E .

TYPICAL 6900 VOLT AND 4160 VOLT UNIT AUXILIARY – SINGLE LINE DIAGRAM

OCONEE NUCLEAR STATION

Figure 8 - 3

Rev. 9 8/11/70

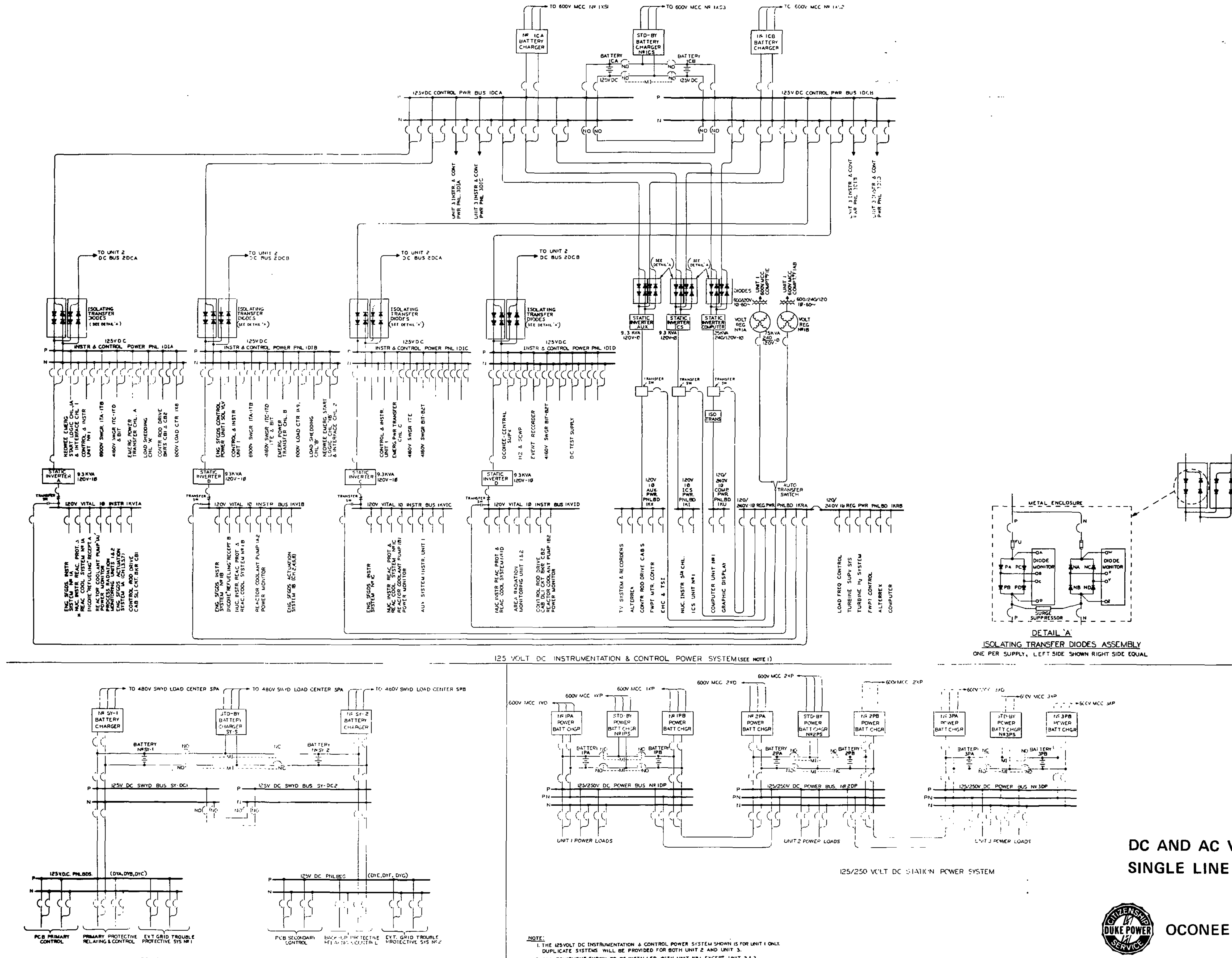


Rev. 9 Added load center 1XS-10 of feeder to 1XS-3. Removed second feeder breakers to 1XS-1 and 1XS-2.

Rev. 16 Added Duke value numbers

TYPICAL 600 VOLT AND 208 VOLT ESG AUXILIARIES – SINGLE LINE DIAGRAM

OCONEE NUCLEAR STATION
Figure 8 - 4
Rev. 9 8/11/70
Rev. 16 7/30/71



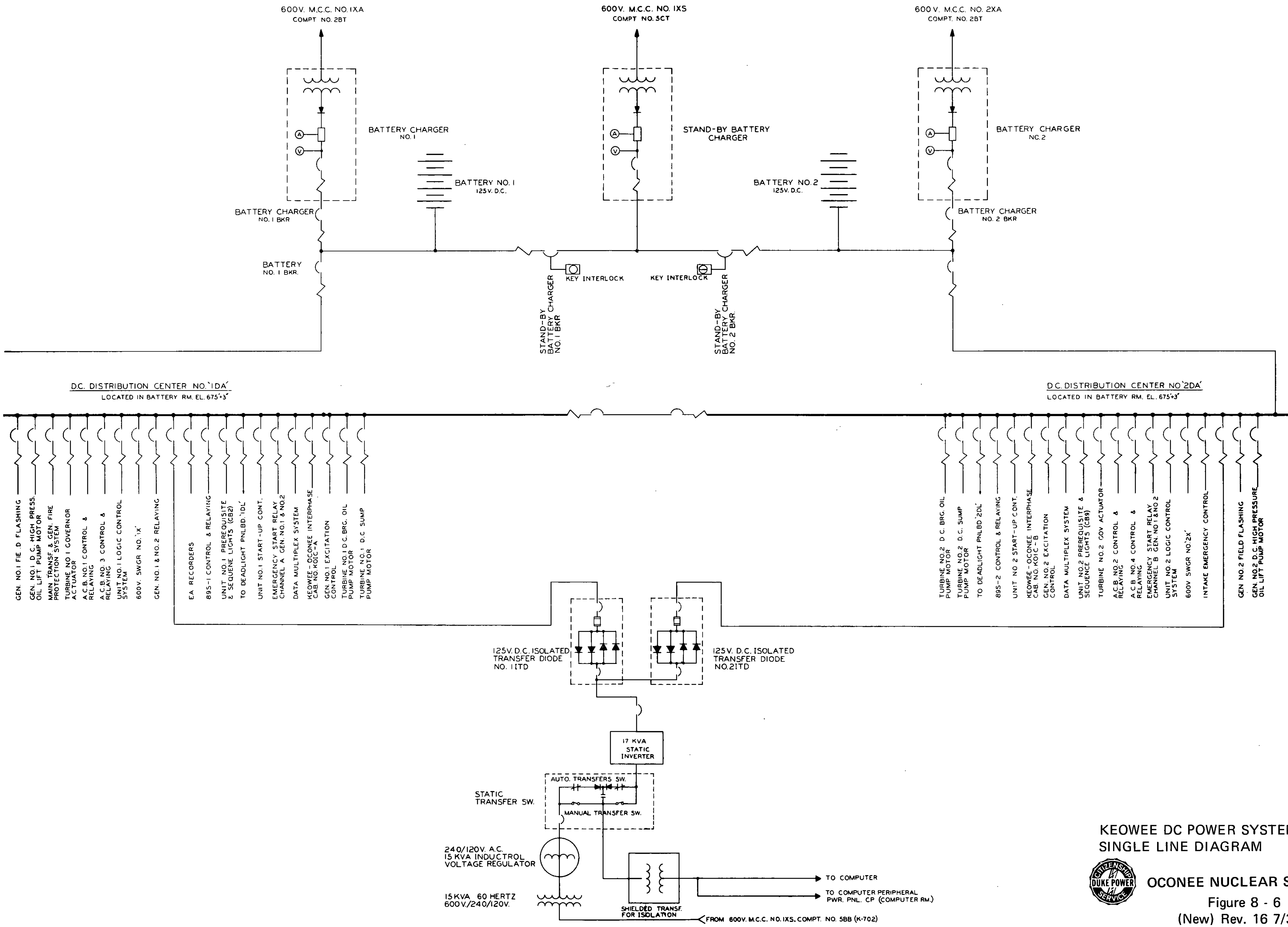
NOTE:

1. THE 125VOLT DC INSTRUMENTATION & CONTROL POWER SYSTEM SHOWN IS FOR UNIT 1 ONLY. DUPLICATE SYSTEMS WILL BE PROVIDED FOR BOTH UNIT 2 AND UNIT 3.
2. ALL EQUIPMENT SHOWN TO BE INSTALLED WITH UNIT Nº1 EXCEPT UNIT 2 & 3. 125/250VDC. STATION POWER SYSTEM ALL OF WHICH WILL BE INSTALLED WITH THEIR RESPECTIVE UNIT.
   THE UNIT 2 INSTRUMENTATION AND CONTROL BATTERIES, BATTERY CHARGERS AND AND DISTRIBUTION CENTER BUSES WILL BE INSTALLED WITH UNIT 1 TO PROVIDE AN ALTERNATE SUPPLY TO UNIT 1 D.C. PANELBOARDS.
3. INSTRUMENTATION OUTPUTS AVAILABLE FOR CONTROL ONLY FROM REACTOR PROTECTIVE CHANNEL AS SHOWN WITH ASTERISK ON VITAL INSTRUMENT BUS

DC AND AC VITAL POWER SYSTEM
SINGLE LINE DIAGRAM

OCONEE NUCLEAR STATION

Figure 8 - 5
Rev. 2 2/9/70
Rev. 6 6/22/70



KEOWEE DC POWER SYSTEM –
SINGLE LINE DIAGRAM

OCONEE NUCLEAR STATION

Figure 8 - 6

(New) Rev. 16 7/30/71

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9-43	Original
9-44	Original
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9-48	Original
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Fig. 9-6	Rev. 18
Fig. 9-7	Original
Fig. 9-8	Rev. 16
Fig. 9-9	Rev. 16
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9. AUXILIARY AND EMERGENCY SYSTEMS

The auxiliary and emergency systems required to support the reactor during normal operations and servicing of the Oconee Nuclear Station are described in this section. Some of these systems have also been described and discussed in Section 6, since they serve as engineered safeguards. The information in this section deals primarily with the functions served by these systems during normal operation.

The design of the auxiliary systems has included consideration of system sharing, where feasible, between the three Oconee Nuclear Station units. This section describes the equipment for each unit and states where equipment is shared. Equipment shared between Units 1 and 2 will be installed with Unit 1.

The majority of the components in these systems are located within the auxiliary building. Those systems connected by piping between the reactor building and the auxiliary building are equipped with reactor building isolation valves as described in Section 5.2.

The systems considered in this section are:

a. High Pressure Injection System

b. Chemical Addition and Sampling System.

c. Component Cooling System.

d. Spent Fuel Cooling System.

e. Low Pressure Injection System.

f. Cooling Water Systems.

g. Fuel Handling System.

h. Station Ventilation Systems.

i. Coolant Storage System.

j. Coolant Treatment System.

The following codes and standards are used as applicable in the design, fabrication, and testing of components, and structures associated with the auxiliary systems:

a. ASME Boiler and Pressure Vessel Code, Section II, Material Specifications.

b. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

c. ASME Boiler and Pressure Vessel Code, Section VIII, Unfired Pressure Vessels and ASME Nuclear Case Interpretations.

d. ASME Boiler and Pressure Vessel Code, Section IX, Welding Qualifications.

e. Standards of the American Society for Testing Materials.

f. USA Standard Code for Pressure Piping, USAS B31.1.

g. USA Standard Code for Nuclear Power Piping, USAS B31.7.

h. Standards of the Institute of Electrical and Electronics Engineers.

i. Standards of the National Electrical Manufacturers Association.

j. Hydraulic Institute Standards.

k. Southern Standard Building Code.

l. Standards of Tubular Exchanger Manufacturers Association.

m. Air Moving and Conditioning Association Standards.

The specific codes for components in safety related systems are listed in the various tables of this section. To assist in review of the system drawings, a standard set of symbols and abbreviations has been used and is summarized in Figure 9-1.

9.1 HIGH PRESSURE INJECTION SYSTEM

9.1.1 DESIGN BASES

The High Pressure Injection System is designed to accommodate the following functions during normal reactor operation:

a. Supply the Reactor Coolant System with fill and operational makeup water.

b. Provide seal injection water for the reactor coolant pumps.

c. Provide for purification of the reactor coolant to remove corrosion and fission products.

d. Control the boric acid concentration in the reactor coolant.

e. In conjunction with the pressurizer, the system will accommodate temporary changes in reactor coolant volume due to small temperature changes.

f. Maintain the proper concentration of hydrogen and corrosion inhibiting chemicals in the Reactor Coolant System.

The specific design bases for various parts of the system are as follows:

Letdown Capability

The system will accommodate letdown required as a result of coolant volume expansion when heating the reactor coolant to operating temperature at a rate of 100 F/h while maintaining constant pressurizer level. The letdown is cooled before leaving the Reactor Building.

Purification

Filters and demineralizers are provided to remove reactor coolant impurities. The letdown filters and purification demineralizers are sized for full flow through the letdown orifice.

Makeup

The system will accommodate makeup requirements during design reactor coolant system transients and for reactor coolant system cooldown at the design rate.

9.1.2 SYSTEM DESCRIPTION AND EVALUATION

The High Pressure Injection System is shown schematically on Figure 9-2. Tables 9-1 and 9-2 list the system performance requirements and data for individual components. The following is a brief functional description of system components.

Letdown Cooler

The letdown cooler reduces the temperature of the letdown flow from the Reactor Coolant System to a temperature suitable for demineralization and injection to the reactor coolant pump seals. Heat in the letdown coolers is rejected to the Component Cooling System.

Letdown Flow Control

The letdown flow rate at reactor operating pressures is limited by a fixed block orifice. At reduced pressures a parallel, normally closed, remotely operated valve can be opened to maintain the desired flow rate. In addition there is a second parallel, normally closed valve which may be manually positioned for flow control.

Letdown Flow Radiation Monitor (RIA-36)

Upon leaving the letdown coolers, a one- to two-gallon per minute sample flow is continuously bypassed around the block orifice through a sampler. There it is monitored for gamma activity before being returned to the letdown stream upstream of the purification demineralizer (11.1.2.4).

Purification Demineralizer

The letdown flow is passed through the purification demineralizer to remove reactor coolant impurities other than boron. The design purification letdown flow is equal to one Reactor Coolant System volume in 24 hours. One demineralizer is provided for each unit. In addition, a spare demineralizer is shared between Units 1 and 2, and another spare is installed for Unit 3. One use of the spare demineralizers will be the removal of cesium from the Reactor Coolant System in the event of fuel defects. Section 11 describes coolant activities, coolant handling and storage, and expected limits on activity discharge.

Letdown Filters

Two letdown filters in parallel are provided to prevent particulates from entering the Reactor Coolant System and subsequently the pump seal filters. One filter is normally in use.

High Pressure Injection Pumps

The high pressure injection pumps are designed to return coolant which is letdown for purification to the Reactor Coolant System, and to supply the seal water to the reactor coolant pumps. The pumps are sized to permit one pump to provide normal operating makeup and seal water flow.

16. Reactor Coolant Pump Seal Injection Filters

Two reactor coolant pump seal filters are provided to prevent particulates from entering the pump seals. One is normally in use.

Seal Return Filter

16. A single filter is installed in the seal return line upstream of the seal return coolers to remove particulate matter. A bypass is installed to permit servicing during operation.

Reactor Coolant Pump Seal Return Coolers

The seal return coolers are sized to remove the heat added by the high pressure injection pumps and the heat picked up in passage through the reactor coolant pump seals. Heat from these coolers is rejected to the Recirculated Cooling Water System. Two coolers are provided and one is normally in operation.

Letdown Storage Tank

The letdown storage tank serves as a receiver for letdown, seal return, chemical addition, and system makeup. The tank also accommodates temporary changes in system coolant volume.

9.1.2.1 Mode of Operation (Reference Supplement 6 Revisions for Oconee 1.)

During normal operation of the Reactor Coolant System, one high pressure injection pump continuously supplies high pressure water from the letdown storage tank to the seals of each of the reactor coolant pumps and to a makeup line connection to one of the reactor inlet lines. Makeup flow to the Reactor Coolant System is regulated by a flow control valve, which operates on signals from the pressurizer level controller.

A control valve in the common injection line to the pump seals automatically maintains the desired total injection flow to the seals. Throttle valves in each
16. pump seal injection line provide a capability to balance the seal injection flow rates. A portion of the water supplied to the seals enters the Reactor Coolant

16. System. The remainder returns to the letdown storage tank after passing through one of the two reactor coolant pump seal return coolers.

Seal water inleakage to the Reactor Coolant system requires a continuous letdown of reactor coolant to maintain the desired pressurizer level. Letdown is also required for removal of impurities and boric acid from the reactor coolant. The letdown is cooled by one of the letdown coolers, reduced in pressure by the letdown orifice, and then passed through the purification demineralizer to a three-way valve which directs the coolant to the letdown storage tank or to the coolant storage system.

Normally, the three-way valve is positioned to direct the letdown flow to the letdown storage tank. If the boric acid concentration in the reactor coolant is to be reduced, the three-way valve is positioned to divert the letdown flow to the coolant storage system. Boric acid is removed by directing the letdown flow through a deborating demineralizer with the effluent returned directly to the letdown storage tank, or by the feed and bleed method. Feed and bleed is the process of directing the letdown flow to a coolant bleed holdup tank and maintaining the level in the letdown storage tank with demineralized water pumped from a supply of unborated water. The flow of demineralized water is measured and totaled by inline flow instrumentation. The flow of demineralized or borated water returning to the letdown storage tank is controlled remotely by the makeup control valve. During normal operation the inline instrumentation or the control rod drive interlock will terminate makeup flow.

The letdown storage tank also receives chemicals for addition to the reactor coolant. A hydrogen overpressure is maintained in the tank to assure a slight amount of excess hydrogen in the circulating reactor coolant. Other chemicals are injected in solution into the tank.

System control is accomplished remotely from the control room with the exception of the reactor coolant pump seal return cooling. The letdown flow rate is set by remotely positioning the letdown flow control valve to pass the desired flow rate. The spare purification demineralizer can be placed in service by remote positioning of the demineralizer isolation valves. The letdown flow to the Coolant Storage System is diverted by remote positioning of the three-way valve
16. and the valves in the Coolant Storage System. The reactor coolant volume control valve is automatically controlled by the pressurizer level controller.

For emergency operation as a high pressure injection system, the normal letdown coolant flow line and the normal pump seal return line are closed, and makeup flow is diverted to the emergency high pressure injection lines. The pumps and pump motors are designed to operate at the higher flow rates and lower discharge pressures associated with the high pressure injection requirements. Emergency operation of this system is described in Section 6.

9.1.2.2 Reliability Considerations

This system provides essential functions for the normal operation of the unit. Redundant components and alternate flow paths have been provided to improve system reliability.

In addition to the letdown orifice the system has two full-capacity control valves in parellel with the orifice. One of these control valves is manually operated and one is remotely operated.

Each unit has three high-pressure injection pumps, each capable of supplying the required reactor coolant pump seal and makeup flow. One is normally in operation while another is in standby status to be used as needed. The third pump is used only for emergency injection. There are two letdown coolers and two seal return coolers. One cooler in each group will perform the required duty while the other may be used as a spare.

One of the two letdown filters or reactor coolant pump seal filters is normally in use while the other is a spare.

9.1.2.3 Codes and Standards

Each component of this system will be designed to the code or standard, as applicable, noted in Tables 9-2 and 6-5.

9.1.2.4 System Isolation

The letdown line and the reactor coolant pump seal return line are outflow lines which penetrate the Reactor Building. Both lines contain electric motor-operated isolation valves inside the Reactor Building and pneumatic valves outside which are automatically closed by an engineered safeguards signal. The injection line to the reactor coolant pump seals is an inflow line penetrating the Reactor Building. This line contains a stop-check valve inside the Reactor Building and a remotely operated valve on the outside of the Reactor Building. Check valves in the discharge of each high pressure injection pump provide further backup for Reactor Building isolation. The two emergency coolant injection lines are used for injecting coolant to the reactor vessel after a loss-of-coolant accident. After use of the lines for emergency injection is discontinued, the electric motor-operated isolation valves in each line outside the Reactor Building may be closed remotely by the control room operator for isolation.

9.1.2.5 Leakage Considerations

Design and installation of the components and piping in the High Pressure Injection System considers the radioactive service of this system. Except where flanged connections have been installed for ease of maintenance, the system is an all-welded system. Valves have double packing with provisions for leak-off connections.

9.1.2.6 Failure Considerations

The effects of failures and malfunctions in the High Pressure Injection System concurrent with a loss-of-coolant accident are presented in Section 6. Section 9.1.2.4 describes system isolation. These analyses show that redundant safety features are provided where required.

For pipe failures in the High Pressure Injection System, the consequences depend upon the location of the rupture. If the rupture were to occur between the reactor coolant loop and the first isolation valve or check valve, it would lead to an uncontrolled loss of coolant from the Reactor Coolant System. The analysis of this loss-of-coolant accident is included in Section 14. If the rupture were to occur beyond the first isolation valve or outside the Reactor Building, the release of radioactivity would be limited by the small line sizes and by closing of the isolation or check valve.

A single failure will not prevent boration when desired for reactivity control, since several alternate paths are available for adding boron to the Reactor Coolant System. These are: (a) through the normal makeup lines, (b) through the reactor coolant pump seals, and (c) through the high pressure injection connections. If pump suction is unavailable from the letdown storage tank, a source of borated water is available from the borated water storage tank during reactor power operation.

9.1.2.7 Operational Limits

Alarms or interlocks are provided to limit variables or conditions of operation that could cause system upsets. The variables or conditions of operation that are limited are as follows:

a. Letdown Storage Tank Level

Low water level in the letdown storage tank is alarmed and interlocked to the three-way bleed valve. Low water level will switch the three-way valve from the bleed position to its normal position.

b. Letdown Line Temperature

A high letdown temperature in the letdown line downstream of the letdown coolers is alarmed and interlocked to close the pneumatic letdown isolation valve, thus protecting the purification demineralizer resins.

c. Dilution Control

The dilution cycle is initiated by the operator. Several safeguards are incorporated into the design to prevent inadvertent excessive dilution of the reactor coolant.

1. The dilution valves are interlocked so that the operator must preset the desired quantity of dilution batch size before initiating the dilution cycle. The dilution cycle will automatically terminate when the dilution flow has integrated to the preset batch size.

6.

2. Interlocks on the regulating control rod bank automatically terminate the dilution cycle if the regulating rod group (Group 6) is inserted into the core beyond 25 percent.

6. 3. The operator may manually terminate the dilution cycle at any time.

(Reference Supplement 6 revisions for Oconee 1.)

Table 9-1
High Pressure Injection System Performance Data

Nominal Letdown Flow, Cold, gpm	45
Letdown Flow Maximum, Cold, gpm	140
16. Total Seal Flow to Each Reactor Coolant Pump, gpm	8
Seal Inleakage to Reactor Coolant System per Reactor Coolant Pump, gpm	7
Injection Pressure to Reactor Coolant Pump Seals, psig	2,190
Temperature to Seals, normal/maximum, F	120/150
Purification Letdown Fluid Temperature, normal/maximum, F	120/135
Letdown storage Tank Normal Operating Pressure, psig	15 - 35
Letdown Storage Tank Volume Between Minimum and Maximum Operating Levels, ft^3	250

Table 9-2
High Pressure Injection System Component Data

16. High Pressure Injection Pump	
Type	Vertical, multistage, centrifugal, mechanical seal
Capacity, gpm	(See Figure 6-6)
Head, ft H_2O (at sp. gr. = 1)	(See Figure 6-6)
Motor Horsepower, nameplate hp	600
Pump Material	SS wetted parts
3. Design Pressure, psig	2,800/3,050
Design Temperature, F	200/150
Letdown Cooler	
Type	Shell and spiral tube
Heat Transferred, Btu/h	16.0×10^6
Letdown Flow, lb/h	3.5×10^4
Letdown Cooler Inlet/Outlet Temperature, F	555/120
Material, shell/tube	CS/SS

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Table 9-2 (Cont'd)

High Pressure Injection System Component Data

Design Pressure, psig	2,500
Design Temperature, F	600
Component Cooling Water Flow (ea.), lb/h	2×10^5
Code	ASME Sec. III-C & VIII
Reactor Coolant Pump Seal Return Cooler	
Type	Shell and tube
Heat Transferred, Btu/h	2.2×10^6
Seal Return Flow, lb/h	1.25×10^5
Seal Return Temperature Change, F	145 x 127
Material, shell/tube	CS/SS
Design Pressure, psig	150
Design Temperature, F	200
Recirculated Cooling Water Flow (ea.), lb/h	1.25×10^5
Code	ASME Sec. III-C & VIII
Letdown Storage Tank	
Volume, ft^3	600
Design Pressure, psig	100
Design Temperature, F	200
Material	SS
Code	ASME Sec. III-C
Purification Demineralizer	
Type	Mixed bed, boric acid saturated
Material	SS
Volume, ft^3	85
Flow, gpm	70
Vessel Design Pressure, psig	150
Vessel Design Temperature, F	200
Code	ASME Sec. III-C
Letdown Filter	
Design Flow Rate, gpm	80
Material	SS
Design Temperature	200

Table 9-2 (Cont'd)

High Pressure Injection System Component Data

Design Pressure	150
Code	ASME Sec. III-C
16. Reactor Coolant Pump Seal Injection Filter	
Design Flow Rate, gpm	300
Material	SS
Design Temperature, F	200
Design Pressure, psig	3,050
Code	ASME Sec. III-C
Reactor Coolant Pump Seal Return Filter	
Design Flow Rate, gpm	50
Material	SS
16. Design Temperature, F.	200
Design Pressure, psig	150
Code	ASME III-C

9.2 CHEMICAL ADDITION AND SAMPLING SYSTEM

9.2.1 DESIGN BASES

Chemical addition and sampling operations are required to change and monitor the concentration of various chemicals in the Reactor Coolant System and auxiliary systems. The Chemical Addition and Sampling System is designed to add boric acid to the Reactor Coolant System for reactivity control, lithium hydroxide for pH control, and hydrazine for oxygen control.

9.2.2 SYSTEM DESCRIPTION AND EVALUATION

The Chemical Addition and Sampling System is shown schematically on Figure 9-3. The Sampling System has separate sampling stations for reactor coolant and steam generator sampling for each of the three units. Two auxiliary systems sampling stations are provided, one for Units 1 and 2 and one for Unit 3.

Two chemical addition systems are also provided, one for Units 1 and 2 and one for Unit 3. These systems permit chemical addition to and sampling of the Reactor Coolant System and other reactor auxiliary systems during normal reactor operation.

The Chemical Addition and Sampling System performs no emergency functions. The steam generator feedwater quality is maintained within the limits noted in Table 9-3, and the reactor coolant quality is maintained by this system within the limits given in Table 9-4. A brief functional description of the major system components follows.

Boric Acid Mix Tank

A single boric acid mix tank is provided as a source of concentrated boric acid solution. The volume of the tank provides sufficient boric acid solution to increase the reactor coolant system boron concentration to that required for cold shutdown. Tank heaters and electrically heat traced transfer lines maintain the fluid temperature above that required to assure solubility of the boric acid.

Boric Acid Pumps

Three boric acid pumps are provided to transfer the concentrated boric acid solution from the boric acid tank to the borated water storage tank, letdown storage tanks, spent fuel storage pool, or the core flooding tanks. One pump, with a 1-gpm capacity, supplies boric acid to the core flooding tanks. The other two pumps, which have 10-gpm capacities, supply boric acid to other tanks, systems, and locations (Figure 9-3).

Caustic Mix Tank

The caustic mix tank is used to prepare solution which neutralizes the feed to the waste evaporator. It also supplies sodium hydroxide to the deborating demineralizer for regeneration.

Caustic Pump

The caustic pump transfers sodium hydroxide from the caustic mix tank to the intended destination.

Lithium Hydroxide Tank

Lithium hydroxide is mixed and added to the Reactor Coolant System from the lithium hydroxide tank.

Lithium Hydroxide Pump

The lithium hydroxide pump transfers lithium hydroxide from the $LiOH$ tank to the letdown line upstream of the letdown filters.

Hydrazine Drum

A 55-gallon drum supplies hydrazine to the Reactor Coolant System; the hydrazine is used to scavenge dissolved oxygen, primarily following a reactor shutdown.

Hydrazine Pump

The hydrazine pump transfers hydrazine to the letdown line upstream of the letdown filters.

Pressurizer Sample Cooler

This cooler cools the effluent sample taken from the pressurizer steam or water space.

Steam Generator Sample Cooler

This cooler cools the effluent sample taken from the secondary side of the steam generator.

9.2.2.1 Mode of Operation

The chemical addition portion of this system delivers the necessary chemicals to other systems as required. Boric acid is provided to the spent fuel pool, borated water storage tank, letdown storage tank, and core flooding tanks as makeup for leakage or to change the concentration of boric acid in the associated systems. Sodium hydroxide is added to the waste evaporator feed tank during evaporator operation and to the deborating demineralizer during demineralizer resin regeneration. The sampling portion of this system is used to take samples to assure that water qualities and boric acid concentrations are maintained. Sampling locations and the samples taken at each location are as follows:

Liquids

a. Steam Generator Sample Sink.

   1. Secondary Side of Steam Generator.

b. Reactor Coolant Sample Sink.

   1. Pressurizer Water Space.
   2. Pressurizer Steam Space.
   3. Low Pressure Injection Cooler Outlet.
   4. Core Flooding Tanks.
   5. Total Gas Sample 18.

c. Auxiliary Systems Sample Sink.

   1. Purification Demineralizer Inlet and Outlet.
   2. Deborating Demineralizer Outlet.
   3. Letdown Storage Tank Water Space.
   4. RC Bleed Evaporator Feed Pump Discharge.
   5. Deborating Demineralizer Outlet (Regeneration).
   6. Waste Evaporator Feed Pump Discharge.
   7. RC Bleed Evaporator (Concentrate).
   8. Concentrated Boric Acid Transfer Pump Discharge.
   9. RC Bleed Evaporator (Distillate).
   10. Waste Evaporator (Concentrate).

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11. RC Bleed Transfer Pump Discharge.

12. Waste Transfer Pump Discharge.

13. High Acitivity Waste Transfer Pump Discharge.

14. Low Activity Waste Transfer Pump Discharge.

15. Condensate Test Tank Pump Discharge.

16. RC Bleed Evaporator Demineralizer Outlet.

Gaseous

a. Hydrogen and Oxygen Gas Analyzer.

1. Waste Holdup Tank.
2. High Activity Waste Tank.
3. RC Bleed Holdup Tank.
4. RC Bleed Holdup Tank (Demineralized Water).
5. Waste Gas Vent Header.
6. RC Bleed Evaporator.
7. Waste Evaporator.
8. Waste Gas Tanks.

b. Sample Containers (to be analyzed for a variety of substances).

1. Letdown Storage Tank Gas Space.
2. Letdown Line Upstream of Purification Demineralizer.
3. In addition, all gas analyzer samples can be containerized for further analysis.

9.2.2.2 Reliability Considerations

The Chemical Addition and Sampling System is not required to function during an emergency condition. Redundant boric acid pumps and flow paths are provided to guard against a single component failure rendering the system inadequate for boron addition. In addition to the boric acid mix tank, boric acid is also available for boration in 5-per-cent-by-weight solution from the concentrated boric acid storage tank. To prevent precipitation, heat tracing is installed on components and lines used to transfer concentrated boric acid. The pumps, tanks, coolers, and instrumentation are located in the Auxiliary Building and are accessible for inspection and maintenance.

9.2.2.3 Codes and Standards

The components of the Chemical Addition and Sampling System are designed to the codes and standards noted in Table 9-5.

9.2.2.4 System Isolation

The pressurizer sample line and both steam generator sample lines are the only system lines that penetrate the Reactor Building. All three lines contain electric motor-operated isolation valves inside the Reactor Building and pneumatic valves outside, which are automatically closed by an engineered safeguards signal.

9.2.2.5 Leakage Considerations

Leakage of radioactive reactor coolant from this system within the Reactor Building will be collected in the Reactor Building normal sump. Leakage of radioactive gases from this system outside the Reactor Building is collected by placing the sampling stations under hoods exhausting to the unit vent.

9.2.2.6 Failure Considerations

Since the system serves no engineered safeguards function, the only consideration following a loss-of-coolant accident is the operation of the isolation valves. Redundant isolation valves are provided to assure isolation of the Reactor Building as described in 9.2.2.4.

9.2.2.7 Operational Limits

The Chemical Addition and Sampling System provides certain chemicals to several systems in proper quantities and concentrations and provides a capability to sample fluids in various systems. The limits that must be maintained on these operations are described below.

The boric acid mix tank solution is to be maintained above an average temperature of 105 F in order to maintain boric acid in solution at a concentration of 7 per cent by weight. The capacity of the boric acid mix tank is 500 cubic feet. Approximately 216 cubic feet is required to borate the Reactor Coolant System for cold shutdown near the end of core life.

Table 9-3

Normal Steam Generator Feedwater Quality

Total Dissolved and Suspended Solids (max), ppb	50
Dissolved Oxygen (max), ppb	7
Total Silica as SiO_2 (max), ppb	20
Total Iron as Fe (max), ppb	10
Total Copper as Cu (max), ppb	2
pH at 77 F (adjusted with ammonia)	9.3 - 9.5

Table 9-4

Reactor Coolant Quality

Total Solids, including dissolved and undissolved material, but excluding ^7LiOH and H_3BO_3 (max), ppm	1.0
Boron (max), ppm	2,270
Lithium as ^7Li, ppm (when required for pH adjustment)	0.5 - 2.0[a]
pH at 77 F	4.8 - 8.5[b]
Dissolved Oxygen as O_2	See note[c]
Chlorides as Cl^- (max), ppm	0.1
Hydrogen as H_2, std cc/liter H_2O	15 - 40
Fluorides as Fl^- (max), ppm	0.1

[a] Equivalent range as ^7LiOH is 1.455 to 5.82 ppm.

[b] Equivalent pH at 600 F is 6.8 to 7.8.

[c] With proper H_2 specification at critical condition, dissolved O_2 is assumed not to be present.

Table 9-5

Chemical Addition and Sampling System Component Data

Tanks	
Boric Acid Mix Tank	
Type	Vertical cylindrical
Volume, ft^3	500
Design Pressure, psig	Atmospheric
Design Temperature, F	250
Material	Al
Code	USAS B96.1
Lithium Hydroxide Mix Tank	
Type	Vertical cylindrical
Volume, gal.	50
Design Pressure, psig	Atmospheric
Design Temperature, F	140
Material	SS

Table 9-5 (Cont'd)

Chemical Addition and Sampling System Component Data

Caustic Mix Tank	
Type	Vertical cylindrical
Volume, gal.	150
Design Pressure, psig	Atmospheric
Design Temperature, F	200
Material	SS
16. **Pumps**	
Boric Acid Pump	
Type	Reciprocating, variable stroke
Capacity, gpm	10
16. Maximum Discharge Pressure, Psig	75
Design Pressure, psig	100
Design Temperature, F	200
Pump Material	SS
Boric Acid Pump (Core Flood Tanks)	
Type	Reciprocating, variable stroke
Capacity, gpm	1
16. Maximum Discharge Pressure, Psig	630
Design Pressure, psig	700
Design Temperature, F	300
Pump Material	SS
Lithium Hydroxide Pump	
Type	Reciprocating, variable stroke
Capacity, gph	10
Maximum Discharge Pressure, Psig	75
16. Design Pressure, psig	100
Pump Material	SS

Table 9-5 (Cont'd)
Chemical Addition and Sampling System Component Data

Hydrazine Pump	
Type	Reciprocating, variable stroke
Capacity, gph	10
16. Maximum Discharge Pressure, Psig	75
Design Pressure, psig	100
Design Temperature, F	200
Pump Material	SS
Caustic Pump	
Type	Reciprocating, variable stroke
Capacity, gpm	2
16. Maximum Discharge Pressure, Psig	50
Design Pressure, psig	100
Design Temperature, F	200
Pump Material	SS
Pressurizer Sample Cooler	
Type	Shell and spiral tube
Rated Capacity, Btu/h	2.1×10^5
Sample Flow Rate, lb/h	200
Maximum Sample Inlet Temperature, F	650
Sample Outlet Temperature, F	120
Cooling Water Flow, lb/h	5,000
Design Temperature Shell/Tube, F	250/670
Design Pressure Shell/Tube, psig	150/2,500
Code	ASME Sec. III-C & VIII
Steam Generator Sample Cooler	
Type	Shell and spiral tube
Rated Capacity, Btu/h	2.3×10^5
Sample Flow Rate, lb/h	500
Sample Inlet/Outlet Temperature, F	535/100
Cooling Water Flow, lb/h	5,000
Design Temperature Shell/Tube, F	300/600
Design Pressure Shell/Tube, psig	150/1,050
Code	ASME Sec. VIII

9.3 COMPONENT COOLING SYSTEM

9.3.1 DESIGN BASES

The Component Cooling System is designed to provide cooling water for various components in the Reactor Building as follows: letdown coolers, reactor coolant pump cooling jacket and seal coolers, quench tank cooler, and control rod drive cooling coils. The design cooling requirement for the system is based on the maximum heat loads from these sources. The system also provides an additional barrier between high pressure reactor coolant and service water to prevent an inadvertent release of activity.

9.3.2 SYSTEM DESCRIPTION AND EVALUATION

The Component Cooling System is shown schematically on Figure 9-4, and the performance requirements of the system are tabulated in Table 9-6. The following is a brief functional description of the major components of the system and their sharing between nuclear units of the station:

Component Cooler

Each component cooler is designed for the total Component Cooling System heat load for a reactor unit. Units 1 and 2 each have a single component cooler with a shared common spare. Unit 3 has two coolers. The coolers reject the heat load to the low pressure service water system.

Component Cooling Pumps

Each component cooling pump is designed to deliver the necessary flows to the letdown coolers, reactor coolant pump cooling jackets and seal coolers, quench tank cooler, and control rod drive cooling coils. Each unit has one operating pump and one spare.

Component Cooling Surge Tank

This tank allows for thermal expansion and contraction of the water in this closed-loop system. It also provides the required NPSH for the component cooling pumps.

Control Rod Drive Filter

A filter is provided in the cooling water circuit to the control rod drives to prevent particulates from entering the drive cooling coils. A single filter with a bypass is provided.

9.3.2.1 Mode of Operation

During operation, one component cooling pump and one component cooler recirculate and cool water to accommodate the system heat loads for each reactor unit. The component cooling surge tank accommodates expansion, contraction, and leakage of coolant into or out of the system. The surge tank provides a reservoir of component cooling water until a leaking cooling line can be isolated (9.3.2.5). Makeup water and corrosion inhibiting chemicals are added to the system in the surge tank.

9.3.2.2 Reliability Considerations

The Component Cooling System performs no emergency functions. Redundancy in active components is provided to improve system reliability. The pumps, coolers, surge tank, and most of the instrumentation are located in the Auxiliary Building and are accessible for inspection and maintenance.

9.3.2.3 Codes and Standards

The components of the system are designed to the codes and standards given in Table 9-7.

9.3.2.4 System Isolation

Since the Component Cooling System is not an engineered safeguards system, Reactor Building isolation valves are automatically closed on a high Reactor Building pressure signal to provide building isolation. The Reactor Building inlet lines are isolated by two check valves, one on the outside and one on the inside of the Reactor Building. The Reactor Building outlet line is isolated by an electric motor-operated valve on the inside and by a pneumatic valve on the outside of the Reactor Building.

9.3.2.5 Leakage Considerations

Water leakage from piping, valves, and other equipment in the system is not considered to be detrimental since the cooling water is normally nonradioactive. Welded construction is used throughout the system to minimize the possibility of leakage except where flanged connections are required for servicing.

Inleakage of reactor coolant to the system is detected by a radiation monitor (RIA-50) located in the recirculation line from the pumps to the surge tank (11.1.2.4), and is also indicated by an increase in surge tank level. A defective coil of a coolant pump can be remotely isolated by an electric motor-operated valve on the outlet cooling line and a stop-check valve on the inlet line. A letdown cooler leak can be remotely isolated with motor-operated valves on the reactor coolant side of the cooler. The cooling water side can be completely isolated by closing a remotely operated, motor-actuated valve on the inlet of the cooler and the manual valves on the outlet cooling lines. Leakage from the quench tank cooler can be isolated by manual valves on the reactor coolant side. The cooling water side can be completely isolated by two manual valves.

9.3.2.6 Failure Considerations

Since the system serves no engineered safeguards function, the only consideration following a loss-of-coolant accident is the operation of the isolation valves. Redundant isolation valves are provided as described in 9.3.2.4.

Failures and malfunction of components during normal operation were evaluated. Operation of the Component Cooling System is essential to normal reactor operation. In the event of loss of a component cooling pump, the standby pump will automatically start and maintain cooling water flow. The complete loss of cooling water flow does not require immediate reactor shutdown. However, procedures will require the operator to shut down the reactor to protect the control rod drive coils. The reactor coolant pumps can be operated indefinitely without component cooling water if seal injection flow is available.

Table 9-6
Component Cooling System Performance Data
(For Normal Operation on a Per Unit Basis)

Number of Component Cooling Pumps	2
Number of Pumps Normally Operating	1
Design Flow, gpm	766
Number of Component Coolers	
Units 1 and 2	1 + 1 Shared Spare
Unit 3	2
Number of Coolers Normally Operating	1
Design Heat Removal Requirements, Btu/h per cooler	19×10^6

Table 9-7
Component Cooling System Component Data
(Component Data on a Per Unit Basis)

Component Cooling Pumps	
Type	Centrifugal
Rated Capacity, gpm	766
Rated Head, ft, H_2O	220
Motor Nameplate Horsepower, hp	60
Casing Material	CS
Design Pressure, psig	100
Design Temperature, F	225
Component Coolers	
Type	Shell and Tube
Capacity, Btu/h	19×10^6
Component Cooling Water Inlet Temp, F	150
Component Cooling Water Outlet Temp, F	100
Code	ASME Section VIII
Surge Tank	
Volume, ft^3	50
Material	CS
Design Pressure, psig	Atmospheric
Design Temperature, F	200
Code	AWWA D-100
Control Rod Drive Filter	
Design Flow Rate, gpm	140
Code	ASME Section VIII

9.

(Reference Supplement 6 revisions for Oconee 1.)

9.4 SPENT FUEL COOLING SYSTEM

9.4.1 DESIGN BASES

The Spent Fuel Cooling System for Units 1 and 2 is designed to remove the decay heat from the stored fuel in the spent fuel pool. It is designed to maintain the spent fuel pool at approximately 120°F, with a heat load based on removing the decay heat generated from two 1/3-core batches of fuel assemblies, one of which has been irradiated for 930 days and cooled for 150 hours, and one of which has been irradiation for 930 days and cooled for 420 hours.

In meeting the foregoing design bases, the system has the capability to maintain the spent fuel pool at 150°F while removing decay heat from the following combination of stored fuel assemblies:

a. 2/3 core irradiated for 930 days and cooled for 100 days.

b. 1/3 core irradiated for 720 days and cooled for 150 hours.

c. 1/3 core irradiated for 410 days and cooled for 150 hours.

d. 1/3 core irradiated for 100 days and cooled for 150 hours.

In addition to its primary function, the system provides for purification of the spent fuel pool water, the fuel transfer canal water, and the contents of the borated water storage tank in order to remove fission and corrosion products and to maintain water clarity for fuel handling operations. The system also provides for filling the fuel transfer canal and the incore instrument handling tank.

The Spent Fuel Cooling System for the Unit 3 spent fuel pool duplicates the equipment used for the Units 1 and 2 system. It is designed to maintain the spent fuel at 150°F or less, with a heat load based on the decay heat generated by as many as 474 fuel assemblies with 1/3 of a core being discharged into the pool at each refueling interval.

In meeting the foregoing design bases, the system has the capability to maintain the spent fuel pool at 150°F while removing decay heat from the following combination of stored fuel assemblies:

a. 1/3 core irradiated 720 days and decayed 6 days

b. 1/3 core irradiated 410 days and decayed 6 days

c. 1/3 core irradiated 100 days and decayed 6 days

d. 1/3 core irradiated 930 days and decayed 118 days

e. Four 1/3 core batches irradiated 930 days and decayed one, two, three, and four years respectively, longer than batch (d) above.

For conservatism, the above evaluation is based on the assumption that the Unit 3 spent fuel pool is used only for Unit 3 fuel and without inputs of decayed batches from Units 1 and 2.

9.4.2 SYSTEM DESCRIPTION

The Spent Fuel Cooling System (Figure 9-5) provides cooling for the spent fuel pool to remove fission product decay heat energy. System performance data are shown in Table 9-8. Major components of the system are briefly described below.

Spent Fuel Coolers

The spent fuel coolers are designed to maintain the temperature of the spent fuel pool as noted in Section 9.4.1. There are two coolers for Units 1 and 2, and two for Unit 3.

Spent Fuel Coolant Pumps

The spent fuel coolant pumps take suction from the spent fuel pool and recirculate the fluid back to the pool after passing through the coolers, demineralizer and filters in various combinations depending on conditions. There are two pumps for Units 1 and 2, and two pumps for Unit 3. The spent fuel pumps are also used for filling the fuel transfer canal or incore instrumentation handling tank with borated water from the borated water storage tank.

Spent Fuel Coolant Demineralizer

The spent fuel coolant demineralizer will process approximately one-half of the spent fuel pool volume in 24 hours. There is one demineralizer for Units 1 and 2, and one for Unit 3.

Spent Fuel Coolant Filters

The spent fuel coolant filters are designed to remove particulate matter from the spent fuel pool water. They are sized for the same flow rate as the demineralizer (180 gpm). There are two filters for Units 1 and 2, and two for Unit 3.

Borated Water Recirculation Pump

This pump removes water from the borated water storage tank for demineralization and filtering. The pump may also be used while demineralizing and filtering the water in the fuel transfer canal during a transfer of fuel. It may also be used for emptying the fuel transfer canal if both spent fuel coolant pumps are unavailable for use. There is one pump for Units 1 and 2, and one for Unit 3.

9.4.2.1 Modes of Operation

9.4.2.1.1 Normal Operation

The normal operation of the Spent Fuel Cooling System serves two main functions. The first is to maintain the pool water at temperatures of approximately 125°F or less for Units 1 & 2 spent fuel pool and 150°F for Unit 3 spent fuel pool with stored fuel at expected operating histories and with both coolers operating. The second function is to provide purification of the spent fuel pool coolant for clarity during fuel handling operations.

The first function is accomplished by recirculating spent fuel coolant water from the spent fuel pool through the pumps and coolers and back to the pool. The spent fuel pumps take suction from the spent fuel pool and deliver the water through the shell side of two coolers arranged in parallel and back to the pool (Figure 9-5).

A bypass purification loop is provided to maintain the purity of the water in the spent fuel pool. This loop is also utilized to purify the water in the borated water storage tank following refueling, and to maintain clarity in the fuel transfer canal during refueling. Water from the borated water storage tank or fuel transfer canal can be purified by using the borated water recirculation pump.

9.4.2.1.2 Refueling Conditions

Normally, either 1/3 or 2/3 of the spent fuel assemblies from a core will be stored in the units 1 and 2 pool. When 1/3 of a core is stored, assuming the conditions in Section 9.4.1, one of the pumps and one of the coolers will maintain pool coolant at 125°F or less. When 2/3 of a core is stored, as assumed in Section 9.4.1, both pumps and both coolers will be operated to maintain the 120°F temperature.

When 1-2/3 cores are stored under the conditions of Section 9.4.1 due to the assumed unloading of a complete reactor core, two pumps and two spent fuel coolers will maintain the spent fuel storage pool temperature at 150°F. If both a pump and a cooler are assumed to be unavailable when this storage condition exists, the water temperature would eventually rise to 205°F, although considerable time would be required to heat the large spent fuel pool to this temperature. If all cooling is lost, the time required for the spent fuel pool to reach 205°F for the specified quantities and conditions of stored fuel is as follows:

One-third of a core	> 40 hours
Two-thirds of a core	> 15 hours
One and two-thirds cores	9 hours

These time periods are considered to be long enough to assure that maintenance repairs or temporary arrangements to provide adequate cooling can be carried out by operator action.

Normally, 1/3 of a core will be discharged into the Unit 3 spent fuel pool at each refueling of Unit 3. Transfers of spent fuel assemblies from the Unit 1-2 pool will also be made. When the first batch is introduced, one pump and one cooler will be capable of maintaining the pool at 125°F or less. As fuel accumulates, two pumps and two coolers will be utilized to maintain the pool at 150°F or less. The heat load will vary and be primarily dependent on the age of the most recently irradiated fuel in the pool. In the event one pump and one cooler were unavailable at the time of maximum thermal loading, the remaining cooling train would limit the bulk pool temperature to 205°F or less. In the unlikely event of total loss of all forced cooling, considerable time would be required to heat the large pool water mass to the boiling temperature. With the most severe expected heat load and loss of all pumps and coolers, the Unit 3 spent fuel pool (assumed to be adiabatic) requires 6.2 hours to reach 205°F. This is considered ample time to effect repairs or to provide alternate cooling.

9.4.2.2 Reliability Considerations

The Spent Fuel Cooling System provides adequate capacity and component redundancy to assure the cooling of stored spent fuel, even when unexpectedly large amounts of fuel are in storage. Multiple component failures or complete cooling failures permit ample time to assure that protective actions are taken. The system is arranged so that loss of water by piping or component failure is impossible. The system performs no emergency function and is not directly connected with the Reactor Coolant System.

9.4.2.3 Codes and Standards

Each component of this system is designed to the code or standard, as applicable, as noted in Table 9-9.

9.4.2.4 Leakage Considerations

If a leaking fuel assembly is transferred from the fuel transfer canal to the spent fuel pool, a small quantity of fission product activity may enter the spent fuel pool cooling water, even though the assembly's cladding temperature is lowered and leakage may reasonably be expected to be minimized. The purification loop removes these fission products and other contaminants from the water. Special provisions are made for the storage of defective fuel assemblies.

The fuel handling and storage area housing the spent fuel pool will be monitored (RIA-41) for gaseous activity (11.1.2.4) and gases will be exhausted to the environment through the unit vent. Provisions have been made in the design to air-test the flanged end of the fuel transfer tube for leak-tightness after use.

9.4.2.5 System Isolation

The Spent Fuel Cooling System has no process lines connecting to the Reactor Coolant System. Its major penetration to the Reactor Building is through the fuel transfer tube. The fuel transfer tube is isolated inside the Reactor Building by a blind flange connection in the fuel transfer canal. On the outside of the Reactor Building, the fuel transfer tube is isolated by a closed, manually-operated valve in the spent fuel pool.

9.4.2.6 Failure Considerations

Failure of a single component in this system will not permit the uncovering of the stored spent fuel under normal operating conditions since the system is designed with redundant components. With a complete loss of cooling, considerable time is required to raise the temperature of the pool water to boiling.

9.4.2.7 Operational Limits

There are alarms provided for the spent fuel system to indicate coolant level and temperature.

Table 9-8
Spent Fuel Cooling System Performance Data

System Cooling Capacity, Btu/h	
Normal	15.5×10^6
Maximum	28.6×10^6
System Design Pressure, psig	125
System Design Temperature, F	250

Table 9-9
Spent Fuel Cooling System Component Data

Spent Fuel Cooler	
Type	Tube and shell
Material Tube/Shell	CS/SS
Capacity, Btu/h/cooler	7.75×10^6
Cooling Water Flow, lb/h/cooler	5×10^5
Code	ASME VIII/III-C
Spent Fuel Pump	
Type	Horizontal, centrifugal
Material	SS

Table 9-9 (Cont'd)
Spent Fuel Cooling System Component Data

Flow, gpm	1,000
Head, ft H_2O	100
Motor Horsepower, hp	40
Spent Fuel Pool Volume, ft^3	73,000 (Units 1 & 2) 50,000 (Unit 3)
Spent Fuel Filter	
Design Flow Rate, gpm	180
Material	SS
Design Temperature, F	250
Design Pressure, psig	125
Code	ASME III-C
Borated Water Recirculation Pump	
Type	Vertical, inline, centrifugal
Material	SS
Flow, gpm	180
Head, ft H_2O	140
Motor Horsepower, hp	15
Design Temperature, F	250
Design Pressure, psig	125
Spent Fuel Demineralizer	
Type	Mixed bed
Material	SS
Resin Volume, ft^3	21
Flow, gpm	180
Design Temperature, F	250
Design Pressure, psig	125
Code	ASME III-C

9.5 LOW PRESSURE INJECTION SYSTEM

9.5.1 DESIGN BASES

The Low Pressure Injection System removes decay heat from the core and sensible heat from the Coolant System during the latter stages of cooldown. The system also provides auxiliary spray to the pressurizer for cooling, maintains the reactor coolant temperature during refueling, and provides a means for filling and draining the fuel transfer canal. In the event of a loss-of-coolant accident, the system injects borated water into the reactor vessel for long-term emergency cooling. The emergency functions of this system are described in Section 6.

9.5.2 SYSTEM DESCRIPTION AND EVALUATION

The Low Pressure Injection System is shown schematically in Figure 9-6. An independent system is provided for each unit. The Low Pressure Injection System normally takes suction from the reactor coolant outlet line and delivers the water back to the reactor through the core flooding nozzles after passing through the low pressure injection pumps and coolers. The Low Pressure Injection System may be lined up when the reactor pressure is below the system suction piping design pressure for cooldown of the system to refueling temperatures. The pressurizer is cooled by an auxiliary spray line. The decay heat is transferred to the Low Pressure Service Water System bv the low pressure
16. injection coolers. Component data are shown in Table 9-11.

The major system components are described as follows:

Low Pressure Injection Pumps

Three low pressure injection pumps are arranged in parallel with electric motor-operated valves in the suction line to each pump. The two outboard pumps are normally available for emergency operation, and the center pump is valved off on both the suction and discharge sides of the pump. During decay heat removal, any two of the three pumps are lined up to the low pressure injection coolers.

The design flow is that required to cool the reactor coolant system from 250 F to 140 F in 14 hours. The steam generators are used to reduce the Reactor Coolant System from operating temperature to the 250 F temperature.

Low Pressure Injection Coolers

The low pressure injection coolers, during a routine shutdown, remove the decay heat from the circulated reactor coolant. Both coolers are designed to cool the circulated reactor coolant from 250 F to 140 F in 14 hours.

Borated Water Storage Tank

The borated water storage tank is located outside the Reactor Building and the Auxiliary Building. It contains 2200 ppm boron in solution and is used for filling the fuel transfer canal during refueling and filling the incore instrumentation handling tank. The borated water storage tank also provides borated water for emergency core cooling and the Reactor Building Spray System.
16. Liquid level in the borated water storage tank is monitored by redundant level instrumentation.

9.5.2.1 Mode of Operation

(Reference Supplement 6 Revisions for Oconee 1.)
Two pumps and two coolers normally perform the decay heat cooling function for each reactor unit. After the steam generators have reduced the reactor coolant

temperature to 250 F, decay heat cooling is initiated by aligning pumps to take suction from the reactor outlet line and discharge through the coolers into the reactor vessel. The equipment utilized for decay heat cooling is also used for low pressure injection during accident conditions.

During refueling, the decay heat from the reactor core is rejected to the low pressure injection coolers in the same manner as it is during cooldown to 140 F. At the beginning of the refueling period, both coolers and both pumps are required to maintain 140 F in the core and fuel transfer canal. Later, as core decay heat decreases, one cooler and pump can maintain the required 140 F.

The fuel transfer canal may be filled by switching the suction of the low pressure injection pumps from the reactor outlet to the borated water storage tank. When the transfer canal is filled, suction to the pumps is switched back to the reactor outlet pipe. (Normally filled with the spent fuel cooling pumps as described in 9.4.2.)

After refueling, the transfer canal is drained by switching the discharge of one of the pumps from the reactor injection nozzle to the borated water storage tank. The other pump will continue the recirculation mode of decay heat removal.

9.5.2.2 Reliability Considerations

Since the equipment is designed to perform both normal and emergency functions, separate and redundant flow paths and equipment are provided to prevent a single component failure from reducing the system performance below a safe level. All rotating equipment and most valves are located in the Auxiliary Building to facilitate maintenance and periodic operational testing and inspection.

9.5.2.3 Codes and Standards

Each component of this system will be designed to the code or standard, as applicable, as noted in Table 9-11.

9.5.2.4 System Isolation

The Low Pressure Injection System is connected to the reactor outlet line on the suction side and to the reactor vessel on the discharge side. The system is isolated from the Reactor Building on the suction side by two electric motor-operated valves located inside the Reactor Building and one electric motor-operated valve located outside the Reactor Building. The discharge side is isolated from the Reactor Building by a check valve inside and an electric motor-operated valve outside the Reactor Building. All of these valves are normally closed whenever the reactor is in the operating condition. In the event of a loss-of-coolant accident, the valve on the discharge side opens, but the valves between the reactor vessel and the suction side of the pumps remain closed throughout the accident.

9.5.2.5 Leakage Considerations

During reactor power operation, all equipment of the Low Pressure Injection System is idle, and all isolation valves are closed. Under loss-of-coolant accident conditions, fission products may be recirculated in the coolant through the exterior piping system. Potential leaks have been evaluated to obtain the total radiation dose to the public due to leakage from this system. The evaluation is discussed in Sections 6 and 14.

9.5.2.6 Failure Considerations

Failures and malfunctions in the Low Pressure Injection System in conjunction with a loss-of-coolant accident are discussed in Section 6.

9.5.27 Operational Limits

Alarms or interlocks are provided to limit variables or conditions of operation that might affect system or station safety. These variables or conditions of operation are as follows:

Low Pressure Injection Flow Rate

Low flow from the pumps during the decay heat removal mode of operation is alarmed to signify a reduction or stoppage of flow and cooling to the core.

Reactor Coolant Pressure Interlock

The first valve from the reactor coolant system in the suction line to the low pressure injection pumps is interlocked with the Reactor Coolant System pressure instrumentation to prevent inadvertent overpressurization of the Low Pressure Injection System piping while the reactor coolant system is still above Low Pressure Injection System design pressure.

Reactor Coolant Leaving Low Pressure Injection Coolers

High temperature of the reactor coolant discharging from the low pressure injection coolers is alarmed to signal a loss of cooling capability in the respective low pressure injection cooler.

Table 9-10
Low Pressure Injection System Performance Data

Reactor Coolant Temperature at Startup of Decay Heat Removal, F	250
Time to Cool Reactor Coolant System From 250 F to 140 F, h	14
Refueling Temperature, F	140
Fuel Transfer Canal Fill Time, h	1
Fuel Transfer Canal Drain Time, h	2
Boron Concentration in Borated Water Storage Tanks, ppm	2,200

Table 9-11
Low Pressure Injection System Component Data

Pump (each)	
Type	Single stage, centrifugal
Capacity, gpm	3,000
Head at Rated Capacity, ft H_2O	350
Motor Horsepower, hp	400
Material	SS (wetted parts)
Design Pressure, psig	470/505
16. Design Temperature, F	300/250
Cooler (each) (Reference Supplement 9 Revisions for Oconee 3.)	
Type	Shell and tube
Capacity (at 140 F), Btu/h	30×10^6
Reactor Coolant Flow, gpm	3,000
Low Pressure Service Water Flow, gpm	3,000
Low Pressure Service Water Inlet Temp, F	75
Material, Shell/Tube	CS/SS
Design Pressure, Shell/Tube	150/370
Design Temperature, F	300
Code	ASME Section III-C, III and VIII
Borated Water Storage Tank (each)	
Capacity, gal	388,000
Material	CS/Coated inside
Design Pressure, psig	Atmospheric
Design Temperature, F	150
Code	AWWA D-100

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9.6 COOLING WATER SYSTEMS

9.6.1 DESIGN BASES

The cooling water systems for the station are designed to provide redundant cooling water supplies to insure continuous heat removal capability both during normal operation and during accident conditions.

The cooling water systems are designed so no single component failure will curtail normal station or impair emergency safeguards operation. Redundant pumping capability is provided, heat exchangers and pumps can be isolated and pressure reducing valves are provided with bypasses.

All cooling systems are designed to be operated and monitored from the control room. Component design parameters are given in Table 9-12.

The design purpose of each of the cooling water systems is outlined below:

Condenser Circulating Water (CCW) System - This system provides for cooling of the condensers during normal and emergency operation of the plant. The system also serves as the heat sink for decay heat removal during cooldown of the plant.

High Pressure Service Water (HPSW) System - This system provides a source of water for fire protection throughout the station.

Low Pressure Service Water (LPSW) System - These systems provide cooling water for normal and emergency services throughout the station. Safety related functions served by this system are:

(a) Reactor building cooling units.
(b) Low pressure injection coolers.

Recirculated Cooling Water (RCW) Systems - This is a closed loop system to supply inhibited cooling water to various components. This system has no direct safety related functions.

9.6.2 SYSTEM DESCRIPTION AND EVALUATION

9.6.2.1 Condenser Circulating Water System

The Little River arm of Lake Keowee is the source of water for the CCW systems. Figure 9-7 shows the arrangement of the systems with respect to the two branches of Lake Keowee. Each unit has four condenser circulating water pumps supplying water via two 11 ft. conduit into a common condenser intake header under the turbine building floor. The discharge from the condenser is returned to the Keowee River arm of Lake Keowee.

The suction of the condenser circulating pumps extends below the maximum drawdown of the lake. The intake structure is provided with screens which can be manually removed for periodic cleaning.

The CCW systems are designed to take advantage of the siphon effect so the pumps are required only to overcome pipe and condenser friction loss. The siphon is initiated at start-up by plant vacuum pumps and sustained during operation by the continuous priming vacuum pumps.

The CCW system has a 48 inch emergency discharge line to the Keowee hydro tailrace. This discharge line is connected to each of the three condensers of each unit. Under a loss-of-power situation, the emergency discharge line will automatically open and the CCW system will continue to operate as an unassisted siphon system supplying sufficient water to the condenser for decay heat removal and emergency cooling requirements. The vacuum is sustained by steam air ejectors.

9.6.2.2 High Pressure Service Water System

The schematic arrangement of the HPSW system is shown on Figure 9-8. This system is used primarily for fire protection throughout the Oconee station.

Two full size (6000 gpm at 117 psig) and one reduced size (500 gpm at 117 psig) high pressure service water pumps supply the high pressure system.

The 500 gpm pump will normally operate to keep pressure on the fire headers. In the event of a fire, one full size pump provides adequate capacity for fire protection service. The second full size pump is an installed spare. The HPSW pumps take suction from the CCW system. The HPSW and LPSW pump suctions are connected to the 42 inch cross-connection between the condenser cooling water inlet headers for the three units. Manual isolation valves are provided so that service water may be supplied from any or all of the inlet headers.

A 100,000 gallon elevated storage tank provides a back-up supply of water for fire protection.

9.6.2.3 Low Pressure Service Water System

The schematic arrangement of the LPSW system is shown on Figures 9-8 and 9-9. Units 1 and 2 share three 15,000 gpm LPSW pumps. The LPSW pumps and the HPSW pumps take suction from the 42 inch crossover line between the condenser inlet headers; two LPSW pumps are supplied by one suction line and the other pump is supplied by the other suction line. The HPSW system is connected to the LPSW at the LPSW pump discharge to provide emergency back-up.

The LPSW provides cooling for components in the turbine building, the auxiliary building, and in the reactor building. Two separate 24 inch lines provide the LPSW to the components in the auxiliary and reactor buildings. These two supply lines are further divided into four separate supply headers, two supplying the components in Unit 1 and two supplying the components in Unit 2. The low pressure injection coolers and the reactor building cooling units are supplied by separate LPSW supply lines. The return lines from the low pressure injection coolers and the reactor building coolers are also separated to beyond a remote operated valve.

For Unit 3, each of the two 15,000 gpm LPSW pumps take their suction from the CCW crossover. These pumps provide cooling water via separate supply lines to engineered safeguards equipment in the reactor building and the auxiliary building similar to Units 1 and 2. The return lines from the Unit 3 engineered safeguards are also separate to beyond a remote operated isolated valve.

The turbine building requirements for LPSW are supplied from other separate headers. The three pumps associated with Units 1 and 2 have a turbine building header serving the turbine building requirements for Units 1 and 2. The two pumps associated with Unit 3 also have a turbine building header to supply the Unit 3 requirements.

The separate flow paths serving the emergency safeguards equipment can be isolated by remote operated isolation valves.

The LPSW system is monitored and operated from the control room. Isolation valves are incorporated in all LPSW lines penetrating the reactor building.

The reactor building coolers are supplied by individual lines from the separate LPSW supply headers. Each inlet line is provided with a motor operated shutoff valve located outside the reactor building. Similarly, each discharge line from the coolers is provided with a motor operated valve located outside the reactor building. This allows each cooler to be isolated individually. During normal operation, flow through the coolers will be throttled for reactor building temperature control by the motor operated throttle valves on the discharge of each cooler. These valves open automatically upon engineered safeguards signal to provide full emergency flow through the coolers.

The LPSW flow to and from each reactor building cooler is measured. The inlet and outlet flow indications are compared and any excessive deviations are annunciated as an indication of cooler leakage.

24. The LPSW returned from the auxiliary building is monitored for radioactivity (11.1.2). Upon any indication of radioactivity in the effluent, the component suspected of leaking may be individually isolated.

The LPSW pumps are connected to the 4160 volt buses (8.2.2) which supply power to engineered safeguards equipemnt. The emergency power supply is adequate to operate all LPSW pumps upon a loss of off-site power.

During normal operation, the cooling requirements are supplied by operating one LPSW pump per unit. The LPSW requirement following a loss of coolant accident can also be supplied by one pump. The spare pump is started by the engineered safeguards actuation signal to provide redundancy for single failure criteria.

9.6.2.4 Recirculated Cooling Water System

The RCW system for the Oconee station is shown schematically in Figure 9-10. This system provides inhibited closed cycle cooling water to various components outside the reactor building including:

(a) R. C. pump seal return coolers
(b) Spent fuel cooling

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(c) Sample coolers
(d) Evaporator systems
(e) Various pumps and coolers in the turbine building

The RCW system consists of four (4) motor-driven pumps and four RCW heat exchangers to supply cooling water service to the three Oconee units. A 25,000 gallon surge tank provides a surge volume to accommodate temperature changes and leakage. Condenser circulating water is used to cool the RCW heat exchangers.

The RCW effluent from the auxiliary building is monitored for radioactivity (11.1.2). Leakage of radioactive fluids from any of the coolers in the auxiliary building will be indicated by these monitors. Separate monitors are provided on the return lines from the Unit 1 and 2 auxiliary building and the Unit 3 auxiliary building.

During normal operation (three units) three RCW pumps and three RCW heat exchangers will be in service. One pump and one heat eachanger are installed spares.

The RCW provides no engineered safeguards functions and does not penetrate the reactor building.

Table 9-12

Cooling Water Systems Component Data

Parameter	Value
Condenser Circulating Water Pumps	4 per unit
Flow (per pump), gpm	177,000
Design temperature, F	75
Design pressure, psig	15.7
High Pressure Service Water Pumps	2 for all units
Flow (per pump), gpm	6,000
Design temperature, F	75
Design pressure, psig	117
High Pressure Service Water Jockey Pump	1 for all units
Flow (per pump), gpm	500
Design temperature, F	75
Design pressure, psig	117
Low Pressure Service Water Pumps	5 for all units
Flow (per pump), gpm	15,000
Design temperature, F	75
Design pressure, psig	65
Recirculated Cooling Water Pumps	4 for all units
Flow (per pump), gpm	2,400
Design temperature, F	105
Design pressure, psig	76
Recirculated Cooling Water Heat Exchangers	4 for all units
Type	Shell and tube
Recirculating cooling water flow, each (shellside), gpm	2,400
Recirculating cooling water inlet temperature, F	105
Recirculating cooling water outlet temperature, F	90
Condenser circulating water inlet temperautre, F	75
Design pressure, shell/tube, psig	100/50
Design temperature, shell/tube, F	200/200
Tube material	Admiralty metal (SB-111)
Shell material	Carbon steel (SA-285C)

9.7 FUEL HANDLING SYSTEM

9.7.1 DESIGN BASES

9.7.1.1 General System Function

The fuel handling system shown on Figure 9-11 is designed to provide a safe, effective means of transporting and handling fuel from the time it reaches the station in an unirradiated condition until it leaves the station after postirradiation cooling. The system is designed to minimize the possibility of mishandling or maloperations that could cause fuel assembly damage and/or potential fission product release.

Separate fuel handling equipment is provided for each reactor. A common fuel storage area serves Units 1 and 2, while a separate fuel storage area is provided for Unit 3.

The reactors are refueled with equipment designed to handle the spent fuel assemblies underwater from the time they leave the reactor vessels until they are placed in a cask for shipment from the site. Underwater transfer of spent fuel assemblies provides an effective, economic, and transparent radiation shield, as well as a reliable cooling medium for removal of decay heat. Use of borated water assures reactor subcriticality during refueling.

9.7.1.2 New Fuel Storage

New fuel is stored in the spent fuel pool serving the respective unit. Maximum new fuel storage requirements occur in storage of the initial core for each unit, at which time the respective spent fuel pool remains otherwise unused. New fuel required for refueling can be accommodated without infringing on the spent fuel storage capability described in 9.7.1.3.

9.7.1.3 Spent Fuel Pool

Each spent fuel pool is a reinforced concrete pool lined with stainless clad plate, located in its respective auxiliary building. Each pool is sized to accommodate a full core of irradiated fuel assemblies in addition to the concurrent storage of the largest quantity of new and spent fuel assembiles predicated by the fuel management program, as shown below. The spent fuel assemblies are stored in racks in parallel rows having a center to center distance of 21 inches in both directions. This spacing is sufficient to maintain a k_{eff} of less than 0.92 when flooded. This value of k_{eff} is based on a
8. criticality calculation for an infinite 20.5 inch square array of 3.50 w/o fuel assemblies submerged in cold unborated water. Control rod assemblies requiring removal from the reactors are stored in the spent fuel assemblies.

	Units 1-2 Spent Fuel Pool	Unit 3 Spent Fuel Pool
Maximum Predicted Fuel Storage Requirements, assemblies	297	237
Actual Fuel Storage Capability, assemblies	336	216

9.7.1.4 Fuel Transfer Tubes

Two horizontal tubes are provided to convey fuel between each reactor building and the respective auxiliary building. These tubes contain tracks for the fuel transfer carriages, gate valves on the spent fuel pool side, and a means for flanged closure on the Reactor Building side. The fuel transfer tubes penetrate the spent fuel pool and the fuel transfer canal at its lower depth, where space is provided for the rotation of the fuel transfer carriage baskets.

9.7.1.5 Fuel Transfer Canal

The fuel transfer canal is a passageway in the Reactor Building extending from the reactor vessel to the Reactor Building wall. It is formed by an upward extension of the primary shield walls. The enclosure is a reinforced concrete structure lined with stainless clad plate to form a canal above the reactor vessel which is filled with borated water for refueling.

Space is available in the deeper portion of the fuel transfer canal for underwater storage of the reactor vessel internals upper plenum assembly. This portion of the fuel transfer canal can also be used for storage of the reactor vessel internals core barrel and thermal shield assembly by storing the upper plenum assembly in the upper end of the fuel transfer canal.

9.7.1.6 Fuel Handling Equipment

This equipment consists of fuel handling bridges, fuel handling mechanisms, fuel storage racks, control rod handling mechanisms, fuel transfer mechanisms, and shipping casks. In addition to the equipment directly associated with the handling of fuel, equipment is provided for handling the reactor vessel closure head and the upper plenum assembly to expose the core for refueling.

9.7.2 SYSTEM DESCRIPTION AND EVALUATION

9.7.2.1 Receiving and Storing Fuel

New fuel assemblies are received in shipping containers, unloaded and stored in the appropriate spent fuel pool. Special provisions will be made for storage of the initial core of Unit 1 prior to transferring fuel into the pool. After reactor shutdown, new fuel assemblies can be transferred from the spent fuel pool to the reactor building with the use of the fuel transfer mechanisms and the fuel transfer tubes.

9.7.2.2 Loading and Removing Fuel

Following the reactor shutdown and reactor building entry, the refueling procedure is begun by removal of the reactor closure head. The first step in this operation is to uncouple the control rods from the drive mechanism. An auxiliary
16. hoist is used for this and any other special purposes that may be required during refueling. The electrical and water connections to the head assembly are disconnected.

To close the annular space between the reactor vessel flange and fuel transfer canal floor, a seal plate supported by the head stand is lowered into position and bolted to the canal shield flange with appropriate gaskets.

Head removal and replacement time is minimized by the use of two stud tensioners which are also supported by the head assembly. The stud tensioners are hydraulically operated to permit preloading and unloading of the reactor vessel closure studs at cold shutdown conditions. The studs are tensioned to their operational load in two steps in a predetermined sequence. Required stud elongation after tensioning is verified by an elongation gage.

Following removal of the studs from the reactor vessel tapped holes, the studs and nuts are supported in the closure head bolt holes with specially designed spacers. Removal of the studs with the reactor closure head minimizes handling time and reduces the chance of thread damage. Two special alignment studs are installed in stud locations Nos. 15 and 45. The lift of the head and replacement after refueling is guided by these studs. These studs are also used to locate the index fixture used for aligning the plenum assembly during removal and replacement.

The reactor closure head is lifted out of the canal onto a head storage stand on the operating floor by a head and internals handling fixture attached to the polar crane. The stand is designed to protect the gasket surface of the closure head. The studs and nuts can be removed from the reactor closure head at the storage location for inspection and cleaning using special stud and nut handling fixtures. A storage rack is provided for several closure head studs and the alignment studs. The stud holes except for locations Nos. 15 and 45 are closed with special plugs that prevent water and/or other foreign substances from entering the holes. The fuel transfer canal is then filled with borated water.

The upper plenum assembly is removed from the reactor and stored underwater on a stand on the fuel transfer canal floor using the head and internals handling fixture and adaptor attached to the polar crane with an internals handling extension.

Refueling operations are carried out from two fuel handling bridges which span the fuel transfer canal in each reactor building. One bridge, the Main Bridge, is used to shuttle spent fuel assemblies from the core to the fuel transfer station and new fuel assemblies from the transfer station to the core. During this operation, the second bridge, the Auxiliary Bridge, is occupied with relocating partially spent fuel assemblies in the core as specified by the fuel management program.

The main bridge is equipped with two trolley-mounted hoists. One hoist (fuel handling mechanism) is equipped with a fuel grapple and the second hoist (control rod handling mechanism) houses the control rod grapple. The auxiliary bridge has only one trolley-mounted hoist equipped with a fuel grapple and is used primarily for shuffling or rearranging partially spent fuel assemblies from one position in the core to another.

The main bridge moves a spent fuel assembly from the core underwater to the transfer station where the fuel assembly is lowered into the fuel transfer carriage fuel basket. The control rod handling mechanism is used to transfer the control rod or orifice rod assembly to a new fuel assembly waiting in the second fuel transfer carriage basket. This new fuel assembly with control rod or orifice rod assembly is carried to the reactor by the main bridge and located in the core while the spent fuel assembly is being transferred to the spent fuel pool.

Spent fuel assemblies removed from the reactors are transported to the spent fuel pool from the reactor building via fuel transfer tubes by means of the fuel transfer mechanism. The fuel transfer mechanisms are underwater chain and sprocket driven carriages that run on tracks extending from each spent fuel pool through the transfer tubes and into the respective reactor building. Each of the two independently operated fuel transfer mechanisms which serve Units 1 and 2 is designed to operate in two directions so that either of the two reactor buildings can be serviced by one or two mechanisms as required. A rotating fuel basket is provided on each end of each fuel transfer carriage to receive fuel assemblies in a vertical position. The hydraulically operated fuel basket is rotated to a horizontal position for passage through the transfer tube, and then rotated back to a vertical position in the spent fuel pool or reactor building for vertical removal or insertion of the fuel assembly.

The spent fuel assemblies are removed from the fuel transfer carriage fuel basket using a fuel handling bridge equipped with a fuel handling mechanism and fuel grapple. This bridge spans the spent fuel pool and permits the refueling crew to store or remove new and spent fuel assemblies in any one of the storage rack positions.

Once refueling is completed, the fuel transfer canal is drained through a pipe located in the deep transfer station area. The canal water is pumped to the borated water storage tank to be available for the next refueling.

During operation of the reactors, the fuel transfer carriages are stored in the respective spent fuel pools, thus permitting the gate valve on the spent fuel pool side of each transfer tube to be closed and a blind flange to be installed on the reactor building side of each tube.

Space is provided in each spent fuel pool to receive a spent fuel shipping cask as well as provide for required fuel storage. Following a decay period, the spent fuel assemblies are removed from storage and loaded into the spent fuel shipping cask underwater for removal from the site.

Space is provided in each spent fuel pool to receive a spent fuel shipping cask as well as provide for required fuel storage. The layout of the fuel pool is shown on Figure 1-4 through 1-8. The cask area is located at the north end of the fuel pools and adjacent to the fuel racks. Following a decay period, the spent fuel assemblies are removed from storage and loaded into the spent fuel shipping cask under water for removal from the site. The spent fuel shipping cask does not pass over fuel storage racks, or any systems or equipment important to safety when being moved to or from the spent fuel pool.

The spent fuel cask handling facility consists of a 100-ton capacity overhead bridge crane with an 11 feet, 6 inch span. The hoist controls are five step magnetic, contactor reversing, secondary resistor type with time delay acceleration and a maximum speed of 9 feet per minute. The hoist is equipped with AC solenoid-operated brake system and an eddy-current brake. The bridge controls are the same as the hoist controls and are equipped with AC solenoid-operated brake system and has a maximum speed of 50 feet per minute. The trolley is a single speed, four feet/minute, magnetic contactor reversing type controller with AC solenoid-operated brake system. The cranes were designed in accordance with Electric Overhead Crane Institute's Specification No. 61, Class A. The cranes were tested in the shop by performing a running test, and load tested at the Oconee site to 98 percent of capacity. The running and load test results were satisfactory. Maintenance of the cranes is in accordance with ANSI B30.2. The structural and mechanical components of the crane are designed to have a minimum factor of safety of 2.5 based on yield strength and rated capacity. The hoist brake system consists of the dynamic AB 707 eddy-current control brake and a 13-inch solenoid-operated shoe brake (Whiting SESA). The bridge is equipped with a hydraulic brake for operating the crane from the cab and a solenoid-operated shoe brake for operating the crane by pendant control from the floor. The trolley is equipped with a solenoid-operated shoe brake. The hoist system is equipped with a 75 horsepower motor that produces 328 foot-pounds of torque at full load, 1200 rpm. The starting and instantaneous stalling torque is 902 foot-pounds. The hoist is equipped with a geared lower limit switch for block travel and a paddle-type upper limit switch to prevent a two-blocking situation from occurring.

The cranes are equipped with a sister type hook with safety latch. The hook was load tested and non-destructive tested in the shop. Bethanized wire rope with a safety factor of 6 was used. A lifting adaptor to be used between the yoke and the crane hook is also designed to support three times the load. The lifting adaptor is a stainless steel member approximately 24 feet long, used to lift the cask from the platform to the bottom of the spent fuel pool.

A decontamination area is located in the building adjacent to each spent fuel pool where the outside surfaces of the casks can be decontaminated prior to shipment by using high pressure spray water, detergent solutions and manual scrubbing to the extent required.

9.7.2.3 Safety Provisions

Safety provisions are designed into the fuel handling system to prevent the development of hazardous conditions in the event of component malfunctions accidental damage or operational and administrative failures during refueling or transfer operations.

All fuel assembly storage facilities maintain an eversafe geometric spacing of 21 inches between assemblies. The fuel storage racks are designed so that it is impossible to insert fuel assemblies in other than the prescribed locations, thereby assuring the necessary spacing between assemblies. Fuel handling and transfer containers are also designed to maintain an eversafe geometric array. Under these conditions, a criticality accident during refueling or storage is not considered credible.

Fuel handling equipment is designed to minimize the possibility of mechanical damage to the fuel assemblies during transfer operations. If fuel damage should occur, the amount of radioactivity reaching the environment will present no hazard. The fuel handling accident is analyzed in Section 14.

All spent fuel assembly transfer operations are conducted underwater. The water level in the fuel transfer canal provides a minimum of 9-1/2 ft. of water over the active fuel line of the spent fuel assemblies during movement from the core into storage to limit radiation at the surface of the water to less than 10 mrem/hr. The fuel storage racks provide a minimum of 23-1/2 ft. of water shielding over stored assemblies. The minimum depth of the water over the fuel assemblies and the thickness of the concrete walls of the storage pool are sufficient to limit the maximum continuous radiation levels in the working area to 2.5 mrem/hr.

Water in the reactor vessel is cooled during shutdown and refueling by the low pressure injection system described in 9.5. Adequate redundant electrical power supply assures continuity of heat removal. The spent fuel pool water is cooled by the spent fuel cooling system described in 9.4 A power failure during the refueling cycle will create no immediate hazardous condition due to the large water volume in both the transfer canal and spent fuel pool. With a normal quantity of spent fuel assemblies in the storage pool and no cooling available, the water temperature in the spent fuel pool would increase very slowly (9.4.2.1.2).

During reactor operations, bolted and gasketed closure plates, located on the reactor building flanges of the fuel transfer tubes, assure that spent fuel pool water will not leak into the transfer canal in the event of a leak through the transfer tube valves. Both the spent fuel pool and the fuel transfer canals are completely lined with stainless clad steel plate for leaktightness and for ease of decontamination. The fuel transfer tubes will be appropriately attached to these liners to maintain leak integrity. The spent fuel pool cannot be accidently drained by gravity since water must be pumped out.

During the refueling period the water level in both the fuel transfer canal and the spent fuel pool is the same, and the fuel transfer tube valves are open. This eliminates the necessity for interlocks between the fuel transfer carriages and transfer tube valve operations except to verify full-open valve position.

The fuel transfer canal and spent fuel pool water will have a boron concentra-
16. tion of at least 1800 ppm. Although this concentration is sufficient to maintain core shutdown if all of the control rod assemblies were removed from the core, only a few control rods will be removed at any one time during the fuel shuffling and replacement. Although not required for safe storage of spent fuel assemblies, the spent fuel pool water will also be borated so that the transfer canal water will not be diluted during fuel transfer operations.

The fuel transfer mechanisms permit initiation of the fuel basket rotation from the building in which the fuel basket is being loaded or unloaded. Carriage travel and fuel basket rotation are interlocked to prevent inadvertent carriage movement when the fuel baskets are in the vertical position. Rotation of the fuel baskets is possible only when the carriages are in the rotating frame at the end of travel.

Interlocks are provided to prevent operation of the bridges or trolleys until the assemblies have been hoisted to the upper limit in the mast tube. Mandatory slow zones are provided for the hoisting mechanisms as the grapples approach the core and fuel baskets during insertion of fuel assemblies. The slow zones will be in effect during entry into the reactor core or fuel storage rack and just before and during bottoming out of the fuel assemblies. The controls are appropriately interlocked to prevent simultaneous movement of the bridge, trolley or hoists. The grapple mechanisms are interlocked with the hoists to prevent vertical movement unless the grapples are either fully opened or fully closed. The fuel grapple is so designed that when loaded with the fuel assembly, the fuel grapple cannot be opened as a result of operator error, electrical or hydraulic failure.

All operating mechanisms of the system are located in the fuel handling and storage area for ease of maintenance and accessibility for inspection prior to start of refueling operations. All electrical equipment, with the exception of some limit switches, is located above water for greater integrity and ease of maintenance. The hydraulic systems which actuate the fuel basket rotating frame use pool water for operation.

Relief valves are provided on each stud tensioner to prevent overtensioning of the studs due to excessive pressure.

Suspected defective fuel is removed from the core and tested for leakage. Leakage verification will utilize a container which can be sealed and allow for sampling for fission products. If sampling shows the presence of fission products indicative of clad failures, the fuel element may be transferred to the spent fuel pool in a transfer container which will prevent the grass dispersion of fuel particles in the transfer carriage. The defective fuel assemblies may be stored to await shipment utilizing required precautions to minimize release of gaseous fission products.

The fuel handling bridges are limited to handling of fuel assemblies and control rod and orifice rod assemblies only. All lifts for handling of reactor closure heads and reactor internal assemblies will be made using the reactor building cranes.

Travel speeds for the fuel handling bridges, hoists and fuel transfer carriages will be controlled to assure safe handling conditions.

The spent fuel cask currently planned for use at Oconee is equipped with a detachable impact absorber, which is attached during transport, and which permits the cask to qualify under 10CFR71. This impact structure is not in place during in-plant cask movements due to considerations for decontamination and pool and lifting clearances. Following decontamination, the cask is lowered approximately 49 feet from the elevation of the edge of the spent fuel pool to the elevation of the cask loading area. The shipping cask planned for use at Oconee can sustain the 30 foot drop described in 10CFR71, Appendix B; however, this is one of a series of acceptance tests for cask design, rather than a criterion for station design. In the very unlikely event of an accidental cask drop, the postulated release of all of the volatile fission products present does not exceed, at the site boundary, the limits of 10CFR100, the applicable regulation for accidents occurring inside the station.

9.8 STATION VENTILATION SYSTEMS

9.8.1 DESIGN BASES

The station ventilation systems are designed to provide a suitable environment for equipment and personnel. The path of ventilating air in potentially radioactively contaminated areas of the Auxiliary Building is from areas of low activity toward areas of progressively higher activity for ultimate discharge to the unit vent.

The design is based upon the following ambient conditions:

	Winter, °F		Summer, °F	
	Outside	Inside	Outside	Inside
Turbine Building				
All Areas	5	50	95	104
Service Building				
Shops and Locker Area	5	70	95	104
Warehouse	5	50	95	104
Administration Building	5	74	95	74
Auxiliary Building	5	70	95	104
Control Room	5	74	95	74
Cable Room	5	74	95	74
Equipment Room	5	74	95	86

In addition to the ventilation systems serving the above areas, which are discussed here, the Reactor Building Ventilation System is discussed in 5.3 and the Penetration Room Ventilation System is discussed in 6.5.

9.8.2 SYSTEM DESCRIPTION AND EVALUATION

The station ventilation systems discussed in this section are shown on Figures 9-12 and 9-13, with the design parameters for major components given in Table 9-13. The following is a brief description of each system:

9.8.2.1 Auxiliary Building Ventilation System

The Auxiliary Building is served by a low pressure fan-duct system shown on Figure 9-12 consisting of inlet louvers, roughing filters, steam coils, cooling coils supplied with low pressure service water, 100 percent capacity supply fans, supply duct, exhaust ducts, and exhaust fans including one spare full capacity exhaust fan for each main exhaust duct. The exhaust fans and supply fans are manually balanced to achieve a negative pressure throughout the Auxiliary Building. Desired temperature is maintained by throttling steam to the steam coils or low pressure service water to the cooling coils as required. Standby spot cooling is provided in the pump rooms by recirculating fan-coil units.

The Auxiliary Building Ventilation System is monitored by a radiation monitor, RIA-32. Air from up to 24 locations is pulled through the detector by a sample pump with the detector output logged on a recorder in the control room. High radiation level and loss of sample flow are annunciated. Spent fuel building air is continuously monitored by RIA-41.

9.8.2.2 Control Area Ventilation System

The control room is served by two separate 100 percent capacity low pressure
1. fan-duct systems shown on Figure 9-13, composed of outside air filter unit,
roughing filters, chilled water coil, fan, electric heater and duct system.
The outside air for control room pressurization is filtered through the outside
air filter unit before entering the ventilation system. The outside air filter
8. unit consists of a roughing filter, high efficiency particulate absolute filters,
charcoal filters and a booster fan to pressurize the control room with outside
air. The chilled water for cooling the control area is supplied from the station
chilled water system. This system is capable of supplying sufficient chilled
water for all necessary systems with 50 percent of the system out of service.

The radiation monitor, RIA-39, has a continuous sample of control room air pumped through the detector. High radiation level and loss of sample flow are annunciated.

9.8.2.3 Turbine Building Ventilation System

The Turbine Building is ventilated by 12 roof mounted exhaust fans. Ventilation air is pulled through outside air louvers in the exterior Turbine Building wall and is discharged through the roof mounted exhaust fans. The Turbine Building temperature is controlled in winter by steam unit heaters.

9.8.2.4 Administration Building Ventilation System

The Administration Building is served by a low pressure multizone type air handling unit with a chilled water cooling coil for building cooling and a low pressure steam coil for heating. This system is shown on Figure 9-13. Makeup ventilation air is taken through an outside wall louver into the filter casing and discharged directly to the atmosphere through the building exhaust system. Chilled water for this cooling coil is from the station chilled water system and the low pressure steam for building heating is from the auxiliary steam header.

9.8.3 CODES

The work, equipment, and materials conform to the requirements and recommendations of the following codes and standards as applicable:

American Society of Heating, Refrigeration and Air Conditioning Engineers Guide
National Electric Code
Air Moving and Conditioning Association
National Fire Protection Association Pamphlet 90A

Table 9-13

Ventilation System Major Component Data

System	Equipment	Capacity	Number Installed	Number Required Normal Operation
Auxiliary* Building	Supply Fan	45,000 cfm	1	1
	Supply Fan	46,500 cfm	1	1
	Supply Fan	9,500 cfm	1	1
	Supply Fan	17,230 cfm	1	1
	Exhaust Fan	31,000	2	1
	Exhaust Fan	31,000	2	1
	Exhaust Fan	29,000	3	2
Control* Area	Supply Fan	41,100 cfm	2	1
	Booster Fan	2,700 cfm	2	1
	Outside Air Filter Units	2,700 cfm	2	1
	Roughing Filter	41,100 cfm	2	1
	Chilled Water Coil	41,100 cfm	2	1
	Electric Heater	56 kw	2	1
Turbine Building	Exhaust Fans	69,000 cfm	12	12
Administration Building	Supply Fan	24,900 cfm	1	1
	Roughing Filter	24,900 cfm	1	1
	Chilled Water Coil	24,900 cfm	1	1
	Steam Coil	24,900 cfm	1	1

* For First Two Units

9.9 COOLANT STORAGE SYSTEM

9.9.1 DESIGN BASES

The coolant storage system for each unit is designed to accommodate the accumulated coolant bleed over a core cycle, including startup expansion and coolant letdown to storage for boric acid reduction.

Two coolant bleed holdup tanks, each with a capacity of 11,000 ft^3, are provided for each unit. One tank provides storage for the reactor coolant prior to treatment by the Coolant Bleed Evaporator. The other tank provides additional storage for unprocessed bleed or can be used for storage of evaporator distillate awaiting reuse as feed to the reactor coolant system. An additional tank is provided for storage of the concentrated boric acid from the evaporator.

The storage of both unprocessed and processed bleed requires approximately 55 percent of the volume of the bleed holdup tanks for each unit. The tanks for all three units are arranged so that they can be utilized to store liquid from the other units if so desired.

The design volume of coolant removed from one unit during heatup and dilution from cold shutdown is approximately 9600 ft^3. This occurs near the end of the core cycle when boric acid concentrations are reduced. Earlier in life the bleed quantity would be less.

An additional requirement for coolant storage is the partial drain which occurs during refueling. The coolant is removed in a batch of approximately 6100 ft^3 per unit and returned to the reactor coolant system upon completion of refueling. Thus, it occupies storage capacity only during the period of refueling. The required storage volume for refueling operations of 6100 ft^3 is less than 10 percent of the total available capacity.

A quench tank, located inside the reactor building, condenses and contains any effluent from the pressurizer safety valves. The quench tank is sized to condense one normal pressurizer steam volume without relieving to the reactor building atmosphere.

9.9.2 SYSTEM DESCRIPTION AND EVALUATION

The Coolant Storage System is used for the collection and storage of reactor coolant liquid. The liquid is received from the high pressure injection system both as a result of reactor coolant expansion during startup and for boric acid concentration reduction during startup and normal operation. It is either conveyed to coolant bleed holdup tanks for storage or passed through deborating demineralizers for boric acid removal and returned as unborated makeup to the high pressure injection system. A spray nozzle in the coolant bleed tanks on the inlet line allow some of the gases to be released. Recirculating the tank allows further stripping action to occur. Liquid from the coolant bleed holdup tanks is pumped to the coolant treatment system for processing. A flow diagram with the necessary instrumentation and controls for operation is shown in Figure 9-14. Component data is shown in Table 9-14.

The quench tank, located inside the reactor building, condenses and contains effluent from the pressurizer safety valves and various vents. Liquid in the quench tank can be circulated through a cooler for temperature control, sampled and excess liquid pumped to the coolant bleed holdup tanks or the liquid waste disposal system.

The coolant bleed holdup tanks and the concentrated boric acid storage tanks are vented to the gaseous waste vent header to provide for filling and emptying without overpressurization or causing a vacuum to exist. In addition, each tank is equipped with a relief valve and a vacuum breaker. Pressurized nitrogen can be supplied to each tank to enable purging.

Instruments and controls for operation of this system are located in the control rooms. Instruments and controls for the coolant bleed holdup tanks and pumps and for the concentrated boric acid storage tanks and pumps are duplicated on the auxiliary control boards.

9.10 COOLANT TREATMENT SYSTEM

9.10.1 DESIGN BASES

The Coolant Treatment System is used for the recovery of boron and purification of water that is drained or bled from the reactor coolant system for reuse in the plant as needed.

The system is sized to process all of the reactor coolant bled from the reactor coolant system, as stated in Section 11, operating on an 8-hour per day basis. The coolant bleed evaporator feed tank is sized to hold sufficient feed for about 5 hours of evaporator operation. The evaporator is sized to process 5060 lb/h of feed with a concentration of 2200 ppm boron. A flow diagram with the necessary instrumentation and controls for operation is shown in Figure 9-15. Component Data is shown in Table 9-14.

9.10.2 SYSTEM DESCRIPTION AND EVALUATION

The Coolant Treatment System is operated on a batch basis. The system receives liquid from the coolant bleed holdup tanks through the coolant bleed evaporator demineralizers into the feed tank. The liquid is then pumped into the coolant bleed evaporator by the evaporator feed pump which maintains a level in the evaporator while the recirculating pump recirculates the liquid until the temperature is stabilized. The distillate pump returns the distillate to the feed tank until the distillate is of the desired quality for pumping to the condensate test tanks. The evaporator concentrate is sampled and normally pumped to the concentrated boric acid storage tanks at approximately 8700 ppm boron. The
1. evaporator concentrate can be allowed to increase to 26000 ppm boron and pumped to the drumming station (See 11.1.2.1) for ultimate disposal as solid waste.

The condensate test tanks are used as storage or holdup tanks. One tank contains clean (Demineralized Water) and is used to supply water to the demineralized water header and for makeup to the coolant bleed holdup tanks. The other tank is used for water of less purity and is used or disposed of in one of the following ways:

(1) Returned to the waste evaporator feed tank

(2) Returned to the coolant bleed evaporator feed tank

(3) Returned to the coolant bleed holdup tanks

(4) Released to the liquid waste effluent header.

Items 3 and 4 can be routed through the condensate demineralizers if desired.

9.11 AUXILIARY SERVICE WATER SYSTEM

9.11.1 SYSTEM DESCRIPTION:

The auxiliary service water system utilizes the plant CCW intake and discharge conduits as a source of raw cooling water for decay heat removal. (Fig. 9-16) These conduits are interconnected by crossovers and unwatering lines. An auxiliary service water pump located in the auxiliary building at Elev. 771 takes its suction from the Unit 2 intake conduit and discharges into the steam generators of each unit via separate lines into the auxiliary feedwater headers. The raw water is vaporized in the steam generator removing residual heat and dumped to the atmosphere.

The auxiliary service water pump is an end suction centrifugal pump with a rated capacity of 3000 gpm at a total head of 176 feet.

It has been submitted to the following tests:

1. A non-witness ASME hydro test
2. Witnessed performance test
3. Sonic testing of shaft
4. Mill test certificates for casing, impeller, and shaft
5. Certified caliper measurements

The pump power supply is taken from the 4160 volt standby Bus No. 1.

All valves required for operation of the auxiliary service water system are either check valves or manually operated. The pump suction is equipped with a normally open butterfly valve and the discharge with a check valve and normally open gate valve. The pump bypass is equipped with a globe valve. The individual lines to each steam generator auxiliary feedwater header are equipped with a check valve and two normally closed gate valves which are used to control flow.

Atmospheric steam dumps on each main steam lead are equipped with two normally closed gate valves which must be opened to reduce steam generator shell side pressure before placing the auxiliary service water system into operation. All non-embedded piping is Class F.

9.11.2 DESIGN BASIS:

The auxiliary service water system is designed for decay heat removal following a concurrent loss of the main feedwater system, auxiliary feedwater system, and decay heat removal system. The system will maintain decay heat removal for a minimum of 37 days.

SYSTEM ABBREVIATIONS

HP HIGH PRESSURE INJECTION
LP LOW PRESSURE INJECTION
CC COMPONENT COOLING WATER
RCW RECIRCULATED COOLING WATER
RC REACTOR COOLANT
SF SPENT FUEL COOLING
LWD LIQUID WASTE DISPOSAL
GWD GASEOUS WASTE DISPOSAL
CA CHEMICAL ADDITION & SAMPLING
LPSW LOW PRESSURE SERVICE WATER
HPSW HIGH PRESSURE SERVICE WATER
CCW CONDENSER CIRCULATING WATER
RBV REACTOR BUILDING VENTILATION
BS BUILDING SPRAY
CF CORE FLOODING
CS COOLANT STORAGE
CT COOLANT TREATMENT
FDW FEEDWATER
M MAIN STEAM
PR PENETRATION ROOM

MISC. ABBREVIATIONS

VB VACUUM BREAKER
RV RELIEF VALVE
E EXHAUSTER
H STEAM GENERATOR
S STRAINER
C COOLER OR HEAT EXCHANGER
M MIXER (AGITATOR)
ES ENGINEERED SAFEGUARD SIGNAL
D DRAIN
DW DEMINERALIZED WATER
G GAS ANALYZER
H_2 HYDROGEN
N_2 NITROGEN
S SAMPLING
VH VENT HEADER
V VALVE
-RB- REACTOR BUILDING
-SH- SHIELDING WALLS
V VENT
T TANK
P PUMP
IX DEMINERALIZER
F FILTER
AB AUXILIARY BUILDING
LO LOCKED OPEN

INSTRUMENTATATION SYMBOLS

INSTRUMENT FUNCTION (SECOND, THIRD & FOURTH LETTERS) / PROCESS VARIABLE (OR ACTUATION) FIRST LETTER		MEASURING											CONTROLLING						SWITCH		ALARM
			RECORDING	INDICATING	OBSERVATION GLASS	PRIMARY ELEMENT	TEST POINT	TRANSMITTER	INTEGRATOR	AMPLIFIER	LOGGING		RECORDING	INDICATING	BLIND	CONTROL VALVE	LOGGING		BLIND	INDICATING	ALARM
			R	I	G	E	X	T	Z	AM	L		RC	IC	C	CV	LC		S	IS	A
TEMPERATURE	T		TR	TI		TE	TX	TT			TL		TRC	TIC	TC	TCV	TLC		TS	TIS	TA
AVERAGE TEMP.	Ta		TaR	TaI				TaT			TaL		TaRC	TaIC	TaC		TaLC		TaS		TaA
DIFF. TEMP.	cT		dTR	dTI				dTT		dTAM			dTRC	dTIC	dTC					dTIS	dTA
PRESSURE	P		PR	PI			PX	PT			PL		PRC	PIC	PC	PCV	PLC		PS	PIS	PA
DIFF. PRESS.	dP		dPR	dPI			dPX	dPT			dPL		dPRC	dPIC	dPC	dPCV	dPLC		dPS	dPIS	dPA
FLOW	F		FR	FI	FG	FE	FX	FT	FZ	FAM	FL		FRC	FIC	FC	FCV	FLC		FS	FIS	FA
LEVEL	L		LR	LI	LG		LX	LT		LAM	LL		LRC	LIC	LC	LCV	LLC		LS	LIS	LA
MANUAL	M													MIC					MS	MIS	
CONDUCTIVITY	C		CR	CI		CE		CT													CA
PH	PH		PH R	PH I		PH E	PH X	PH T		PH AM	PH L		PH RC	PH IC	PH C		PH LC				PH A
HUMIDITY	H		HR	HI		HE		HT			HL								HS		HA
POSITION	Po		PoR	PoI				PoT			PoL		PoRC	PoIC	PoC		PoLC		PoS	PoIS	PoA
RADIATION	R		RR	RI		RE				RAM	RL										RA
OXYGEN	O_2		O_2R	O_2I			O_2X	O_2T			O_2L								O_2S		O_2A
SPEED	S		SR	SI				ST			SL		SRC	SIC	SC		SLC				SA
VIBRATION	V		VR	VI				VT		VAM	VL										VA
HYDROGEN	H_2		H_2R	H_2I			H_2X	H_2T			H_2L								H_2S		H_2A
GAS ANALYSIS	G							GT													
QUANTITY	Q														QC						

LOCATION OF CONTROL POINT FOR REMOTE OPERATED VALVES AND OTHER EQUIPMENT DESIGNATED BY * (SEE LOCATION LEGEND) ON VALVE OPERATOR OR SWITCH SYMBOL.

VALVE SYMBOLS

GATE VALVE
GLOBE VALVE
BUTTERFLY VALVE, OR
FOOT VALVE
TILTING DISC CHECK
SWING CHECK VALVE
LIFT CHECK VALVE
STOP CHECK VALVE
NEEDLE VALVE
RELIEF VALVE
VACUUM BREAKER
3-WAY VALVE
ANGLE VALVE
SPRING ACTUATED EXHAUST VALVE
SAUNDERS VALVE

VALVE POSITION SYMBOLS

NORMALLY OPEN
NORMALLY CLOSED
NORMALLY THROTTLED

VALVE OPERATOR SYMBOLS

* E S P

DIAPHRAGM OPERATOR
ELECTRIC OPERATOR
SOLENOID OPERATOR
PISTON OPERATOR
PO POWER OPERATOR UNESTABLISHED TYPE
VB-VACUUM BREAKER
RV-RELIEF VALVE
PA-PILOT ACTUATED
PCV PRESSURE CONTROL
FLOAT OPERATOR

FLOW DEVICES

NOZZLE
ORIFICE
FLOW METER IN LINE DISPLACEMENT
ROTARY FLOW INDICATOR

SWITCHES

*
DENOTES REMOTE MOUNTED SWITCH
DENOTES LOCALLY MOUNTED SWITCH

PROCESS INSTRUMENT IDENTIFICATION

SEE LOCATION LEGEND
*
DENOTES REMOTE MOUNTED INSTRUMENT
DENOTES LOCALLY MOUNTED INSTRUMENT

* LOCATION LEGEND

A - AUXILIARY CONTROL PANEL
C - CONTROL ROOM PANELS
R - AUXILIARY RACK
ES - ENGINEERED SAFEGUARDS PANEL
L - LOCAL
T - AUXILIARY SHUTDOWN PANEL

WHEN VALVES OR EQUIPMENT ARE CONTROLLED FROM MORE THAN ONE POINT, BOTH POINTS SHOULD BE SHOWN AS FOLLOWS:

C ES 2

NUMBER(S) BELOW "ES" DESIGNATE SAFEGUARDS ACTUATION CHANNEL(S).

C P 12 ES 2 A 2

LOCATION
PROCESS VARIABLE AND FUNCTION
IDENTICAL MEASUREMENT
ENGINEERED SAFEGUARDS CHANNEL
LOOP DESIGNATION IF REQUIRED
INSTRUMENT SUB SYSTEM OR VALVE NUMBER

ALARMS

PA HI LO
PRESSURE ALARM (HIGH-LOW)

EQUIPMENT IDENTIFICATION METHOD

XX-X # X

FIRST AND SECOND SYMBOLS ARE LETTERS AND ARE USED TO DESIGNATE A SYSTEM ABBREVIATION.

THIRD SYMBOL IS A LETTER AND IS USED TO DESIGNATE THE COMPONENT TYPE. (SEE ABBREVIATIONS).

FOURTH SYMBOL IS NUMERICAL AND REPRESENTS THE NUMBER OF DIFFERENT COMPONENTS OF A SIMILAR TYPE IN ANY GIVEN SYSTEM.

FIFTH SYMBOL IS A LETTER AND REPRESENTS THE NUMBER OF COMPONENTS SERVING IDENTICAL FUNCTIONS. IT IS OMITTED WHEN ONLY ONE COMPONENT IS INSTALLED FOR A GIVEN FUNCTION.

PROCESS EQUIPMENT SYMBOLS

DEMINERALIZER
HEAT EXCHANGER
W WATER FILTER
DUPLEX STRAINER
POSITIVE DISPLACEMENT PUMP
CENTRIFUGAL PUMP
HC FLUID HEATING COIL
CC FLUID COOLING COIL
VACUUM PUMP OR COMPRESSOR
EXHAUSTER
BLOWER OR FAN
CHECK DAMPER
LOUVER
AIR REGISTER
P A C AIR FILTERS P-PRE FILTER A-ABSOLUTE FILTER C-CHARCOAL FILTER
AGITATOR
VERTICAL SUMP PUMP

MISCELLANEOUS PIPE AND FITTING SYMBOLS

FLANGED JOINT
SPRAY HEAD
BLIND FLANGE
Y-STRAINER
CAPPED JOINT
LOOP SEAL
BLOCK ORIFICE
VENT PIPE
RESTRICTING ORIFICE
SCREEN AT PIPE INLET
EDUCTOR
PIPE REDUCER
HEAT TRACING
MH MANHOLE
T TRAP
RUPTURE DISC
FLEXIBLE HOSE
MH MANHOLE WITH RUPTURE DISC
CHEMICAL ADDITION FUNNEL
PIPE SYMBOL
SCHEDULE (WHEN SHOWN)
MATL.
SIZE
REACTOR BUILDING PENETRATION IDENTIFICATION NO.
COMPUTER SIGNAL

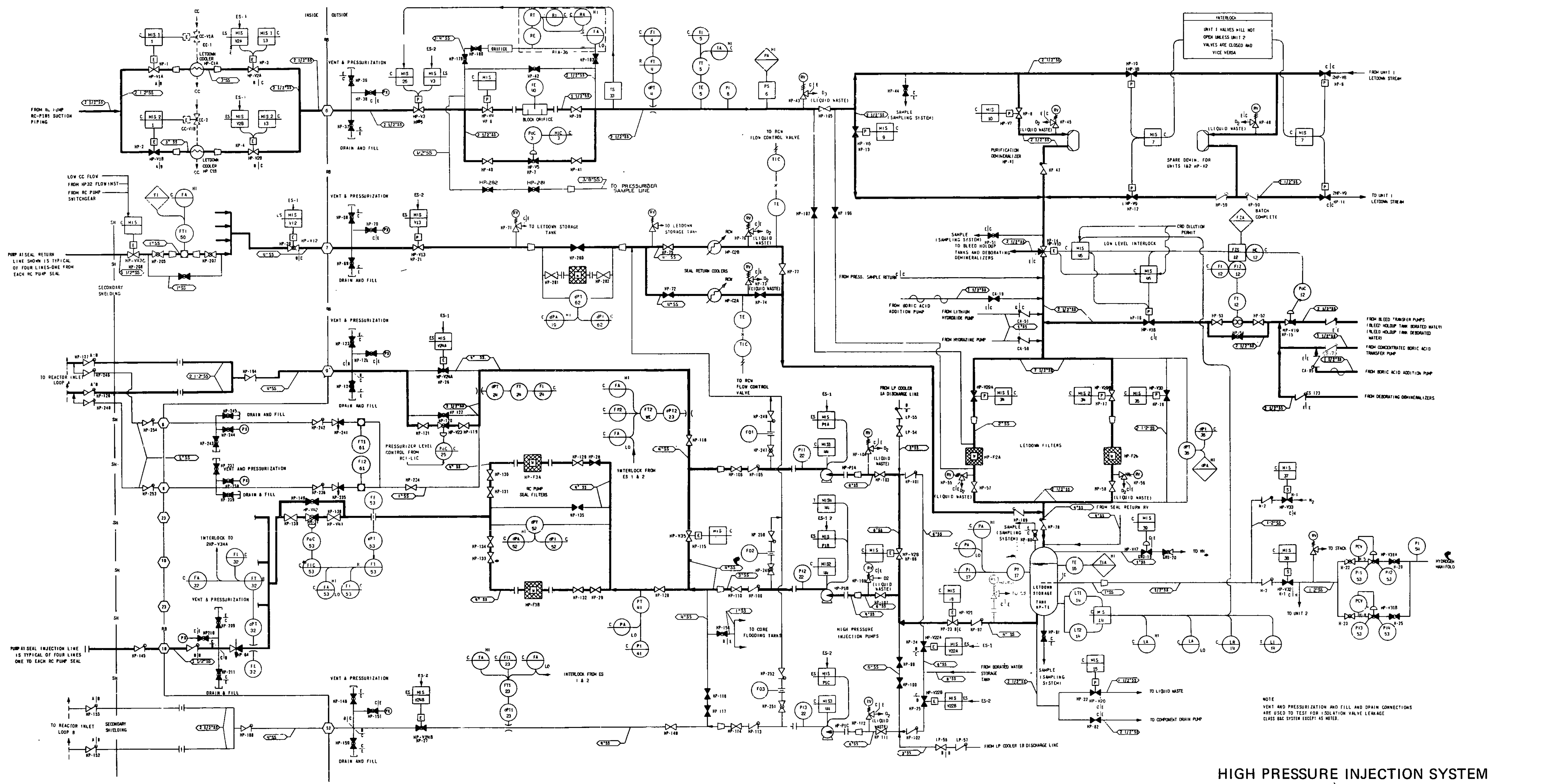
FLOW DIAGRAM IDENTIFICATIONS

DUKE POWER CITIZENSHIP SERVICE

OCONEE NUCLEAR STATION

Figure 9 - 1

Rev. 9 8/11/70

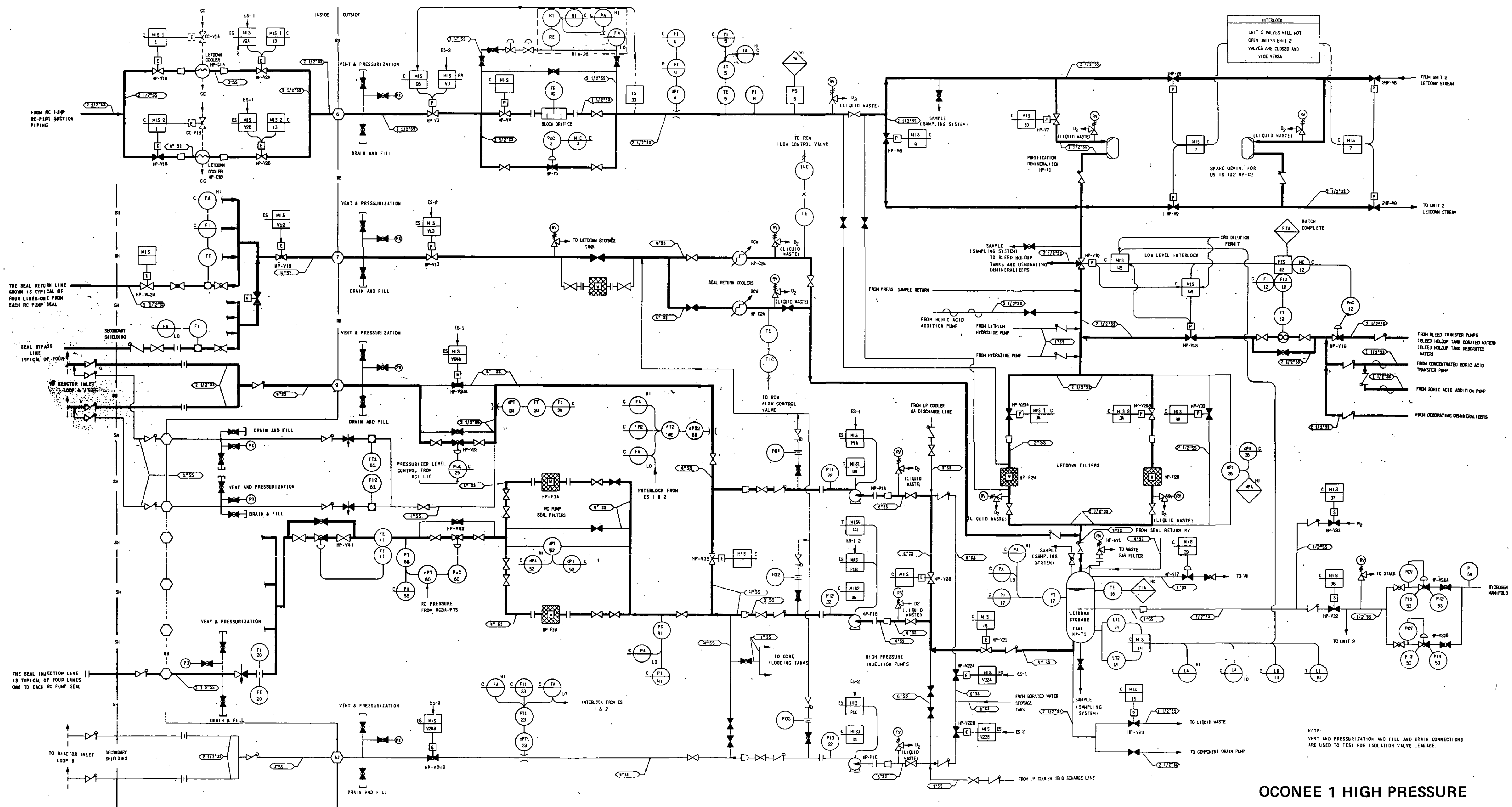


HIGH PRESSURE INJECTION SYSTEM
(Reference Supplement 6 Revisions for Oconee 1)

OCONEE NUCLEAR STATION

Figure 9 - 2

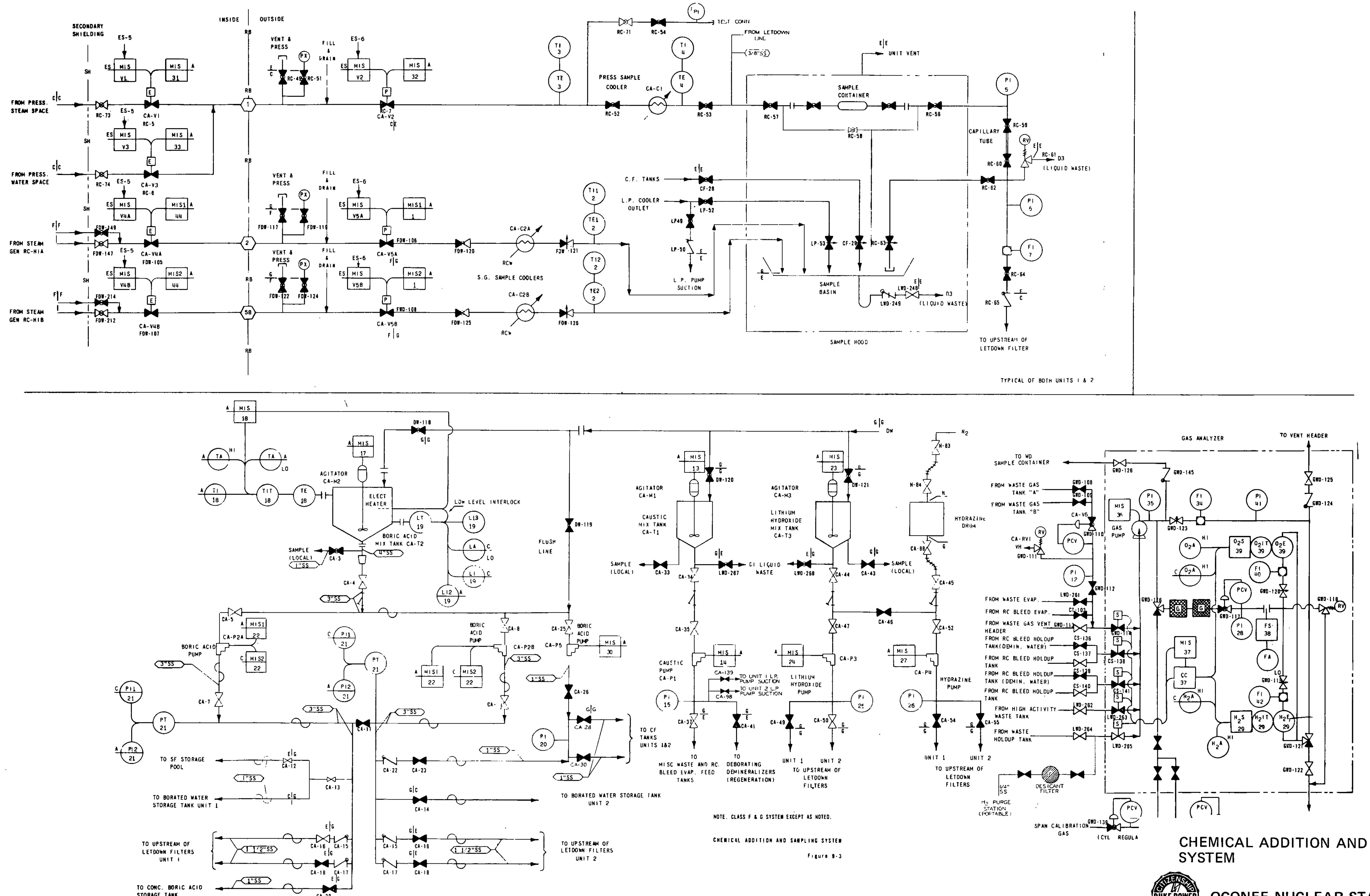
Rev. 3 3/16/70
Rev. 9 8/11/70
Rev. 16 7/30/71
Rev. 18 3/10/72



OCONEE 1 HIGH PRESSURE INJECTION SYSTEM

Figure 9 - 2A

(New) Rev. 9 8/11/70



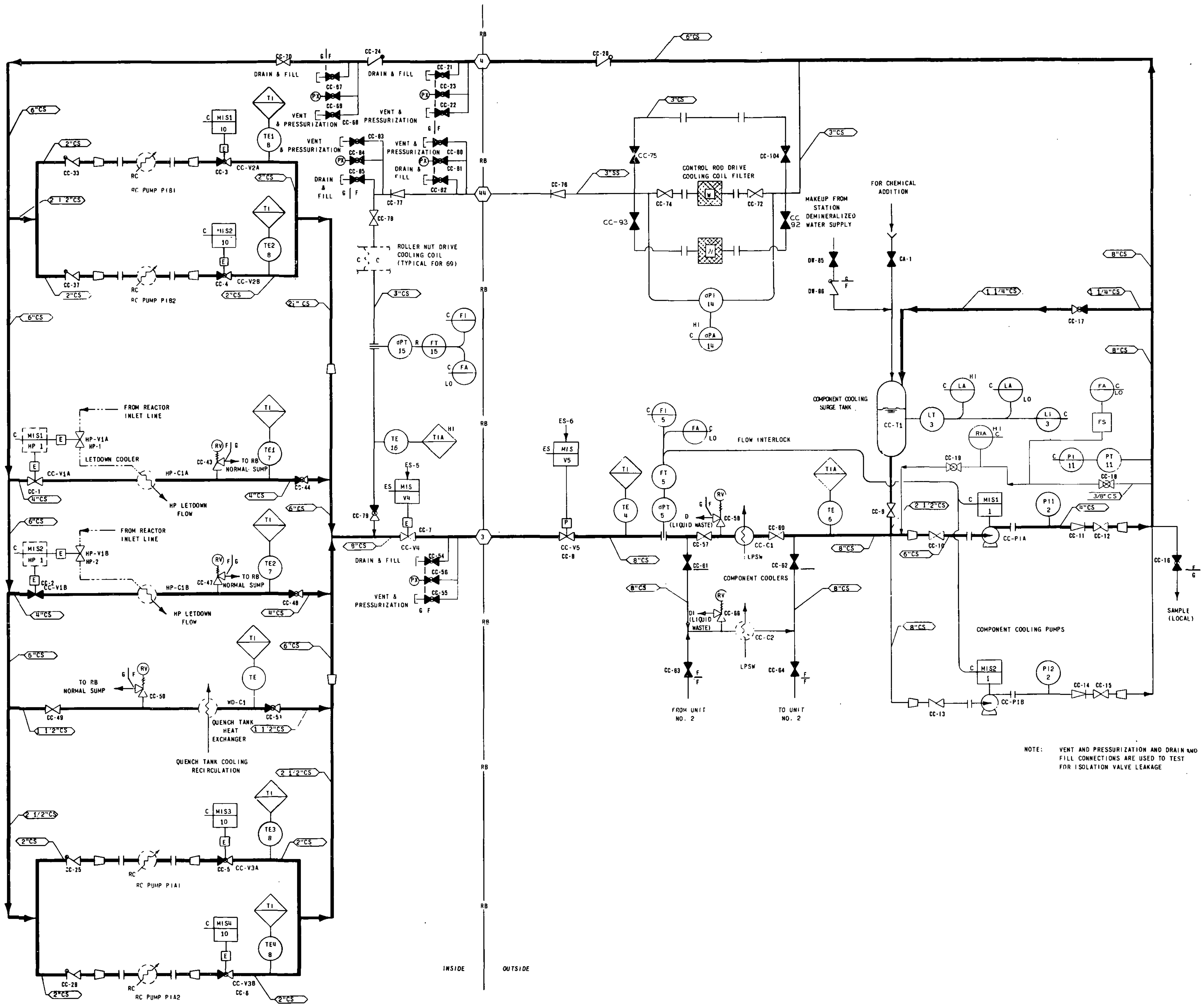
CHEMICAL ADDITION AND SAMPLING SYSTEM

OCONEE NUCLEAR STATION

Figure 9 - 3

Rev. 16 7/30/71

Rev. 18 3/10/72

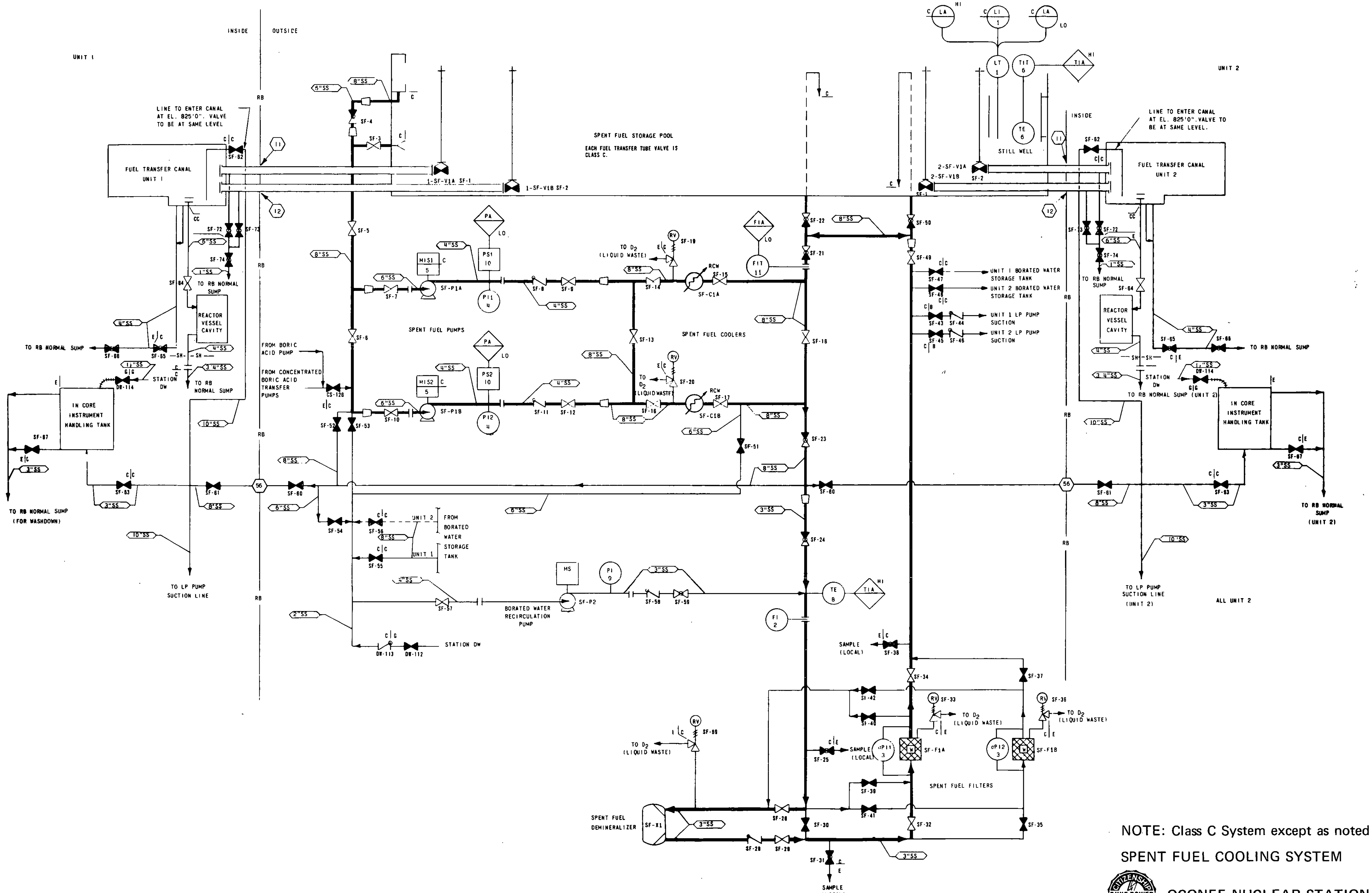


NOTE: Class F System except as noted

COMPONENT COOLING SYSTEM

OCONEE NUCLEAR STATION

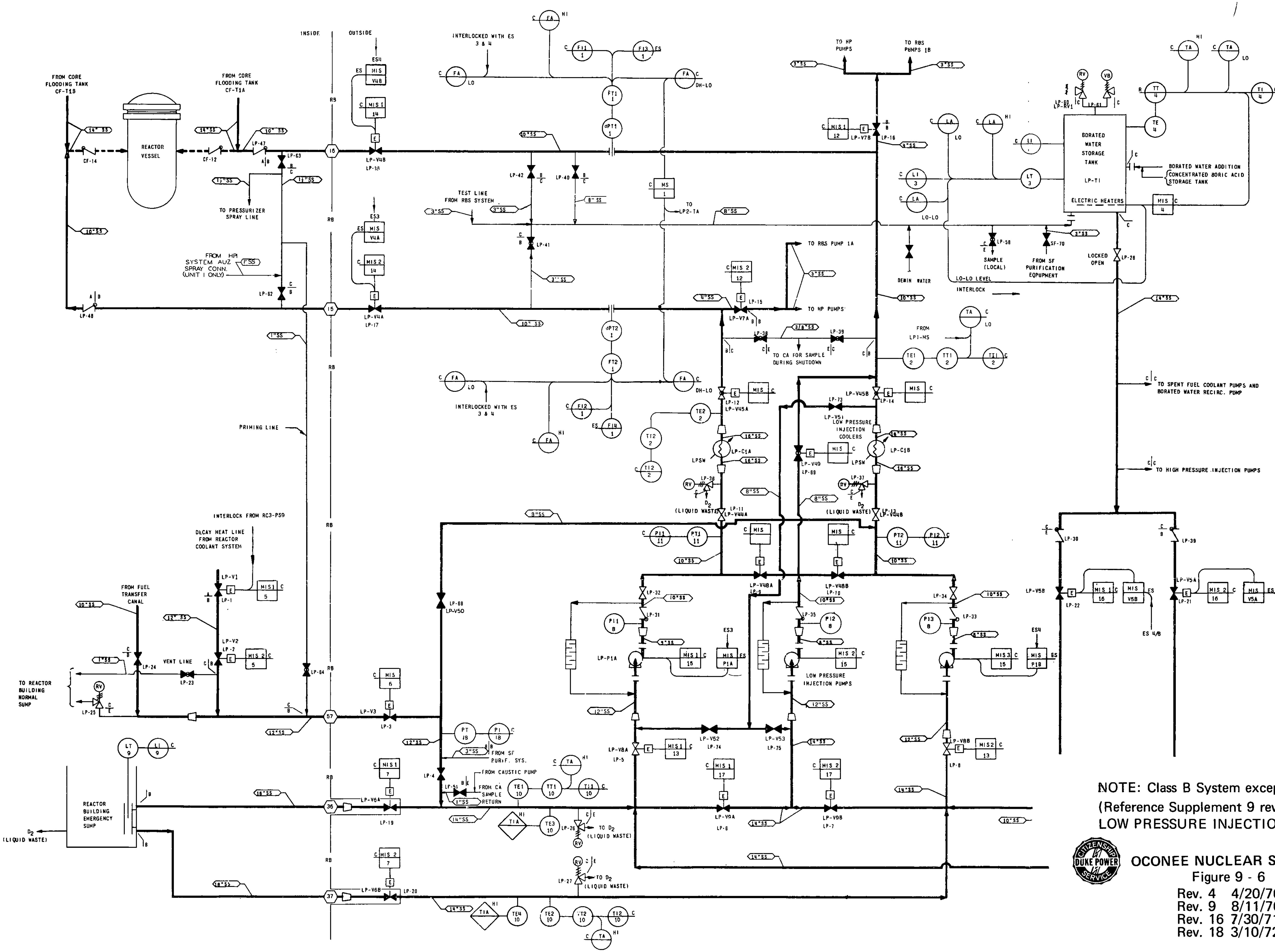
Figure 9 - 4
Rev. 16 7/30/71
Rev. 24 11/15/72
Rev. 26 1/29/73



NOTE: Class C System except as noted

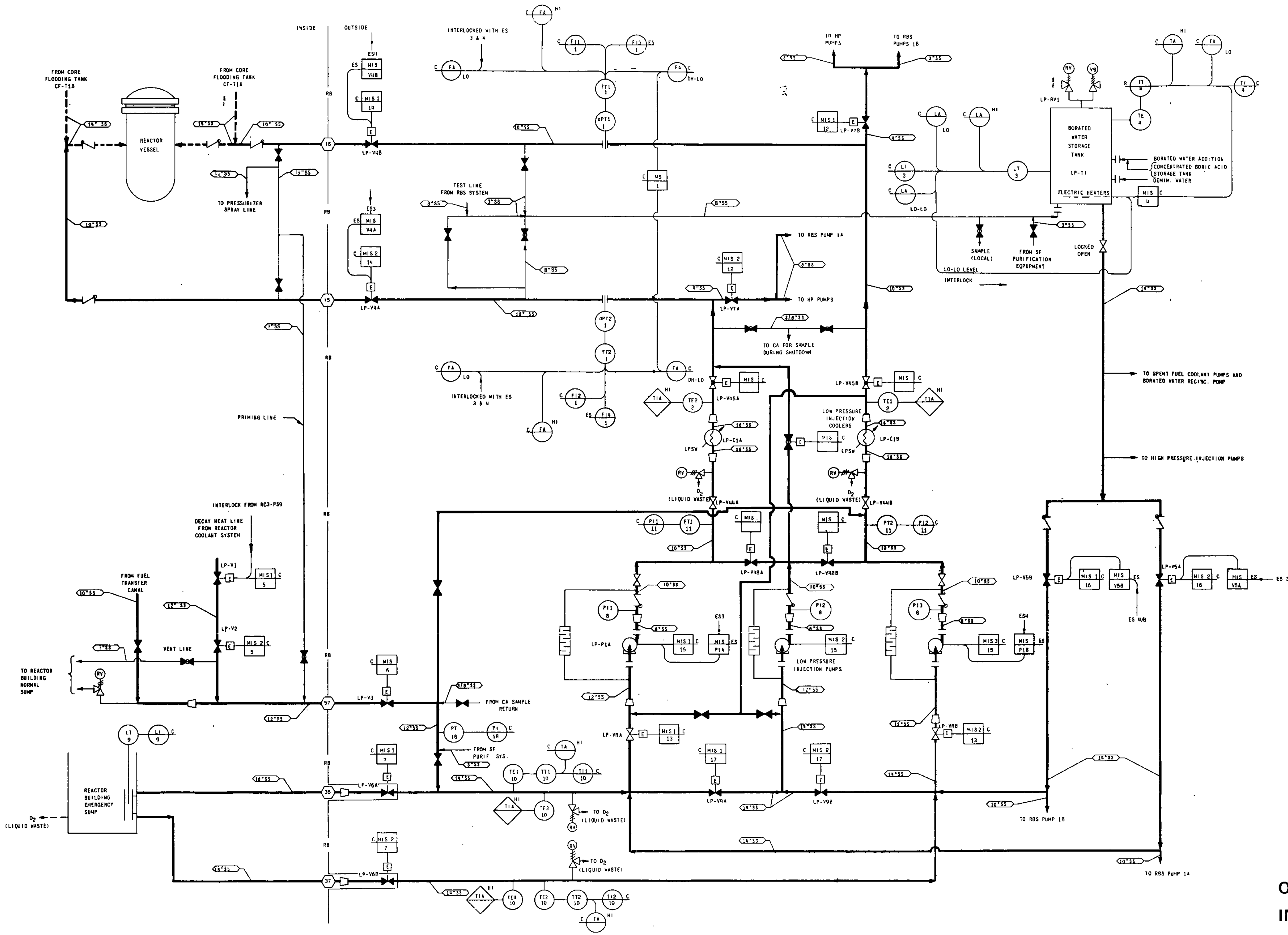
SPENT FUEL COOLING SYSTEM

OCONEE NUCLEAR STATION
Figure 9 - 5
Rev. 16 7/30/71



NOTE: Class B System except as noted.
(Reference Supplement 9 revisions for Oconee 3)
LOW PRESSURE INJECTION SYSTEM

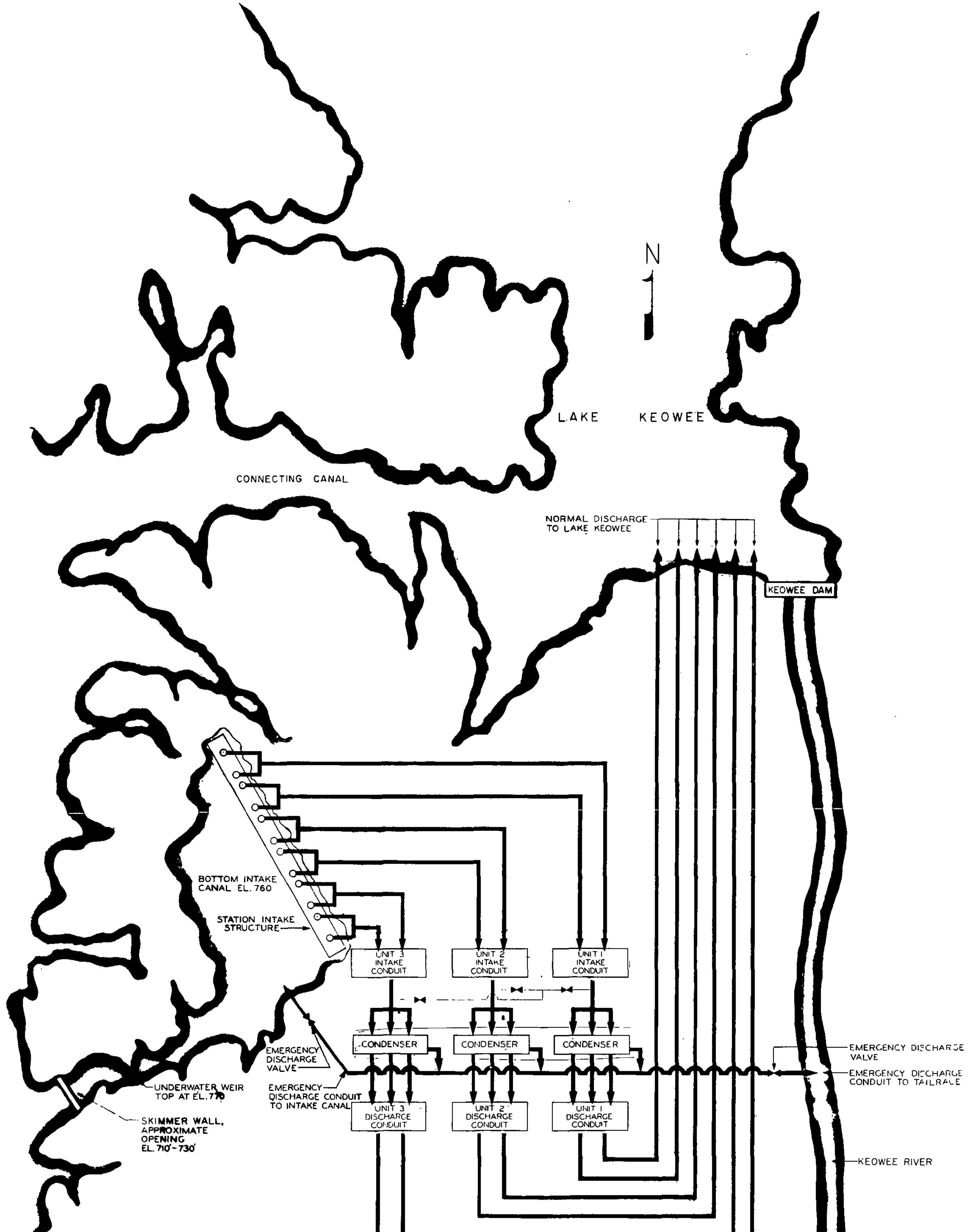
OCONEE NUCLEAR STATION
Figure 9 - 6
Rev. 4 4/20/70
Rev. 9 8/11/70
Rev. 16 7/30/71
Rev. 18 3/10/72



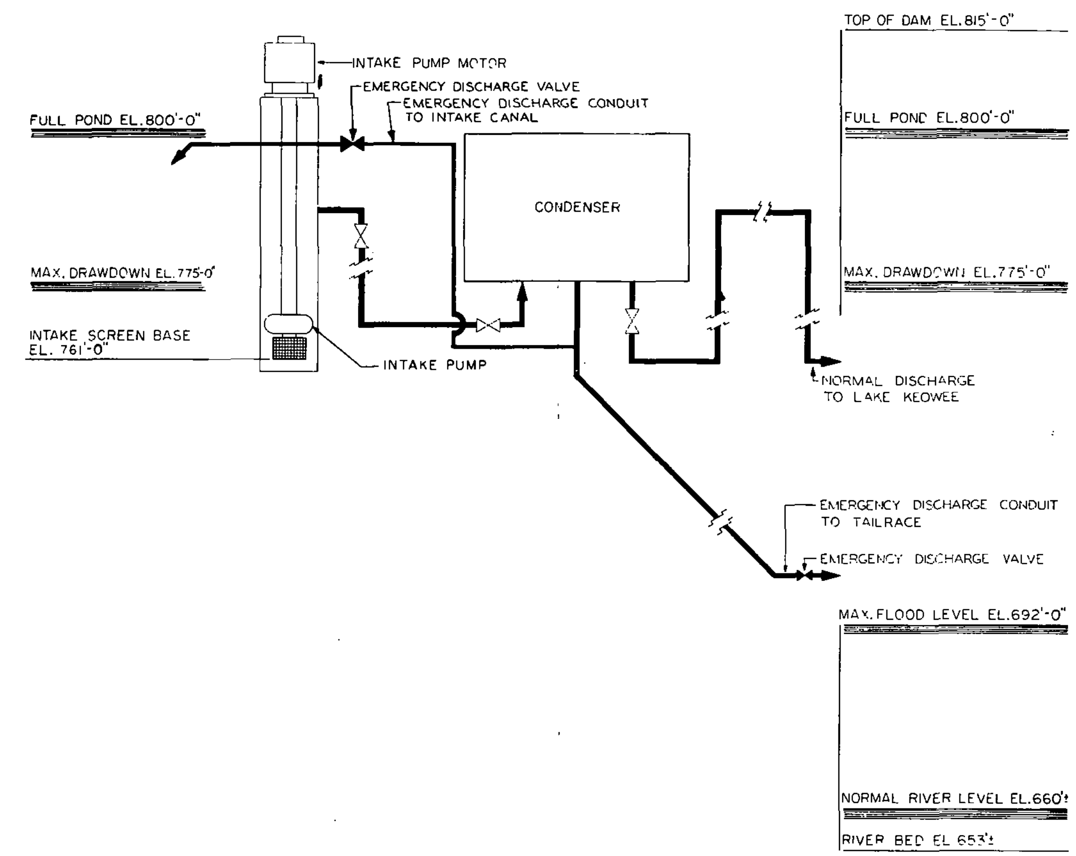
OCONEE 1 LOW PRESSURE INJECTION SYSTEM

Figure 9 - 6A

(New) Rev. 9 8/11/70



SCHEMATIC PLAN
NO SCALE



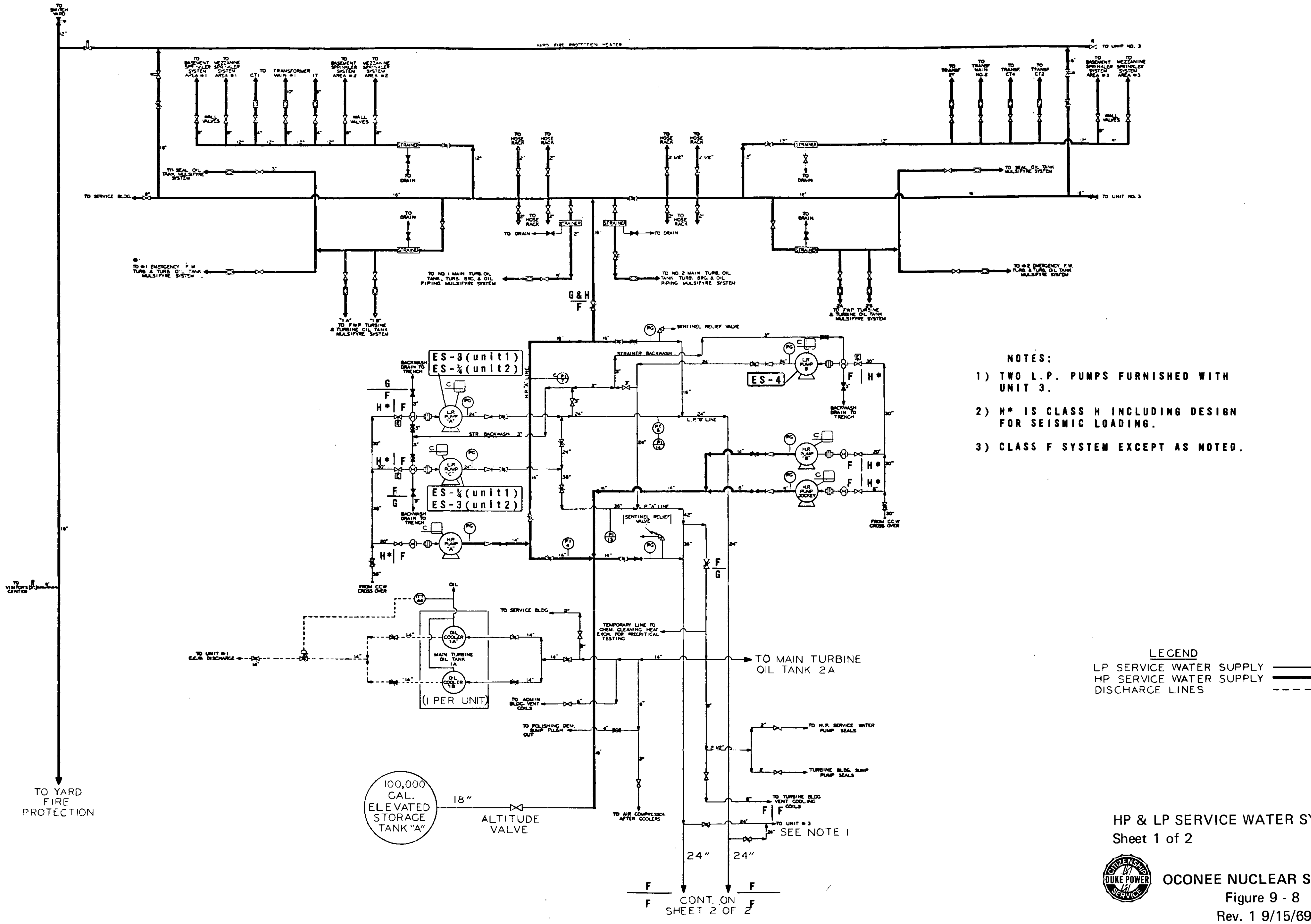
SCHEMATIC ELEVATION
NO SCALE

REVISED 5-29-67

CONDENSER CIRCULATING WATER SYSTEM

OCONEE NUCLEAR STATION

Figure 9 - 7



NOTES:

1) TWO L.P. PUMPS FURNISHED WITH UNIT 3.

2) H* IS CLASS H INCLUDING DESIGN FOR SEISMIC LOADING.

3) CLASS F SYSTEM EXCEPT AS NOTED.

LEGEND

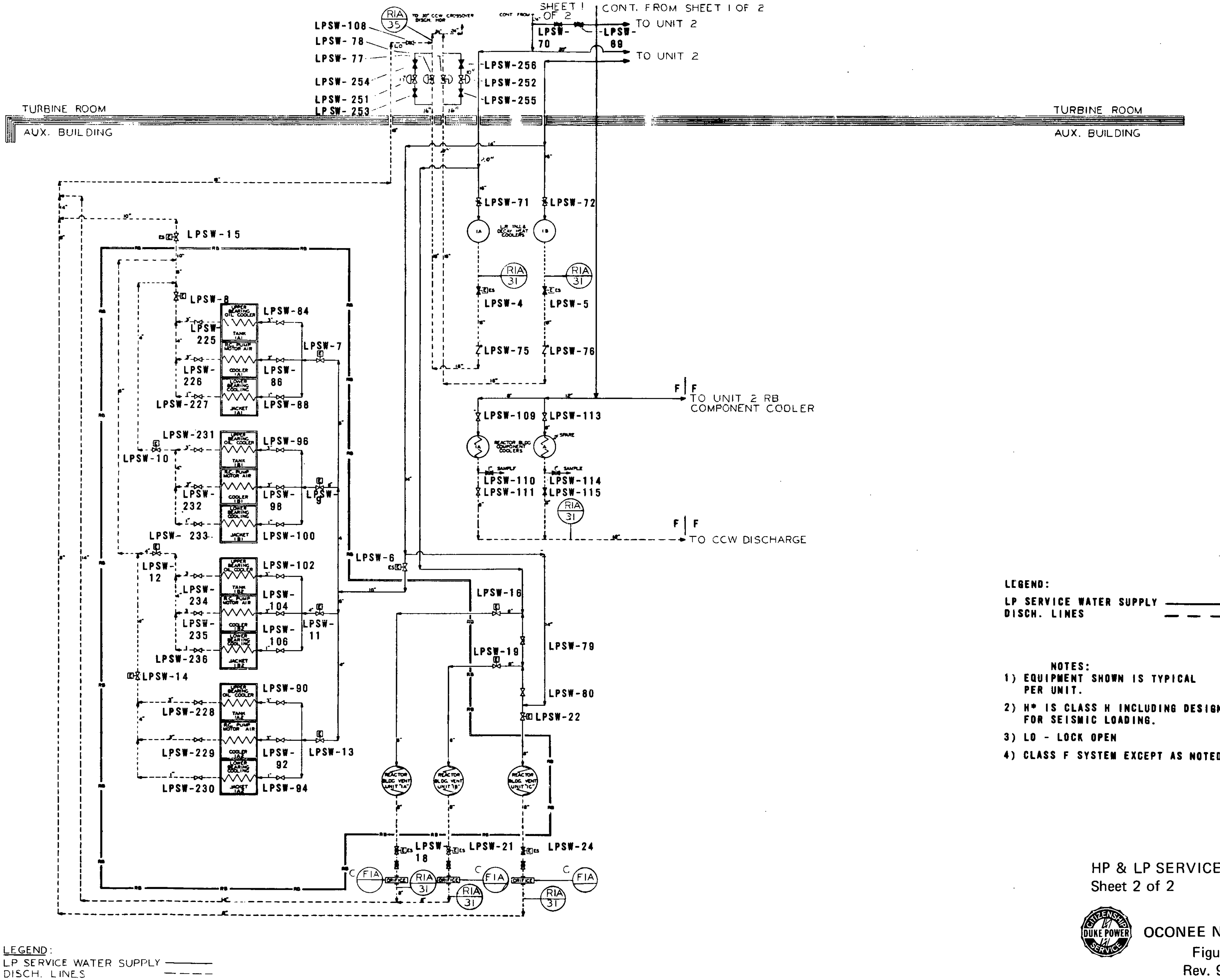
LP SERVICE WATER SUPPLY

HP SERVICE WATER SUPPLY

DISCHARGE LINES

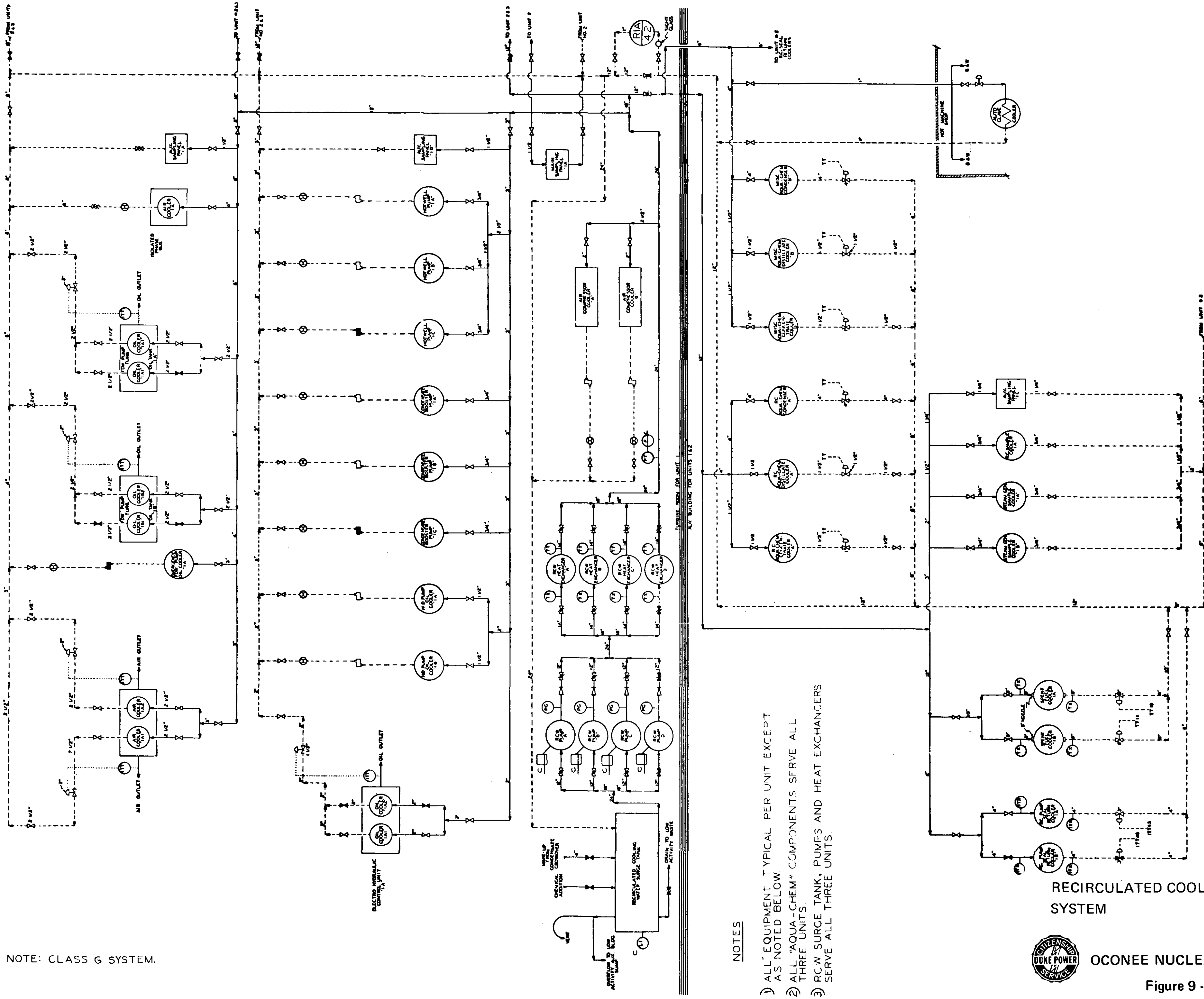
HP & LP SERVICE WATER SYSTEMS
Sheet 1 of 2

OCONEE NUCLEAR STATION
Figure 9 - 8
Rev. 1 9/15/69
Rev. 16 7/30/71



HP & LP SERVICE WATER SYSTEMS
Sheet 2 of 2

OCONEE NUCLEAR STATION
Figure 9 - 9
Rev. 9 8/11/70
Rev. 16 7/30/71



NOTES

1) ALL EQUIPMENT TYPICAL PER UNIT EXCEPT AS NOTED BELOW.
2) ALL "AQUA-CHEM" COMPONENTS SERVE ALL THREE UNITS.
3) RCW SURGE TANK, PUMPS AND HEAT EXCHANGERS SERVE ALL THREE UNITS.

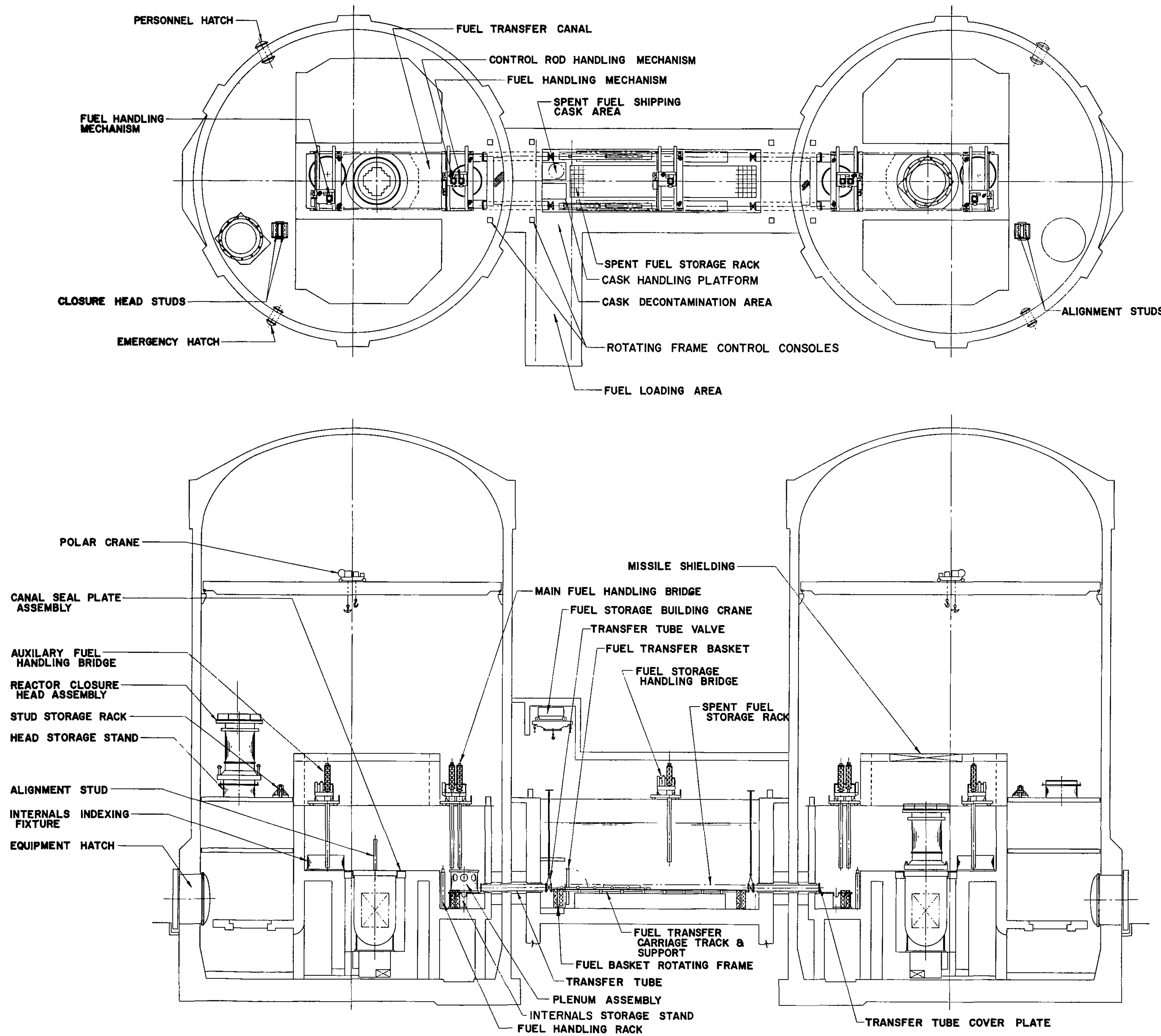
NOTE: CLASS G SYSTEM.

RECIRCULATED COOLING WATER SYSTEM

OCONEE NUCLEAR STATION

Figure 9 - 10

Rev. 16 7/30/71



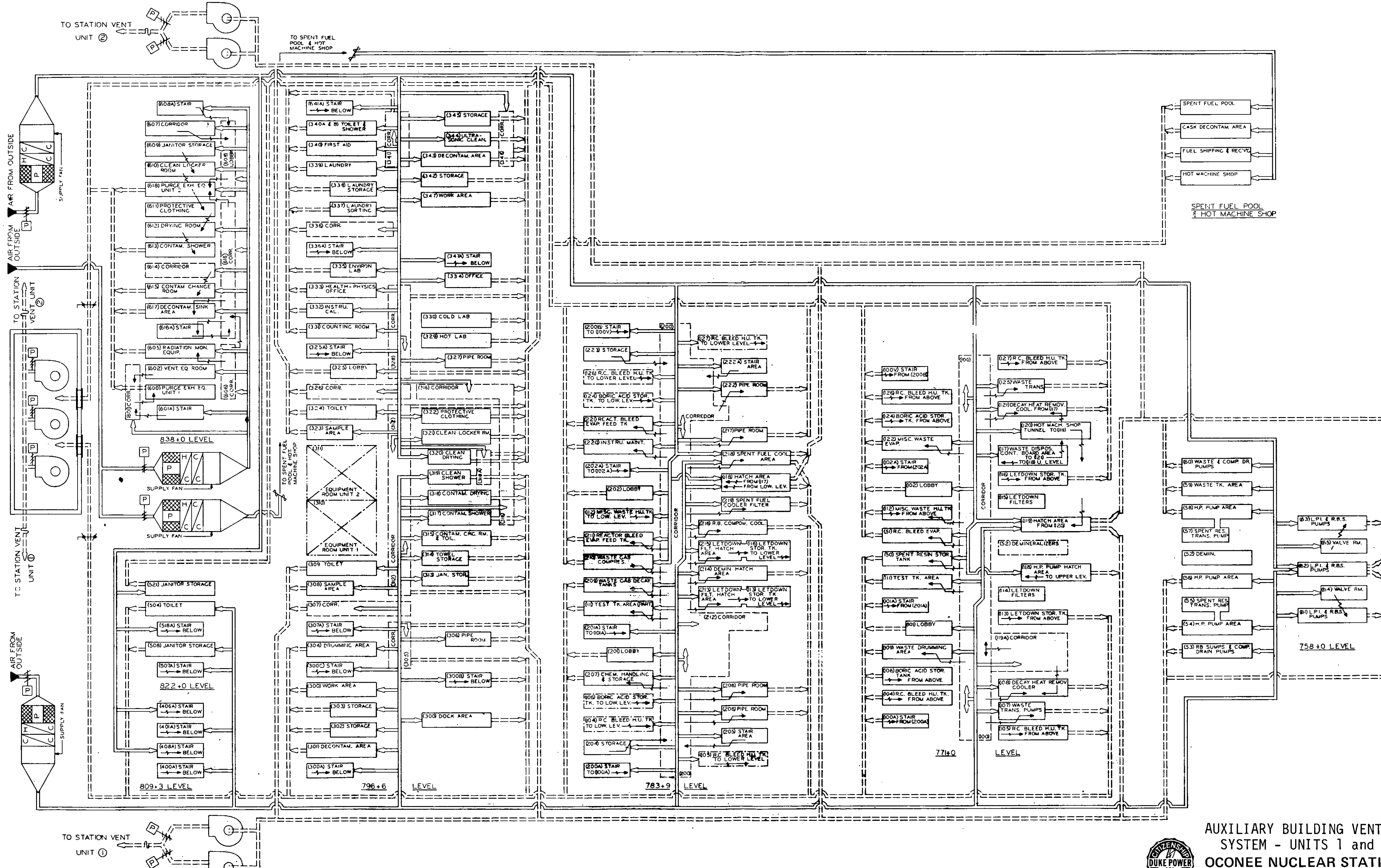
NOTES:

1 UNITS ONE AND TWO SHOWN UNIT THREE LATER.

FUEL HANDLING SYSTEM

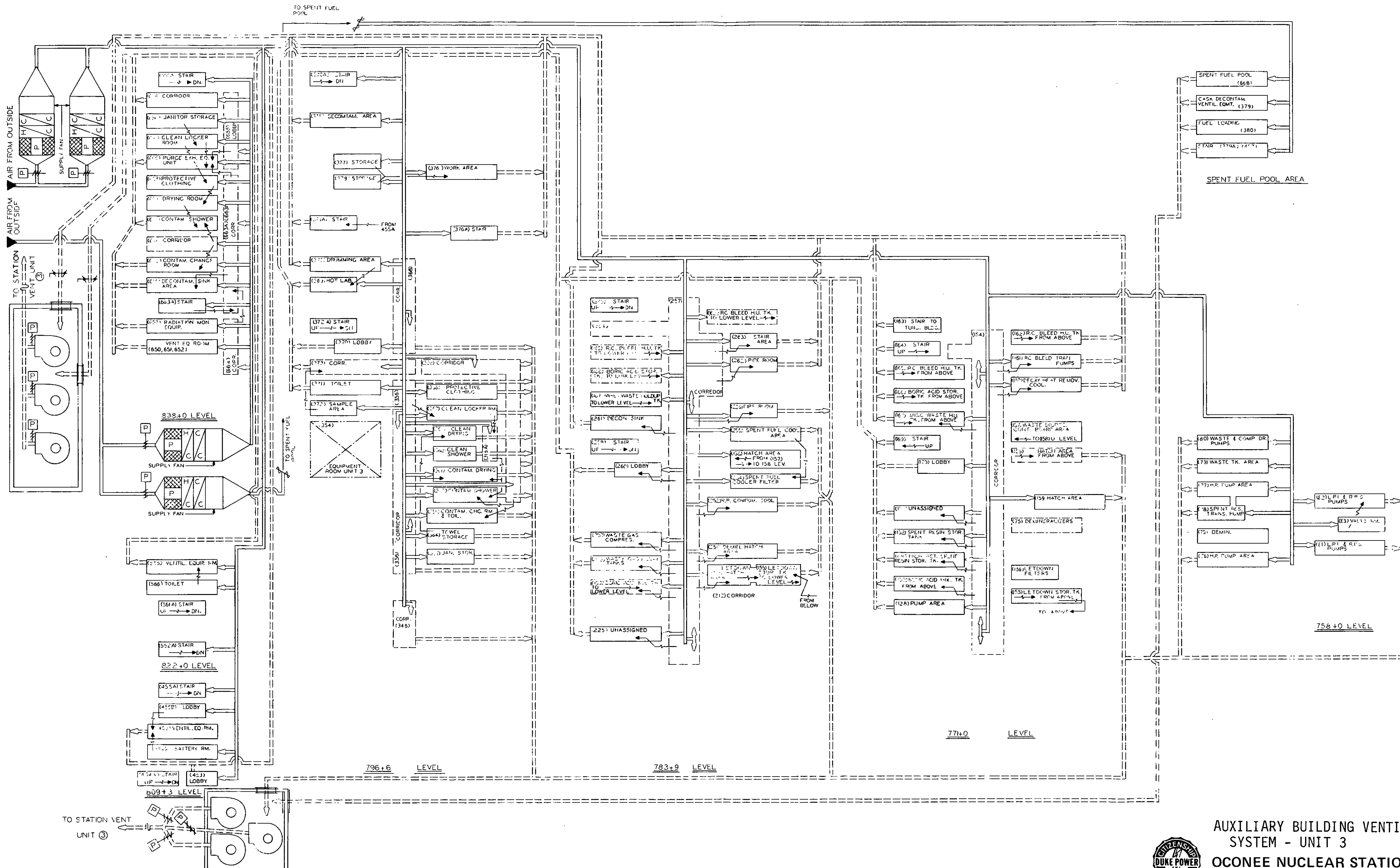
OCONEE NUCLEAR STATION

Figure 9 - 11



AUXILIARY BUILDING VENTILATION SYSTEM - UNITS 1 and 2
OCONEE NUCLEAR STATION
Figure 9-12

Rev. 32. 4/12/74

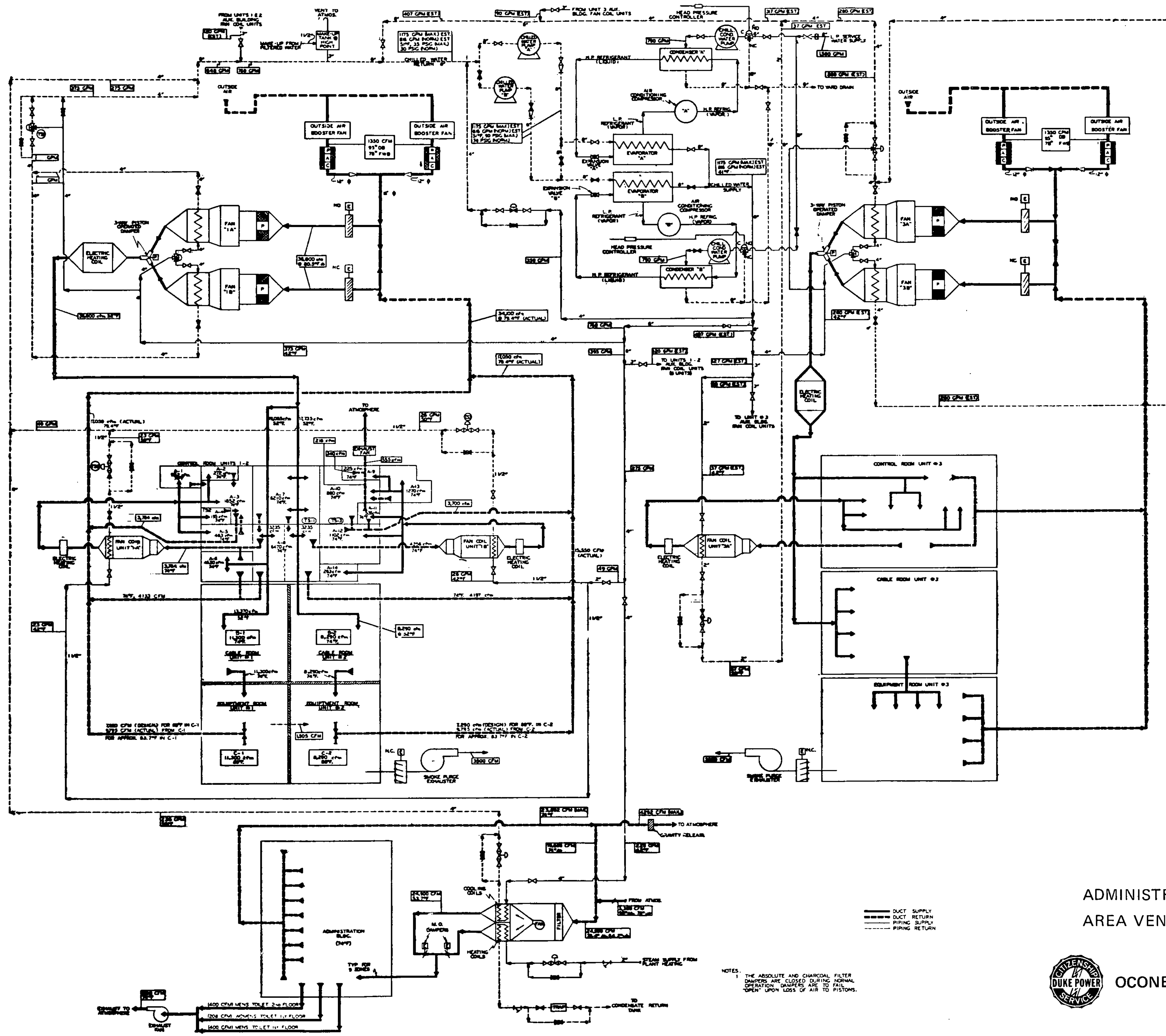


AUXILIARY BUILDING VENTILATION SYSTEM - UNIT 3

OCONEE NUCLEAR STATION

Figure 9-12A

Rev. 32. 4/12/74

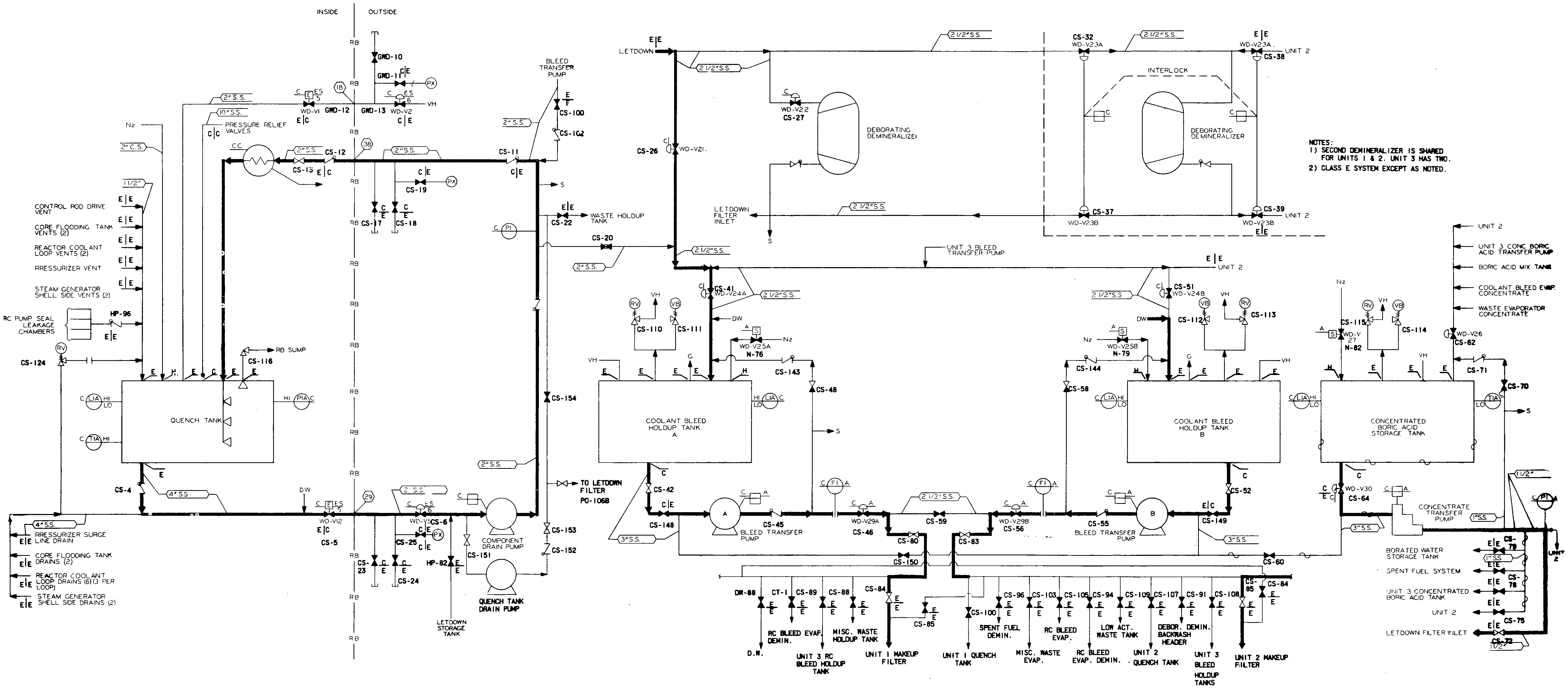


ADMINISTRATION BUILDING & CONTROL AREA VENTILATION SYSTEM

OCONEE NUCLEAR STATION

Figure 9 - 13

Rev 2 2/9/70

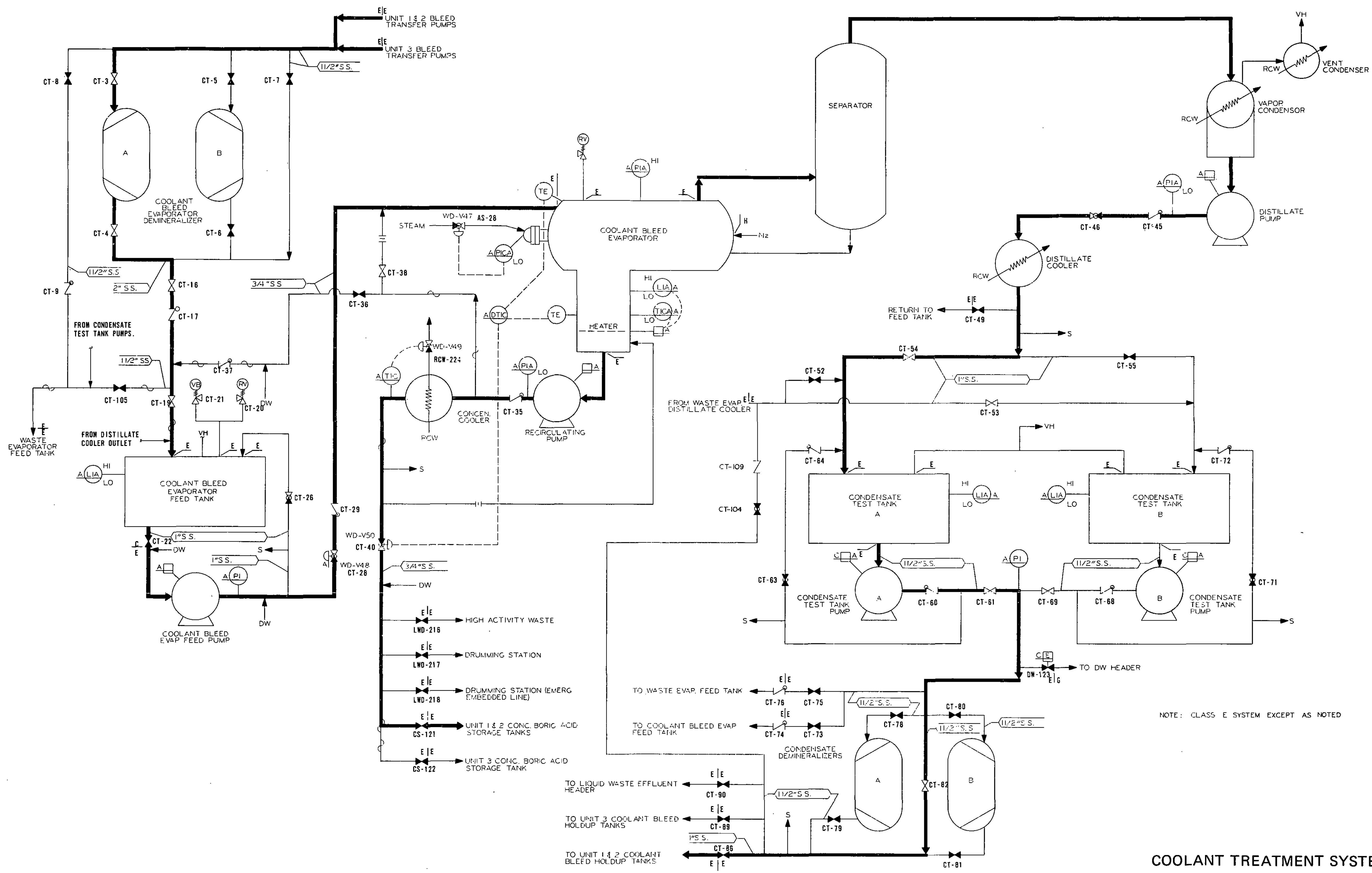


COOLANT STORAGE SYSTEM

OCONEE NUCLEAR STATION

Figure 9 - 14

Rev. 16 7/30/71



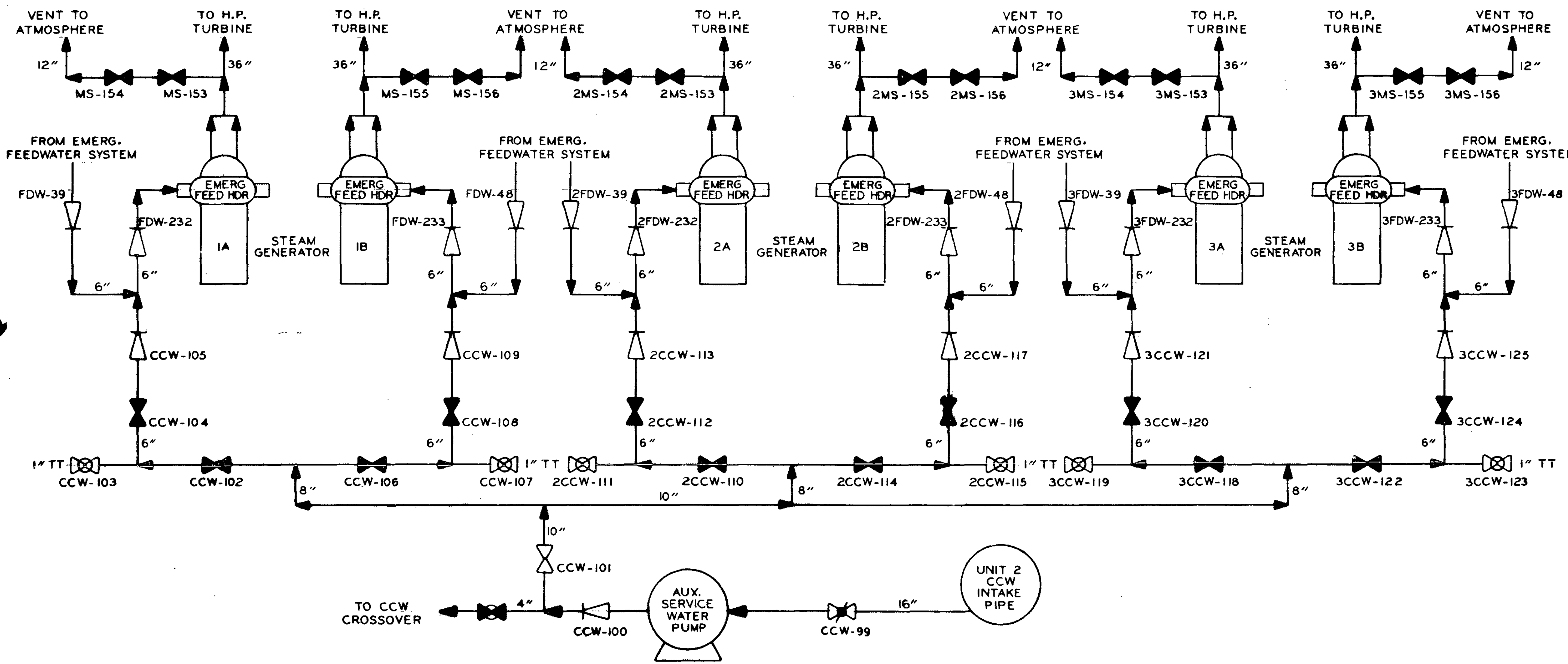
COOLANT TREATMENT SYSTEM

OCONEE NUCLEAR STATION

Figure 9 - 15

Rev. 16 7/30/71

Rev. 18 3/10/72



AUXILARY SERVICE WATER SYSTEM

OCONEE NUCLEAR STATION

Figure 9 - 16

Rev. 26 1/29/73

LIST OF EFFECTIVE PAGES
FSAR SECTION 10

<u>Steam and Power Conversion System</u>

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10-ii	Original
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10-2	Rev. 31
10-3	Rev. 16
10-4	Rev. 7
10-4a	Rev. 7
10-5	Original
10-6	Rev. 5
10-7	Rev. 5
Fig. 10-1	Rev. 16
Fig. 10-2	Rev. 18
Fig. 10-3	Rev. 16

Rev. 24. 11/15/72
Rev. 31. 2/15/74
Rev. 36. 7/21/75

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21. 10.3.1	TURBINE TRIPS, AUTOMATIC CORRECTION ACTIONS AND ALARMS	10-4a
10.3.2	TRANSIENT CONDITIONS	10-5
10.3.3	INTERACTIONS WITH REACTOR COOLANT SYSTEM	10-5
10.3.4	OVERPRESSURE PROTECTION	10-6
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LIST OF FIGURES

Figure No.	Title
10-1	Steam and Power Conversion System
10-2	Auxiliary Feedwater Supply
10-3	Emergency Feedwater Pump Turbine Steam Supply

10 STEAM AND POWER CONVERSION SYSTEM

10.1 DESIGN BASES

10.1.1 OPERATING AND PERFORMANCE REQUIREMENTS

The steam and power conversion system for each unit is designed to remove heat energy from the reactor coolant in the two steam generators and convert it to electrical energy. The closed feedwater cycle condenses the steam and the heated feedwater is returned to the steam generators. The system is designed to utilize the entire output from the nuclear steam supply system.

A maximum rate of turbine load change of 10 percent full load per minute is permitted by the turbine electro-hydraulic control (EHC) system without restriction if the minimum load involved in the change is 46 percent full load or greater. Below 46 percent full load, the maximum rate of change is still 10 percent full load per minute, but the total load change may be restricted by turbine metal temperature considerations.

The rate of change of reactor power is limited to values consistent with the characteristics of the reactor coolant system and its control systems. These limitations are imposed by the integrated control system on the steam and power
5. conversion system. See 7.2.3 and Table 7-4.

36.

10.1.2 ELECTRICAL SYSTEM CHARACTERISTICS

The station is designed for load following operation. The maximum rate of change of load is noted in 10.1.1.

10.1.3 SECONDARY FUNCTIONS

The steam and power conversion system provides steam for driving the two one-half capacity feedwater pumps for each unit. Steam is also used for the condenser air removal equipment and the emergency feedwater pump.

10.2 SYSTEM DESIGN AND OPERATION

10.2.1 SCHEMATIC FLOW DIAGRAM

The steam and power conversion system is shown in Figure 10-1. The closed cycle feedwater heaters are half-size units (two paralled strings), with the exception of "F" heater. There are three "F" heaters, one in each condenser neck. Deaeration is accomplished in the condenser. A bypass capability of 25 percent of full load main steam flow to the condenser is provided.

Two of the three one-half capacity hotwell pumps and two of the three one-half capacity condensate booster pumps are in normal use. Each of two feedwater pumps is more than one-half capacity.

There will be a total of approximately ten minutes condensate storage (142,000 gallons) at full load in the condenser hotwells. The upper surge tank (72,000 gallons) will provide approximately five minutes of additional condensate storage at full load.

The main steam lines and the normal and auxiliary feedwater lines are the only lines of the steam and power conversion system which penetrate the Reactor Building. These lines can be isolated by the turbine stop valves and the normal and auxiliary feedwater line valving. Each of the lines leaving the main steam lines before the turbine stop valves has motor operated valves to complete the isolation of a steam generator. These lines are:

(a) Steam bypass to condenser and steam supply for auxiliary steam header (See Figure 10-3 for line to auxiliary steam header)

(b) Supply to feed pump turbines and condenser air ejectors

(c) Supply to steam reheaters

(d) Supply to emergency feedwater pump turbine.

The arrangement of the valving and parallel piping shown schematically in Figure 10-1 prevents blowdown of both steam generators from a single leak in the system.

10.2.2 FEEDWATER SUPPLY

Feedwater supply to the steam generators following a reactor shutdown is assured by one of the following methods:

(a) Either of the two half-size feedwater pumps is capable of supplying both steam generators at full secondary system pressure.

(b) The hotwell and condensate booster pump combination has discharge shutoff head of approximately 700 psia. Three sets of half-size pumps are provided. If required, the turbine bypass system can be used to reduce secondary system pressure to the point where one of the hotwell and condensate booster pump combinations can supply feedwater to both steam generators.

(c) The 7-1/2 percent capacity, turbine driven, emergency feedwater pump will supply feedwater to both steam generators at full system pressure. The emergency feedwater pump takes its suction initially from the upper surge tank and, when desired, suction can be shifted to the hotwell. The auxiliary feedwater is pumped through separate supply lines to both steam generators, as shown in Figure 10-2. On loss of both feedwater pumps and/or loss of all four reactor coolant pumps, the integrated control system automatically closes valves FDW-36 and FDW-45 and opens valves
31. FDW-38 and FDW-47. On loss of both feedwater pumps, automatic startup of the emergency feedwater pump turbine occurs, and valves FDW-33 and FDW-42 are automatically closed. If all four

reactor coolant pumps fail, the emergency control signal for steam generator level is substituted for the normal control signal to permit higher water level and promote natural circulation.

The steam supply for the emergency feedwater pump turbine (Figure 10-3) will come from either of two sources (the main steam line or the auxiliary steam header) and will exhaust to the condenser or atmosphere. The sole-
16. noid operated valve which controls the steam shutoff valve MS-93 is de-energized on loss of both feedwater pumps, thus opening the steam shutoff valve. As the steam shutoff valve leaves the closed position, a limit switch starts the emergency feedwater pump turbine oil pump. This same
16. signal permits valves FDW-33 and FDW-42 to close when the integrated control system has opened valves FDW-38 and FDW-47.

(d) Alternate auxiliary feedwater supplies are available from the emergency feedwater pump of each of the other units.

10.2.3 CODES AND STANDARDS

The turbine-generator equipment conforms to the applicable ASA, ASME and IEEE standards.

The design, material, and details of construction of the feedwater heaters are in accordance with both the ASME Code, Section VIII, Unfired Pressure Vessels, and the Standards of Feedwater Heater Manufacturers Association, Inc.

The condenser equipment is in accordance with the Standards for Steam Surface Condensers as published by the Heat Exchange Institute.

The upper surge tanks, condensate storage tank, and heater drain flash tanks are in accordance with the ASME Code, Section VIII, Unfired Pressure Vessels.

The following portions of the system are designed to withstand seismic loading (criteria for seismic loading defined in Appendix 1C):

(a) Main steam lines from steam generator through the turbine stop valves

(b) Main steam line relief valves

(c) The steam supply from the main steam lines to the emergency feed pump tur-
16. bine including valve AS-38 and that portion of the auxiliary steam supply downstream from the valve

(d) Through the first valve of all other lines leaving the main steam lines

(e) Both supply lines from the upper surge tank to the emergency feedwater pump, including piping through the first valve of any connections to these lines

(f) Both discharge lines from the emergency feedwater pump to the steam generators, including piping through the first valve of any connections to these lines.

10.2.4 SHIELDING

No radiation shielding is required for the components of the steam and power conversion system. Continuous access to the components of this system located outside the Reactor Building is possible during normal conditions.

10.2.5 CORROSION PROTECTION

Hydrazine will be added to the feedwater for oxygen control, and ammonia will be used to maintain the pH at the optimum value for the materials of construction for the system. No other additives are contemplated.

10.2.6 IMPURITIES CONTROL

The polishing demineralizer utilizes the Powdex process, developed by Graver Water Conditioning Company as a unique, high quality water purification system. The Powdex units will function as a combination demineralizer and high purity filter, treating 100 percent of the feedwater flow to the steam generator under conditions of startup and reduced load, and treating approximately 70 percent of the total flow during normal full-load operation.

The Powdex process uses extremely fine particle-size (60-400 mesh) ion exchange resins which are applied to the external surface of specially designed filter elements. The rapid ion exchange rates of these fine resins allows the use of a thin coating (1/16 inch to 1/2 inch) on the elements and permits a greater utilization of the ultimate capacities of the resins than is the case for bead type resins.

7. The Powdex resins are not chemically regenerated for repeated use but are replaced with fresh resins upon exhaustion. This continued resin replacement allows complete flexibility in the selection of the most advantageous type of resin or combination of resins for the removal of specific impurities.

The resins will be selected for the effective removal of dissolved metallic cations and also anions such as halides, silicates, and sulfates. In addition, the resin will also remove by filtration the suspended and colloidal trace impurities such as corrosion products.

Exhaustion of each batch of resins will be monitored and will be indicated by an increase in pressure drop or by a decrease in treated water quality. Exhausted resins will be backwashed from the units and pumped to a disposal facility.

10.2.7 RADIOACTIVITY

The once-through nature of this recirculating steam condensate cycle is utilized in the removal of contaminants resulting from steam generator leaks, since it allows the flow through the steam generator to be subjected to purification. Radioactive contaminants will be removed by the Powdex polishing demineralizers as described for the control of impurities (10.2.6). Provision is made for transferring the backwashed resins, when they contain radioactive material, to the radioactive waste disposal system.

10.3 SYSTEM ANALYSIS

10.3.1 TURBINE TRIPS, AUTOMATIC CORRECTIVE ACTIONS AND ALARMS

Trips, automatic corrective actions, and alarms will be initiated by deviations of system variables within the steam and power conversion system. In the case of automatic corrective action in the steam and power conversion system, appropriate automatic corrective action will be taken to protect the reactor coolant system. The more significant malfunctions or faults which cause trips, automatic actions or alarms in the steam and power conversion system are:

(a) Turbine Trips

1. Loss of d-c supply to trip circuits
2. Loss of condenser vacuum
3. Thrust bearing wear.
4. Loss of generator stator coolant (if runback fails)
5. Loss of both feedwater pumps
6. Turbine overspeed
7. Reactor trip
8. Bearing oil low pressure
9. EHC hydraulic fluid low pressure.

(b) Automatic Actions

1. Feedwater flow lagging feedwater demand results in a reduction in power demand.
2. Low feedwater temperature results in a reduction in power demand.
3. High level in steam generator results in a reduction in feedwater flow.
4. Low level in steam generator results in an increase in feedwater flow.
5. Loss of pressure at feedwater pump discharge header results in starting the emergency feedwater pump turbine.
6. Low pressure at feedwater pump suction trips feedwater pump turbines.

(c) Principal Alarms

1. Low pressure at feedwater pump suction
2. Low vacuum in condenser
3. Low water level in condenser hotwell
4. High water level in condenser hotwell
5. High water level in steam generator
6. Low water level in steam generator
7. High pressure in steam generator
8. Low pressure in steam generator
9. Low feedwater temperature
10. Electrical malfunctions in the EHC.

10.3.2 TRANSIENT CONDITIONS

The analysis of the effects of loss of full load on the reactor coolant system is discussed in 14.1.2.8. Analysis of the effects of partial loss of load on the reactor coolant system is discussed in 7.2.3.3.4.

10.3.3 INTERACTIONS WITH REACTOR COOLANT SYSTEM

The effects of inadvertent steam relief or steam bypass are covered by the analysis of the steam line failure given in 14.1.2.9. The effects of an inadvertent rapid throttle valve closure are covered by the loss of full load discussion in 14.1.2.8.

Following a turbine trip, the control system will reduce reactor power output immediately. The safety valves will relieve excess steam until the output is reduced to the point at which the steam bypass to the condenser can handle all the steam generated.

In the event of failure of a single feedwater pump, there will be an automatic runback of the power demand. The one feedwater pump remaining in service will carry approximately 60 percent of full load feedwater flow. If both feedwater pumps fail, the turbine will be tripped, and the emergency feedwater pump started. If reactor coolant system conditions reach trip limits, the reactor will trip.

On failure of a hotwell pump or a condensate booster pump, the spare hotwell pump or the spare condensate booster pump will be automatically started.

10.3.4 OVERPRESSURE PROTECTION

Pressure relief is required at the system design pressure of 1050 psig, and the first safety valve bank will be set to relieve at this pressure. The design pressure is based on the operating pressure of 925 psia plus a 10 percent allowance for transients and a 4 percent allowance for blowdown. Additional safety valve banks will be set at pressures up to 1104 psig, as allowed by the ASME Code. Pressure relief is provided by eight safety valves on each main steam line, and the valve relief pressures are:

Number of Valves	Relief Pressure (psig)
1	1050
1	1065
1	1080
1	1090
2	1100
2	1104

The relief valve capacity (13,105,000 lb hr total for 16 valves) is such that the energy generated at the reactor high power level trip setting can be dissipated through this system.

10.4 TESTS AND INSPECTIONS

Steam from the steam generators is admitted to the turbine through four cast 24 inch main steam stop valves, arranged in parallel and located in the main steam lines upstream from the turbine control valves (see Figure 10-1). In the event of a steam line rupture accident, the stop valves serve to isolate the unaffected steam generator.

The main steam stop valve is designed for tight seating throughout its life. The valve stem extends through a guide bushing which centers the disc on the stem with some degree of freedom, permitting self alignment of the disc on its seat. The valve seat and disc have spherical seating surfaces so that perfect contact is made even if they are not in precise alignment. The use of stem sealing permits relatively large stem to bushing clearance, minimizing the pos-
5. sibility of stem sticking. The seating surfaces of the valve and the stem seal are hardened inlay contact areas which resist erosion and mechanical damage and assure tightness. A coarse-mesh internal screen strainer with removable fine-mesh startup strainer is provided for each stop valve.

The main steam stop valves are fail-safe, requiring hydraulic pressure to open and closure is spring-assisted. Each stop valve has two positions: fully open and fully closed. Each stop valve will be tested periodically (while the turbine is in operation) and any tendency of the valve to remain open in opposition to a control signal will be detected. A stop valve will be disassembled, inspected, and required corrective action taken when a valve test warrants such action. Stop valves are also disassembled and inspected during turbine inspections.

The main steam stop valves are designed and tested to assure proper functioning. In the event of a steam line rupture accident, the two stop valves serving the unaffected steam generator will close in the presence of steam flow in the normal direction, thus precluding the possibility of reverse flow through the other two stop valves.

The motor operated valve on each of the lines connected to the main steam lines can be tested for operability when the unit is shutdown. These valves along with the main steam stop valves prevent uncontrolled blowdown of both steam generators in the unlikely event of a main steam line break. Their ability to close will be verified at periodic intervals.

Proper operation of the emergency feedwater pump and turbine, the steam shutoff valve (Figure 10-3), and the valves in the auxiliary feedwater supply to the steam generators (10.2.2) can be demonstrated when the unit is shutdown. The emergency feedwater pump and turbine, and the steam shutoff valve can be tested anytime by utilizing the recirculation test line (Figure 10-2). Proper functioning of the auxiliary feedwater supply will be verified at periodic intervals.

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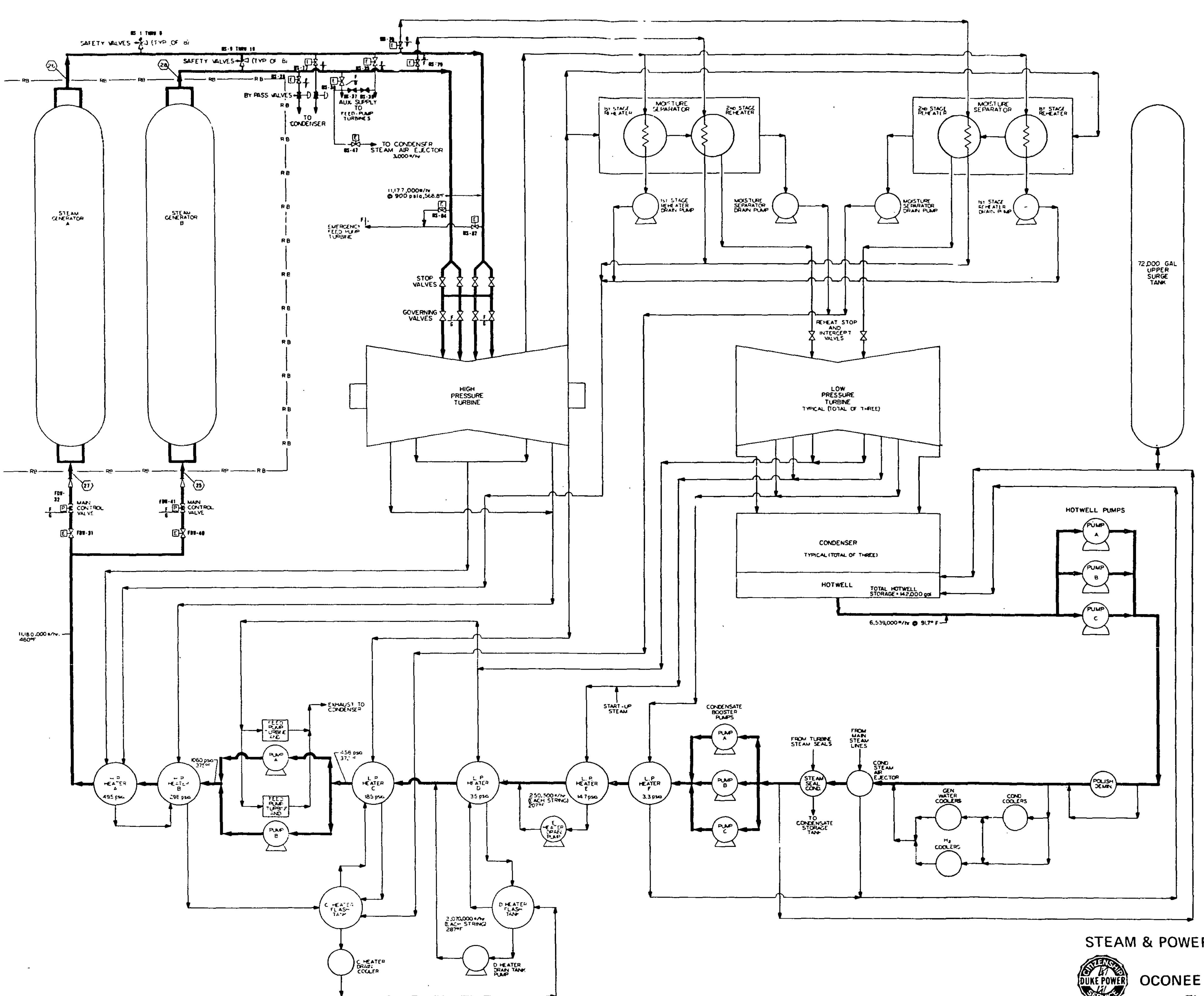

NOTES:

1. THIS SCHEMATIC OMITS STANDARD VALVES, SMALL PIPING, AND MINOR DETAILS.
2. FLOWS, TEMPERATURES AND PRESSURES SHOWN FOR APPROXIMATELY 2584 MWT OUTPUT FROM NUCLEAR STEAM SUPPLY SYSTEM, 1" Hg, 0% MAKEUP.
3. ONE STRING OF FEEDWATER HEATERS SHOWN - TOTAL OF TWO STRINGS.
4. SYSTEM SHOWN IS TYPICAL OF EACH GENERATING UNIT
5. ALL EQUIPMENT AND VALVES SHOWN ARE CONTROLLED FROM CONTROL ROOM.
6. CLASS F SYSTEM EXCEPT AS NOTED.

STEAM & POWER CONVERSION SYSTEM

OCONEE NUCLEAR STATION

Figure 10 - 1

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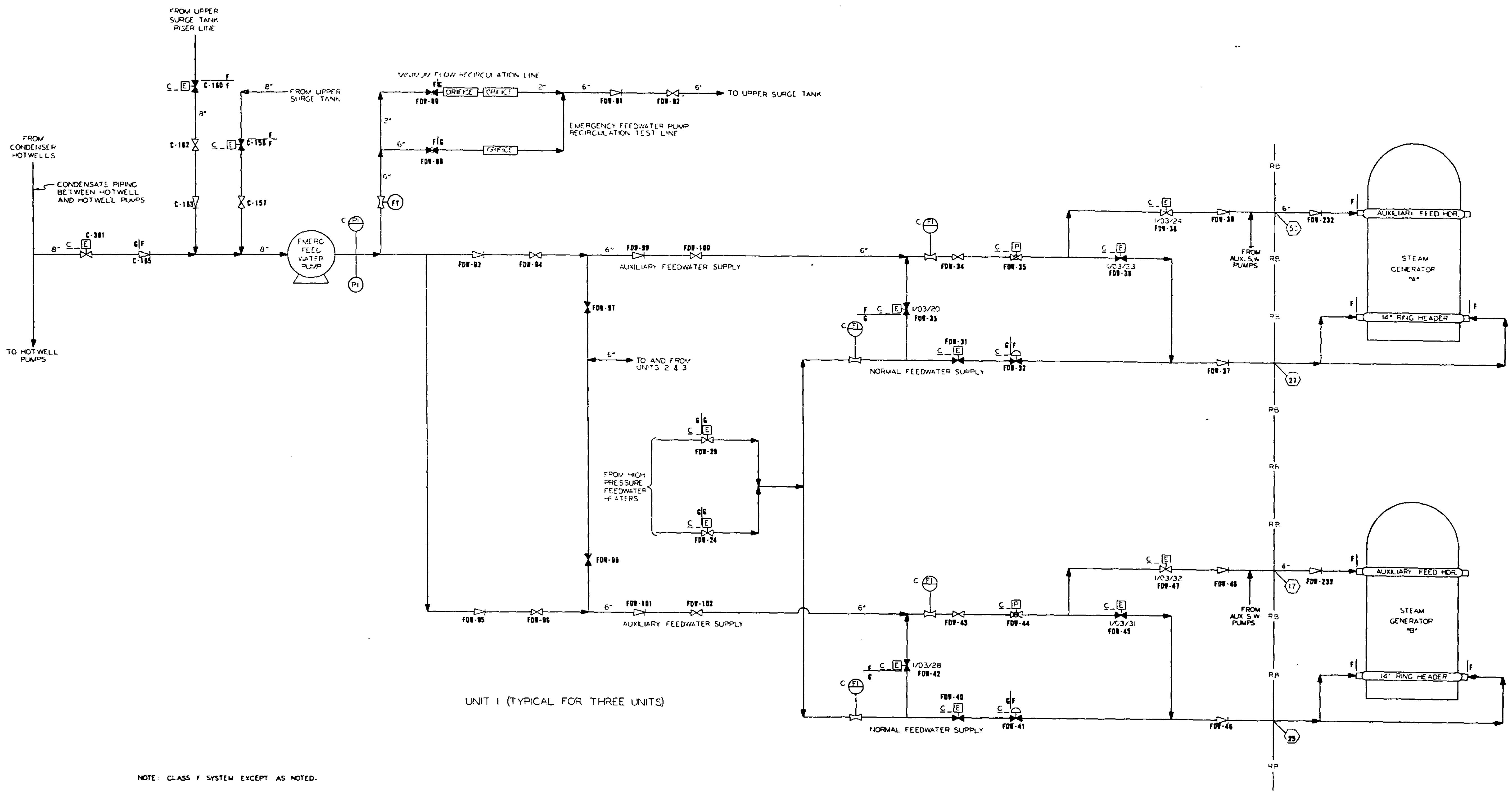

AUXILIARY FEEDWATER SYSTEM

OCONEE NUCLEAR STATION

Figure 10 - 2
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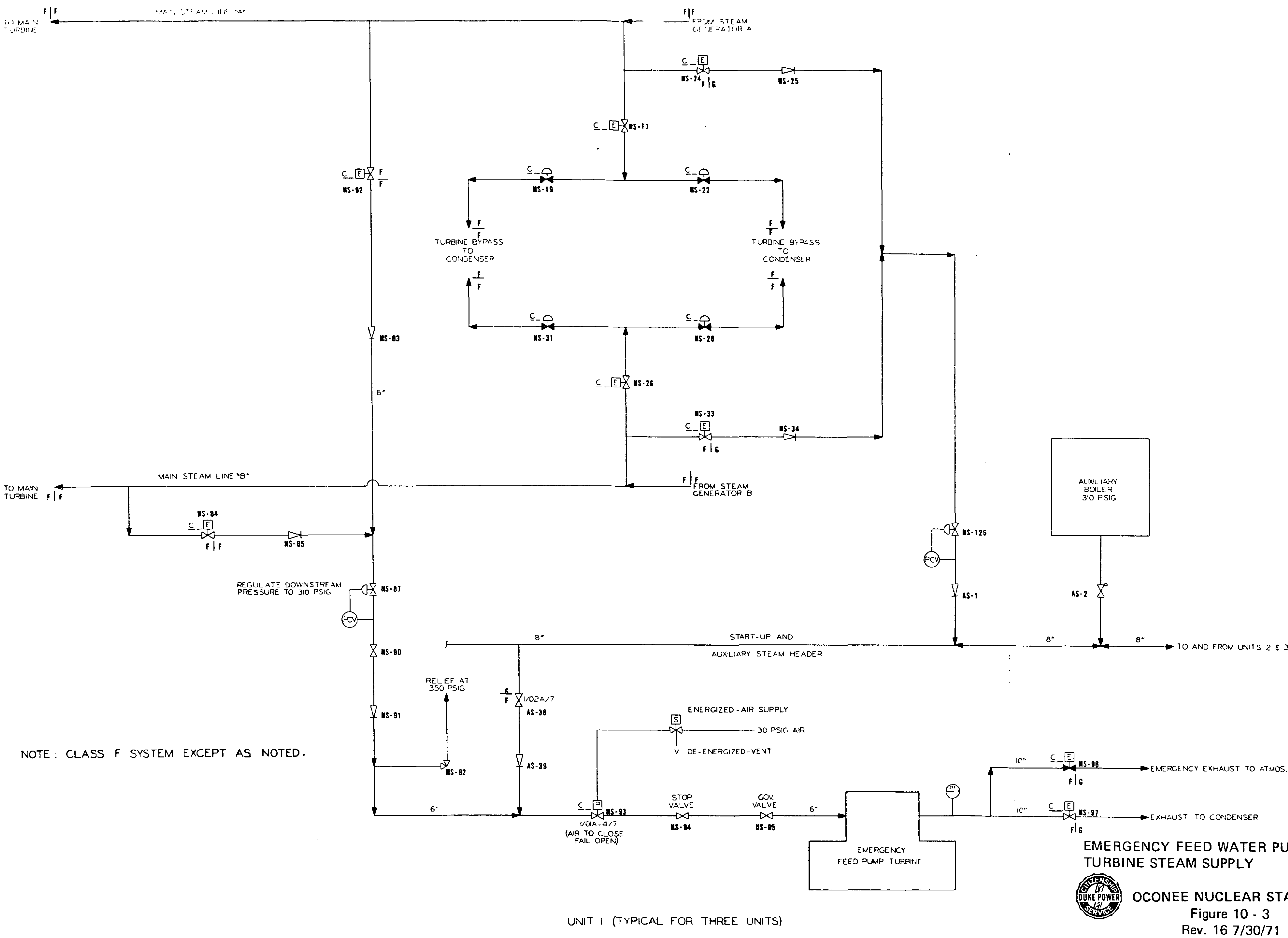

EMERGENCY FEED WATER PUMP
TURBINE STEAM SUPPLY

OCONEE NUCLEAR STATION
Figure 10 - 3
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11 RADIOACTIVE WASTES AND RADIATION PROTECTION

11.1 RADIOACTIVE WASTES

11.1.1 DESIGN BASES

11.1.1.1 Performance Objective

The waste disposal systems are designed to provide for controlled handling and disposal of radioactive liquid, gaseous, and solid wastes. The principal design criterion (1A.70) assures that station personnel and the general public are protected against excessive exposure to radioactive material on the basis of: (a) 10CFR20 requirements for normal operations and anticipated transient conditions and (b) 10CFR100 guidelines for potential reactor accidents.

11.1.1.2 Radioactive Waste Quantities

A conservative model of the operation of the reactor coolant system, coolant treatment system, and the liquid and gaseous waste disposal systems, summarized below, was employed as the design basis. Using this model, the liquid and gaseous waste volumes, activity levels, and activity inventories released through normal station operation were computed.

(1) Assumptions regarding operation and cleanup of the reactor coolant system are given in 11.1.1.3.1.

(2) All three units have identical operations.

(3) Initial unit startup from refueling boron concentration of approximately 2270 ppm.

(4) Startup from cold shutdown condition at equal intervals during the 310 full power day cycle, i.e., at 77.5, 155, and 232.5 days. Shutdown boron concentration is sufficient to maintain 1 percent subcritical margin.

(5) Use of evaporator for treatment of reactor coolant bleed until the boron concentration reaches approximately 180 ppm, at which time the deborating demineralizers are placed into service.

(6) Operation of coolant bleed evaporator at full capacity for 8 hours per day.

The estimated volumes and maximum rates of accumulation of the various forms of radioactive wastes generated in the station are listed in Tables 11-1a and 11-1b.

11.1.1.3 Radioactive Waste Activities

11.1.1.3.1 Liquid Wastes

Design activity accumulation in the reactor coolant system and associated waste handling equipment has been determined on the basis of fission product leakage through clad defects in 1 percent of the fuel. The activity levels were computed assuming full core power operation for two cycles with no defective fuel followed by operation over the third cycle with 1 percent defective fuel.

Table 11-1a

Estimated Radioactive Waste Quantities From Three Units

Waste Source	Quantity/Year ft^3	Assumptions and Comments
Reactor Coolant System[a]		
Startup Expansion	39,800	Four cold startups per unit
Startup Dilution	49,000	One startup from cold condition at beginning of cycle, 77.5, 155 and 232.5 full power days, respectively, per unit
Lifetime Shim Bleed[b]	43,800	Dilution 1070 to 180 ppm boron in each unit
System Drain	18,300	Drain of each unit to level of outlet nozzles during refueling
	150,900	
Liquid Waste		
Sampling and Laboratory Drains	1,250	36 samples per week at 5 gal per sample
Demineralizer Sluice	700	2 ft^3/ft^3 resin
Deborating Demineralizer Regeneration and Rinse	7,600	20 ft^3/ft^3 resin regeneration
Miscellaneous System Leakage	11,700	10 gph leakage
Laundry	14,600	300 gpd
Showers	29,200	20 showers per day at 30 gal per shower
	65,050	
Gaseous Waste		
Off-Gas from Reactor Coolant System[b]	6,040	Degas at 40 cc H_2 per liter concentration
Off-Gas from Liquid Sampling	50	Degas at 40 cc H_2 per liter concentration
Letdown Storage Tank	2,700	Vent once per year
Pressurizer	180	Vent once per year
	8,970	
Solid Waste		
Demineralizer Resin	350	Resin replacement twice per year
Evaporator Bottoms	2,500	Evaporation to 25 percent solids for reactor coolant system liquid effluent

(a) Treated as waste for purposes of evaluation

(b) Wastes processed through holdup tanks

Table 11-1b

Estimated Maximum Rate of Accumulation of Radioactive Wastes Per Operation

Waste Source	Maximum Rate of Accumulation	Assumptions and Comments
Reactor Coolant System[(a)]		
Startup Expansion & Dilution	9900 ft^3/22 hours	Cold startup immediately prior to placing deborating demineralizers into service
Lifetime Shim Bleed[(b)]	1200 ft^3/10 days	Last 10 days of processing bleed prior to placing deborating demineralizers into service
System Drain	6100 ft^3/refueling	
Liquid Waste		
Demineralizer Sluice	100 ft^3/resin change	One change of purification demineralizer resin
Deborating Demineralizer Regeneration and Rinse	1260 ft^3/regeneration	One regeneration of deborating demineralizer
Gaseous Waste		
Off-Gas from Reactor Coolant System[(b)]	400 ft^3/22 hours	Cold startup immediately prior to placing deborating demineralizers into service
Letdown Storage Tank	900 ft^3/purge	
Pressurizer	60 ft^3/purge	
Solid Waste		
Demineralizer Resin	50 ft^3/resin change	One change of purification demineralizer resin

(a) Treated as waste for purpose of evaluation

(b) Wastes processed through holdup tanks

Continuous reactor coolant purification at a rate of one reactor system volume per day was used with a zero removal efficiency for Kr, Xe, Mo, and Y; a 99 percent removal efficiency for I and Sr; and a 90 percent removal efficiency for all other elements. Te was assumed to plate out on the pipe walls, but its daughter I was put back into the reactor coolant. The reactor coolant bleed for boron removal is processed through demineralizers and/or the evaporator. Removal efficiencies of essentially 100 percent for all isotopes were assumed. Activity levels are relatively insensitive to small changes in removal efficiencies, e.g. use of 90 percent instead of 100 percent would result in less than a 10 percent increase in the coolant activity.

The quantity of fission products released to the reactor coolant during steady state operation is based on the use of "escape rate coefficients" (sec^{-1}) as determined from experiments involving purposely defected fuel elements (References 1,2,3,4). Values of the escape rate coefficients used in the calculations are shown in Table 11-2.

Calculations of the activity released from the fuel were performed with a digital computer program which solves the differential equations for a five-member radioactive chain for buildup in the fuel, release to the coolant, removal from the coolant by purification and leakage, and collection on a resin or in a hold-up tank. Further description of the analytical method can be found in answer to question 6.7, Supplement 2, Oconee PSAR. The resulting design activity levels in the reactor coolant are shown in Table 11-3 as a function of time during full power operation of the third cycle. The activity levels tabulated for 200 full power days were used in all activity inventory calculations.

The liquid waste generated by leakage, sampling, and demineralizer sluice or rinse is assumed to have an activity concentration equal to the concentration in the reactor coolant. Reactor coolant bleed is taken from the downstream side of the purification demineralizer and is assumed to have the same specific activity as the reactor coolant reduced by the decontamination factor of the purification demineralizer.

11.1.1.3.2 Gaseous Waste

Gaseous activity is generated by the evolution of radioactive gases from liquids stored in tanks throughout the station. Therefore, the activity of the gases is dependent upon the liquid activity. The assumptions for liquid activity are described above. The resulting gaseous activities are described in 11.1.2.5, Design Evaluation.

11.1.1.3.3 Solid Wastes

Solid radioactive wastes consist of demineralizer resins, evaporator concentrate, and miscellaneous solids. This activity is not released to the environment and influences only the shielding required to meet criteria stated in 11.2.1.

Table 11-2

Escape Rate Coefficients for Fission Product Release

Element	Escape Rate Coefficient, sec^{-1}
Xe	1.0×10^{-7}
Kr	1.0×10^{-7}
I	2.0×10^{-8}
Br	2.0×10^{-8}
Cs	2.0×10^{-8}
Rb	2.0×10^{-8}
Mo	4.0×10^{-9}
Te	4.0×10^{-9}
Sr	2.0×10^{-10}
Ba	2.0×10^{-10}
Zr	1.0×10^{-11}
Ce, and other rare earths	1.0×10^{-11}

Table 11-3

Reactor Coolant Activity

(Reactor Coolant Activity for the Third Cycle, Based on 1 Percent Defective Fuel)

μCi/ml at Operating Conditions

Time, Full Power Days Isotope	100	150	200	260	310
Kr 85 m	1.5	1.5	1.5	1.5	1.5
Kr 85	8.7	9.7	9.1	6.3	1.3
Kr 87	.85	.85	.85	.85	.84
Kr 88	2.7	2.7	2.7	2.7	2.7
Rb 88	2.7	2.7	2.7	2.7	2.7
Sr 89	.041	.041	.041	.041	.038
Sr 90	.0027	.0028	.0030	.0032	.0031
Sr 91	.046	.046	.046	.046	.045
Sr 92	.017	.017	.017	.017	.017
Xe 131m	2.3	2.2	2.2	2.1	1.2
Xe 133m	2.8	2.8	2.8	2.8	2.3
Xe 133	248	246	242	234	166
Xe 135m	.94	.94	.94	.94	.93
Xe 135	6.6	6.6	6.6	6.6	6.3
Xe 138	.51	.51	.51	.51	.51
I 131	3.2	3.2	3.2	3.2	3.0
I 132	4.8	4.8	4.7	4.6	3.8
I 133	3.8	3.8	3.8	3.8	3.7
I 134	.50	.50	.50	.50	.50
I 135	1.9	1.9	1.9	1.9	1.9
Cs 136 13d.	.045	.045	.045	.045	.042
Cs 137 30y.	.29	.29	.29	.28	.26
Cs 138 32m.	.72	.72	.72	.72	.72
Mo 99	5.4	5.4	5.3	5.2	4.2
Ba 139	.082	.082	.082	.082	.082
Ba 140	.072	.072	.072	.072	.067
La 140	.025	.025	.025	.025	.022
Y 90	.24	.37	.51	.69	.84
Y 91	.18	.20	.20	.17	.076
Ce 144	.0028	.0029	.0029	.0030	.0028
Bleed Rate in Reactor Coolant volume/sec	$6.7x10^{-8}$	$5.7x10^{-8}$	$7.8x10^{-8}$	$1.3x10^{-7}$	$5.1x10^{-7}$

11.1.1.4 Disposal Methods and Limits

Liquid wastes from the station are disposed of, under continuous radiation monitoring and control, in any of the following three ways depending on the concentration of radioactivity and quantities involved:

(a) Collected, sampled and analyzed, and discharged directly to the tailrace of the Keowee Hydroelectric Plant.

(b) Collected, sampled and analyzed, held up for decay, resampled and analyzed, and discharged directly to the tailrace of the Keowee Hydroelectric Plant.

(c) Collected, sampled and analyzed, concentrated by evaporation, the evaporator bottoms packaged and shipped off-site to an AEC licensed burial ground.

Liquid waste effluent is diluted, as necessary, in the hydroelectric plant tailrace to permissible concentration limits in accordance with the requirements of 10CFR20.

Gaseous wastes are disposed of, at a permissible rate, under continuous radiation monitoring and control, by any of the following methods depending on the concentration of radioactivity, quantities, and source of the material involved:

(a) Release of Auxiliary Building ventilation air and Reactor Building purges to the unit vents.

(b) Release of Reactor Building purges through high efficiency particulate and charcoal iodine filters to the unit vents.

(c) Release of waste gas directly or through high efficiency particulate and charcoal iodine filters to the unit vents.

(d) Diversion to waste gas tanks with controlled release after sampling and analysis through the waste gas system high efficiency particulate and charcoal iodine filters to the unit vents.

Gaseous wastes are released from the station at a controlled rate so that permissible concentration limits for Unrestricted Areas will not be exceeded at the Exclusion Area boundary, when averaged over a year in accordance with the requirements of 10CFR20. The concentrations at the boundary are determined after applying appropriate dilution factors derived from on-side meteorological studies (2.3 Meteorology). The Exclusion Area is considered as a Restricted
8. Area for Gaseous Waste disposal purposes only; because of precautions that will be taken, the general public, construction forces and station office staff, within the Exclusion Area, will be subject only to Unrestricted Area dose limits.

Waste releases from the three units are integrated and controlled by process radiation monitors, interlocks, and by the operator so as not to exceed the appropriate station release limits. Where effluents can be released from more than one location, administrative controls are also provided to insure that station limits are not exceeded. In addition, each control room has a continuous indication of the releases from all three units.

Solid wastes, which are principally expected to be low level contaminated trash, evaporator bottoms, and spent ion exchange resins will be packed in Department of Transportation (DOT) specification containers and/or AEC approved containers for a "large quantity" of radioactive material and shipped in accordance with DOT regulations to an AEC licensed burial ground.

11.1.2 SYSTEM DESIGN AND EVALUATION

11.1.2.1 Solid Waste Disposal System

The solid waste disposal system provides the capability to package solid wastes for shipment to an offsite disposal facility. The waste drumming room is provided to handle evaporator concentrate and spent resin (Figure 11-1). Within the waste drumming room provision is made to utilize a range of containers from the size of a 55 gallon drum, shielded or unshielded, up to a 15 ton cask. These containers utilize an internal filter for retention of essentially all undissolved solids. Evaporator concentrate and spent resin can also be discharged directly into a truck-mounted cask outside the building. Exposure to personnel is minimized through the use of quick-disconnect feed, vent, and drain connections to the containers and operation of the valves in the waste drumming room from a control panel outside the room.

Evaporator concentrate is pumped directly from the evaporator to the shipping container. Spent resin is accumulated in the spent resin storage tanks which have a capacity exceeding that amount of spent resin normally expected to be generated in two years of operation. Sufficient flexibility is provided in the design of the system that evaporator concentrate and spent resin can be transferred to an appropriately sized and shielded container inside or outside the building, as dictated by the volume and activity of the waste material.

Soft solid wastes such as contaminated clothing, rags, paper, gloves, and shoe coverings will be compressed in 55 gallon drums by a hydraulic compactor.

Loaded containers will be monitored for surface radiation and stored in the low level drumming area.

11.1.2.2 Liquid Waste Disposal System

11.1.2.2.1 General Description

Liquid wastes are accumulated in storage tanks according to the expected activity concentrations. Liquids with the lowest expected activity concentrations are directed into drain header D-1 (Figure 11-2) which empties into the low activity waste tank. Intermediate activity concentration liquids are directed into drain header D-2 which empties into the high activity waste tank. The liquid waste with the highest expected activity concentrations is directed into drain header D-3 which empties into the waste holdup tank. As liquid wastes are accumulated in any of these tanks, the contents are mixed by a mixer and/or by recirculation with the transfer pumps. After thorough mixing of the contents, a sample is drawn for laboratory analysis. Based on this laboratory analysis, the water is:

(1) Discharged to the Keowee Hydro tailrace.

(2) Directed into a higher or lower activity waste tank as appropriate.

(3) Stored in the waste holdup tank for radioactive decay.

(4) Processed by evaporation to concentrate impurities for ultimate disposal and to provide return of purified water to the coolant storage system or release. The evaporator concentrate, if desired, can be pumped to the concentrated boric acid storage tank. If recovery of the concentrate is not desired, it will be pumped to the drumming station for drumming (11.1.2.1) and off-site disposal.

All piping and equipment in contact with reactor coolant are constructed of corrosion-resistant material. This equipment is arranged and located to permit detection and collection of system losses and to prevent escape of any unmonitored radioactive liquid to the environment. A flow diagram of this system with the necessary instrumentation and controls for operation is shown in Figure 11-2. Component data are shown in Table 11-4.

7. The liquid waste discharge header to the Keowee Hydro tailrace is shown in Figure 11-2A.

All waste tanks are vented to the gaseous waste vent header to provide for filling and emptying without overpressurization or creating a vacuum. In addition, each waste tank is equipped with a relief valve and vacuum breaker. Nitrogen is supplied to each waste collection tank for purging to the Gaseous Waste Disposal System as needed.

Demineralized water connections are provided at appropriate locations in the system for flushing of piping and components.

Spent resins from various demineralizers containing radioactive materials are sluiced into the spent resin storage tanks. Excess water from the spent resin storage tanks is utilized in flushing out the demineralizers. As a spent resin storage tank accumulates shipable quantities of resins, the resins are pumped to the waste drumming room or outside area for processing as described in 11.1.2.1.

11.1.2.2.2 Operation

Instrumentation and controls necessary for the operation of the liquid waste disposal system for Units 1 and 2, and for Unit 3 are located on an auxiliary control board near the equipment. The instrumentation and controls for the low activity waste tanks are located on the auxiliary control board and are duplicated in the respective control rooms. However, the only controls for the release of liquid waste are in the control rooms. A selector switch mounted in the control room for Units 1 and 2 prevents operation of the liquid effluent
16. discharge valve LWD-132 from both control rooms simultaneously.

All liquid wastes are released from the low activity waste tanks or the condensate test tanks (Coolant Treatment System 9.10). After the liquid is mixed, sampled, and analyzed, a release rate consistent with dilution flow from the Keowee Hydro Station is determined and the radiation monitor alarm set points adjusted to comply with limits specified in 10CFR20. The release is controlled
16. from the control room with valve LWD-132 which will close on a high radiation

Table 11-4

Waste Disposal Systems Component Data
(Component Quantities for Three Units)

Low Activity Waste Tank	
18. Number	2
Volume each, cu. ft.	398
19. Material	Concrete with Stainless Steel Liner
High Activity Waste Tank	
Number	2
18. Volume each, cu. ft.	262
19. Material	Concrete with Stainless Steel Liner
Waste Holdup Tank	
Number	2
Volume each, cu. ft.	2,700
Material	Carbon Steel with Stainless Clad
Design Pressure	Vessel Full Plus 10 ft. Hydro Head
Spent Resin Storage Tank	
Number	2
Volume, cu. ft.	450 (Unit 1 and 2); 200 (Unit 3)
Material	Stainless
Design Pressure	Vessel Full Plus 10 ft. Hydro Head
Reactor Building Normal Sump	
Number	3
Volume each, cu. ft.	45
Material	Concrete
Reactor Building Emergency Sump	
Number	3
Volume each, cu. ft.	540
Material	Concrete
Waste Gas Tank	
Number	4
Volume each, cu. ft.	1,100
Material	Carbon Steel
Design Pressure, psig	100
Waste Evaporator Feed Tank	
Number	1
Volume, cu. ft.	400
Material	Stainless Steel
Design Pressure	Vessel Full Plus 10 ft. Hydro Head
Waste Evaporator	
Number	1
Process Rates, lb/h	5,060
Material	Stainless Steel
Design Pressure, psig	15

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Waste Disposal Systems Component Data
(Component Quantities for Three Units)

Low Activity Waste Tank Pump	
Number	4
Capacity each, gpm	50
Diff. Head, ft.	100
High Activity Waste Tank Pump	
Number	4
Capacity each, gpm	50
Diff. Head, ft.	40
Waste Transfer Pump	
Number	4
Capacity each, gpm	50
Diff. Head, ft.	50
Spent Resin Sluicing Pump	
Number	2
Capacity each, gpm	50
Diff. Head, ft.	50
Spent Resin Transfer Pump	
Number	2
Capacity each, gpm	10
Diff. Head, ft.	100
Reactor Building Normal Sump Pump	
Number	6
Capacity each, gpm	25
Diff. Head, ft.	28
Waste Evaporator Feed Pump	
Number	1
Capacity, gpm	7-1/2
Diff. Head, ft.	60
Waste Evaporator Recirculating Pump	
Number	1
Capacity, gpm	160
Diff. Head, ft.	53
Waste Evaporator Distillate Pump	
Number	1
18. Capacity, gpm	9-1/2
Diff. Head, ft.	62
Waste Gas Filter	
Number	2
Rating, scfm	200
Type	Prefilter, Absolute and Charcoal
Material	11 Gauge Galvanized Steel

Table 11-4 - Continued

Waste Disposal Systems Component Data
(Component Quantities for Three Units)

Waste Gas Exhauster	
Number	2
Rating, scfm	200 at 6" W.G. External Static Pressure
Type	Backward Curved - Centrifugal
Waste Gas Compressor	
Number	4
18. Capacity each, cfm	48 at 85 psig
Type	Centrifugal Displacement

level interlock. The dilution flow from the Keowee Hydroelectric Plant is controlled from the Oconee Nuclear Station Unit 1 and 2 control room. The release flow is recorded and integrated in the Unit 1 and 2 control room.

11.1.2.2.3 Liquid Waste Holdup Capacity

The estimates of liquid waste holdup times are based on the following assumptions:

(1) The liquid waste generation rates are as follows (See Table 11-1a):

(a) Sampling and laboratory drains	1,250 ft^3 per year for 3 units
(b) Demineralizer sluice	700 ft^3 per year for 3 units
(c) Deborating demineralizer regeneration and rinse	7,600 ft^3 per year for 3 units
(d) Miscellaneous system leakage	11,700 ft^3 per year for 3 units
(e) Equipment decontamination	500 ft^3 per year for 3 units
(f) Laundry and shower drain (one-half the value of Table 11-1a is assumed to have sufficient activity to require processing)	21,900 ft^3 per year for 3 units
(g) Floor drains	10,000 ft^3 per year for 3 units
TOTAL	53,650 ft^3 per year for 3 units

(2) Holdup capacity equals the contents of the high activity waste tank and waste holdup tank which is 2,962 ft^3 for Unit 1 and for Units 1 and 2. Holdup capacity equals two high activity waste tanks and two waste holdup tanks for Units 1, 2, and 3 which is 5,924 ft^3.

(3) Credit is taken for the holdup volume equal to one-half of the capacity since on the average the tanks are half-filled.

(4) The times for filling and discharging the tanks are negligible.

The liquid waste generation rates from (1) above are 17,883 ft^3 for Unit 1, 35,767 ft^3 for Units 1 and 2, and 53,650 ft^3 for Units 1, 2, and 3. The holdup volume for Unit 1 is 1,481 ft^3, for Units 1 and 2 is 1,481 ft^3, and for Units 1, 2, and 3 is 2,962 ft^3. Using the following equation, holdup times calculated for the three units are:

$$\text{Holdup Time} = \frac{\text{Holdup Volume x 365 Days/Year}}{\text{Waste Generation}}$$

Unit 1 Average Holdup Time=30 days
Units 1 and 2 Average Holdup Time=15 days
Units 1, 2, and 3 Average Holdup Time=20 days

It should be noted that these holdup times are based on average expected operating conditions. Unusual conditions may significantly shorten the liquid waste holdup time. It should also be pointed out that the tanks are assumed to fill in a linear rate and that the contents are discharged instantaneously when the tank becomes full.

11.1.2.3 Gaseous Waste Disposal System

11.1.2.3.1 General Description

All components that can contain potentially radioactive gases are vented to a vent header. The vent gases are subsequently drawn from this vent header by one of two waste gas compressors or a waste gas exhauster. The waste gas compressor discharges through a waste gas separator to one of two waste gas tanks. The waste gas tanks and the waste gas exhauster discharge to the unit vent after passing through a filter bank consisting of a prefilter, an absolute filter, and a charcoal filter. A flow diagram of this system with the necessary instrumentation and controls for operation is shown in Figure 11-3. Component data are shown in Table 11-4.

Units 1 and 2 share a gaseous waste disposal system. Unit 3 has a separate waste gas disposal system, which can be interconnected to the gaseous waste disposal system for Units 1 and 2 through double isolation valves between the vent headers. These are normally operated separately, but may be tied together to facilitate maintenance of either of the systems.

The purpose of the gaseous waste disposal system is to:

(1) Maintain a non-oxidizing cover gas of nitrogen in tanks and equipment that contain potentially radioactive gas.

(2) Hold up radioactive gas for decay.

(3) Release gases (radioactive or non-radioactive) to the atmosphere under controlled conditions.

11.1.2.3.2 Operation

One waste gas compressor is normally in continuous operation with the other
31. compressor in a standby condition. The waste gas compressor takes suction
on the vent header and normally discharges into waste gas tank "A" which
16. is used as a surge tank. The vent header pressure control operates a bleed-
back valve (GWD-1) allowing a continuous circulation of gas through the vent
header. As liquid storage tanks connected to the systems are filled, the
excess gas is stored in the waste gas tank. As liquid storage tanks are
emptied, gas flows from the waste gas tank back into the vent header. As
16. waste gas tank "A" is filled, the inlet valve on waste gas tank "B" (GWD-3)
is opened and waste gas tank "A" inlet valve (GWD-2) is closed. The gas in
waste gas tank "A" is allowed to bleed back into the vent header and is
directed into waste gas tank "B" by the waste gas compressor until the
pressure in waste gas tank "A" is at the desired operating pressure. The
valves are then repositioned to utilize waste gas tank "A" as a surge tank
and waste gas tank "B" for radioactive decay. Gas in waste gas tank "B"
is sampled for laboratory analysis to determine the permissible release rate
or need for holdup for radioactive decay.

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16. Release of gas from the waste gas tanks to the unit vent is controlled by the waste gas tank outlet valves GWD-4 and GWD-5. The volume of gas discharged to the unit vent is recorded in the control room. Monitoring of the gas discharged to the unit vent for radioactivity is provided by a radiation monitor which, on a high radiation signal, will close the valves through which the gas is being discharged.

The waste gas exhauster is used when large volumes of gas containing little or no radioactivity are available for release to the unit vent. The waste gas exhauster and its isolation valves are interlocked to trip the exhauster and close the isolation valves in case of a high radiation level in the line going
4. to the unit vent. The waste gas exhauster does not normally operate and is
16. normally valved off by the manual valve upstream of GWD-6. Therefore, no unintentional release of significant activity is possible through this line.

All instrumentation and controls for this system are located in the control room.

11.1.2.3.3 Gaseous Waste Holdup Capacity

The estimates of gaseous waste holdup times are based on the following assumptions:

(1) An annual waste gas volume of 8970 ft^3 (STP) is evolved from three units. (Reference Table 11-1a) Each unit contributes 2990 ft^3 (STP) per year.

(2) Holdup capacity equals the contents of two waste gas tanks which is 2200 ft^3 for Unit 1 and Units 1 and 2 and four waste gas tanks which is 4400 ft^3 for Units 1, 2, and 3.

19. (3) Credit is taken for 75% of the holdup capacity since, for example, in Unit 1, one waste gas tank is used as a surge tank which is on the average one half filled while the other is used as a decay tank.

(4) The times for filling and venting the waste gas tanks are negligible.

(5) The waste gas tanks are initially filled with nitrogen at 10 psig and 100°F. The tanks may be filled to 85 psig and 100°F.

(6) A quantity of 7000 ft^3 (STP) of gas is generated by filling and venting the reactor coolant system. It is assumed that on the average the reactor coolant system will be filled and vented 1.25 times per year.

(7) Due to pumps inside the waste holdup tank, low activity waste tank, and high activity waste tank, it is assumed that these tanks will have to be entered once per year for maintenance and that five times the volume of the tank is needed to properly purge the tank with nitrogen. Therefore, 16,800 ft^3 of waste gas is generated per year.

(8) Other tanks, such as coolant bleed holdup, concentrated boric acid storage tank, coolant bleed evaporator feed tank, condensate test tank, quench tank, letdown storage tank, waste evaporator feed tank, spent resin storage tank, are assumed to require repair on the average of once each five years.

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Five tank volumes of nitrogen are again assumed for proper purging. The volumes of these tanks are 28,510 for Unit 1, 54,890 for Units 1 and 2, and 81,470 for Units 1, 2, and 3.

Therefore 28,510 ft^3 for Unit 1, 54,890 for Units 1 and 2, and 81,470 ft^3 for Units 1, 2, and 3 of waste gas would be generated from this source.

19. From the assumptions above, the quantities of waste gas generated per year are:

Assumption Number	Unit 1	Units 1 & 2	Units 1, 2, & 3
(1)	2,990	5,980	8,970
(5)	3,500	3,500	7,000
(6)	8,750	17,500	26,250
(7)	16,800	16,800	33,600
(8)	28,510	54,890	81,470
Total	60,550	98,670	157,290

From assumptions (2) and (3), the effective holdup volumes are 1,650 ft^3 for Unit 1 and Units 1 and 2 and 3300 ft^3 for Units 1, 2, and 3. At 85 psig and 100°F the effective holdup volume at STP is10,590 ft^3 for Unit 1 and Units 1 and 2 and 21,180 ft^3 for Unit 1, 2, and 3.

Therefore, the gaseous waste holdup times are calculated from the following equation:

$$\text{Holdup time} = \frac{\text{Holdup Volume x 365 days/year}}{\text{Waste Generation}}$$

Unit 1 Average Holdup Time = 63 days
Units 1 and 2 Average Holdup Time = 39 days
Units 1, 2, and 3 Average Holdup Time = 49 days

Therefore, for Unit 1 one waste tank is vented every 42 days. It should be noted that the holdup times are based on average expected operating conditions. Unusual conditions may significantly shorten the gaseous waste holdup time.

11.1.2.4 Process Radiation Monitoring System

11.1.2.4.1 Design Bases and Evaluation

Radiation monitoring of process systems provides early warning of equipment, component, or system malfunctions or potential radiological hazards. The Process Radiation Monitoring System includes alarms, indications, and recording of data in the control rooms. In some cases automatic action is taken upon an alarm condition; in others the alarm serves as a warning to the operator so that manual corrective action can be taken. Radioactive liquid and gaseous waste effluents, particularly, are monitored, coordinated between control rooms,
4. and controlled to assure that radioactivity released does not exceed 10CFR20 limits for the station as a whole.

The sensitivity and the ranges of the detectors have been coordinated with system and environmental dilution factors to assure that releases due to normal, transient, and accident conditions will be monitored and that normal releases will not exceed permissible concentrations. The release of radioactive waste will generally be on a batch basis. Waste releases will also be integrated and recorded. Interlocks are provided to terminate any release of liquid or gaseous waste if a pre-set radiation level is reached. The monitoring and controls exerted by the Process Radiation Monitoring System and the operator during the release will also be supplemented by manual sampling, laboratory analysis, and counting prior to release.

Various detectors are also shielded against ambient background radiation levels that would exist in their location due to normal transient or accident conditions, so that accurate readings of radioactivity will be obtained.

The process monitors have been given a primary calibration with the particular radionuclides that they are expected to monitor. Their energy response has been determined as an aid in measurement of other radionuclides that may also be encountered. A built-in, long-half life source, which is related to the primary calibration, serves as a secondary calibration or check source for each detector. Spectrometer grade amplifiers have been supplied with all of the sodium iodide scintillation detectors so that they can be used with a gamma analyzer for the identification of the specific radionuclides being monitored.

Monitors are also provided on various cooling water systems to detect leakage from normally radioactive systems due to any component failures and thus prevent their accidental release to the environment. In addition to the manual sampling of waste prior to release, mentioned above, the measurement of radioactivity in other process fluids is also supplemented by manual sampling, laboratory analysis, and counting. This is particularly necessary for tritium.

11.1.2.4.2 Description

The radiation monitoring equipment indications and alarms are located in the control rooms from which the systems being monitored are operated. Radiation monitor indications for liquid waste disposal and unit vent effluents are displayed in both control rooms. Outputs from all process monitor channels are recorded on multipoint recorders except for the main steam line monitors which alarm only. Control room annunciation of high radiation level is provided for each channel. Each detector assembly is equipped with a control room operated secondary calibration check source.

Table 11-5 lists the process radiation monitors and gives the following information:

(a) Channel Number and Function - A Radiation Indicating Alarm (RIA) number has been assigned to each detector. Monitors serving the same function have the same number. Prefix numbers indicate the unit on which the detector is used. No prefix number indicates that the detector is utilized for Units 1 and 2. The function shows the system in which the monitor is employed.

(b) Type of Detector - The standard detector type identification is given followed by the size of the crystal or the length of the detector. The lead shield thickness which has been applied to obtain the sensitivities indicated is also given.

(c) Sensitivity - Monitor sensitivities are indicated in terms of background equivalent concentrations and count rate for the radionuclides listed. Background equivalent information shown in the table defines the ability of the monitor to detect the indicated radionuclide concentrations inside the sampler at a count rate that is equal to that resulting from a gamma field outside the sampler. The lead shielding is designed to reduce the count rate resulting from 1.5 MeV gammas in order to obtain the sensitivities shown.

(d) Range - Readout range of monitoring instrumentation, upper range limits, and range overlap between different detectors monitoring the same sample are indicated.

The following is a description of the various applications of these monitors as they are applied to systems:

(a) RIA-31 and 3RIA-31 monitor low pressure service water effluents for gross gamma from each low pressure injection cooler, each reactor building cooler, and component coolers for each unit (9.6.2.3). Samples from the coolers are automatically valved through one of the shielded samplers. The outputs from the detectors are indicated, alarmed, and recorded in the control rooms. Two twelve-point recorders scan the sample valves at an adjustable rate of from two to five minutes per sample and log the output from the detectors. One recorder is used for Units 1 and 2 and one recorder is used for Unit 3. The detectors are located inside the turbine building and are shielded to function during a loss of coolant accident including 100 percent release of fission gases inside the reactor building. These monitors are shown in Figure 1-3.

(b) RIA-32 and 3RIA-32 monitor air from up to 24 locations within the auxiliary building for early detection and location of equipment malfunctions (9.8.2). They also are designed to warn personnel of the presence of radiological hazards. Each monitor incorporates a sample pump that continuously draws samples through a three-way valve manifold at the detector. Sample valves are sequenced by twelve point recorders to direct individual samples to shielded beta sensitive detectors. Detector outputs are logged by the same recorders, which are located in the appropriate control rooms. Loss of sample flow is annunciated in the control rooms.

(c) RIA-33 and RIA-34 are two separate instrument strings that are used to monitor total liquid waste effluent from the station (Figure 11-2). The dynamic range of the two gross gamma monitors is designed to cover normal and abnormal releases. Each monitor is equipped with two adjustable alarm points. One is used to annunciate a high radiation level, and the other automatically terminates the release.

(d) 1RIA-35, 2RIA-35, 2RIA-35A and 3RIA-35 continuously monitor samples of low pressure service water (LPSW) for gross gamma in each LPSW effluent header from the auxiliary building (9.6.2.3). These monitors are provided to supplement laboratory analysis and indications from RIA-31 and 3RIA-31 to assure that LPSW effluents are maintained within safe limits. The detectors are located inside the turbine building. Loss of sample flow is annunciated in the appropriate control room.

(e) IRIA-36, 2RIA-36, and 3RIA-36 monitor reactor coolant letdown upstream of the purification demineralizers (9.1.2). The detectors and associated equipment cover reactor coolant concentrations from approximately 10^{-3} μCi/ml to 10^{3} μCi/ml. For initial reactor operation and low power physics
5. tests, the detectors are designed to be directly adjacent to the sample tube. For monitoring concentrations of gamma emitters, resulting from greater than one percent defective fuel, provisions are made to change the sampler geometry by inserting an attenuator between the sample tube and the detector. Calibration data are available for both configurations.

(f) RIA-37 and RIA-38 monitor waste gas effluent from Units 1 and 2. One instrument channel using a plastic beta scintillation detector and one instrument channel using a Geiger-Mueller (G-M) tube provide the dynamic range indicated on Table 11-5. This range covers normal and abnormal operating conditions with overlap as indicated. Interlocks from these monitors automatically terminate release at preset levels. 3RIA-37 and 3RIA-38 are functionally identical and serve the same purpose for Unit 3. These monitors are shown on Figure 11-3.

(g) RIA-39 and 3RIA-39 monitor control room ventilation using beta sensitive detectors (9.8.2). Samples of control room air are continuously pumped through shielded samplers. Loss of sample flow is annunciated in the appropriate control room.

(h) 1RIA-40, 2RIA-40, and 3RIA-40 monitor condenser air ejector off gas effluent to each unit vent (10.2.7) to detect activity in the steam system resulting from a steam generator tube leak. In addition to this protection, 1RIA-16 and 1RIA-17, which are coaxial ion chambers described in Table 11-7, are located adjacent to the main steam headers in the turbine building. N-16 carryover resulting from a 1 gpm reactor coolant leak into the secondary side of a steam generator at full power operation will produce a field of approximately 35 mR/hr at the detector located 24 inches from the piping. For units 2 and 3, this monitoring function is served by 2RIA-16, 2RIA-17, 3RIA-16, and 3RIA-17.

(i) RIA-41 and 3RIA-41 monitor ventilation air in both spent fuel buildings using beta sensitive detectors (9.8.2). Samples of spent fuel building air are continuously pumped through shielded detectors. Loss of sample flow is annunciated in the appropriate control room.

(j) RIA-42 and 3RIA-42 monitor recirculated cooling water return from auxiliary building for gross gamma activity (9.6.2.4).

(k) 1RIA-43, 1RIA-44, 1RIA-45 and 1RIA-46 monitor unit 1 vent for radioactive air particulates, gas, and iodine. A vent monitor incorporates a sample nozzle, a dual pumping system, and four detector channels. Both pumps are driven from a single motor. One pump supplies samples to an air particulate monitor. The second pump supplies a sample to a fixed charcoal filter
26. that is monitored for iodine and to two gas monitors. Air particulates are detected by monitoring a moving filter paper with a plastic beta scintillator. Iodine is monitored with a NaI scintillator and provisions are made in the readout device for monitoring a selected gamma energy range. Gaseous activity is detected by a plastic beta scintillator for normal ranges. A small volume G-M tube is used in a separate instrument channel to extend the dynamic range of the system. Sensitivity and overlap of the gaseous monitoring ranges are indicated in Table 11-5. Collection efficiency for the air particulate filter is 99 percent for particles 0.5 micron and larger. The activated charcoal cartridge type filter has a rated collection efficiency of at least 90 percent for radio-iodine in forms anticipated.

Malfunctions involving loss of sample flow and depleted, torn, or clogged filter paper are alarmed in the control room.

For Units 2 and 3, this monitoring function is served by 2RIA-43, 44, 45, 46, and 3RIA-43, 44, 45, 46.

Interlocks from the gas monitors automatically terminate a reactor building purge and close the purge isolation valves on high radiation level (5.2.1). These monitors are shown on Figure 11-3.

(l) 1RIA-47, 1RIA-48, and 1RIA-49 and associated equipment make up the reactor building airborne activity monitoring system for Unit 1. The equipment provided is functionally identical to that described for the vent monitors except that the G-M detector is not included. For Units 2 and 3 this monitoring function is performed by 2RIA-47, 48, 49, and 3RIA-47, 48, 49. On high radiation level, interlocks from the gas monitors automatically close the Reactor Building Sump line isolation valves (5.2.1).

(m) 1RIA-50 monitors Unit 1 component cooling system for gross gamma using a NaI scintillator (9.3.2.5). Sample flow loss is alarmed in the Control Room. For Units 2 and 3, this monitoring function is performed by 2RIA-50 and 3RIA-50.

(n) 1RIA-51 is designed to monitor the discharge from the penetration room fans (6.4.2.1). The gamma detector is shielded to permit detection of gaseous activity resulting from reactor building design leakage following a reactor coolant system failure and subsequent release of all fission gases into the reactor building. For Units 2 and 3, this monitoring function is performed by 2RIA-51 and 3RIA-51.

Power for radiation monitoring instrumentation is provided from the station vital buses. Power for sample pumps and solenoid valves is from Engineered Safeguard buses.

Table 11-5

Process Radiation Monitors

Channel Number and Function	Type Detector	Sensitivity (Background Equivalent Concentration and Count Rate)	Range
RIA-31 3RIA-31 Monitors LPSW discharge	Na I 1-1/2"D x 1"L 7-1/2" Pb shield	10 mR/hr=2.5 x 10^{-6} μCi/ml I 131 = 250cpm	(10-10^{6}cpm)
RIA-32 3RIA-32 16. Aux. Bldg. Gas Monitor	Plastic beta Scint. (Ne-102) 2"D x .01"T 4" Pb shield	1 mR/hr=5.5 x 10^{-7} μCi/ml Kr 85 = 56cpm 1 mR/hr=1.4 x 10^{-6} μCi/ml Xe 133 = 56cpm	(10-10^{6}cpm)
RIA-33 Waste Disposal (Normal)	Na I 1-1/2"D x 1"L	1 mR/hr=1.4 x 10^{-5} μCi/ml I 131 = 900cpm	(10-10^{6}cpm) (1.5 x 10^{-2} μCi/ml I 131=10^{6}cpm)
RIA-34 Waste Disposal Liquid (High)	G.M. 5" Pb shield	1 mR/hr=1.5 x 10^{-3} μCi/ml I 131 = 10cpm	(10-10^{6}cpm) (1 x 10^{2} μCi/ml I 131=7 x 10^{5}cpm)
1RIA-35 2RIA-35 2RIA-35A 3RIA-35 Total LPSW discharge from Aux. Bldg.	Na I 1-1/2"D x 1"L 6" Pb shield	1 mR/hr=4.3 x 10^{-6} μCi/ml I 131 = 420cpm	(10-10^{6}cpm)
1RIA-36 2RIA-36 3RIA-36 Reactor Coolant Letdown	Na I 1-1/2"D x 1"L 6" Pb shield	7.7 x 10^{-4} μCi/ml FP(1) = 100cpm 8.9 x 10^{-2} μCi/ml FP(1) = 100cpm	(10-10^{6}cpm) (7.7 μCi/ml FP(1) = 10^{6}cpm) (8.9 x 10^{2} μCi/ml FP(1) = 10^{6}cpm)

(1) Fission Product Spectrum

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Table 11-5 - (Continued)

Process Radiation Monitors

	Channel Number and Function	Type Detector	Sensitivity (Background Equivalent Concentration and Count Rate)	Range
16.	RIA-37 3RIA-37 Waste Disposal Gas (Normal)	Plastic beta scint. (Ne-102) 2"D x .01"T 4" Pb shield	1 mR/hr=11.7 x 10^{-7} μCi/ml Kr 85 = 103cpm 1 mR/hr=2.8 x 10^{-6} μCi/ml Xe 133 = 103cpm	($10-10^6$cpm) (1.1 x 10^{-2} μCi/ml Kr 85=10^6cpm) (2.6 x 10^{-2} μCi/ml Xe 133=10^6cpm)
16.	RIA-38 3RIA-38 Waste Disposal Gas (high)	G.M. 4" Pb shield	1 mR/hr=5.5 x 10^{-2} μCi/ml Kr 85 = 10cpm 1 mR/hr=3.3 x 10^{-3} μCi/ml Xe 133 = 10cpm	($10-10^6$cpm) (5.5 x 10^3 μCi/ml Kr 85=10^6cpm) (3.3 x 10^2 μCi/ml X3 133=10^6cpm)
16.	RIA-39 3RIA Control Room Gas	Plastic beta scint. (Ne-102) 2"D x .01"T 4" Pb shield	1 mR/hr=5.2 x 10^{-7} μCi/ml Kr 85 = 53cpm 1 mR/hr=1.3 x 10^{-6} μCi/ml Xe 133 = 53cpm	($10-10^6$cpm)
	1RIA-40 2RIA-40 3RIA-40 Condenser Air Ejector off gas	Na I 1-1/2"D x 1"L 6" Pb shield	1 mR/hr=4 x 10^{-5} μCi/ml Xe 133 = 400cpm	($10-10^6$cpm)
16.	RIA-41 3RIA-41 Spent Fuel Bldg. Gas	Plastic beta scint. (Ne-102 2"D x .01"T 4" Pb shield	1 mR/hr=5.5 x 10^{-7} μCi/ml Kr 85 = 56cpm 1 mR/hr=1.4 x 10^{-6} μCi/ml Xe 133 = 56cpm	($10-10^6$cpm)
	RIA-42 3RIA-42 Recirculating Coolant Water	Na I 1-1/2"D x 1"L 6" Pb lead	1 mR/hr=6.3 x 10^{-6} μCi/ml I 131 = 420cpm	($10-10^6$cpm)

Table 11-5 - (Continued)

Process Radiation Monitors

Channel Number and Function	Type Detector	Sensitivity (Background Equivalent Concentration and Count Rate)	Range
1RIA-43 2RIA-43 3RIA-43 Unit Vent Particulates	Plastic beta scint. (Ne-102) 2"D x .01"T 2" Pb shield	1 mR/hr=1.2 x 10^{-10} µCi/ml Cs 137 = 200 cpm (15 minute sample)	(10-10^6cpm)
1RIA-44 2RIA-44 3RIA-44 Unit Vent Iodine	Na I 1-1/2"D x 1"L 3" Pb shield	1 mR/hr=8.2 x 10^{-10} µCi/ml I 131=360cpm (15 minute sample)	(10-10^6cpm)
16. 1RIA-45 2RIA-45 3RIA-45 Unit Vent Gas (Normal)	Plastic beta scint. (Ne-102) 2"D x .01"T 4" Pb shield	1 mR/hr=5.5 x 10^{-7} µCi/ml Kr 85 = 56cpm 1 mR/hr=1.4 x 10^{-6} µCi/ml X3 133 = 56cpm	(10-10^6cpm) (9.8 x 10^{-3} µCi/ml Kr 85 = 10^6cpm) (2.5 x 10^{-2} µCi/ml Xe 133 = 10^6cpm)
16. 1RIA-46 2RIA-46 3RIA-46 Unit Vent Gas (High)	G.M. 4" Pb shield	8.1 x 10^{-1} µCi/ml Kr 85 = 200cpm 4.7 x 10^{-2} µCi/ml Xe 133 = 200cpm	(10-10^6cpm) (8.1 x 10^2 µCi/ml Kr 85 = 2 x 10^5cpm) (47 µCi/ml Xe 133 = 2 x 10^5cpm)
1RIA-47 2RIA-47 3RIA-47 Reactor Building Particulate	Plastic beta scint. (Ne-102) 2"D x .01"T 2" Pb shield	1 mR/hr=1.2 x 10^{-10} µCi/ml Cs 137 = 200cpm (15 minute sample)	(10-10^6cpm)

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Table 11-5 - (Continued)

Process Radiation Monitors

Channel Number and Function	Type Detector	Sensitivity (Background Equivalent Concentration and Count Rate)	Range
1RIA-48 2RIA-48 3RIA-48 Reactor Building Iodine	Na I 1-1/2"D x 1"L 3" Pb shield	1 mR/hr=8.2 x 10^{-10} μCi/ml I 131 = 360cpm (15 minute sample)	(10-10^6cpm)
16 1RIA-49 2RIA-49 3RIA-49 Reactor Building Gas	Plastic Beta Scint. (Ne-102) 2"D x .01"T 4" Pb shield	1 mR/hr=5.5 x 10^{-7} μCi/ml Kr 85 = 56cpm 1 mR/hr=1.4 x 10^{-6} μCi/ml Xe 133 = 56cpm	(10-10^6cpm)
1RIA-50 2RIA-50 3RIA-50 Component Cooling Water	Na I 1-1/2"D x 1"L 6" Pb shield	1 mR/hr=4.3 x 10^{-6} μCi/ml I 131 = 420 cpm	(10-10^6cpm)
1RIA-51 2RIA-51 3RIA-51 Penetration Room Gas	GM Tube 10" Pb shield	(MHA), Noble Gas Conc. = <5 x 10^5cpm	(10-10^6cpm)
1RIA-16 1RIA-17 2RIA-16 2RIA-17 3RIA-16 3RIA-17 Steam Header N16	Coaxial ion chamber		0.1-10^7 mR/hr

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11.1.2.5 Design Evaluation

All analyses on liquid and gaseous waste disposal were performed on the basis of three units operating with one percent defective fuel. Although it is not expected that the number of clad defects will ever approach one percent of the total fuel, the objective is to demonstrate the capability of safe station operation within the limits of 10CFR20 with these design quantities of radioactive fission products in the system. Thus, the adequacy of the waste disposal system design is demonstrated.

A summary of the various operations considered in the analyses and the total concentrations (given as fractions of the MPC for unrestricted areas) resulting in the station effluent from operation of three units with defective fuel are given in Table 11-6. The concentration of each radioactive nuclide has been divided by its respective Maximum Permissible Concentration for discharge into unrestricted areas as set forth in 10CFR20, and the total concentration was obtained by summation over all isotopes considered.

11.1.2.5.1 Liquid Wastes

The accumulated coolant bleed per unit over a core cycle is about 44,000 ft^3, including startup expansion and dilution and coolant letdown to storage for reduction of boron concentration. The reactor coolant bleed, including startup expansion and dilution, is processed by the coolant bleed evaporator for reuse and is not discharged.

Two coolant bleed holdup tanks, (Coolant Storage System, 9.9) each with a capacity of 11,000 ft^3, are provided for each unit. Considering the rate of bleed input to the tank and the rate at which it is processed by the evaporator, one tank provides adequate storage for the entire 44,000 ft^3 of reactor coolant bleed prior to treatment by the evaporator. The remaining tank provides adequate storage for the processed bleed awaiting reuse as feed to the reactor coolant system. The storage of both unprocessed and processed bleed requires approximately 55 percent of the volume of the bleed holdup tanks for each unit.

However, in the event that discharge of the coolant bleed is necessary, the activity level in the station effluent was determined by assuming that the entire amount of reactor coolant system liquid was discharged to the Keowee Hydro tailrace. Letdown through the purification demineralizer was assumed with the same decontamination factors as those in 11.1.1.3.1. Holdup and decay for a 30 day period was assumed. Dilution through the tailrace was by the average stream flow of 1100 cfs. The results were a concentration level of 0.026 MPC for this discharge from one unit into unrestricted areas. Adequate holdup volume for both processed and unprocessed bleed was established in the preceding paragraph.

The maximum volume of coolant removed from one unit during heatup and dilution from a cold shutdown is approximately 9900 ft^3. This occurs at the end of the period during which the evaporator is used for treatment of reactor coolant bleed. Earlier in life, the quantity removed would be less. Even if this startup and dilution storage requirement occurs within a short time period of one of the four startups for each unit already included in the above analysis, less than 15 percent of the total storage capacity will be required.

Table 11-6

Maximum Activity Concentrations in the Station Effluent for Three Units, Each Operating with One Percent Defective Fuel

Liquid Waste

Operation	Yearly Average Concentration in Tailrace Discharge, Fraction of MPC
Lifetime Shim Bleed Including Startup Expansion and Dilution	0.077
Discharge of Miscellaneous Wastes	0.16

Gaseous Wastes

Operation	Yearly Average Concentration at Site Boundary, Fraction of MPC
Lifetime Shim Bleed	0.058
Startup Expansion and Dilution	0.18
Venting of Letdown Storage Tank	0.015
Venting of Pressurizer	0.011
Reactor Building Purge	0.11
Steam Generator Tube Leakage of 1 gpm in one unit	0.089

The remaining coolant removed from the reactor coolant system is the partial drain which occurs once per year during refueling. The coolant is removed in a batch of 6100 ft^3 per unit and returned to the reactor coolant system upon completion of refueling. Thus, it occupies storage capacity only during the period of refueling. The required storage volume for refueling operations of 6100 ft^3 is less than 10 percent of the available capacity.

It is unlikely that operating conditions could occur which would require simultaneous storage for all of the above liquid wastes. However, even if simultaneous storage were required, it could be accommodated by less than 80 percent of the available storage capacity. This demonstrates that the six tanks provide adequate capacity to accommodate radioactive wastes as well as providing extra capacity for liquid storage when desired.

The station storage facilities for miscellaneous liquid wastes include:

(1) Two waste holdup tanks, each 2700 ft^3 = 5400 ft^3

(2) Two low activity waste tanks, each 450 ft^3 = 900 ft^3

(3) Two high activity waste tanks, each 300 ft^3 = 600 ft^3

(4) Three reactor building normal sumps, each 45 ft^3 = 135 ft^3

(5) Three reactor building emergency sumps, each 540 ft^3 = 1620 ft^3

Miscellaneous waste activity levels were determined by assuming that all liquid collected was reactor coolant containing the design fission product activity. Collection was assumed to take place at the rate of 435 gpd for 60 days resulting in an accumulation of 3490 ft^3 of liquid, compared to a storage capacity of approximately 8100 ft^3. An additional holdup of 30 days for this liquid prior to discharge was assumed, followed by discharge to the Keowee Hydro tailrace with a dilution flow of 1100 cfs. The resulting station effluent concentration averaged over 60 days was 0.16 of the MPC for unrestricted areas.

The above analysis demonstrates the station capability for handling large quantities of liquid wastes within allowable limits. For the reactor coolant bleed, the purification demineralizers, the coolant bleed evaporator, and the large system storage capacity provide ample means of collection and disposal for liquid wastes, even in the remote case of one percent fuel failure. Similarly, the miscellaneous wastes are shown to present no problem when analyzed on this conservative basis. It is concluded that the capacity of the liquid waste disposal system is large enough to permit flexibility in station operations while providing a means for safe disposal of wastes with activity well below the acceptable limits.

11.1.2.5.2 Gaseous Wastes

In determining the activity concentrations in the gaseous effluent, the values for atmospheric dilution, evaluated at the Exclusion Area Boundary, were taken
8. from Section 2.3 (Meteorology). All gases handled by the waste gas system pass through the waste gas filter prior to discharge via a unit vent, and a 90 percent removal efficiency for iodine is included in all the analyses.

The collection of gaseous activity was determined for those components representing the maximum potential radiation hazard, including the reactor building, letdown storage tank, pressurizer, and coolant bleed holdup tanks.

The discharge of activity to the atmosphere as a result of reactor coolant bleed was determined for two situations: (1) continuous bleed over life for reduction of boron concentration, and (2) dilution and expansion following shutdown and startup. For the case of continuous bleed, a total coolant volume of approximately 15,000 ft^3 per unit is letdown to the coolant bleed holdup tanks over a core cycle. All of the Kr, Xe and I in the coolant letdown was assumed to come out in the void space of the coolant bleed holdup tanks. The activity levels in the coolant letdown to the coolant bleed holdup tanks were the design activity levels reduced by the purification demineralizer removal efficiencies (11.1.1.3.1). The activity was released to the atmosphere, without holdup, at a rate equal to the yearly average shim bleed rate of approximately 12 gph. With three units releasing activity at this rate, the total average annual con-
8. centration at the Exclusion Area Boundary is about 0.058 MPC for unrestricted areas.

In the case of unit shutdown and startup, it was postulated that a cold shutdown occurred at a time in lifetime just prior to the beginning the use of the deborating demineralizer for boric acid removal. This results in the maximum quantity of coolant bleed during startup. No coolant activity decay was assumed during the shutdown. As a result of this operation, a bleed quantity of approximately 9,900 ft^3 occurs from one unit. The activity levels in the coolant letdown to the coolant bleed holdup tanks were the design activity levels reduced by the purification demineralizer efficiencies. As the coolant is let down to the bleed holdup tanks, all of the Kr, Xe and I is assumed to come out of the water and go into the waste gas tanks. The 9,900 ft^3 of gas displaced from the bleed holdup tank would require approximately 33 percent of the capacity of the four waste gas tanks, each with a volume of 1,100 ft^3 and a design pressure of 100 psig. The gaseous activity could then be discharged over a period of one week to allow dispersion in accordance with the seven day atmospheric diffusion
8. model. The average yearly concentration at the Exclusion Area Boundary, for one such operation for one unit, would be about 0.020 of the MPC. Four startups, spaced equally over the cycle, were considered for each unit. The average annual concentration resulting from three units operating in this manner would be 0.18 MPC.

The gaseous concentrations in the letdown storage tank void were determined from Henry's Law assuming the tank gas space is in equilibrium with the reactor coolant. The fraction of total activity in the reactor coolant system which collected in the letdown storage tank was approximately 30 percent for Kr, 19 percent for Xe, and 0.09 percent for I. The activity levels used for sources in the letdown storage tank were based on the reactor coolant system design activity. It is assumed that the tank in each unit is vented, once a year, to the waste gas tank. The volume of gas in the letdown storage tank is about 300 ft^3 at 45 psia. This gas would require only 3 percent of the capacity of the four waste gas tanks. This gas can be discharged to the atmosphere over a period of one week, allowing dispersion in accordance with the seven day atmospheric diffusion model. The average yearly concentration of activity at the
8. Exclusion Area Boundary resulting from this operation for three units is computed to be 0.015 of the MPC.

Calculations similar to those used for the letdown storage tank were performed to determine the activity in the pressurizer. The fraction of total activity in the reactor coolant system which collected in the pressurizer was approxi-
3. mately 21 percent for Kr, 15 percent for Xe, and 0.08 percent for I. Venting of the pressurizer results in only about 60 ft^3 of gas, which can be released from the waste gas tank over a period of one week to give a yearly average con-
8. centration (from three units) of 0.011 of the MPC at the Exclusion Area Boundary.

The activity level in the reactor building atmosphere was computed assuming a reactor coolant system leakage to the reactor building air of 10 gpd per unit. All of the Kr and Xe, and 50 percent of the I that leaked from the reactor coolant system, based on coolant design activities, were dispersed throughout the reactor building atmosphere. Activity buildup in the reactor building was computed over a 30 day period without reactor building purge. This quantity of activity was then discharged to the atmosphere, without additional decay, by
8. way of the reactor building purge system. The concentration at the Exclusion Area Boundary averaged over 30 days was computed to be 0.037 of the MPC for one unit. Venting three reactor buildings once each 30 days would give an average yearly dose of 0.11 of the MPC.

This evaluation demonstrates that the total yearly average concentration of
8. activity at the Exclusion Area Boundary from all modes of gaseous release, including reactor building purge, pressurizer venting, venting of the letdown storage tank, startup expansion and dilution, and chemical shim bleed, is a maximum of about 0.12 of the MPC for one unit. Even in the remote instance of one percent fuel defects in three units concurrently, the average yearly con-
8. centration at the Exclusion Area Boundary would be about 0.38 of the MPC. The evaluation also demonstrated that equipment capacities are adequate to accommodate and store radioactive gases as necessary. Thus, the system design is adequate to insure safe disposal of gaseous wastes.

An analysis has been made to examine the consequences of reactor operation with steam generator tube leakage and one percent defective fuel rods. In the air ejector exhaust, the controlling isotope is xenon-133. The analysis assumed that the xenon passed directly from the reactor coolant system leak to the condenser with all the activity ultimately released to the unit vent with no radioactive decay. The analysis was based on a reactor coolant leakage rate of one
8. gpm in one unit continuously over a year. The concentration at the Exclusion Area Boundary was computed to be 0.089 MPC. Therefore, much higher tube leakages could be permitted for shorter periods of time.

During release of gas from the waste gas tank to the unit vent, if a failure of
16. the control or protective instrumentation for the control valve (GWD-4 or GWD-5) should occur, the following alarms and features are available to prevent an inadvertent release of activity.

4. (a) Radiation indicators and alarms monitor both the gas release line from the waste gas tank and the unit vent.

16. (b) Control valves (GWD-4, GWD-5) fail closed on loss of air or loss of power to the 3-way solenoid valve in the air line.

16. (c) Control valves (GWD-4, GWD-5) can be closed from the control room either by the pneumatic valve controller or by the manual switch for the 3-way solenoid valve which will block the air supply to and exhaust the air
4. from the control valve.

(d) Manually operated valves are located both upstream and downstream of the
16. control valves (GWD-4, GWD-5).

16. Should bleedback valve (GWD-1) fail instantaneously full open, a maximum gas volume of 593 SCFM (waste gas tank @ 85 psig) could pass through it to the waste gas compressor suction header. An immediate consequence of this event would be the sounding of an alarm in the control room, indicating a high pressure in the waste gas compressor suction header. The control room operator
16. 18. has the capability to close valve (GWD-1) from the console control station. In the event of operator delay in the manual-remote closing of valve GWD-1 the relief valve mounted in the waste gas compressor suction header will begin to relieve at 21 inches W.G. pressure and pass excess gas to the station stack via the waste gas filter. This relief valve is sized to fully relieve at 24 inches W.G. and pass 600 SCFM. Bleedback valve could be passing a maximum of 593 SCFM. However,
5. the compressor would pump 40 SCFM of this back to the tank so that the maximum of gas through the relief valve would be the difference between 593 and 40 SCFM, or 553 SCFM.

The consequence of GWD-1 failing open without operator action would be the same as the rupture of one waste gas tank as analyzed in Section 14.2.2.5 except that the relief valve discharges through a particulate-absolute-carbon
19. filter chain. The iodine dose is therefore limited by the efficiency of the carbon filter. Consequently, the off-site two hour dose is .013 rem to the thyroid assuming 90 percent filter efficiency and .19 rem to the whole body.

The liquid waste tanks, connected to the vent header, are designed for an internal pressure of 10 feet of water (120 inches W.G.). Even if the gas compressor were not operating, the relief valve would pass 56 SCFM with the vent header pressure at less than 120 inches W.G.

11.1.3 TESTS AND INSPECTIONS

Each process radiation monitoring channel will be functionally tested and calibrated periodically to verify proper operation of components and to insure that the desired detector sensitivities are maintained.

A signal generator located within the process monitor panel will be used to check the alignment of electronic modules. After the electronic alignment is completed, a remote operated calibration source is actuated to determine proper channel response as referenced to primary calibration data.

The flow measuring instrument and controls associated with the gaseous and liquid waste effluent lines will be calibrated periodically to insure proper accuracy, measurement, and control of radiography releases from the station.

Efficiency of the particulate filters is determined in the factory, as well
5. as in-place, in accordance with USA DOP (Dioctyl Phtholate) test method and

UL standard 586. DOP smoke is introduced upstream of the filter and the quantity detected downstream of the filter is measured. This test is conducted at full rated flow capacity, minimum acceptable test efficiency for the particulate filter is 99.97%. The difference between factory and in-place tests is the test duration time: 30 seconds and 2 minutes, respectively.

Efficiency of the iodine filter is determined by two different methods in the factory. One employs I_2-131, I_2, and CH_3I; the other uses refrigerant-112.

TEST WITH IODINE

The filter shall remove at least 99.9% of molecular iodine-131 (I_2-131) in the presence of a gaseous concentration of 50 mg per m^3 of non-radioactive molecular iodine (I_2) plus 5 mg per m^3 of non-radioactive methyl iodide (CH_3I). This performance level is maintained until the amount of non-radioactive I_2 having reached the test unit is equivalent to 200 gm in the full scale system. Following this loading, feeding of non-radioactive I_2 and CH_3I is halted, and air at 70% RH and 150°F is drawn through the test unit at its rated flow for two hours. The integrated I_2-131 removal efficiency for the test unit, including both iodine feed and elution periods, shall be no less than 90.0%. The I_2-131 activity during I_2-131 feed periods is between 10 and 100 microcuries per gram of non-radioactive I_2 fed.

The filter is required to remove at least 99.0% of methyl iodide-131 (CH_3I-131) in the presence of a gaseous concentration of 50 mg per m^3 of non-radioactive methyl iodide (CH_3I). This performance level is maintained until the amount of CH_3I having reached the test unit is equivalent to 200 gm in the full scale system. Following this loading, feeding of I_2 and CH_3I is halted, and air at 70% RH and 150°F is drawn through the test unit at its rated flow for two hours. The integrated efficiency in the removal of CH_3I-131 by the test unit, including both feed and elution periods, is required to be no less than 65%. The CH_3I-131 activity during CH_3I-131 feed periods is between 10 and 100 microcuries per gram of non-radioactive CH_3I fed.

TEST WITH REFRIGERANT-112

Refrigerant-112 is injected into an air flow of 333 cfm upstream of the filter until the concentration is 50 ppm. After 2 minutes the refrigerant-112 concentration downstream of the filter is required to be less than 0.1 ppm.

Field tests for efficiency will be performed using refrigerant-112 only. The system will be operating at rated flow. Refrigerant-112 is introduced upstream of the filter to produce an R-112 concentration of 50 ppm. With an upstream concentration of 50 ppm and a test of 2 minutes, the maximum allowable downstream concentration is 0.1 ppm.

11.2 RADIATION PROTECTION

11.2.1 SHIELDING

11.2.1.1 Design Bases

The shielding is designed to perform two primary functions: (1) to insure that, during normal operation, the radiation dose to operating personnel and to the general public is within the limits set forth in 10 CFR 20, and (2) to insure that operating personnel are adequately protected in the event of a reactor accident so that the accident can be terminated without undue hazard to the general public.

Operation of each unit at rated power was assumed. System activity levels are based on the reactor coolant system design activity levels defined in 11.1.1.3. Other parameters employed in shielding analyses are listed in Table 11-7.

Each area in the station is classified according to the dose rate allowable in the area, based on the expected frequency and duration of occupancy. These radiation zones are summarized below.

	Location	Dose Rate, mrem/hr
8.	Exclusion area boundary	0.05
	Offices, control room turbine building	0.5
	Normally accessible areas in auxiliary building	2.0
	Above fuel storage pool with normal complement of fuel assemblies	2.5
	Above reactor vessel and over fuel storage pool when handling fuel assemblies	10
	Normally accessible areas in reactor building during full power operation	25
9.	Inside control room following maximum hypothetical accident	3 rem whole body total dose;(integrated over first 90 days after accident, assuming 8 hours per day per shift.)

Piping and equipment components are shielded by concrete walls and floors of varying thickness, depending on the magnitude of the sources in each pipe section and component and on the access requirements in a particular area. In some areas local shielding in the form of removable lead or concrete blocks will be utilized to facilitate maintenance or repair operations.

11.2.1.2 Reactor Building Shielding

11.2.1.2.1 Primary Shield

The primary shield consists of reinforced concrete which surrounds the reactor vessel and extends upward from the reactor building floor to form the walls of

Table 11-7

Parameters Used for Shielding Analyses

Core Thermal Power	2568 MWt
Effective Core Diameter	128.9 in.
Active Fuel Length	144 in
Thickness of Core Liner	0.75 in.
Thickness of Core Barrel	2 in.
Thickness of Thermal Shield	2 in.
Reactor Vessel I.D.	171 in.
Reactor Vessel O.D.	188 in.
Volume of Reactor Coolant	11,478 Ft^3
Normal Letdown Flow	1 reactor coolant system volume per day
Time Between Shutdown and Fuel Handling	100 hrs.
Coolant Transit Times (Seconds):	
Core Inlet to Core Exit	0.8
Core Exit to S.G. Inlet	2.8
S.G. Inlet to S.G. Outlet	5.2
S.G. Outlet to Core Inlet	4.0
Total Loop Time	12.8

the fuel transfer canal. The shield thickness is 5 ft up to the height of the reactor vessel flange, where the thickness is reduced to 4.5 ft. The primary shield is designed to meet the following objectives:

(a) To attenuate the neutron flux in order to limit the activation of component and structural materials.

(b) To limit the radiation level after shutdown so that access to the reactor coolant system equipment is permissible.

(c) To reduce, in conjunction with the secondary shield, the radiation level from sources within the reactor vessel to allow access to the reactor building during normal full power operation.

11.2.1.2.2 Secondary Shield

The secondary shield is a 4 ft thick reinforced concrete structure which surrounds the reactor coolant equipment, including the piping, pumps, and steam generators. The shielding is designed to reduce radiation levels from activity in the reactor coolant and to supplement the primary shield in the attenuation of neutrons and secondary gamma rays to permit limited access to the reactor building during full power operation.

11.2.1.2.3 Reactor Building Shield

The reactor building shield is a reinforced, prestressed concrete structure with 3.75 ft thick cylindrical walls and a 3.25 ft thick dome. In conjunction with the primary and secondary shields, it will limit the radiation level outside the reactor building from all sources inside the reactor building to no more than 0.5 mrem/hr at full power operation. The shielding is also designed to protect station personnel from radiation sources inside the reactor building following the Maximum Hypothetical Accident. Additional shielding is provided around the control room to insure that exposure of operating personnel in the control room is within the design limits following the MHA.

Other significant shielding inside the reactor building is listed in Table 11-8.

11.2.1.2.4 Materials

The material used for the primary, secondary, and reactor building shields is ordinary concrete with a density of approximately 140 lbs/ft^3. Since the primary and secondary shielding walls serve as the refueling structure, give support for the reactor coolant components under pipe rupture conditions, and provide missile shielding, they are reinforced and designed to be self-supporting.

11.2.1.2.5 Design Evaluation

The thickness of the reactor building shielding, in accordance with the design dose rate criteria, is based upon radiation levels due to fission product release following a reactor accident. For the evaluation of the MHA, it was assumed that 100 percent of the gases, 50 percent of the halogens, and 1 percent of the solid fission products were instantaneously released to the reactor building following a buildup period in the core of 600 full power days.

Table 11-8

Principal Shielding

Reactor Building

Component	Concrete Thickness (ft)
Primary Shield (Below Flange)	5
(Above Flange)	4.5
Secondary Shield	4
Reactor Building Vertical Walls	3.75
Reactor Building Dome	3.25
Side Walls of Fuel Transfer Canal	4.5
End Walls of Fuel Transfer Canal	2.5, 4
Floor of Fuel Transfer Canal	4
Minimum Water Over Active Fuel During Transfer	9 ?

Auxiliary Building

Component	Concrete Thickness (ft)
Spent Resin Tanks	4
Evaporator Equipment	3.0-4.0
Purification Demineralizers	4
Deborating Demineralizers	4
Component Drain Pump	1.5-4.0
Spent Resin Transfer Pump	2-4
Spent Resin Sluicing Pump	2-4
Waste Transfer Pump	1-4
Low Pressure Injection Pumps	2.5-3.5
High Pressure Injection Pumps	1.5-4.0
Coolant Bleed Holdup Tanks	1.5-4.0
Concentrated Boric Acid Storage Tank	1.5-2.0
Waste Drumming Area	1.5-2.0
Low Pressure Injection Coolers	3
Letdown Storage Tanks	4
Waste Holdup Tank	2.0
Waste Gas Tank	3.0-3.5
Waste Gas Compressors	3.0-3.5
Bleed Evaporator Feed Tank	2.0-3.0
Spent Fuel Coolant Pumps & Coolers	2.5-3.5
Side Walls of Storage Pool	3.5
End Walls of Storage Pool	5.5

The fission product activity was assumed to be uniformly dispersed throughout the reactor building volume, and the reactor building was represented by a cylindrical source for the dose calculations. The integrated dose over various time intervals was computed as a function of distance from the reactor building. (14.2.2.4 and Figure 14-67).

11.2.1.3 Auxiliary Building Shielding

The major radiation sources will be piping and equipment components handling potentially contaminated fluid, practically all of which are located on the 758'-0", 771'-0", and 783'-9" levels (Figures 1-2, 1-3, and 1-4). Groups of equipment or individual equipment items are separated by shielding walls such that systems and equipment can be isolated for maintenance with no significant radiation interference from other systems or equipment. During normal operation, there is no need to occupy these potentially radioactive equipment areas. Potential radiation sources and associated shielding are listed in Table 11-8.

11.2.2 AREA RADIATION MONITORING SYSTEM

11.2.2.1 Design Bases

The Area Radiation Monitoring System, consisting of coaxial ion chambers, is designed to indicate radiation levels that exist in various remote locations throughout the station where personnel are most likely to be exposed and to alarm upon high radiation levels. Indications from the monitors are used in conjunction with station operating procedures to assure that radiation exposure of personnel does not exceed 10CFR20 limits.

11.2.2.2 Description

Numbers and locations of the Area Radiation Monitors are shown in Table 11-9.

Control room indication is provided for each chamber on readout modules that indicate 10^{-1} to 10^{7} mR/hr, or any three consecutive decades of that range. Readout modules for Unit 1 and 2 monitors are located in Unit 1-2 control room. Readout modules for Unit 3 monitors are located in Unit 3 control room.

Each detector assembly is equipped with a check source that is automatically actuated on a periodic basis. The failure of any channel to respond to the source will initiate an alarm in the control room. High radiation level at any detector is alarmed at the detector location and in the control room.

11.2.2.3 Evaluation

The Area Radiation Monitoring System detectors are located throughout the station in locations where significant radiation levels may exist, which may change with time and with the operation being performed. They are designed primarily for the protection of personnel performing such operations as routine coolant sampling, refueling, reactor building entry, radioactive waste disposal operations and for certain other operating and maintenance work. The system has sufficient range and flexibility to permit readout during routine operations and during any transient or emergency conditions that may exist. The equipment is

Table 11-9

Area Radiation Monitors

Number of Detectors	Monitor	Location
2	RIA-1	Control Room Unit 1 & 2, Unit 3
3	RIA-2	Main Fuel Handling Bridge Unit 1, 2, 3
3	RIA-3	Auxiliary Fuel Handling Bridge Unit 1, 2, 3
3	RIA-4	Reactor Building Entrance Unit 1, 2, 3
3	RIA-5	Incore Instrument Handling Area Unit 1, 2, 3
2	RIA-6	Spent Fuel Building Bridge Unit 1 & 2, Unit 3
1	RIA-7	Hot Machine Shop
1	RIA-8	Hot Laboratory
1	RIA-9	Low Level Drumming Area
3	RIA-10	Sample Area Unit 1, 2, 3
2	RIA-11	Auxiliary Building Corridor Elevation 796' Unit 1 & 2, Unit 3
2	RIA-12	Chemical Addition Area Unit 1 & 2, Unit 3
2	RIA-13	Waste Disposal Control Area Unit 1 & 2, Unit 3
1	RIA-14	Waste Drumming Area
2	RIA-15	Auxiliary Building Corridor Elevation 758' Unit 1 & 2, Unit 3
3	RIA-16*	Steam Generator "A" Main Steam Header Unit 1, 2, 3
3	RIA-17*	Steam Generator "B" Main Steam Header Unit 1, 2, 3

* These are functionally Process Radiation Monitors, but are identical physically to the Area Monitors and are contained in the same cabinets.

self-checking for proper operation and alarms both in the local area and in the respective control room. Where necessary or desirable, readout is also provided locally.

11.2.3 HEALTH PHYSICS

The Station Superintendent is responsible for all health physics control measures at the Oconee Nuclear Station. This includes the protection of all persons against radiation and compliance with all AEC regulations and license conditions. This responsibility is in turn shared by all supervisors. All personnel assigned to the station and all visitors are required to follow rules and procedures established by administrative control for protection against radiation and contamination.

The Duke Power Company Staff Health Physicist has established the Health Physics Program at the Oconee Nuclear Station. He will also provide technical guidance for conducting this program and will periodically audit the effectiveness and the results of the program.

The station Health Physics Supervisor is responsible for conducting the established Health Physics Program. It is the duty of the Health Physics Section at the station to train personnel in radiation safety; to continuously evaluate and review the radiological status of the station; to make recommendations for control or elimination of radiation hazards; and to control the radiation exposure of personnel. The Health Physics Section also functions in an advisory capacity to assist all personnel in carrying out their radiation safety responsibilities and to audit all aspects of station operation and maintenance to assure safe conditions and compliance with AEC and other Federal and State regulations concerning radiation protection.

Administrative controls relating to radiation protection are subject to the same review and approval as those that govern other station procedures, (Section 12.5, Administrative Control). These procedures include a Radiation Work Permit System. All work on systems or locations where exposure to radiation or radioactive materials is or may be involved, requires an appropriate Radiation Work Permit initiated by Health Physics and approved by the Operations Shift Supervisor before work can begin. The radiological hazards associated with the job are determined and evaluated prior to issuing the permit. The work permit lists the precautions to be taken, the protective clothing to be worn, and any other radiation control and safety precautions that are required.

11.2.3.1 Personnel Monitoring Systems

Personnel monitoring equipment consisting of film badges and/or their equivalent, (thermoluminescent dosimeters, TLD's), are assigned by the Health Physics Section and worn by all personnel at Oconee whose job involves significant levels of radiation exposure as defined in 10CFR20. In addition, pocket chambers, self reading dosimeters, pocket high radiation alarms, wrist badges, and/or finger tabs are readily available for use by those persons who ordinarily work in the Controlled Area or whose job requires frequent access to this area. This equipment is worn as required by the Radiation Work Permit or routinely as determined by station radiological conditions. Neutron sensitive film is also issued or assigned as required by plant conditions. This personnel monitoring equipment is also available on a day-to-day basis for those persons, employees or visitors not assigned to the station who have occasion to enter the Controlled Area or to perform work involving possible exposure to radiation. Records of radiation exposure history and current occupational exposure are maintained by the Health Physics Section for each individual for whom personnel monitoring is required. The external radiation dose to personnel is determined on a daily and/or weekly basis, as necessary, by means of the pocket chamber and dosimeter. Film badges are processed monthly or more frequently when conditions indicate it is necessary. Any personnel monitoring equipment worn by station personnel who do not enter the Controlled Area are evaluated on a quarterly basis.

11.2.3.2 Personnel Protective Equipment

Special "protective" or "anti-contamination" clothing is furnished and worn as necessary to protect personnel against contact with radioactive contamination. This consists of coveralls, lab coats, surgeon caps, hoods, gloves, and shoe covers. Change rooms are conveniently located throughout the station for proper utilization of this protective clothing. Respiratory protective equipment is also available to supplement process containment and ventilation controls, for the protection of personnel against airborne radioactive contamination and the possibility of internal radiation exposure. This equipment consists of full face air purifying respirators and self-contained breathing apparatus. Also, a Breathing Air System has been installed in the station, and respiratory protective equipment consisting of air line full face respirators, hoods, and plastic suits is provided, should its use become necessary or desirable.

Maintenance of the above equipment is in accordance with the manufacturer's recommendations and rules of good practice, such as those published by the American Industrial Hygiene Association in its "Respiratory Protective Devices Manual". The use and maintenance of protective clothing and respiratory protective equipment is under the direct control of the Health Physics Section and personnel are trained in the use of this equipment before using it in the performance of their work. The use of this equipment is in accordance with the proposed Section 20.103 and Appendix E of 10CFR20.

11.2.3.3 Facilities and Access Provisions

Change room facilities are provided where personnel obtain clean protective clothing and other equipment required for station work. The major change rooms serve the Reactor Buildings and the Auxiliary Buildings and a separate change room is provided for each Spent Fuel Pool. These facilities are divided into clean and contaminated sections. The contaminated section of the change rooms is used for the removal and handling of contaminated protective clothing after use. A small change room is also provided for the Hot Machine Shop. Provisions for change and personnel decontamination are also available in the First Aid Room within the Controlled Area. Showers, sinks, and necessary monitoring equipment are provided in all of the change rooms to aid in the decontamination of personnel.

Equipment decontamination facilities are also provided at the station for large and small items of plant equipment and components. There is one such facility for each unit, and in addition, a Cask Decontamination Area is provided adjacent to each Spent Fuel Pool. A Decontamination Laundry is also provided.

Provision is also made for decontamination of work areas throughout the station. This consists of several janitor's sinks on each floor level in the Auxiliary Buildings and in the Reactor Buildings. Drains from all of these facilities go to the various radioactive liquid waste drain tanks. Appropriate written procedures govern the proper use of protective clothing, the change rooms, and the decontamination facilities.

In order to protect personnel from radiation, the Controlled Area of the plant is divided into areas of increasingly controlled access depending on radiation levels. Protection of personnel from access to radiation area and high radiation areas that exist temporarily or permanently as a result of station operations and maintenance is by means of appropriate radiation warning signs, barricades, closed doors, audible and visual indicators, and alarms as required by 10CFR20. Administrative controls are also used in conjunction with the above and keys are issued for access to the Controlled Area of the plant and to limited access areas within the Controlled Area under certain conditions. The Radiation Work Permit system is also utilized to control access to high radiation areas.

11.2.3.4 Health Physics and Chemistry Facilities

The Health Physics and Chemistry Facilities are centrally located within the station and are clustered in the Auxiliary Building midway between the three reactors for efficiency of operation. In addition to the Health Physics Supervisor's office and the Chemist's office, a Health Physics Operations office is also provided. Laboratory facilities consist of a Conventional Chemistry Laboratory, a Radio-Chemistry Laboratory, a shielded Counting Room, and Environmental Monitoring Laboratory, and a shielded Radiation Survey Instrument Calibration Room. These facilities are equipped for conducting the health physics and chemistry programs for the station and for detecting, analyzing, and measuring all types of radiation and for evaluating any radiological problem that may reasonably be expected at a pressurized water reactor. Equipment for analyzing environmental radioactivity samples, performing measurements for internal personnel dosimetry purposes, and for radio-bioassay is also included.

11.2.3.5 Health Physics Instrumentation

Different types of instruments have been selected to cover the entire spectrum of radiation measurement problems expected at the Oconee Nuclear Station. This includes instruments for detecting and measuring alpha, beta, gamma, and neutron radiation. Sufficient quantities have been obtained to allow for use, calibration, maintenance, and repair.

Counting room instruments for radioactivity measurements include:

- a multi-channel gamma analyzer
- several automatic and manual beta-gamma counter-scalers
- an alpha scintillation counter-scaler
- a gamma counter-scaler
- a liquid scintillation counter for tritium counting

a vibrating reed electrometer
a TLD reader
a low background gas-proportional counter
a body-burden analyzer

Portable radiation survey and monitoring instruments for routine use include:

Beta-gamma survey meters (geiger counters)
Low and high range beta-gamma ionization chamber survey meters
Neutron rem dosimeter instruments
Thermal and fast neutron detectors
Alpha scintillation counters

Airborne gaseous, particulate, and iodine samplers are also available for routine use as well as an assortment of special purpose and emergency type radiation survey instruments.

In addition to the portable radiation monitoring instruments, fixed monitoring instruments, i.e. beta-gamma count rate meters (geiger counters), are located at exits from the Controlled Area. These instruments are intended to prevent any contamination on personnel, materials, or equipment from being spread into the uncontrolled or conventional areas of the plant. Appropriate monitoring instruments are also available at various locations within the Controlled Area for contamination control purposes. Portal monitors are also utilized, as appropriate, to monitor personnel leaving the Controlled Area and to monitor persons leaving the station.

The station has permanently installed area and process radiation monitoring systems. These systems monitor airborne particulate and gaseous radioactivity, including iodine, as well as external radiation levels. The systems present an audible alarm and radiation level indication in the areas of concern, in addition to reading out in the respective control rooms.

11.2.3.6 Radio-Bioassay and Medical Programs

Counting and gamma analysis equipment for screening personnel for internal exposure are available on site, and outside services for radio-bioassay and whole body counting will be utilized as required for backup and support of this program. The plant equipment is sufficiently sensitive to detect in thyroid, lungs or whole body a small fraction of the permissible body burden for those gamma emitting radionuclides expected at a reactor facility.

No special medical examination is considered to be necessary for radiation workers whose exposure is maintained within permissible dose limits. However, a pre-employment physical is required of each prospective radiation worker to determine his health status and his ability to perform the job. Body burden scans are also made on personnel, prior to their beginning work, who have a history indicating likely internal exposure.

Medical observation and treatment are available in case of over-exposure and contamination. Physicians, a medical clinic, and hospital facilities are available for the treatment of injuries. A local physician has been retained and trained in the care and treatment of radiation injuries and facilities have been established in a local hospital for the handling and treatment of possibly contaminated patients. Health physics and chemistry personnel are responsible for the radio-bioassay program and will assist the physicians and the hospital in maintaining medical control of over-exposed or contaminated personnel.

These programs are designed to monitor and protect the health of all employees concerned, to confirm the adequacy of the radiation control methods employed at the station and to provide for the treatment of injuries.

11.2.3.7 Tests and Inspections

Complete radiation surveys are made at low, intermediate, and full power levels during the initial start-up and operation of each unit to evaluate the effectiveness of shielding and to determine the radiological status of the plant and the plant systems for radiation protection purposes. Radiation survey data is reviewed before increasing power. Process and area radiation monitoring system instruments are tested as required by procedures and by the technical specifications for reactor operation using built-in or portable calibration sources. The operation of this equipment is self-checking and alarms upon instrument failure in the respective Control Rooms. Counting Room instruments are routinely calibrated and checked frequently for proper operation. Radiation survey and monitoring instruments are calibrated at least every three months and after all repairs. Pocket chambers and self-reading dosimeters are also calibrated at least every six months when routinely assigned and worn
28. and are checked for proper operation prior to initial issuance and use, in accordance with Regulatory Guide 8.4.

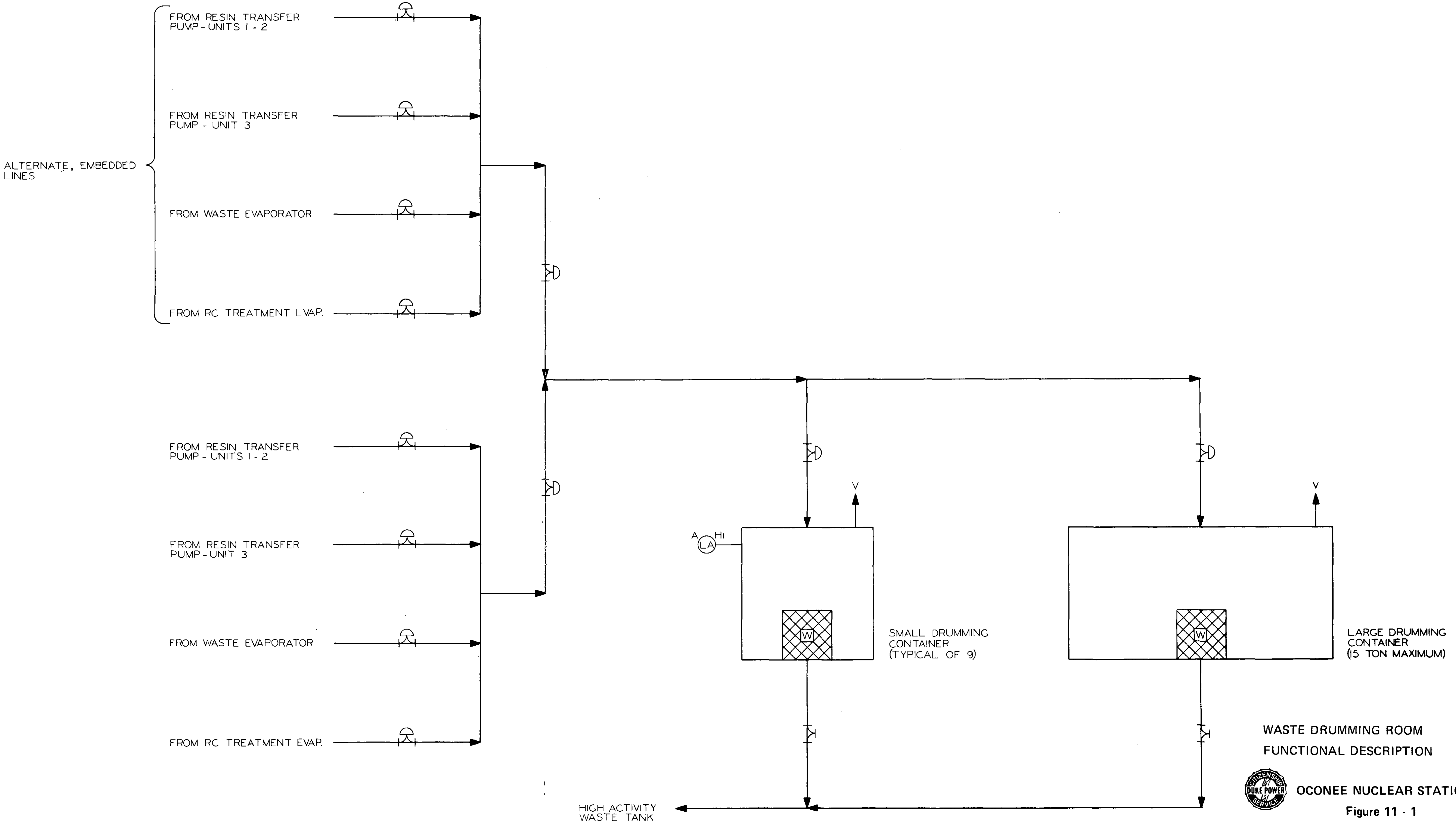
11.3 REFERENCES

(1) Frank, P. W., et al., Radiochemistry of Third PWR Fuel Material Test - X-1 Loop NRX Reactor, WAPD-TM-29, February 1957.

(2) Eichenberg, J. D., et al., Effects of Irradiation on Bulk UO_2, WAPD-183, October 1957.

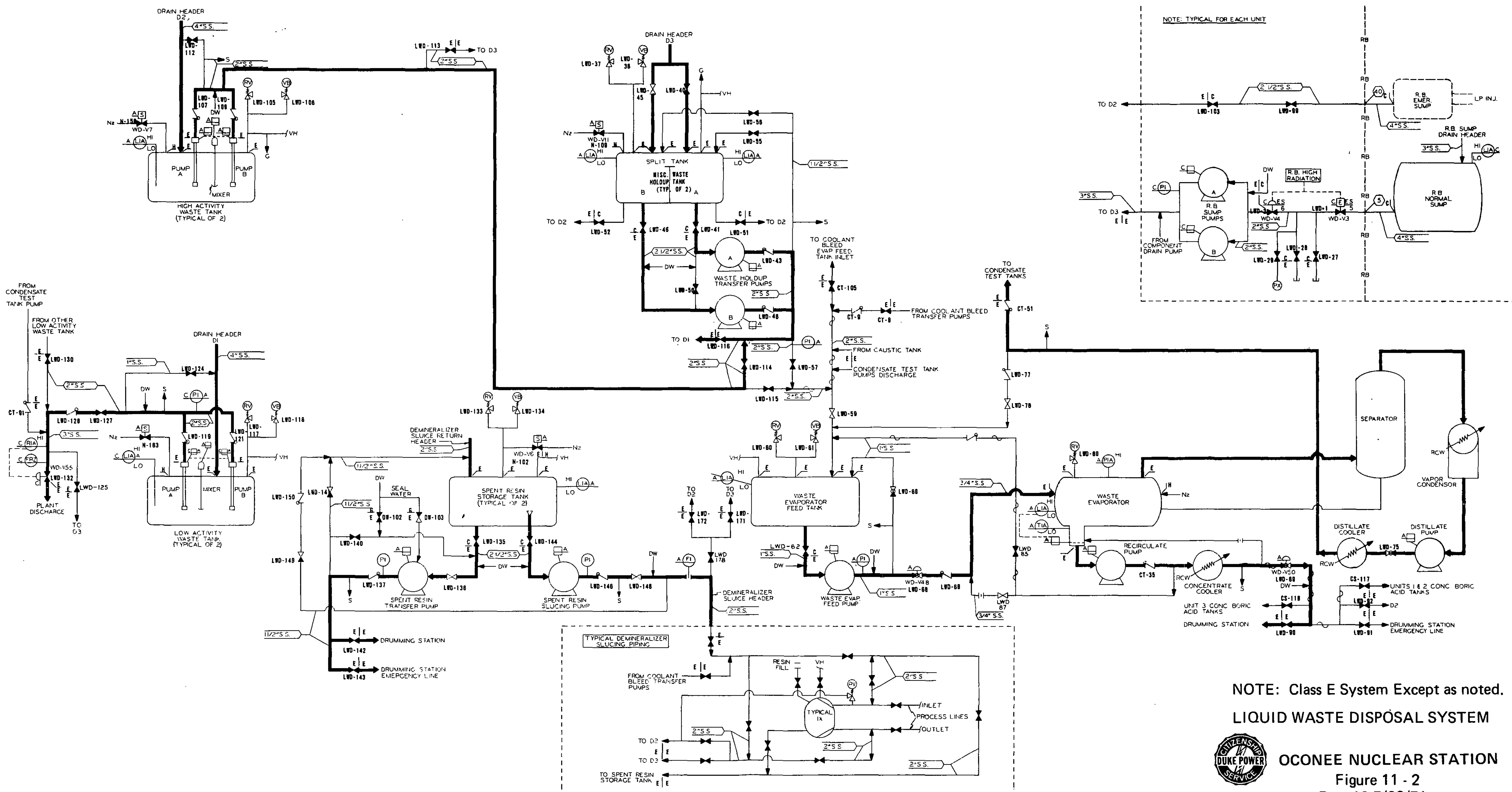
(3) Allison, G. M. and Robertson, R. F. S., The Behavior of Fission Products in Pressurized-Water Systems. A Review of Defect Tests on UO_2 Fuel Elements at Chalk River, AECL-1338, 1961.

(4) Allison, G. M. and Roe, H. K., The Release of Fission Gases & Iodines From Defected UO_2 Fuel Elements of Different Lengths, AECL-2206, June 1965.



WASTE DRUMMING ROOM
FUNCTIONAL DESCRIPTION

OCONEE NUCLEAR STATION
Figure 11 - 1



NOTE: Class E System Except as noted.

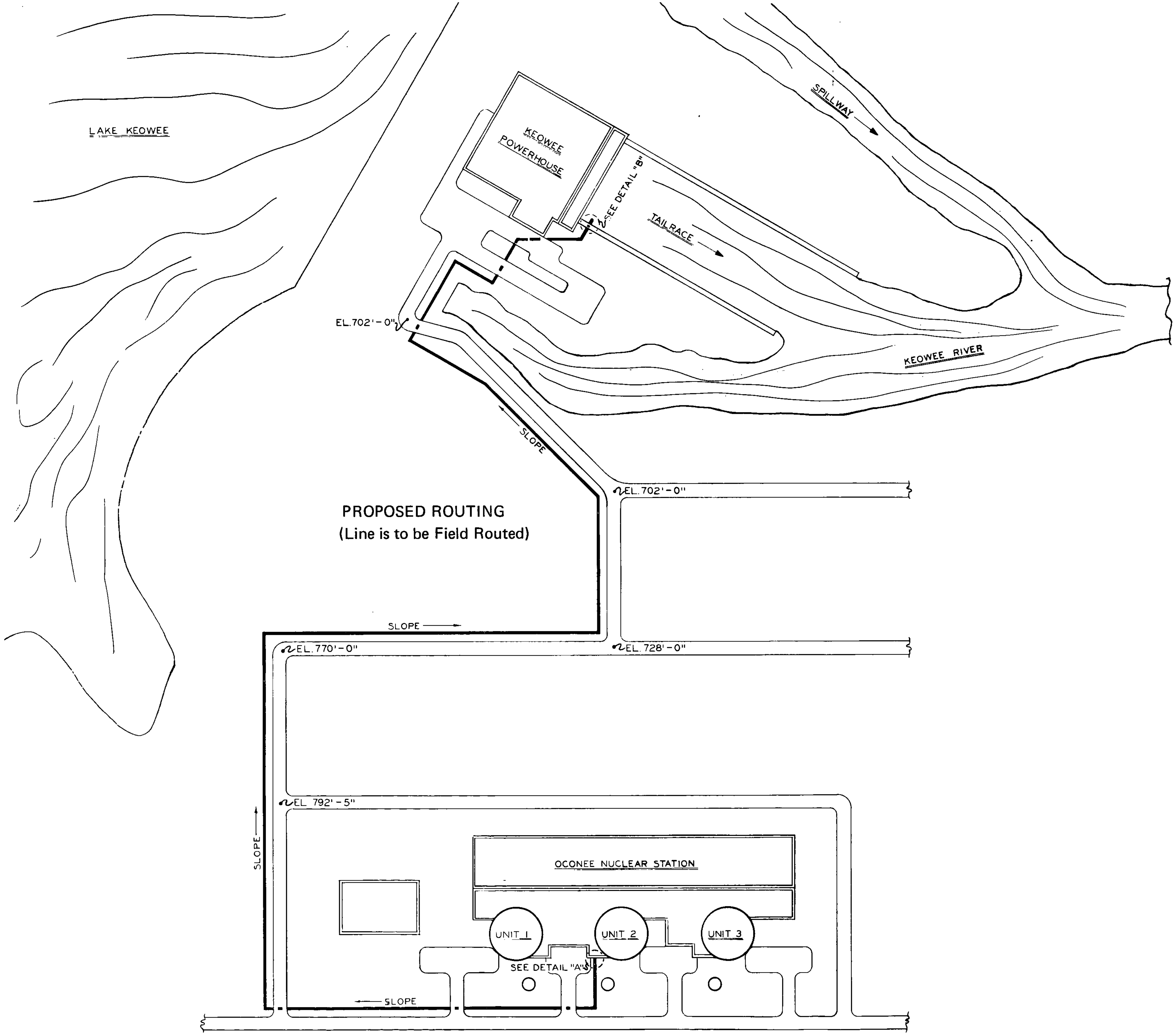
LIQUID WASTE DISPOSAL SYSTEM

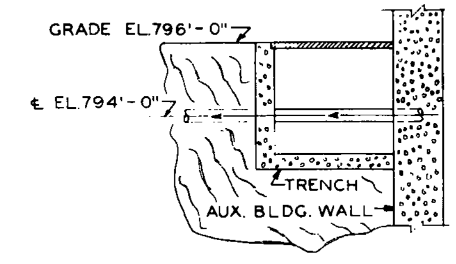
OCONEE NUCLEAR STATION

Figure 11 - 2

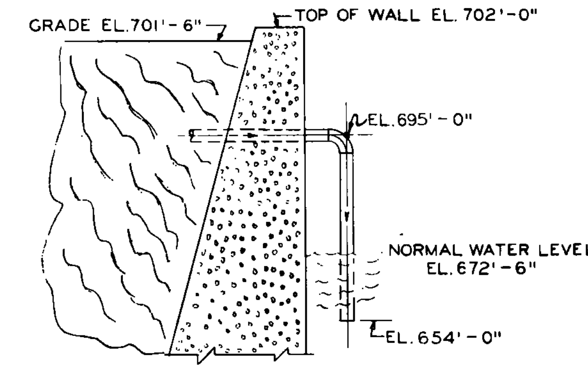
Rev. 16 7/30/71

Rev. 27 4/13/73





DETAIL "A"



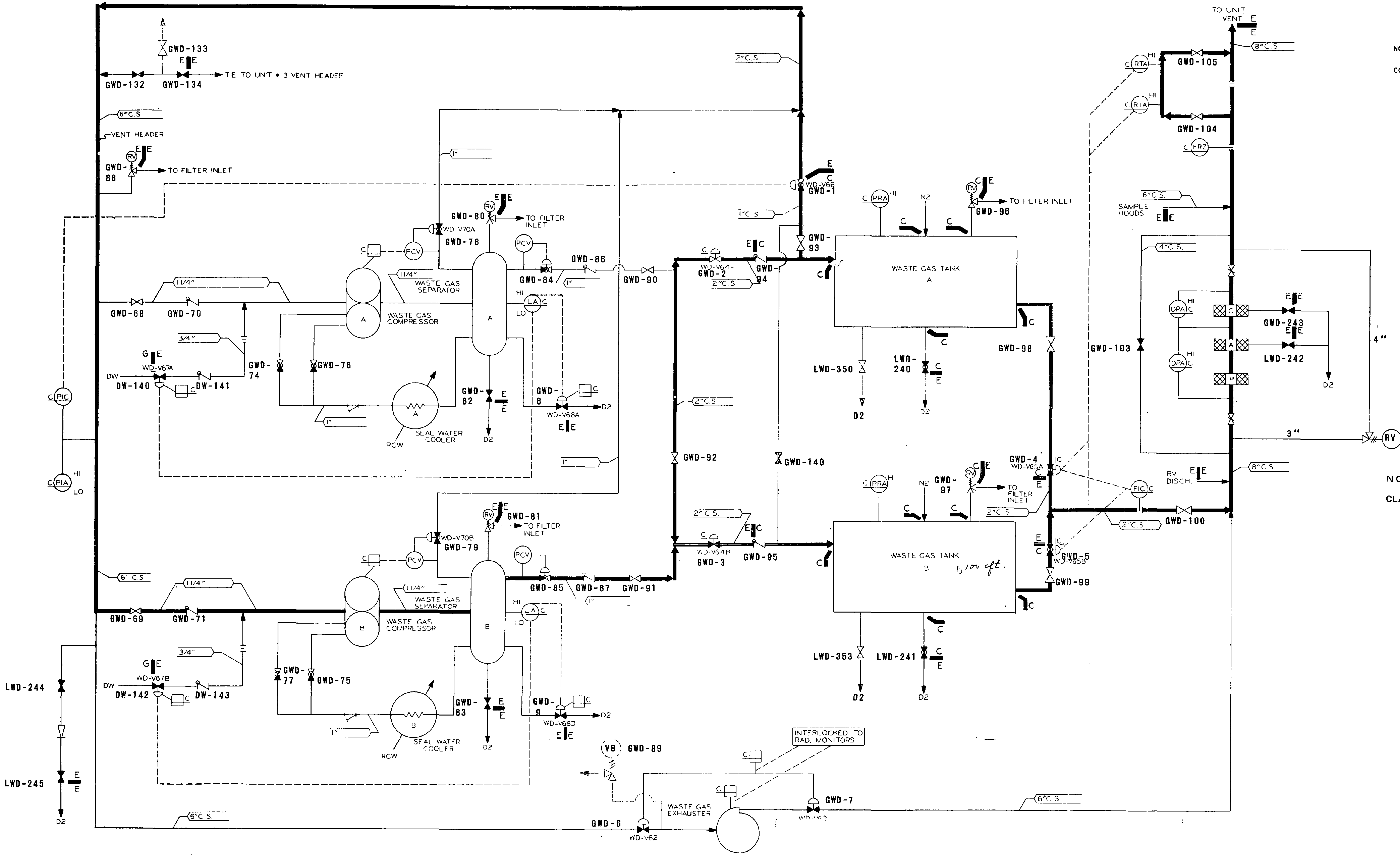
DETAIL "B"

3" LIQUID WASTE DISCHARGE

OCONEE NUCLEAR STATION

Figure 11 - 2A

(New) Rev. 7 7/9/70



NOTE 1

CONNECTIONS TO VENT HEADER

- MISC WASTE EVAPORATOR UNIT VENT
- FROM UNIT 1 QUENCH TANK VENT
- PURIFICATION DEMINERALIZER - TYP OF 3
- SF DEMINERALIZER
- SF FILTER - TYP OF 2
- LETDOWN STORAGE TANK - TYP OF 2
- DEBORATING DEMINERALIZER - TYP OF 3
- AUTOMATIC GAS ANALYZER - TYP OF 2
- MISC WASTE EVAPORATOR FEED TANK VENT
- MISC WASTE EVAP FEED TK RELIEF VALVE DISCH
- RC BLEED EVAPORATOR FEED TANK VENT
- RC BLEED EVAP FEED TK RELIEF VALVE DISCH
- MISC WASTE HOLDUP TANK VENT
- MISC WASTE HOLDUP TK RELIEF VALVE DISCH
- LETDOWN STORAGE TK GAS SAMPLE LINE - TYP OF 2
- EVAPORATOR CONDENSATE DEMINERALIZER - TYP OF 2
- RC BLEED EVAPORATOR UNIT VENT
- LOW ACTIVITY WASTE TANK VENT
- MAKEUP FILTERS UNIT 1 - TYP OF 2
- MAKEUP FILTERS UNIT 2 - TYP OF 2
- BORATED WATER RECIRCULATION PUMP
- SPENT FUEL PUMP VENTS - TYP OF 2
- SPENT FUEL COOLER VENTS - TYP OF 2
- CONTROL ROD DRIVE SEAL FILTER VENT - TYP OF 4
- FROM UNIT 2 QUENCH TANK VENT
- RC PUMP SEAL RETURN COOLER VENT UNIT 1 - TYP OF 2
- RC PUMP SEAL RETURN COOLER VENT UNIT 2 - TYP OF 2
- LP PUMPS - TYP OF 6
- LP COOLER - TYP OF 4
- RC BLEED EVAPORATOR DEMINERALIZER - TYP OF 2
- RC BLEED HOLDUP TANK VENT - TYP OF 4
- RC BLEED HOLDUP TANK RELIEF VALVE DISCH - TYP OF 4
- EVAPORATOR CONDENSATE TEST TANK VENT
- CONCT'D BORIC ACID STORAGE TANK VENT - TYP OF 2
- CONCT'D BORIC ACID STORAGE TK RELIEF VALVE DISCH - TYP OF 2
- SPENT RESIN STORAGE TANK VENT
- SPENT RESIN STORAGE TANK RELIEF VALVE DISCH
- HIGH ACTIVITY WASTE TANK VENT
- HIGH ACTIVITY WASTE TANK RELIEF VALVE DISCH
- LOW ACTIVITY WASTE TANK RELIEF VALVE DISCH
- EVAPORATOR CONDENSATE TK RELIEF VALVE DISCH
- BUILDING SPRAY PUMP VENTS - TYP OF 4

NOTE 2

CLASS E SYSTEM EXCEPT AS NOTED

Revised: Added connections to vent header.
Revised: Manual valves shown.
Revised: Valve numbers added.

GASEOUS WASTE DISPOSAL SYSTEM

OCONEE NUCLEAR STATION

Figure 11 - 3

Rev. 2 2/9/70
Rev. 4 4/20/70
Rev. 16 7/30/71

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12.0 CONDUCT OF OPERATIONS

12.1 ORGANIZATION

12.1.1 CORPORATE ORGANIZATION

The corporate structure of Duke Power Company is shown in Figure 12-1. The Executive Vice President and General Manager has corporate responsibility for nuclear station design, construction and operation. Reporting to the Executive Vice President and General Manager are two senior vice presidents, one having line responsibility for Engineering and Construction and the other for Production and Transmission.

Company involvement in nuclear power began in the early 1950's with various personnel receiving nuclear training. Selected personnel have been involved full time in nuclear projects since the mid-1950's. Duke participated in the Carolinas-Virginia Nuclear Power Associates (CVNPA), which resulted in a 17,000 kWe nuclear steam-electric unit at Parr, South Carolina. This unit, the Carolinas-Virginia Tube Reactor (CVTR), produced electricity over the period 1963 to 1967 as part of a five-year operating research program.

12.1.1.1 Corporate Functions, Responsibilities and Authorities

As Executive Vice President and General Manager, Mr. B. B. Parker has overall responsibility for corporate functions involving planning, design, construction and operation of the Company's generation, transmission and distribution facilities, as well as other staff functions. Mr. Parker was an active participant in the Savannah River Nuclear Study Group which examined the feasibility of developing power from a production reactor. Reporting to Mr. Parker are Mr. W. S. Lee, Senior Vice President, Engineering and Construction and Mr. A. C. Thies, Senior Vice President, Production and Transmission.

Mr. A. C. Thies, Senior Vice President, Production and Transmission, directs the operation of Duke's power generation facilities and the design, construction and operation of the transmission facilities. Mr. Thies graduated from Georgia Institute of Technology in 1943 with a BS in Mechanical Engineering and joined Duke's Steam Production Department in 1946. He was named Manager of Steam Production in 1963 and Assistant Vice President, Steam Production, in 1965. In 1967, he was made Vice President, Production and Operation, with the responsibility for system generation and electrical operation. In 1971, he was appointed Senior Vice President, Production and Transmission, and a Director of the Company. He served on the Technical Advisory Committee of CVNPA from 1960 and was Chairman in 1962-63. This committee represented the four owning companies in exercising direct supervision over operation of the CVTR. Mr. Thies is a former member of the Atomic Power Subcommittee of the EEI and is active in a number of industry and engineering organizations. Reporting to Mr. Thies are Mr. W. O. Parker, Jr., Vice President, Steam Production and others not having direct responsibility for nuclear station operation.

Mr. W. S. Lee, Senior Vice President, Engineering and Construction, directs the design and construction of Duke's new power generating facilities. Mr. Lee was graduated from Princeton University in 1951 with a BS in Civil Engineering and joined Duke's design staff in 1955. With design engineering experience at progressive levels of responsibility, he was appointed Vice President, Design Engineering, in 1965; a Director of the Company in 1968; and Senior Vice President, Engineering and Construction in 1971. In 1957, Mr. Lee was temporarily assigned to CVNPA to participate in selection of reactor type, cost estimating and conceptual design of the CVTR. Mr. Lee is a registered Professional Engineer in North Carolina and South Carolina, and is a member of a number of industry and engineering organizations. Reporting to Mr. Lee are Mr. W. H. Owen, Vice President, Design Engineering and Mr. R. L. Dick, Vice President, Construction.

Mr. W. H. Owen, Vice President, Design Engineering was graduated from Clemson University with a BME in 1947 and joined Duke Power in the Steam Production Department in 1948. He served in various engineering capacities at two of the Company's steam stations and was then assigned to the General Office staff of the Steam Production Department. In 1962, he was transferred to the Design Engineering Department. In 1962, he was transferred to the Design Engineering Department as Senior Engineer responsible for instrumentation and controls, and was appointed Principal Mechanical Engineer in 1966. He became Vice President, Design Engineering, in 1971. He participated in the Company's 1955 and 1966 nuclear training programs. He was a member of ANS-20, Systems Engineering Committee, and participated in writing "Nuclear Safety Criteria for Design of Stationary Pressurized Water Reactors," and is now a member of the Nuclear Standards Policy Board. Mr. Owen is a registered Professional Engineer in North Carolina and South Carolina. Mr. Owen directs the design of Duke's new power generating facilities.

Mr. R. L. Dick, Vice President, Construction, directs the construction of Duke's new power generating facilities. A 1949 graduate from North Carolina State University with a BCE, Mr. Dick joined Duke's Construction Department upon graduation. Since that time, all of his assignments have been in the area of generating station construction. In 1971, he was appointed Vice President and placed in charge of all construction work performed by the Construction Department. Since graduation, he has attended nuclear courses at North Carolina State University and the University of California and the Company's 1966 nuclear training program. He is presently Chairman of the American National Standards Subcommittee N45-3, Nuclear Quality Assurance Standards, and a member of the ASME Boiler and Pressure Vessel Code Committee's Subcommittee on Nuclear Power (SC-111). Mr. Dick is a registered Professional Engineer in North Carolina and South Carolina.

12.1.1.2 Organization for Design and Construction

12.1.1.2.1 Organization for Design

The Design Engineering Department, directed by Mr. W. H. Owen, Vice President Design Engineering, has the specific responsibility for design of structures and systems, specification of materials and equipment, and preparation of construction and installation drawings. Line responsibility for the design of electric generating stations rests with the Chief Engineers who report to Mr. Owen and direct the Civil and Environmental, Electrical, and Mechanical and Nuclear engineering divisions.

Mr. L. C. Dail, Chief Engineer, Civil and Environmental Division, was graduated from North Carolina State University with a BCE in 1949. Following design experience with the U. S. Bureau of Reclamation and the Tennessee Valley Authority, he joined Duke Power in 1953 and was appointed Principal Civil Engineer in 1959. He participated in the 1955 and 1966 nuclear training programs given at Duke and in the Nuclear Safety Program given at Massachusetts Institute of Technology in 1966. Mr. Dail is a registered Professional Engineer in North Carolina and South Carolina.

Mr. S. K. Blackley, Jr., Chief Engineer, Mechanical and Nuclear Division, was graduated from Duke University with a BSCE in 1954 and joined Duke in the Engineering Department in 1959. He was appointed Principal Mechanical Engineer in 1971 after having been engaged in civil engineering design of hydro, fossil and nuclear generating stations for more than 11 years. He has participated in siting work, environmental studies and licensing of all Duke generating stations undertaken since 1960. He participated in the 1966 nuclear training program and has been a member of Duke's review committee on fast breeder reactors. Mr. Blackley is a registered Professional Engineer in North Carolina and South Carolina.

Mr. C. J. Wylie, Chief Engineer, Electrical Division, was graduated from the University of South Carolina with a BS in Electrical Engineering, and joined Duke's Engineering Department after graduation in 1950. He was appointed Principal Electrical Engineer in 1959. Mr. Wylie was temporarily assigned in 1957 to the conceptual design phase of the CVTR, and he participated in Duke's courses in nuclear engineering in 1955 and 1966. Mr. Wylie is actively engaged in various Institute of Electrical and Electronics Engineers Technical Committees, including the Power Engineering Committee, and is Chairman of the Power Engineering Committee. He is a registered Professional Engineer in North Carolina and South Carolina.

12.1.1.2.2 Organization for Construction

The Construction Department has the responsibility for all site construction activities. The department is organized by projects and each nuclear power station has a separate project group. The Construction Department is also responsible for field testing.

12.1.1.3 Station Support Organization

Duke's Steam Production Department, headed by the Vice President, Steam Production, has direct line responsibility for fossil and nuclear station production operations. The Steam Production Department organization is shown in Figure 12-2. A General Office staff of engineers experienced in all phases of both conventional and nuclear plant operation furnishes general supervision and technical support to each of the generating stations.

Other departments of the Company are available for consultation and assistance on a day-to-day basis as required. The Design Engineering Department is available to furnish technical assistance as requested in areas such as nuclear physics, shielding, inservice inspection, environmental services and the general area of station design. Other departments which regularly supply assistance and services for the station are: Electrical Maintenance and Construction; Relay, Meters and Communications; and Construction.

Departments furnishing assistance to the station organization are permanently staffed with competent engineers and technicians with many years of experience in their respective fields. Qualifications and nuclear experience of certain key personnel in the Steam Production Department are detailed below.

Mr. W. O. Parker, Jr., Vice President, Steam Production, has direct line responsibility for all fossil and nuclear operations. Mr. Parker graduated from Illinois Institute of Technology in 1946 with a BSME and moved through a succession of jobs in Steam Production to his present position. Mr. Parker served on a technical review committee for the CVTR. He is a member of the ASME Ad Hoc Committee on Inservice Inspection and is a registered Professional Engineer. Mr. Parker participated in the 1966 Nuclear Training Program, an 80-hour course given at Duke to a group of over 80 engineers by industry specialists in the nuclear power field.

Among those reporting to Mr. Parker are Mr. P. H. Barton, Mr. R. W. Bostian, Mr. H. B. Tucker and Mr. W. A. Haller.

Mr. H. B. Tucker, Manager, Nuclear Production Division, is a 1949 EE graduate of Georgia Institute of Technology. Mr. Tucker has held various positions in the operating organization and has 17 years experience in the supervision of steam station operation including five years as a station Superintendent. He has taken the one-month course at the University of Michigan in nuclear power reactor engineering. Mr. Tucker has line responsibility for operating and maintaining the nuclear generating facilities, planning and startup of new facilities. The nuclear station managers report to Mr. Tucker.

Mr. P. H. Barton, Manager, System Operation and Maintenance, has had extensive nuclear experience in addition to 15 years of active experience in the design, preoperational checkout and operation of steam stations. A 1948 BME graduate of Clemson University and a registered Professional Engineer in North and South Carolina, Mr. Barton was Duke's representative on the CVNPA Technical Advisory Committee from 1957 to 1960, when he was transferred to CVNPA as Operation Supervisor. He completed the Reactor Operation Supervisory Program at the Oak Ridge School of Reactor Technology in the class of 1960-61; and, as a licensed senior operator, was intimately involved in the successful startup and operation of the CVTR, where he also held the position of Maintenance Engineer. After returning to Duke in 1964, he was involved in technical training for steam station personnel, selection and training of Oconee plant personnel, as well as reviewing Oconee design. He is responsible for establishing policies and procedures for operation and maintenance of nuclear and fossil generating facilities; planning and coordinating system schedules, providing support for major station maintenance, materials management, and new facility startup; and coordinating station operation with the Operating Department.

Mr. R. W. Bostian, Manager, Systems Results and Fuel Management, is a 1949 ME graduate of North Carolina State University. He has worked in various fossil stations in Duke's system since 1950 and was promoted to Superintendent of Allen Steam Station in 1962. In 1971, Mr. Bostian became System Production Engineer and was assigned to the General Office staff. Mr. Bostian was promoted in 1973 to Manager, Fuels Utilization, where he is responsible for the management of fossil and nuclear fuels.

Dr. W. A. Haller, Manager, Environmental, Technical Services and Projects, joined Duke Power Company early in 1974 following extensive experience in the nuclear industry. He obtained a B.S. degree in chemistry from LaSalle College (1960) and an M.S. degree in nuclear chemistry from the University of Notre Dame in 1963. From 1963-65, while employed by General Electric and 1965-71 while employed by Battelle Northwest, he gained experience in the chemical and environmental aspects of operating reactors at the Hanford NRC reservation. During this same period, he completed course and experimental work at the Hanford Graduate Center which led to the granting of a Ph.D. in Nuclear Chemistry by Washington State University. During 1971-73 Dr. Haller was employed by Babcock and Wilcox as a group leader/acting section manager in its Chemical and Radiological Department.

Mr. Lionel Lewis, System Health Physicist, joined Duke in 1967, following six years at the Carolinas Virginia Tube Reactor, (CVTR), as Health Physicist and Safety Coordinator. Prior to joining CVNPA in 1961, he held various positions in the health physics field with Combustion Engineering, the Martin Company, and Brookhaven National Laboratory. Mr. Lewis received a BA in Pre-Medical Sciences from the University of Vermont in 1949 and obtained an MS in Radiological Biophysics from the University of Rochester in 1955. He was certified in Health Physics by the American Board of Health Physics in 1961. Mr. Lewis' present position as System Health Physicist includes the establishment and direction of the Health Physics and Environmental Surveillance Programs.

12.1.2 OPERATING ORGANIZATION

12.1.2.1 Station Organization

The organization of the Oconee staff generally follows the pattern already proven to be successful in Duke's conventional steam stations. The underlying philosophy is that the station staff is fully capable and equipped to handle all situations involving safety of the station and public. The organization of the station staff is shown in Figure 12-3.

12.1.2.2 Personnel Functions, Responsibilities and Authorities

The functions and responsibilities of the station supervisory staff are described in the succeeding paragraphs.

(a) Station Manager

The station Manager reports to the Manager, Nuclear Production, and has direct responsibility for operating the station in a safe, reliable and efficient manner. He is responsible for protection of the station staff and the general public from radiation exposure and/or any other consequences of an accident at the station. He bears the responsibility for compliance with the facility operating license.

(b) Operating Superintendent

The Operating Superintendent has the responsibility for directing the actual day-to-day operation of the station. In the event of the absence of the station Manager, the Operating Superintendent, if so designated, assumes the responsibilities and authority of the station Manager.

(c) Operating Engineer

An Operating Engineer assists the Operating Superintendent in directing station operation.

(d) Shift Supervisor

A Shift Supervisor is responsible for the actual operation of the station on his assigned shift. He directs the activities of the operators on his shift and must be cognizant of all maintenance activity being performed while he is on duty. The Shift Supervisor on duty has both the authority and the obligation to shut down a unit if, in his opinion, conditions warrant this action. When all three units, or Unit 3 and one of the other units are in operation, the Shift Supervisor will, although having overall responsibility for total station operation, have the responsibility for operations from one of the Control Rooms and the Assistant Shift Supervisor has responsibility for operations from the other.

(e) Assistant Shift Supervisor

The Assistant Shift Supervisor, upon assignment, may assume the responsibilities of the Shift Supervisor. His normal duties involve supervision over various operating tasks as assigned. He will hold a Senior Reactor Operator License. The Assistant Shift Supervisor's responsibilites for station operations during multiple unit operation is described in 12.1.2.2.(d).

(f) Technical Services Superintendent

The technical services group at Oconee is headed by a Technical Services Superintendent. He reports directly to the station Manager and has reporting to him the Performance Engineer, the Chemist, and the Health Physics Supervisor.

(g) Performance Engineer

The Performance Engineer directs data gathering and evaluation in the areas of equipment and plant performance. Specifically included in this are core physics and core performance, from both nuclear and thermal-hydraulic considerations. He assists in setting up fuel shuffling patterns and participates in other phases of fuel management.

(h) Health Physics Supervisor

The Health Physics Supervisor has the responsibility for conducting the Health Physics Program for Oconee. His duties include training of personnel in use of equipment, control of radiation exposure of all personnel, continuous determination of the radiological status of the plant, surveillance of all radioactive waste disposal operations, conduting the Environmental Monitoring Program, and maintaining all required records. He has direct access to the station Manager in matters concerning any phase of radiological protection.

(i) Chemist

The Chemist is responsible for overall chemistry and radiochemistry requirements, with special emphasis on primary and secondary system water chemistry.

(j) Maintenance Superintendent

The Maintenance Superintendent reports to the station Manager and is responsible for organizing and conducting preventive maintenance and repairs of all equipment at Oconee with the exception of certain electrical instrumentation applying only to the generators and switching station, which are maintained on a system basis by other departments. Reporting to the Maintenance Superintendent are the Planning Engineer, Instrument and Control Engineer, and Maintenance Engineer.

(k) Maintenance Engineer

The Maintenance Engineer has responsibility for preventive maintenance and repair of all mechanical and non-instrument related electrical equipment.

(l) Instrument and Control Engineer

The Instrument and Control Engineer has responsibility for preventive maintenance and repair of all instrumentation and controls at Oconee with the exception of certain electrical instrumentation applying only to the generators and switching station, which are maintained on a system basis by other departments. He will also supervise computer maintenance.

(m) Planning Engineer

The Planning Engineer reports to the Maintenance Superintendent and is responsible for the coordinating, planning and scheduling of all preventive maintenance and repairs of all equipment at Oconee with the exception of certain electrical instrumentation maintained by other departments. He is also responsible for materials management at Oconee. Reporting to him are the Planners, Stores Supervisor and Preventive Maintenance Supervisor.

12.1.2.3 Shift Crew Composition

The shift crew consists of a Shift Supervisor, an Assistant Shift Supervisor, Control Room operators, Assistant Control Room operators and utility operators. The composition of the shift crew and NRC licenses required are specified in Technical Specification 6.1.

12.1.3 QUALIFICATIONS OF STATION PERSONNEL

12.1.3.1 Minimum Qualification Requirements

The required training and qualifications of key personnel in the operating staff of Oconee are summarized below. The educational and experience backgrounds of all candidates for positions on the plant staff are carefully evaluated and training programs are initiated where required. The result is an exceptionally well-trained and well-qualified staff for operation.

The following minimum qualifications for key supervisory positions are consistent with the requirements of ANSI N.18.1-1971.

Position	Minimum Requirements for Position
Manager	1. Graduate in Engineering, a related physical science or equivalent experience (1 year's experience for 1 year of college education)
	2. Minimum of 6 years of responsible power plant experience, including a minimum of 3 years nuclear experience
	3. Meets training and experience requirements for Senior Reactor Operator License. (But will not be required to have an active license)
Operating Superintendent	1. Graduate in Engineering, a related physical science or equivalent experience
	2. Minimum of 6 years responsible power plant experience including at least 3 years nuclear experience
	3. Hold, or be qualified for and acquire, a Senior Reactor Operator License
Operating Engineer	1. Same as Operating Superintendent
	2. Minimum of 2 years plant experience including at least 1 year nuclear experience
	3. Same as Operating Superintendent
Technical Services Superintendent	1. Graduate in Engineering or a related physical science or equivalent in experience

Position	Minimum Requirement for Position
Technical Services Superintendent (Cont'd)	2. Minimum of 4 years in a technically related position, including at least 1 year nuclear experience
Performance Engineer	1. Graduate in Engineering, a related science or equivalent experience and training. (If the Technical Services Superintendent is not a graduate in Engineering or related physical science, then the Performance Engineer shall be.)
	2. Minimum of 2 years nuclear experience
Chemist	1. Graduate chemist of Chemical Engineer or equivalent experience
	2. Minimum of 1 year experience in radiochemistry
Health Physics Supervisor	1. High school graduate with 2 years technical training in Health Physics
	2. Three years experience in Health Physics
Maintenance Superintendent	1. High school graduate with 2 years technical school or equivalent training or experience. (1 year experience for 1 year technical school)
	2. Minimum 5 years experience in plant maintenance, including at least 1 year of nuclear plant maintenance
Maintenance Engineer	1. High school graduate with 2 years technical school or equivelent training or experience (1 year experience for 1 year technical school
	2. Minimum of 1 year experience in nuclear plant maintenance
Instrument & Control Engineer	1. Graduate in Engineering or a related physical science or equivalent in experience

Position	Minimum Requirement for Position
Instrument & Control Engineer (Cont'd)	2. Minimum of 1 year experience with power plant instrumentation and controls, ½ year of which is with nuclear instrumentation and controls
Shift Supervisor	1. High school graduate with 2 years technical school or equivalent experience
	2. Minimum of 2 years power plant experience, 1 of which will be nuclear experience
	3. Hold, or be qualified for and acquire, a Senior Reactor Operator License
Assistant Shift Supervisor	(Same as Shift Supervisor)
Control Operator	1. High school graduate or equivalent
	2. Minimum of 2 years power plant experience, 1 of which will be nuclear experience
	3. Hold, or be qualified for and acquire, a Reactor Operator License
Assistant Control Operator	1. High school graduate or equivalent
	2. Minimum of 2 years power plant experience, 1 year of which shall be nuclear experience
	3. Hold, or be qualified for and acquire, a Reactor Operator License

12.1.3.2 QUALIFICATIONS OF STATION PERSONNEL

Mr. J. Ed Smith, Manager, has had extensive nuclear experience as well as a background of steam station operation experience. He joined Duke following graduation in 1949 with a BME from N. C. State College. He held positions of Testman, Shift Supervisor and Plant Engineer in Duke steam stations prior to completing the Reactor Operation Supervisory Program at the Oak Ridge School of Reactor Technology in the class of 1960-61. Assigned to the CVTR, he held positions of increasing responsibility as Shift Supervisor Engineer, and Plant Supervisor. He was intimately involved in the startup, operation and de-commissioning of the CVTR, and held a senior reactor operators license. As plant Supervisor, he directed the maintenance and operations group and was responsible for operator training. In November 1967, Mr. Smith was named Superintendent of Oconee Nuclear Station, so that he has been personally involved in reviewing the plant design and layout and in the selection of his operating staff. Mr. Smith has a MNE Degree from the University of South Carolina, which he received in 1967.

Mr. James W. Hampton, Director, Administrative Services Group, is a BSME graduate of Clemson College in 1958, has held positions as Junior Engineer, Assistant Plant Engineer and Plant Engineer at Duke steam stations. As Plant Engineer for four years at Lee Steam Station, he was responsible for turbine-generator and switchyard operation. Mr. Hampton was named Assistant Superintendent of Oconee in 1968, following completion of one year of graduate study in Nuclear Engineering at N. C. State University along with three other members of the Oconee staff. He and other Oconee supervisors conducted classes and study for the "cold license" trainees full time for the six-month period immediately prior to the six-month work experience at an operating reactor. Mr. Hampton holds a senior reactor operator license for Oconee Units 1, 2 and 3. Mr. Hampton was named Director of Administrative Services at Oconee Nuclear Station in September, 1974.

Mr. Loyd E. Schmid, Superintendent of Operations, has over 19 years of reactor operations experience. He joined Duke in 1967 following six years at the CVTR in the operations group. His CVTR experience in positions of increasing responsibility as Reactor Technician, Shift Supervisor, Relief Supervisor, Operation Supervisor, and Assistant Operating Engineer has involved all shift operations duties associated with the safe and proper operations of the reactor and other plant systems. He held one of the first "hot licenses" and a senior operator license and played a key role in training CVTR personnel for their operating licenses. His reactor operations experience prior to joining CVNPA in 1961 was at the National Reactor Testing Station in Idaho, where he served two years as Reactor Technician and three years as Reactor Operator. Mr. Schmid is a licensed senior reactor operator on Oconee Nuclear Station's Units 1, 2 and 3.

Mr. William M. Harris, Operating Engineer, joined Duke in 1952 and gained experience in a large fossil fuel generating plant as a laboratory technician for five years and as an operator for eleven years. Mr. Harris entered Duke's nuclear training program in 1968 and obtained a Letter of Certification on the Saxton Nuclear Experimental Reactor in 1970. As a Control Operator, Mr. Harris participated in the training program and pre-operational testing at Oconee. Mr. Harris is a licensed Senior Reactor Operator on Oconee Units 1, 2 and 3.

Mr. James Norman Pope, Operating Engineer, joined Duke Power Company in 1965 and gained experience in a large fossil fuel generating plant as an operator prior to entering Duke's nuclear training program in 1968. Mr. Pope participated in the training program and preoperational testing at Oconee in positions of increased responsibility prior to being named Operating Engineer. Mr. Pope is a licensed Senior Reactor Operator on Oconee Units 1, 2 and 3.

Mr. Henry R. Lowery, Shift Supervisor, joined Duke in 1963 and gained experience as a hydroelectric plant operator prior to entering Duke's nuclear training program in 1968. Mr. Lowery participated in the training program and preoperational testing at Oconee in positions of increased responsibility prior to being named Shift Supervisor. Mr. Lowery is a licensed Senior Reactor Operator on Oconee Units 1, 2 and 3.

Mr. James T. Campbell, Shift Supervisor, joined Duke Power Company in January 1966 and has progressed through positions of increased responsibility prior to being named Shift Supervisor in February 1974. Mr. Campbell is a licensed Senior Reactor Operator on Oconee Units 1, 2 and 3.

Mr. William Mack McClain, Shift Supervisor, joined Duke in 1952 and progressed through positions of increased responsibility at large fossil fuel generating plants prior to being named Shift Supervisor in 1970. Mr. McClain participated in Duke's nuclear training program beginning in 1968 and obtained a Letter of Certification on the Saxton Nuclear Experimental Reactor in 1969. Mr. McClain is a licensed Senior Reactor Operator on Oconee Units 1, 2 and 3.

Mr. Ned Fincher Edwards, Shift Supervisor, joined Duke in 1967 and has progressed th-ough positions of increased responsibility prior to being named Shift Supervisor in February 1974. Mr. Edwards has participated in the training program and preoperational testing at Oconee. Mr. Edwards is a licensed Senior Reactor Operator on Oconee Units 1, 2 and 3.

Mr. George A. Ridgeway, Shift Supervisor, joined Duke Power Company in June 1966 and has progressed through positions of increased responsibility prior to being named Shift Supervisor in February 1974. Mr. Ridgeway is a licensed Senior Reactor Operator on Oconee Units 1, 2 and 3.

Mr. Barry V. Earnhardt, Shift Supervisor, joined Duke Power Company in March 1967 and has progressed through positions of increased responsibility prior to being named Shift Supervisor in November 1974. Mr. Earnhardt is a licensed Senior Reactor Operator on Oconee Units 1, 2 and 3.

Mr. Harold W. Morgan, Shift Supervisor, joined Duke Power Company in December 1969 and has progressed through positions of increased responsibility prior to being named Shift Supervisor in October 1975. Mr. Morgan is a licensed Senior Reactor Operator on Oconee Units 1, 2 and 3.

Mr. T. D. Patterson, Shift Supervisor, joined Duke Power Company in July 1967 and has progressed through positions of increased responsibility prior to being named Shift Supervisor in October 1975. Mr. Patterson is a licensed Senior Reactor Operator on Oconee Units 1, 2 and 3.

Mr. Ollie S. Bradham, Superintendent of Maintenance, has over 22 years experience in the nuclear field. He joined Duke in 1967 following six years at the CVTR. As the CVTR Instrument/Electrical Supervisor, he was responsible for the safe and accurate performance of the nuclear instrumentation and control systems. In addition to the supervisory and technical work normally associated with this position, Mr. Bradham was involved in several major modifications to the CVTR, including conversion of Control Rod Drive amplifiers from tube type to transistorized type, and the design and fabrication of amplifiers and power supplies for the Source Range Nuclear System. His experience from 1953 to 1961 at the Savannah River Plant as Instrument Technician included work in electronic and pneumatic instrumentation in the reactor area and technical labs, and formal training in electronic and pneumatic instruments. Since joining Duke's Steam Production Department, Mr. Bradham has worked closely with Duke's Steam Engineering Department in the design and review of Oconee instrumentation and control systems.

Mr. J. M. Davis, Planning Engineer, received his BS Degree in Chemical Engineering from Clemson University in 1970, and joined Duke Power Company in October 1970. He was assigned to Marshall Steam Station prior to being transferred to Oconee Nuclear Station in July 1973.

Mr. Danny Thompson, Maintenance Engineer, has a background of over 17 years in the electrical maintenance field, 12 of these years in maintenance of Duke Power Company's steam stations. His experience includes work as a mechanic and machinist at the steam stations.

Mr. Roy C. Adams, Instrument and Control Engineer, has 21 years experience in this field. He joined Duke in 1967 following five years at the CVTR. Prior to coming to the CVTR in 1962, Mr. Adams accumulated nine years of electronic and pneumatic experience at the Savannah River Plant. Since joining Duke, he has served as an Instrument and Control Technician at Lee Station and Instrument and Control Supervisor at Oconee Nuclear Station.

Mr. Robert M. Koehler, Superintendent of Technical Services, obtained a BSME from N. C. State in 1964. He served in the U. S. Army's Corps of Engineers for two years prior to joining Duke as a Junior Engineer in plant operation at Marshall Steam Station. Mr. Koehler was named Performance Engineer after receiving a Professional Degree in Nuclear Engineering from N. C. State University in 1968. Mr. Koehler has been involved in Oconee operator training and review of core physics test specifications and procedures. In addition to Performance Engineer, Mr. Koehler has served in positions of increased responsibility, and was named Superintendent of Technical Services in September 1974.

Mr. Charlie T. Yongue, Health Physics Supervisor, has 13 years of practical experience in Health Physics. His experience began with the CVTR facility, 1963 to 1967, which encompassed both health physics and operations. His experience as a Reactor Operator involved completion of the Reactor Operators Training Course and considerable on-the-job training prior to receiving the Operator License. As a Health Physics Technician, he was involved in all related activities of the Health Physics Program at CVTR. Mr. Yongue joined Duke Power in 1967 and progressed through positions of increased responsibility prior to being named Health Physics Supervisor in October 1975.

Mr. Richard L. Wilson, Performance Engineer, obtained a BSNE from N. C. State University in 1962 and a MSE (nuclear engineering) from the University of Michigan in 1970. He worked in the Nuclear Power Division of the Charleston Naval Shipyard for 18 months and served in the U. S. Navy for four years, receiving training in the nuclear power program and serving in the engineering department on board a destroyer. Mr. Wilson joined Duke in May 1970 and was named Performance Engineer in April, 1971. Mr. Wilson has been involved in Oconee operator training, review and preparation of core physics test procedures, and developing cleaning requirements for the primary system components and piping.

Mr. Donald C. Smith, Chemist, received his BS Degree in Chemistry in 1966 from Memphis State University. Mr. Smith joined Duke Power Company's Oconee Nuclear Station in 1972 and has progressed through positions of increased responsibility prior to being named Chemist in January 1974.

12.2 TRAINING PROGRAM

12.2.1 REQUALIFICATION PROGRAM

The requalification program for the Oconee Nuclear Station is designed to maintain and demonstrate the continued competence of all licensed personnel. This program is conducted on an annual basis and includes a comprehensive exam, formal requalification lectures, on-the-job training. Simulator training will be given at least every two years. The program is implemented so as to minimize scheduling difficulties that are incurred by site management.

12.2.1.1 Program Content

Annually all SRO and RO licensed personnel take a written examination which closely parallels the NRC written examination. On the basis of the examination results, any operator failing to make a grade of 80 percent on any category is required to participate in requalification lectures. Any operator scoring less than 70 percent overall is removed from licensed duties and placed in an accelerated requalification program and subsequently participates in all scheduled requalification lectures.

Certain licensed personnel, in the performance of their normal duties, may be very much involved with one or more of the areas covered in classroom lectures. These individuals are not required to attend the applicable classroom lectures.

12.2.1.2 Practical Reactor Operation

In order to insure the continued proficiency of licensed operators in meeting all operating situations, on-the-job training plays a major role in the Oconee Requalification Program. Technical Specifications, operating procedures and emergency procedures are reviewed on shift according to a formal schedule. Short oral and/or written quizzes are used to demonstrate the effectiveness of this review. Changes to operating procedures, facility design changes and revisions to Technical Specifications are reviewed on shift also.

All licensed operators participate, to the maximum extent possible, in unit evolutions involving reactivity changes. A licensed Reactor Operator that performs or a licensed Senior Reactor Operator that performs or directs any one of the manipulations listed below meets the requirements of 10CFR55, Appendix A, Section 3(a):

(a) Critical approach from subcritical on source range instrumentation to critical on the intermediate range instrumentation.

(b) Any power level change (increased or decreased) of 10 percent of rated power or greater with control rods in manual.

(c) Reactor shutdown from critical at 15 percent of rated power to subcritical shutdown on source range instrumentation.

(d) Boration or deboration during critical operation.

(e) Operation of refueling bridge to change core geometry during refueling.

During the two-year license term, a minimum of ten such evolutions shall be conducted by each licensed Reactor Operator and either conducted or directed by each licensed Senior Reactor Operator with no more than three (3) such evolutions being any combination of Items (d) and (e) above.

Backup licensees (staff engineers, General Office personnel) acquire four hours/month on shift assigned the duties of Assistant Shift Supervisor or Assistant Control Operator, which are positions requiring a Senior Reactor Operator's and Reactor Operator's License respectively, at this facility. The assignment to Assistant Shift Supervisor or Assistant Control Operator is consistent with the license held by the backup licensee.

12.2.1.3 Simulator Training

All licensed personnel participate in a one-week (40 hour) simulator training course, consisting of 20 hours classroom and 20 hours simulator training during the two-year license term. This training includes operation during emergency or abnormal conditions including the following:

(a) Reactor Startup
(b) 15-100 Percent PWR Maneuver
(c) 100-15 Percent PWR Maneuver
(d) Power Operations (Manual and Auto)
(e) Unit Shutdown/Cooldown
(f) Reactor Trip
(g) Turbine Trip
(h) Steam Line Rupture
(i) RC System Leaks

Control manipulations at the simulator are credited in the same manner as manipulations on the Oconee units.

12.2.2.4 Evaluations

(a) The performance and competency of Reactor Operators and Senior Reactor Operators is evaluated by the annual written exam as well as with personal evaluations from supervisors and training staff personnel.

(b) Biannually each Shift Supervisor submits a report to the Operating Engineer, evaluating the performance of each licensed man under his supervision during normal and abnormal operating conditions.

(c) The performance of Reactor Operators and Senior Reactor Operators is evaluated by the simulator training staff following simulator training.

(d) The Operating Superintendent reviews these reports biannually. On the basis of the evaluations, the Operating Superintendent can recommend special training classes and removal from license duties if necessary. Prior to the license renewal date, the Operating Superintendent reviews each licensed operator's training record. Based on this evaluation he makes recommendations for license renewal or specialized training prior to license renewal.

12.2.2 REPLACEMENT TRAINING

Replacement raining is conducted to fill vacancies and prepare individuals for increased responsibility in the supervisory, technical and maintenance staff. A comprehensive work-study training and advancement program is designed to supply technically qualified and experienced personnel to meet all the job requirements. Replacement personnel receive training in areas in which they are not already qualified by reason of experience, technical training, and/or on-the-job training. In the overall program, the replacement personnel receive training comparable to that received by the initial staff. This ensures that the required level of proficiency is maintained in all operating positions.

12.2.3 RECORDS

Records are maintained on each employee's participation in training activities. It is the Training Supervisor's responsibility to see that training records of individuals are accurate and retrievable. The contents of an individual's file include the following:

(a) Job Qualifications
(b) Job-Related Experience
(c) Training Courses Participated In

The station training programs are evaluated by one or more of the following methods:

(a) On-the-job observation made by supervisor and simulator personnel
(b) Written examinations
(c) Oral examinations
(d) Independent audit of the training course
(e) Records of the number and type of reactivity changes

In addition, the Operator Training Program's effectiveness is evaluated by administering simulated NRC license examinations. The results of these examinations indicate the training program's effectiveness and are a major factor in determining whether the trainee takes the NRC examination. Permanent records are not maintained during the initial selection stages of the training program.

12.3 EMERGENCY PLANNING

12.3.1 OBJECTIVES

The Emergency Plan for the Oconee Nuclear Station is established for the protection of life and property in all emergency and accident situations. It particularly applies to those situations involving radiation and contamination where the health and safety of station personnel and the general public may be involved, in accordance with NRC regulations; but in addition it also includes other general industrial emergency and accident conditions. The Emergency Plan is a coordinated effort involving station personnel, facilities and equipment; the emergency resources and capabilities of Duke Power Company; outside emergency services; and various local, state and federal agencies having appropriate jurisdiction or concern for the public health and safety. (See Figure 12-4)

The Emergency Plan is compatible with facility design features, site layout, and site location with respect to such considerations as access routes, surrounding population distributions, and lake and land use.

12.3.2 EMERGENCY CONDITIONS CONSIDERED

The emergency and accident situations covered in the Emergency Plan include the following situations, among others, that may affect the site and the surrounding area: fire, vehicular accidents, natural disasters, medical injury or illness, radiation and contamination incidents and reactor accidents.

Radiological emergency situations, if they occur at all, are expected to be highly localized and only station property and station personnel are subjected to any major hazard. Members of the public are also within the Exclusion Area at various times (highway and traffic, Visitors Center, boating and recreation on Lake Keowee, etc.). In case of a major accidental release of activity, the general public and property in locations beyond the Exclusion Area may also be affected. The plan also includes the protection of all these groups.

12.3.3 PROTECTIVE MEASURES AND PREPARATIONS

(a) The Availability of Warning Systems:

This includes process and area radiation monitoring instrumentation which alarms both the control rooms and in the locally affected area within the station. The unit vent radioactivity monitors are shielded and have extended ranges and are threfore capable of monitoring any release that, by calculation, might be expected to result from a major reactor accident. Also included is an electric signal horn that has been strategically placed in the Exclusion Area to serve as a warning and evacuation signal for the general public.

(b) Reliable Communications:

(1) For communication within the station:

These include the station telephone (with connections throughout the plant as well as between the control rooms, the Visitors Center, and the Keowee Hydro Plant) and the public address system.

(2) For communication outside the station:

These include the Bell Telephone; microwave communications with the Duke Power Company General Office and with the Duke Power Company Spartanburg Dispatcher; two-way radio communications between the Unit 1-2 control room, the sub-station at Central, S. C., and to an emergency vehicle, and a boat.

(c) Established Outside Emergency Services:

These services or agencies are the South Carolina State Board of Health and Environmental Control, Division of Radiological Health; the Civil Preparedness Agency for Oconee County; the Civil Preparedness Agency for Pickens County; the Sheriff's Department of Oconee County, the Sheriff's Department of Pickens County, the South Carolina Highway Patrol; the Oconee County Rural Fire Control System; the Oconee Memorial Hospital and the Memorial Clinic; the NRC Emergency Radiological Monitoring Team.

Many of these agencies have considerable resources at their disposal for handling emergency situations and several have drawn up plans of their own for handling county disaster situations of one type or another.

(d) Provision for the Treatment of Medical and Radiation Injury and for Decontamination:

A first aid room has been established within the Restricted Area of the station. This room is particularly equipped for the decontamination of injured persons. A physician at Memorial Clinic in Seneca, South Carolina, will serve as the company doctor for the Oconee Nuclear Station. He has received training in medical planning and care in radiation accidents at an NRC sponsored seminar at Brookhaven National Laboratory. Also, hospital personnel have been trained and facilities have been established at the Oconee Memorial Hospital in Seneca, S. C. for the treatment of radiation injuries and overexposures and for the handling and care of possibly contaminated patients. This training will be repeated periodically, to maintain the ability of the hospital staff to handle any radiation emergency situation and to update the procedures and facilities required.

(e) Training of Personnel in Emergency Procedures:

Personnel on each shift have been trained and qualified in Health Physics, radiation monitoring, first-aid and fire fighting. In addition to their responsibility to familiarize themselves with the Emergency Plan, drills are held routinely as specified in Technical Specifications to develop and

and maintain the competence of operating personnel in handling of each category of emergency situation. Off-site agencies involved in the Emergency Plan are contacted on an annual basis to verify the correctness of telephone numbers.

(f) Emergency Equipment:

Equipment for handling a radiation and contamination accident or a reactor accident include high range radiation monitoring instruments and dosimeters, self-powered air sampling equipment, supplies of protective clothing and respiratory protective equipment. These are placed in strategic locations and replaced and serviced as necessary. Also included are an emergency vehicle and a boat. These are equipped with two-way radios for communication with the control room and are intended for emergency radiation monitoring and sampling, and for warning the public, both within and outside the Exclusion Area.

12.3.4 ORGANIZATION FOR COPING WITH EMERGENCY SITUATIONS: RESPONSIBILITIES AND DELEGATION OF AUTHORITY

All emergency or accident situations at the Oconee Nuclear Station are handled by the Shift Supervisor on duty. He is informed immediately of all emergencies and assumes control of the situation; he alerts and warns station personnel and others, takes necessary remedial on-site action, obtains necessary outside aid and notifies management. The Shift Supervisor continues this responsibility until he is relieved by the Oconee Station Manager appointing another responsible person to act in this capacity. The person who relieves the Shift Supervisor coordinates with outside agencies. This then enables the Shift Supervisor to devote his full attention to remedial measures within the station.

The normal operating crew is staffed and qualified to perform all actions that are necessary to institute immediate protective measures and to implement the Emergency Plan. The normal operating crew is designated as the responsible group for such actions. The operating crew is augmented by all other station personnel (maintenance, technical support, etc.) as required and as called upon for assistance.

Additional emergency assistance from the Steam Production Department General Office staff (Health Physics, Operations and Maintenance, etc.) and other departments within the company, as well as the full resources of Duke Power Company and backup from corporate management, are available to the Oconee Nuclear Station. Lines of authority, responsibilities and functions, are as normally established within the Duke Power Company organization. Personnel and groups from other departments within the company responding to a request for emergency aid at the station or in the surrounding area work through the local District Manager, under the direction of the Oconee Station Manager.

In the event of a major reactor accident situation, an Emergency Control Center will be established in the town of Walhalla, S. C. as called for by the Oconee County Disaster Plan. Responsibility for the health and safety of the public will be assumed by those outside agencies manning the Emergency Control Center.

12.3.5 COORDINATION WITH OUTSIDE GROUPS

During the first critical hours until the Emergency Control Center is manned and operating, the Shift Supervisor will assume responsibility for the protection of the general public in any locations beyond the Exclusion Area that may be affected as a result of the accident situation. As aids in evaluation of any possible hazards in off-site locations, he will utilize readout data available from the instrumented weather tower in the Unit 1 and 2 control room and from the unit vent radioactivity monitors. He will also utilize meteorological overlays, nomographs or other calculational aids, local area maps and population data for this purpose. An emergency vehicle and/or boat will also be available to him for off-site monitoring. If evacuation of any off-site locations is required immediately, the Shift Supervisor will call upon the services of the Sheriff's Department, the S. C. Highway Patrol, and the Oconee County Board of Education (school buses) as necessary.

As various agencies responsible for the public health and safety respond and the Emergency Control Center becomes operable, responsibility for the protection of the general public will be transferred from the Shift Supervisor to the Emergency Control Center. The Shift Supervisor will then be responsible only for the protection of station property and station personnel.

The agency initially staffing, and primarily responsible for the operation of the Emergency Control Center and for the protection of the health and safety of the public, will be the Oconee County Civil Defense Agency. They will be joined and assisted by the S. C. State Board of Health and Environmental Control Later, it may also be staffed by NRC Emergency Radiological Monitoring Team personnel and Duke Power Company personnel who will serve in an advisory capacity. Once the emergency situation is over, recovery operations at the Oconee Nuclear Station may also be directed from this Emergency Control Center. However, NRC personnel and Duke Power Company personnel may want to use an on-site location for directing plant recovery operations in which case the service building, Visitors Center or other suitable location within the Exclusion Area may be used for this purpose.

12.3.6 DESCRIPTION OF PROTECTIVE MEASURES, SPECTRUM OF ACCIDENTS

The Control Room is shielded and the ventilation is controlled so as to afford protection for operating personnel against external radiation and airborne radioactive materials in the event of a major reactor accident. This enables them to take remedial action necessary to protect personnel on-site, to take remedial action in regard to the station and to protect the general public in the surrounding area. All necessary communications equipment (station and Bell telephones, radio, public address system) is provided in the Control Room as well as instruments and controls for reactor and station operation. This enables operating personnel to continue operation of the second unit in the event of a major accident. As aids in evaluation of any possible radiological hazards in off-site locations, the Shift Supervisor utilizes site meteorological data (windspeed, direction, delta t) available to the Control Room from site weather instruments and information available from the station process radiation monitoring system. He also utilizes meteorological nomographs, local area maps, and population data for this purpose. An emergency vehicle and a boat are also maintained for emergency radiation monitoring and warning purposes. Each has radio communication with the Control Room. A site evacuation horn can also be actuated from the Control Room. Protective measures are implemented at pre-established protective action levels.

All of the Health Physics radiation counting and survey instruments are available for emergency purposes. In addition high range radiation monitoring instruments, dosimeters and self-powered air samplers are maintained in various locations on site for emergency purposes, i.e., Control Room, Administration Building, Auxiliary Building.

As a result of a radiation or contamination accident within the station, area or process monitoring instrumentation alarms sound, personnel evacuate the immediate area to the nearest safe shelter area or to an area where emergency facilities and monitoring are available such as the contaminated side of the Personnel Change Room or the First-Aid Room. The Shift Supervisor alerts other persons by means of the public address system. In case of a major release of radioactivity, the first evacuation is to the nearest suitable place of cover, as directed by the Shift Supervisor, to await further instructions, i.e., visitors on the grounds are asked to enter the Administration Building; station personnel on the grounds enter the Auxiliary, Service or Turbine Buildings; operations personnel assemble in the protected Control Room.

12.3.7 PROTECTIVE ACTION LEVELS AND EMERGENCY MEASURES REQUIRED

The fundamental basis for handling any accident or emergency situation is to prevent or minimize any hazard to station personnel or the general public and to prevent or minimize any damage to property. To accomplish this it is essential that:

(a) all personnel be trained and familiar with emergency procedures and equipment;

(b) each emergency situation be quickly and effectively evaluated; and that

(c) protective action, appropriate to the emergency situation be taken promptly.

In line with the above basis, the Protection Action Levels contained in this procedure have been conservatively chosen, so that they in themselves do not represent any serious hazard to persons. These Protective Action Levels are based on the recommendations of the International Commission on Radiological Protection, the National Council on Radiation Protection, the Federal Radiation Council and NRC Regulations. They further utilize guidance provided by the British Medical Research Council and other agencies that have prepared recommendations for the handling of radiological emergency situations.

12.3.7.1 Radiological Situations Involving Station Personnel Only

(a) Evacuation of Personnel from Affected Local Areas of the Station

Evacuation is required for any situation that is likely to expose personnel to doses in excess of the limits specified in Section 20.101 or 20.103 of 10CFR20. That is, any unplanned situation that will result for example, in whole body doses in excess of 2.5 mRem in an hour, 100 mRem in a week, 1250 mRem in a quarter, or an internal exposure in excess of the equivalent of mpc for 40 hours.

Generally, these situations will be due to sudden unexpected increases in radiation, surface or air contamination levels above existing levels, as a result of spills or leaks or changes in conditions.

(b) Allowable Dose to Personnel for Handling Emergency Situations

Planned exposures necessary for the protection of lives or to prevent serious property damage should not exceed 5 Rems whole body, 30 Rems to the skin of the whole body, or 75 Rems to the hands, forearms, feet and ankles. However, doses up to 25 Rems whole body, 150 Rems to the skin of the whole body, or 375 Rems to the hands, forearms, feet, and ankles may be incurred if necessary to save lives or to prevent further property damage that otherwise will place lives in jeopardy.

NOTE: The above doses represent the sums of external plus internal exposure. Appropriate respiratory protective equipment must be used to minimize internal exposure in any planned emergency action.

12.3.7.2 Radiological Situations Involving the General Public

In evaluating any possible hazards within the Exclusion Area or the Low Population Zone, the Shift Supervisor will utilize data available from the weather tower instruments in the Unit 1 and 2 control room and from the unit vent radioactivity monitors. He will also utilize the meteorological overlays, nomographs or other calculational aids that are available, local area maps and population data to evaluate the effects off site of a reactor accident. He will also utilize the emergency vehicle and/or boat for off-site monitoring.

NOTE: Off-site monitoring is also required as a follow-up to all accidental releases equal to or in excess of that described in 20.403 (b)(2) of 10CFR20.

(a) Within the Exclusion Area

All members of the general public and other persons who are not subject to occupational radiation exposure at the Oconee Nuclear Station must be warned (public address system, station telephone, etc.) and protected (shelter, health physics monitoring, etc.) and evacuated (evacuation horns, public address system) as necessary from the Exclusion Area, if as a result of any emergency condition they may possibly be exposed to doses in excess of any in the following table:

Organ or Tissue	Dose
Whole body, or major portion of body	0.5 Rem
Skin of whole body, or thyroid	3 Rems (1.5 Rems to thyroid of infants and children)
Hands and forearms, feet, and ankles	7.5 Rems
Other single organs	1.5 Rems

NOTE: Members of outside emergency services responding to a call from the station for assistance such as for evacuating people or blocking traffic into the Exclusion Area must also be protected from excessive radiation exposure. In these cases, their doses must not exceed 10 times the above limits.

(b) Outside the Exclusion Area and Within the Low Population Zone

NOTE: In the event of a major reactor accident situation, an Emergency Control Center will be established in the town of Walhalla, S. C. This center will be staffed by various agencies having responsibilities for health and safety of the public. During the first critical hours until the Emergency Control Center is manned and operating the Shift Supervisor will assume responsibility for the protection of the general public in locations within the Low Population Zone that may be affected by the accident situation. If evacuation or closing of areas is required immediately, he will call upon the services of the Sheriff's Department, the S. C. Highway Patrol, and the Oconee County Board of Education (school buses) as necessary for this purpose.

Evacuation of persons from the affected area in the Low Population Zone shall be undertaken for any reactor accident situation that is likely to result in doses in excess of any of those listed on the following page. However, the South Carolina State Board of Health and Environmental Control, Division of Radiological Health, will be advised of any accident condition or release that is likely to result in an external dose in the excess of 2.5 Rem or in exposure to a concentration of radioactive materials in excess of the equivalent of 500 times mpc (10CFR20 App. B, Table II) for 24 hours; and their advice will be sought for subsequent action required to assure the health and safety of the public.

EXTERNAL EXPOSURE

Exposure Group	Gamma Radiation(1)	Beta Radiation(2)	Skin Contamination(3)
Children up to 16 years and pregnant women	1.5 rads*	6.0 rads to superficial tissue	7.5 rads to superficial tissue
Other persons	2.5 rads	12.5 rads to superficial tissue	15.0 rads to superficial tissue

(1) whole body
(2) in addition to 1.5 or 2.5 rads gamma (whole body exposure)
(3) limited to 1/10 of the body surface in addition to (2)

* This corresponds to a time integral of concentration in air of 5.7 Ci-sec/m^3 of total gaseous and volatile fission products related to the time of release.

INTERNAL EXPOSURE

Radionuclide	Critical Organ	Limit for Individuals in General Population (RADS)	Limit for Suitable Sample, Children 1 Yr. of Age (RADS)
Strontium^{-89} or Strontium^{-90} or Cesium^{-137}	bone marrow bone marrow whole body	10 in first year; total dose not to exceed 15	3 in first year (1 for Strontium^{-90}) total dose not to exceed 5
Iodine^{-131}	thyroid	30**	10**

** This corresponds to a time integral of concentration in air of 0.043 Ci-sec/m^3 (for 30 rad, adult) and 0.003 Ci-sec/m^3 (for 10 rad, child) measured at the time of release and assumes other iodine isotopes are present in equilibrium proportions.

12.3.8 TRAINING PROGRAM, TESTING, DRILLS, REVIEWS AND UPDATING

All participating outside services have been familiarized with the Oconee Nuclear Station and their part in helping with an emergency situation. They have been given copies of the Emergency Plan and have discussed the plan with Duke Power Company at some length. All station personnel have received copies of the Emergency Plan and are responsible to familiarize themselves with the plan and to know their part in handling any emergency situation. This plan also is a part of the Reactor Operator and Health Physics training programs. Personnel on each shift are trained and qualified in Health Physics radiation monitoring and exposure control, first-aid, and firefighting. The capabilities are maintained.

In addition to the responsibility to familiarize themselves with the Emergency Plan, operating personnel conduct drills to develop and maintain their competence in handling each category of emergency situation. Insofar as possible, simulated drills involving off-site agencies are also performed in accordance with Technical Specifications.

Nurses and hospital attendants are also trained in the handling and care of possible contaminated patients by Duke Power Company Health Physics personnel. Ambulance drivers also receive appropriate training. This is done by use of a training film combined with a lecture and a demonstration of decontamination techniques. The initial training of hospital, ambulance, and rescue personnel is repeated periodically to maintain the ability of these persons to handle irradiated and contaminated patients.

The maintenance of a viable emergency plan is the responsibility of the Health Physics Supervisor.

12.3.9 PUBLIC RELATIONS AND NEWS RELEASES

Whenever possible, prompt and factual news releases are issued through normal channels by the Public Relations Department of Duke Power Company. When time does not permit release through established channels, the Manager of the Oconee Nuclear Station may authorize the issuance of a completed news release form. Warnings of the need for evacuation of an affected area are given promptly by local radio and TV spot announcements.

12.3.10 RECOVERY PLANS

Once any major emergency is over and the accident situation is under complete control, detailed plans will be developed, in accordance with the specific circumstances involved, to return the station to normal or near as normal conditions as possible. Comprehensive plans for recovery from any major accident situation will be formulated as a result of agreements between Duke Power Company, the NRC, the Division of Radiological Health of the S. C. State Board of Health and Environmental Control, and the heads of any local agencies involved with public health and safety matters as designated by the County Supervisor.

12.3.11 IMPLEMENTING PROCEDURES

Detailed procedures for implementing the Emergency Plan and for achieving its objectives have been prepared by the staff of the Oconee Nuclear Station. The plan is reviewed annually and updated as required. This responsibility is assigned to the Health Physics Supervisor. Temporary and permanent changes are made in accordance with station operating procedure change requirements.

12.4 REVIEW AND AUDIT OF OPERATIONS

Administrative controls assure that all operations, tests, and emergencies are handled in accordance with written procedures which have been properly reviewed and approved.

A daily review of the Shift Supervisor's log and other operating data is made by the Operating Superintendent or his assistant. All non-routine operations and conditions will also be reviewed by the technical services staff. In addition to these reviews, periodic station staff meetings will be held to keep all operating personnel advised of conditions in the station.

The Station Review Committee is appointed by the station Manager and serves as an independent review body reporting to the station Manager. The organizational structure, administrative requirements, responsibilities and authorities of the Station Review Committee are specified in the Oconee Technical Specifications. The Station Review Committee has the following general responsibilities:

(a) Review safety-related procedures and significant revisions to approved safety-related procedures.

(b) Review proposed safety-related station modifications.

(c) Review proposed changes to the station's Technical Specifications.

(d) Review abnormal occurrences, unusual events and violations of the station's Technical Specifications.

(e) Review safety-related station operations.

(f) Make recommendations concerning station safety to the station Manager.

The Executive Vice President and General Manager appoints a Nuclear Safety Review Committee for Oconee Nuclear Station. The nuclear safety review committee is an independent body to serve as a safety review and audit backup to the normal operating organization. The organizational structure, administrative requirements, responsibilities and authorities of the Nuclear Safety Review Committee are detailed in a written charter and in the Technical Specifications. The Nuclear Safety Review Committee has the following general responsibilities:

(a) Review proposed tests and experiments, and results thereof, when these involve an unreviewed safety question.

(b) Review proposed station modifications which involve an unreviewed safety question.

(c) Review requests to the Atomic Energy Commission for changes in the station's Technical Specifications that involve an unreviewed safety question.

(d) Review abnormal occurrences, unusual events and violations of the Technical Specifications.

(e) Periodically review the functioning of the operational quality assurance program.

(f) Conduct special reviews and investigations as requested by management.

12.5 STATION PROCEDURES

Safety-related operating, maintenance and testing activities at Oconee are conducted in accordance with approved, written procedures.

12.5.1 DESCRIPTION OF STATION PROCEDURES

12.5.1.1 Operating Procedures

Operating activities which affect the proper functioning of the station's safety-related structures, systems and components are performed in accordance with approved, written procedures. These procedures are intended to provide a preplanned method of conducting station operations, in order to eliminate errors due to on-the-spot analyses and judgments.

Operating procedures are sufficiently detailed that qualified individuals can perform the required functions without direct supervision. Written procedures, however, cannot address all contingencies and operating procedures, therefore, contain a degree of flexibility appropriate to the activities for which each is applicable.

12.5.1.2 Emergency Procedures

Emergency procedures are written which specify steps to be taken during foreseeable emergency situations. These procedures are based on a sequence of observations and actions, with emphasis placed on operator responses to indications in the Control Room. When immediate operator actions are required to prevent or mitigate the consequences of an emergency situation, procedures require that those actions be implemented at the earliest possible time, even if full knowledge of the emergency situation is not yet available.

The actions outlined in emergency procedures are based on a conservative course of action to be followed by the operating crew. Written procedures, however, cannot address all contingencies and emergency procedures, therefore, contain a degree of flexibility consistent with the fact that an emergency situation may not follow an anticipated sequence.

12.5.1.3 Periodic Test Procedures and Instrument Procedures

Safety-related testing conducted at Oconee on a periodic basis to determine various station parameters and to verify the continuing capability of structures, systems and components to meet performance requirements is conducted in accordance with approved, written procedures. Periodic test procedures and instrument procedures are utilized to perform such testing and are sufficiently detailed that qualified personnel can perform the required functions without direct supervision.

12.5.1.4 Maintenance Procedures

Maintenance of station safety-related structures, systems and components is performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances (for example, skills normally possessed by qualified maintenance personnel may not require detailed step-by-step delineation in a written procedure) which conform to applicable codes, standards, specifications, criteria, etc. Where appropriate sections of related vendor manuals, instructions or approved drawings with acceptable tolerances do not provide adequate guidance to assure the required quality of work, an approved, written maintenance procedure is provided.

Each procedure is sufficiently detailed that qualified workers can perform the required functions without direct supervision. Written procedures, however, cannot address all contingencies and maintenance procedures, therefore, contain a degree of flexibility appropriate to the activities for which each is applicable.

12.5.2 ADMINISTRATION OF STATION PROCEDURES

12.5.2.1 Preparation of Procedures

Each procedure is assigned to a member of the station staff for development. Initial procedure drafts are reviewed by members of the station staff; the Steam Production Department General Office, the Design Engineering Department and other departments within Duke, as appropriate. Following resolution of review comments, a revised procedure is prepared and forwarded to the station Manager for review and comment. The station Manager may transmit the procedure to the Station Review Committee and/or the Nuclear Safety Review Committee for their review and comment. After all required and appropriate reviews have been completed a final version of a procedure is prepared. Upon approval by the station Manager, a procedure becomes available for use.

12.5.2.2 Changes to Procedures

Minor changes to an approved procedure, without prior approval of the appropriate supervisory personnel, are permitted only to the extent that such changes correct errors in the applicable approved procedure of a typographical or editorial nature.

Major changes or revisions, i.e., all changes other than minor changes, to an approved procedure require final approval by the Operating Superintendent, the Instrument and Performance Engineer, the Maintenance Superintendent, the Chemistry-Health Physics Supervisor or the station Manager, as appropriate. Such changes may also be reviewed by the Station Review Committee. A major change or revision, however, may receive temporary approval for use by an appropriate station supervisor.

12.6 STATION RECORDS

The Oconee Nuclear Station records are maintained in a controlled and systematic manner in order to adequately document station operation. This maintenance is in accordance with the intent of the following:

(a) Title 10, Code of Federal Regulations, Part 50, Appendix B, Criterion XVII;

(b) Title 18, Code of Federal Regulations, Part 125, Section 125.3, Subsection 22.2.

12.6.1 STATION RECORDS ADMINISTRATION

The Oconee Nuclear Station Manager has the final responsibility for the proper management of station records. The various station supervisors are responsible to the station Manager for assuring the proper management, as applicable, of records within their purview.

Station records are maintained in an identifiable and retrievable manner in the station Master File. Access and use of the Master File are controlled.

12.6.2 STATION RECORDS REQUIREMENTS

12.6.2.1 Administrative Records

12.6.2.1.1 Abnormal Occurrences

Records of abnormal occurrences are retained for a minimum of six years.

12.6.2.1.2 Unusual Events

Records of unusual events are retained for a minimum of six years.

12.6.2.1.3 Station Review Committee

Records of Station Review Committee activities, including, but not necessarily limited to, minutes of Station Review Committee meetings, are retained for a minimum of six years.

12.6.2.1.4 Nuclear Safety Review Committee

Minutes of meetins of the Oconee Nuclear Safety Review Committee are retained for a minimum of six years.

12.6.2.2 Environmental Records

Records of off-site environmental surveys are retained for the life of the station.

2.6.2.3 Maintenance Records

12.6.2.3.1 Modifications

Records of modifications to the station as described in the FSAR are retained for the life of the station.

12.6.2.3.2 Maintenance Histories

Maintenance histories are maintained on station safety-related structures, systems and components. These histories contain a description of maintenance performed and sufficient documentation to assure identification of any replacement parts used. These records are retained for a minimum of six years.

12.6.2.3.3 Inspections

Safety-related inspections, such as equipment inservice inspections, cleanliness inspections and procedure compliance inspections, are documented in such a manner as to allow identification of the individual performing the inspection, when the inspection was performed, the type and purpose of the inspection, and the results of the inspection. These records are retained for a minimum of six years.

12.6.2.4 Operating Records

12.6.2.4.1 Changes to Operating Procedures

Records of changes to safety-related operating procedures are retained for the life of the station.

12.6.2.4.2 Switchboard Record

The Switchboard Record contains data on station and unit electrical power loadings and generation, generator temperatures, bus voltages and transmission line voltages, and is retained for a minimum of six years.

12.6.2.4.3 Reactor Operations Logbook

The Reactor Operations Logbook is maintained for each unit at the station. The logbook contains information concerning changes in reactivity. Notations are made of any abnormal conditions of operation due to auxiliary equipment and of releases of radioactive waste, both gaseous and liquid. Alarms received are normally logged by computer, however, those pertaining to reactor core conditions are also listed in this log with appropriate explanation. The Reactor Operations Logbook is retained for a minimum of six years.

12.6.2.4.4 Shift Supervisor Logbook

The Shift Supervisor Logbook contains a summary of station operation for each shift. Significant abnormalities which occur are explained in greater detail than would be expected in the Reactor Operations Logbook. The Shift Supervisor Logbook is retained for a minimum of six years.

12.4.2.5 Radiological Records

12.4.2.5.1 Personnel Radiation Exposure

Personnel radiation exposure records, including an individual's past radiation exposure history and current radiation exposure, for the appropriate personnel as required by 10CFR20, are retained for the life of the station.

12.4.2.5.2 Radiation Monitoring

Radiation monitoring records, including records of radiation and contamination surveys identifiable as to location and date, are retained for the life of the station.

12.4.2.5.3 Radioactive Releases

Records of radioactive releases and waste disposal are retained for the life of the station.

12.4.2.6 Special Nuclear Material Records

12.4.2.6.1 Physical Inventory

Position maps, photographs, television tapes or other suitable devices showing the location and identification of each fuel assembly are maintained for each fuel storage area. These records are updated at the completion of each major fuel handling operation. Also, a record is maintained of the storage location history of each fuel assembly from the time it is received at the station until it is shipped off site. Special nuclear material physical inventory records are maintained for the life of the station.

12.4.2.6.2 Isotopic Inventory

Records are kept which show the isotopic content of each fuel assembly as follows:

(a) When received.

(b) Semi-annually while being irradiated in a reactor.

(c) At the time of shutdown prior to each refueling outage.

(d) At the time of the fuel assembly is shipped off site.

Isotopic inventory records are retained for the life of the station.

12.6.2.7 <u>Testing Records</u>

12.6.2.7.1 Preoperational Testing

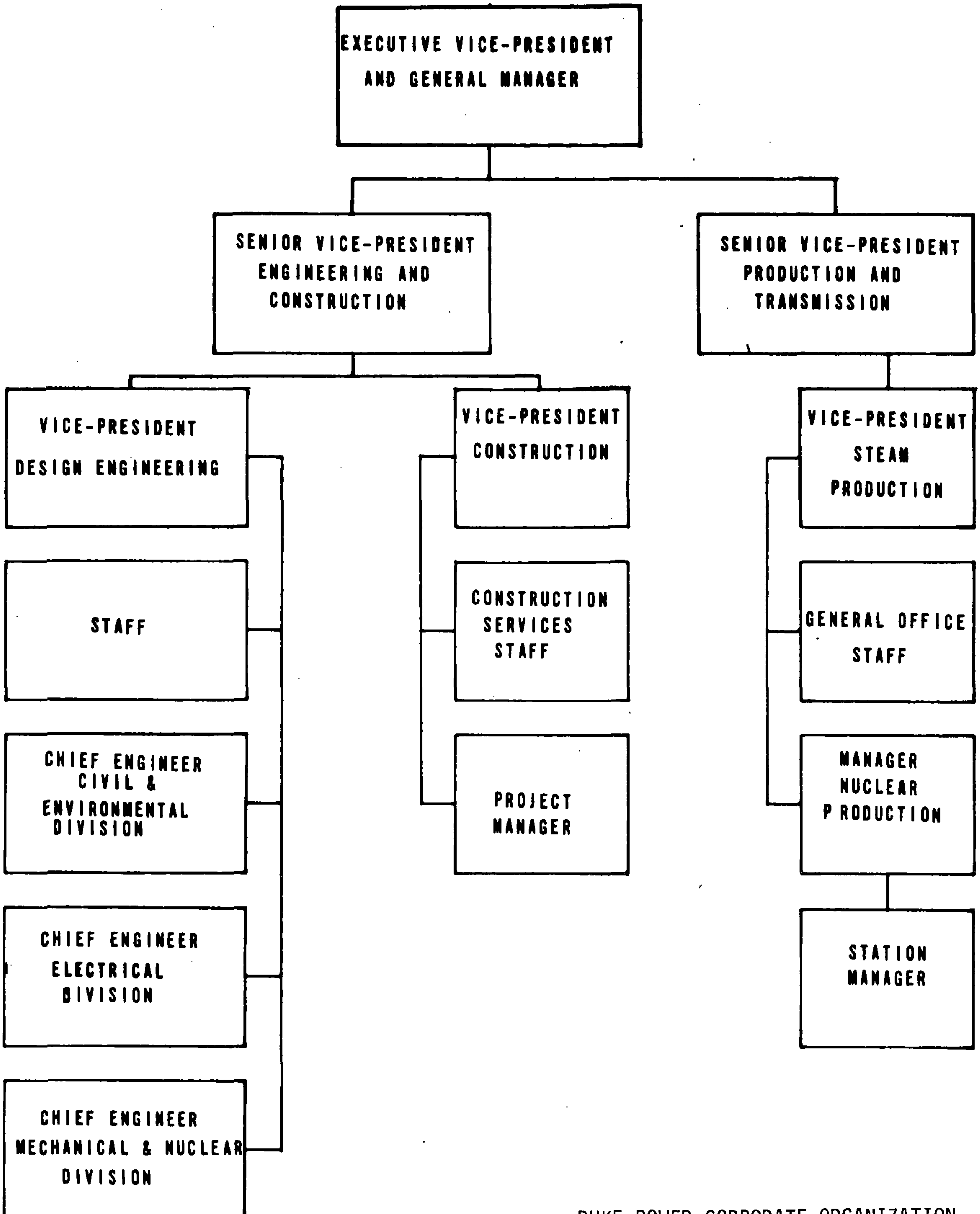
Safety-related preoperational testing records are retained in sufficient detail to permit adequate confirmation of the testing program. In particular, these records identify the data taker, the results of the testing and whether or not the results were acceptable, discrepancies and their cause, and any corrective action resulting from the test. Preoperational testing records are retained for the life of the station.

12.4.2.7.2 Special Testing

Records of special reactor tests or experiments are retained for the life of the station.

12.4.2.7.3 Periodic Testing

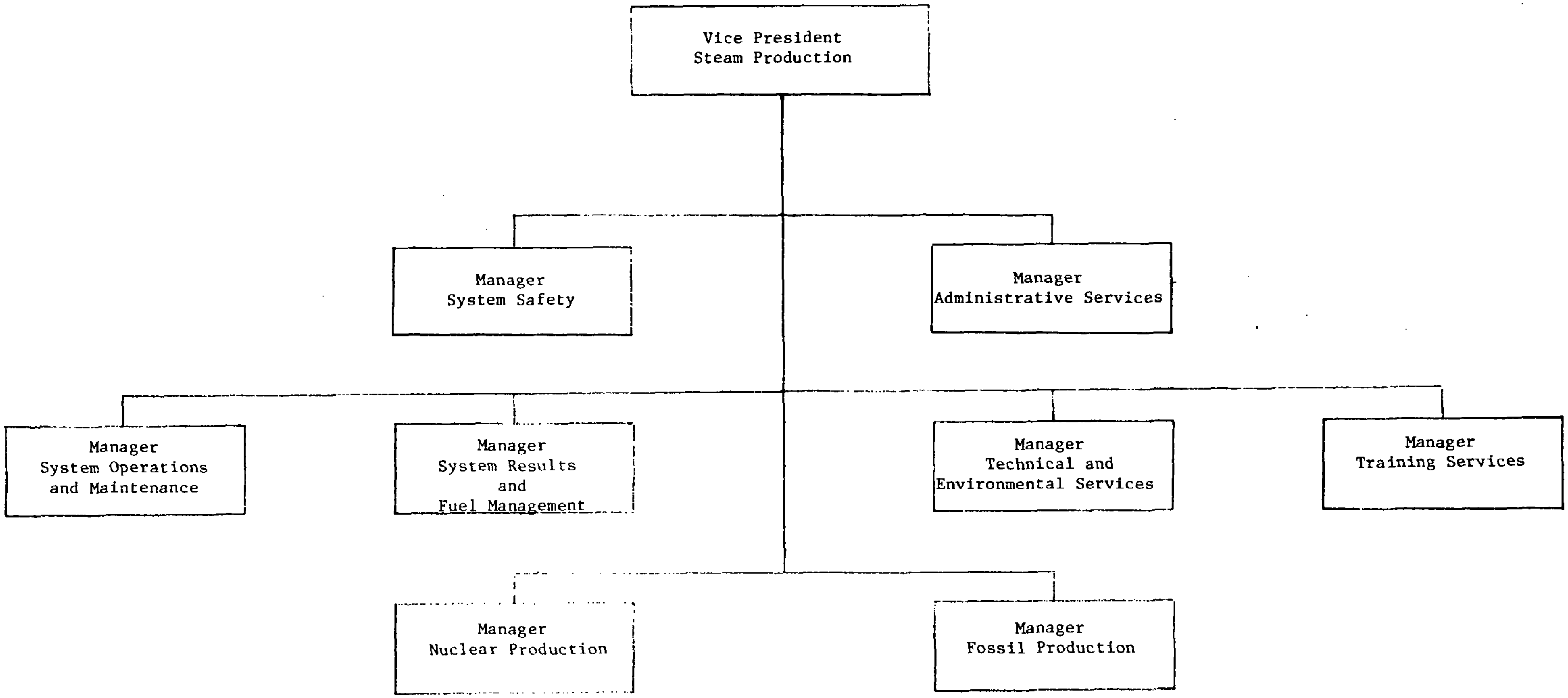
Safety-related periodic testing records are retained in sufficient detail to permit adequate confirmation of the testing program. In particular these records identify the data taker, the results of the testing and whether or not the results were acceptable, discrepancies and their cause, and any corrective action resulting from the test. Also, records of other periodic checks, calibrations, etc. performed in accordance with surveillance requirements for safety-related parameters, structures, systems and components are retained. Periodic testing and surveillance records are retained a minimum of six years.



DUKE POWER CORPORATE ORGANIZATION

OCONEE NUCLEAR STATION

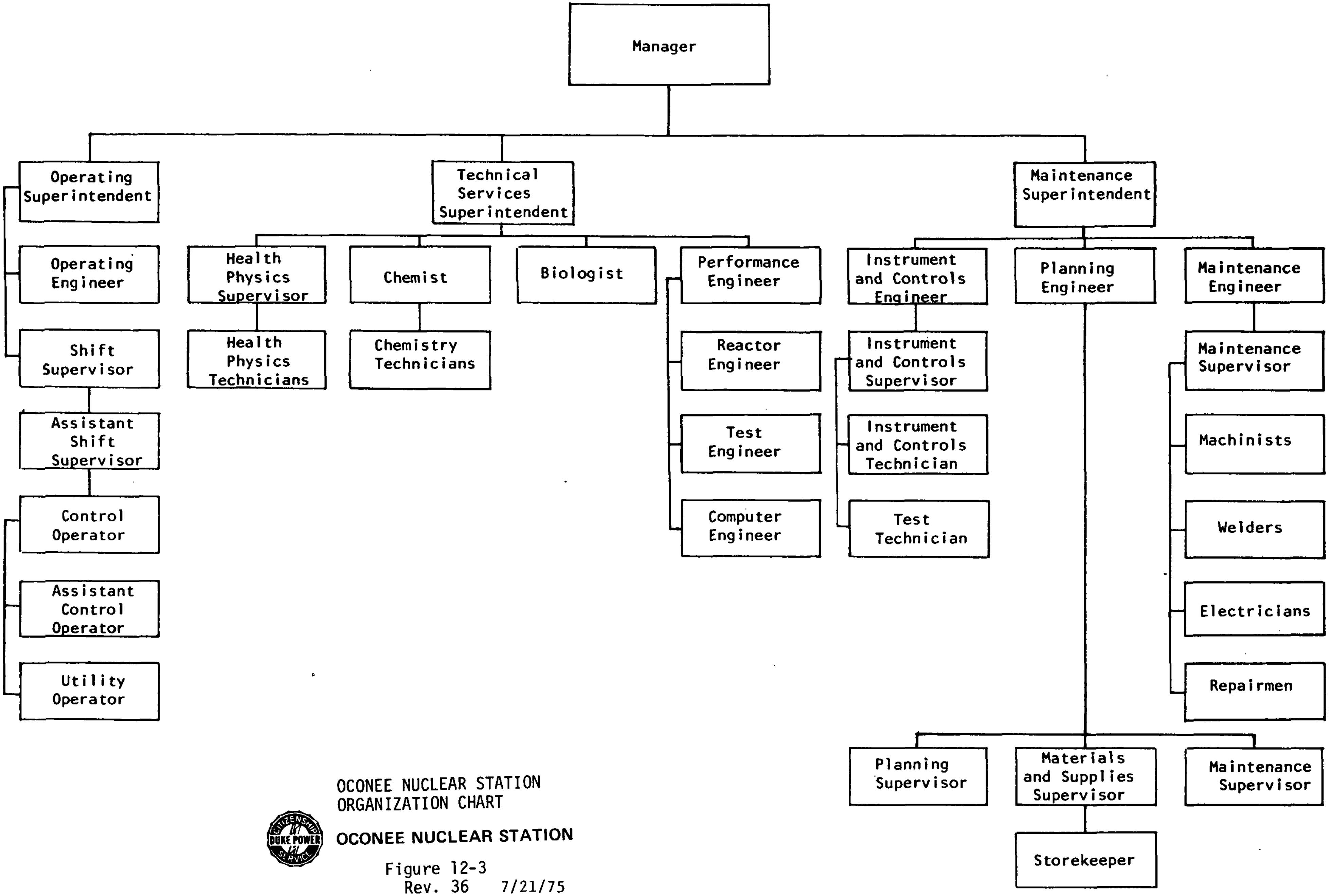
Figure 12-1
Rev. 36 7/21/75



STEAM PRODUCTION DEPARTMENT
ORGANIZATION CHART

OCONEE NUCLEAR STATION

Figure 12-2
Rev. 36. 7/21/75

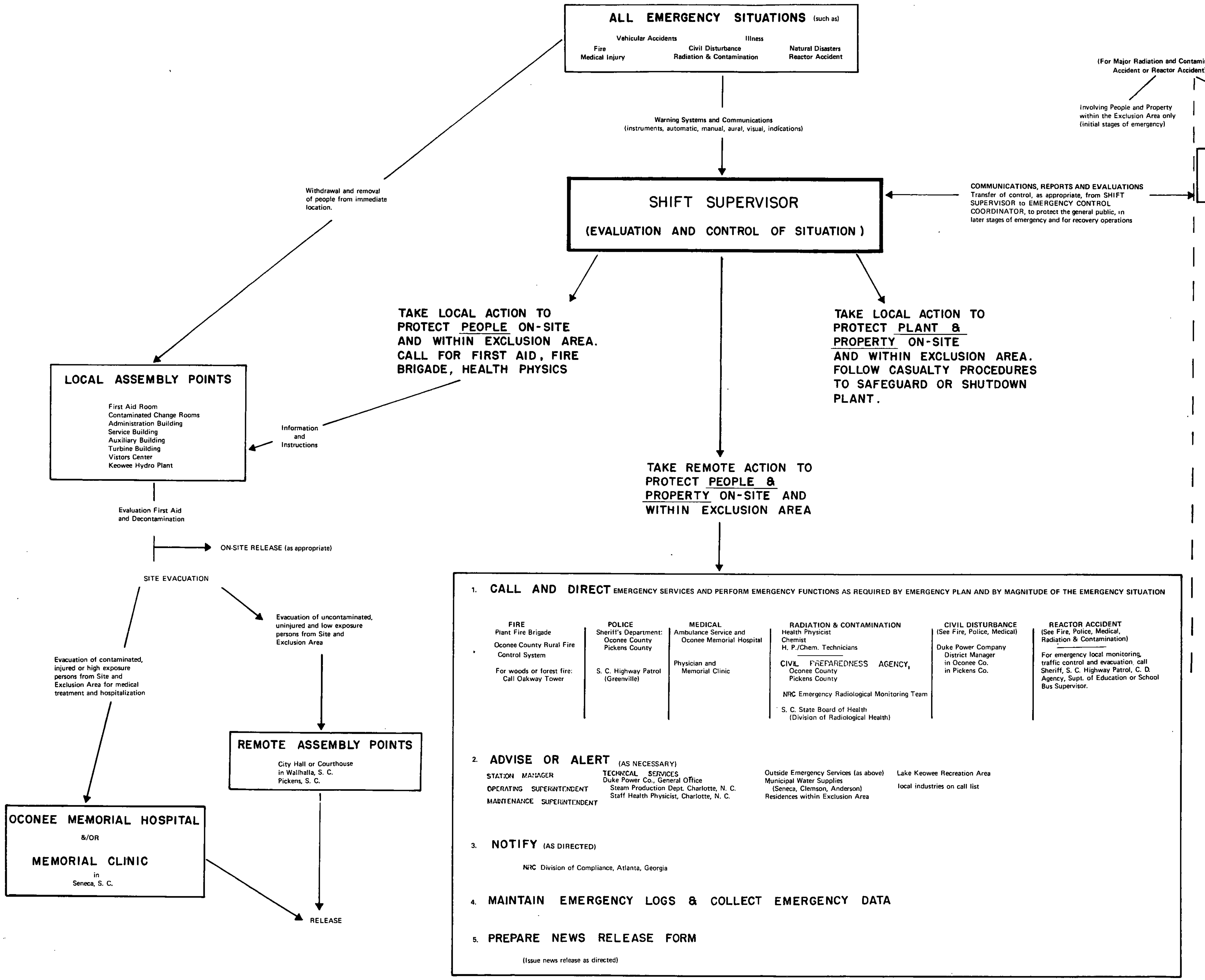


OCONEE NUCLEAR STATION
ORGANIZATION CHART

DUKE POWER

OCONEE NUCLEAR STATION

Figure 12-3
Rev. 36 7/21/75



OUTLINE OF BASIC EMERGENCY PLAN

OCONEE NUCLEAR STATION

Figure 12-4
Rev. 36. 7/21/75

SECTION 13

INITIAL TESTS AND OPERATION

Complete Section Revised

Submitted With FSAR Revision No. 4

April 20, 1970

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13 INITIAL TESTS AND OPERATION

A comprehensive initial testing and operation program will be conducted at the Oconee Nuclear Station. The purpose of this program is (1) to assure that the equipment and systems perform in accordance with design criteria, (2) to effect initial fuel loading in a safe efficient manner, (3) to determine the nuclear parameters, and (4) to bring the unit to rated capacity.

The test program will begin as installation of individual components and systems is completed. The individual components and systems will be tested and evaluated according to written test procedures. An analysis of the test results will verify that each component and system performs satisfactorily.

The written procedures for the initial tests and operation include the purpose, conditions, precautions, limitations, prerequisites, and the acceptance criteria.

13.1 ORGANIZATION OF TEST PROGRAM

13.1.1 GENERAL ORGANIZATION

The organization for development and execution of the test program has major participants from the Oconee Nuclear Station operating personnel, the Steam Production Department General Office staff, and Babcock & Wilcox (B&W) Site Operations. Additional participants will be from the Duke Engineering Department; Construction Department; and Electrical, Maintenance and Construction Department. Bechtel Corporation will participate in the tests associated with the reactor building.

The Oconee Nuclear Station organization for the test program consists of the Superintendent, Assistant Superintendent, Station Review Committee (SRC), and a station test coordinator assigned for each test.

The Steam Production Department General Office staff organization for the test
16. program consists of a Steam Production General Office test coordinator assigned for each test.

The B&W Site Operations organization for the test program consists of the Site Operations Manager, Site Operations Engineer, and Site Service Engineers who will be working in the specific areas of test procedures, testing, startup, operations, maintenance, fueling, field analysis, and reports. The test program will have technical support from B&W Nuclear Power Generation Division engineers. This support will include technical analysis of the test results of certain tests with the result analyses transmitted to the Steam Production Department through normal channels of communication for checking and final analyses prior to test completed approval. Special rapid channels of communication will be utilized where results are needed as soon as possible for other operations to proceed. The qualifications for the B&W Site Operations organization are listed below:

A. The minimum qualifications for the B&W Site Operations Manager are:

1. Graduate in engineering, or related physical science, or equivalent experience. (2 years experience for one year of college).

2. Four years of responsible power plant experience or two years of responsible nuclear reactor experience.

3. One year engineering or test program preparation experience for this or similar nuclear plant.

B. The minimum qualifications for the B&W Site Operations Engineer are:

1. Graduate in engineering, or related physical science or equivalent experience. (2 years experience for one year of college).

2. Two years of responsible power plant experience or one year of responsible nuclear reactor experience.

3. One year engineering or test program preparation experience for this or similar nuclear plant.

Either the Site Operations Manager or Site Operations Engineer will meet the nuclear e xperience requirements of A2 or B2 above.

Various individuals from the Mechanical, Electrical, and Civil sections of the Duke Engineering Department will furnish technical support as needed in specific areas. Similarly, individuals in the Duke Construction Department, Duke Electrical, Maintenance and Construction Department; and Bechtel Corporation will furnish techncial support as needed. This support will principally apply to the review of test procedures prior to approval, analyses of test results, and the development and installation of modifications to the equipment and systems as required and identified during the test program. Qualifications for Duke personnel are contained in Appendix 12A.

During the initial criticality (including fuel loading) and post-criticality phases of the test program, the nuclear physics and thermal hydraulics aspects of the reactor operation will be under the technical responsibility of the Steam Production Department Nuclear Engineer and the Oconee Technical Support Engineer with assistance from B&W Site Operations, B&W Nuclear Power Generation Division, and Duke Engineering Department nuclear engineers as needed. A very close coordination between these groups will exist with the appropriate support available when needed.

13.1.2 RESPONSIBILITIES

Superintendent

9. The Superintendent or his authorized representative has final responsibility for the overall test program which includes the approval of the test procedures, modification of test procedures, scheduling, completion of the tests, and approval of the test results. Approval of test procedures, modifications of test procedures, and approval of test results will not be made without giving proper consideration to recommendations of Babcock and Wilcox and Bechtel in their areas of interest.

Test Working Group

A Test Working Group (TWG) will coordinate the activities of B&W, Duke Construction, and the Steam Production Department during the preoperational test program. Representatives will be from Oconee Nuclear Station and B&W (Site Operations Engineer). Duke Engineering; Construction; Steam Produc-
16. tion General Office; and Electrical, Maintenance and Construction Depart-
21. ment will have representatives participate as required. The Oconee representative will be chairman of the TWG. The TWG will meet at regular intervals; approximately every week during the most active phases of the program.

Station Test Coordinator

A station test coordinator is designated for each test. His responsibility will be to develop the test procedure, coordinate the performance of the test, analyze results, identify discrepancies in test and acceptance criteria, initiate action to correct discrepancies, obtain approval of other parties when test has been completed satisfactorily, and file results in the master final documentation file.

Nuclear Test Engineer

20. A general office nuclear test engineer is designated for the testing program. His responsibility will be to furnish technical guidance for the test program; to assist in the development of the approved procedures; and to assist the station personnel in conducting and evaluating the tests. Other members of the general office staff will assist in the test program as necessary.

29. Nuclear Safety Review Committee

An audit of safety related tests and their results will be performed by the Nuclear
29. Safety Review Committee

13.1.3 RESOLUTIONS OF DISCREPANCIES

Any discrepancies in systems or equipment found during the Test Program will be promptly reported by the station test coordinator to the Superintendent.

A corrective action request will be made to the appropriate departments by the
9. Superintendent to initiate any revision or repair deemed necessary. After the corrective action has been completed the Superintendent or his authorized representative will be notified. Retests will be performed on systems and components as necessary to verify the adequacy of the corrective action.

Prior to any revisions relating to the health and safety of the public or plant personnel, structural integrity of plant components and systems, and items covered by codes and nuclear standards, review and approval is necessary by the Duke Power Design Engineering Department with assistance from vendors or consultants as necessary.

13.2 TESTS PRIOR TO REACTOR FUEL LOADING

The tests prior to reactor fuel loading assure that systems are complete and operate in accordance with design. The test program has been divided into two phases: Preheatup Test Phase and Hot Functional Test Phase. In many instances systems are tested during both the Preheatup Test Phase and the Hot Functional Test Phase. A summary of the test program prior to fuel loading is listed in Table 13-1.

The types of tests are classified as hydro/leak, operational, electrical, and functional with the following definitions for each classification:

Hydro/Leak Test - Structural integrity leak test of the various systems and components at the appropriate pressure.

Operational Test - Operation of systems and equipment under operating conditions.

Electrical Test - Consists of: grounding, megger, continuity, and phasing checks; circuit breaker operation and control checks; potential measurement and energizing of buses and equipment to ensure continuity, circuit integrity, and proper functioning of electrical apparatus.

Functional Test - Tests to verify that systems and equipment will function as intended.

Instruments and controls of each system or component will also be subjected to
9. a preoperational instrumentation and controls calibration prior to the
initial operation of that system or component to assure proper operation.

An engineered safeguard actuation system test will be performed to assure actuation and proper operation of the engineered safeguards system and to evaluate the test method and frequency for future testing.

13.2.1 PREHEATUP TEST PHASE

The objective of the Preheatup Test Phase is to assure that the equipment and systems perform as required for hot functional testing. This phase of the testing will include certain preoperational calibration, hydro/leak, operational, electrical, and functional tests as required. The reactor building containment system will undergo a structural integrity and integrated leakage rate test to verify the building design and to ensure that leakage is within the design limit.

13.2.2 HOT FUNCTIONAL TEST PHASE

The Hot Functional Test Phase will be a period of hot operation of the reactor coolant system and the associated auxiliary systems prior to the initial fueling of the reactor. The reactor coolant system will be heated up to no-load operating pressure and temperature.

The Hot Functional Test Phase will continue the preparation toward the initial fuel loading. The objectives of this phase of the test program are:

(a) Operational test of systems, components, and non-nuclear instrumentation and controls at no load operating pressure and temperature.

(b) Operator training.

(c) Verification of normal operating procedures.

(d) Verification of emergency operating procedures.

9. Following the hot functional test, the reactor vessel intervals will be removed and inspected for signs of distress e.g. loose parts, cracking or fretting.

13.3 INITIAL CRITICALITY TEST PROGRAM

The Initial Criticality Test Program consists of the initial fuel loading followed by initial criticality.

13.3.1 INITIAL FUEL LOADING

9. Fuel will be loaded into the reactor in accordance with a step-by-step written procedure. This procedure contains a number of safety precautions and operating limitations.

The fuel loading procedure includes:

(a) A sequence of loading temporary detectors, sources, control rods, and fuel assemblies in order to maintain shutdown margin requirements.

(b) The conditions under which fuel loading may continue after any step.

(c) An identification of responsibility and authority.

(d) During any reactivity changes, a minimum of two detectors will be operating and indicating neutron level after the source has been inserted. At all other times, at least one detector shall be indicating neutron level.

(e) Two completely independent plots of reciprocal neutron multiplications as a function of the parameter causing reactivity change will be maintained.

(f) Reactivity effects for each fuel assembly addition will be checked prior to the release of the fuel assembly by the fuel handling grapple.

(g) An estimate of the reactivity effect for the next fuel addition will be made prior to insertion of the next fuel assembly.

(h) The boron concentration in the reactor vessel, spent fuel pool, and reactor coolant system will be maintained at a value to assure the required subcritical margin at all times.

(i) The valve alignment of the auxiliary systems connected to the reactor coolant system will be checked periodically to prevent dilution of the reactor coolant boron concentration.

(j) Chemical analysis and water level monitoring will be used to assure that inadvertent dilution of the reactor coolant boron concentration has not occurred.

(k) Communication between control room and fuel handling areas will be maintained.

(l) The plant radiation monitoring systems will be in operation.

(m) Health physics and chemistry monitoring and services will be provided.

13.3.2 PREPARATION FOR INITIAL CRITICALITY

Upon completion of the initial fuel loading, prestartup checks will be completed prior to the approach to initial criticality. The prestartup checks will include:

(a) Control rod trip test
(b) Reactor coolant flow test
(c) Reactor coolant flow coastdown test

9.

A reactor coolant flow test and a reactor coolant flow coastdown test will be conducted under cold reactor conditions to assure that the flow characteristics of the reactor coolant system have not materially changed as a result of the reactor core installation.

13.3.3 INITIAL CRITICALITY

A written procedure will be followed during the approach to initial criticality. This procedure will specify in detail the sequence to be followed, the limitations and precautions, the required plant status, and the prerequisite system conditions. (This procedure also specifies the alignment of fluid systems
19. to assure controlled boron dilution and core conditions under which the approach to criticality may proceed.)

Permissible rod group withdrawal and deboration are based on calculated
19. reactivity effects. Two independent plots of inverse multiplication characteristics are maintained during rod group withdrawal and deboration. A predicted rod group position or boron concentration for criticality is determined before the next rod group withdrawal or deboration is started.

19.

13.4 POSTCRITICALITY TEST PROGRAM

The Postcriticality Test Program will be performed to provide assurance that the plant is operating in a safe and efficient manner. Systems and components which could not be operationally tested prior to initial criticality will be tested during the postcriticality test program to verify reactor parameters and to obtain information required for plant operation. A summary of the postcriticality testing is included in Table 13-2.

13.4.1 ZERO POWER PHYSICS TESTS

Following initial criticality, a program of reactor physics measurements will be undertaken to verify the physics parameters. Measurements will be made under zero power condition at sufficient temperature plateaus to verify calculated worths of individual control rods and control rod groups, moderator temperature coefficient, boron worth, and excess reactivity of the core. In addition, the response of the source and intermediate range nuclear instrumentation will be verified.

Detailed written procedures specifying the sequence of tests, parameters to be measured, and conditions under which each test is to be performed will be followed. These tests will involve a series of prescribed control rod configurations and boron concentrations with intervening measurements of control rod and/or boron worth during boron dilution or boron injection.

13.4.2 POWER ESCALATION TEST PROGRAM

Following determination of the operating characteristics and physics parameters of the reactor at zero power, a detailed power escalation test program will be conducted. This program will consist of specified incremental increases in power levels up to full power with appropriate testing conducted at each power level. An analysis of the significant parameters at each step will be made prior to initiating an additional power escalation.

9. At selected power levels, the following tests will be performed:

(a) Unit heat balance test
(b) Power coefficient measurement
(c) Core power distribution measurement
(d) Unit load steady state test
(e) Unit transient test

Other Power Escalation Tests will be performed at one or more power levels in the test sequence.

13.5 OPERATING RESTRICTIONS

During initial operations and associated testing, the normal plant safety procedures and technical specifications will be in effect. In addition, special safety precautions and limitations will be included in the test procedures and more restrictive operating limitations than those in the technical specifications will be imposed where required from initial criticality through the power escalation program. The reactor protective system power level trip point will initially be set at a low value and raised as the power escalation program progresses.

TABLE 13.1

Tests Prior to Initial Fuel Loading

Test Title	Abstract Page Number
Penetration Room Pressure Drop and Filter Test	13-11
Reactor Building Structural Integrity Test	13-12
Reactor Building Leak Tests	13-13
Reactor Building Cooling System Functional Test	13-15
Reactor Building Purge System Functional and Operational Test	13-16
Reactor Internals Vent Valve Inspection Test	13-17
Core Flooding System Engineered Safeguards Test	13-18
High Pressure Injection System Engineered Safeguards Test	13-19
Low Pressure Injection System Functional Test	13-20
Low Pressure Injection System Engineered Safeguards Test	13-21
Reactor Building Spray System Functional Test	13-22
Soluble Poison Control Test	13-23
Low Pressure Service Water System Test	13-24
Condenser Circulating Water System Gravity Flow Test	13-25
Steam Generator Hydrostatic Test	13-26
Integrated Control Rod Drive System Test	13-27
Control Rod Drive Trip Test	13-28
Process Radiation Monitoring System Calibration and Functional Test	13-29
Area Radiation Monitoring System Calibration and Functional Test	13-31
Reactor Coolant Hot Leakage Test	13-32
Control Rod Drive Operational Test	13-33
Electrical System Normal & Emergency Operation Test	13-34

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TABLE 13-1 (Continued)

Tests Prior to Initial Fuel Loading

Test Title	Abstract Page Number
Integrated Engineered Safeguards Actuation Test	13-35
Reactor Protective System Functional Test	13-36
Keowee Hydro Emergency Power Functional Test	13-37
Control Rod Drive Mechanism Temperature Test	13-38

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ABSTRACT OF

PENETRATION ROOM PRESSURE DROP AND FILTER TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To verify the integrity of the Penetration Room and filter train.

2. Prerequisites

Penetration Room shall be sealed as it would be sealed during normal operation.

3. Test Method

3.1 Start fans and adjust flow control elements to satisfy the design filter train flow rate and Penetration Room pressure.

3.2 Inject smoke into the particulate filter bed and measure the smoke penetration.

3.3 Inject refrigerant-112 into the carbon filter bed and measure the refrigerant-112 penetration.

4. Data Required

4.1 Record Penetration Room pressure and filter train flow rates.

4.2 Record penetration through the absolute and carbon filters.

5. Acceptance Criteria

5.1 The ability to meet design Penetration Room pressure and filter train flow rates must be demonstrated.

5.2 Penetration of generated smoke through absolute filters shall not be greater than .06%.

5.3 Penetration of refrigerant-112 through the carbon bed shall not be greater than 0.2%.

ABSTRACT OF

REACTOR BUILDING STRUCTURAL INTEGRITY TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To verify the structural integrity of the reactor building at 115% of design pressure.

2. Prerequisites

2.1 Completion of reactor building penetration leak tests.

2.2 Verification of the leak tightness of the reference system.

3. Test Method

The reactor building will be pressurized in steps up to a maximum of 115% of design pressure. At selected pressure levels, data will be recorded and inspections will be made to verify the structural integrity. After remaining at 115% of design pressure for approximately one hour, depressurization will begin. Data will be recorded at selected pressure levels during depressurization. During the test, visually inspect reactor building, hatches, penetrations, and gaskets.

4. Data Required

4.1 Record readings of strain gages, load cells, and deflection rods at selected pressure level.

4.2 Record temperature and pressure of reactor building air and the temperature of the reactor building liner plate.

5. Acceptance Criteria

Structure is capable of withstanding internal pressure of 1.15 times the design pressure.

ABSTRACT OF

REACTOR BUILDING LEAK TESTS

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

   1.1 To leak test each reactor building penetration and seal at calculated peak accident pressure or above.

   1.2 To measure integrated reactor building leakage rate at calculated peak accident pressure and at 1/2 calculated peak accident pressure.

2. Prerequisites

   2.1 Penetrations are installed.

   2.2 Penetrations tests completed before integrated leak rate test.

   2.3 Reactor building cooling system operable.

3. Test Methods

   3.1 The reactor building leak rate will be determined at calculated peak accident pressure and at 1/2 calculated peak accident pressure. The reference vessel method and/or absolute pressure method will be used. The "known leak" and "pump back" techniques may be used to verify the results.

   3.2 The penetration leak rates will be determined by pressurizing the
8 volume between the appropriate boundaries to 1/2 the calculated peak accident pressure and observing the pressure decay.

4. Data Required

   4.1 Integrated Leak Rate Test -

      4.1.1 Reactor building temperature pressure and humidity.
      4.1.2 Reference vessel temperature and pressure.
      4.1.3 Atmospheric pressure and temperature.
      4.1.4 "Known leak" and "pump back" air flow.
      4.1.5 Time.

4. Data Required (Continued)

4.2 Penetrations Leak Test -

4.2.1 Pressure.
4.2.2 Atmospheric pressure and temperature.
4.2.3 Time.

5. Acceptance Criteria

The leak rates must not exceed the allowable limits as given in Technical Specification.

31.

ABSTRACT OF

REACTOR BUILDING COOLING SYSTEM

FUNCTIONAL TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To verify functional operation of the reactor building cooling system.

2. Prerequisites

The low pressure service water system shall be operable.

3. Test Method

Fans are run at full speeds in combinations of two while one fan is idle. All three fans are run simultaneously at half speed.

4. Data Required

4.1 Fan speeds.

4.2 Fan/cooler air flows.

4.3 Reactor building coolers LPSW flow.

5. Acceptance Criteria

To provide emergency heat removal capacity a combination of three fan/coolers shall be capable of removing 240 x 10^6 BTU/hr under accident conditions as described in Section 14 of the FSAR. This requires:

5.1 A LPSW flow of 1400 gpm through each cooler.

5.2 A flow of 54,000 cfm through each fan.

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31. ABSTRACT OF

16. REACTOR BUILDING PURGE SYSTEM FUNCTIONAL AND OPERATIONAL TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To verify the isolation capability of the reactor building purge system and to verify the operation of the Radiation monitoring system interlock to valves in the purge line.

2. Prerequisites

2.1 Reactor building purge system operable.

2.2 Radiation monitoring system electrical checkout complete.

3. Test Method

3.1 Open valves in purge lines.

3.2 Start purge fan.

3.3 Simulate high radiation alarms to isolate purge system.

4. Data

4.1 Valve positions.

5. Acceptance Criteria

Upon high radiation alarm - proper valves close in purge line and fan trips.

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ABSTRACT OF

REACTOR INTERNALS VENT VALVE

INSPECTION TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

19. (REQUIRED FOR UNIT 1 ONLY)

1. Purpose

To demonstrate removal, replacement, and exercising of internals vent valves.

2. Prerequisites

The reactor vessel closure head and the internals upper plenum will be removed.

3. Test Method

For the removal and replacement test a valve will be removed and re-installed using normal handling tools and procedures.

For the exercise test each valve disc will be cycled using the normal exercise tool.

4. Data Required

Actual steps and tools required to remove and replace valves. For the exercise test the forces required to move the valve off its seat and to hold the disc wide open.

5. Acceptance Criteria

The tools and procedures required to remove and replace a valve are validated. The forces required to move each disc off its seat and to hold each disc wide open shall be within the specified limits of BAW 10005.

31. ABSTRACT OF

CORE FLOODING SYSTEM ENGINEERED SAFEGUARDS TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

19. (REQUIRED FOR UNIT 1 ONLY)

1. Purpose

To demonstrate core flooding system discharge capability.

2. Prerequisites

2.1 Reactor vessel filled with head removed and fuel transfer canal filled to level which will accept core flooding tank volumes without exceeding "full" level.

2.2 Core Flooding System operable.

3. Test Method

Pressurize core flooding tanks and open tank stop valves allowing contents to discharge into reactor vessel.

4. Data Required

Tank level and pressure vs time; stop valve cycle time.

5. Acceptance Criteria

Core flood discharge vs time is within design values.

ABSTRACT OF

HIGH PRESSURE INJECTION SYSTEM

ENGINEERED SAFEGUARDS TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To demonstrate that the high pressure injection pumps will be actuated by a 1500 psig Engineered Safeguards signal and to determine high pressure injection flow to the Reactor Coolant System.

2. Prerequisites

2.1 High Pressure Injection System and borated water storage tank operable.

2.2 The Reactor Coolant System filled and pressurized to $\geq$ 1650 psig.

3. Test Method

3.1 Reduce Reactor Coolant System pressure to approximately 1500 psig and actuate the high pressure injection pumps.

3.2 With the Reactor Coolant System pressure at approximately 1475 psig, operate each high pressure injection pump and determine flow.

4. Data Required

4.1 Reactor Coolant System pressure at which the high pressure injection pumps are actuated.

4.2 High Pressure Injection System flow versus time and Reactor Coolant System pressure versus time.

5. Acceptance Criteria

5.1 The high pressure injection pumps actuate when the Reactor Coolant System pressure reaches 1500 $\pm$ 37.5 psig.

5.2 The high pressure injection pumps deliver approximately 345 gpm flow at a Reactor Coolant System pressure of approximately 1500 psig.

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31. ABSTRACT OF

LOW PRESSURE INJECTION SYSTEM FUNCTIONAL TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To functionally test the operation of the low pressure injection system in the decay heat mode.

2. Prerequisites

2.1 The system is lined up for the decay heat mode of operation.

2.2 Reactor coolant system at elevated pressure and temperature.

3. Test Method

Operate the system for decay heat removal during reactor coolant system cool-down.

4. Data Required

Reactor coolant system temperature and pressure and the low pressure injection system temperature and flow.

5. Acceptance Criteria

The system parameters fulfill the calculated requirements for decay heat removal.

31. ABSTRACT OF

LOW PRESSURE INJECTION SYSTEM ENGINEERED SAFEGUARDS TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To demonstrate emergency low pressure injection flow to the reactor coolant system.

2. Prerequisites

2.1 The low pressure injection system and borated water storage tank operable.

16. 2.2 The reactor coolant system filled and pressurized to between 25 and 50 psig with nitrogen in the pressurizer.

3. Test Method

Operate the low pressure injection system in the engineered safeguards mode for each pump/flow path combination.

4. Data Required

Low pressure injection system flow and reactor coolant system pressure vs. time.

5. Acceptance Criteria

Each pump/flow path combination will be tested to assure that low pressure injection system flows will provide emergency core cooling as described in Section 6 of the FSAR.

31. ABSTRACT OF

REACTOR BUILDING SPRAY SYSTEM FUNCTIONAL TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

20.

1. Purpose

   1.1 To functionally check the reactor building spray pumps.

   1.2 To verify spray nozzles are open.

2. Prerequisites

   System filled and lined up recirculating to the borated water storage tank.

3. Test Method

   3.1 Each reactor building spray pump will be operated recirculating to the borated water storage tank.

   3.2 Force air or smoke through each spray header.

4. Data Required

   4.1 Suction pressure and discharge pressure of each pump at different flow rates.

   4.2 Observe flow through each spray nozzle.

5. Acceptance Criteria

   5.1 Pump characteristic curve is in accordance with FSAR Figure 6-9.

   5.2 Spray nozzle flow paths are open.

31.

ABSTRACT OF

SOLUBLE POISON CONTROL TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To test the equipment used to change the boron concentration in the reactor coolant system.

2. Prerequisites

Reactor coolant system and high pressure injection system in operation. Concentrated boric acid storage tank, boric acid mix tank, bleed holdup tanks and deborating demineralizers operable.

3. Test Method

Increase and decrease the boron concentration in the reactor coolant system using the required boron control equipment.

4. Data Required

Boron concentration versus time in the reactor coolant system. Volumes and concentrations of liquids required.

5. Acceptance Criteria

Boron concentration can be controlled as required to meet operational requirements.

31. ABSTRACT OF

LOW PRESSURE SERVICE WATER SYSTEM TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To test the normal operation and engineered safeguards function of the low pressure service water system.

2. Prerequisites

Low pressure service water system operable.

3. Test Method

Operate the system in its normal and emergency modes.

4. Data Required

Record temperature, pressure, and flow rates through reactor building coolers and decay heat removal coolers.

5. Acceptance Criteria

Flow through reactor building coolers and decay heat removal coolers is equal to or greater than minimum required for engineered safeguards operation.

31.

ABSTRACT OF

CONDENSER CIRCULATING WATER SYSTEM GRAVITY FLOW TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To demonstrate cooling water flow through condenser circulating water system by siphon action to satisfy emergency requirements after loss of power to the condenser circulating water pumps.

2. Prerequisites

48" emergency condenser circulating water discharge line to Keowee tailrace has been filled with water and is free of air.

3. Test Methods

3.1 Remove condenser circulating water pumps from service, one at a time. When the last pump is tripped, specified valves close in sequence and the discharge valve in the 48" line opens.

3.2 Trip power to continuous priming vacuum pumps, causing steam to be admitted to the emergency steam jet air ejector and the specified valves open and close in sequence.

4. Data Required

4.1 Condenser circulating water, continuous priming vacuum, and valve positions.

4.2 Verify condenser circulating water flow to Keowee tailrace.

4.3 Vacuum in continuous priming tank.

5. Acceptance Criteria

5.1 Valves assume proper position.

5.2 Flows and vacuum to meet emergency requirements as per FSAR Section 9.6.

31.

ABSTRACT OF

STEAM GENERATOR HYDROSTATIC TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. <u>Purpose</u>

   To hydrostatically test the non-isolable feed and steam lines connected to the secondary side of the steam generator.

2. <u>Prerequisites</u>

   Steam generator secondary side filled and vented. Reactor coolant system above design transition temperature.

3. <u>Test Method</u>

   Using hydro pumps, raise system pressure to code requirements, check system for leakage, and depressurize.

4. <u>Data Required</u>

   Pressure, temperature, and results of visual inspection.

5. <u>Acceptance Criteria</u>

   Conformance to USAS B31.1.0.

ABSTRACT OF

INTEGRATED CONTROL ROD DRIVE SYSTEM TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To ascertain that the Control Rod Drive System with all related inputs, outputs, and system responses is operating satisfactorily.

2. Prerequisites

2.1 The Control Rod Drive System has been electrically tested.

2.2 No fuel is in the reactor for the initial test.

3. Test Method

The Control Rod Drive System will be exercised in all operating modes.

4. Data Required

A detailed record will be maintained for each operational mode of the Control Rod Drive System and external equipment.

5. Acceptance Criteria

System operates satisfactorily in all modes of control.

31.

ABSTRACT OF

CONTROL ROD DRIVE TRIP TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

20.

1. Purpose

To measure control rod drive trip time from fully withdrawn to 3/4 control rod drive insertion and to verify control rod drive attains fully inserted position.

2. Prerequisites

2.1 Control Rod Drive System shall be operable.

2.2 Reactor shutdown margin in accordance with technical specifications.

3. Test Method

Withdraw each control rod or group to upper limit and trip.

4. Data Required

4.1 Reactor coolant temperature, pressure, flow and trip time for each control rod drive.

4.2 Visually verify that the position indication follows the rod up and down.

4.3 After each rod is tripped, verify that rods bottom.

5. Acceptance Criteria

Trip times do not exceed the value in the Technical Specifications.

31. 19.

ABSTRACT OF

PROCESS RADIATION MONITORING SYSTEM CALIBRATION

AND FUNCTIONAL TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To adjust the system, fluid flows, alarms, self-checking features, interlock functions, and tape speeds. To calibrate the system to primary radioactivity calibration data.

2. Prerequisites

2.1 Radiation Monitoring System operable.

2.2 Process systems being monitored must be in operation before final adjustments can be made.

3. Test Method

3.1 Calibration of Detectors - using secondary standard radioactivity sources, calibrate output to primary radioactivity calibration data.

3.2 Adjust alarms, interlocks, and recorders.

3.3 Adjust sampling parameters including fluid flow rates and tape speeds. Verify sampling valve sequences, and self-checking functions.

4. Data Required

4.1 Output from each channel for secondary standard radioactivity sources and installed operational check sources.

4.2 Alarm and interlock settings.

4.3 Fluid flow rates, tape speeds and flow alarm settings.

5. Acceptance Criteria

5.1 Outputs in agreement with primary calibration data.

5.2 Proper sampling, alarm, interlock, and self-checking functions operate as designed.

31. 18.

ABSTRACT OF

AREA RADIATION MONITORING SYSTEM CALIBRATION

AND FUNCTIONAL TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To adjust the system alarms, self-checking features, and interlock functions. To verify manufacturers initial calibration.

2. Prerequisites

Radiation Monitoring System operable.

3. Test Method

3.1 Using standard radiation source, calibrate detectors.

3.2 Adjust alarms, self-checking features, interlocks, and recorders.

4. Data Required

4.1 Output from each channel for standard radiation source and installed operational check source.

4.2 Alarm and interlock settings.

4.3 Verification of self-checking features.

5. Acceptance Criteria

5.1 Outputs in agreement with standard source radiation levels.

5.2 Proper alarm, interlock, and self-checking functions.

31.

ABSTRACT OF

REACTOR COOLANT HOT LEAKAGE TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To demonstrate that reactor coolant leakage, at hot pressurized system conditions, is within the limits set in the Technical Specifications.

2. Prerequisites

2.1 Hydrostatic tests of systems which form the reactor coolant boundary are complete.

2.2 The high pressure injection system and the reactor coolant system are operating as a closed system.

2.3 The reactor coolant system is hot and pressurized.

3. Test Method

Calculate reactor coolant inventory change by measuring changes in pressurizer and letdown storage tank levels during a specified time interval, with correction for reactor coolant temperature change.

4. Data Required

Record reactor coolant system average temperature, pressurizer level, and letdown storage tank level at the beginning and end of the time interval.

5. Acceptance Criteria

Reactor coolant leakage does not exceed limits set by the Technical Specifications.

31.

ABSTRACT OF

CONTROL ROD DRIVE OPERATIONAL TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To demonstrate operation of the control rod drive system in all modes of control at Hot Functional Test conditions.

2. Prerequisites

2.1 Control Rod Drive System operable.

2.2 Reactor Coolant System is at hot conditions .

2.3 Dummy drive guide assemblies in reactor vessel.

3. Test Method

3.1 Exercise Control Rod Drive System in all modes of control.

3.2 Continue periodic operation of control rod drive mechanisms during Hot Functional Test period. Verify trip functions after sustained mechanism operation.

4. Data Required

4.1 Control rod positions, stator temperature, and mode of control.

4.2 Record operational cycles imposed on each control rod drive.

5. Acceptance Criteria

5.1 Control Rod Drive System operates satisfactorily at hot operational conditions.

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ABSTRACT OF

ELECTRICAL SYSTEM NORMAL & EMERGENCY OPERATION TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To prove normal operation of the station electrical system and to verify automatic switchover to emergency sources when required.

2. Prerequisites

2.1 All normal and emergency power source complete.

2.2 Reactor shutdown for initial test.

2.3 Unit is on normal power source.

3. Test Method

Simulate signals to the relays to cause transfer to each emergency power source.

4. Data Required

Bus transfer time for each mode of transfer.

5. Acceptance Criteria

Successful automatic switchover to each emergency power source.

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ABSTRACT OF

INTEGRATED ENGINEERED SAFEGUARDS ACTUATION TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To demonstrate the full operational sequence of the Engineered Safeguards Actuation System and to demonstrate the transfer to alternate power sources.

2. Prerequisites

All systems actuated by the Engineered Safeguards System should have their individual functional tests completed.

3. Test Method

3.1 Actuate all Engineered Safeguard channels on the normal Engineered Safeguards power sources.

3.2 Actuate all Engineered Safeguard channels with simultaneous simulated failure of all normal Engineered Safeguard power sources.

3.3 Demonstrate by calibration or actual pressures the actuation of Engineered Safeguards instrument channels.

4. Data Required

Initial status and time required for the change of status for each Engineered Safeguards component during operation with normal power and with emergency power.

5. Acceptance Criteria

The Engineered Safeguards System and Electrical Power Systems shall respond as described in Section 6 of the FSAR.

31.

ABSTRACT OF

REACTOR PROTECTIVE SYSTEM

FUNCTIONAL TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To demonstrate functional performance of the reactor protective system.

2. Prerequisites

2.1 Reactor protective system is operable.

2.2 Reactor shutdown margin in accordance with technical specifications.

3. Test Method

Utilizing installed test equipment, vary each trip parameter until channel trips.

4. Data Required

Trip value for each parameter.

5. Acceptance Criteria

5.1 Trip setting for each parameter is within tolerances of technical specifications.

5.2 Reactor protective system functions as described in Section 7 of FSAR.

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ABSTRACT OF

KEOWEE HYDRO EMERGENCY POWER FUNCTIONAL TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

26. (UNIT 1 ONLY)

1. Purpose

To verify that the Keowee Hydro units supply emergency power to Oconee Nuclear Station.

2. Prerequisites

2.1 Oconee and Keowee switchgear is set up for normal operation.

2.2 Keowee Hydro operable.

3. Test Method

Each Keowee Hydro unit is started and brought up to speed in the following ways:

3.1 Automatically from Keowee control room.

3.2 Automatically from Oconee 1 and 2 control room.

3.3 Automatically by simulated engineered safeguards signal.

4. Data Required

Time to synchronous speed and bus voltage at Oconee.

5. Acceptance Criteria

Successful startup and automatic synchronization of both units in each mode.

ABSTRACT OF
CONTROL ROD DRIVE MECHANISM
TEMPERATURE TEST
FOR
DUKE POWER COMPANY
OCONEE NUCLEAR STATION
(UNIT II ONLY)

1. Purpose

To verify that the maximum motor tube extension temperatures on selected control rod drive mechanisms are less than the allowable value.

2. Prerequisites

2.1 The reactor coolant system at hot, pressurized conditions.

2.2 The temperature monitoring system is installed on the selected CRDM motor tube extensions for both portions of the test.

3. Test Method

3.1 During the Hot Functional Test phase, specified hold, cycle and trip operations will be conducted on the safety, regulating and axial power shaping drives. The motor tube extension temperatures will be monitored during these operations.

3.2 During first fuel cycle power operation, the motor tube extension temperatures will be monitored.

4. Data Required

The selected motor tube extension temperatures, as indicated by contact thermocouples. There are two thermocouples on the selected tube extensions, one above the stator and one below the seismic tie plate assembly.

5. Acceptance Criteria

The maximum motor tube extension temperatures are less than the allowable value in Topical Report BAW 10047, Revision 1, "Study of Discontinuities on Motor Tube Extensions".

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31.

ABSTRACT OF

ZERO POWER PHYSICS TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To verify design physics parameters.

2. Prerequisites

2.1 The reactor core is installed.

2.2 The reactor coolant system is operable.

2.3 Reactor building integrity is established.

2.4 Precritical check list is complete.

3. Test Method

The reactivity effect of control rod assembly, boron and temperature changes will be measured.

4. Data Required

4.1 Initial criticality data at cold conditions.

4.2 Differential control rod assembly group worths as a function of positions.

4.3 Differential boron worths as a function of boron concentration.

4.4 The "all control rod assemblies out" boron concentration for excess reactivity data.

4.5 The temperature coefficients of reactivity.

4.6 The reactivity change associated with the plant heatup from cold to hot conditions.

4.7 The worth of single control rod assemblies simulating an ejected control rod assembly in the hot condition.

4. <u>Data Required</u> (Continued)

4.8 Verify the stuck rod margin shutdown capability.

4.9 The total control rod assembly worth for shutdown margin verification in the cold and hot conditions.

5. <u>Acceptance Criteria</u>

5.1 Applicable design features in FSAR Section 3 are verified.

5.2 Verify parameters in technical specifications.

31. ABSTRACT OF

REACTOR COOLANT FLOW TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To demonstrate the flow characteristics of the reactor coolant system and the reactor coolant pumps. Test will be performed using various pump combinations during cold and hot conditions with the core installed.

2. Prerequisites

The reactor is shut down and all reactor coolant system parameters are satisfied for reactor coolant pump operation.

3. Test Method

Operate various combinations of Reactor Coolant Pumps.

4. Data Required

Reactor coolant flow, pressure, and temperature.

5. Acceptance Criteria

Flow under the various combinations must reach acceptable values for the
18. power levels shown in Technical Specification 2.1.

31.

ABSTRACT OF

REACTOR COOLANT FLOW COASTDOWN TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To determine flow versus time after trip for various reactor coolant pump combinations.

2. Prerequisites

2.1 Reactor coolant flow is established with desired pumps.

2.2 The reactor is shut down.

3. Test Method

For each desired operating pump combination, trip pumps and measure flow vs time.

4. Data Required

Record reactor coolant initial temperature, flow, and pressure. Record time of reactor coolant pump trip and flow during coastdown.

5. Acceptance Criteria

Flow during coastdown is equal to or greater than the calculated minimum allowable flow required to preserve the minimum allowable departure from nucleate boiling ratio.

31.

18. ABSTRACT OF

CONTROLLING PROCEDURE FOR POWER ESCALATION

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To startup the nuclear plant from zero power to operation at normal temperature at full power. This test sequences concurrent tests.

2. Prerequisites

Individual tests for all nuclear systems and components, including pre-operational calibration of the integrated control system (ICS), shall be completed. Zero power physics tests shall be completed.

3. Test Method

3.1 Increase reactor power from zero to about 15%. Thermal energy will be removed by the letdown coolers and steam dump to the main condenser via the bypass valves.

3.2 Calibrate nuclear instrumentation with a heat balance.

3.3 Adjust boron concentration to maintain control rod positions within design limits.

3.4 Adjust steam generator minimum level setpoint and reactor coolant average temperature setpoint to provide the combination of temperature and power required for plant design operating conditions.

3.5 Continue increasing power using turbine to power levels of 40%, 75%, 95%, and 100%.

19.

4. Data Required

As required by concurrent tests.

5. Acceptance Criteria

5.1 Nuclear instrumentation ranges overlap. Nuclear instrumentation is calibrated against a heat balance.

5.2 Nuclear Steam Supply System is operating within design limits.

19. 5.3 As required by concurrent tests.

ABSTRACT OF

LOSS OF CONTROL ROOM

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To demonstrate procedures and emergency controls used to maintain the unit in a hot shutdown condition during a simulated loss of Control Room incident.

2. Prerequisites

2.1 The unit is operating at approximately 15 percent power.

2.2 The controls and instrumentation required to maintain hot shutdown under postulated conditions are operable.

3. Test Method

3.1 Simulate condition requiring Control Room evacuation.

3.2 Carry out actions required prior to evacuating Control Room.

3.3 Control unit from Auxiliary Shutdown Panel and local stations.

4. Data Required

4.1 Reactor coolant pressure, temperature and flow, pressurizer level, letdown storage tank level, feedwater flow and main steam header pressure.

4.2 Log of operator actions.

5. Acceptance Criteria

Controls located outside the Control Room are adequate to effectively maintain the unit in a hot shutdown condition.

ABSTRACT OF

NUCLEAR INSTRUMENTATION POWER CALIBRATION

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To calibrate the power range nuclear instrumentation indication to the thermal power level of the reactor as determined by heat balance.

2. Prerequisites

Nuclear instrumentation preoperational calibration and testing complete.

3. Test Method

3.1 Obtain heat balance.

3.2 Adjust channel gain until indicated power level agrees with heat balance.

3.3 Repeat at various power levels up to full power.

4. Data Required

Indicated power level, channel gain, and heat balance data.

5. Acceptance Criteria

The power range nuclear instrumentation indicates the power calculated by the heat balance within the limits of the FSAR, Section 14.1.2.6.

31.

ABSTRACT OF

BIOLOGICAL SHIELD SURVEY TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

1.1 To measure radiation in all accessible locations of the plant outside of biological shields.

1.2 To obtain base-line radiation levels for comparison with future measurements of activity buildup with operation.

2. Prerequisites

2.1 Radiation survey instruments calibrated.

2.2 Background radiation levels measured in designated locations prior to initial criticality.

3. Test Method

Measure gamma and neutron dose rates at low, intermediate, and full power.

4. Data Required

Power level, gamma and neutron dose rates at each specified location.

5. Acceptance Criteria

Accessible areas and occupancy times during power operation are defined.

31.

ABSTRACT OF

NATURAL CIRCULATION TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
(REQUIRED FOR UNIT 1 ONLY)

19.

1. Purpose

To verify the capability for decay heat removal in the absence of all forced circulation.

2. Prerequisites

2.1 The reactor coolant flow coastdown test is completed.

2.2 The zero power physics test is completed.

3. Test Method

3.1 Determine the magnitude of natural circulation using the main feedwater pumps as the perturbation for measuring the induced temperature transient circulation time.

3.2 Determine the magnitude of natural circulation using the reactor as a heat source.

4. Data Required

Reactor coolant temperatures and pressure; OTSG water levels, and steam temperatures and pressure; reactor coolant pressurizes level; and feedwater temperatures and flows.

5. Acceptance Criteria

The reactor coolant system establishes adequate natural circulation in the absence of all forced circulation.

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31.

ABSTRACT OF

REACTIVITY COEFFICIENTS AT POWER

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To measure the power doppler and moderator coefficients of reactivity at selected power levels.

2. Prerequisites

The reactor is operating at power.

3. Test Method

3.1 Measure the power doppler coefficient by determining the reactivity change associated with a change in power level at constant reactor coolant average temperature.

3.2 Measure the moderator coefficient by determining the reactivity change associated with a change in reactor coolant average temperature at constant power level.

4. Data Required

Power changes, temperature changes, and reactivity changes.

5. Acceptance Criteria

The power doppler and moderator coefficients are consistent with the limits specified in the FSAR, Section 14.

31. ABSTRACT OF

XENON REACTIVITY WORTH AND RAPID DEPLETION TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

19. (REQUIRED FOR UNIT 1 ONLY)

1. Purpose

   1.1 To measure the peak xenon reactivity worth associated with predetermined transients in core power.

   1.2 To measure the core power redistribution associated with transient xenon conditions.

2. Prerequisites

   2.1 The reactor is operating at power.

   2.2 Incore monitor readout system operable.

   2.3 Initial induced power oscillation test must be completed.

3. Test Method

   Measure transient xenon effects by establishing steady state operation to achieve equilibrium xenon conditions, reducing core power to a lower power level, and returning back to the original power level at a specified xenon condition.

4. Data Required

   Control rod assembly differential worth data and incore instrumentation data during xenon transient.

5. Acceptance Criteria

   Xenon reactivity worth and core power distributions within design limits.

ABSTRACT OF

CORE POWER DISTRIBUTION TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To measure core power distribution using incore instrumentation.

2. Prerequisites

2.1 Reactor at power.

2.2 Incore monitor system operable.

3. Test Method

Read required data at selected conditions of reactor power level, xenon, and control rod positions.

4. Data Required

Incore instruments data, reactor power, control rod assembly positions, power history, and reactor coolant pressure, temperature, and flows.

5. Acceptance Criteria

Maximum/average power ratios within design limits.

31.

ABSTRACT OF

REACTOR/TURBINE TRIP TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To measure unit response to a turbine or reactor trip.

2. Prerequisites

Unit operating at power.

3. Test Method

Trip reactor or turbine and record transient response from a steady state condition.

4. Data Required

4.1 Reactor coolant system flow, pressure, temperatures, pressurizer level, and power vs time.

4.2 Steam temperature, pressure, steam generator water level, and feedwater flow vs time.

5. Acceptance Criteria

No equipment or system limits exceeded.

31.

ABSTRACT OF

INDUCED POWER OSCILLATION TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

UNIT 1

(REQUIRED FOR UNIT 1 ONLY)

19. 1. Purpose

1.1 To induce a xenon power oscillation and measure resultant core power distribution.

1.2 To demonstrate use of axial power shaping rod assemblies for control of axial power distribution.

2. Prerequisites

2.1 The reactor will be at equilibrium xenon conditions at the selected power level.

2.2 The power imbalance detector correlation test is complete for the test power.

2.3 Incore monitor readout system operable.

3. Test Method

3.1 The reactor power distribution will be perturbed by movement of control rods with soluble boron reactivity compensation at a constant power level.

3.2 The power redistribution will be monitored.

3.3 After several oscillations power distribution will be controlled by movement of the axial power shaping rod assemblies.

4. Data Required

Incore instrument data, reactor power, control rod assembly positions, and reactor coolant pressure, temperatures, and flows.

5. <u>Acceptance Criteria</u>

Core power distribution can be controlled.

ABSTRACT OF

ROD WORTH AT POWER TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

   1.1 To measure differential control rod assemblies reactivity worths during zero power operations.

   1.2 To measure differential control rod assemblies reactivity worths during power operation.

   1.3 To determine differential boron worth during control rod assembly measurements.

   1.4 To measure control rod assembly integral worth by rod drop.

2. Prerequisites

   Reactor critical.

3. Test Method

   3.1 Measure zero power differential control rod assembly reactivity worths by withdrawals or insertions of specified control rod assemblies and concurrently recording reactivity.

   3.2 Measure differential control rod assembly reactivity worth at power by using withdrawal and immediate insertion technique to compensate for temperature feedback.

4. Data Required

   Control rod assembly positions, reactivity, reactor coolant pressure, temperature, and flows, and boron concentrations.

5. Acceptance Critieria

   Rod worths as specified in FSAR, Section 14.

31.

ABSTRACT OF

POWER IMBALANCE DETECTOR CORRELATION TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

20.

1. Purpose

To verify the ability of the out-of-core power range detectors to detect and indicate an imbalance in the axial power distribution of the reactor core.

2. Prerequisites

2.1 Reactor at power.

2.2 Incore monitor readout system operable.

3. Test Method

Establish preplanned axial power imbalances and use incore monitoring system to measure the imbalance.

4. Data Required

Incore instrument data, out-of-core detector readings, total reactor power. Reactor coolant pressure and temperature.

5. Acceptance Criteria

The out-of-core detector system is calibrated to detect and indicate imbalances in the core power.

31.

ABSTRACT OF

NUCLEAR STEAM SUPPLY SYSTEM HEAT BALANCE

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

1. Purpose

To measure core thermal power.

2. Prerequisites

Unit at power.

3. Test Method

Core thermal power will be calculated from secondary heat balance at each stable power level during power escalation.

4. Data Required

4.1 Feedwater flow, temperature, and pressure.

4.2 Steam generator outlet pressure and temperature.

4.3 Letdown flow, temperature, and pressure.

4.4 Reactor coolant pump power.

5. Acceptance Criteria

Core thermal power is determined.

31.

ABSTRACT OF

UNIT LOAD STEADY STATE TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

20.

1. Purpose

To measure reactor coolant system (RCS) and steam generator steady state parameters as a function of reactor power.

1.1 Determine turbine steam temperature versus reactor coolant system average temperature.

1.2 Measure reactor coolant system and steam generator parameters with selected reactor coolant pumps operating.

2. Prerequisites

All systems ready for power operation.

3. Test Method

At selected power levels, measure reactor coolant system performance by:

3.1 Establishing steady state at test power level with

- ΔT_c set to zero to balance feed flow,
- T_{ave} constant,
- Feed flow constant,
- Megawatt demand constant,
- Integrated control system in auto, unit load demand in manual.

3.2 Measuring power by heat balance.

3.3 Predicting trend of Nuclear Steam Supply System parameters from all measured steady state data for next selected power level.

3.4 Repeating test at next selected power level.

4. Data Required

4.1 Reactor coolant pressure, temperatures, flows, pressurizer level, and reactor power.

4. Data Required (Continued)

4.2 Steam generator steam temperatures, pressures, levels.

4.3 Turbine header pressure.

4.4 Feedwater temperature, flow, and feed pumps operating.

4.5 Valve positions in feedwater system and steam system.

4.6 Control rod drive positions.

4.7 Generated megawatts.

5. Acceptance Criteria

5.1 Results of measurements at each power level are sufficient to predict performance at the next power level.

5.2 Reactor outlet temperature increases and reactor inlet temperature decreases monotonically with increasing power level.

5.3 Unit parameters do not exceed equipment or safety limits.

31. ABSTRACT OF

UNIT TRANSIENT TEST

FOR

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

20.

1. Purpose

Demonstrate that load changes can be made at the design rates.

2. Prerequisites

Unit operating at power.

3. Test Method

3.1 Perform load reduction followed by load increase at selected power levels between 15% and 100% power.

3.2 Repeat 3.1 with reduced number of reactor coolant pumps in operation.

4. Data Required

4.1 Reactor power.

4.2 Reactor coolant system pressure, pressurizer level, flow, temperatures, and boron concentration.

4.3 Steam generator levels, temperatures, pressure.

4.4 Steam header pressure.

4.5 Feedwater flow and temperature.

5. Acceptance Criteria

No equipment or system limits exceeded.

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14 SAFETY ANALYSIS

14.1 CORE AND COOLANT BOUNDARY PROTECTION ANALYSIS

14.1.1 ABNORMALITIES

In previous sections of this report both normal and abnormal operations of the various systems and components have been discussed. This section summarizes and further explores abnormalities that are either inherently terminated or require the normal protection systems to operate to maintain integrity of the fuel and/or the reactor coolant system. These abnormalities have been evaluated for core power of 2,568 MWt. Fission product dispersion in the atmosphere is assumed to occur as predicted by the dispersion models developed in 2.3. Table 14-1 summarizes the potential abnormalities studied.

Table 14-1
Abnormalities Affecting Core and Coolant Boundary

Event	Analysis Assumptions	Effect
Uncompensated Operating Reactivity Changes	Automatic control system inoperative or unused.	Reduction in reactor system average temperature. Automatic reactor trip if uncompensated. No equipment damage or radiological hazard.
Start-Up Accident	Uncontrolled single-group and all-group rod withdrawal from subcriticality with the reactor at zero power. Only high flux and high pressure trips were used to terminate the accident.	Power rise terminated by negative Doppler effect, control rod inhibit on short period, high reactor coolant system pressure, or overpower. No equipment damage or radiological hazard.
Rod Withdrawal Accident at Rated Power Operation	Uncontrolled single-group and all-group rod withdrawal with the reactor at rated power. Only high flux and high pressure trips were used to terminate the accident.	Power rise terminated by overpower trip or high-pressure trip. No equipment damage or radiological hazard.
Moderator Dilution Accident	Uncontrolled addition of unborated water to the reactor coolant system due to failure of equipment designed to limit flow rate and total water addition.	Slow change of power terminated by reactor trip on high temperature or pressure. During shutdown a decrease in shutdown margin occurs, but criticality does not occur. No radiological hazard.

Table 14-1 (Cont'd)

Event	Analysis Assumptions	Effect
Loss of Coolant Flow	Reactor coolant system flow decreases because of mechanical or electrical failure in one or more reactor coolant pumps. Reactor protection systems are the flux-flow and power-pump trips.	None. Core protected by reactor low-flow trip or loss-of-power trip. No radiological hazard.
Stuck-Out, Stuck-In, or Dropped-In Control Rod	Asymmetric rod monitor operates to inhibit rod out-motion and run back of secondary load.	None. Subcriticality can be achieved if one rod is stuck out. If stuck in or dropped in, continued operation is permitted if effect on power peaking is not severe. No radiological hazard.
Loss of Electric Power	Both a blackout condition and a complete loss of all station power are considered. One per cent defective fuel plus a 1 gpm steam generator tube leakage are assumed.	Possible power reduction or reactor trip, depending on condition. Redundancy provided for safe shutdown. Integrated thyroid dose at exclusion distance is 0.0008 Rem for loss of external load.
18. Steam Line Failure	Reactor coolant leakage into the steam generator continues for 3 hours following reactor operation with 1% defective fuel and 1 gpm steam generator tube leakage.	Reactor trips following a large rupture. Integrated doses at exclusion distance are 0.002 Rem whole body and 0.19 Rem thyroid.
Steam Generator Tube Failures	Reactor coolant leakage into the steam generator continues for 1.7 hours following reactor operation with 1% defective fuel.	Reactor automatically trips if leakage exceeds normal makeup capacity to reactor coolant system. Integrated doses at exclusion distance are 0.07 Rem whole body and 0.001 Rem thyroid.

14.1.2 ANALYSIS OF EFFECTS AND CONSEQUENCES

14.1.2.1 Uncompensated Operating Reactivity Changes

14.1.2.1.1 Identification of Cause

During normal operation of the reactor, the overall reactivity of the core changes because of fuel depletion and changes in fission product poison concentration. These reactivity changes, if left uncompensated, can cause operating limits to be exceeded. In all cases, however, the reactor protection system prevents safety limits from being exceeded. No damage occurs from these conditions.

14.1.2.1.2 Analysis and Results

During normal operation, the Integrated Control System senses any reactivity change in the reactor. Depending on the direction of the reactivity change, the reactor power increases or decreases. Correspondingly, the reactor coolant system average temperature increases or decreases, and the Integrated Control System acts to restore reactor power to the power demand level and to reestablish this temperature at its set point. If manual corrective action is not taken or if the control system malfunctions, the reactor coolant system average temperature changes to compensate for the reactivity disturbance. It is assumed in the analysis that the secondary system follows the temperature changes in the reactor coolant System. Table 14-2 summarizes these disturbances.

Table 14-2
Uncompensated Reactivity Disturbances

Cause	Maximum Reactivity Rate, $(\Delta k/k)$/min	Rate of Average Temperature Change (Uncorrected), F/min
Fuel Depletion	-2.9×10^{-7}	-0.0007
Xenon Buildup	-2.2×10^{-5}	-0.060

These results are based on $+0.5 \times 10^{-4}$ $(\Delta k/k)$/F moderator coefficient and -1.17×10^{-5} $(\Delta k/k)$/F Doppler coefficient, representative of beginning of core life for the first cycle. These reactivity changes are extremely slow and allow the operator to detect and compensate for the change.

16. 14.1.2.2 Start-Up Accident (Reference Supplement 9 Revisions for Oconee 3)

14.1.2.2.1 Identification of Cause

The objective of a normal start-up is to bring a subcritical reactor to the critical or slightly supercritical condition, and then to increase power in a controlled manner until the desired power level and system operating temperature are obtained. During a start-up, an uncontrolled reactivity addition could cause a nuclear excursion. This excursion is terminated by the strong negative Doppler effect if no other protective action operates.

The following design provisions minimize the possibility of inadvertent continuous rod withdrawal and limit the potential power excursions:

a. The control system is designed so that only one control rod group can be withdrawn at a time, except that there is a 25 per cent overlap in travel between two regulating rod groups successively withdrawn. This overlap occurs at the minimum worth positions for each group since one group is at the end of travel and the other is at the beginning of travel. The maximum calculated worth of any single control rod regulating group is 1.5 per cent $\Delta k/k$ when the reactor is critical.

b. Control rod withdrawal rate is limited to 30 in./min.

c. A short-period withdrawal stop and alarm are provided in the source range.

d. A short-period withdrawal stop and alarm are provided in the intermediate range.

e. A high flux level and a high-pressure trip are provided.

The criterion for the analysis of this accident is that the reactor protection system shall be designed to limit (a) the reactor thermal power to 114 per cent of rated power, and (b) the reactor coolant system pressure so as not to exceed code pressure limits.

14.1.2.2.2 Methods of Analysis

A B&W digital computer model of the reactor core and coolant system was used to determine the characteristics of this accident. This model used full reactor coolant flow, but no heat transfer out of the system and no sprays in the pressurizer. The rated-power Doppler coefficient [-1.17×10^{-5} $(\Delta k/k)/F$] was used although the Doppler is much larger than this for the principal part of the transient. The rods were assumed to be moving along the steepest part of the rod-worth versus rod-travel curve. The values of the principal parameters used were: 0.5 s trip delay for high-pressure trip, 0.3 s trip delay for high flux trip, $+0.5 \times 10^{-4}$ $(\Delta k/k)/F$ moderator coefficient, and -1.17×10^{-5} $(\Delta k/k)/F$ Doppler coefficient.

In addition, the criterion for minimum movable control rod worth is that a shutdown margin of 1 per cent $\Delta k/k$ at the hot standby condition is required (3.2.2.1.3). The start-up accident has been analyzed using the minimum tripped rod worth as part of the analysis.

The start-up accident was analyzed from 1 per cent $\Delta k/k$ subcritical at the hot, pressurized condition.

14.1.2.2.3 Results of Analysis

Figure 14-1 shows the results of withdrawing the maximum worth control rod group at a rod speed of 30 in./min from 1 per cent subcritical. This group is worth a maximum of 1.5 per cent $\Delta k/k$. This rod velocity and worth result in a maximum reactivity addition rate of 1.09×10^{-4} $(\Delta k/k)/s$. The Doppler effect begins to slow the neutron power(*) rise, but the heat to the coolant increases the pressure past the trip point, and the transient is terminated by the high-pressure trip.

Figure 14-2 shows the results of withdrawing all 61 control rod assemblies at the maximum speed (with a total worth of 10 per cent $\Delta k/k$) from 1 per cent subcritical. Although the calculated total rod worth (Table 3-2) is slightly higher, the sensitivity analysis in Figures 14-3 and 14-4 indicates that the difference will have little effect on the analysis. This results in a maximum reactivity addition rate of 7.25×10^{-4} $(\Delta k/k)/s$. The neutron power peaks at 275 per cent, where the power rise is stopped by the negative Doppler effect. The high neutron flux trip takes effect 0.25 s after the peak power is reached and terminates the transient. The peak thermal heat flux is only 37 per cent of the rated power heat flux.

A sensitivity analysis was performed on both of these start-up accidents to determine the effect of varying several key parameters. Variation of the trip delay time from 0.1 to 0.7 s resulted in a change in peak thermal power of only 5 per cent. Figures 14-3 through 14-6 show typical results for the single group, 1.5 per cent $\Delta k/k$ start-up accident.

Figures 14-3 and 14-4 show the effect of varying the reactivity addition rate on the peak thermal power and peak neutron power. This reactivity rate was varied from more than an order of magnitude below the nominal single rod-group rate used for analysis (i.e., the rate for the maximum-worth 1.5 per cent $\Delta k/k$ group) to a rate above that for simultaneous withdrawal of all rods. The slower rates--up to about 2.0×10^{-4} $(\Delta k/k)/s$--will result in the pressure trip being actuated. Only the very fast rates actuate the high neutron flux level trip.

Figures 14-5 and 14-6 show the peak thermal power variation as a function of a wide range of Doppler and moderator coefficients for the 1.5 per cent $\Delta k/k$ rod group. The peak thermal power varied about 10 per cent from the nominal case for the moderator coefficient variation, and also by about 10 per cent from the nominal for the range of Doppler coefficients. Figures 14-7 and 14-8 are the corresponding results from the withdrawal of all rods (10 per cent $\Delta k/k$).

None of these postulated start-up accidents, except for reactivity addition rates greater than 2×10^{-3} $(\Delta k/k)/s$, which is three times greater than for withdrawal of all rods at once, causes a thermal power peak in excess of 100 per cent rated power or a nominal fuel rod average temperature greater than 1,150 F. The nominal 1.5 per cent $\Delta k/k$ rod group withdrawal causes a peak pressurizer pressure of 2,515 psia, the relief valve set point. The capacity

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(*)Neutron power is defined as the total energy release from fission.

of the relief valves is adequate to handle the maximum rate of coolant expansion resulting from a start-up accident at 2 x 10^{-3} ($\Delta k/k$)/s.

It is concluded that the reactor is completely protected against any start-up accident involving the withdrawal of any or all control rods, since in no case does the thermal power approach 114 per cent, and the peak pressure never exceeds code allowable limits.

16. 14.1.2.3 Rod Withdrawal Accident at Rated Power Operation (Reference Supplement 9 Revisions for Oconee 3)

14.1.2.3.1 Identification of Cause

A rod withdrawal accident presupposes an operator error or an equipment failure which results in accidental withdrawal of a control rod group while the reactor is at rated power. As a result of this assumed accident, the power level increases, the reactor coolant and fuel rod temperatures increase, and, if the withdrawal is not terminated by the operator or protection system, core damage would eventually occur.

The following provisions are made in the design to indicate and terminate this accident:

a. High reactor coolant outlet temperature alarms.

b. High reactor coolant system pressure alarms.

c. High pressurizer level alarms.

d. High reactor coolant outlet temperature trip.

e. High reactor coolant system pressure trip.

f. High power level (i.e., neutron flux level) trip.

The rod withdrawal accident analysis is performed with the criterion that the reactor protection system will limit (a) the reactor thermal power to 114 per cent of rated power, and (b) the reactor coolant system pressure to code allowable limits.

14.1.2.3.2 Methods of Analysis

A B&W digital computer code was used to determine the characteristics of this accident. A complete kinetics model, pressure model, average fuel rod model, steam demand model with secondary coastdown to decay heat level, coolant transport model, and a simulation of the instrumentation for pressure and flux trip were included. The initial conditions were normal rated power operation without automatic control. Only the moderator and Doppler coefficients of reactivity were used as feedback. The nominal values used for the main parameters were as follows:

Trip delay time (high pressure trip), s	0.5
Trip delay time (high flux trip), s	0.3
CRA insertion time (2/3 insertion), s	1.4
Doppler coefficient, $(\Delta k/k)/F$	-1.17×10^{-5}
Moderator coefficient, $(\Delta k/k)/F$	$+0.5 \times 10^{-4}$
Control rod speed, in./min	30
Control rod group worth, % $(\Delta k/k)$	1.5

The criterion for minimum movable control rod worth is that a shutdown margin of 1 per cent $\Delta k/k$ at the hot stand-by condition is required (3.2.2.1.3). The rod withdrawal accident has been analyzed using the minimum tripped rod worth as part of the analysis.

14.1.2.3.3 Results of Analysis

Figure 14-9 shows the results of the nominal rod withdrawal from rated power using the 1.5 per cent $\Delta k/k$ rod group at 1.09×10^{-4} $(\Delta k/k)/s$. The transient is terminated by a high neutron flux level trip, and the reactor thermal power is limited to 108 per cent, well below the design overpower of 114 per cent of rated power. The changes in the parameters are all quite small. For example, the average reactor coolant temperature rise is only about 3 F and system pressure increase is only 120 psi.

A sensitivity analysis of important parameters was performed around this nominal case, and the resultant reactor coolant system pressure responses are shown in Figures 14-10 through 14-13.

Figure 14-10 shows the pressure variation for a very wide range of rod withdrawal rates--more than an order of magnitude smaller and greater than the nominal case. For very rapid rates, the neutron flux level trip is the primary protective device for the reactor core. It also protects the system against high pressure during fast rod withdrawal accidents. The high-pressure trip is relied upon for the slower transients. In no case does the thermal power exceed 111 per cent rated power.

Figures 14-11 through 14-13 show the pressure response to variations in the trip delay time, Doppler coefficient, and moderator coefficient. In all cases the neutron flux level trip is actuated.

An analysis has been performed extending the evaluation of the rod withdrawal accident for various fractional initial power levels up to rated power. This evaluation has been performed assuming simulated withdrawal of all 61 control rods giving a maximum reactivity addition rate of 7.25×10^{-4} $(\Delta k/k)/s$. This rate is a factor of seven higher than that used in the cases evaluated for withdrawal of a single group. The results of this analysis are shown in Figures 14-14 and 14-15.

As seen in Figure 14-14, the peak thermal power occurs for the rated power case and is well below the maximum design power of 114 per cent. The peak neutron power for all cases is approximately 119 per cent of rated power and represents a slight overshoot above the assumed trip level of 114 per cent. Figure 14-15 shows that the maximum fuel temperature reached in the average rod and the hot spot are well below melting. Even in the most severe case at rated power, the average fuel temperature increases by only 28 F. It is, therefore, readily concluded that no fuel damage would result from simultaneous withdrawal of all rods from any initial power level.

This analysis demonstrates that the high-pressure trip and the high flux level trip adequately protect the reactor against any rod withdrawal accident from power.

14.1.2.4 Moderator Dilution Accident

14.1.2.4.1 Identification of Cause

The reactor utilizes boron in the form of boric acid in the reactor coolant to control excess reactivity. The boron content of the reactor coolant is periodically reduced to compensate for fuel burnup. The dilution water is supplied to the reactor coolant system by the high pressure injection system. This system is designed with several interlocks and alarms to prevent improper operation. These are as follows:

a. Flow of dilution water to the letdown storage tank must be initiated by the operator. The dilution water addition valve can be opened only when the Group 6 control rods have been withdrawn to a preset (95 percent withdrawn) position and the timing device to limit the integrated flow has been set. Dilution water is added at flow rates up to 70 gpm at 2,200 psia.

b. Flow of dilution water is automatically stopped when either the flow has integrated to a preset value or when the Group 6 rods have been inserted to a preset position (at 75 per cent full stroke).

c. A continuous Dilute Permit light and feed and bleed valve position lights on the console are on whenever continuous dilution is in progress.

The high pressure injection system normally has one pump in operation which supplies makeup to the reactor coolant system and the required seal flow to the reactor coolant pumps. Thus, the total makeup flow available is normally limited to 70 gpm. When the makeup rate is greater then the letdown rate, the net water increase will cause the pressurizer level control to close the makeup valves. The nominal moderator dilution event considered is the pumping of water with zero boron concentration into the reactor coolant system.

It is possible, however, to have a slightly higher flow rate during transients when the system pressure is lower than the nominal value and the pressurizer level is below normal. This flow might temporarily be as high as 100 gpm.

Furthermore, with a combination of multiple valve failures or maloperations, plus more than one high pressure injection pump operating with reduced reactor coolant system pressure, the resulting inflow rate could be as high as 500 gpm. This constitutes the maximum dilution accident. A reactor trip would terminate unborated water addition to the letdown storage tank, and total flow into the coolant system would be terminated by a high pressurizer level.

The criteria for reactor protection in this accident are:

a. The reactor thermal power will be limited to less than the design overpower of 114 per cent rated power.

b. The reactor coolant system pressure will be limited to less than the code allowable limit.

c. The reactor minimum shutdown margin of 1 per cent Δk/k subcritical will be maintained.

14.1.2.4.2 Analysis and Results

The reactor is assumed to be operating at rated power with an initial boron concentration of 1,400 ppm in the reactor coolant system. The dilution water is uniformly distributed throughout the reactor coolant volume. Uniform distribution results from a discharge rate of 70 to 500 gpm into the reactor coolant flow. A change in concentration of 75 ppm produces a 1 per cent Δk/k reactivity change. The analysis is based on $+0.94 \times 10^{-4}$ (Δk/k)/F moderator coefficient, -1.17×10^{-5} (Δk/k)/F Doppler coefficient, and 1,400 ppm boron concentration. Both the moderator coefficient and boron concentration values used are conservative. The effects of the three dilution rates discussed above on the reactor are as follows:

Dilution Water Flow, gpm	Reactivity Rate, (Δk/k)/s	Average Reactor Coolant System Temp Change, F/s
70	$+2.2 \times 10^{-6}$	0.007
100	$+3.2 \times 10^{-6}$	0.010
500	$+1.6 \times 10^{-5}$	0.049

The highest rate of dilution can be handled by the Integrated Control System, which would insert rods to maintain the power level and thus limit the reactor coolant system temperature rise. If an interlock failure occurred while the reactor was under manual control, these reactivity additions would cause a high reactor coolant temperature trip or a high-pressure trip. In any event, the thermal power will not exceed 114 per cent rated power, and the system pressure will not exceed code allowable limits. Therefore, moderator dilution accidents will not cause any damage to the reactor system.

During refueling or maintenance operations when the reactor closure head has been removed, the sources of dilution water makeup to the letdown storage tank--and therefore to the reactor coolant system--are locked closed, and the high pressure injection pumps are not operating. At the beginning of core life when the boron concentration is highest, the reactor is about 9.5 per cent Δk/k subcritical with the maximum worth rod stuck out. To demonstrate the ability of the reactor to accept moderator dilution during shutdown, the consequences of accidentally filling the letdown storage tank with dilution water and starting the high pressure injection pumps have been evaluated. The entire water volume from the letdown storage tank could be pumped into the reactor coolant system (assuming only the coolant in the reactor vessel is diluted), and the reactor would still be 4.9 per cent $\Delta k/k$ subcritical.

14.1.2.5 Cold-Water Accident

14.1.2.5.1 Identification of Cause

An evaluation of possible accident modes has established the worst case for the idle pump startup accident. This case occurs when one loop is in operation at 50% power with the down steam generator operating at minimum level. For this case back flow through the down steam generator will enter the steam generator at the reactor inlet temperature (557 F) and is assumed to leave the steam generator at the saturation temperature of the secondary side (532 F). Therefore, if the two idle pumps are started under these conditions the potential for a cold water accident exists.

It is also possible to reduce the average reactor coolant temperature in the core; when the reactor is operated with one or more pumps not running, and these pumps are then turned on, the increased flow rate will cause the average core temperature to decrease. If the temperature coefficient is negative, reactivity will be introduced into the core and a power rise will occur.

14.1.2.5.2 Protective Procedure and Interlocks

7. For the case where one or more pumps are idle, an administrative procedure
21. is initiated whereby the operator must reduce power to less than 20% before starting a pump. This operational procedure assures that a trip will not occur before reaching equilibrium conditions upon restarting the pump. In addition, the pump control circuitry has an interlock to prevent turning on
21. an idle pump if the power is above 22%.

14.1.2.5.3 Method of Analysis

A detailed B&W Proprietary digital simulation of the plant was used to evaluate the transient response to this accident. The model includes point kinetics and a multi-region fuel pin model connected through time delays to a pressurizer model. A steam generator model was included. The reactor was assumed to be operating normally at 50% power with two pumps in one loop not operating. It was assumed that the two idle pumps were started and accelerated instantaneously to their normal flow value. It was assumed that no additional heat was removed from the steam generators. Conservative values of the moderator coefficient (-3.0×10^{-4} $\Delta k/k/°F$) and Doppler coefficient (-1.3×10^{-5} $\Delta k/k/°F$) were used. The protection criteria for this accident are that the minimum DNBR be greater than 1.3 and that system pressure limits not be exceeded.

14.1.2.5.4 Results of Analysis

This accident has been evaluated under the conditions outlined in 14.1.2.5.3.

The results are presented in Figure 14.15a. The curve clearly shows that the
cold water that will enter the core approximately 5 seconds after pump start-
up has an irrelevant effect when compared to the effect of starting two pumps
with the assumption of instantaneous pump acceleration. As the results show,
the power increases rapidly in response to the decreased temperature that
results from the step change in flow and reaches the trip setpoint about
0.5 seconds after the initiation of the accident. The maximum neutron power
reached was about 130%, however, the thermal power reached a maximum value
of only 62%. The pressure decreases initially in response to the system
7. temperature decrease but recovers due to the increased thermal power. The
swing from maximum to minimum pressure is about 100 psi, which represents a
change of pressurizer level of about 1.8 feet.

Based on the analysis, which conservatively assumes that the interlock does not function, it is concluded that the core is protected in the event that an idle pump is started. Additionally, the automatic control system could serve to limit the imbalance between the reactor and steam generator powers and reduce the severity of the accident. Since the thermal power does not exceed 65% of rated power at full flow and the pressure does not exceed 2300 psia, it is concluded that the protection criteria are met.

16. 14.1.2.6 Loss-of-Coolant Flow (Reference Supplement 9 Revisions for Oconee 3)

14.1.2.6.1 Identification of Cause

A reduction in the reactor coolant flow rate occurs if one or more of the reactor coolant pumps should fail. A pumping failure can occur from mechanical failures or from a loss of electrical power. With four independent pumps available, a mechanical failure in one pump will not affect operation of the others.

Each reactor coolant pump receives electrical power from one of the two electrically separate buses of the 6,900 volt system discussed in 8.2.2.3. Loss of the unit auxiliary transformer to which the 6,900 volt buses are normally connected will initiate a rapid transfer to the start-up transformer source without loss of coolant flow. Faults in an individual pump motor or its power supply could cause a reduction in flow, but a complete loss of forced flow is extremely unlikely. In spite of the low probability of this event, the nuclear unit has been designed so that such a failure would not lead to core damage.

The reactor protection criterion for loss-of-coolant-flow conditions
resulting from electrical malfunction of the reactor coolant pumps or their
7. power supply is that the minimum DNBR experienced by the core shall not be
less than 1.3. The protection criterion for loss-of-coolant-flow conditions
resulting from mechanical malfunction of a coolant pump or its motor is that
the minimum core DNBR shall be greater than 1.0. In addition, an analysis
has been made for the case where a reactor internals vent valve is stuck
open. This malfunction reduces the effective core flow by about 5% which
results in a DNBR of 1.40 at the steady-state design overpower.

14.1.2.6.2 Methods of Analysis

The loss-of-coolant-flow accident is analyzed by a combination of analog and digital computer programs. Analog simulation is used to determine the reactor flow rate following loss of pumping power. Reactor power, coolant flow, and inlet temperature are input data to the digital program which determines the core thermal characteristics during the flow coastdown.

The B&W digital computer model used to determine the neutron power following reactor trip includes six delayed neutron groups, control rod worth and rod insertion characteristics, and trip delay time. The analog model used to determine flow coastdown characteristics includes description of flow-pressure drop relations in the reactor coolant loop. Pump flow characteristics are determined from manufacturers' zone maps. Flow-speed, flow-torque, and flow-head relationships are solved by affinity laws.

A computer code has been developed by B&W to calculate fuel temperature, clad temperature and DNB ratio as a function of time for reactor system transients. Input to the code consists of flow, power, inlet temperature and system pressure as a function of time. The flow resulting from a coast-down is calculated by a standard B&W analog model that includes a simulation
7. of the pump and its associated inertia along with all pressure drops around the loop. The power, inlet temperature and system pressure is calculated by a standard B&W digital code that includes point kinetics with a closed loop simulation. The loop simulation includes a pressurizer model and a steam generator model, the whole of which is connected through time delays to the kinetics calculations.

The core transient analysis code simulates the reactor core through the use of a two-channel model. Each channel consists of one fuel rod with its associated flow area and spacer grid geometry. Given the necessary input as stated above, the code will calculate a pressure drop across a typical
4. reactor channel (average channel) as a function of time. This pressure drop is then imposed on the second channel (usually a hot channel) to determine hot channel flow, DNB ratio, in addition to fuel and clad temperatures. Compared to the average channel, the hot channel has greater heat generation and reduced flow area, as well as statistical hot channel factors. The analytical fuel pin model contains a transient response calculation while the hydraulic model considers the steady-state solutions of energy, mass, and momentum balances at each time step.

The transient response is obtained by applying the changing flow, power, inlet temperature, and system pressure to the initial conditions of the average channel. This calculation yields the average channel pressure drop as a function of time. This pressure drop as a function of time applied to the hot channel yields clad temperature, fuel temperature and DNB ratio as a function of time.

The loss-of-coolant-flow analysis has been carried out in the power range for coastdown from power levels between 100 to 114 per cent rated power. Conditions utilized in this analysis are as follows:

a. Initial core inlet temperature for given power level is assumed to be plus 2 F in error.

b. Initial system pressure is assumed to be minus 65 psi in error.

c. Trip delay time, i.e., time from sensor detection of loss of power to the pumps until initial downward movement of control rod, is 500 milliseconds.

d. The percentage of neutron power at beginning of life as a function
7. of time after initiation of control rod insertion is as shown in
Figure 3-6. This figure also contains the shutdown characteristics for a minimum of 1.0 per cent shutdown margin at the hot standby condition.

e. The pump inertia is 70,000 $lb\text{-}ft^2$.

14.1.2.6.3 Results of Analysis

The results of this analysis show that the reactor can sustain a loss-of-coolant-flow accident without damage to the fuel. The results of the evaluation are presented in Figures 14-16 and 14-17. Figure 14-16 shows the per cent reactor flow as a function of time after loss of all pump power. Figure 14-17 shows the minimum DNBR which occur during the coastdown from various initial power levels using the minimum tripped rod worth assuming 1% $\Delta k/k$ subcritical margin at hot standby. The degree of core protection during coastdown is indicated by comparing the minimum DNBR for the coast-
down (1.82) with the criterion value of 1.30. This DNBR (1.30) in the hot
7. channel corresponds to a 95 per cent confidence that 95 per cent of the
fuel rods in the core will not experience a departure from nucleate boiling.

Under normal conditions, the maximum indicated reactor power level from which a loss-of-coolant-flow accident could occur is 102 per cent rated power (as indicated by reactor instrumentation). This power level provides an allowance of plus 2 per cent rated power for transient overshoot. This power level also represents the maximum power demand that will be permitted to the reactor control system. The 102 per cent rated power is an instrument-indicated value and is subject to the following maximum errors: (a) $\pm$ 2 per cent heat balance and (b) $\pm$ 4 per cent nuclear instrumentation. The true power level could theoretically be as high as 108 per cent at 102 per cent indicated power. As shown in Figure 14-17, however, the DNBR at
108 per cent is 1.53 for the minimum rod worth available. The coastdown
7. from rated power results in a minimum DNBR of 1.82 which is considerably
larger than the 114 per cent overpower minimum DNBR of 1.55.

The reactor coolant system is capable of providing natural circulation flow after the pumps have stopped. The natural circulation characteristics of the reactor coolant system have been calculated using conservative values for all resistance and form loss factors. No voids are assumed to exist in the core or reactor outlet piping. The following tabulation shows the natural circulation flow capability as a function of the decay heat generation.

Time After Loss of Power, s	Decay Heat Core Power, %	Natural Circulation Core Flow Available, % Full Flow	Flow Required for Heat Removal, % Full Flow
3.6×10^{1}	5	4.1	2.3
2.2×10^{2}	3	3.3	1.2
1.2×10^{4}	1	1.8	0.36
1.3×10^{5}	0.5	1.2	0.20

The flows above provide adequate heat transfer for core cooling and decay heat removal by the reactor coolant system.

The reactor is protected against reactor coolant pump failure(s) by the reactor protective system and the integrated control system. The integrated control system initiates a power reduction on pump failure to prevent reactor power from exceeding that permissible for the available flow. The reactor is tripped if insufficient reactor coolant flow exists for the power level.

One additional loss-of-coolant mechanism has been analyzed in the reactor design evaluation. This involves possible flow or leakage past the seat of a reactor internals vent valve. These valves are designed to be closed during all normal operations and all normal and accident transients except for those accident transients where reverse flow through the core would occur. The design provides for positive closure even with no flow, and several rotational clearances are provided to assure free motion and to prevent any tendency to stick. The valves also have a self-alignment feature to prevent reactor coolant leakage. Hydrostatic and vibrational tests have
21. been made to demonstrate that the valves will operate as designed. However,
3. an analysis has been made for the case where a reactor internals vent valve remains open. This malfunction reduces the effective core flow by about 5% which results in a DNBR at the steady-state design overpower of 1.40.

The locked rotor accident has been investigated for all pump configurations with the exception of a single pump operation. The investigation has been carried out by calculating flow versus time for the various configurations for a locked rotor using a standard B&W analog technique. This information was in turn used as part of the input to a standard B&W digital code that
7. yields as part of its output the neutron power, system pressure and various core temperatures as a function of time. Hot channel temperatures and DNB ratios were calculated for these transients and the most adverse conditions were found to exist for the rated power case. This is to be expected since it is known that power is more influential than flow on the thermal behavior of the hot channel. The principal assumptions involved in the analysis are:

1. The initial power is 102% of rated power.

2. The initial pressure is 2135 psia, which is 65 psia below the normal operating pressure.

3. The initial reactor inlet temperature is 556 F, which is 2 F above the normal operating temperature.

4. The initial flow is the design value.

5. Maximum design conditions given in 3.2.3.2.2 were assumed for the thermal calculations.

6. Film boiling is assumed to occur at DNBR = 1.0.

7. The trip delay time was assumed to be 650 msec.

8. The locked rotor was simulated by stepping the pump inertia from its normal value to infinity.

7. Figure 14.17a shows the neutron power, pressure and flow for a locked rotor accident initiated from 102% of rated power. The flow decreases from its steady-state 4-pump value to its steady-state 3-pump value in about 2 seconds. The power-to-flow trip point is reached after about 300 milliseconds; however, no rod motion was assumed until 650 milliseconds later; this represents the delay in the power-to-flow instrumentation. The neutron power rises slightly from its initial value to about 103% under the influence of the assumed positive moderator coefficient of 5.25×10^{-5} $\Delta k/k/F$. The increased power results in a slight increase in system pressure that eventually reaches 2216 psia about 1.5 seconds after trip.

Figure 14.17b shows the hot channel DNB ratio versus time for the locked rotor accident during the principal part of the transient. The DNBR decreases rapidly to its minimum value of 1.15 which occurs about 1.5 seconds after the initiation of the transient. The hot spot cladding temperature does not vary appreciably from its steady-state value of 650 F. The centerline fuel temperature at the hot spot increases by 5 F from its initial value of 4066 F.

Based on the analysis of the locked rotor accident, it is concluded that:

1. The most severe single pump failure accident is the mechanical failure of one of four reactor coolant pumps at rated power.

2. No fuel melting will occur as a result of this accident.

14.1.2.7 Stuck-Out, Stuck-In, or Dropped-In Control Rod Accident

14.1.2.7.1 Identification of Cause

In the event that a control rod cannot be moved, localized power peaking and subcritical margin must be considered.

If a control rod is dropped into the core while operating, a rapid decrease in neutron power would occur, accompanied by a decrease in core average coolant temperature. In addition, the power distribution might be distorted due to the new control rod pattern. In the presence of a distorted power distribution, the return to full power might lead to localized power densities and heat fluxes in excess of design limitations.

14.1.2.7.2 Protective Basis

Adequate hot subcritical margin is provided by requiring a subcriticality of 1 percent $\Delta k/k$ with the control rod of greatest worth fully withdrawn from the core. The nuclear analysis reported in 3.2.2 demonstrates that this criterion can be satisfied. This criterion has been analyzed in terms of the minimum tripped rod worth available in the loss-of-coolant-flow, startup, rod withdrawal, and steam-line-failure accidents. In all cases, the available rod worth is sufficient to provide margins below any damage threshold.

For protective purposes a dropped control rod is defined as the deviation of a control rod from its group reference position by more than a maximum of 9 inches. This definition then covers both the action of dropping a rod and sticking a rod while moving a group. The action taken by the ICS is:

a. All rod-out motion is inhibited.

b. The steam generator load demand is run back to 60 percent of rated load at 5%/s.

The details of these actions are described in Section 7.2.2 and 7.2.3.

Although these ICS actions are available to mitigate the consequences of the accident, they are not required functions for safe plant operation as the results of the accident analysis demonstrate.

The criterion for plant protection during this transient is that the DNB ratio will not be less than 1.3 and the system pressure will not exceed code limits.

14.1.2.7.3 Method of Analysis

The transient response to a dropped control rod has been analyzed using a detailed B&W digital model. This program includes fuel pin, point kinetics, pressurizer, and loop models, including the steam generators.

The reactor is assumed to be operating at 100 percent of rated power when the control rod is dropped. In order to achieve the most adverse response the most negative values of moderator coefficient [3.0×10^{-4} $(\Delta k/k)/F$] and Doppler coefficient [-1.3×10^{-5} $(\Delta k/k)/F$] occurring at end-of-core life were

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used. The maximum rod worths expected to occur during full operation were used to examine the effects of ICS protective action. These rod worths correspond to operation at full power without xenon (0.46% $\Delta k/k$) and with xenon (0.36% $\Delta k/k$). It was assumed that the steam generator load demand was reduced linearly to 60 percent at 5%/s. The effects of a dropped rod without ICS action were also examined for a very conservative rod worth of 0.65% $\Delta k/k$. The rod was assumed to drop to 2/3 insertion in 1.4 seconds.

14.1.2.7.2 Results of Analysis

The results of the analysis with ICS action are presented in Figures 14-18 and 14-19. Figure 14-18 shows the response to a 0.46% $\Delta k/k$ dropped rod. The neutron power decreases rapidly to about 55 percent of rated power. This causes rapid decreases in the core moderator temperature and fuel temperature. These temperature decreases over-compensate for the worth of the control rod, and the power rises until the reduced steam generator demand begins to increase the inlet temperature and decreases the power. The thermal power levels out briefly at about 78 percent of its initial value but soon begins to decrease in response to the decreased steam generator demand. The pressurizer pressure swing is about $\pm$ 60 psi before returning to equilibrium.

Figure 14-19 shows the results of the 0.36% $\Delta k/k$ rod drop. The initial neutron power decrease is slightly less in this case, resulting in the thermal power leveling off at 83 percent, a slightly higher value than in the 0.46% $\Delta k/k$ case. The pressurizer pressure peaks at a higher value due to this higher thermal power.

Figure 14-19a shows the results of a 0.65% $\Delta k/k$ dropped rod analysis conservatively based on no ICS action and operation at higher than rated power level of 2772 mwt. The neutron power decreases causing a rapid decrease in both the core moderator temperature and the fuel temperature. These temperature decreases overcompensate for the worth of the control rod, and the neutron power rises slightly above the initial neutron power level. The neutron power then decreases to below the initial power level and eventually levels out at the initial power level. The thermal power response is similar to the neutron power; however, the thermal power level never exceeds the initial power value. Both the core moderator temperature and pressurizer pressure decrease during the transient and level out at a value lower than the initial value.

Several cases have been run for rod drops at beginning of life conditions. These transients yield new power levels that are lower than the end of life conditions and may result in reactor trip. These are therefore not included in this discussion because they represent less severe conditions.

14.1.2.7.5 Conclusions

Control rod malfunctions are accommodated by the core design without ICS action. Since the most severe case analyzed for the dropped rod does not result in reactor trip nor does the thermal power exceed its initial value, core and reactor coolant system boundary protection is assured. Additional protection for the dropped rod accident is provided through the ICS which detects a dropped rod and inhibits out-motion of the control rods. The ICS is designed to run back the steam generator load demand upon receiving the dropped rod signal from the rod drop detection circuitry. The reactor thermal power will assume a lower value that matches the load demand and will provide additional margin toward not exceeding any design limit.

14.1.2.8 Loss of Electric Power

14.1.2.8.1 Identification of Cause

Each unit is designed to withstand the effects of a loss of electric load or electric power. Emergency power systems are described in 8.2.3. Two types of power losses are considered:

a. A loss of load condition, caused by separation of the unit from the transmission system.

b. A hypothetical condition which results in a complete loss of all system and station power.

The reactor protection criteria for those conditions are that fuel damage will not occur from an excessive power-to-flow ratio nor will the reactor coolant system pressure exceed design pressure.

14.1.2.8.2 Results of Loss-of-Load Conditions Analysis

36. The effect of a loss-of-load condition on a unit would be that the unit generator breakers would open and thus disconnect the unit from the transmission system. When this occurs, a runback signal causes an automatic power reduction to 15 percent power. Depending on the initial power level at the time of the loss of load, the Reactor Protective System may initiate a reactor trip on high reactor coolant temperature or pressure. Electrical power would then be supplied from any of the available sources as described in 8.2.3. Assuming that no Reactor Protective System actuation is required, the following actions occur:

a. All vital electrical loads, including power to the reactor coolant pumps, condenser circulating water pumps, hot well and condensate booster pumps, and other auxiliary equipment, will continue to obtain power from the unit generator. Feedwater is supplied to the steam generators by steam-driven feed pumps.

b. As the electrical load is dropped, the turbine generator accelerates and closes the governor valves and intercept valves. The unit frequency will peak at less than the overspeed trip point and decay back to set frequency in 40-50 seconds.

c. Following closure of turbine governor valves and intercept valves, steam pressure increases to the turbine bypass valve set point, and may increase to steam system safety valve set point. Steam is relieved to the condenser and to the atmosphere. Steam venting to the atmosphere occurs for about two minutes following loss of load from 100 per cent initial power until the turbine bypass can handle all excess steam generated. About 148,000 pounds of steam will be relieved to the atmosphere. Steam relief permits energy removal from the reactor coolant system to prevent a high pressure reactor trip. The initial power runback is to 15 per cent power, which is a higher power level than needed for the unit auxiliary load. This allows sufficient steam flow for regulating turbine speed control. Excess steam above unit auxiliary load requirements is rejected by the turbine bypass valve to the condenser.

d. During the short interval while the turbine speed is high, the vital electrical loads connected to the unit generator will undergo speed increase in proportion to the generator frequency increase. All motors and electrical gear so connected will withstand the increased frequency.

e. After the turbine generator has been stabilized at auxiliary load and set frequency, the station operator may reduce reactor power to the auxiliary load as desired.

The loss-of-load accident does not result in any fuel damage or excessive pressures on the reactor coolant system. There is no resultant radiological hazard to station operating personnel or to the public from this accident as only secondary system steam is discharged to the atmosphere.

Unit operation with 1 per cent defective fuel and 1 gpm steam generator tube leakage is shown to be safe by the analysis presented in 11.1.2.5. For the same conditions, the steam relief accompanying a loss-of-load accident would not change the whole body dose. The whole body dose is primarily due to the release of Xe and Kr. Release of these gases is not increased by the steam relief because, even without relief, all of these gases are released to the atmosphere through the condenser air ejector. The rate of release of iodine during the approximately two minutes of relief would be increased by almost a factor of 10^4, because the iodine is released directly to the atmosphere rather than through the condenser and station vent. However, the quantity released during this short time is small--0.04 dose equivalent curies of I-131. Atmospheric dilution is calculated using the 2-hour elevated release dispersion factor of 3.35×10^{-5} s/m^3 developed in 2.3. The total integrated thyroid dose from this release is 0.0008 Rem at the 1 mile exclusion distance.

14.1.2.8.3 Results of Complete Loss of All Station Power Analysis

The second power loss considered is the hypothetical case where all station power except the station batteries is lost. The sequence of events and the evaluation of consequences relative to this accident are:

a. A loss of power results in gravity insertion of the control rods and trip of the turbine stop valves.

b. The main steam safety valves actuate after the turbine stop valves trip and prevent excessive temperatures and pressures in the reactor coolant system. These safety valves will close after 20 seconds when the flow through the turbine bypass valve is sufficient to relieve excess steam and provide for decay heat removal. The potential personnel thyroid exposure from this accident is only 1/6 of that discussed under the loss-of-load conditions in 14.1.2.8.2, as the safety valves relieve for only 20 seconds compared to two minutes for a loss-of-load condition.

c. The reactor coolant system flow decays without fuel damage occurring. Decay heat removal after coastdown of the reactor coolant pumps is provided by the natural circulation characteristics of the system. This capability is discussed in the loss-of-coolant-flow evaluation (14.1.2.6).

d. A turbine-driven emergency feedwater pump is provided to supply feedwater any time the main feed pumps cannot operate. The emergency feed pump takes suction from the condenser hotwell and the upper surge tank (condensate storage) and is driven by steam from either or both steam generators. The emergency feedwater system is discussed in section 10.2.2. The controls and auxiliary systems for the emergency feed pump operate on d-c power from the battery-backed d-c bus.

e. Each unit condenser cooling water system is arranged to provide cooling water even in the unlikely event that all power is lost. This is
24. accomplished by the arrangement shown in Figure 9-7. Condenser cooling water intake is obtained from a point below minimum level in the Little River branch of Lake Keowee. Circulating water pumps are provided during normal operations to overcome line friction and to discharge condenser flow to Lake Keowee. An emergency line, discharging

to the Keowee Dam tailrace and normally closed by a power-to-close valve, is provided to obtain cooling water circulation by gravity head even if power is lost.

The features described above would permit decay heat cooling of the unit for an extended period of time following a hypothetical complete loss of electric power.

The above evaluation demonstrates the features incorporated in the design to sustain loss of power conditions with only the station batteries to operate system controls. Immediate operation of the emergency feedwater pump and the emergency condenser cooling water system is not of critical nature. Each reactor can sustain a <u>complete electric power loss without emergency cooling</u> for about 23 minutes before the steam volume in the pressurizer is filled with reactor coolant. These 23 minutes are derived as follows:

(1) Steam generators evaporated to dryness	9 min
(2) Pressurizer relief valves open	5 min
(3) Pressurizer fills with water(due to reactor coolant system expansion)	9 min
	23 min

Beyond this time reactor coolant will boil off, and an additional 83 minutes will have elapsed before the boiloff will start to uncover the core.

With the turbine-driven emergency feedwater pump but without either electric power
24. or presence of Lake Keowee, 20 hours are available before exhausting the supply of
stored condensate pumped into steam generators and steam vented to the atmosphere.

The emergency feedwater pump and the emergency condenser cooling water system are automatically actuated on loss of power within this period of time. As noted above, gravity flow of emergency condenser cooling water will be available from Lake Keowee with all power lost. Accordingly, core protection is insured for the unlikely condition of total loss of station electric power.

14.1.2.9 <u>Steam Line Failure</u>

14.1.2.9.1 Identification of Cause

Analyses have been performed to determine the effects and consequences of loss of secondary coolant due to a 34-in. double-ended, steam line rupture. Mechanical effects are evaluated in 4.3.4.

The criteria for unit protection and the release of fission products to the environment are as follows:

a. The core will remain intact for effective core cooling, assuming minimum tripped rod worth with a stuck rod.

b. No steam generator tube loss of primary boundary integrity will occur due to the loss of secondary side pressure and resultant temperature gradients.

c. Doses will be within 10CFR100 limits.

14.1.2.9.2 Analysis and Results

Accident Dynamics

The loss of secondary coolant due to a failure of a steam line between the steam generator and the turbine causes a decrease in steam pressure, thus placing a demand on the control system for increased feedwater flow. The turbine control valves will open to maintain power generation. Increased feedwater flow, accompanied by steam flow through the turbine stop valves and the break, lowers the average reactor coolant temperature. The limiting action in this condition is the 102 per cent power demand to the rod drive control system.

The steam line failure was analyzed for the maximum break size to determine the maximum cooling effects on reactor coolant and related core reactivity effects. In addition, two plant conditions were considered; one reflects the beginning-of-core-life (BOL) condition where the moderator coefficient is positive, the other represents end-of-core-life (EOL) condition with a negative moderator coefficient.

The rate of reactor system cooling following a steam line break accident is also a function of the steam generator water inventory available for cooling. The steam generator inventory increases with power level. The inventory at rated power is 55,000 lb. This decreases linearly to 20,000 lb at 15 per cent of rated power. The larger inventory results in a greater mass available for cooling.

A steam line rupture of small area causes a relatively slow decrease in steam pressure. With a positive moderator temperature coefficient (BOL condition) the reactor power will decrease when the control system reaches the power demand limit because of the continuing temperature decrease. The reactor will then trip on low reactor coolant system pressure, causing a turbine trip. It is impossible for the reactor in this condition to return to criticality.

When the moderator temperature coefficient is negative, the reactor power will increase with decreasing average coolant temperature. This will cause control rod insertion in an attempt to limit reactor power to 102 per cent. The decrease in coolant temperature causes a reduction in reactor coolant system pressure. The reactor trips on low reactor coolant pressure or high neutron power. Following reactor trip, the turbine stop valves and feedwater control valves close. The steam generator in the steam loop associated with the rupture blows dry, and decay heat is removed by the unaffected steam generator by steam flow through the turbine bypass valve.

The analysis for the maximum break size (34 in. diameter) at rated power shows results similar to those discussed above, but represents the worst condition for a steam line rupture accident. With a positive moderator coefficient (BOL) the reactor will trip on low pressure, making it impossible for the reactor to return to criticality because of continuing decrease in reactor coolant temperature. A negative moderator coefficient (EOL) would result in a reactor power increase with decreasing average reactor coolant temperature. A controlled cooldown rate can be established by feedwater isolation.

Analysis Inputs

The following initial conditions are assumed:

a. Before the accident, the reactor is operating at 100% power (2,568 MWt) with a high flux trip of 114%.

b. Doppler coefficient (EOL) is -1.2×10^{-5} $(\Delta k/k)/F$.

c. Moderator coefficient is -3.0×10^{-4} $(\Delta k/k)/F$, corresponding to EOL.

d. Rod drop starts 0.3 s after the trip point is reached, and 2/3 insertion occurs in 1.4 s. The tripped rod worth (3.46% $\Delta k/k$) corresponds to the minimum worth available with the maximum worth rod stuck out at EOL.

The following sequence of events occurs following a steam line rupture:

a. As a result of reactor trip, the turbine stop valves close.

b. Feedwater flow is held at 100% until reactor trip occurs. The trip also causes the closing of the feedwater main control valves and feedwater startup valves.

c. The emergency feedwater pump is started on a loss of discharge pressure on the main feedwater pumps (800 psia).

d. Auxiliary feedwater isolation valves open and normal feedwater isolation valves close. (See Figure 10-2).

e. The feedwater isolation valves upstream of startup valves close when the emergency feedwater pump has started and auxiliary feedwater isolation valve is opening.

f. The feedwater startup and main control valves follow control system requirements for steam generator minimum level control.

g. The unaffected steam generator is isolated, on the steam side, by the automatic closing of the turbine stop valves. The turbine bypass valve opens on high pressure. Steam flow through this valve to the condenser is the means of removing core decay heat.

h. The operator insures that the feedwater valves are closed and remain closed on the affected steam generator.

Evaluation Results

After a steam line rupture both steam generators blow down at the same rate until a reactor trip occurs (6 seconds), which closes the turbine stop valves and feedwater startup and main control valves on both steam generators. The unaffected steam generator is isolated, resulting in a pressure increase until the turbine bypass valve opens. This bypass valve remains open until the reactor coolant temperature goes below 550 F at which time this valve closes.

The steam generator with the assumed break will continue to blow down after the turbine stop valves and feedwater valves are closed (see Figure 10-1). The operator insures that the startup and main control valves in this generator remain closed by switching the feedwater valve controller to manual. The steam generator blows dry approximately 50 seconds after the steam line rupture has occurred. Since this steam generator is dry there is no means by which heat can be removed from the reactor coolant through this generator.

The unaffected steam generator (that has been isolated on the steam side) has the capability of removing core decay heat by venting steam through the turbine bypass valve. When the reactor coolant temperature is above 550 F, the turbine bypass valve will remain open allowing up to 25 per cent steam flow, which is more than adequate to remove core decay heat. With continuing steam flow through the turbine bypass valve, the steam generator downcomer water level will reach 2 feet, at which time the feedwater control, which is left in automatic, will open the feedwater startup and main control valves to meet the 2 ft downcomer minimum level control system requirement. Sufficient feedwater to satisfy decay heat requirements can be obtained from the feedwater pumps or the emergency feedwater pump.

Figure 14-20 shows the response of the reactor coolant system for an assumed 34-in. double-ended steam line rupture. Initially, both steam generators blow down until a high flux reactor trip occurs with a maximum thermal power of 106% of rated power. The reactor coolant temperature leaving the unaffected steam generator increases after the turbine stop valves close as a result of pressure recovery and a reduction of feedwater flow. The coolant temperature leaving the affected steam generator decreases until it has blown dry (50 seconds), at which time it approaches the inlet temperature. Since the unaffected steam generator turbine stop valves are closed, and the steam generator with the rupture is dry, the reactor coolant system temperature can only be lowered as a result of the steam flow from the isolated steam generator through the turbine bypass valve. At 108 seconds after the rupture, thermal equilibrium is reestablished; i.e., the heat removal rate (steam flow through the turbine bypass valve) is equal to the heat input (core decay heat). The maximum cooling rate occurs during the first 10 seconds of blowdown. After the reactor trip the core remains 0.4% $\Delta k/k$ subcritical.

The accident described above has also been analyzed considering no operator action. The affected steam generator blows dry 50 seconds after the rupture has occurred, at which time the minimum level controller has opened the startup and main control valves supplying feedwater from the main feedwater pumps. In addition, the emergency feed pump has started and is supplying feedwater at 90 F to the auxiliary header. The resulting cooling effect returns the reactor to 35% of rated power in approximately 65 seconds. Assuming a nominal tripped rod worth of 5.66% $\Delta k/k$ with the maximum rod stuck out, the reactor will return to less than 25% rated power in about 350 seconds. If the nominal tripped rod worth is used without consideration for the stuck rod, the reactor will remain subcritical.

During the first minute following the break, the average tube temperature in the steam generator remains above the shell temperature (Figure 14-21). Since thermal equilibrium is established, the average reactor coolant temperature will remain near the saturated temperature corresponding to the pressure at which the turbine bypass valve is set. Therefore, the tube-to-shell temperature difference will approach zero. The resulting tube stresses will remain less than 7,350 psi, corresponding to a pressure difference of 2,200 psi.

14.1.2.9.3 Environmental Consequences

The environmental consequences from this accident are calculated by assuming that:

a. The unit has been operating with a 1-gpm steam generator tube leak.

b. The unit has been operating with 1 per cent defective fuel rods.

c. The steam line break occurs between the reactor building and a turbine stop valve.

d. Reactor coolant leakage into the steam generator continues unabated for 3 hours before the reactor coolant system can be cooled down and the leakage terminated.

With these assumptions, the iodine inventory in the steam generator containing the 1-gpm tube leak is 0.018 dose equivalent curie I-131. The steam line break is assumed to result in the release of the activity contained in the steam generator inventory, the activity contained in 12,800 lb feedwater, the activity contained in 35,500 lb of feedwater in the feedwater line, and the activity contained in 180 gal. reactor coolant. The iodine, a total of 3.11 dose equivalent curies I-131, primarily resulting from continued steam generator leakage in the 3 hours following the accident, is assumed to be released directly to atmosphere where it mixes in the wake of the reactor building. Atmospheric dilution is calculated using the 2-hour ground release dispersion factor of 1.16×10^{-4} s/m^3 as developed in 2.3. With these assumptions, the total in-
18. tegrated dose to the thyroid is 0.19 Rem at the exclusion distance. The corresponding whole body dose is 0.002 Rem.

14.1.2.9.4 Building Pressure

The effect of a steam line rupture inside the reactor building has been evaluated by conservatively assuming an instantaneous release to the reactor building of the energy associated with this accident. The mass and energy releases for the steam generator in this analysis are:

		Mass, lb	Energy Btu x 10^{-6}
3.	Steam Generator (fouled condition)	62,600	36.0
	Feedwater Flow (6 s full flow plus coastdown to 0% flow at 16 s)	12,800	5.6
3.	Reactor Coolant System Energy Transferred	--	76.0
	Available Mass in Feedwater Line Between Feedwater Control Valves and Steam Generator	35,500	15.5
3.		110.900	133.1

Based on the above, a single steam generator release would result in about a 13 psi rise in reactor building pressure.

14.1.2.9.5 Conclusions

This analysis has shown that the reactor trips and remains subcritical with no stuck rod. With a stuck rod and prompt operator action, the reactor also remains subcritical. With a stuck rod and without operator action, the reactor returns to low power level.

The initial blowdown results in a maximum thermal power of 106 per cent of rated power; therefore, no fuel damage will occur. The maximum temperature differential that occurs in the steam generator does not produce excessive stresses and steam generator integrity is maintained. The environmental doses are within acceptable limits.

14.1.2.10 <u>Steam Generator Tube Failures</u>

14.1.2.10.1 Identification of Accident

The environmental effects associated with steam generator tube leakage and subsequent releases to the environment are evaluated in the preceding sections. An evaluation has also been performed for the complete severance of a steam generator tube. For this occurrence, the activity contained in the reactor coolant would be released to the secondary system. Radioactive gases and some of the radioactive iodine would be released to the atmosphere through the condenser air removal system.

14.1.2.10.2 Analysis and Results

In analyzing the consequences of this failure, the following sequence of events is assumed to occur:

a. A double-ended rupture of one steam generator tube occurs with unrestricted discharge from each end.

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b. The initial leak rate, approximately 435 gpm, exceeds the normal makeup of 45 gpm to the reactor coolant system, and system pressure decreases. No operator action is assumed, and a low reactor coolant system pressure trip will occur in about 8 minutes.

c. Following reactor trip, the reactor coolant system pressure continues to decrease until high-pressure injection is actuated at a pressure of 1,500 psig. The capacity of the high-pressure injection is sufficient to compensate for the leakage and maintains both pressure and volume control of the reactor coolant system. Thereafter, the reactor is assumed to be cooled down and depressurized at the normal rate of 100 F per hour.

d. Following reactor trip, the turbine stop valves will close. Since a reactor coolant to secondary system leak has occurred, steam line pressure will increase, opening the steam bypass valves to the condenser. The bypass valves actuate at a lower pressure than do the steam safety valves. The reactor coolant that leaks as a result of the tube failure is condensed in the condenser. Only the fission products that escape from the condensate are released to the atmosphere.

e. The reactor coolant system is cooled and depressurized until the affected steam generator can be isolated by closing the steam bypass isolation valves. Cooldown continues with the unaffected steam generator until the temperature is reduced to 250 F. Thereafter, cooldown to ambient conditions is continued using the low pressure injection system.

f. At the design cooling rate of 100 F per hour, depressurization of the reactor coolant system to the steam line safety valve set point requires approximately 34 minutes. During this time period, 1,980 ft^3 of reactor coolant leaks to the secondary system. This leakage
8. corresponds to approximately 10,600 curies of xenon-133 if the reactor has been operating for 200 days with 1 per cent of the fuel pins in the core defective.

The radioactivity released during this accident is discharged through the turbine bypass to the condenser and then out the unit vent. A partition factor of 10^4 is assumed for iodine in the condenser.[1,2] Noble gases are assumed to be released directly to the unit vent. Atmospheric dilution is calculated using the 2-hour elevated release dispersion factor of 3.35×10^{-5} s/m^3 developed in 2.3. The total dose to the body from all the xenon and krypton
8. released is only 0.023 Rem at the 1 mile exclusion distance. The corresponding dose to the thyroid at the same distance is only 0.00034 Rem.

14.2 STANDBY SAFEGUARDS ANALYSIS

14.2.1 SITUATIONS ANALYZED AND CAUSES

In this section accidents are analyzed in which one or more of the protective barriers are not effective and standby safeguards are required. All accidents evaluated are based on the core power level of 2,568 MWt. Table 14-3 summarizes the potential accidents studied.

Table 14-3
Situations Analyzed and Causes

	Event	Analysis Assumptions	Effect
1.	Fuel Handling Accidents	Gap activity is released from the outer row of fuel rods in one assembly (operated at 22.8 MWt for 930 days and then zero power for 3 day), while in spent fuel storage pool. No retention of noble gases and only 99% retention of iodine is considered.	Integrated dose at exclusion distance boundary is 0.43 Rem thyroid and 0.027 Rem whole body.
18.	Rod Ejection Accident	All fuel rods which experience DNB are assumed to release their total gap activity to the reactor coolant (following operation with 1% defective fuel).	Some fuel clad failure. Two-hour dose at exclusion distance is 0.19 Rem thyroid.
18.	Loss-of-Coolant Accident	Double-ended rupture of 36 in. diam. reactor coolant system pipe. Release of all gap activity.	No clad melting. Two-hour dose at exclusion distance is 4.6 Rem thyroid.
18.	Maximum Hypothetical Accident	Release of 100% rare gases, 50% iodine, and 1% solid fission products.	Two-hour dose at exclusion distance is 186 Rem thyroid. Thirty-day dose at 6 mile low population zone distance is 144 Rem thyroid.

14.2.2 ACCIDENT ANALYSES

14.2.2.1 Fuel Handling Accidents

14.2.2.1.1 Identification of Accident

Spent fuel assemblies are handled entirely under water. Before refueling, the reactor coolant and the fuel transfer canal water above the reactor are increased in boron concentration so that, with all control rods removed, the k_{eff} of a core is no greater than 0.99. In the spent fuel storage pool, the fuel assemblies are stored under water in storage racks having an eversafe geometric array. Under these conditions, a criticality accident during refueling is not considered credible. Mechanical damage to the fuel assemblies during transfer operations is possible but improbable. The mechanical damage type of accident is considered the maximum potential source of activity release during refueling operations.

14.2.2.1.2 Analysis and Results

The fuel assembly is assumed to have operated for 930 days at 22.8 MWt. The
1. reactor is assumed to have been shut down for 72 hours, which is the minimum
time for reactor coolant system cooldown, reactor closure head removal, and removal of the first fuel assembly. It is further assumed that the entire outer row of fuel rods in the assembly, 56 of 208, suffers mechanical damage to the cladding. Since the fuel pellets are cold, only the gap activity is released. The fuel rod gap activity is calculated using the escape rate coefficients and calculational methods discussed in 11.1.1.3.

The gases released from the fuel assembly pass upward through the spent fuel storage pool water prior to reaching the auxiliary building atmosphere. As a minimum, the gases pass through at least 9 ft of water. Although there is experimental evidence that a portion of the noble gases will remain in the water, no retention of noble gases is assumed. In experiments whereby air-steam mixtures were bubbled through a water pond, Diffey, et al.,[3] demonstrated decontamination factors of about 1,000 for iodine. Similar results for iodine were demonstrated by Barthoux, et al.,[4] and predicted by Eggleton.[5] Based on these references, 99 per cent of the iodine released from the fuel assembly is assumed to remain in the water. The iodine activity released to the auxiliary
1. building atmosphere is therefore 25.2 dose equivalent curies of I-131. The
noble gas activity released is:

Table 14-4
Fuel Handling Accident; Noble Gas Release

Isotope	Activity Released, Dose Equivalent Curies
Kr-85	1.63×10^3
Xe-131m	1.64×10^2
1. Xe-133m	0.93×10^2
Xe-133	1.36×10^4

The activity is assumed to be released as a puff from the unit vent. Atmospheric dilution is calculated using the 2-hour elevated release dispersion factor of 3.35×10^{-5} s/m^3 developed in Section 2.3. The total integrated dose to the whole body at the 1 mile exclusion distance is 0.0 27 Rem, and the
1. thyroid dose at the same distance is 0.43 Rem. In evaluating the sensitivity of this analysis, the thyroid dose at the site boundary is directly proportional to the quantity of iodine released. For example, if only 90 per cent retention of iodine is assumed by the spent fuel storage pool water, the dose at the exclusion distance is increased by a factor of 10. The dose from this increased iodine release would still be a factor of 50 below the 10 CFR 100 guidelines.

16. 14.2.2.2 <u>Rod Ejection Accident</u> (Reference Supplement 9 Revisions for Oconee 3)

14.2.2.2.1 Identification of Accident

Reactivity excursions initiated by uncontrolled rod withdrawal (14.1) were shown to be safely terminated without damage to the reactor core or reactor coolant system integrity. For reactivity to be added to the core at a more rapid rate, physical failure of a pressure barrier component in the control rod drive assembly must occur. Such a failure could cause a pressure differential to act on a control rod assembly and rapidly eject the assembly from the core region. The power excursion due to the rapid increase in reactivity is limited by the Doppler effect and terminated by reactor protection system trips.

Since control rod assemblies are used to control load variations only and boron dilution is used to compensate for fuel depletion, only a few control rod assemblies are inserted (some only partially) at rated power level. Thus, the severity of a rod ejection accident is inherently limited because the amount of reactivity available in the form of control rod worth is relatively small.

The criterion for reactor protection in this assumed accident is that the reactor will be operated in such a manner that a control rod ejection accident will not further damage the reactor coolant system.

a. <u>Accident Bases</u>

Using an analytical method based on diffusion theory (3.2.2.2.1) the worth of the most reactive control rod assembly in each rod group was determined for different control rod configurations.

The maximum rod worths and other important parameters used in the study are as follows:

Table 14-5
Rod Ejection Analysis Parameters

Worth of Ejected Rod, % Δk/k	
Rated Power, No Xenon	0.46
Rated Power, With Xenon	0.36
Hot, Zero Power, Critical	0.56
Rod Ejection Time, s	0.150
9. Rated Power Level, MWt	2,772
Reactor Trip Delay Time, s	
High Flux Trip	0.3
High Pressure Trip	0.5
Trip Time to 2/3 Insertion, s	1.4

The tripped rod worth used corresponds to the minimum worth available with the maximum worth rod stuck out at BOL and EOL.

The severity of the rod ejection accident is dependent upon the worth of the ejected rod and the reactor power level. The control rod group of greatest worth is the first of the entire rod pattern to be withdrawn. The maximum worth of a rod in this group can be as high as 2.5 per cent Δk/k but would only have this worth when the reactor was subcritical. The details of the control rod worth calculations and the methods of selecting the number of control rods in each group are presented in 3.2.2 and 7.2.2.1.2.

When the reactor is subcritical, the boron concentration is maintained at a level which insures that the reactor is at least 1 percent subcritical with the control rod of greatest worth fully withdrawn from the core. Thus, a rod ejection will not cause a nuclear excursion when the reactor is subcritical and all the other rods are in the core.

As criticality is approached, the worth of the remaining rods decreases so that at criticality the maximum reactivity addition from a rod ejection would be 0.56 per cent Δk/k.

The rod worth continues to decrease as rated power is attained. Before equilibrium xenon is established, the total pattern worth remaining in the core at rated power is 2.8 per cent Δk/k, and the greatest single control rod worth is 0.46 per cent Δk/k. At equilibrium xenon the pattern worth is 1.8 per cent Δk/k and the maximum rod worth is 0.36 per cent Δk/k. A detailed analysis has been performed at worths up to 0.7 per cent Δk/k, however, to show the large margin that exists between the actual rod worths and those worths needed to approach any failure thresholds.

A rod must be fully inserted in the core to have the foregoing reactivity worth values. Assuming that the failure occurs so that the pressure barrier no longer offers any restriction to the ejection and that there is no viscous drag force limiting the rate of ejection, the control rod travel time to the top of the active region of the core is calculated to be 0.176 s. Since most of the reactivity is added during the central 75 per cent of this travel, only this distance is used in the analysis, resulting in an ejection time of 0.15 second for the analysis.

b. Fuel Rod Damage

The consequences of a rod ejection accident are largely dependent upon the rate at which the thermal energy resulting from the nuclear excursion is released to the coolant. If the fuel rods remain intact while the excursion is being terminated by the negative Doppler coefficient and by reactor trip, then the energy release rate is limited by a relatively low surface-to-volume ratio for heat transfer. The energy stored in the fuel rods will then be gradually released to the coolant (over a period of several seconds) at a rate which poses no threat to the integrity of the reactor coolant system. However, if the magnitude of the nuclear excursion is such that the fuel rod cladding does not remain intact, then fuel and clad may be dispersed into the coolant to such an extent as to cause a significant increase in the heat transfer rate.

Power excursions caused by reactivity disturbances of the order of magnitude occurring in rod ejection accidents could lead to three potential modes of fuel rod failure. Failure by the first mode occurs when internal pressures developed in the fuel rod are insufficient to cause cladding rupture, but subsequent heat transfer from fuel to cladding raises the temperature of the cladding and weakens it until local failure occurs. "Departure-from-nucleate-boiling" (DNB) usually accompanies and contributes to this mode of failure, and little or no fuel melting would be expected. In this mode of failure, fuel fragmentation is usually only minor, and any dispersal of fuel to the coolant would occur very gradually, with system contamination being the worst probable consequence.

The second failure mode occurs when significant fuel melting causes a rapid increase in internal fuel rod pressure (*) which, combined with clad loss of strength at higher temperatures, causes the fuel rod clad to rupture. Some fuel vaporization may occur, contributing to the pressure buildup. Considerable fragmentation and dispersal of the fuel would be expected in this mode.

The third and most serious mode of fuel rod failure occurs when, as a result of a very large and rapid reactivity transient in which

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(*)The increase in volume associated with the melting of UO_2 is 9.6 per cent.[6]

there is insufficient time for heat to be transferred from fuel to cladding, extensive fuel melting followed by vaporization occurs. Destructive internal pressures can be generated without increasing cladding temperatures significantly in this mode.

In evaluating the effects of the failure modes discussed above, two failure thresholds are considered. The first is associated with a gradual, and usually minor, cladding failure and may be approximately defined by the minimum heat flux for DNB at the cladding surface. The second failure threshold, defined as the enthalpy threshold for prompt fuel failure with significant fragmentation and dispersal of fuel and cladding into the coolant, is used to describe the energy required to cause failure by either the second or the third failure mode described above.

A correlation of the results of different experiments conducted on Zircaloy-2-clad UO_2 fuel rods at TREAT[7] has been interpreted by the experimenters to show a threshold at 280 cal/g of fission energy input. That is, below this value the fuel rod can be expected to remain intact, and above this value fragmentation can be expected. The enthalpy corresponding to the melting point of UO_2 is about 260 cal/g[8], and the heat of fusion is at least 78 cal/g[9]. Thus the 280 cal/g represents a condition where only part of the fuel is molten. Also of interest as a probable indication of the degree and rapidity of fuel and cladding dispersal are the measurements of pressure rise rates in the autoclave in the TREAT experiments.[7] Preliminary analysis indicates that there is only a modest pressure rise up to an energy input of 400 cal/g. Above 500 cal/g, however, there is a very definite pressure pulse. Thus between 400 and 500 cal/g there is a transition which probably corresponds to the change from the second to the third failure mode discussed previously. A fuel failure threshold of 280 cal/g, at the pellet radius corresponding to the average temperature of the hottest fuel pellet, has been used in this study to define the extent of fuel failure.

In computing the average enthalpy of the hottest fuel pellet during the excursion for the rated power cases, it is assumed that no heat is transferred from the fuel rod between the time the accident is initiated and the time when the neutron power returns to the rated power level. For the zero-power cases, the enthalpy increase was based on the peak value of the average fuel temperature. In all cases the average enthalpy rise -- from the integrated energy or the fuel temperature traces -- is multiplied by the maximum peaking factor to obtain the enthalpy increase in the hottest fuel pellet.

The latest correlation of the ANL TREAT data[7] for the meltdown experiments on Zircaloy-2-clad UO_2 fuel rods shows the threshold for the zirconium-water reaction to be 210-220 cal/g energy input. A conservative threshold value of 200 cal/g is used in this study.

In calculating the volume of the core that experiences burnout in a given rod ejection accident, it is assumed that any DNB conditions result in burnout for each rod where the DNB occurs. DNB in a rod ejection transient is assumed to occur whenever the peak thermal power of a given fuel rod exceeds the peak at steady-state conditions which could result in a DNB, which in turn is assumed to occur for a DNBR of 1.3 using the W-3 correlation.

In determining the environmental consequences from this accident, an even more conservative approach is taken in computing the extent of DNB experienced in the core. All fuel rods that undergo DNB to any extent are assumed to experience cladding failure with subsequent release of all the gap activity. Actually, most of the fuel rods will recover from DNB and no fission product release will occur. The fuel rods that experience DNB at BOL are assumed to have EOL gap activities.

14.2.2.2.2 Method of Analysis

A B&W digital computer program has been used to analyze the rod ejection accident. This program agrees to within a few percent in all cases with CHIC-KIN.[10] The B&W program is a point kinetics model with a primary loop and pressurizer model. The core heat transfer model allows for up to 30 radial mesh points in the fuel and clad, and the mesh size can be different in the two regions. The model accounts for the gap conductivity and film coefficient of heat transfer. Reactivity feedback is calculated in each mesh point and in the coolant and is weighted for inclusion in the kinetics simulation. The thermal properties are input separately for each mesh point but remain constant with time. The loop model includes a simulation of the steam generator which can have a nonlinear heat demand input on the secondary side. Trip action is initiated on high or low reactor coolant system pressure or on high neutron flux. Decay heat can be taken into account as well. This code was used to calculate the neutron and thermal power, integrated energy, reactivity components, pressure, and fuel rod and loop temperatures. Six delayed neutron groups are considered. The control rod trip is represented by a 25-segment curve of reactivity insertion during trip versus time, obtained by combining the actual rod worth curve with a rod velocity curve. Nominal values for the various nuclear and physical parameters used as inputs are listed in Table 14-6.

As a check on the point kinetics calculation, the rod ejection accident was also analyzed for a limited number of cases using the exact, one-dimensional, space-and-time dependent WIGL2 digital computer program.[11] The point kinetics model assumes that the flux shape remains constant during a transient. This flux shape contains peaking factors which reflect unusual rod patterns such as the flux adjacent to a position where a high worth rod has been removed. Therefore, these point kinetics peaking factors are much higher than any that would actually occur in the core during normal operation. The purpose of using an exact space-time calculation is to find the flux shape during a transient. But to have a transient where a rod is ejected from the core, one must start with a flux shape that is necessarily depressed in the region of the ejected rod. In fact, the higher the worth of the rod, the more severe becomes the depression. This flux depression also causes a fuel temperature depression. When the rod is ejected from this position, the flux quickly assumes a shape that shows some local peaking.

Table 14-6
Nominal Values of Input Parameters for Rod Ejection Accident Analysis (Core Power 2,568 MWt)

	BOL	EOL
Delayed Neutron Fraction, β_{eff}	0.0071	0.0053
Neutron Lifetime, μs	24.8	23.0
Moderator Coefficient, (Δk/k)/F	$+0.5 \times 10^{-4}$	-3.0×10^{-4}
Doppler Coefficient, (Δk/k)/F	-1.17×10^{-5}	-1.33×10^{-5}
Coolant Inlet Temperature, F	554	554
Initial System Pressure, psia	2,200	2,200
Total Nuclear Peaking Factor, F_q	3.24	2.92
Average Fuel Temperature of Average Pellet, F	1,540	1,670
Average Fuel Temperature of Hottest Pellet, F	2,810	2,685
Center Line Temperature of Hottest Pellet, F	3,759	3,553

Results from WIGL2 indicate that for rod worths greater than 0.2 per cent Δk/k this local peaking is in excess of the maximum peaking applied to the point kinetics results. However, when this "exact" peaking is applied to a region initially at depressed fuel temperatures, as it is in the case of the region adjacent to the ejected rod, the resultant energy deposited in this region causes a lower peak temperature and peak thermal power than does applying a lower maximum peaking factor to an undepressed peak power region. The result is that this local region simulated in the WIGL2 code actually undergoes a less severe transient than the hottest fuel rod assumed in the point kinetics model. As seen in Table 14-7, this result is uniformly true for all rod worths up through 0.5 per cent Δk/k.

Thus it can be seen that the space-time dependent code gives a less conservative treatment of the accident analysis than does the point kinetics code.

For certain cases where the ejected rod has a low worth, or where at least one reactivity coefficient is very negative, or the initial power level is low, there is considerable pressure buildup in the reactor coolant system because of the increased heat being added to the coolant with no increase in heat demand. Many of these transients never reach the overpower trip point. For this class of possibilities, the high-pressure trip must be relied on, and this is incorporated in the calculation.

Table 14-7
Comparison of Space-Dependent and Point Kinetics Results on the Fuel Enthalpy

Rod Worth (%Δk/k)	BOL Rated Power Peak-to-Average Values WIGL2	Point Kinetics	Fuel Enthalpy (cal/g) WIGL2	Point Kinetics
0.2	4.2	3.24	72	141
0.3	4.8	3.24	86	149
0.4	5.4	3.24	109	159
0.5	6.0	3.24	143	172

14.2.2.2.3 Analysis and Results

a. Zero Power Level

This analysis was performed at 10^{-3} rated power. For the nominal case of a 0.56 per cent Δk/k rod ejection, the neutron power reaches 60 per cent and the thermal power peaks at 39 per cent for BOL; the excursion is terminated by high-pressure trip. The nominal EOL case trips on high flux because it is a faster transient. This results in peak neutron and thermal powers of 165 and 11 per cent, respectively. No DNB and no fuel damage would result from these transients.

A sensitivity analysis has been performed around these two cases in which the Doppler and moderator coefficients, trip delay time, and rod worth were varied. Figure 14-22 shows the peak neutron power as a function of ejected rod worth from 0.2 to 0.8 per cent Δk/k. The curve shows two distinct parts corresponding to worths less than and values near to and greater than β. Figure 14-23 shows the corresponding results for the peak thermal power. It is seen that for rod worth values near prompt critical, the period is small enough to carry the transient through the high neutron flux trip. For lower values the pressure trip is relied on. In no case does the thermal power exceed 70 per cent. Therefore, no DNB would occur.

Figure 14-24 shows that the peak enthalpy in the fuel for the rod worths in the range being evaluated never exceeds 75 cal/g. Therefore, no threshold for damage is approached.

Figures 14-25 and 14-26 show the peak neutron and thermal power as a function of Doppler coefficient from -0.9 to -1.7×10^{-5} (Δk/k)/F. It is seen that the variation is relatively small. Similar results are shown in Figures 14-27 and 14-28 for the variation of the moderator coefficient from -4.0 to $+1.5 \times 10^{-4}$ (Δk/k)/F. The slope of the curve for 10^{-3} rated power at BOL is the greatest slope for any of the four curves because this case relies on the pressure trip, which makes it a longer transient. It is also steeper because of the effect of the positive moderator coefficient, which is only noticeable in long transients due to the long time constant from fuel to coolant.

Similarly, it is seen that the peak neutron power is higher for the EOL cases in both the Doppler and moderator studies, whereas the peak thermal powers are higher for the BOL cases. This again is because the EOL rod ejection cases are faster and the neutron power overshoots the trip point by a greater margin. It also trips more quickly, however, terminating the transient faster.

Figure 14-29 shows the effect of the trip delay time on the peak thermal power. It is seen that there is very little effect.

b. Rated Power

An analysis was performed for a 0.46 per cent $\Delta k/k$ rod ejection, although this worth is available at rated power only when no xenon is present. For BOL rod ejection, the neutron power peaks at 277 per cent, and the thermal power at 126 per cent. This causes only 0.5 per cent of the core to be in DNB, as described above. A sensitivity study was made around this case and around the same rod worth at EOL. Figures 14-22 through 14-29 show these results.

As seen in Figure 14-23, the peak thermal power shows relatively little change with increased rod worth. The peak neutron power in Figure 14-22 does show a marked change with increased worths, but the thermal effect is small because the transients are rapidly terminated by the Doppler effect. As further evidence of this small thermal effect, the peak fuel enthalpies are given in Figure 14-24. The threshold for the zirconium-water reaction is not reached until values of 0.64 and 0.69 per cent $\Delta k/k$ for BOL and EOL ejected rod worths, respectively, are encountered. These worths are well above any that are considered feasible.

The results of varying the Doppler and moderator coefficients and trip delay time show very little effect on the peak neutron and thermal powers.

The only situation in which DNB occurs is for the related power case at BOL. The results of the DNB calculation are shown in Figure 14-30. For the nominal rod ejection analysis which considers a worth of 0.46 per cent $\Delta k/k$, only 0.5 per cent of the core volume is in DNB. This corresponds to 4.1 per cent of the rods.

14.2.2.2.4 Energy Required to Produce Further Reactor Coolant System Damage

The reactor vessel has been analyzed to estimate the margin that exists between the rod worths assumed for the calculated rod ejection accident transients and those worths that could initiate reactor coolant system failure. The pressure vessel material is SA-533 Grade B steel. Table 14-8 lists the values used in this analysis. The radial deformation which is assumed to represent failure of the vessel is 50 per cent of the total elongation, or 0.13 in./in. To calculate the weight of an explosive charge required to reach 50 per cent elongation, the vessel is simulated by a single cylinder with the same OD as the actual vessel, but with an increased thickness to account for the thermal shield and core barrel.

Table 14-8
Reactor Vessel Parameters

Vessel Temperature, F	600
Yield Strength (0.2% Offset), psi	55,000
Ultimate Strength, psi	80,000
Ultimate Strain (ε_u), %	26
Strain Energy (E_s) per Unit Volume Up to a Strain Equal to 1/2 Ultimate Strain, in.-lb/in.^{3}	8,000
Strain Energy (E_s) per Unit Volume Up to Ultimate Strain, in.-lb/in.^{3}	17,000
Equivalent Pressure Vessel Dimensions	
OD, in.	188.25
ID, in.	166.69
Thickness, in.	10.78

The expression used for the weight of explosive required to strain the vessel a given amount is[12]

$$W = \left[\frac{1.407E_s(3.41 + 0.117R_i/t)(R_e^2 - R_i^2)^{1.85}}{10^5 w^{-0.85}(1.47 + 0.0373R_i/t)^{0.15}(R_i)^{0.15}}\right]^{0.811}$$

where

W = charge weight (TNT or Pentolite), lb

w = weight density of vessel material, lb/ft^3

R_i = initial internal radius of vessel, ft

R_e = initial external radius of vessel, ft

t = initial wall thickness of vessel wall, ft

E_s = wall strain energy, in.-lb/in.^{3}

Using this formula on the equivalent vessel, the required weight of explosive charge was calculated. The results of this calculation indicate that 1,410 pounds of TNT would strain the mid-meridian ring up to the 50 per cent ε_u, i.e., 0.13 in./in. The 1,410 pounds of TNT have an energy equivalent of 6.74 x 10^8 cal.

An analysis of ejected rod worths higher than those reported in the preceding sections has been made to estimate the transient required to generate the deformation energy equivalent to 1,410 pounds of TNT. These cases were evaluated to find the amounts of fuel melting and zirconium-water reaction. Using the conservative assumption that all the fuel that exceeds the melting threshold is fragmented, dispersed into the coolant, and quenched to the coolant average temperature, a total thermal energy release can be determined. The conversion of this energy release to an equivalent deformation energy is dependent upon the duration of the release. TNT has an energy release in microseconds, and a deformation conversion efficiency of about 50 per cent. The energy generated during a reactor transient from the zirconium-water reaction and a molten fuel dispersal is in the range from milliseconds to seconds. Thus, the conversion efficiency to deformation energy would be considerably less, and is assumed to be 1/5 that of TNT.(13) Using these figures, the reactor vessel capability is 3.37×10^8 cal, and, under the foregoing assumptions, a reactivity addition of 1.52 per cent $\Delta k/k$ is required to release this much energy to vessel deformation.

14.2.2.2.5 Conclusions

The hypothetical rod ejection accident has been investigated in detail at two different initial reactor power levels: rated power and zero power; both BOL and EOL conditions were considered. The results of the analysis prove that the reactivity transient resulting from this accident will be limited by the Doppler effect and terminated by the reactor protection system with no serious core damage or additional loss of the coolant system integrity. Furthermore, it has been shown that an ejected rod worth greater than 1.52 per cent $\Delta k/k$ would be required to cause a pressure pulse, due to prompt dispersal of fragmented fuel and zirconium-water reaction, of sufficient magnitude to cause rupture of the pressure vessel.

As a result of the postulated pressure housing failure associated with the accident (14.2.2.2.1), reactor coolant is lost from the system. The rate of mass and energy input to the reactor building is considerably lower than that previously reported for the smallest rupture size considered in the loss-of-coolant analysis (14.2.2.3). The maximum hole size resulting from a rod ejection is approximately 1.75 in. This lower rate of energy input results in a much lower reactor building pressure than those obtained for any rupture sizes considered in the loss-of-coolant accident.

The environmental consequences of this accident are calculated assuming that all fuel rods undergoing DNB release all of their gap activity to the reactor coolant. Subsequently, this gap activity and the activity in the reactor coolant from operation with 1% defective fuel pins is released to the reactor building. For the case of a 0.46 per cent $\Delta k/k$ rod ejection from rated power at BOL, 4.1 per cent of the fuel rods are assumed to fail, releasing noble gas activity to the reactor building as shown in Table 14-9.

The iodine released to the reactor building is 6.05×10^4 dose equivalent curies of I-131.

Fission product activities for this accident are calculated using the methods discussed in Section 11.1.1.3. Using environmental models and dose rate calculational methods discussed under the loss-of-coolant accident, the total integrated 2-hour dose at the 1-mile exclusion distance is 0.19 Rem thyroid and
18. 0.001 Rem whole body. The total integrated thyroid dose at the 6-mile low population zone distance is 0.22 Rem for 30-day exposure. These doses are well below the guideline values of 10 CFR 100.

Table 14-9
Noble Gas Release for Nominal Rod Ejection

Isotope	Activity (curies)
Kr-83m	3.69×10^2
Kr-85m	2.51×10^3
Kr-85	2.06×10^4
Kr-87	1.36×10^3
Kr-88	4.48×10^3
Xe-131m	4.04×10^3
Xe-133m	4.79×10^3
Xe-133	4.26×10^5
Xe-135m	1.43×10^3
Xe-135	3.46×10^3

14.2.2.3 Loss-of-Coolant Accident

14.2.2.3.1 Identification of Accident

Failure of the reactor coolant system pressure boundary would allow partial or complete release of reactor coolant into the reactor building, thereby interrupting the normal mechanism for removing heat from the reactor core. If all the coolant were not released immediately, residual heat, fission product heat, and possible heat from chemical reactions would cause the remaining amount of coolant to be boiled off unless an alternate source of coolant were made available. Coolant from the Emergency Core Cooling Systems (ECCS) is available and the fast cooling action provided by the core flooding tank water prevents any significant chemical reactions and destructive core heatup.

14.2.2.3.2 Accident Bases

All components of the reactor coolant system have been designed and fabricated to insure high integrity and thereby minimize the possibility of their rupture. The reactor coolant system, the safety factors used in its design, and the special provisions taken in its fabrication to insure quality are described in Section 4 and in Appendix 1B.

In addition, emergency core cooling is provided to insure that the core will continue to be cooled and will not lose its geometric configuration, even if the reactor coolant system should fail and release coolant. This emergency core cooling standby safeguard is provided by the core flooding system and two full capacity, independent, emergency core cooling strings. These systems are described in detail in Section 6.

The basic design objective for the ECCS equipment is to terminate the temperature
3. transient and maintain core geometry in the loss-of-coolant accident. This would

be accomplished if the temperature transient were terminated before the melting point of the clad was reached. Emergency core injection is provided to limit the temperature transient below the clad melting point so that fuel geometry is maintained to provide core cooling capability. A design limit has been conservatively established to limit the clad temperature transient to 2,300 F or less. Temperatures in excess of this value promote a faster zirconium-water reaction rate, so the termination of the transient near the melting point would be difficult to demonstrate at this time.

In order to evaluate this accident, a range of rupture sizes from small leaks up to the complete severance of a 36-in. ID reactor coolant system line has been evaluated. A detailed core cooling analysis is presented for the complete severance of the 36-in. ID reactor coolant piping. This largest rupture has the shortest blowdown time, which results in the removal of the least amount of stored energy from the core. This rupture, therefore, represents the minimum temperature margin to core damage and places the most stringent requirements on the core cooling equipment. This is shown by analysis of peak temperature conditions for a spectrum of rupture sizes in both the hot and cold leg piping.

The reactor building pressures have been evaluated for a complete spectrum of rupture sizes with the action of core flooding tanks. The peak reactor building pressure occurs for a 5.0 ft^2 rupture rather than for a 36-in. ID (14.1 ft^2) rupture. Rupture sizes smaller than the 36-in. ID break result in longer blowdown times, and the amount of energy transferred to the reactor building atmosphere is increased. As a result, the intermediate break size results in a reactor building pressure slightly greater than that produced by the 36-in. ID rupture. The entire spectrum of large rupture sizes (1.0 ft^2 to 14.1 ft^2) yields pressures that range from 49.1 to 53.3 psig (14.2.2.3.6). Thus, the building design is not strongly influenced by rupture size.

This analysis demonstrates that in the unlikely event of a failure of the reactor coolant system, both the reactor core geometry and the reactor building integrity are maintained. Accordingly, the public would be protected against potential radiation hazards.

14.2.2.3.3 Accident Simulation - Reactor Coolant System

a. Hydraulic Model

Blowdown of the reactor coolant system following an assumed rupture has been simulated by using a modified version of the FLASH[14] code. This code calculates transient flows, coolant mass and energy inventories, pressures, and temperatures during a loss-of-coolant accident. The code calculates inflow from the emergency cooling strings and calculates heat transferred from the core to the coolant.

Modifications were made to FLASH to make the model more applicable to this system. The changes are as follows:

(1) The calculation of reactor coolant pump cavitation was based on the vapor pressure of the cold leg instead of the hot leg water.

(2) Core flooding tanks have been added. Water flow from the core flooding tanks is calculated on the basis of the pressure difference between the core flooding tanks and the point of discharge into the reactor coolant system. The line resistance and the inertial effects of the water in the pipe are included. The pressures in the tanks are calculated by assuming an adiabatic expansion of the gas above the water level in the tank. Pressure, flow rate, and mass inventories are calculated and printed out in the computer output.

(3) Additions to the water physical property tables (mainly in the subcooled region) have also been made to improve the accuracy of the calculations.

(4) A change in the steam bubble rise velocity has been made from the constant value in FLASH to a variable velocity as a function of pressure. The bubble velocity term determines the amount of water remaining in the system after depressurization is complete. For large ruptures, this change in velocity shows no appreciable change in water remaining from that predicted by the constant value in the FLASH code. For smaller ruptures, an appreciable difference exists. The variable bubble velocity is based on data in Reference 15 and is adjusted to correspond to data from the LOFT semiscale blowdown tests.

Test No. 546 from the LOFT semiscale blowdown tests is a typical case for the blowdown through a small rupture area. A comparison of the predicted and experimentally observed pressures is shown in Figure 14-31. Figure 14-32 shows the per cent mass remaining in the tank versus time, as predicted by the code. At the end of blowdown, the predicted mass remaining is 13 per cent. The measured mass remaining is approximately 22 per cent.

(5) An addition was made which provided for simulation of the vent valves in the reactor vessel core support shield.

~~The FLASH code describes the reactor~~ coolant system by the use of two volumes plus the pressurizer volume. The system was grouped into volumes on the basis of the temperature distribution in the system as follows:

Volume 1 includes half of the core water volume, the reactor outlet plenum, the reactor outlet piping, and 55 per cent of the steam generators.

Volume 2 includes half of the core water volume, the reactor inlet plenum and downcomer section, the reactor inlet piping, pumps, and 45 per cent of the steam generators.

Volume 3 represents the pressurizer.

The resistances to flow were calculated by dividing the reactor coolant system into 24 regions and calculating the volume-weighted resistance to flow for a given rupture location based on normal flow resistances. For the double-ended ruptures, all of the leak was assumed to occur in the volume in which that pipe appeared.

The reactor core power was input as a function of time as determined by the CHIC-KIN code in conjunction with the FLASH output. Steam generator heat removal was assumed to cease when the rupture occurred. Stored heat in the reactor coolant system was considered in the analysis.

The modified FLASH code has the capability of simulating injection flow from the core flooding tanks. Reactor vessel filling was calculated by adding the mass remaining in the vessel as predicted by FLASH to the mass injected from the core flooding tanks. This method of calculation is conservative in that condensation of steam by the cold injection water is not taken into account. An analysis using the FLASH code with condensation effects confirms that conservatism is used in this analysis.

Pressure, temperature, mass and energy inventories, and hydraulic characteristics as determined by FLASH, are input into the core thermal code (QUENCH) and the reactor building pressure buildup code (CONTEMPT).[16]

b. Core Thermal Model

The core heat generation and heat transfer to the fluid are dependent upon the blowdown process. The FLASH program includes a core thermal model and the functional relationship between heat transfer and fluid flow. While the FLASH thermal model is acceptable for determining the effect of core heat transfer on the blowdown process, a more extensive simulation is necessary for evaluation of the core temperature transient.

A digital computer program (QUENCH) was developed to simulate the core thermal transient for the period beginning with the initiation of the leak and ending after the core temperature excursion had terminated.

The QUENCH program determines the clad and fuel temperature as a function of time after a rupture with provisions for a time dependent tabular input of heat transfer coefficient, sink temperature, and power. It also includes the effects of heat generation from the exothermic zirconium-water reaction based on the parabolic rate law.

The basic model structure provides for up to 50 equal-volume core regions with input provisions to allow any choice of power distribution. The model may be used to simulate the entire core or any subdivision of the core.

The following parabolic law for the zirconium-water reaction equation[17] with the following constants is simulated for each of the regions:

$$\frac{dr}{dt} = \frac{K}{(r_o - r)} \exp - \frac{\Delta E}{RT}$$

where

r = radius of unreacted metal in fuel rod (cm)

r_o = original radius of fuel rod (cm)

t = time (s)

K = rate law constant (0.3937 cm^2/s)

ΔE = activation energy (45,500 cal/mole)

R = gas constant (1.987 cal/mole K)

T = temperature, K

The zirconium-water reaction heat is assumed to be generated completely within the clad. The heat necessary to increase the steam temperature from the bulk temperature to the reaction temperature is transferred from the clad at the point of reaction. Although the above equation implies no steam-limiting, the program does have provision for steam rate-limiting to any degree desired. However, no steam-limiting of the reactions has been assumed. All heat from beta and gamma sources and 97.3 per cent of the neutron heat is assumed to be generated within the fuel according to the pre-accident power distribution after infinite irradiation. Within each of the regions there is a single fuel node and a single clad node with simulation of thermal resistance according to the normal fuel rod geometry.

The surface heat transfer coefficient input data are based on results obtained from the blowdown analysis.

c. Reactor Building Pressure Model

The reactor building pressure-temperature analysis is performed using the digital computer code "CONTEMPT" developed by Phillips Petroleum Company in conjunction with the LOFT project. This program and its capabilities are described in Reference 16.

In this model, the reactor building is divided into two regions: the atmosphere (water vapor and air mixture) and the sump region (liquid water). Each region is considered to be well mixed and in thermal equilibrium, but the temperature of each region may be different. The reactor building and its internal structures are subdivided into five segments, as shown in Table 14-10, and treated as slabs with 1-dimensional heat transfer. Each segment, divided into several heat conducting subregions, may act as a heat source or sink. The program includes the capability of cooling the reactor building atmosphere by air coolers (reactor building emergency cooling units) and spray coolers (reactor building spray system), and cooling the liquid region by sump coolers (decay heat removal coolers).

During blowdown, mass and energy are added directly to the atmosphere where the liquid water present is assumed to fall to the liquid region; these input quantities to CONTEMPT are obtained from FLASH. After blowdown is over and emergency injection has been initiated, mass and energy are also added directly to the vapor region as steam. When the water level in the reactor vessel reaches the nozzle height, all mass and energy are added directly to the liquid region since no boiling of the injection water occurs after the core has been covered. When the supply of injection water is depleted, recirculation and cooling of sump water is simulated; all mass and energy input to CONTEMPT after blowdown is obtained from PRIT, a B&W company code. The PRIT program has the same core thermal model as described above for QUENCH, and, in addition to determining fuel and cladding temperatures, it maintains an inventory of mass and energy in the reactor vessel and of mass and energy released to the reactor building.

The reactor building calculations are begun by computing steady-state results using initial atmospheric conditions as the input. Following the rupture, the mass and energy addition is determined from the energy input rates for each time step. Heat losses or gains due to the heat-conducting slabs are calculated. Then the pressure and temperature of the liquid and vapor regions are calculated from the mass, volume, and energy balance equations.

Table 14-10
Reactor Building Structural Heat Capacitance Segments

Segment	Description
1	Reactor Building Walls and Dome
2	Refueling Cavity (Type 304 SS Liner - One Side)
3	Reactor Building Floor
4	Internal Concrete
5	Internal Steel

The model has been developed so that the effectiveness of the natural heat sinks and the engineered safety features can be clearly demonstrated.

14.2.2.3.4 Accident Analysis

a. Emergency Core Cooling System Design Base Accident

The accident selected for emergency core cooling equipment is based on reactor conditions at the rated power level of 2,568 MWt. As noted in the discussion of Accident Bases, the 36-in. ID double-ended pipe rupture produces the fastest blowdown and lowest heat removal from the fuel. This rupture, therefore, presents the most stringent requirements on emergency core cooling equipment and therefore dictates the design of the ECCS.

The QUENCH digital computer program, as described in 14.2.2.3.3 b above, is used to evaluate the core flooding tank performance in terms of core cooling capability. In this analysis, the hottest spot of the core is simulated. This spot is assumed to have a linear heat rate of 17.7 kW/ft.

The transient core flow from the FLASH analysis of the 36-in. ID double-ended pipe rupture was used to determine the heat transfer coefficients used in QUENCH. The very high flow rate during the initial blowdown period provides nucleate boiling conditions. Based upon the average channel flow predicted by FLASH, the core pressure drop was used to establish the hot channel flow conditions, and the W-3 DNB correlation was applied to the hot channel to determine the length of time nucleate boiling would exist. This analysis was made for only the first four seconds following the break because the correlation does not apply at the low pressures and heat fluxes that exist beyond 4 seconds. The minimum DNB ratios predicted by W-3 did not reach a value of 1.0 or less during this 4 second interval. However, a conservative design approach has been taken, and a DNB has been assumed to occur at 0.25 second followed by dispersed-flow film boiling. The degree of conservatism in assuming this early DNB time is shown in the sensitivity analysis of the design base accident included in the latter part of this section.

Prior to reaching DNB conditions, nucleate boiling surface coefficients at high flow rates may exceed 50,000 Btu/h-ft^2-F. A nucleate boiling surface coefficient of 20,000 Btu/h-ft^2-F was used in the analysis. However, the overall heat transfer coefficient from the clad node to the fluid sink is limited to 6,000 Btu/h-ft^2-F by the relatively low conductance of the clad.

From 0.25 until 9.5 seconds, when the flow rate predicted by FLASH drops to a negligible value for the 14.1 ft^2 rupture, Quinn's correlation[18] (given below) for dispersed flow film boiling is used to calculate the heat transfer from the clad. Available data indicate that if there had been a DNB, the heat transfer mechanism for a significant period of time would be transition boiling rather than fully developed film boiling. Tong's correlation for transition boiling predicts a heat transfer coefficient 2 to 4 times larger than Quinn's correlation of dispersed flow film boiling for the same flow conditions. Therefore, the procedure used in this analysis of core cooling is conservative and yields predicted maximum clad temperatures which are in excess of those that would be expected if the piping ruptured.

Quinn's correlation[18] for dispersed flow film boiling is as follows:

$$\frac{1}{h} = \frac{1}{H} + \frac{1}{h_{TP}}$$

where

h = overall heat transfer coefficient from wall to coolant, Btu/h-ft^2-F

H = heat transfer coefficient from vapor phase to the liquid drops. The value ranges from 3,000 to 18,000 Btu/h-ft^2-F. This analysis uses the lower value of 3,000.

h_{TP} = heat transfer coefficient from wall to vapor phase

$$h_{TP} = 0.023 \frac{k_B}{D_e} \left(\frac{G_T D_e X}{\mu_B}\right)^{0.8} \left(\frac{C_p \mu_B}{k}\right)_B^{1/3} \left(\frac{\mu_B}{\mu_W}\right)^{0.14} \left[1 + \frac{1 - X}{X} \left(\frac{\rho_g}{\rho_f}\right)^{2/3}\right]$$

where

k = steam conductivity, Btu/h-ft-F

D_e = hydraulic diameter, ft

G_T = mass velocity, lb/h-ft^2

X = steam quality of fluid

ρ = density

μ = viscosity

C_p = specific heat

Subscript B - "bulk steam phase"

Subscript g - "saturated steam"

Subscript f - "saturated liquid"

Subscript W - "wall"

Figure 14-33 shows the core flow versus time following the rupture of the 36-in. pipe. The clad surface heat transfer coefficient based on the flow of Figure 14-33 and Quinn's correlation is shown on Figure 14-34.

After blowdown, no core cooling is assumed until after core recovering starts. When the water level reaches the core bottom and starts to rise in the core, the submerged portion will be cooled in a pool boiling regime, and any steam thus produced will provide some cooling for that portion of the core above the waterline. However, in determining peak clad temperatures no cooling is assumed for that portion of the core which is above the waterline.

When the quiet water level reaches the hot spot, located 3 feet from the bottom of the core, a pool boiling heat transfer coefficient of 20 Btu/h-ft^2-F is applied even though the equation developed for pool film boiling from vertical plane surfaces[19] predicts a heat transfer coefficient of 38 Btu/h-ft^2-F.

Each of the core flooding tanks has a volume of 1,410 ft^3, of which 940 ft^3 is filled with borated water. The remaining volume is filled with nitrogen gas pressurized to 600 psig. Figure 14-35 shows the reactor vessel water level versus time for the 36-in. ID pipe break. Using a "quiet" water level as the basis for calculation, coolant from the core flooding system will cover the core hot spot at 16.5 seconds, and will cover approximately 80 per cent of the core at 28 seconds after the double-ended rupture of the 36-in pipe. Beyond this time, high-pressure and low-pressure injection will provide a continuous increase in the water level.

A detailed analysis of the void shutdown and core response was made with the digital computer program CHIC-KIN. This program accounts for changes in flow, pressure, enthalpy, and void fraction. It also computes axially weighted Doppler and moderator coefficients of reactivity for the kinetics calculation. The Doppler coefficient is input as a nonlinerar function of fuel temperature, and the moderator void coefficient is input as a function of void fraction. The parameters describing the coolant were obtained from the digital computer program FLASH, which in turn used the neutron power output from CHIC-KIN. The core is assumed to be initially at the rated power level of 2,568 MWt.

The analysis of the core kinetics during the LOCA has been based on a detailed breakdown of the average channel using coefficients generated by assuming uniform void distribution. The assumption of uniform reactivity throughout the core is good if the coolant density has some reasonably uniform distribution. The maximum density variation at the core exit at rated power is only ± 5 per cent; therefore, the distribution is relatively even. The distribution at mid-core is even closer.

There is no analytical model available for calculating the void fraction as a function of radius and azimuth during a LOCA, nor is there any way of accounting for such a distribution in a kinetics analysis. Since there is such an even distribution of the voids, however, the reactivity feedback is also expected to be uniform. Therefore, the reactivity held in any one fuel assembly (or group) is very low and nearly proportional to its volume fraction in the core. It is
4. concluded, then, that the average channel simulation used by the CHIC-KIN computer code is an accurate one. The program considers six axial segments in the fuel, clad, and water; and six segments radially in the fuel. The code also weights the reactivity feedback according to the importance of each region. Since the principal voiding effects are occurring in the axial dimension, the code does a proper calculation of the feedback.

Channels which have a slightly smaller-than-average density will shut the reactor down sooner than assumed because they are normally operating very near peak reactivity. Channels with higher density will not reach the peak reactivity-addition point until all other channels are shutting the reactor down due to undermoderation. Therefore, the cooler channels would actually have a smaller effect than the one assumed. In conclusion, the assumption of a radially-invariant moderator coefficient is a very conservative one.

Figure 14-36 shows the neutron power as a function of time following a 36-in. ID double-ended pipe rupture. The components of reactivity feedback for this transient, which assumes no trip action, are shown on Figure 14-37. The
4. curves shown on these figures are based on a design reference moderator coefficient of +0.5 x 10^{-4} $\Delta k/k/f$.

The clad temperatures experienced during a design basis accident are dependent upon the initial reactor conditions. Before a sensitivity analysis could be made for this accident, a scoping study was made to find the time in core life that resulted in the highest clad temperature during the accident. This study was necessary because the moderator coefficient and fuel temperature vary with core life.

The moderator coefficient decreases from a positive to a negative value during core life. The beginning of core life (BOL) positive coefficient causes an increase in core power during the early stages of the loss-of-coolant accident. This adds more heat to the fuel pins than would be added if the coefficient were negative. The moderator coefficient was calculated as a function of core burnup. As stated in Section 3.2.2.1.4, the maximum expected positive moderator coefficient is +0.5 x 10^{-4} $\Delta k/k/F$. This value occurs at BOL. A conservatively
4. high value of 0.5 x 10^{-4} $(\Delta k/k)/^{o}F$ has been used as a design reference. Figure 14-38 shows the value of the moderator coefficient as a function of fuel burnup. The decrease in the moderator coefficient is primarily due to the change in the coolant boron concentration during the life of the core. The moderator coefficient shown in Figure 14-38 was calculated by taking no credit for the buildup of xenon.

The average fuel temperature was calculated at 17.7 KW/ft as a function of core burnup. Figure 14-39 shows two curves for fuel temperature. The lower curve was calculated on the basis of nominal dimensions of the fuel and clad. This results in a cold diametrical clearance between the fuel and clad of 7 mils. The upper curve was calculated by using the design clearance between the fuel and the clad of 8.5 mils (see 3.2.3.2.3 g).

The maximum clad temperature as a function of core life was evaluated based on the following assumptions:

(1) DNB occurs at 0.25 seconds.

(2) Heat transfer beyond DNB as shown in Figure 14-34.

(3) No credit was taken for steam cooling during core recovering.

(4) No credit was taken for control rod insertion even though the core pressure drop is not large enough to prevent their insertion.

(5) The moderator coefficients used were those as shown on Figure 14-38 which take no credit for the buildup of xenon.

(6) The average fuel temperature were those as shown on Figure 14-39 calculated on the basis of the design clearance.

Using these assumptions, the curves shown on Figure 14-40 of maximum clad temperature as a function of core life were generated. It can be seen that the maximum temperature occurs at the beginning of core life due to the larger power generation resulting from the positive moderator coefficient that exists at this time in life. Using initial fuel temperatures based on nominal tolerances, the maximum clad temperature is 1,700 F. Using the design fuel-to-clad clearance, the maximum clad temperature is 1,850 F.

To demonstrate that the ECCS has more than adequate capability, the clad temperature was calculated using a moderator coefficient of 0.9×10^{-4} $\Delta k/k/F$. This more positive coefficient (0.9×10^{-4} rather than 0.5×10^{-4}) causes a greater amount of power to be generated during the early stages of the accident. Using the design clearance and this more positive moderator coefficient, the resulting maximum clad temperature is 1,994 F.

It will be shown later in this section that the 14.1 ft^2 hot leg rupture is indeed the rupture that causes the maximum clad temperature during a loss-of-coolant accident. Therefore, the sensitivity analysis and the spectrum of break size analysis which follow utilize the maximum expected moderator coefficient of 0.5×10^{-4} $\Delta k/k/F$ that occurs at BOL in conjunction with the fuel temperature that exists at this time. The fuel temperatures are those calculated by using the design clearance between the fuel and the clad.

Figure 14-41 shows hot spot clad temperatures for a range of heat transfer coefficients during reflooding of the core. Without any cooling following blowdown, the clad temperature reaches the melting point in approximately 48 s. An h value of 20 Btu/h-ft^2-F provides immediate quenching action and a slow cooling rate thereafter. An h value of 38 provides fast cooling. Even though the value of h = 38 is realistic for film boiling in a pool,[18] the probable mode for the submerged portion of the core, a more conservative value of h = 20 has been used as the reference for evaluating performance of the core flooding tank. The clad hot spot temperature excursion is terminated at 1,850 F, 1,500 F below the clad melting point.

Two additional analyses were performed to evaluate the sensitivity of the maximum clad temperature to time to DNB and to the heat transfer coefficient. The assumption that DNB occurs at 0.25 s is quite conservative. The duration of the nucleate boiling period has been evaluated to show the sensitivity of the maximum fuel temperature to this parameter. Figure 14-42 shows the effect of variation of time to reach a DNB. It should be noted that if DNB occurred at the time of rupture, then the peak temperature would only increase about 8 F above 1,849 F, while if DNB has not occurred for a more realistic value of 4 s, then the temperature would be reduced by 210 F to a peak value of 1,639 F.

After blowdown, zero cooling is assumed until quenching occurs with a clad surface heat transfer coefficient of 20 Btu/h-ft^2-F. Figure 14-34 is a plot of the nominal clad surface heat transfer coefficient versus time based on the flow of Figure 14-33 and Quinn's correlation. Figure 14-43 shows that this nominal heat transfer is not on the most sensitive part of the curve, and a 20 per cent decrease in the value of h would only result in increasing the peak clad temperature by 117 F to a value of 1,967 F.

The results of the design basis accident (36-in., double-ended rupture) are as follows:

(1) The clad hot spot temperature excursion is terminated at 1,850 F. A negligible amount of zirconium-water reaction occurs, and the maximum clad temperature is 1,500 F below the clad melting point (Figure 14-41). This maximum temperature is based on conservative heat transfer assumptions and is 450 F below 2,300 F. No fuel melting occurs to release large quantities of fission products.

(2) If DNB occurs at time 0, then the maximum hot spot clad temperature increases only 8 F, whereas if DNB occurs at 4 seconds, then the maximum hot spot clad temperature decreases by 210 F (Figure 14-42).

(3) A clad surface heat transfer coefficient of 20 Btu/h-ft^2-F is sufficient to quench the clad temperature rise when the water reaches the 1/4 point of the core, even though a coefficient of 38 Btu/h-ft^2-F has been predicted.

b. Examination of Core Component Structural Integrity

Many of the fuel rods may be expected to experience cladding perforation during the heatup transient in the loss-of-coolant accident, as a result of fission gas internal pressure and weakening of the clad as its temperature increases. The mechanical strength of the Zircaloy cladding is reduced as the temperature exceeds 1,000 F, so that the fuel rods with appreciable fission gas internal pressure will begin to fail locally and relieve the gas pressure when the temperature exceeds 1,100 F. Some local deformation of the rods will occur before perforation. However, cooling would still be effective since the fuel rods are submerged, and cross-channel flow around the deformed area will cool the rod. At worst, a local hot spot may occur.

To verify that the perforation/deformation failure will not inhibit the emergency core cooling system from preventing core cooling, B&W has undertaken a program to evaluate the effects of perforation and deformation of fuel rods during the temperature transient following the loss-of-coolant accident. Preliminary tests have been run on nine samples of unirradiated Zircaloy-4 cladding filled with ceramic pellets, and additional experiments are planned to gain a clearer understanding of the effects of temperature excursions on Zircaloy-clad fuel elements. Current plans include performance of a three-phase program. In the first phase, single-rod excursions are being performed to better establish temperature-pressure relationships at the time of clad perforation. The single-rod tests of the first phase are also investigating the extent of deformation to be expected under the varying conditions associated with simulated in-reactor temperature excursions. These include the effects of hydrogen concentration and oxide films. The second phase of the program consists principally of multirod tests to explore the effect of the restraining action of spacer grids and adjacent fuel rods and to determine the randomization of the localized deformation in an assembly of fuel rods. In the third phase of the program the data obtained from the two experimental phases will be applied to the analysis of the effects in a loss-of-coolant accident.

It is calculated that a small number of fuel rods operating at peak power will experience a cladding temperature transient to 1,850 F in about 17 sec. The major portion of the core will not experience as severe a transient. Heating of the fuel rod spacer grid requires heat flow from the clad to the structure by conduction and radiation; therefore, the structure temperatures will lag the cladding temperature transient. As the fuel rod temperature rises, the fuel rods are expected to experience some bowing due to axial growth between supports. The spacer grids have substantial mechanical strength, even at the maximum expected temperatures, and will therefore retain sufficient strength to insure spacing between fuel rods to allow emergency coolant to reach them. This will suffice to keep the fuel rods in their respective positions in the core to prevent gross change in core geometry.

The ability of the clad to maintain its strength and structural integrity during core reflooding has been confirmed by experimental work at B&W involving the rapid quenching of unirradiated Zircaloy tubing specimens from temperatures as high as 2,300 F. Test results show that, for temperatures as high as 2,300 F, the cladding material will retain its strength and will not experience brittle fracture upon quenching. Irradiation will increase the tensile strength of the material even though it will also promote the tendency towards a brittle fracture mode of deformation at higher pressures and temperatures towards the end of core life.

The temperature transient in the core can produce significantly higher than normal temperatures in components other than fuel rods. Therefore a possibility of eutectic formation between dissimilar core materials exists. Considering the general area of eutectic formation in the entire core and reactor vessel internals, the following dissimilar metals are present, with major elements being in the approximate proportions shown:

Type-304 Stainless Steel

19 per cent chromium
10 per cent nickel
remainder iron

Control Rod Poison Material

80 per cent silver
15 per cent indium
5 per cent cadmium

Zircaloy-4

98 per cent zirconium
1-3/4 per cent tin

Inconel

53 per cent nickel
19 per cent chromium
3 per cent molybdenum
5 per cent Nb-Ta
1 per cent titanium
0.5 per cent aluminum
remainder iron

All these alloys have relatively high melting points (greater than 2,700 F) except those for silver, cadmium, and indium. The melting point of the silver-indium-cadmium alloy is about 1,470 F.

The binary phase diagram indicates that zirconium in the proportion 75 to 80 per cent has a eutectic point with either iron, nickel, or chromium at temperatures of approximately 1,710, 1,760, and 2,370 F, respectively. If these dissimilar metals are in contact and if those eutectic points are reached, then the materials could theoretically melt even though the temperature is below the melting point of either material taken singly.

One point of such dissimilar metal contact is between Zircaloy-clad fuel rods and Inconel 718 spacer grids. The analysis of the performance of the core flooding tanks during a loss-of-coolant accident indicated that some of the cladding will exceed the zirconium-iron and the zirconium-nickel eutectic points. Since the spacers are located at 21-in. intervals along the assembly and each grid has a very small contact area, only a fraction of the hottest fuel rods would be in contact with Inconel 718 spacer grids.

During the experimental test program referred to above, B&W conducted experimental tests in which specimens of Zircaloy-4 tubing in contact with sections of spacer grid material were subjected to a thermal transient closely approximating that of the clad hot spot following a LOCA. These tests verified that the eutectic reaction is limited to the small region of contact between the clad and the spacer grid tips (dimples), and that it terminates as these materials melt at the point of contact. Both the clad and the grid material maintained their structural integrity because the amount of material involved was small and melting was localized.

Another area of dissimilar metal contact is that of a zirconium guide tube with the stainless steel cladding of the control rod. To determine whether the temperatures in the control rod following a LOCA could become high enough to approach either the temperature required for possible eutectic formation between the clad and the guide tube or the melting temperature of the Ag-In-Cd alloy, the thermal performance of a control rod assembly following a LOCA was examined analytically. A conservative approach was taken in this analysis. In spite of the fact that the core flow is as high or higher than normal core flow during the first two seconds following the rupture, normal steady-state cooling of a control rod is assumed. In two seconds, the core power is essentially down to decay heat levels. However, the following assumptions were made.

(1) The average core power after two seconds is 8 per cent of ultimate power and remains at this level.

(2) All decay heat is absorbed in the core and 50 per cent of the decay heat is in the form of gamma rays available for absorption in the control rod. By ratioing the control rod density to the average core density, an average energy deposition rate of 8.50 w/cc in the control assemblies was obtained.

(3) The maximum activation product energy in the control rod itself was estimated to be 2.99 w/cc.

(4) The highest energy deposition rate at the decay heat level was assumed to be the average times the ratio of peak-to-average power, or 36.18 w/cc.

(5) An adiabatic heatup of the control rod with a heat rate of 36.18 w/cc was assumed until the water level reached the point in the core at which the highest peak-to-average power occurs.

(6) The temperature of the control alloy is approximately 650 F at the time the rod is assumed to be insulated (2 s).

Using the assumptions above, the average temperature of the Ag-In-Cd goes up to 1,035 F at about 17 seconds, at which time the water level in the core reaches the elevation of the hottest spot on the control rod. The temperature of the rod then rapidly decreases.

The lowest temperature for a eutectic formation is that for Zr-Fe, which occurs at 1,710 F. Therefore, the integrity of the control rod assemblies is maintained during and following a loss-of-coolant accident.

c. Core Cooling Analysis for Spectrum of Leak Sizes

The loss-of-coolant accident has been analyzed for a spectrum of leak sizes and locations. Power generation is based on the moderator coefficient of $+0.5 \times 10^{-4}$ $\Delta k/k/F$ with BOL fuel temperatures. This information is reported according to the following grouping: (1) hot leg ruptures, (2) cold leg ruptures, (3) small leaks, and (4) injection line failures.

(1) Hot Leg Ruptures

In 14.2.2.3.4 a an analysis of the 36-in. ID, double-ended pipe rupture was presented. This accident produced the fastest blowdown and the lowest heat removal from the fuel, therefore producing the highest cladding temperatures of any loss-of-coolant accident. This was, therefore, the basis for design of the core flooding equipment. A decrease in the rupture size results in decreased maximum clad temperature during a loss-of-coolant accident.

Core cooling evaluations have been performed for a spectrum of five additional rupture sizes using the same basic calculational technique and assumptions as used for the large rupture case. These rupture sizes are 8.5, 5.0, 3.0, 1.0, and 0.4 ft^2. The reactor coolant system average pressures, as a function of time after the rupture, are shown in Figure 14-44 for these rupture sizes.

The reactor vessel water heights as a function of time after the rupture for the various rupture sizes are shown on Figure 14-45. These water height curves were generated utilizing the flow available from the core flooding tanks, one high-pressure injection pump, and one low-pressure injection pump. The two independent pumping strings have a combined capacity of at least 7,000 gpm. However, the analysis is based on only one string in operation with a total flow of 3,500 gpm.

Figure 14-46 shows the hot spot clad temperatures as a function of time for the various rupture sizes. As can be seen from this figure, the smaller size ruptures yield maximum clad temperatures which are considerably lower than those resulting from the larger sizes. The results of this study are shown in Table 14-11.

Table 14-11
Tabulation of Loss-of-Coolant Accident Characteristics
for Spectrum of Hot Leg Rupture Sizes

Rupture Size, ft^2	Full-Power Seconds[b]	Minimum Water Level Below Bottom of Core, ft	Hot Spot Maximum Temperature, F
14.1	1.55	-6.8	1,850
8.5	2.65	-5.2	1,717
5.0	4.12	-4.0	1,529
3.0	3.35[a]	-2.2	1,336
1.0	6.25[a]	+4.7	1,095
0.4	6.50[a]	+12.0	1,100

[a]Blowdown forces on control rods are equal to or less than normal pressure drop, and control rods will insert with normal velocities. These values are for trip shutdown rather than for a void shutdown, but include void reactivity effects.

[b]Includes power generation until power has decayed to 10 per cent of rated power.

(2) Cold Leg Ruptures

A similar analysis for a spectrum of rupture sizes in the cold leg piping has been made. The rupture sizes tabulated are the double-ended, 28-in. ID inlet pipe, which yields 8.5 ft^2 of rupture area, and the 5.0, 3.0, 1.0, and 0.4 ft^2 sizes. The reactor coolant system average pressures, as a function of time after the rupture, are shown in Figure 14-47. Figure 14-47a shows the core flow versus time following the rupture of one of
5. the 28-in. reactor vessel inlet pipes. The clad surface heat transfer coefficient based on the flow of Figure 14-47a is shown in Figure 14-47b.

The core cooling analysis for cold leg ruptures assumes that sufficient water remains in the steam generators to prevent free venting of the steam generated during the reflooding period. The internals vent valves incorporated in the design of the reactor internals provide for pressure equalization and steam venting; however, the flooding rate is somewhat delayed. This cold leg rupture analysis is, therefore, different from the hot leg rupture analysis where the reactor vessel would be refilled at the rate that emergency injection coolant was supplied, and no flooding delays existed.

The hot leg rupture analysis assumed no steam cooling from the end of blowdown until the water level in the core reached the core hot spot. Because the delay in flooding is due to the high steam generation rates that occur after the water enters the core, the core cooling analysis for cold leg ruptures utilizes steam cooling during the reflooding stage.

The potential steam generation rate from all available energy sources was considered in our assessment of the internals vent valves. Figure 14-48 shows the recovery period following the most severe cold leg break as a function of the steam vent area. The steam flow rates through the vent valves are included here as Figure 14-48a. It is conservatively assumed that any steam generated and available as vent valve flow reaches a superheated temperature that is dependent on the core temperature and an energy balance in the upper reactor vessel plenum. This steam flow rate was established in a transient analog computer analysis of the primary loop following blowdown. Consideration has been given to the steam generation rate on the primary side of the steam generators including the cores and reactor coolant system metal. Figure 14-48a presents the total steam flow rate from all sources.

The vent valves open by a very small pressure drop of less than 0.5 psi. This is the value used in the safety analysis. However, as
21. shown in Section 3.2.3.2.4, the valve is full open at 0.26.
This provides a steam flow path that allows steam to
vent directly to the reactor inlet piping where the rupture has
5. occurred. The inclusion of these valves permits a pressure equalization between the point of water injection and the core region to enable cooling water to reflood the core.

The core quiet water level predicted by this analog simulation is presented as Figure 14-48. This level is a representation of the equivalent liquid that would exist in the core following the accident. In actuality, the coolant level would never drop below the core midplane once injection water enters the core and the core would be covered by a two phase mixture at all times during this transient. This so-called "Froth height" can be calculated by considering the specific volumes of water and steam existing in the core, the steam generation rate and the steam rise velocity at any point in the following blowdown.

Following a large cold leg rupture with eight vent valves opened to their maximum position, and assuming that the steam generated is superheated to 600 F at a reactor building pressure of 65 psia, 274 lb/s flowing through the vent valves is necessary to balance a quiet liquid level at the core midplane.

Figure 14-48 shows the water height as a function of time after a double-ended inlet pipe rupture (8.5 ft^2) for the case where sufficient water remains in the steam generator to create a water seal. The steam flow through the eight vent valves provides for pressure equalization but the core flooding rate is restricted by the pressure buildup in the reactor vessel upper plenum. This curve is based on the quiet water level, and no credit is taken for increased elevation due to the existence of a steam-water mixture. This figure also shows the flooding rate if a water seal does not exist. The time to reach the hot spot has only a 3 s difference between the vent valve case and the no-water-seal case.

A curve of fill rate for operation of only seven of the eight vent valves has been included to show the effect of core cooling on the number of vent valves available. This reduced venting area results in a 1 s delay in flooding the hot spot. The resultant peak hot spot temperature is 1,645 F, while that for the double-ended pipe rupture with eight vent valves is 1,622 F.

As discussed above, vent valves have been installed in the reactor vessel internals to provide for pressure equalization between the reactor vessel outlet plenum and the downcomer annulus. These valves are necessary only if a sufficient amount of water remains in the steam generators to form a water seal which prevents steam from being released to the reactor building through the steam generators.

It is reasonable to assume that if enough water remains in the steam generators and associated piping to form a water seal, then water would also be left in the reactor vessel. The reactor vessel fill rates as shown on Figure 14-48 were obtained by assuming that the reactor vessel had no water remaining. The conservatism of this assumption was borne out by one of the LOFT semiscale blowdown tests. This test was made with an internals configuration which roughly simulated the internals configuration of a large PWR with vent valves in the core support shield. Preliminary results from this test indicate that as much as 20% of the water could be left in the reactor vessel if vents, such as the ones incorporated in the reactor vessel internals, are available for the steam.

It has been postulated that the injection water could be diverted out the leak due to carryover by the steam. In order for carry-over to take place, the water will have to intimately mix with steam. The annulus geometry is such that intimate mixing will be difficult to obtain and the water injected should go directly to the bottom of the vessel. Water from the core flooding tanks enters the reactor vessel through two nozzles 180° apart. The center lines of the flooding nozzles and the main reactor coolant pipe nozzles are 30 degrees, 3.6 ft, apart at the closest point. The injection nozzles enter the vessel at an elevation just above the main coolant nozzles and just below the vent valves. The flow deflectors on the core support shield and the vent valve steam flow forces will tend to direct the injection flow toward the bottom of the vessel.

While the geometry makes it very unlikely that injection coolant would be lost by diversion to the leak, it cannot be proved conclusively. Accordingly, an analysis has been made assuming that mixing of the leak flow mixture and the injection water does occur. For the first two seconds of injection, the period from 7.5 to 9.5 seconds after the rupture, the energy content of the steam flowing out the leak is greater than the heat absorption capability of the injected coolant. After the first 2 seconds of injection, all of the leakage steam flow can be condensed by the injection water and cooled down to the saturated condition at 65 psia.

Relating this to volume and assuming that diversion will occur if all of the steam cannot be condensed, 150 ft^3 of injection water could be lost during the first 2 seconds of injection. However, condensation and cooling that occurs during the remainder of the blowdown result in a gain of slightly more than 150 ft^3.

To show that the possible loss of injection water by diversion out of the break has a small effect on the peak clad temperature, an analysis was made which assumed that all the water injected during the first 4 seconds was lost. No credit was taken for any condensation or water remaining and this results in a loss of 380 ft^3 of injection water. The net effect of this is to shift the water level versus time curve approximately 4 seconds and delay core cooling. This increases the peak hot spot temperature to 1,684 F as shown on Figure 14-49.

This analysis shows that the peak clad temperature is not extremely sensitive to the effects of diversion of injection coolant. Even taking the extreme case of 4 seconds of diversion, no water remaining at the end of blowdown, and no condensation only increased the peak hot spot temperature from 1,622 F to 1,684 F which is still significantly below the design limit value of 2,300 F.

All cold leg ruptures have been analyzed by assuming that a water seal existed in the cold leg piping. The water heights as a function of time for the complete spectrum of rupture sizes are shown on Figure 14-50. The water height curves were generated using the flow available from the core flooding tanks, one high-pressure injection pump, and one low-pressure injection pump.

The hot spot temperature as a function of time for the spectrum of cold leg leak sizes is shown in Figure 14-51. The results of this analysis are shown in Table 14-12.

Table 14-12
Tabulation of Loss-of-Coolant Accident Characteristics for Spectrum of Cold Leg Rupture Sizes

Rupture Size, ft^2	Full-Power Seconds[a]	Minimum Water Level Below Bottom of Core, ft	Hot Spot Maximum Temperature, F
8.5	0.60	-6.7	1,622
5.0	0.40	-6.0	1,652
3.0	1.00	-4.8	1,510
1.0	5.38	+3.6	1,284
0.4	5.50	+7.0	1,107

[a] Blowdown forces on control rods are equal to or less than normal pressure drop, and control rods will insert with normal velocity. These values are for trip shutdown rather than void shutdown, but include the void reactivity effects. Values also include power generation until power decays to 10 per cent of ultimate power.

(3) Small Leak Analysis

The spectrum of leak sizes presented above indicates that larger leak sizes impose the most stringent requirements on the emergency core cooling system. The analysis also shows that the core hot spot is never uncovered for leak sizes less than 1 ft^2. For all leak sizes examined, down to 0.4 ft^2 the system pressure decreased below 600 psig so that the core flooding tanks discharged borated water into the reactor vessel. For smaller leak sizes, system depressurization is slower. This results in a longer period of core cooling by the reactor coolant and the high-pressure injection coolant prior to the time that the core flooding tanks start discharging into the reactor vessel.

A leak located in the piping below a steam generator results in the greatest mass loss from the reactor coolant system, since the leak will be discharging subcooled and/or saturated water until the water level in the reactor coolant system is below the reactor vessel nozzles elevation and the steam generator located in the ruptured loop empties. When the water level reaches the reactor vessel nozzles, coolant is lost from the reactor vessel by flashing (because the system pressure is dropping) and by boiloff due to the core decay heat. The high pressure injection system supplies coolant to make up for this loss and to cool the core.

When the pressurizer empties, the mode of heat transfer for core cooling is assumed to be natural convection film boiling. Prior to this time the mode of heat transfer is either nucleate boiling or forced convection.

The natural convection film boiling coefficients are based on the following relationships:(19)

$$Nu = 0.0216\ Pr^{1/3} \left[\frac{Gr^2\ Pr^{-2/3}(0.1212 + 0.1756\theta)}{(0.1212 - 0.1756\theta - 0.0889\ Pr^{-2/3})^2} \right]^{1/5}$$

$$Gr = \frac{g(\rho_f - \rho_v)}{\rho_v} \frac{L^3 \rho_v^2}{\mu_v^2}$$

$$\theta = \frac{h_{fg}}{C_p(T_{wall} - T_{sat.})}$$

Pr, ρ_v, μ_v, and C_p are steam properties evaluated at the film temperature. Using these relationships, values of h are found to range from 95 to 160 Btu/h-ft^2-F.

These values of h provide for heat transfer that is more than sufficient to keep the core cool.

For rupture sizes approximately 4 in. in diameter or larger, the system pressure decreased below 600 psig. Below this pressure, water from the core flooding tanks supplements the flow from the high pressure injection system. To demonstrate the capability of the high pressure injection system, a 4-in.-diameter break was analyzed, taking no credit for the core flooding tanks and making use of only one of the three high pressure injection pumps. Figure 14-52 shows the pressure versus time for this break. It can be seen that the pressure reaches 600 psig in about 1,700 seconds, at which time the core flood tanks would start injecting. However, from Figure 14-53, it can be seen that the water level in the reactor vessel starts to increase at 700 seconds. Therefore, at times greater than 700 seconds, the makeup from one high pressure injection pump is greater than the loss due to boiling even though the pressure is above 600 psig.

Figure 14-54 shows the hot spot clad temperature. Heat transfer by nucleate boiling occurred for 50 seconds, the time required to empty the pressurizer. Thereafter, even though calculations show that nucleate boiling continues, a pool film boiling coefficient of 100 Btu/h-ft^2-F was used, since the core remains covered. The dotted line shown on Figure 14-54 is the temperature that would result if DNB were assumed to occur when the reactor trips (4 s). The reactor coolant pumps were also assumed to trip when the reactor tripped. At this time, dispersed flow film boiling was assumed to begin. At 50 seconds, a heat transfer coefficient of 50 Btu/h-ft^2-F was applied. This conservative approach results in a maximum clad temperature of 1,020 F as opposed to the more realistic but still conservative value of 770 F shown by the solid curve.

For smaller breaks, the rate of mass and energy lost from the system is lower and system depressurization is slower. Therefore, the core is cooled for a longer period of time by the reactor coolant. As stated above, at times greater than 700 seconds, the injection flow rate from one high pressure injection pump is greater than the boiloff rate due to decay heat. This is true even if the pressure does not fall below 1,050 psig (the steam generator's safety valve set point). However, if such an accident were to occur, the operator would have sufficient time to dump steam to the condensers so as to lower the secondary system pressure and temperature and facilitate energy removal from the reactor coolant system. This in turn would cause a faster decrease in the reactor coolant system pressure.

Therefore, this analysis shows that for leaks 4 in. in diameter or smaller, the high pressure injection system provides sufficient core cooling. For leaks larger than 4 in. in diameter, the pressure drops rapidly enough and low enough to cause the core flood tanks to discharge their contents into the reactor vessel.

(4) Evaluation of Injection Line Failure

A low-pressure injection line failure has been evaluated, and the results show that the reactor is protected. The rupture of a pipe that connects a core flooding tank and a low-pressure injection pump to the reactor vessel was assumed at a location adjacent to the reactor vessel and before the first check valve (see Figure 6-1). This pipe has an internal diameter of 11.5 in., and the resultant rupture area is 0.72 ft^2.

Interpolation of available blowdown calculations has been used to evaluate this rupture size, and the data show that a rupture of this size would result in the core being uncovered several feet below the top of the core. However, the hot spot will never be uncovered, and peak cladding temperatures will be slightly less than that shown in Figure 14-51 for the 1.0-ft^2 cold leg rupture.

Since this rupture size leaves a considerable water inventory in the reactor vessel, the remaining core flooding tank inventory is more than adequate to reflood the core completely.

The other low-pressure line can supply 3,000 gpm of water to the reactor vessel and provide coolant to keep the core cooled.
17. Furthermore, even without the low-pressure pump, the combined capacity of the two high-pressure pumps is 900 gpm, which is well in excess of the boiloff rate due to decay heat immediately after blowdown.

The high-pressure injection mode has two independent chains of flow to supply borated coolant to the system. If a rupture of high-pressure injection piping were to occur in one of the four lines between the attachment to the reactor coolant pipe and the check valve, then the other chain of this system would have adequate capacity to protect the core against this small leak. Even in the event of a component failure in the second high-pressure injection line, the flow to the reactor coolant system in the broken string would be adequate to protect the core. This is due to the orifices that are located in each of the two lines in both of the high-pressure injection strings. The orifice in the broken line prevents the full loss of the injection water to the reactor building.

(5) Evaluation of Emergency Core Cooling System Performance for Various Rupture Sizes

The loss-of-coolant analysis is based on the operation of one high-pressure injection pump (450 gpm), one low-pressure injec-
17. tion pump (3,000 gpm), and the operation of both core flooding tanks. It has been shown that this combination of engineered safeguards provides core protection for all leak sizes up to that corresponding to the double-ended rupture of the 36" ID pipe.

The capability of other combinations of engineered safeguards to provide core protection has been evaluated. This capability is shown in Figure 14-55. In this evaluation the core is considered protected if the combination of emergency cooling equipment considered will prevent core damage that would interfere with further core cooling.

The high-pressure injection system, with only one pump operating, can protect the core for leaks up to 4 in. in diameter. A combination of one high-pressure and one low-pressure injection pump will protect the core for leaks up to 10 in. in diameter, (0.5 ft^2) whereas, one high-pressure and two low-pressure pumps provide protection for leak areas up to 1 ft^2. For larger break areas, the operation of one high pressure injection pump, one low pressure injection pump, and the core flood tanks provides the coolant necessary to keep the core protected.

This evaluation of emergency core cooling capability demonstrates that the core is protected for the entire spectrum of leak sizes in both the hot and cold leg piping.

(6) Evaluation of Emergency Core Cooling System Performance with Building Pressure ECCS Actuation Signal

Although the reactor coolant system pressure instrumentation is highly reliable and redundant and furnishes the ECCS actuation signal, a separate and independent ECCS actuation signal is afforded on a high reactor building pressure. Both of these signals have been assessed as to the time to reach the actuation signal for a spectrum of breaks ranging from 0.087 ft^2 (4 in. diameter) to the double-ended rupture of the 36 inch ID reactor coolant system outlet piping (14.1 ft^2). The time to receipt of signal is shown in the table below.

Rupture Size, ft^2	Time to 1500 psig RCS Pressure, sec	Time to 4 psig R.B. Pressure, sec
14.1	< 1	< 1
8.5	< 1	< 1
5.0	1	< 1
3.0	3.5	1
1.0	24	2
0.4	50	7
0.087	30	90

All of the work done to date has assumed that the actuation signal for the ECCS comes from the RCS pressure signal. As can be seen from the table, this is conservative since for breaks larger than 0.4 ft^2, the reactor building pressure signal precedes the RCS pressure signal. The 0.087 ft^2 case (4 inch diameter), shown in the table shows that the 1500 psig RCS pressure signal was reached approximately 1 minute before the reactor building signal. During this minute, approximately 400 gallons of emergency coolant are piped by the ECCS. If this coolant were not available, the minimum level in the reactor vessel would be about 1/2 ft lower. However, this is still above the top of the core and the core would be adequately cooled. Therefore, even though the RCS pressure signal acts faster, the reactor building signal occurs rapidly enough to more than adequately protect the core.

14.2.2.3.5 Reactor Building Design Basis Accident

The reactor building pressure was evaluated for a range of leak sizes between 1.0 ft^2 and 14.1 ft^2. The 14.1-ft^2 leak is equivalent to a double-ended rupture of the 36-in. ID reactor outlet piping. For this analysis the reactor was operating at 2,568 MWt when the rupture occurred. Mass and energy released to the reactor building during blowdown was calculated by FLASH. A 20-region PRIT model was used to simulate the core thermal transient following blowdown. This simulation includes fuel heat generation, metal-water reaction, quenching of the core by injection water, and boiloff from the reactor vessel--all considered as functions of time.

The basis for this analysis is that two core flooding tanks and a single ECCS string are working to provide core cooling; i.e., the high pressure injection system has one of the three pumps available for operation and the low pressure injection system has one of the two pumps available. This ECCS string is assumed to operate on emergency power and begin injecting 3,500 gpm of borated water into the reactor coolant system at the time core flooding tanks
5. empty and emergency power is available.

When the injected coolant water covers approximately 25 per cent of the core, the surface heat transfer coefficient from all fuel cladding to the water is assumed to be 100 Btu/h-ft^2-F. Assuming a pool boiling coefficient of 100 Btu/h-ft^2-F for the whole core when only the lower quarter is covered by coolant is conservative for the reactor building pressure analysis because it compresses overall energy transport into a short period of time.

Heat is also released from the hot metal of the reactor coolant system and the reactor vessel internals. During the blowdown period, a surface heat transfer coefficient of 1,000 Btu/h-ft^2-F is used. After blowdown this coefficient is changed to 100 Btu/h-ft^2-F for the metal at elevations below that of the leak and 5 Btu/h-ft^2-F at elevations above that of the leak. The coolant sink temperature is provided by FLASH for the blowdown period and is assumed to be 280 F thereafter. A multilayer finite difference model is used to calculate the conduction heat transfer from the metal.

All heat transferred from the core and the reactor coolant system metal is assumed to generate steam without taking credit for the subcooled condition of the injection water until the reactor vessel is filled to the leak height. Thereafter, all energy is removed by low pressure injection flow of subcooled water, and the energy release to the reactor building atmosphere is terminated. No delay is assumed in transporting steam to the reactor building.

Both reactor inlet (cold) and reactor outlet (hot) line breaks were analyzed with FLASH. However, a complete reactor building analysis was made only for the hot line breaks because the hot line breaks had longer blowdown and better heat transfer during blowdown than did the cold line breaks. Thus, the hot leg rupture produced more rapid heat transport from the core to the building.

The results of calculations of fluid and heat transport to the reactor building as determined by FLASH, PRIT, and other analytical models were used as input to the reactor building pressure analysis program, CONTEMPT.

In calculating the reactor building pressure, it is assumed that the initial average temperature of the building atmosphere and structural materials is 110 F. The initial reactor building atmosphere is assumed to be at zero pressure and zero relative humidity. The steel and concrete are heat sinks that affect the reactor building pressure during the release of hot reactor coolant. The heat sinks considered in this analysis includes a 7 mil layer
5. of paint on all steel surfaces except stainless steel. For the structural
and thermal properties of the heat sinks, refer to Table 14-13. Heat transfer from the reactor building atmosphere to the steel liner is calculated using a condensing coefficient of 620 Btu/h-ft^2-F until a total heat input of 110 Btu/ft^2 has been achieved. Thereafter, a condensing coefficient of 40 Btu/h-ft^2 is used. Structural and miscellaneous steel heat sinks are evaluated using the same heat transfer coefficients as the liner. The geometry of this metal assumes one heat transfer section with an equivalent thickness of 0.93 inch.

For heat transfer from the reactor building atmosphere to the concrete, a condensing coefficient of 40 Btu/h-ft^2-F is used. For heat transfer from the sump water to the concrete floor, a coefficient of 20 Btu/h-ft^2-F is used. No credit is taken for heat transfer to reinforcing steel in the internal concrete structures.

Following a loss-of-coolant accident the reactor building is cooled by three reactor building air coolers and the spray system. Each cooling arrangement, 3 air coolers or 3,000 gpm sprays, has a capability of removing at least 240 x 10^6 Btu/h from the atmosphere at a vapor temperature of 286 F. The two systems combined offer added flexibility. Two air cooling units plus 1,500 gpm from one of the two sprays, or 3,000 gpm from sprays alone, provide cooling that is at least equivalent to three reactor building air coolers. The reactor
5. building emergency cooling unit characteristics are shown in Section 6.3.2.6.
The effects of varying the steam-air mixture temperature and flow rate are shown in Figures 6-11 and 6-11a, respectively. Each system is designed so that it alone can protect the reactor building against overpressure. Each system is assumed to operate on emergency power, and fans are delayed until 25 s and sprays for 37.5 s after the rupture occurs (6.2.3 and 6.3.3).

Table 14-13
Reactor Building Data for Reactor Building
Pressure Analysis

Structural Properties

	Segment	Material Type	Surface Area, ft^2	Thickness	Description
	1	Paint	79,000	0.007 in.	Reactor Building
		Steel	79,000	0.25 in.	Walls and Dome
		Concrete	79,000	3.25 ft.	
	2	S. Steel	6,770	0.25 ft.	Refueling Canal (stainless steel liner on inside)
		Concrete			
	3	Concrete	8,000	2.0 ft.	Reactor Building Floor
	4	Concrete	88,230	1.37 ft.	Miscellaneous Concrete
5.	5	Paint	22,200	0.007 in.	Miscellaneous Steel Exposed
		Steel	22,200	0.93 in.	

Thermal Properties

Material	Density (lb/ft^3)	Thermal Conductivity ($Btu/h\text{-}ft^2\text{-}F/ft$)	Heat Capacity (Btu/F-lb)
Paint	192	0.50	0.350
Steel	490	29.60	0.120
Concrete	143	0.79	0.156
S. Steel	500	26.00	0.120

Figure 14-56 shows the reactor building pressure for complete severance of a 36-in. ID reactor coolant system pip (14.1-ft^2 rupture area) with two core flooding tanks and 3,500 gpm of borated water injection into the reactor coolant system. Reactor building cooling is provided by three building air coolers. The peak pressure resulting from this accident occurs 50 s after
5. the rupture at a value of 53.3 psig.

To demonstrate the effectiveness of the reactor building spray cooling the 14.1 ft^2 rupture was analyzed with 1,500 gpm spray (1 spray pump and 1 spray header) plus 2 fan coolers operating. The reactor building pressure as a function of time is shown on Figure 14-57. The peak pressure for this case
5. is 53.2 psig.

Figures 14-58 through 14-61 show the reactor building pressure for the other rupture sizes analyzed with 3 fan coolers operating.

Figure 14-62 shows the reactor building energy inventory as a function of time after rupture for 14.1 ft^2 rupture with three fan coolers operating. These curves show the effectiveness of the reactor building structures and emergency cooling units.

Figure 14-63 shows the reactor building vapor temperature and sump temperature following a 14.1 ft^2 rupture.

The peak reactor building pressure shown in this analysis is 53.9 psig and is the result of a 5.0 ft^2 rupture in the reactor outlet piping. This case assumes three reactor building emergency cooling units, having the heat removal capacity shown in Figure 6-11, operate. Figures 14-59, 14-63a, and 14-63b show the reactor building pressure, energy inventory and temperature versus time for this case. For this analysis Table 14-13a lists a chronology of events following initiation of the pipe rupture.

Table 14-13a
Chronology of Events Following the 5.0 ft^2 Rupture of the Reactor Coolant System

		Event	Time, Sec.
	1.	Design Break Occurs	0.0
	2.	Core Flooding Tanks Start Injection	13.5
	3.	Reactor Building Emergency Air Coolers Start	25
	4.	Core Flooding Tanks Empty	36
5.	5.	Pumped Core Injection Starts	36
	6.	Peak Reactor Building Pressure Occurs	61
	7.	Borated Water Storage Tank Empty	6036

To supplement Figure 14-63a, an energy balance at the time peak pressure occurs (61 sec) is listed in Table 14-13b. This includes the energy stored prior to the LOCA, the energy generated and absorbed from t=0 to the time of peak pressure and the energy distribution at the time of peak pressure. Figure 14-63c shows the total mass released to the reactor building as a function of time corresponding to the total energy of Figure 14-63a. Also shown on this figure are plots of water entering the RCS from the core flooding system and the borated water storage tank.

Figure 14-63d shows the mass release rates to the Reactor Building as a function of time for this case. The corresponding BTU's per pound for the above mass rates are shown in Figure 14-63e. For this case the water level reaches the reactor vessel nozzle height at 135 seconds and thereafter full injection mass flow is released to the reactor building.

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Table 14-13b

An Energy Balance Table Before the LOCA
and at the Time of
Peak Reactor Building Pressure for the 5.0 ft^2

	Item	t=0 Before LOCA BTU x 10^{-6}	t=61* At Time of Peak R.B. Pressure BTU x 10^{-6}
1.	Reactor Coolant Internal Energy	306.7	31.7
2.	Core Flood Tanks Internal Energy	9.1	0.0
3.	Stored Energy (280 F Datum):		
	a. Fuel and Cladding	25.1	3.5
	b. Core Internals	10.5	7.8
	c. Reactor Vessel Metal	30.4	30.2
	d. Piping, Pumps and Valves	22.3	22.2
	e. Steam Generator Metal	31.3	31.3
5. 4.	Energy Transferred to Secondary System	--	0.0
5.	Reactor Building Air Energy (110 F Datum)	0.0	3.9
6.	Reactor Building Steam Internal Energy	0.0	242.6
7.	Internal Energy of Water on Floor of Reactor Building	0.0	49.8
8.	Energy Transferred to Steel Structures (110 F Datum)	--	13.4
9.	Energy Transferred to Concrete Structures (110 F Datum)	--	10.4
10.	Energy Removed by Reactor Building Emergency Cooling Units	--	2.4
11.	Secondary Coolant Internal Energy	72.0	72.0

*Includes 15.3 x 10^6 BTU of decay heat and shutdown energy generated from t=0 to 61 sec.

Figures 14-63f through 14-63j show results of a study made in conjunction with the NPSH calculations given in Section 6.5.2. Figures 14-63f through 14-63h show the reactor building sump temperature, vapor temperature, and pressure, respectively, for a 5.0 ft^2 rupture in the reactor outlet piping for various numbers of fan coolers operating (no sprays operating). The 5.0 ft^2 rupture case was chosen for this analysis because it results in the largest reactor
19. building pressure.

Figure 14-63i shows the reactor building pressure for a 5.0 ft^2 rupture in the reactor outlet piping for a case with two fan coolers operating and one reactor building spray pump operating at 1,500 gpm until recirculation switchover and 1,000 gpm thereafter. Figure 14-63j shows the reactor building sump and vapor temperature for this same case.

14.2.2.3.6 Reactor Building Pressure Margin

5. The results of the spectrum analysis (14.2.2.3.5) show that there is a 5.1 psi margin between design pressure and the peak pressure following a LOCA. The margin can be expressed in terms of the reactor building capability to withstand additional energy release without exceeding design pressure (59 psig). For this discussion the double-ended rupture of the 36-in. ID outlet pipe with three air coolers providing building cooling is used as the base case.

Figure 14-64 is a plot of total reactor building pressure as a function of internal energy in the steam-air mixture assuming saturated conditions. Point A on the curve corresponds to the conditions at the end of blowdown for the 14.1 ft^2 rupture in the spectrum analysis. In order for the blowdown pressure peak to reach design pressure (Point B) an additional 35 x 10^6 Btu's in form of saturated steam would have to be added to the reactor building atmosphere. Approximately 27 x 10^6 Btu's of energy transferred from the core to the coolant would produce the amount of saturated steam required to reach Point B on Figure 14-64. At the end of blowdown (10 s) there are 15.0 x 10^6 Btu's stored in the core and another 7.0 x 10^6 Btu's stored in core internals metal. Therefore, if an unrealistically high heat transfer rate could be achieved to cool the entire core region down to a base temperature of 286 F during blowdown the building pressure would not exceed design pressure.

The capability to withstand energy addition in the form of superheat is less than for saturated steam. For example, 9.0 x 10^6 Btu's added as superheat (i.e., no change in mass in vapor region) after blowdown would increase building pressure to 59 psig (point C on Figure 14-64). This would correspond to the amount of superheat generated due to hydrogen-oxygen recombination from a metal-water reaction of 10% of the zirconium in the core. For the core to experience this great a zirconium-water reaction, a large amount of the energy transferred to the building during blowdown for the base case analysis would have to remain in the core, and this would lower the predicted building pressure at the end of blowdown. The emergency core cooling system limits peak clad temperature to well below 2,300 F. The core cooling analysis shows that no zirconium-water reaction occurs during blowdown, and less than 0.1% reaction occurs for the duration of the accident.

Shortly after blowdown is over, the reactor building cooling system(s) goes into operation to reduce building pressure. Figure 14-65 shows the ability of the reactor building vapor, with 3 air coolers operating after a delay of 25 seconds, to absorb additional energy in the form of saturated steam without exceeding design pressure. This margin does not include any increased heat removal rates for the air coolers at higher vapor temperatures.

The analysis of Section 14.2.2.3.5 was performed on the basis that only partial injection flow (i.e., 2 core flooding tanks plus 3,500 gpm pumped injection) is available to provide core cooling. An analysis was made to show the effect of further degraded performance of emergency core cooling. This case assumes that after blowdown there is only enough injection water to sustain boiloff from the reactor vessel. This provides for continuous addition of all energy to the reactor building vapor region in the form of saturated steam as it is generated. This case results in the same peak pressure as the base case but a slower depressurization (Figure 14-66). Even for this case a considerable margin exists for additional energy release to the reactor building without reaching design pressure.

14.2.2.3.7 Environmental Analysis

The analysis of the environmental consequences of any loss-of-reactor-coolant accidents includes the assumption that 1 per cent of the fuel rods in the core have been defective prior to the release of reactor coolant to the reactor building. Table 11-3 lists the resulting activity in the reactor coolant. The fission product release to the reactor building in the event of a 36-in. diameter pipe rupture LOCA is the gap activity of all fuel rods plus the reactor coolant activity which results from reactor operation for 200 days with 1 per cent defective fuel. The noble gas activity released is shown in Table 14-14.

Table 14-14
Noble Gas Release From Maximum Break Size LOCA

Isotope	Activity, curies
Kr-83m	8.99×10^3
Kr-85m	4.95×10^4
Kr-85	4.38×10^5
Kr-87	2.70×10^4
Kr-88	8.87×10^4
Xe-131m	8.13×10^4
Xe-133m	9.32×10^4
Xe-133	8.53×10^6
Xe-135m	2.77×10^4
Xe-135	3.25×10^4

Of the iodine released, 50 per cent is assumed to plate out and the other half is assumed to remain in the reactor building atmosphere where it is available for leakage. No credit is taken for removal of airborne iodine by the reactor building spray systems. To facilitate environmental dose calculations, all isotopes of iodine have been equated into dose equivalent curies of iodine-131. The dose equivalency factor is determined by considering the concentration and specific dose of each iodine isotope present over the period of interest. The iodine dose to the thyroid per curie is obtained from the values given in TID-14844. The iodine activity released to the reactor building is 1.43×10^6 dose equivalent curies of iodine-131.

The reactor building pressure history for this accident is shown in Figures 14-56 through 14-61. While the reactor building leakage rate will decrease rapidly as the pressure decays, the leakage is assumed to remain constant at
18. the rate of .25 per cent of reactor building volume per day for the first 24 hours. Thereafter, since the reactor building has returned to nearly atmospheric pressure, the rate is assumed to be reduced to .125 per cent of the reactor building volume per day and to remain at this value for the duration of the accident.

It is assumed that 50 per cent of the reactor building leakage will go into the penetration rooms which will be maintained at a negative pressure as described in 6.5. The atmosphere in these rooms is discharged through charcoal filters to the unit vent. The charcoal filters are assumed to be 90 per cent efficient for iodine removal. The remaining 50 per cent of the reactor building leakage is assumed to escape directly to the atmosphere. By this method a maximum of 55 per cent of the iodine released from the reactor building is ultimately released to the atmosphere. Atmospheric dilution of the leakage discharged from the unit vent is calculated using the elevated release dispersion factor of 3.35×10^{-5} s/m^3 developed in Section 2.3. Dilution of the other leakage from the reactor building is calculated using the ground release disper-
18. sion factor of 1.16×10^{-4} s/m^3.

A breathing rate of 3.47×10^{-4} m^3/s is assumed for the 2-hour exposure. For the 30-day exposure, a breathing rate of 2.32×10^{-4} m^3/s is assumed.

The total integrated thyroid doses resulting from this LOCA fission product release are 4.6 Rem for the 2-hour exposure at the 1 mile exclusion distance, 18. and 5.0 Rem for the 30-day exposure at the 6 mile low population distance. The corresponding whole body doses are 0.010 Rem and 0.014 Rem. These doses are shown in Table 14-17.

The effect on the dose from an LOCA release was also investigated in the event a LOCA occurred while the reactor building was being purged. Assuming that all the fission products had been initially released to the reactor building, which was at a constant pressure of 59 psig, the flow from the purge vents would resemble flow through a long duct. With an isolation valve closing time of 5 seconds and with no credit taken for flow restrictions due to the closing valve, additional dose equivalent curies of I-131 will be released. This release corresponds to a 0.05 Rem additional thyroid dose at the site boundary.

14.2.2.3.8 Pressure Buildup In Primary And Secondary Due To Loss-Of-Coolant Accident

Following the design basis accident pipe rupture within either compartment (reactor cavity, steam generator), high-enthalpy water flows out of both ends of the pipe, flashing partly to steam. As the pressure builds up within the compartment, the steam-air-water mixture will flow through openings in the compartment into the main containment. The pressure built up in the compartment is dependent on the number and shape of vent areas leading into the main containment, the volume of the compartment, and the blowdown rate from the broken pipe.

14.2.2.3.8.1 Pressure Buildup Within The Reactor Cavity

The cavity model, as analyzed, consists of a 5520 cubic foot compartment, that has a vent (always open) and concrete shield plugs which blowout during the accident. A value of six (6) square feet, representing the effective (always
5. open) vent area of the cavity, was used. The total effective area of the vent and the shield plugs (after shield plugs blow out) is 75 square feet. The effect of the increase in area with time, as the shield plugs blow, has been taken into account in the analysis by calculating the acceleration of the plugs due to the pressure buildup within the reactor cavity. The results of the analysis are presented in Figure 14-68. The indicated rapid pressure drop results from the shield plug blowout.

<u>Largest Break Size Reactor Cavity Can Withstand</u>

An analysis of the preliminary design of the reactor cavity was initially made in June 1967 to determine the largest break size that the cavity could withstand. The documentation established a break size of 8 square feet with the associated peak pressure of 195 psi across the cavity walls. Reanalysis of the final design determined that the cavity could withstand a somewhat higher pressure (205 psi) associated with a larger break size (8.5 square feet).

Largest Break Size Possible Within Reactor Cavity

Similarly, the analysis of the largest break size possible that could occur within the reactor cavity was calculated to be 3.0 square feet which could produce a differential pressure across the cavity walls of 120 psi. This is accomplished by restraints on the primary lines which limit the physical separation of the ends of the ruptured lines. The cavity, therefore, can safely withstand the pressures resulting from the largest break possible in the coolant pipes within the reactor cavity. A chart and tabulation of the reactor cavity pressures associated with various break sizes is shown on Figure 14-68.

14.2.2.3.8.2 Pressure Buildup Within The Steam Generator Compartments

There are two steam generator compartments designated East and West. The idealized geometry used assumes two separate compartments. The East compartment has an "effective" volume of 60,400 cubic feet and an "effective" flow area of 1000 square feet. The West compartment has an "effective" volume of 61,700 cubic feet and an "effective" flow area of 1100 square feet. Only one blowdown was investigated for these calculations: the hot-leg rupture, since this case represents the greatest energy release rate.

The pressure differential across the East and West generator compartment walls as a function of time is presented in Figures 14-69 and 14-70. The values associated with these differentials are 9.68 psi and 7.98 psi, respectively.

Maximum and Design Pressures Within Steam Generator Compartments

The table below gives a summary of the maximum pressure and the design pressure in the steam generator compartments:

STEAM GENERATOR COMPARTMENTS

BREAK SIZE	MAX. PRESS.		DESIGN PRESS.	MAX./DESIGN PRESS.	
	East	West		East	West
14.0 S. F.	9.7 psi	8.0 psi	11.1 psi	87%	72%

14.2.2.3.8.3 Assumptions Were Incorporated In The Analysis

The steam and air throughout the compartment are in thermal equilibrium at all times.

During each time-step the blowdown mass expands isenthalpicly to the total compartment pressure. The water present at that time could form more steam only be (slow) evaporation. The water is assumed to undergo no further change of phase and drops to the floor of the compartment. The steam completes its isenthalpic expansion to the partial pressure of the steam already in the compartment and mixes homogeneously with the air.

During each time-step the mass of water added is temporarily stored. If the equilibrium calculation indicates atomsphere superheating, then a sufficient quantity of this temporarily stored water is flashed into steam such that the atmosphere is just saturated. If there is not sufficient water in the temporary storage to saturate the atmosphere, then the atmosphere is allowed to superheat.

The flow out of the compartment into the main containment is calculated using the compressible flow equations for subsonic and choked flow. The orifice coefficient and expansion factors used are discussed in Section 14.2.2.3.8.4.

The flow is assumed to become sonic at the critical pressure ratio defined by:

$$\frac{P_2}{P_1} = \left[\frac{2}{k + 1}\right]^{\frac{k}{k - 1}}$$

Where:

P_2 = downstream pressure

P_1 = upstream pressure

k = isentropic exponent (see Section 14.2.2.3.8.4)

The volume of the compartment is measured to the minimum cross sectional area of each of the flow openings leading out of the compartment.

14.2.2.3.8.4 Calculation of Orifice Coefficient and Expansion Factor

The orifice coefficients (C) used in the orifice flow relation are sensitive to Reynolds number, orifice size, and orifice shape. Orifice coefficients typically become independent of Reynolds number at high Reynolds numbers. Reynolds numbers through orifices in the present area are greater than 10^6, so the Reynolds number effect does not apply here.

Sufficient experimental information on orifice coefficients for our geometry is not available. However, information is available on the head loss coefficient (K) defined by:

$$P = K \frac{V^2}{2} \qquad (1)$$

Where:

P = Pressure drop across orifice

V == Velocity through orifice

The relation between (C) and (K) is: $C = \frac{1}{\sqrt{K}}$ (2)

The advantage of the head loss coefficient is that the total head loss for a complicated flow system can be determined from an equivalent K obtained by adding K's for separate parts of the system (i.e. bends, expansions, contractions, etc.) as follows:

$$K_{eff} = \Sigma_i K_i \qquad (3)$$

Orifice Coefficient Data

Available information on nominal orifice coefficients is summarized in Figure 14-73.

Head Loss Coefficient Data

The required experimental head loss coefficient data are represented in Figures 14-74 and 14-75.

Expansion Factor

The expansion factor used in the orifice flow relation is that applicable to a converging nozzle:

$$Y = \sqrt{\frac{1 - \left[\frac{A_2}{A_1}\right]^2}{\left[\frac{A_2}{A_1}\right]^2 \left[\frac{P_2}{P_1}\right]^{\frac{2}{k}}} \quad \frac{\frac{k}{k-1}\left[\frac{P_2}{P_1}\right]^{\frac{2}{k}} \left[1 - \left[\frac{P_2}{P_1}\right]^{\frac{k-1}{k}}\right]}{1 - \frac{P_2}{P_1}}}$$

Where:

A_1 = upstream area

A_2 = orifice area

P_1 = upstream pressure

P_2 = downstream pressure

k = isentropic exponent

The isentropic exponent (k) for these calculations is based on the mixture of air and steam.

This calculation is conservative becuase for orifices which are not well rounded compressibility effects increase the expansion factor slightly.

14.2.2.4 Maximum Hypothetical Accident

14.2.2.4.1 Identification of Accident

The analyses in the preceding sections have demonstrated that even in the event of a loss-of-coolant accident, no significant core melting will occur. However, to demonstrate in a still more conservative manner that the operation of a nuclear power plant at the proposed site does not present any undue hazard to the general public, a hypothetical accident involving a gross release of fission products is evaluated. No mechanism whereby such a release occurs is postulated, since this would require a multitude of failures in the engineered safeguards which are provided to prevent such an occurrence. Fission products are assumed to be released from the core as stated in TID-14844, namely, 100 percent of the noble gases, 50 percent of the halogens, and 1 percent of the solids. Further, 50 percent of the iodines released to the Reactor Building are assumed to plate out. Other parameters such as meteorological conditions, iodine inventory of the fuel, Reactor Building leak rate, etc., are the same as assumed for the loss-of-coolant accident in 14.2.2.3.7. The average iodine inventory, in terms of dose equivalent curies of iodine-131 available for leakage at different time periods after the accident, is as follows:

Post-Accident Time	Iodine Activity Available (Dose Equivalent Curies I-131)
0 - 2 hours	29.1×10^6
0 - 1 day	23.1×10^6
1 - 30 days	5.2×10^6

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(Carryover)

The noble gas activity released to the reactor building is shown in Table 14-15.

Table 14-15
Noble Gas Release From MHA

Isotope	Reactor Building Activity, Curies
Kr-83m	7.32×10^6
Kr-85m	2.16×10^7
Kr-85	5.37×10^5
Kr-87	3.95×10^7
Kr-88	5.98×10^7
Xe-131m	5.48×10^5
Xe-133m	3.14×10^6
Xe-133	1.30×10^8
Xe-135m	3.42×10^7
Xe-135	2.61×10^7

14.2.2.4.2 Environmental Analysis

The meterology model is as given in Section 2.3, and the reactor building leak
rate is assumed to be .25 per cent per day by volume for the first 24 hours,
18. and then .125 per cent per day by volume for the next 29 days. The other parameters used are given in TID-14844.

The direct dose to the whole body following the accident is shown in Figure 14-67. No significant dose exists from this source at the exclusion distance.

The total integrated thyroid doses resulting from this MHA fission product release are 186 Rem for 2-hour exposure at the 1-mile exclusion distance and 144
18. Rem for the 30-day exposure at the 6-mile low population distance. The corresponding whole body doses are 1.4 Rem and .65 Rem. These doses are listed in Table 14-17 and are below the guideline values of 10 CFR 100.

14.2.2.4.3 Effect of Washout During the Maximum Hypothetical Accident

To provide a further evaluation of the suitability of the site, the effects of washout on surrounding drinking water reservoirs following the maximum hypothetical accident were analyzed. Calculations were made for the case of continuous rain lasting 24 hours covering the general area of the reservoir and the site. The maximum washout rate as a function of distance is calculated from the following equation:[20]

$$\omega_{max} = \frac{Q_o \, e^{-(y^2/2\sigma_y^2)}}{xe\sigma_y \sqrt{2\pi}}$$

where

ω_{max} = maximum washout rate, curies per s per m^2

x = downwind distance, meters

σ_y = horizontal dispersion, meters

y = crosswind distance from plume axis, meters

Q_o = release rate, curies per s

The equation above is conservative since the results do not consider the wind speed or vertical distribution in the cloud. The wind direction is assumed to remain towards Lake Keowee for the 24-hour period with the plume center lines uniformly distributed over this section. Washout is assumed to occur under neutral stability conditions, Pasquill D, which is typical for a rainy day.[20]

The average release rate from the reactor building during the 24-hour period
18. following the accident is 0.37 equivalent curies of iodine-131 per second.
Using the above equation, the maximum iodine washout is calculated by assuming
that all of the iodine that has washed out remains in the surrounding reservoir
and is not affected by runoff. The average number of curies in the reservoir
during a one-year period is reduced by a factor of 0.0318 due to the natural
decay of iodine. Assuming that this activity mixes in the reservoir and that
an adult drinks 0.8 m^3 per year [21] of the contaminated water, the total dose
to the thyroid has been calculated using the methods of TID-14844. The near-
est drinking water intake is approximately six miles from the site. At this
18. distance, the total integrated one-year ingestion dose to the thyroid is 1.0
rem. This dose is well below the limits of 10 CFR 100.

14.2.2.4.4 Effects of Engineered Safeguards Systems Leakage During the Maximum Hypothetical Accident

An additional source of fission product leakage during the maximum hypothetical accident can occur from leakage of the engineered safeguards systems external to the reactor building during the recirculation phase for long-term core cooling. A detailed analysis of the potential leakage from these systems is presented in 6.4. The analysis demonstrates that the maximum leakage is about 6,140 cc/h. About 250 cc/h of this leakage will flash into steam.

It is assumed that the water being recirculated from the reactor building sump through the external system piping contains 50 per cent of the core saturation iodine inventory. This is the entire amount of iodine release from the reactor coolant system. The assumption that all of the iodine escaping from the reactor coolant system is absorbed by the water in the reactor building is conservative since much of the iodine released from the fuel will be plated out on the building walls. The activity in the recirculation water is equal to 0.034 dose equivalent curies of I-131 per cc of water. It is assumed that all the iodine contained in the 250 cc/h of water which is flashed is released to the auxiliary building atmosphere. Iodine release from the remaining water is calculated using a gas/liquid partition coefficient of 9×10^{-3}. This results in a total release to the atmosphere of 10.3 dose equivalent curies of I-131.

The auxiliary building is ventilated and discharges to the unit vent. The activity is assumed to be released as a puff from the unit vent. Atmospheric dilution is calculated using the 2-hour elevated release dispersion factor of 3.35×10^{-5} s/m^3 developed in Section 2.3. The total integrated dose to the thyroid from this activity is 0.18 Rem for a 2-hour exposure at the 1-mile exclusion distance. This dose is well below the limits of the 10 CFR 100 guidelines.

14.2.2.5 Waste Gas Tank Rupture

Rupture of a waste gas tank would result in the release of its radioactive contents to the plant ventilation system and to the atmosphere through the unit vent. This accident was analyzed in order to evaluate the resultant dose at the site boundary.

A tank is assumed to contain its maximum possible inventory of gas activity. The gaseous activity in the tank is:

Table 14-16
Waste Gas Tank Inventory

Isotope	Activity, Curies
Kr-85m	501
Kr-85	3,050
Kr-87	284
Kr-88	902
Xe-131m	735
Xe-133m	936
Xe-133	80,900
Xe-135m	314
Xe-135	2,210
Xe-138	170
I-131	10.7
I-132	15.7
I-133	12.7
I-134	1.67
I-135	6.35

The auxiliary building is ventilated and discharges to the unit vent. The activity is assumed to be released as a puff. Atmospheric dilution is calculated using the 2-hour elevated release dispersion factor of 3.35×10^{-5} s/m^3 developed in Section 2.3. The total integrated dose at the 1-mile exclusion distance is 0.19 Rem whole body and 0.13 Rem thyroid. These doses are well below the limits of the 10 CFR 100 guideline.

Table 14-17
Dose Calculation Summary

		Dose, Rem	
	Item	2-Hours at Exclusion Distance	30-Days at Low Population Distance
	Maximum Hypothetical Accident		
18.	Thyroid	186	144
	Whole Body	1.4	0.65
	Loss-of-Coolant Accident		
18.	Thyroid	4.6	5.0
	Whole Body	0.010	0.014
	Loss of Electric Power		
	Thyroid	0.0008	--
	Steam Line Failure		
18.	Thyroid	0.19	--
	Whole Body	0.002	--
	Steam Generator Tube Failure		
	Thyroid	0.001	--
	Whole Body	0.007	--
	Fuel Handling Accident		
1.	Thyroid	0.43	--
	Whole Body	0.027	--
	Rod Ejection Accident		
18.	Thyroid	0.19	0.22
	Whole Body	0.001	--
	Waste Gas Tank Rupture		
	Thyroid	0.13	--
	Whole Body	0.19	--

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Rev. 18. 3/10/72

14.3 REFERENCES

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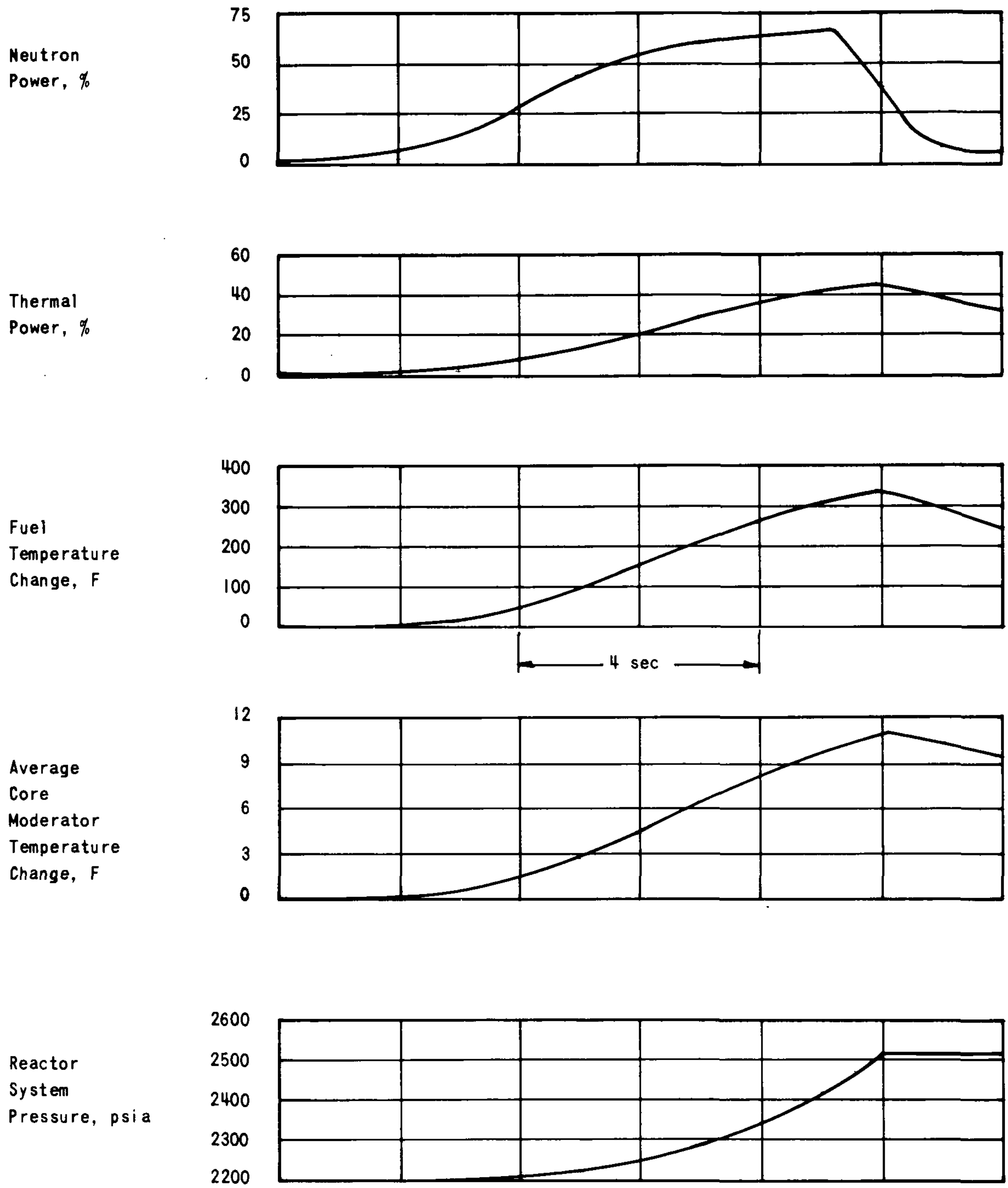
21.├ 3.



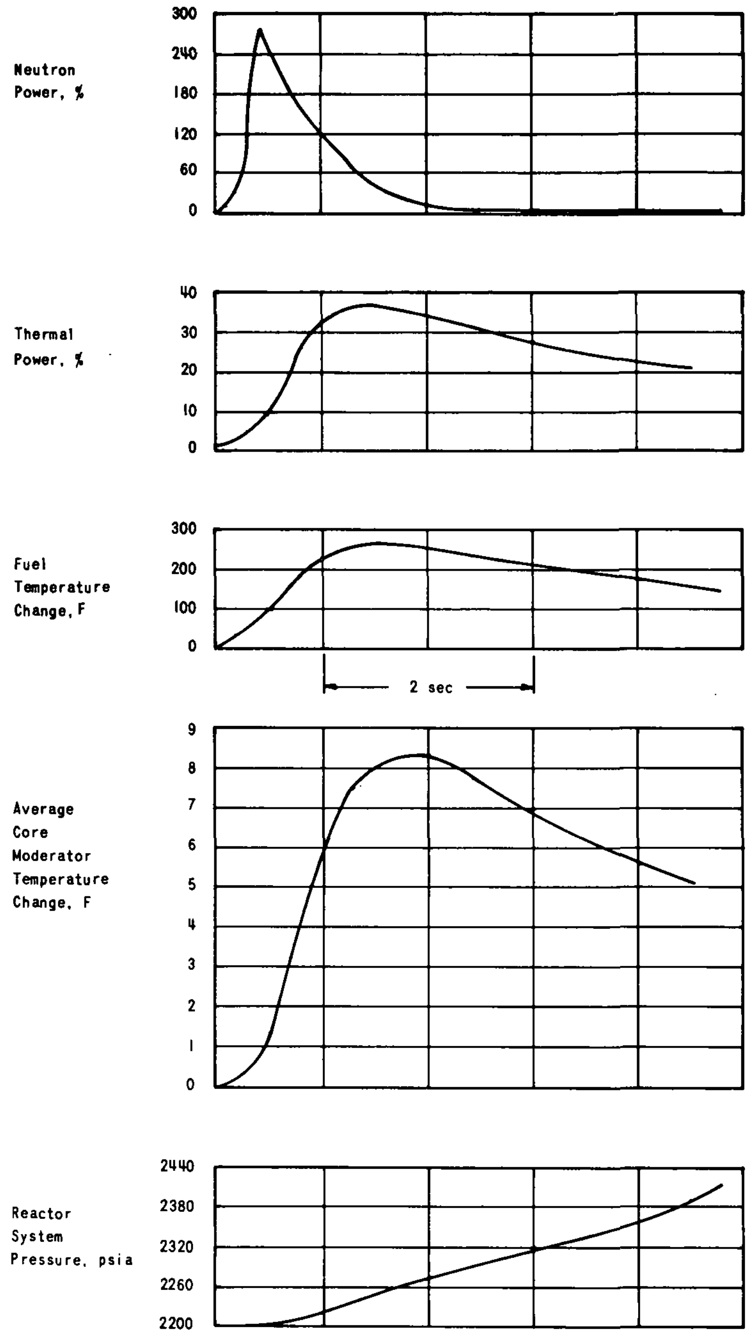
(Reference Supplement 9 Revisions for Oconee 3)

STARTUP ACCIDENT FROM 10^{-9} RATED POWER USING A 1.5% $\Delta k/k$ ROD GROUP: HIGH PRESSURE REACTOR TRIP IS ACTUATED

OCONEE NUCLEAR STATION

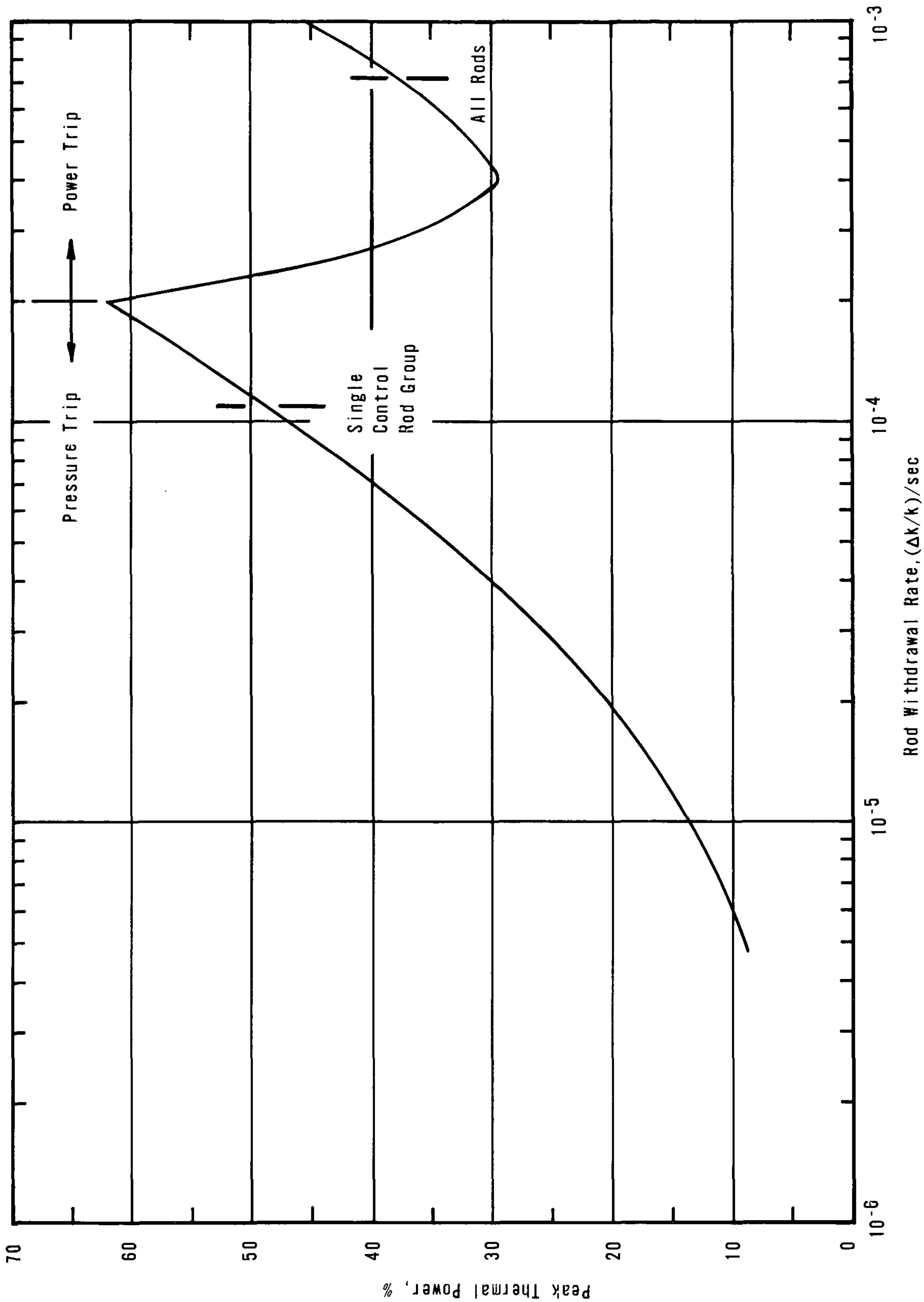
Figure 14 - 1

Rev. 16. 7/30/71



STARTUP ACCIDENT FROM 10^{-9} RATED POWER USING ALL RODS WITH A WORTH OF 10.0% $\Delta k/k$; HIGH FLUX REACTOR TRIP IS ACTUATED

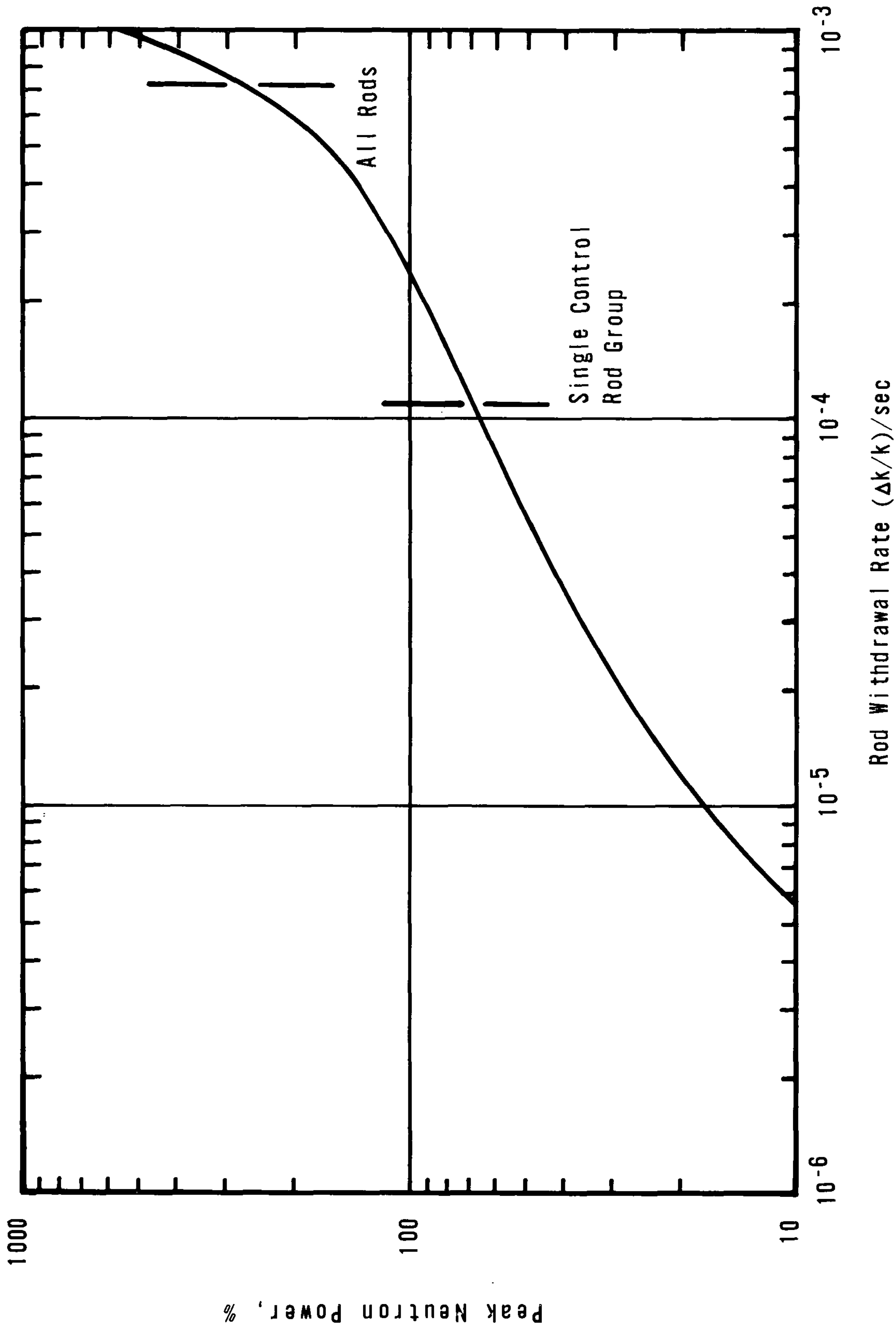
Figure 14 - 2



PEAK THERMAL POWER VERSUS ROD WITHDRAWAL RATE FOR A STARTUP ACCIDENT FROM 10^{-9} RATED POWER

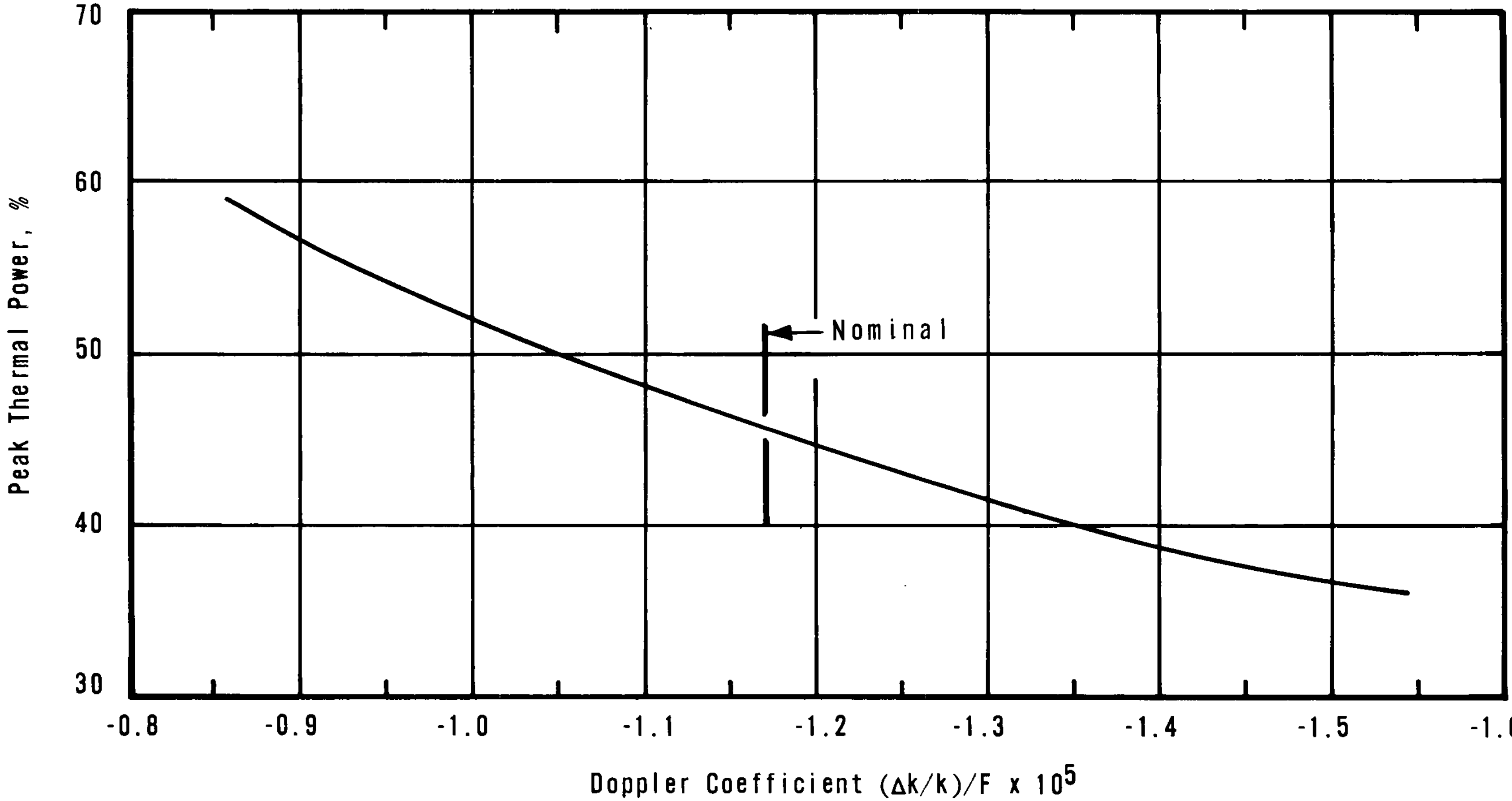
OCONEE NUCLEAR STATION

Figure 14 - 3



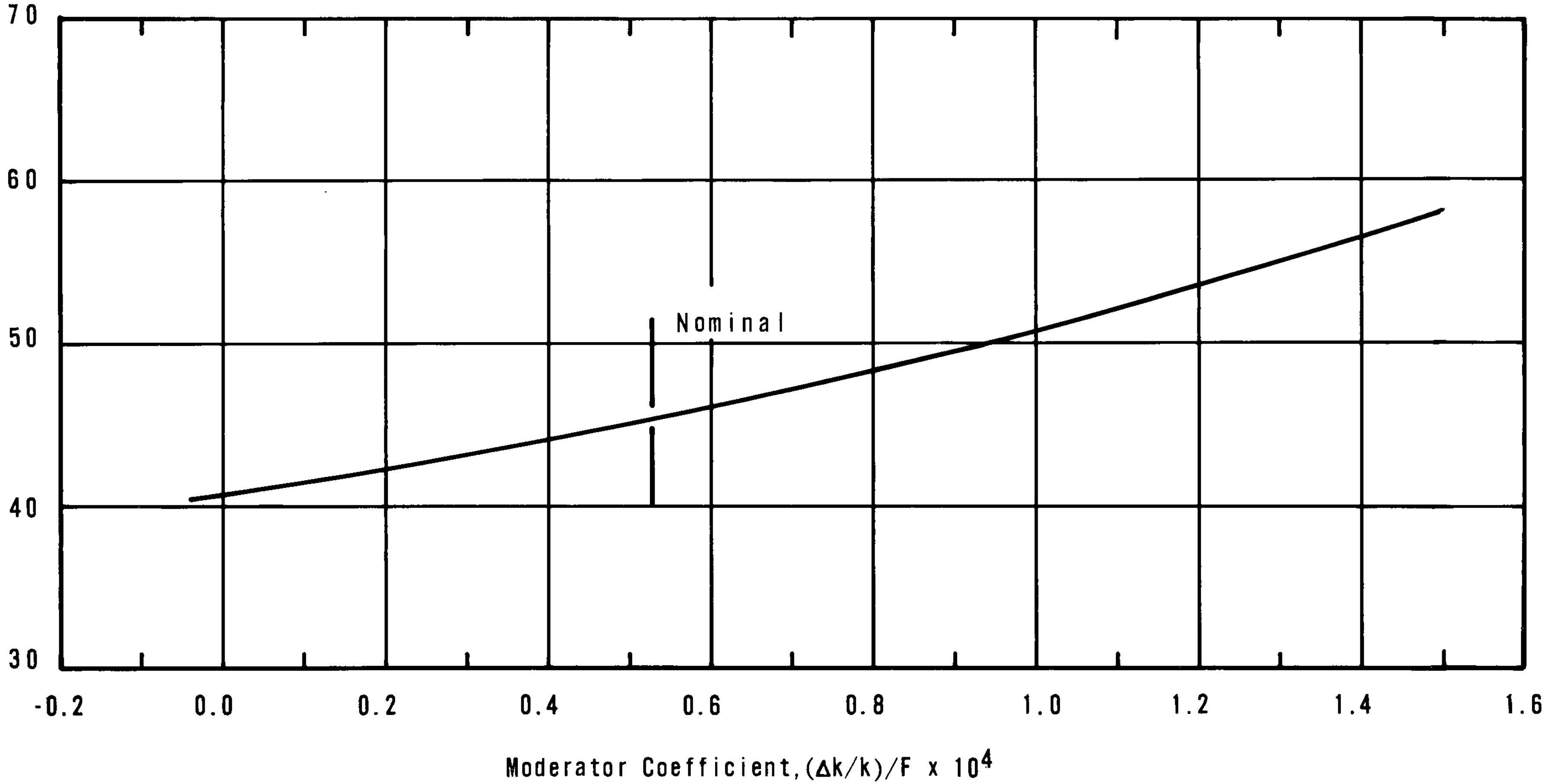
PEAK NEUTRON POWER VERSUS ROD WITHDRAWAL
RATE FOR A STARTUP ACCIDENT FROM 10^{-9}
RATED POWER

OCONEE NUCLEAR STATION
Figure 14 - 4



PEAK THERMAL POWER VERSUS DOPPLER COEFFICIENT FOR A STARTUP ACCIDENT USING A 1.5% $\Delta k/k$ ROD GROUP AT 1.09×10^{-4} $(\Delta k/k)$/SEC FROM 10^{-9} RATED POWER

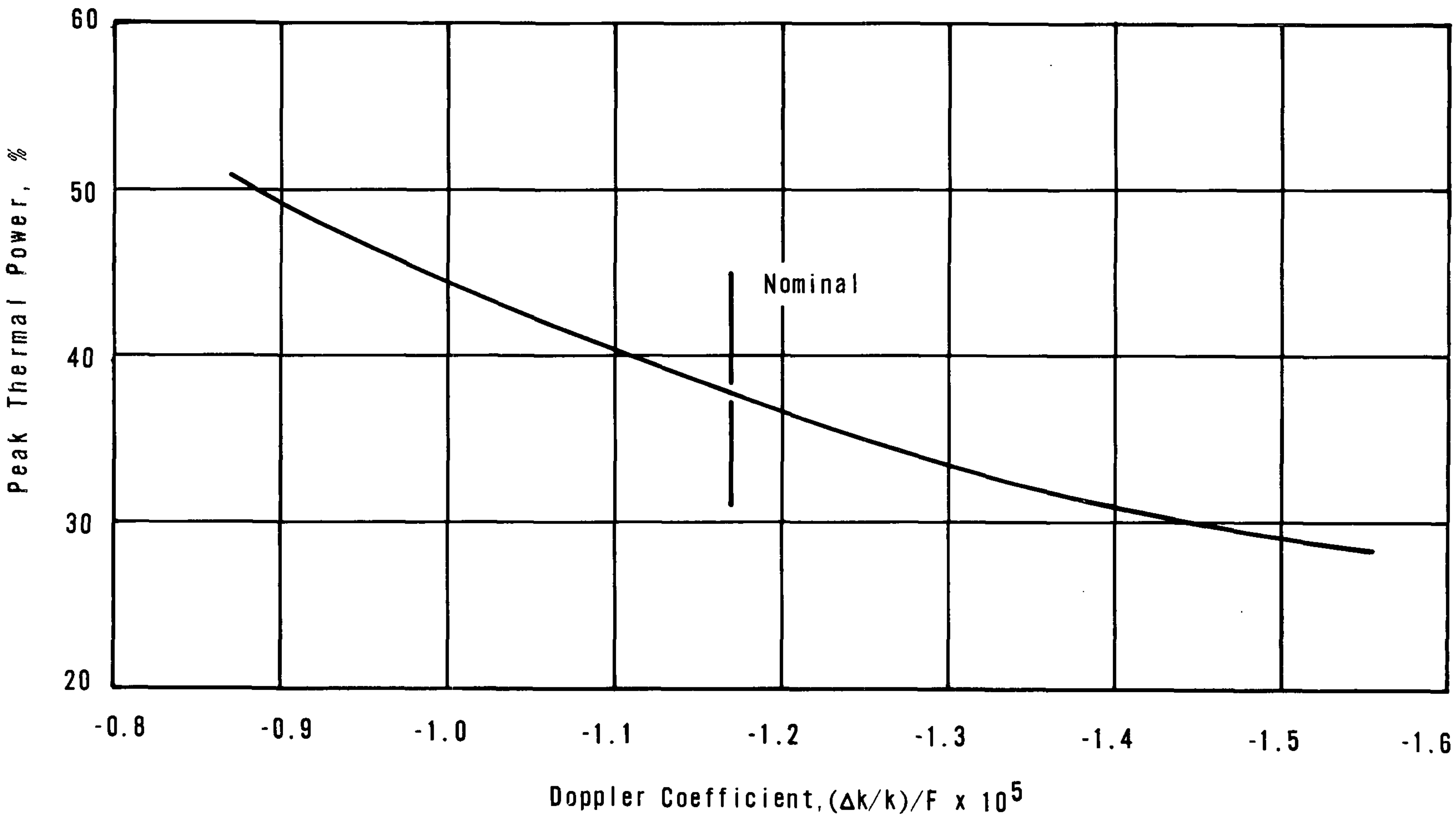
OCONEE NUCLEAR STATION
Figure 14-5



PEAK THERMAL POWER VERSUS MODERATOR COEFFICIENT FOR A STARTUP ACCIDENT USING A 1.5% Δk/k ROD GROUP AT 1.09 x 10^{-4} (Δk/k)/SEC FROM 10^{-9} RATED POWER

OCONEE NUCLEAR STATION

Figure 14 - 6



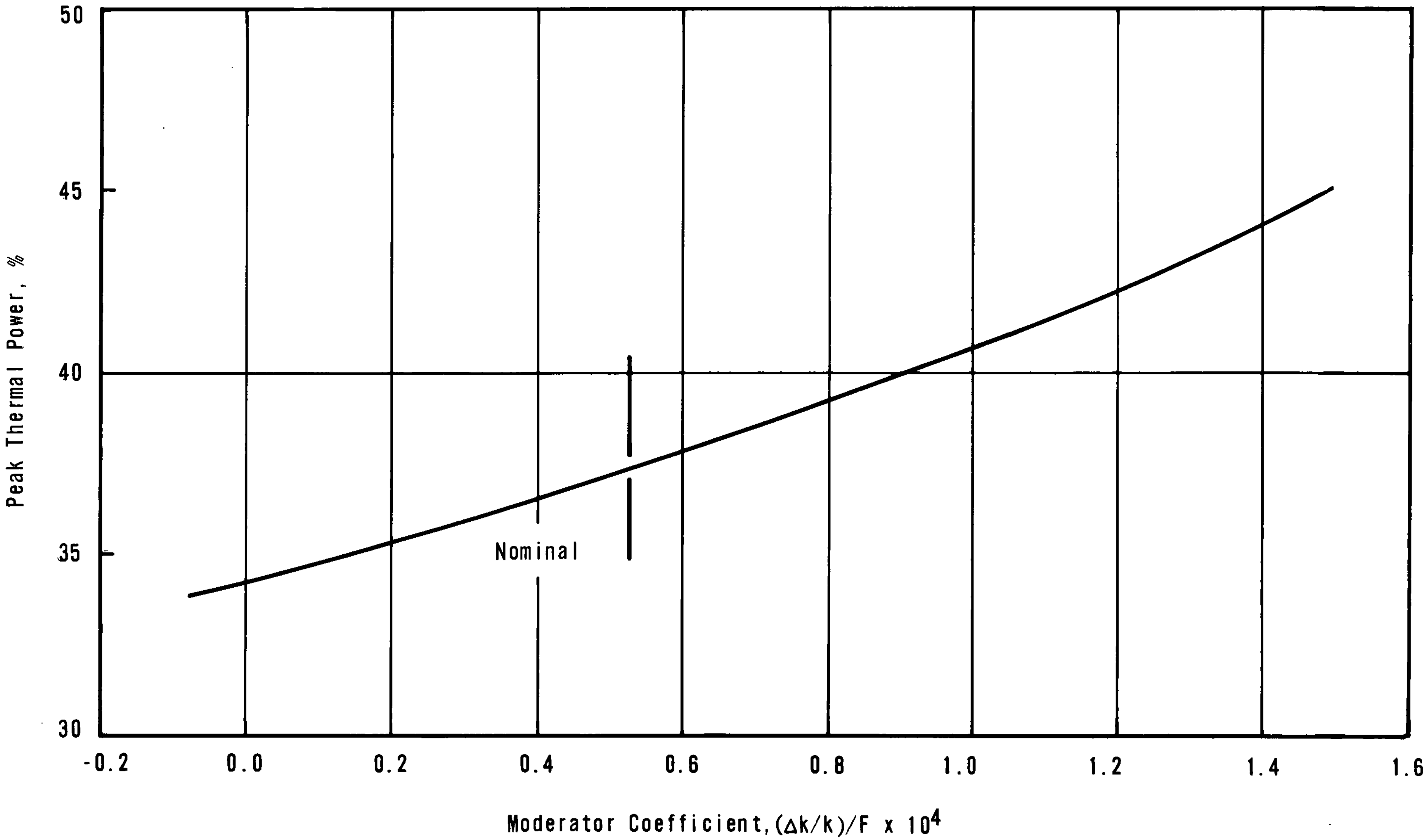
PEAK THERMAL POWER VERSUS DOPPLER COEFFICIENT FOR A STARTUP ACCIDENT USING ALL RODS AT 7.25 $\times 10^{-4}$ $(\Delta k/k)$/SEC FROM 10^{-9} RATED POWER

OCONEE NUCLEAR STATION

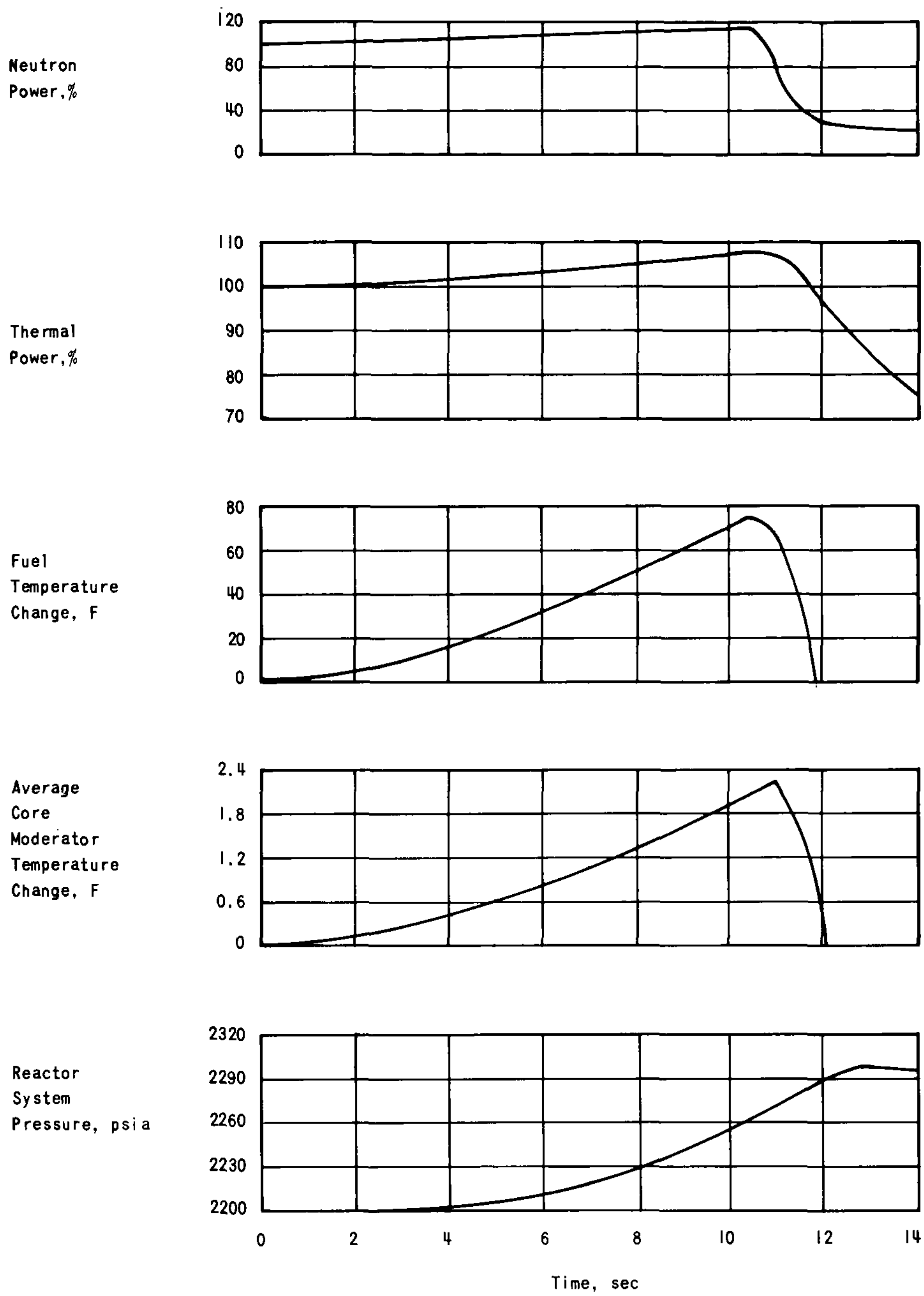
Figure 14-7

OCONEE NUCLEAR STATION

Figure 14-8



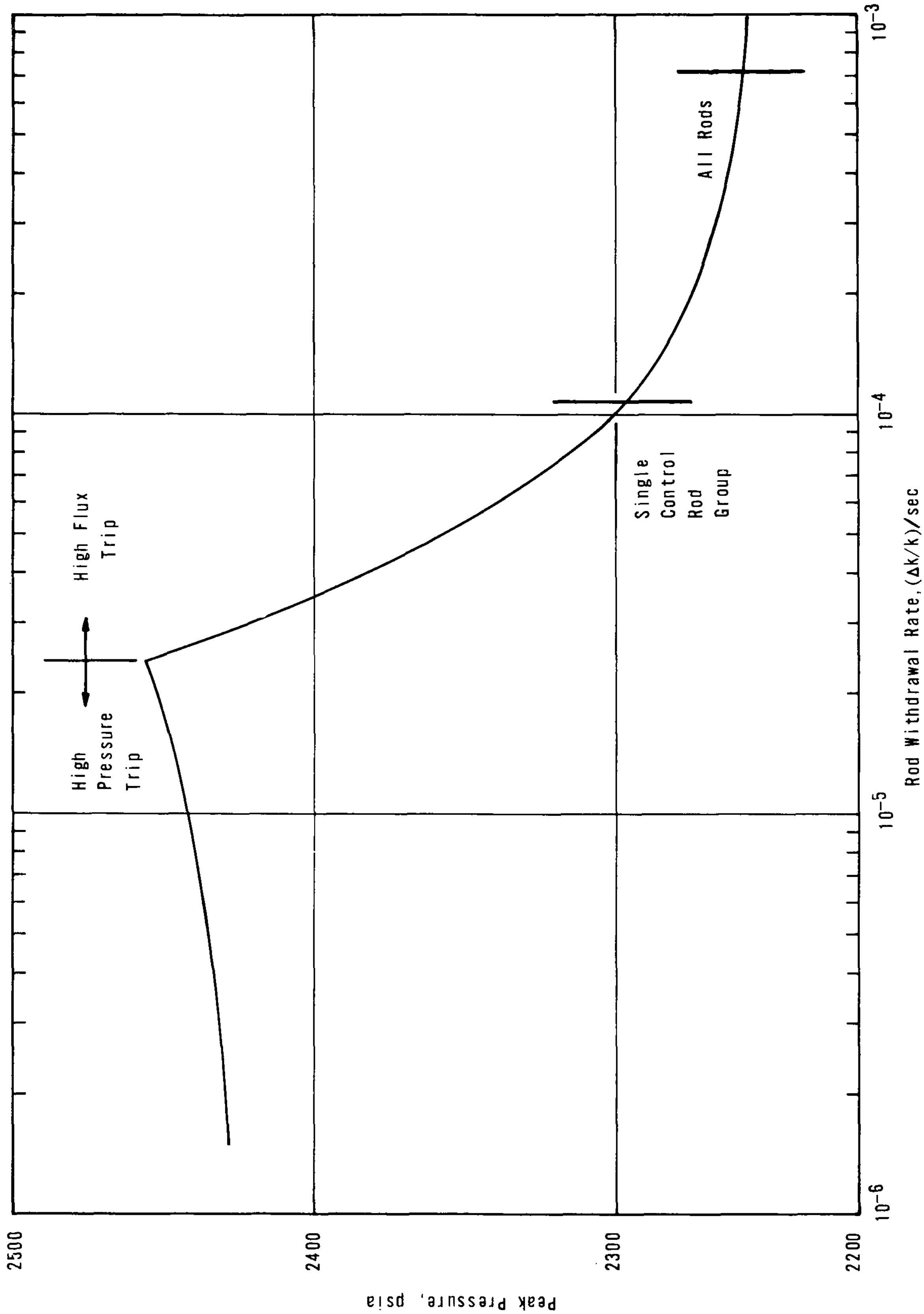
PEAK THERMAL POWER VERSUS MODERATOR COEFFICIENT FOR A STARTUP ACCIDENT USING ALL RODS AT 7.25×10^{-4} $(\Delta k/k)/$SEC FROM 10^{-9} RATED POWER



ROD WITHDRAWAL ACCIDENT FROM RATED POWER USING A 1.5% $\Delta k/k$ ROD GROUP AT 1.09×10^{-4} $(\Delta k/k)$/SEC; HIGH FLUX REACTOR TRIP IS ACTUATED

OCONEE NUCLEAR STATION

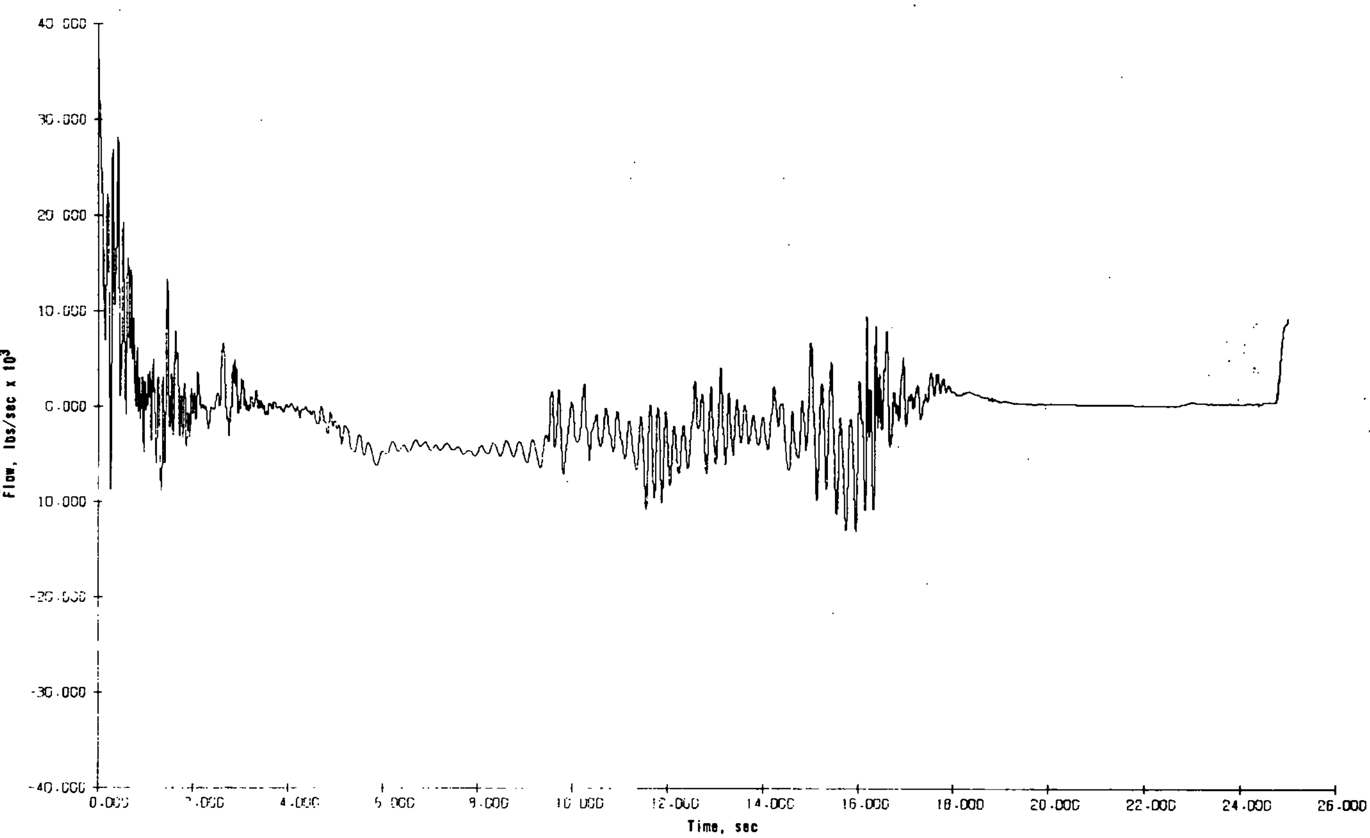
Figure 14 - 9



PEAK PRESSURE VERSUS ROD WITHDRAWAL RATE FOR A ROD WITHDRAWAL ACCIDENT FROM RATED POWER

OCONEE NUCLEAR STATION

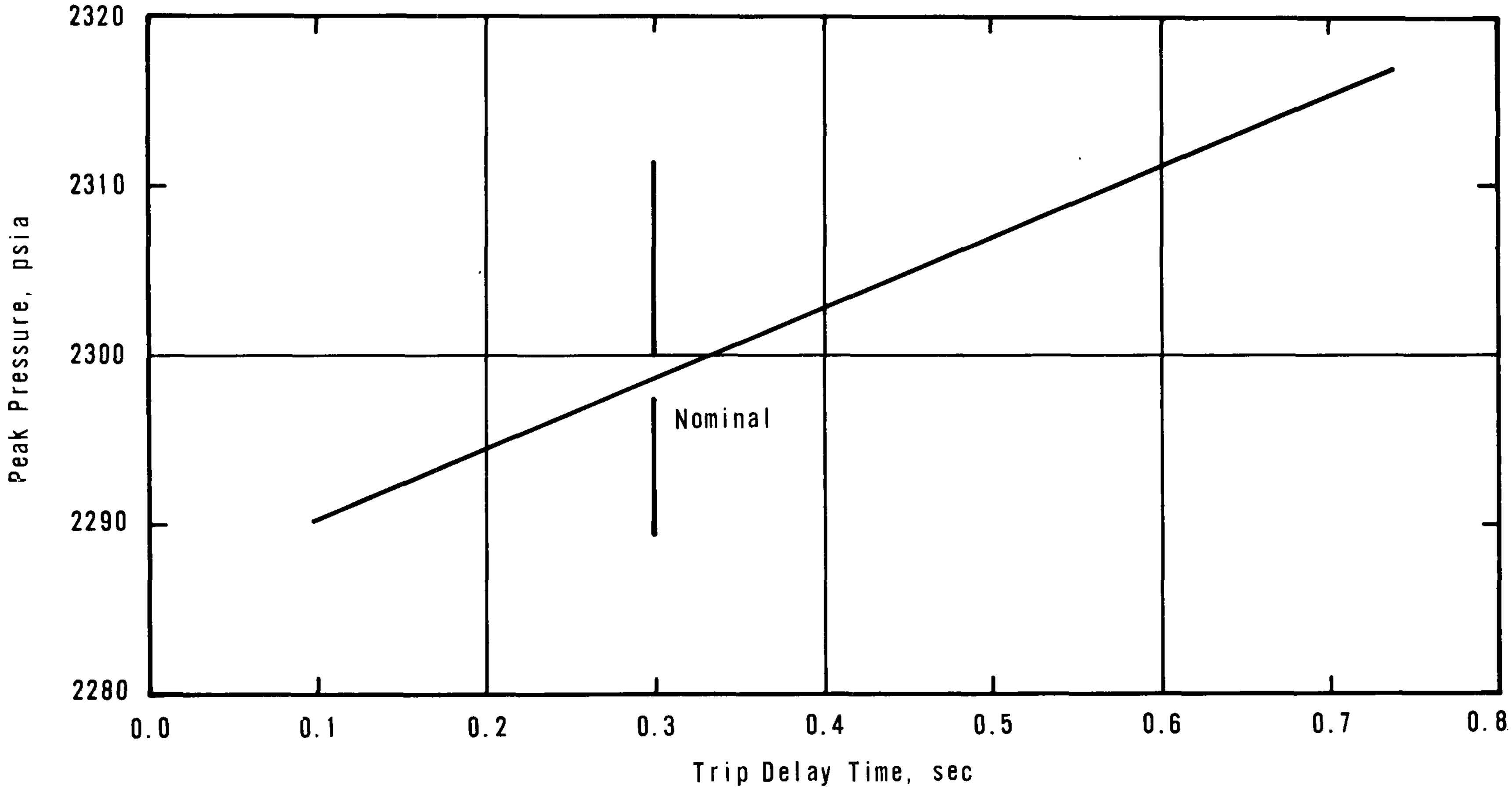
Figure 14 - 10



PATH 4

AVERAGE CORE FLOW (FLOW PATH 4)
8.5 FT^2 COLD LEG BREAK

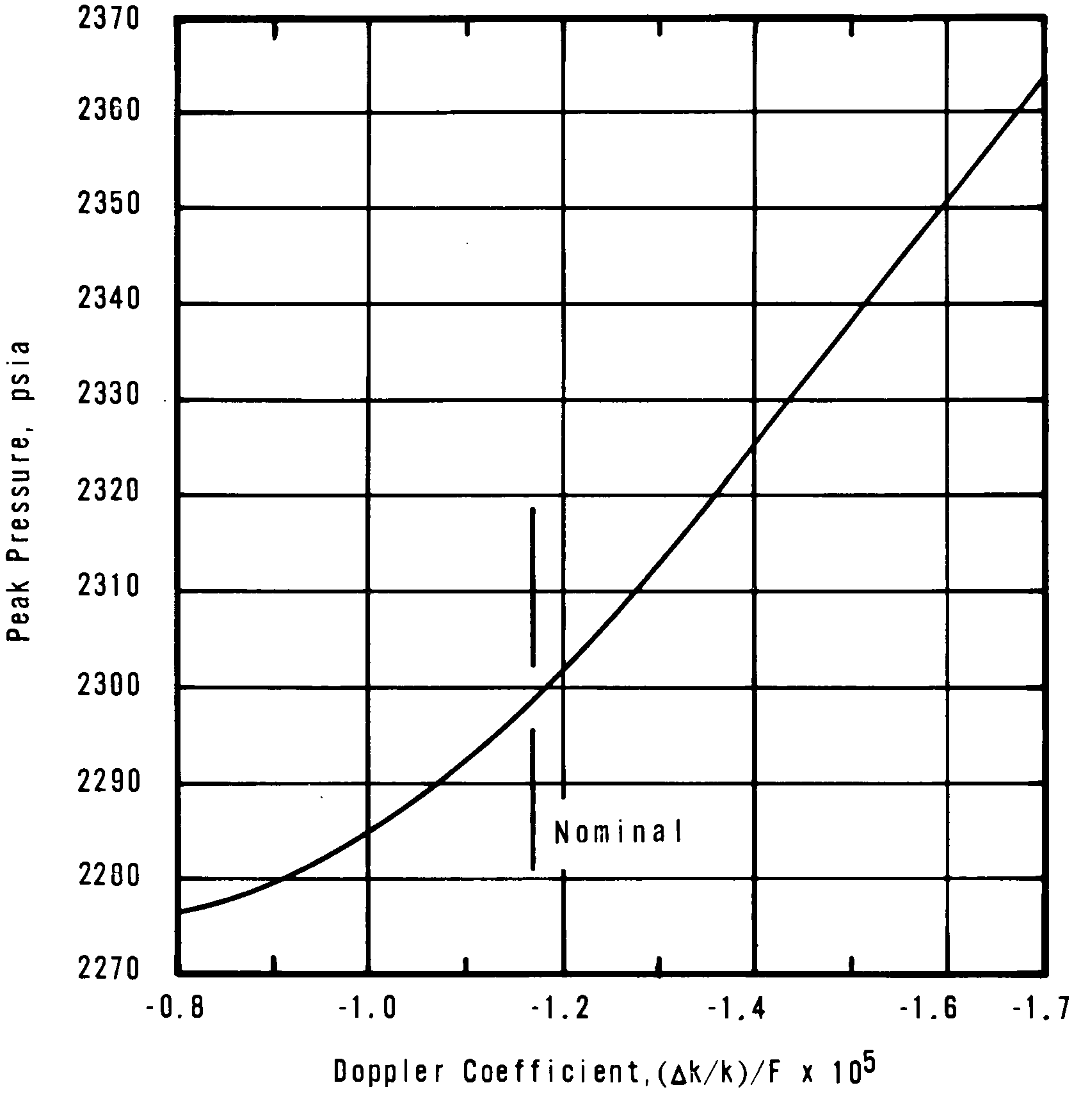
OCONEE NUCLEAR STATION
FIGURE 14B-10



PEAK PRESSURE VERSUS TRIP DELAY TIME FOR A ROD WITHDRAWAL ACCIDENT FROM RATED POWER USING A 1.5% $\Delta k/k$ ROD GROUP

(Reference Supplement 9 Revision for Oconee 3)

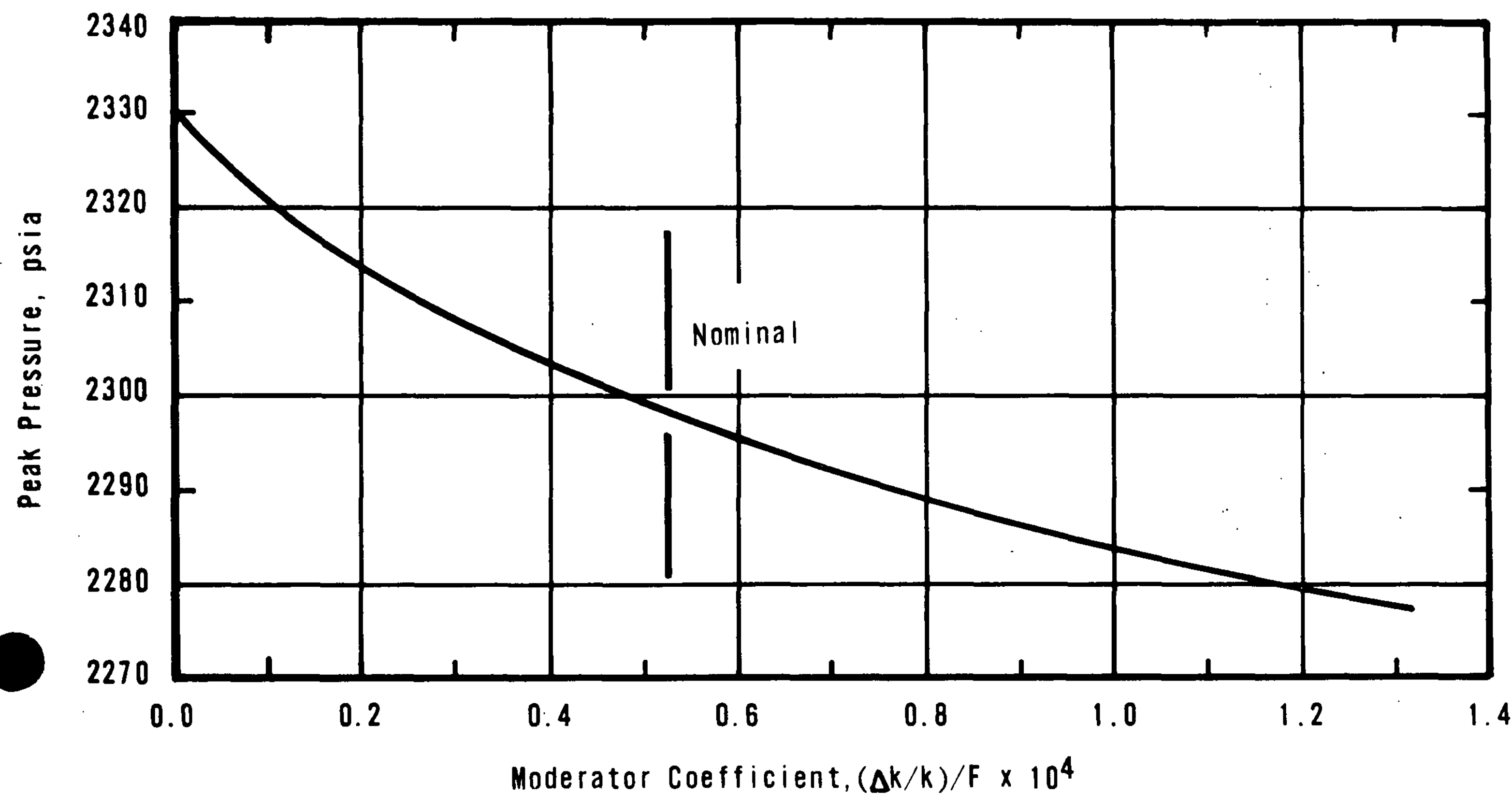
OCONEE NUCLEAR STATION
Figure 14-11
Rev. 16. 7/30/71



PEAK PRESSURE VERSUS DOPPLER COEFFICIENT FOR A ROD WITHDRAWAL ACCIDENT FROM RATED POWER USING A 1.5% Δk/k ROD GROUP

OCONEE NUCLEAR STATION

Figure 14 - 12



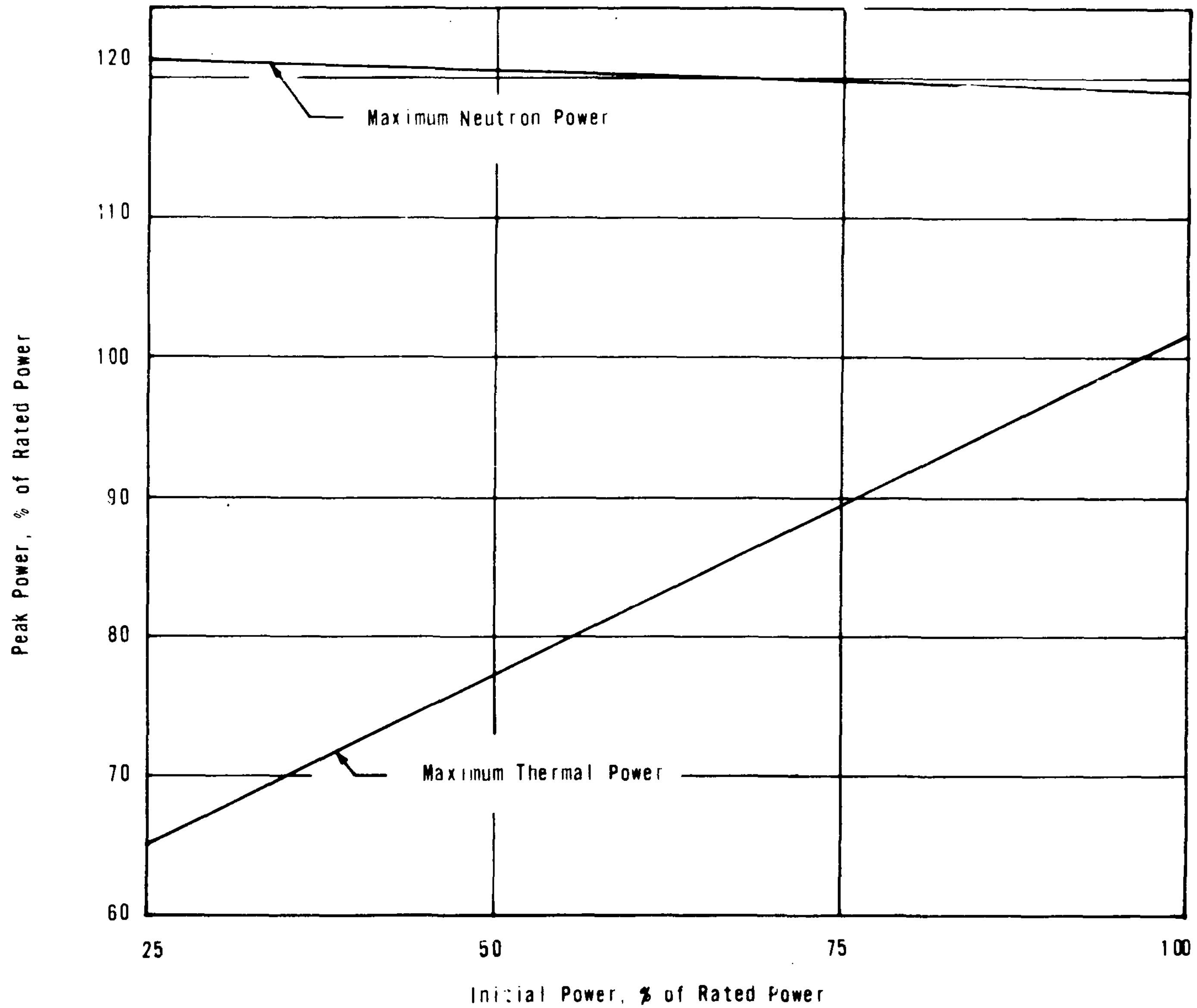
PEAK PRESSURE VERSUS MODERATOR COEFFICIENT FOR
A ROD WITHDRAWAL ACCIDENT FROM RATED POWER USING
A 1.5% Δk/k ROD GROUP

OCONEE NUCLEAR STATION

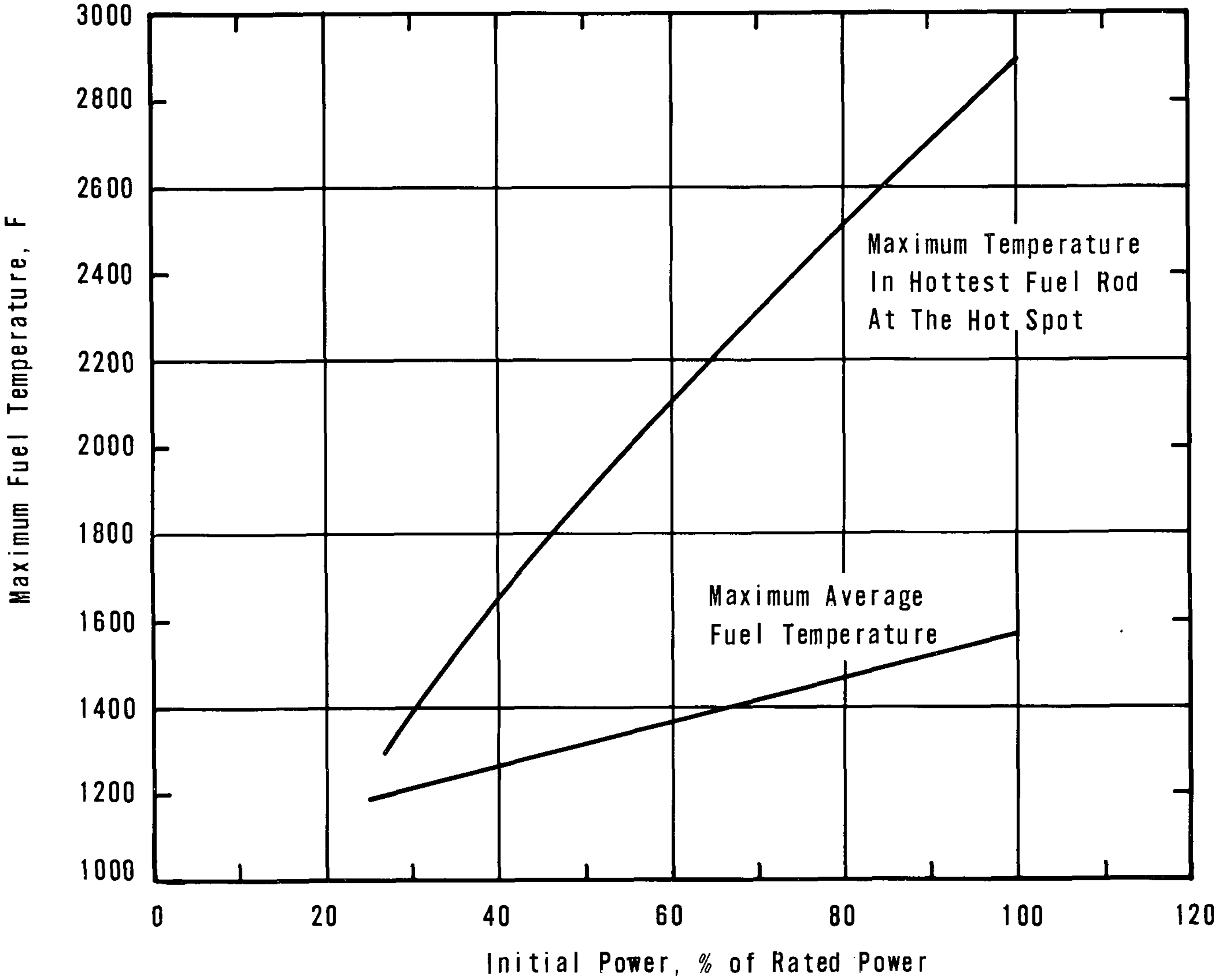
Figure 14 - 13

OCONEE NUCLEAR STATION

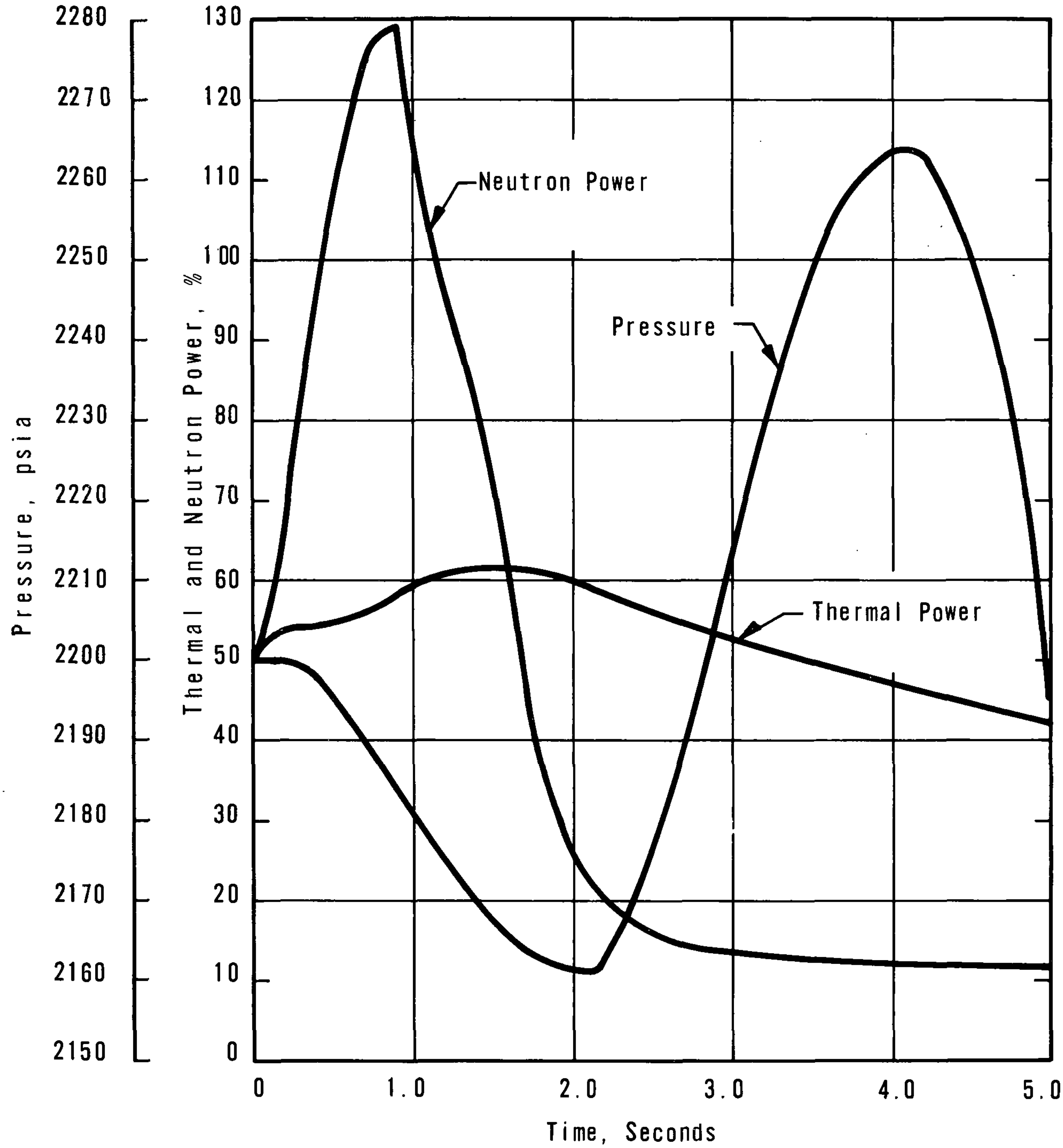
Figure 14-14



MAXIMUM NEUTRON AND THERMAL POWER
FOR AN ALL-ROD WITHDRAWAL ACCIDENT
FROM VARIOUS INITIAL POWER LEVELS



PEAK FUEL TEMPERATURE IN AVERAGE ROD AND HOT SPOT FOR AN ALL-ROD WITHDRAWAL ACCIDENT FROM VARIOUS INITIAL POWER LEVELS

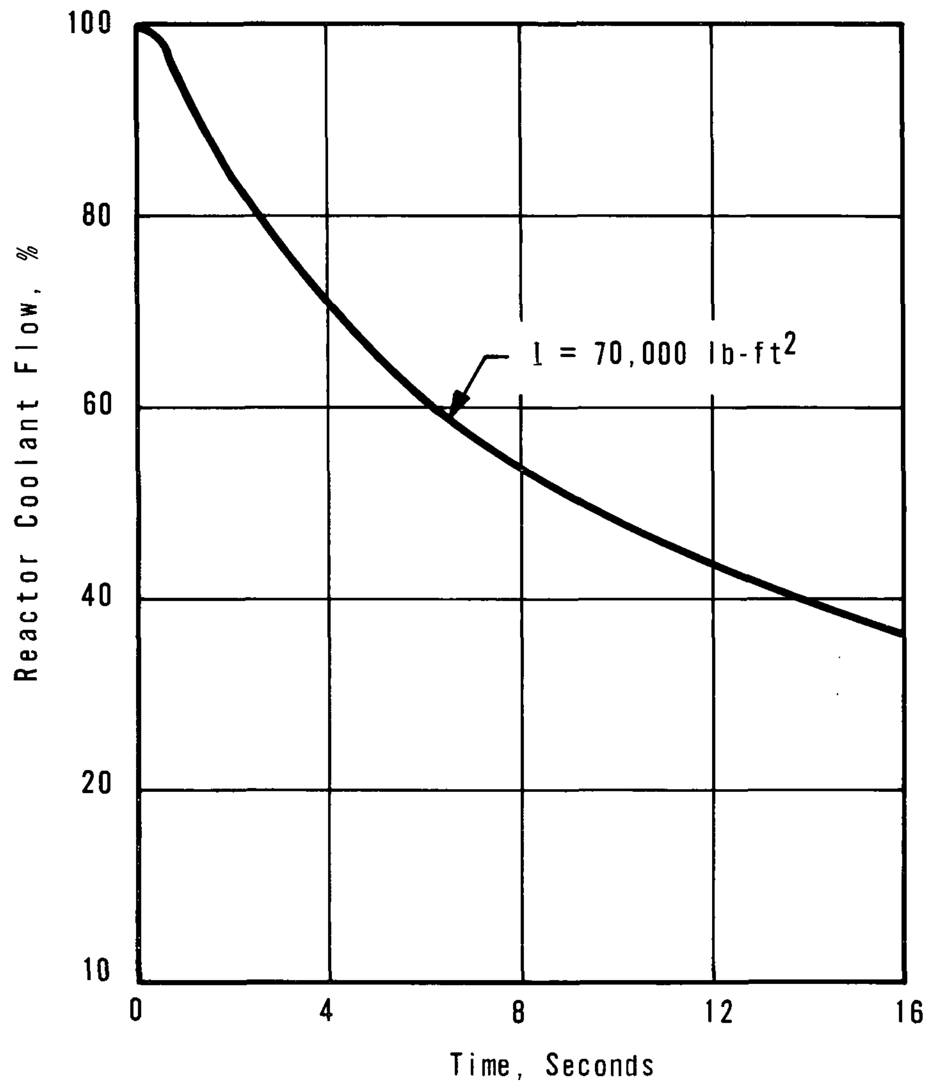


IDLE LOOP STARTUP FROM 50% POWER

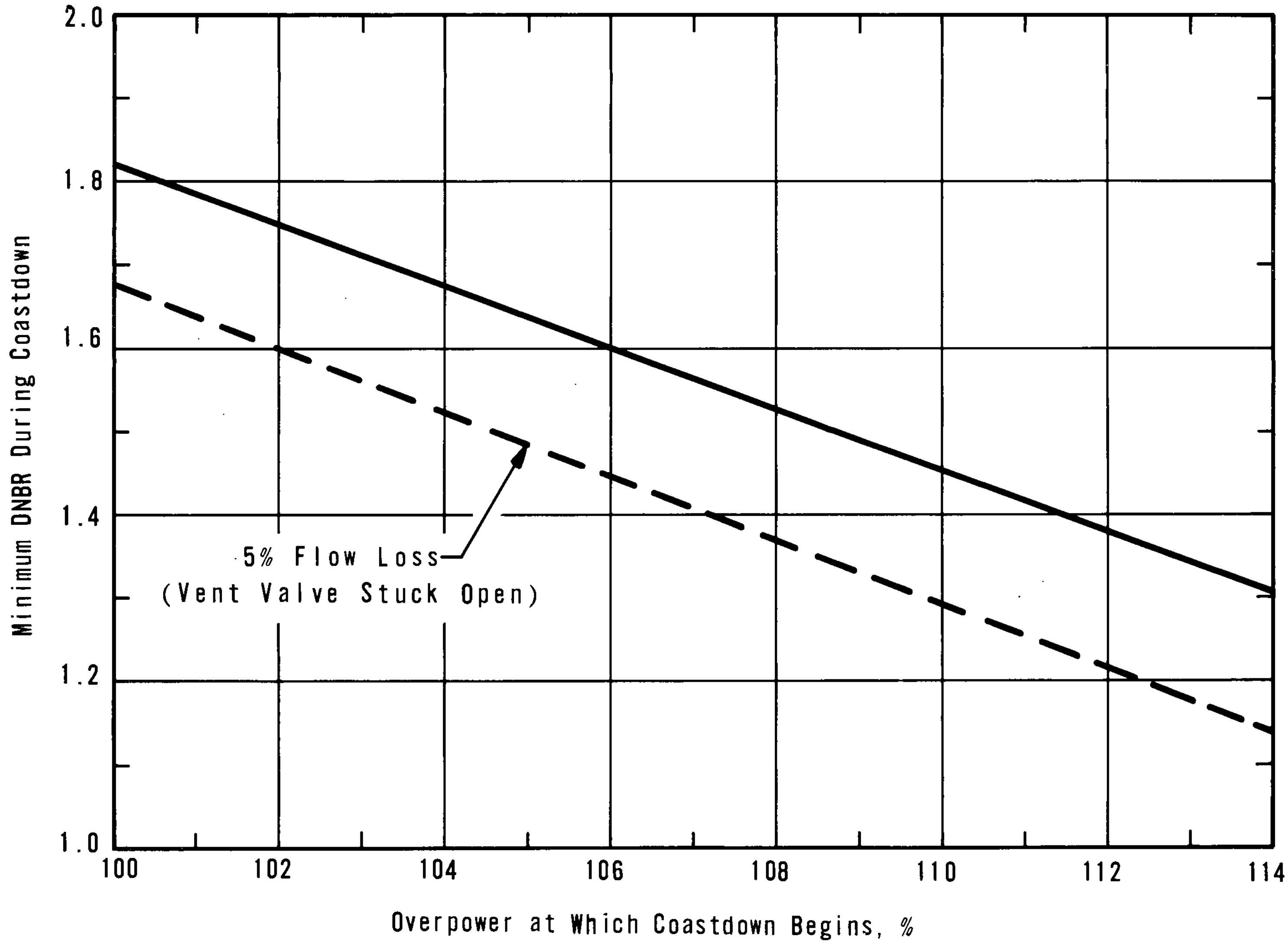
OCONEE NUCLEAR STATION

Figure 14 - 15A

(New) Rev. 7 7/9/70

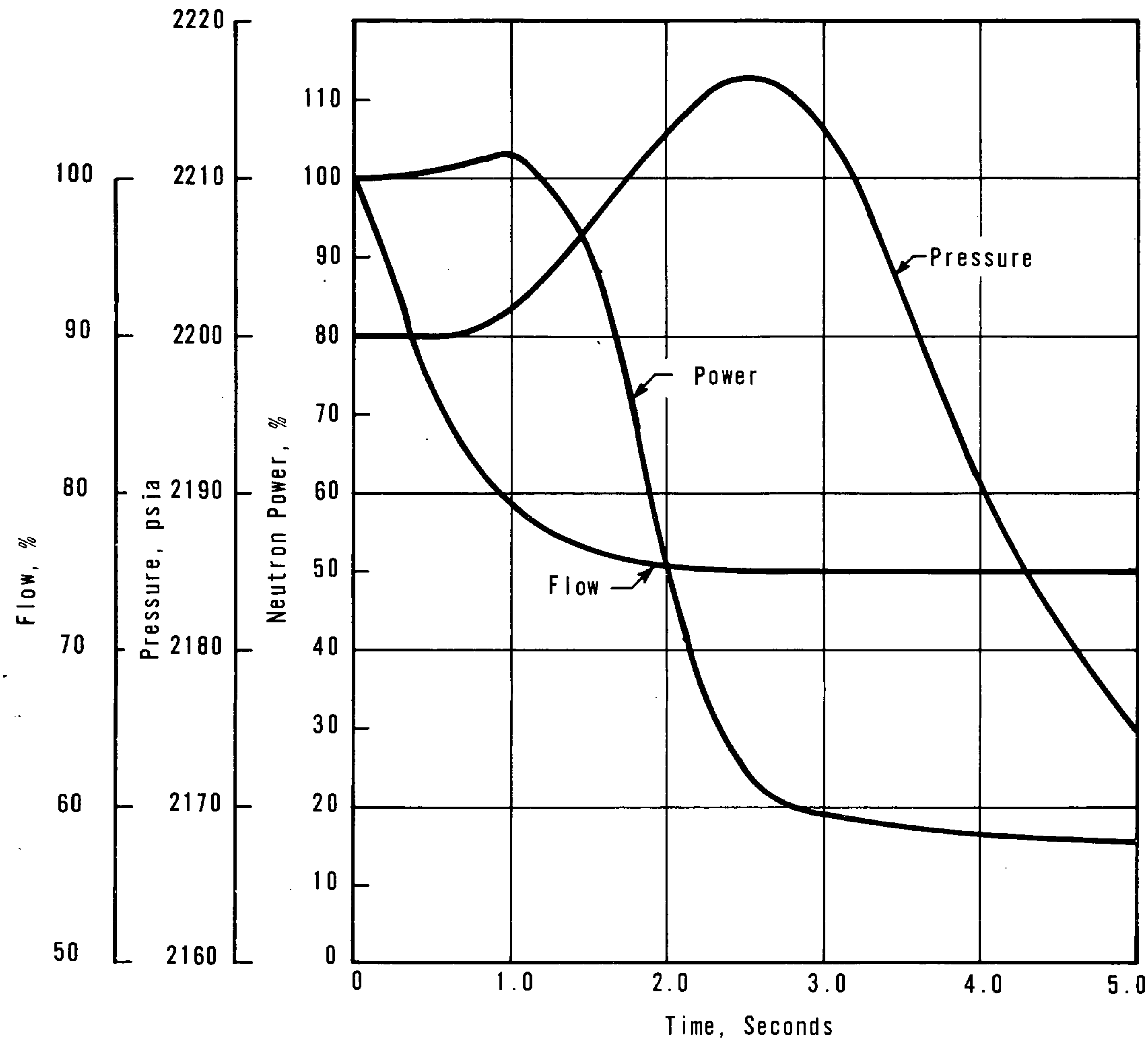


PER CENT REACTOR COOLANT FLOW
AS A FUNCTION OF TIME AFTER
LOSS OF PUMP POWER

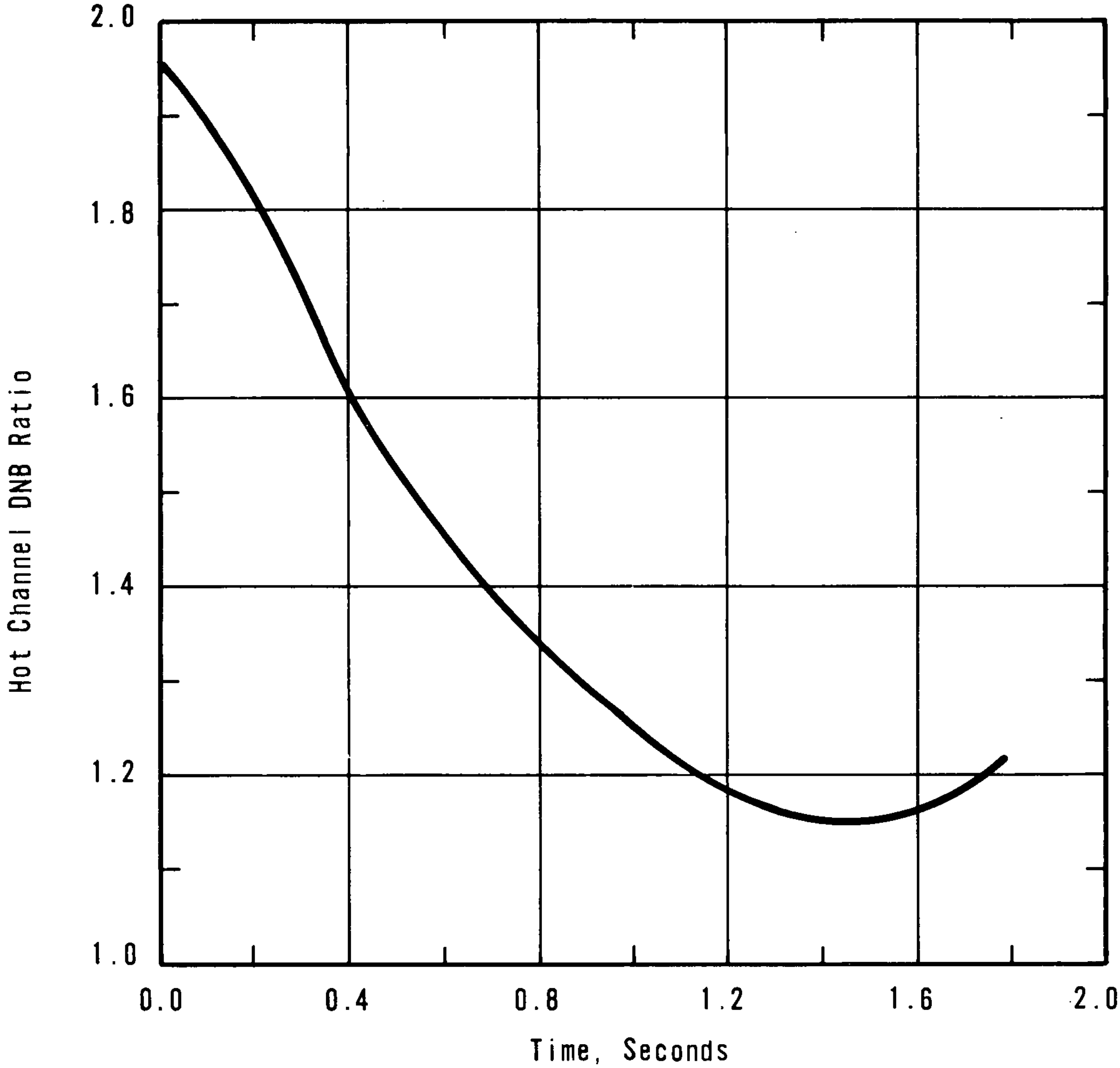


MINIMUM DNBR WHICH OCCURS DURING COASTDOWN
FROM VARIOUS INITIAL POWER LEVELS

OCONEE NUCLEAR STATION
Figure 14 - 17
Rev. 7 7/9/70



LOCKED ROTOR ACCIDENT TRIP ON POWER TO FLOW RATIO, BOL PARAMETERS



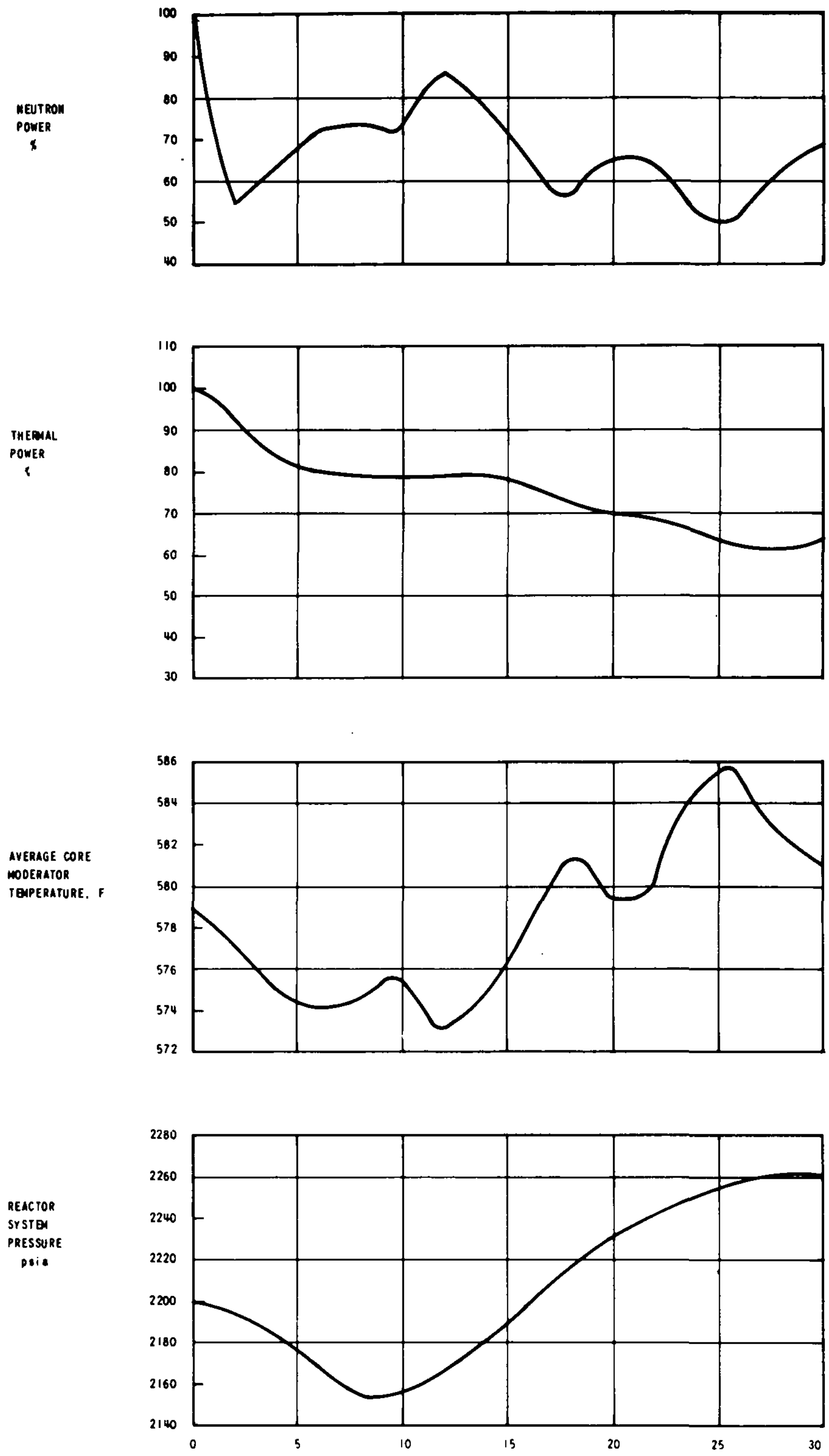
DNB RATIO VERSUS TIME FOR A LOCKED ROTOR
ACCIDENT FROM 102% OF RATED POWER

(Reference Supplement 9 Revisions for Oconee 3)

OCONEE NUCLEAR STATION

Figure 14 - 17B

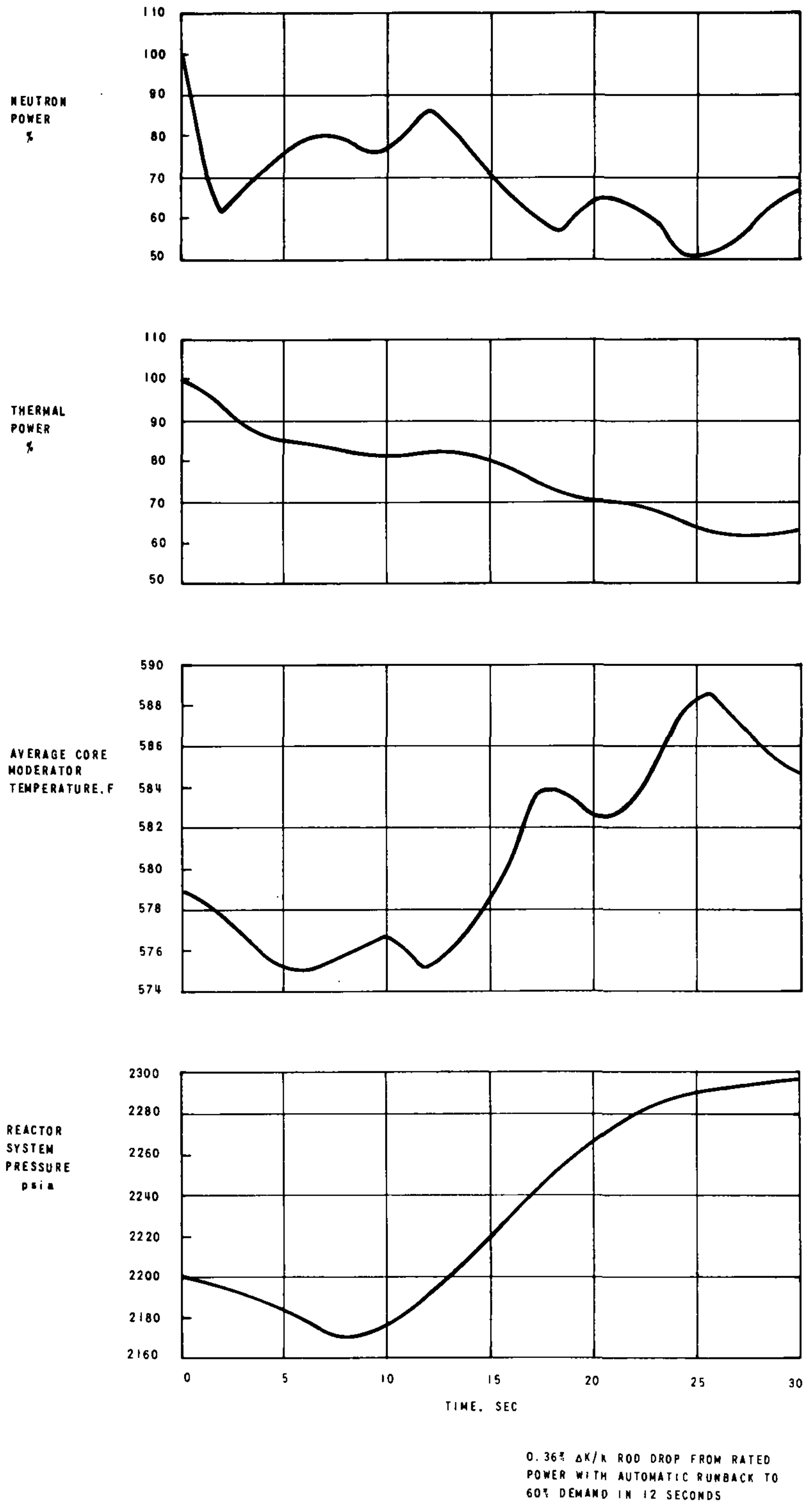
(New) Rev. 7 7/9/70
Rev. 16. 7/30/71



0.46% Δk/k ROD DROP FROM RATED POWER
WITH AUTOMATIC RUNBACK TO 60% DEMAND
IN 12 SECONDS

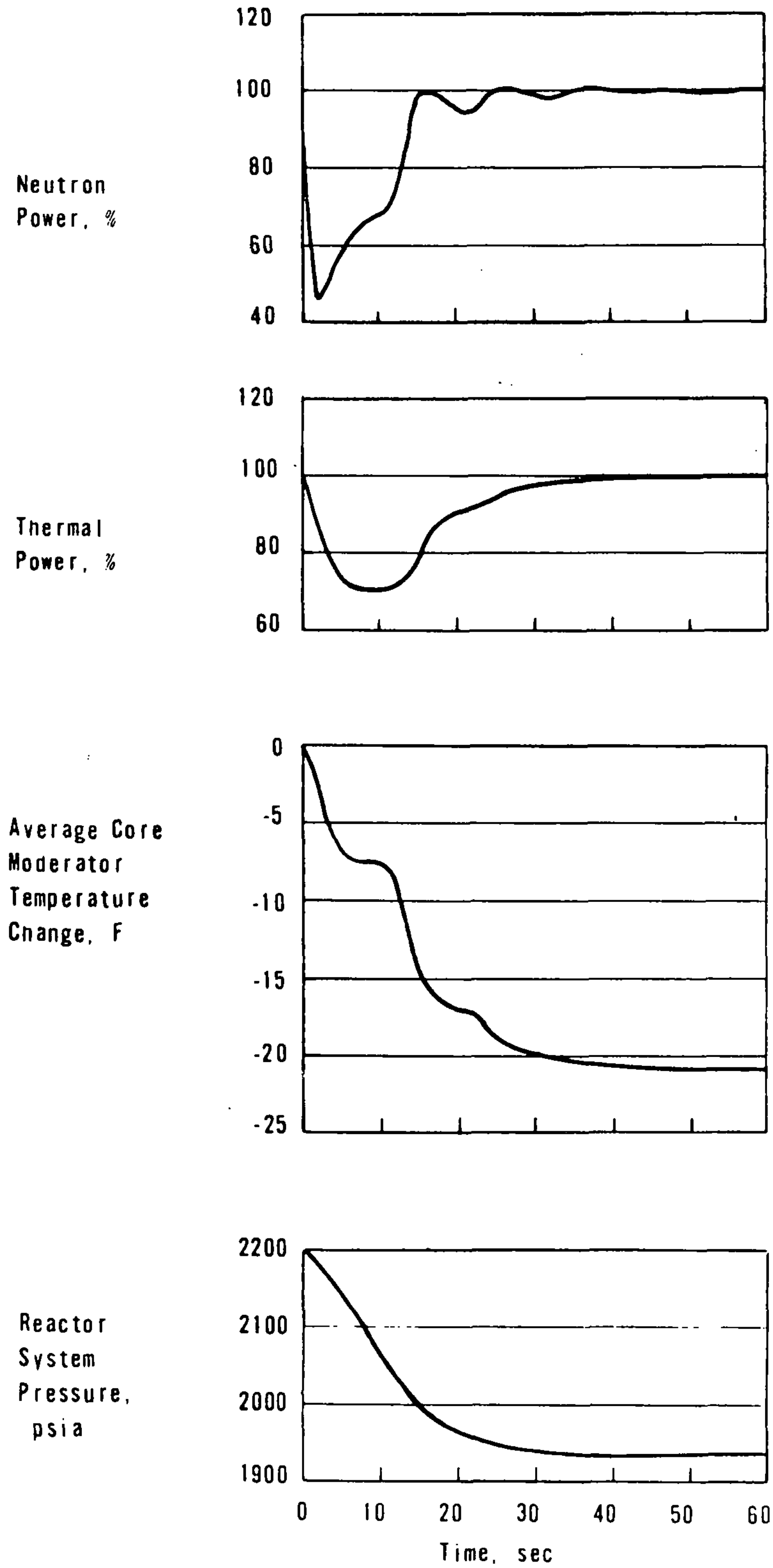
OCONEE NUCLEAR STATION

Figure 14-18



OCONEE NUCLEAR STATION

Figure 14-19



0.65% Ak/k ROD DROP FROM
RATED POWER AT EOL
CONDITIONS (2772 MWT)

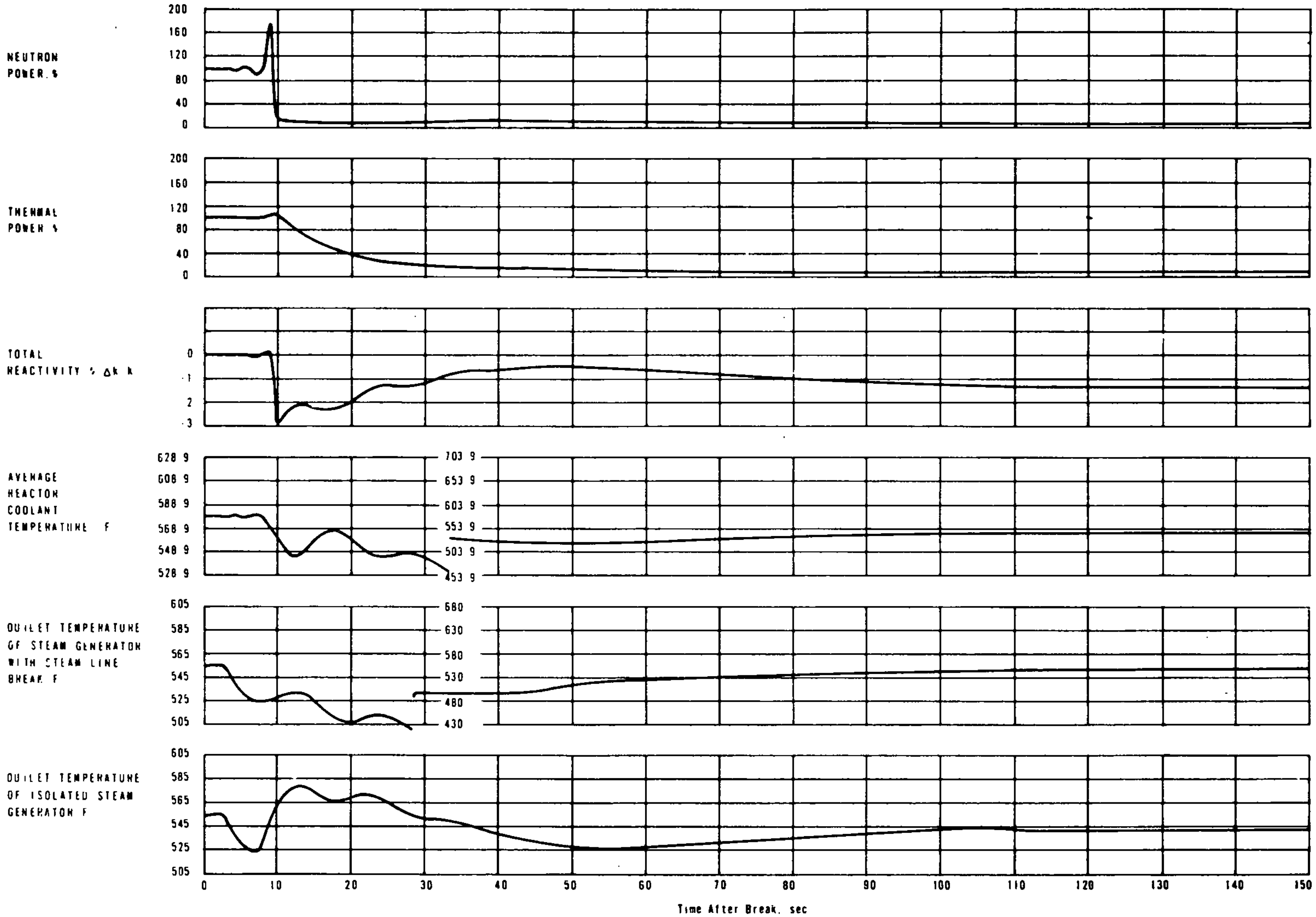
OCONEE NUCLEAR STATION

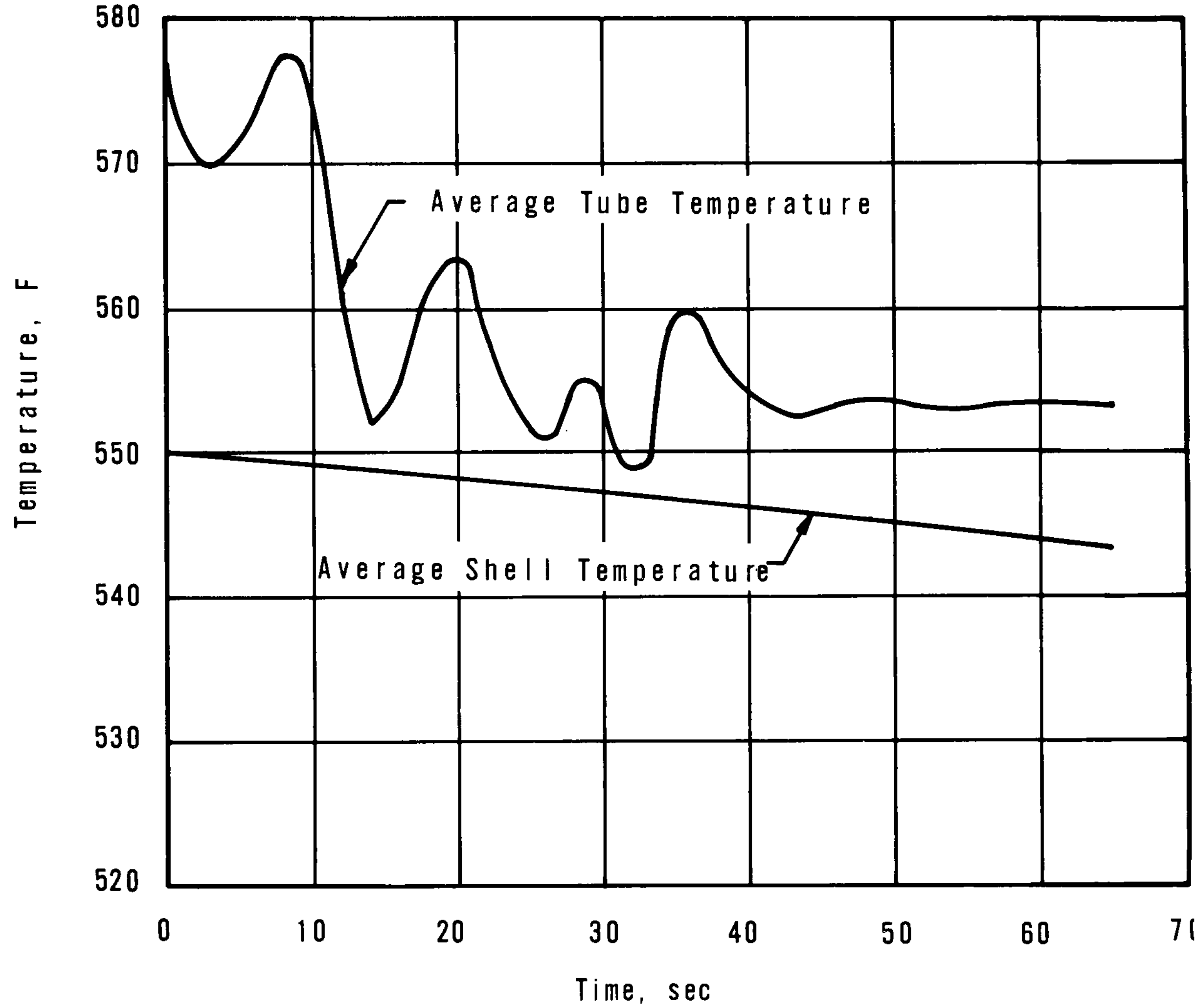
Figure 14-19a

OCONEE NUCLEAR STATION

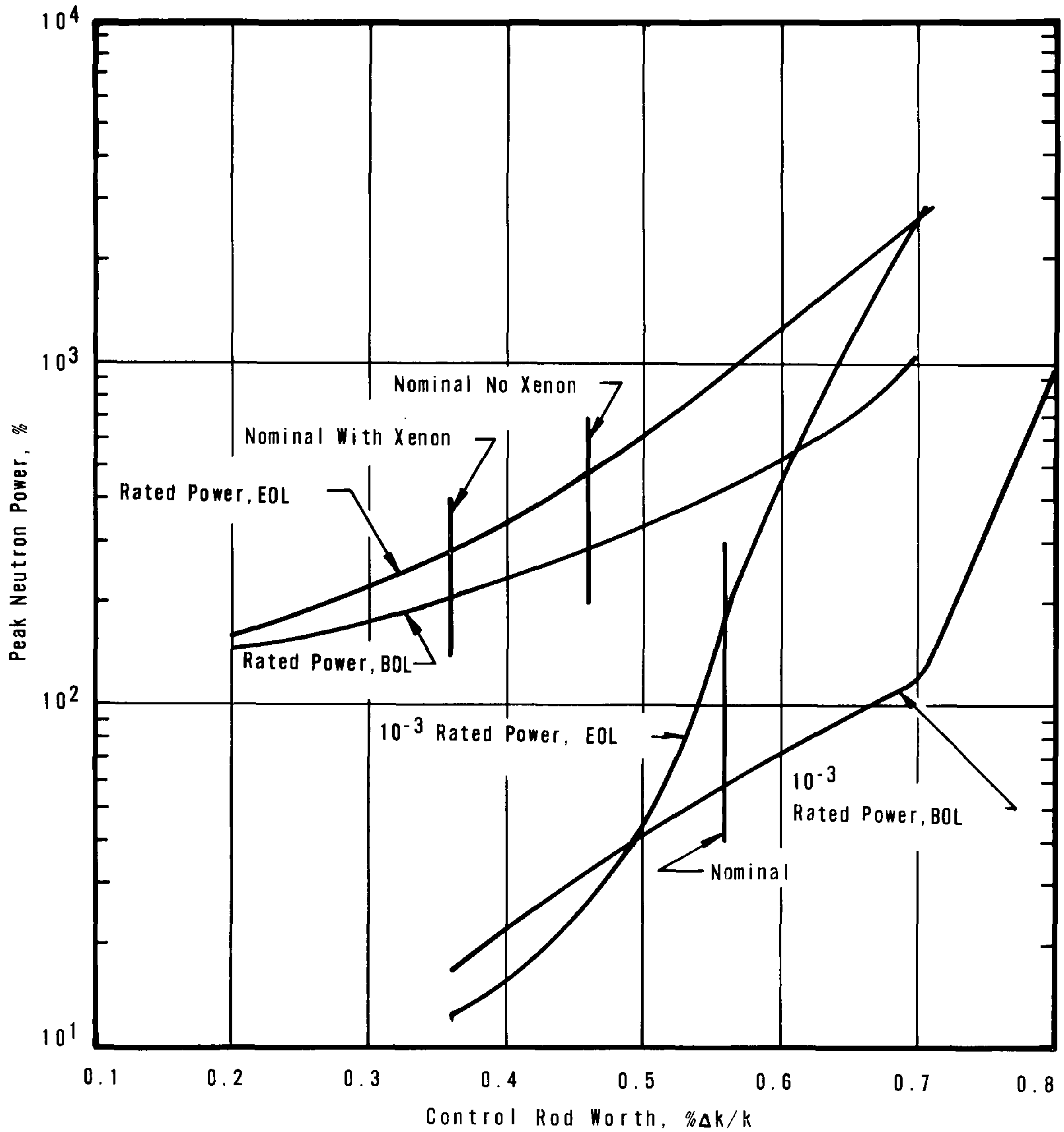
Figure 14 - 20

DOUBLE ENDED RUPTURE OF 34 IN. STEAM LINE BETWEEN STEAM GENERATOR AND STEAM STOP VALVE (WITH FEEDWATER ISOLATION)

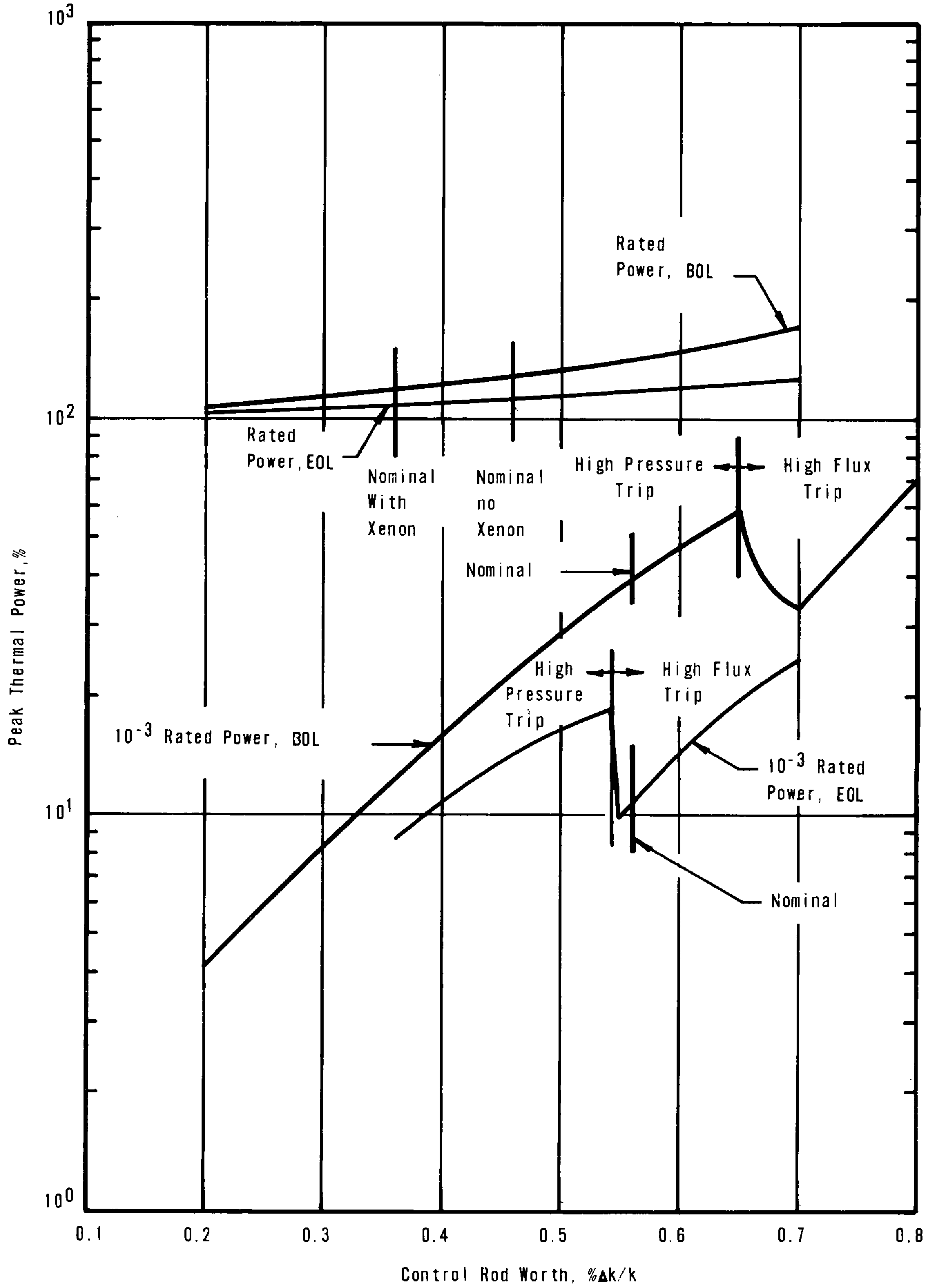




AVERAGE STEAM GENERATOR SHELL AND TUBE TEMPERATURE VERSUS TIME AFTER ASSUMED STEAM LINE BREAK



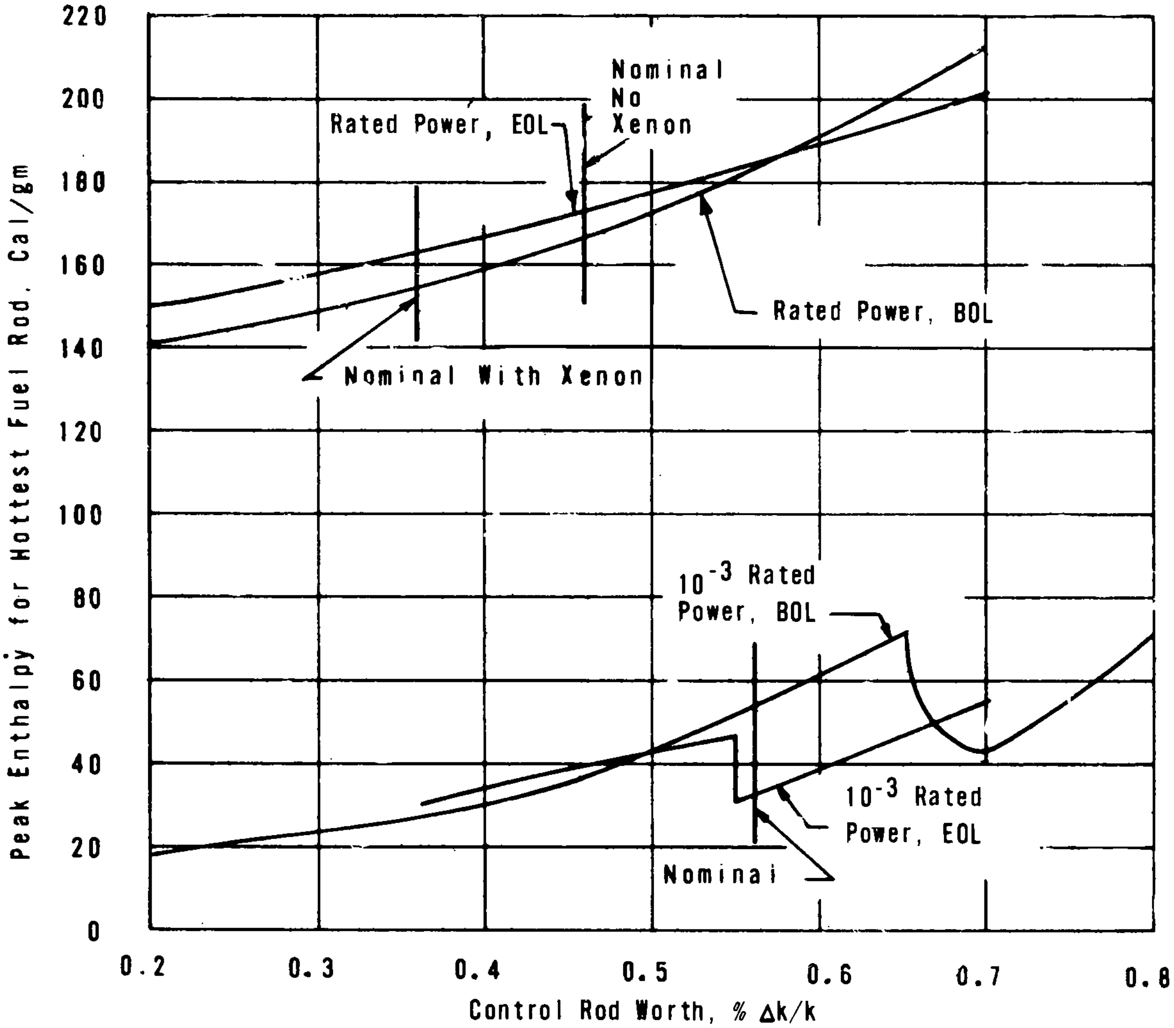
PEAK NEUTRON POWER VARIATION
WITH EJECTED CONTROL ROD WORTH



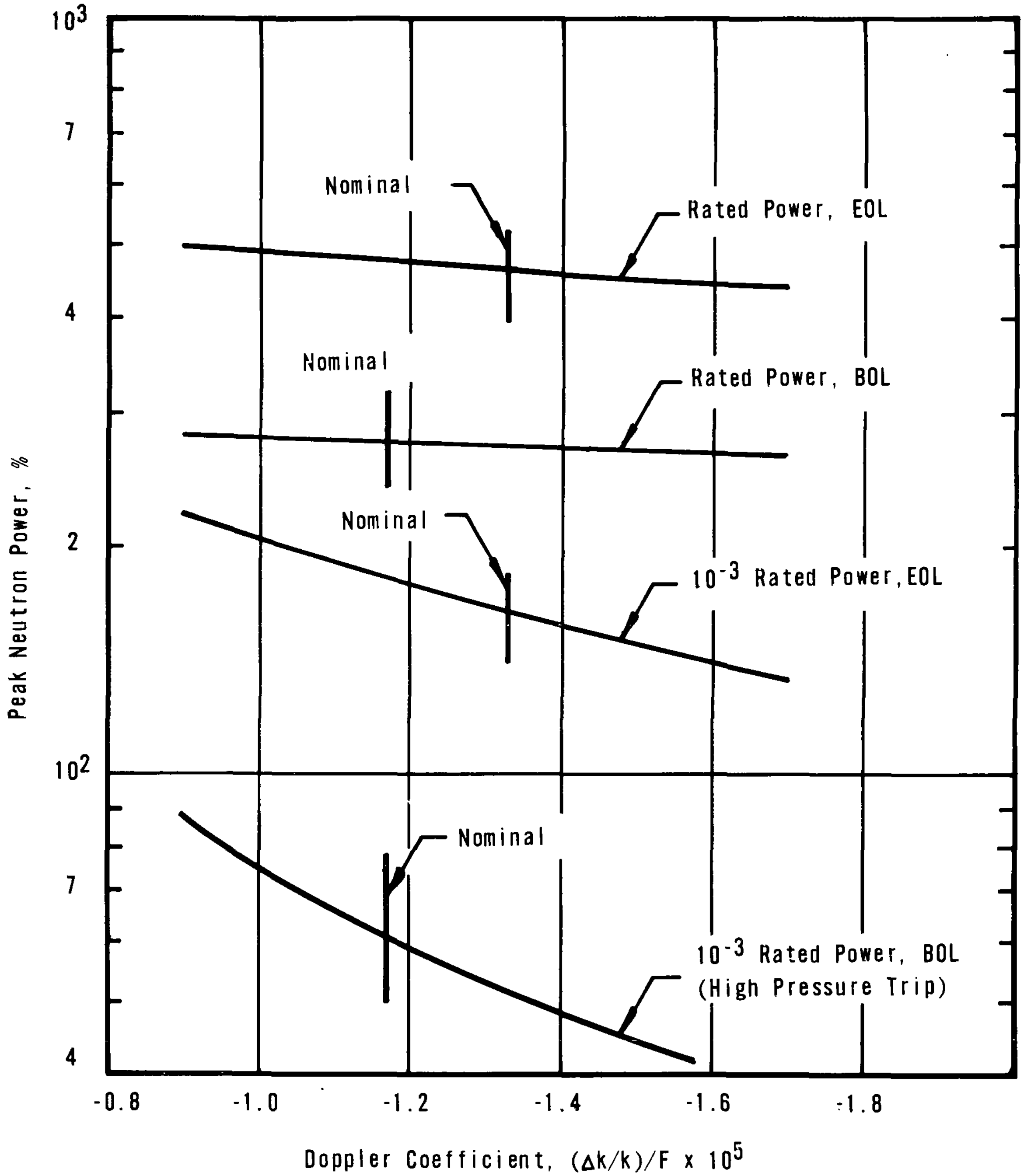
PEAK THERMAL POWER AS A FUNCTION
OF EJECTED CONTROL ROD WORTH

OCONEE NUCLEAR STATION

Figure 14 - 23



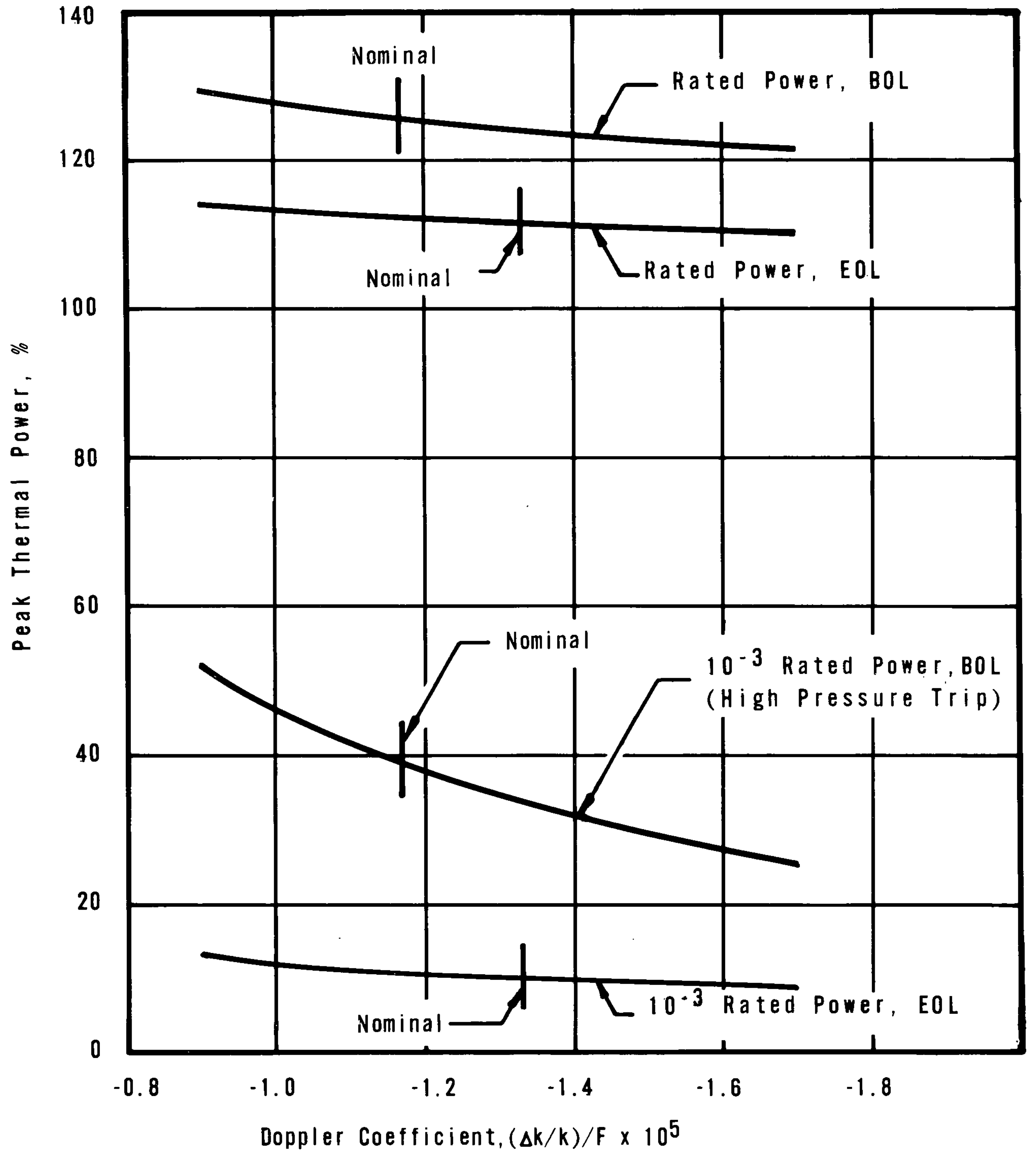
PEAK ENTHALPY OF HOTTEST FUEL ROD
VERSUS EJECTED CONTROL ROD WORTH



EFFECT ON PEAK NEUTRON POWER OF VARYING THE DOPPLER COEFFICIENT FOR AN EJECTED ROD WORTH OF 0.56% $\Delta k/k$ AT 10^{-3} RATED POWER AND 0.46% $\Delta k/k$ AT RATED POWER

OCONEE NUCLEAR STATION

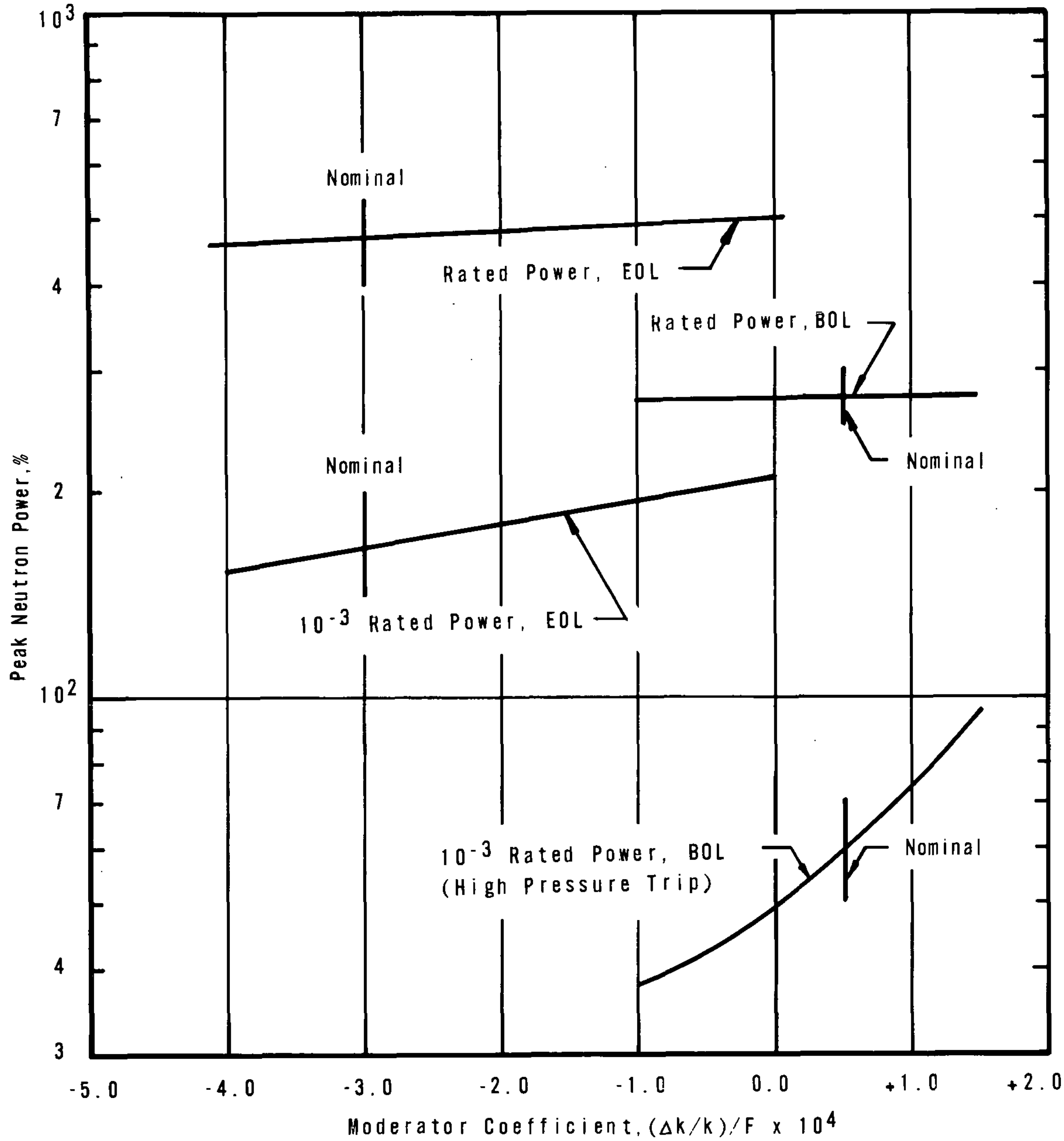
Figure 14 - 25



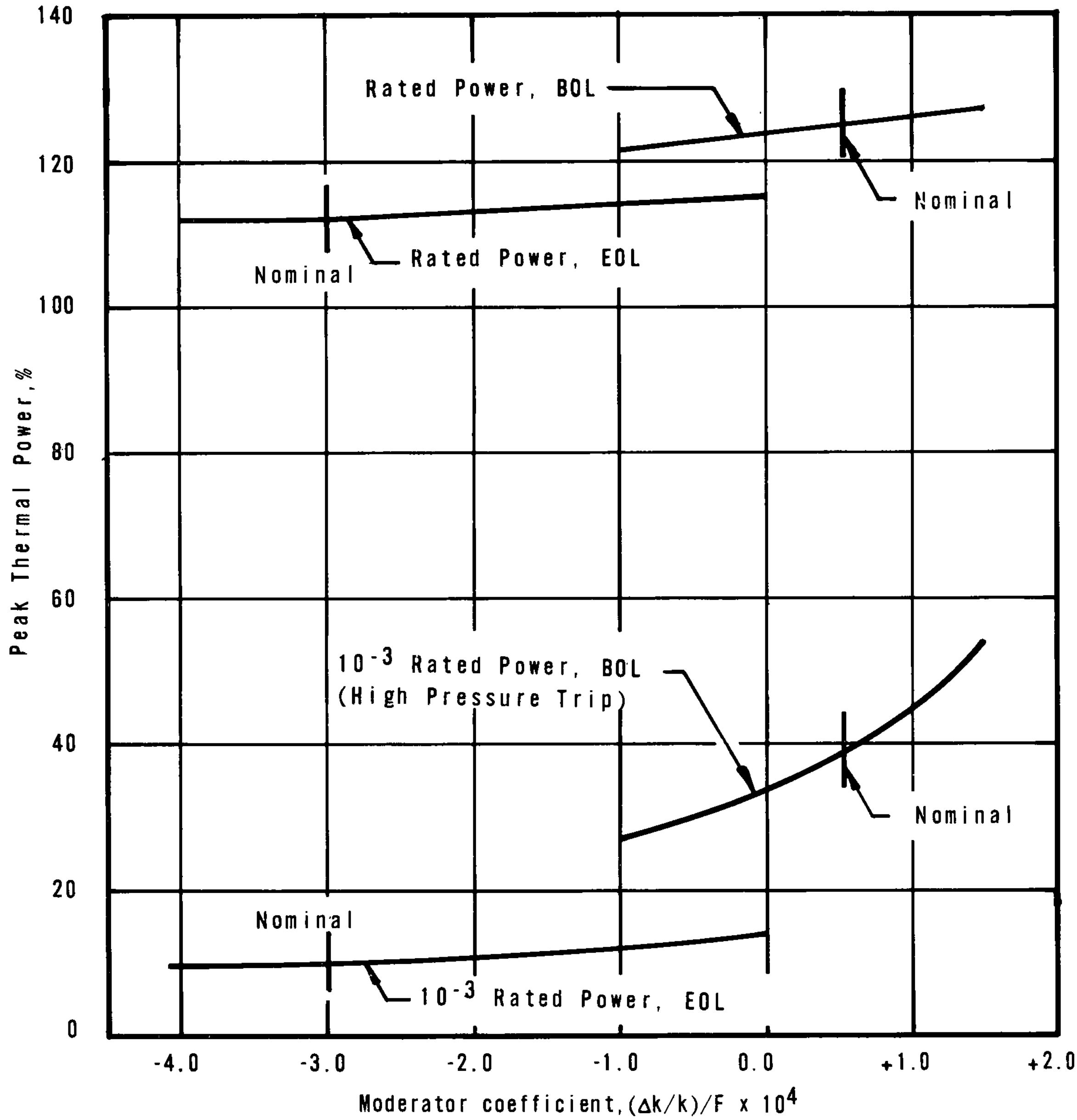
EFFECT ON PEAK THERMAL POWER OF VARYING THE DOPPLER COEFFICIENT FOR AN EJECTED ROD WORTH OF 0.56% $\Delta k/k$ AT 10^{-3} RATED POWER AND 0.46% $\Delta k/k$ AT RATED POWER.

OCONEE NUCLEAR STATION

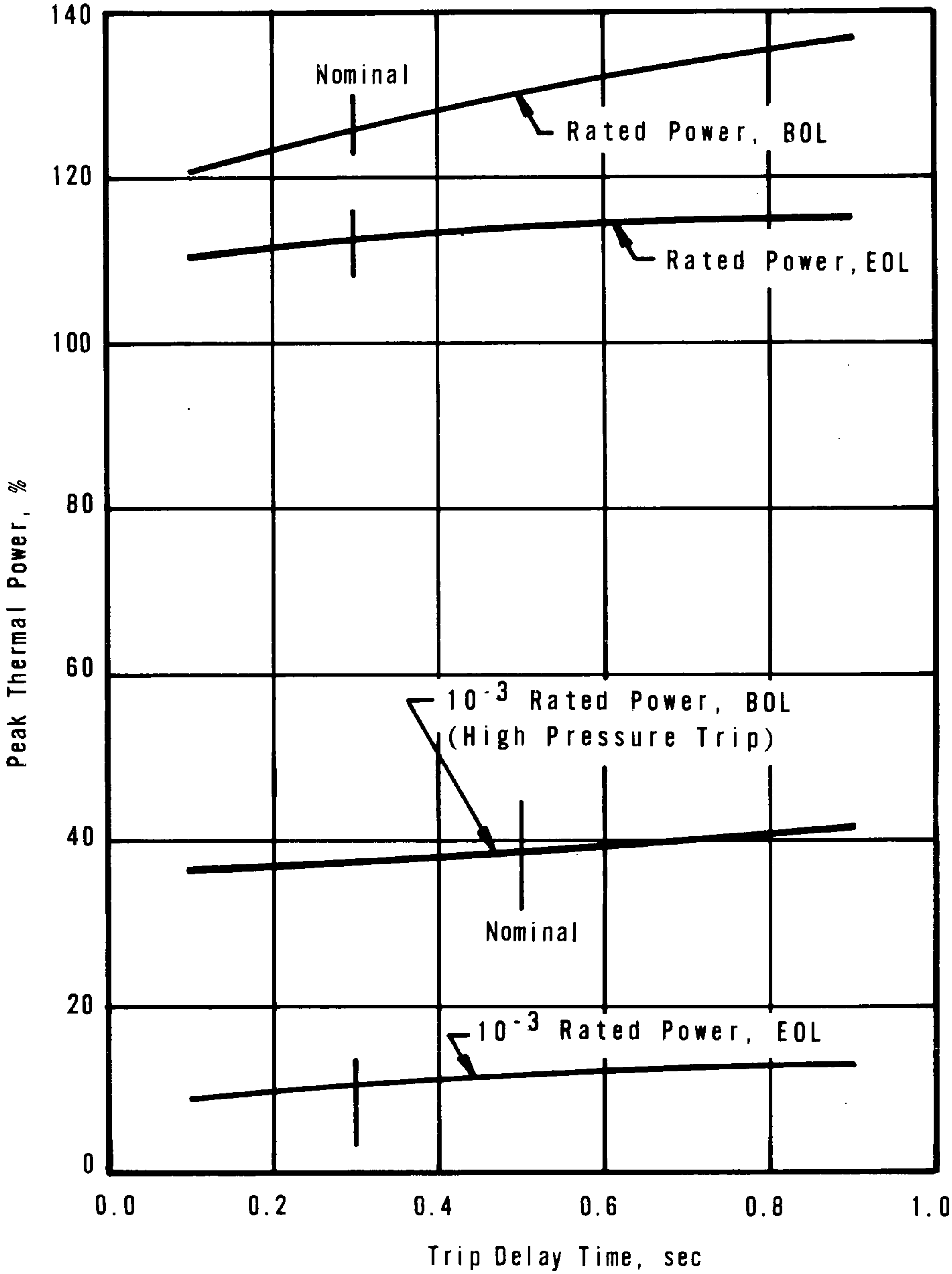
Figure 14 - 26



EFFECT ON PEAK NEUTRON POWER OF VARYING THE MODERATOR COEFFICIENT FOR AN EJECTED ROD WORTH OF 0.56% $\Delta k/k$ AT 10^{-3} RATED POWER AND 0.46% $\Delta k/k$ AT RATED POWER.

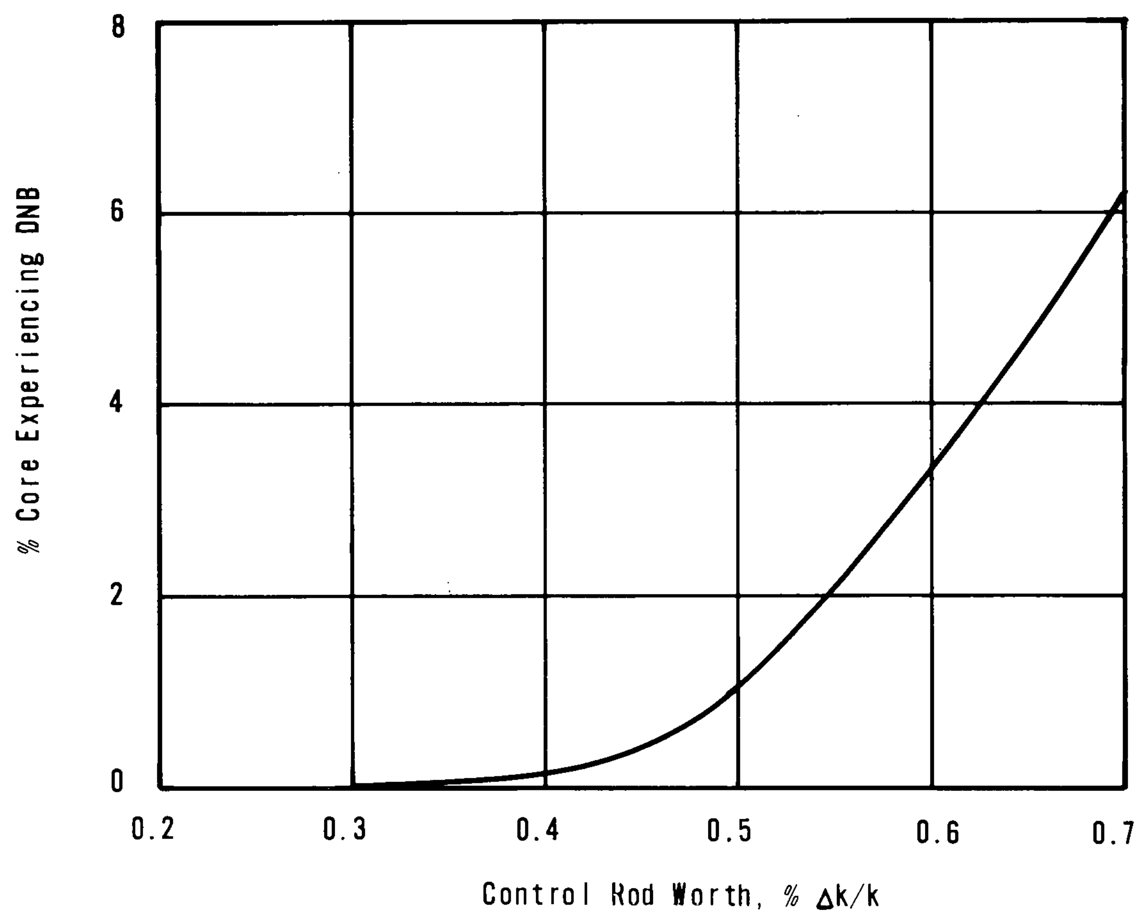


EFFECT ON PEAK THERMAL POWER OF VARYING THE MODERATOR COEFFICIENT FOR AN EJECTED ROD WORTH OF 0.56% $\Delta k/k$ AT 10^{-3} RATED POWER AND 0.46% $\Delta k/k$ AT RATED POWER



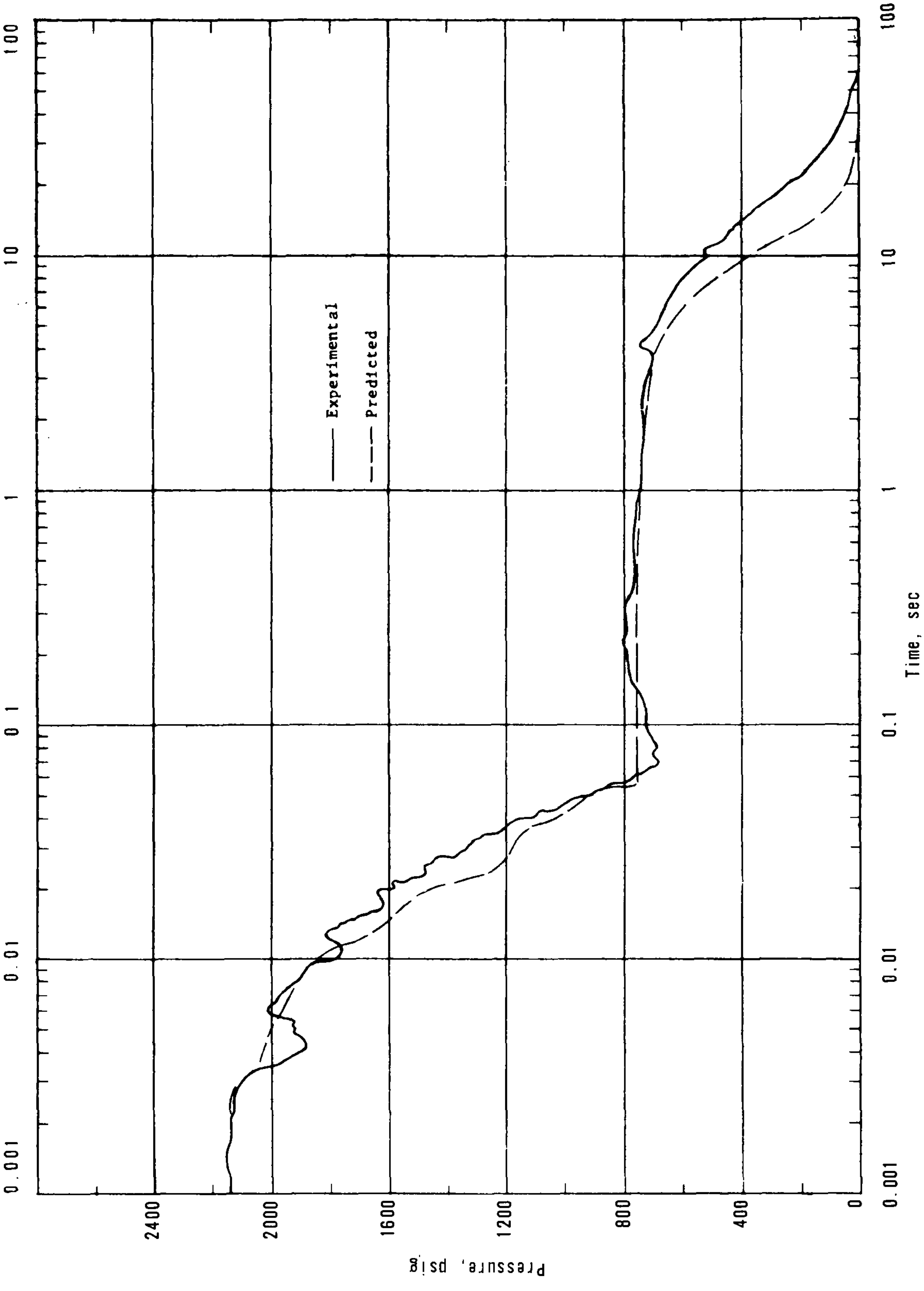
EFFECT ON PEAK THERMAL POWER OF VARYING THE TRIP DELAY TIME FOR AN EJECTED ROD WORTH OF 0.56% $\Delta k/k$ AT 10^{-3} RATED POWER AND 0.46% $\Delta k/k$ AT RATED POWER

(Reference Supplement 9 Revisions for Oconee 3)



PER CENT CORE EXPERIENCING DNB AS A FUNCTION OF EJECTED CONTROL ROD WORTH AT RATED POWER, BOL.

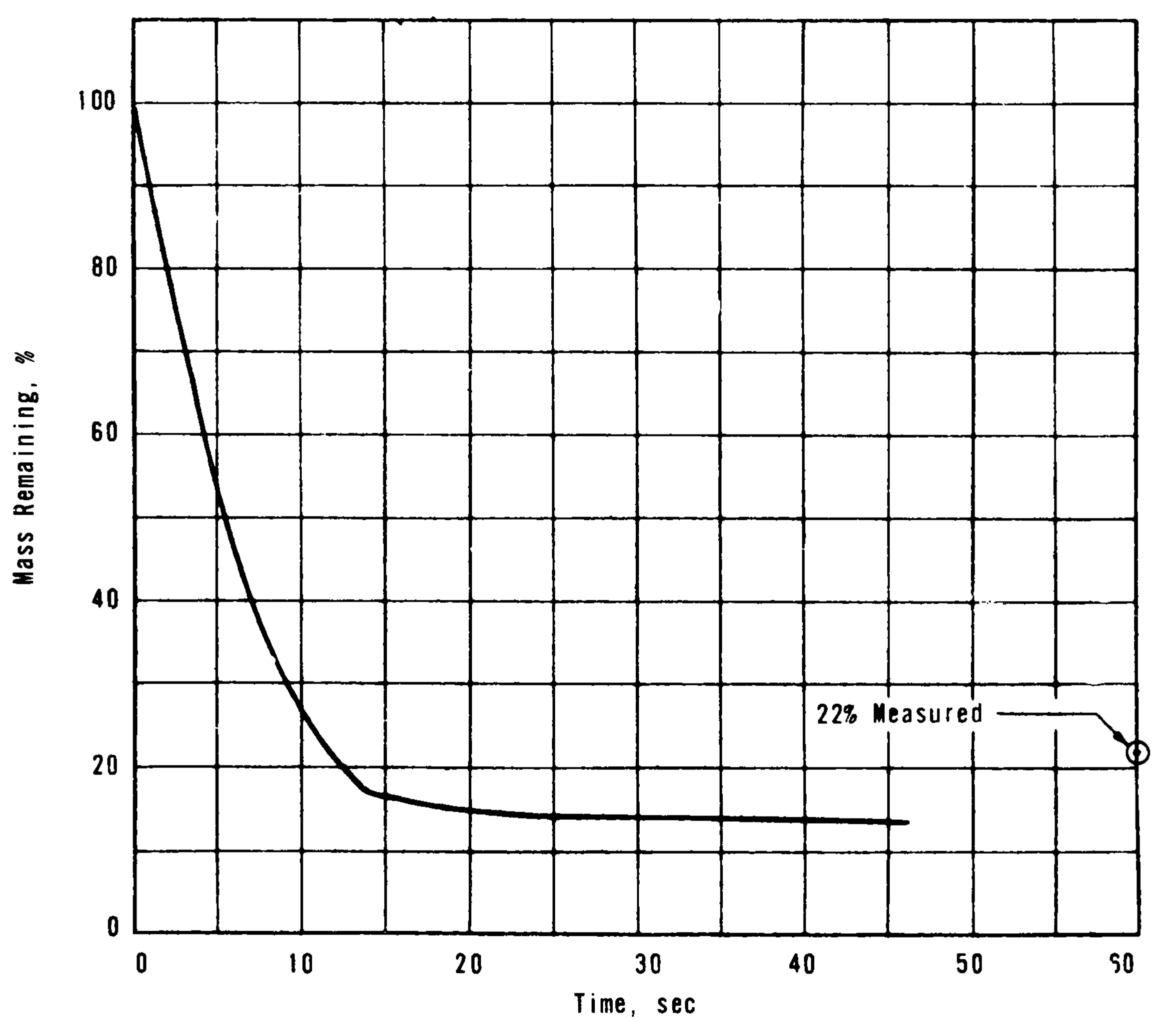
(Reference Supplement 9 Revisions for Oconee 3)



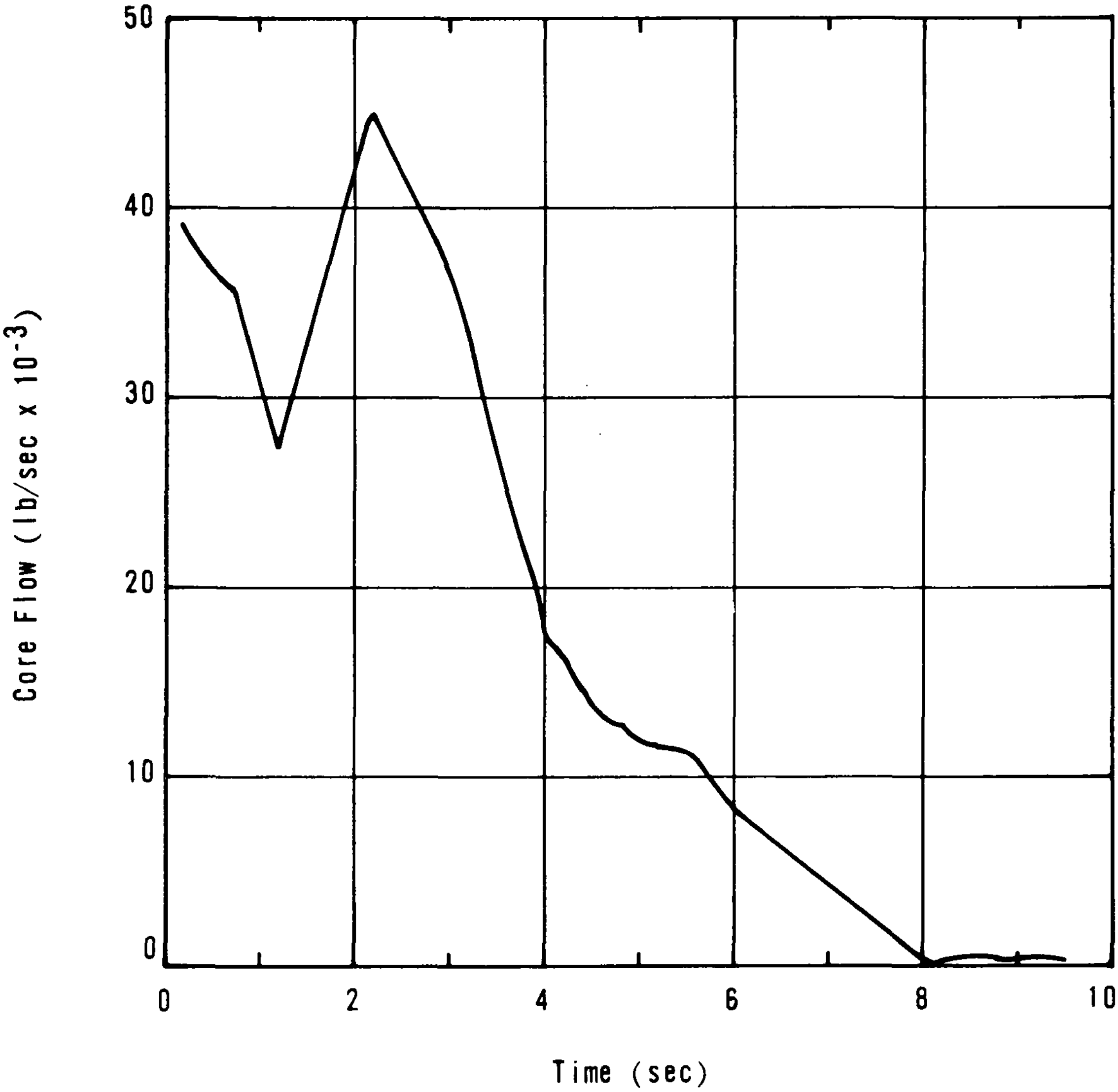
LOFT SEMISCALE BLOWDOWN TEST NO. 546-VESSEL
PRESSURE VERSUS TIME

OCONEE NUCLEAR STATION

Figure 14 - 31



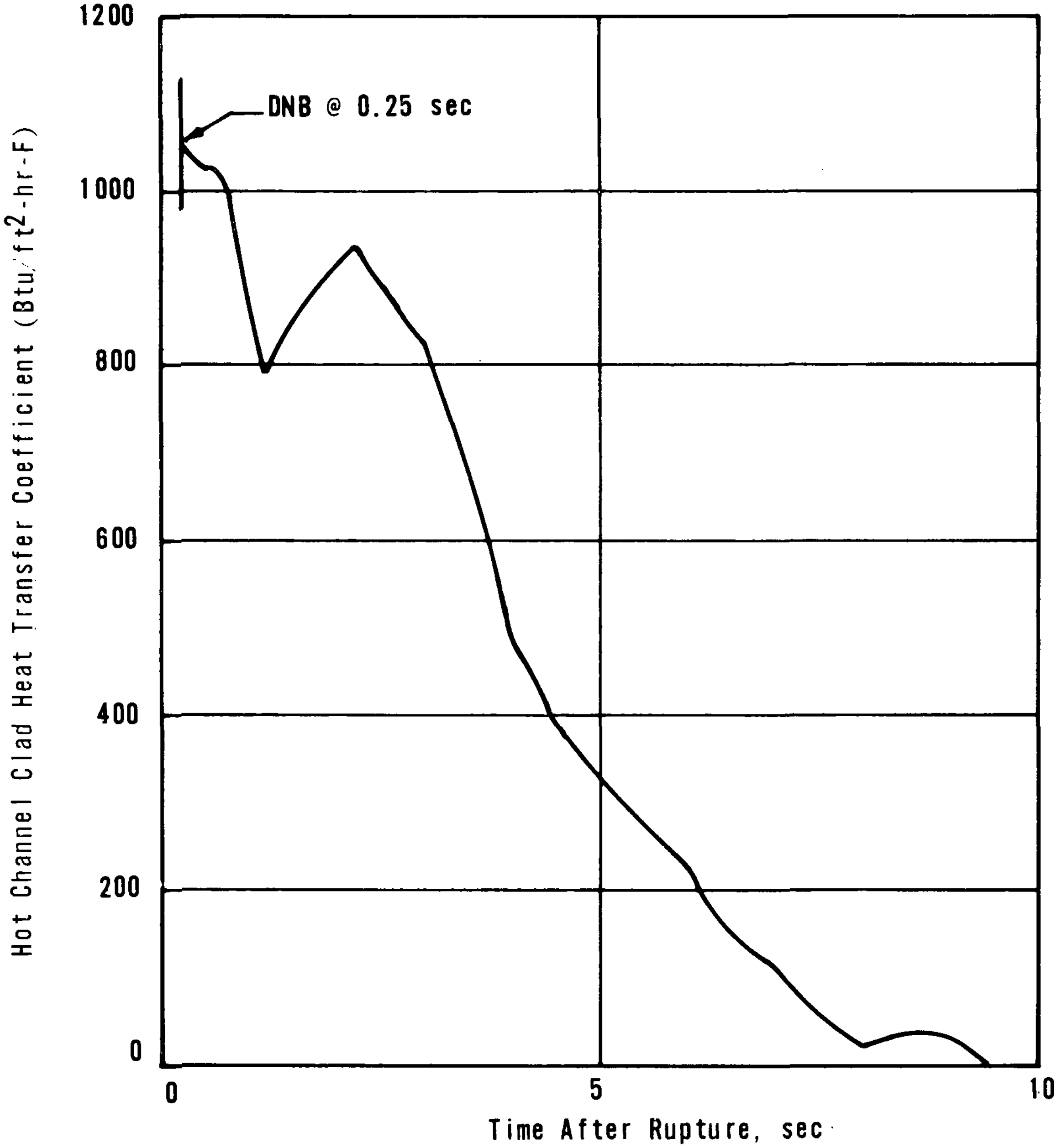
PREDICTED PER CENT MASS REMAINING
VERSUS TIME – LOFT TEST NO. 546



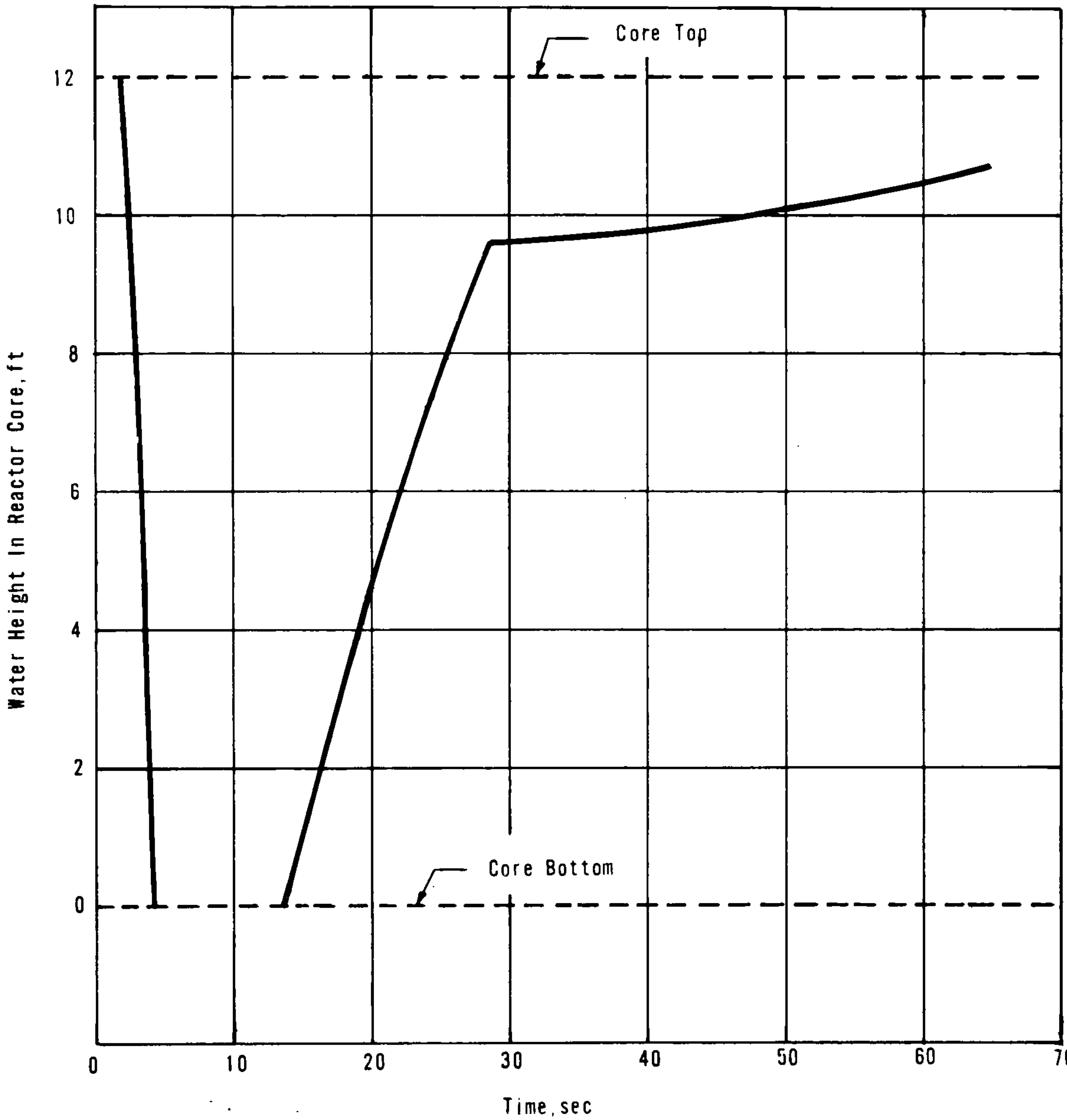
CORE FLOW VERSUS TIME FOR A 36-IN. ID, DOUBLE-ENDED, HOT LEG PIPE RUPTURE

OCONEE NUCLEAR STATION

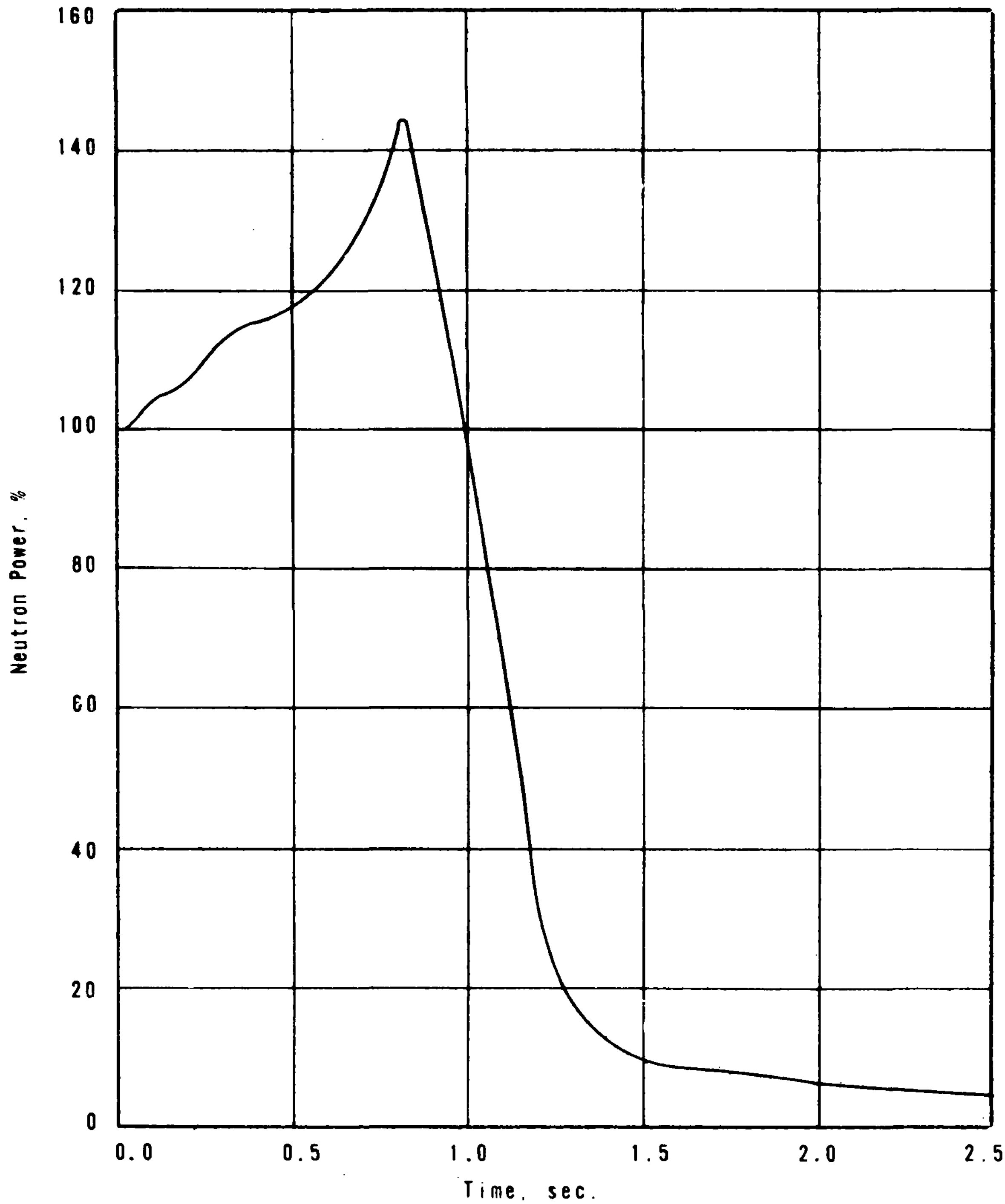
Figure 14 - 33



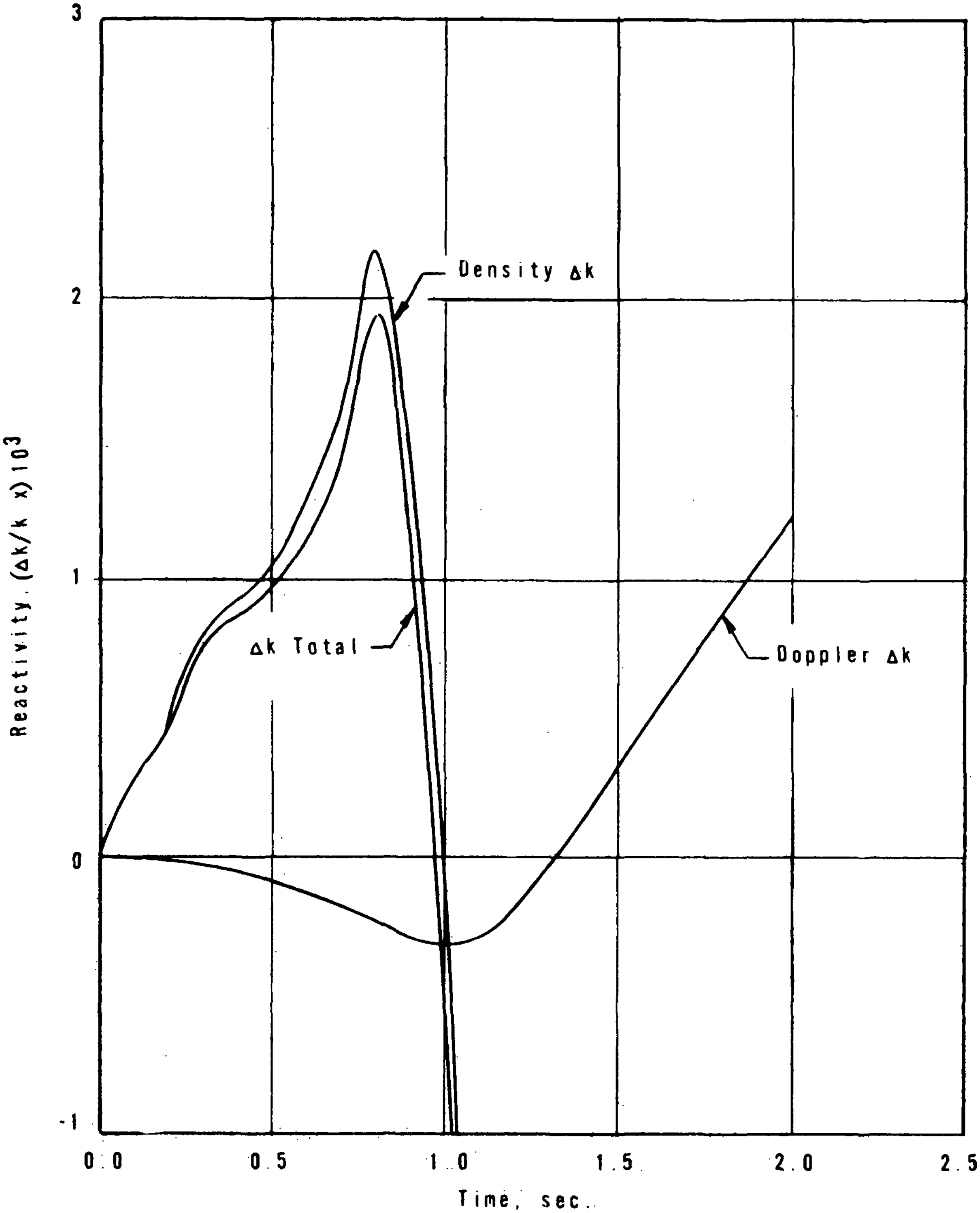
HOT CHANNEL CLAD SURFACE HEAT TRANSFER COEFFICIENT AFTER DNB VERSUS TIME FOR A 36-IN. ID, DOUBLE-ENDED, HOT LEG PIPE RUPTURE.



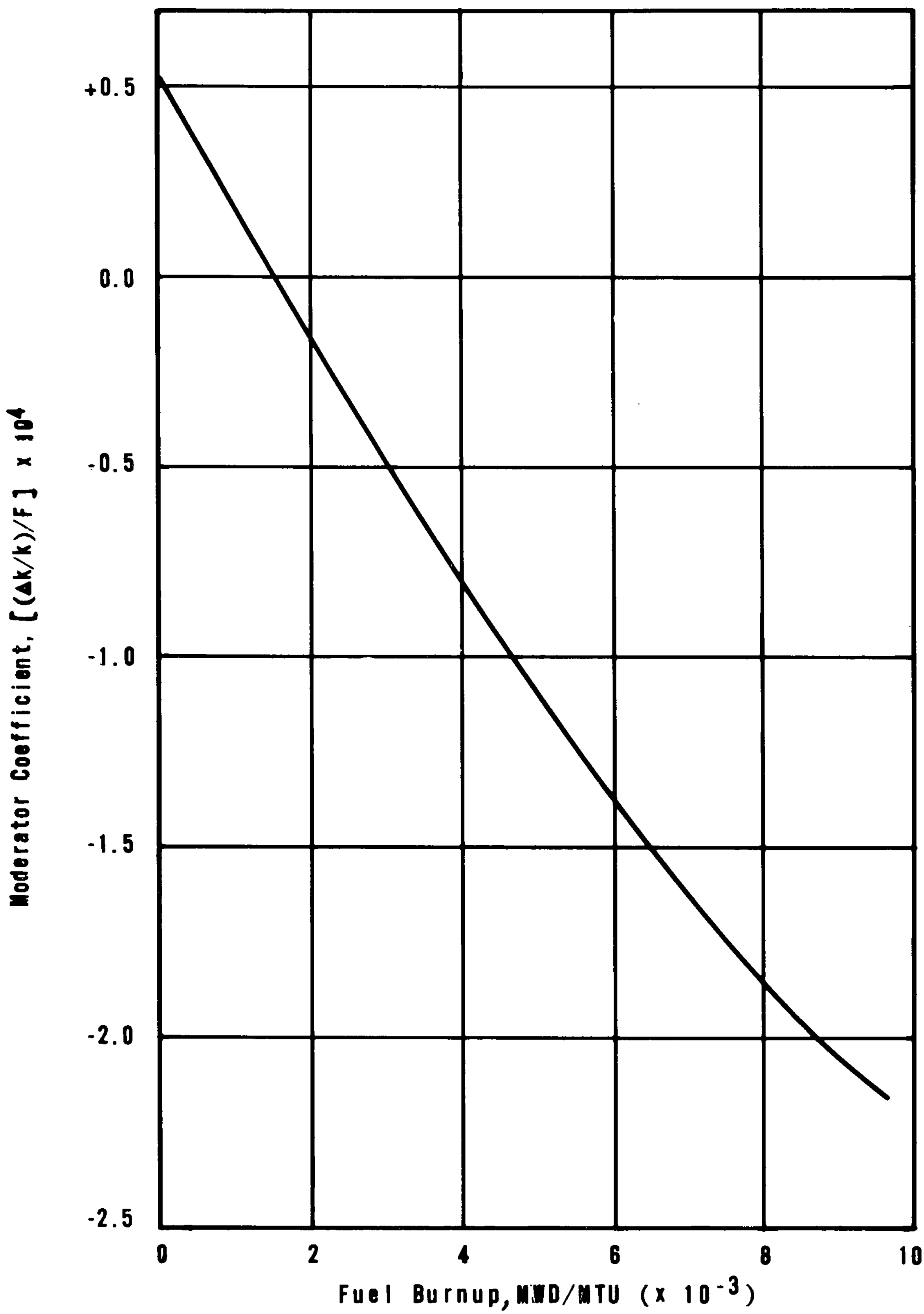
REACTOR VESSEL WATER HEIGHT VERSUS TIME FOR A 36-IN. ID, DOUBLE-ENDED, HOT LEG PIPE RUPTURE FOR 600 PSIG CORE FLOODING TANK OPERATING PRESSURE



NEUTRON POWER VERSUS TIME FOR A 36-IN, ID, DOUBLE-ENDED, HOT LEG PIPE RUPTURE AT RATED POWER WITHOUT TRIP



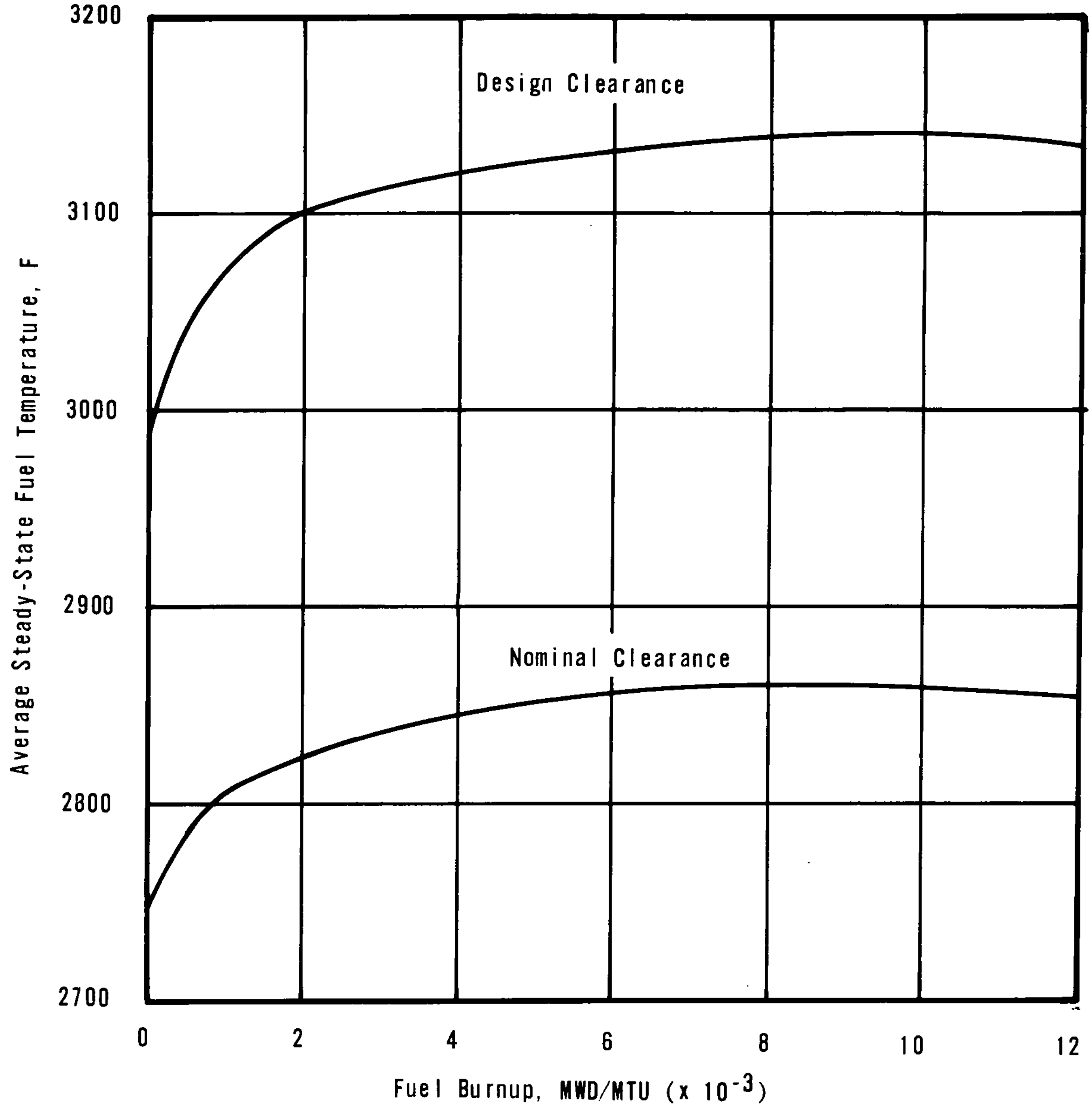
REACTIVITY VERSUS TIME FOR A 36-IN, ID, DOUBLE-ENDED, HOT LEG PIPE RUPTURE AT RATED POWER WITHOUT TRIP



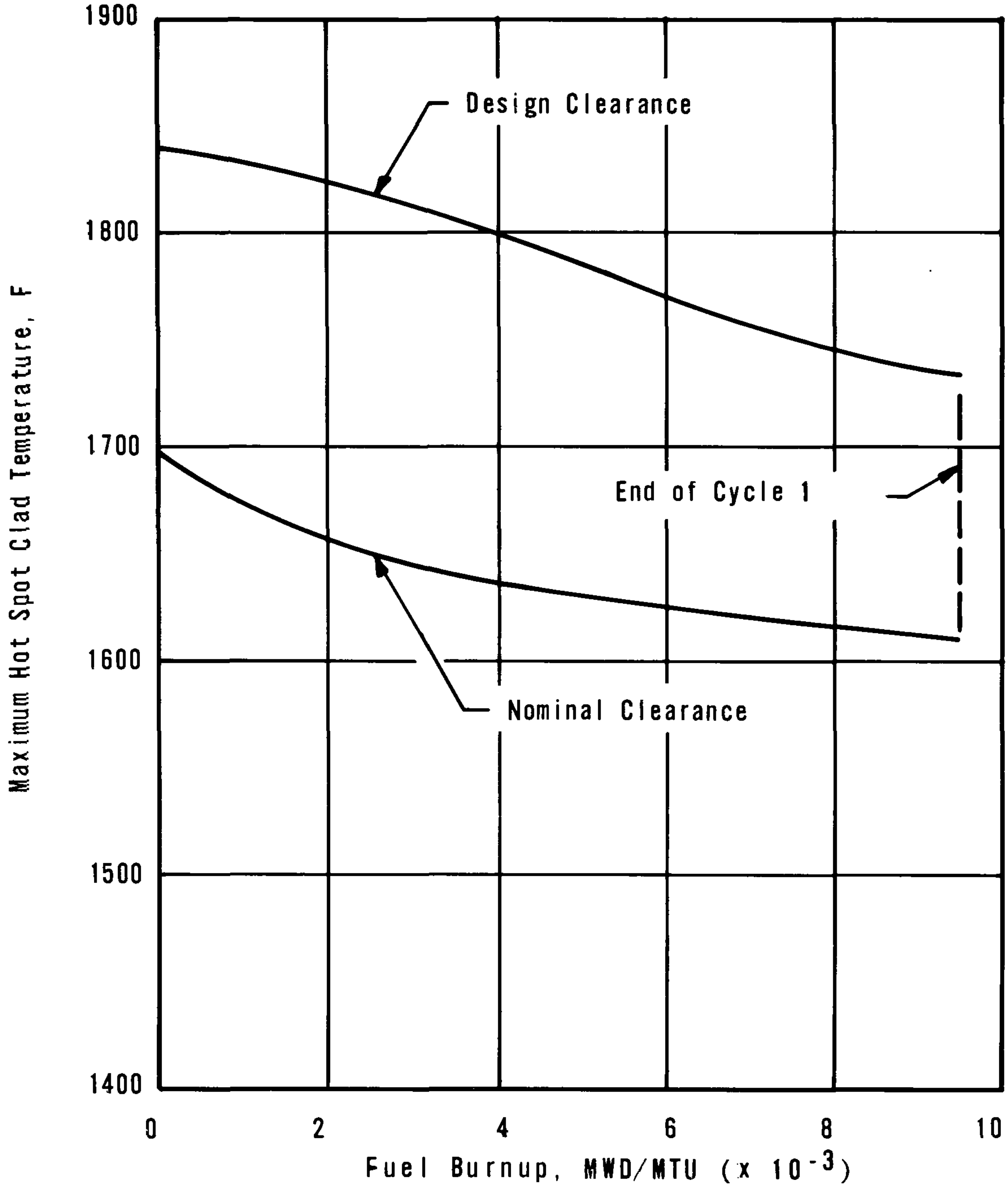
MODERATOR COEFFICIENT AS A FUNCTION OF BURNUP (NO XENON)

OCONEE NUCLEAR STATION

Figure 14 - 38



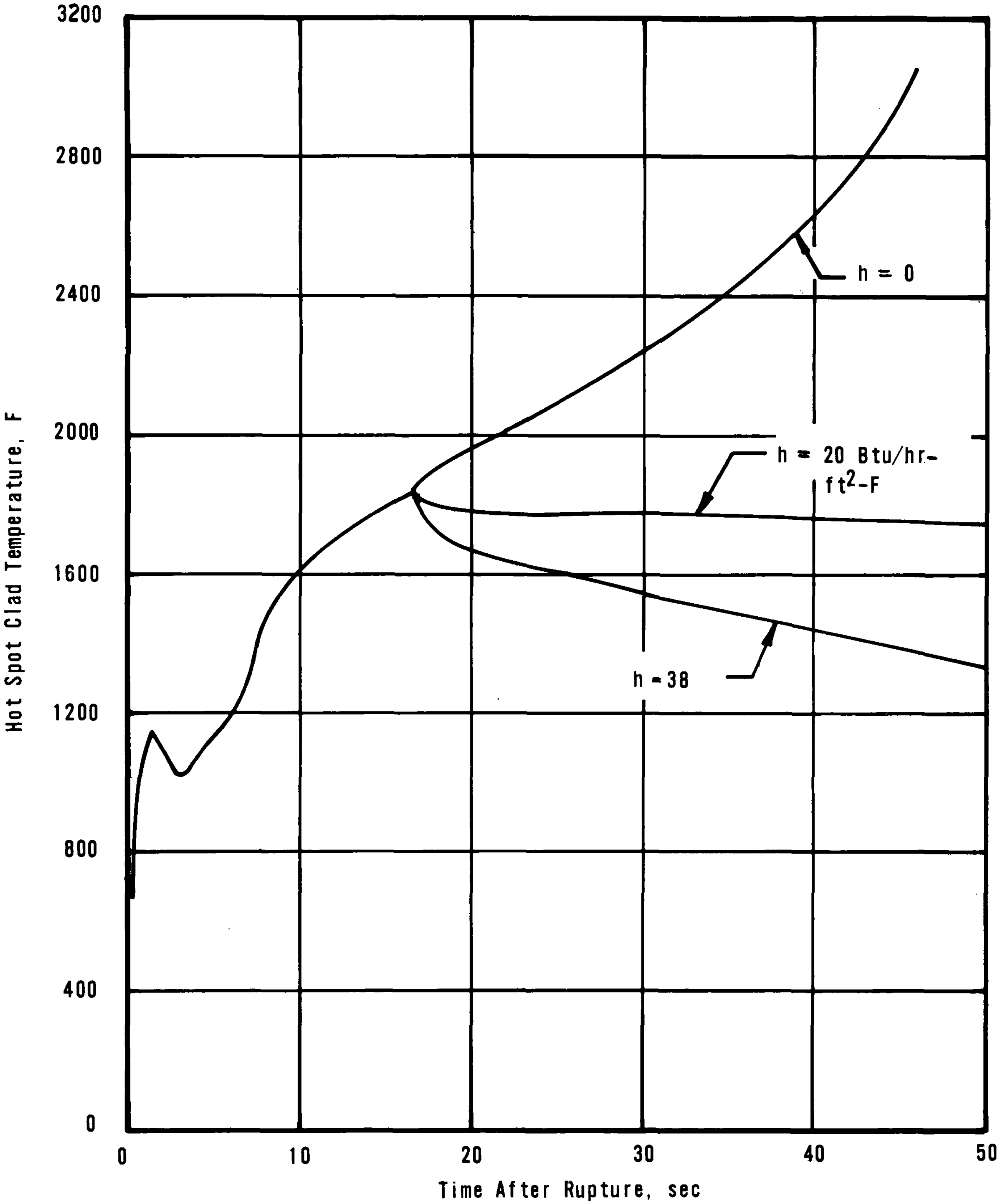
AVERAGE STEADY-STATE FUEL TEMPERATURE VERSUS FUEL BURNUP (MWD/MTU) FOR NOMINAL AND STATISTICAL FUEL-CLAD GAP TOLERANCES



MAXIMUM HOT SPOT CLAD TEMPERATURE VERSUS BURNUP FOR A 36 - IN. ID, DOUBLE-ENDED, HOT LEG PIPE RUPTURE WITHOUT TRIP.

OCONEE NUCLEAR STATION

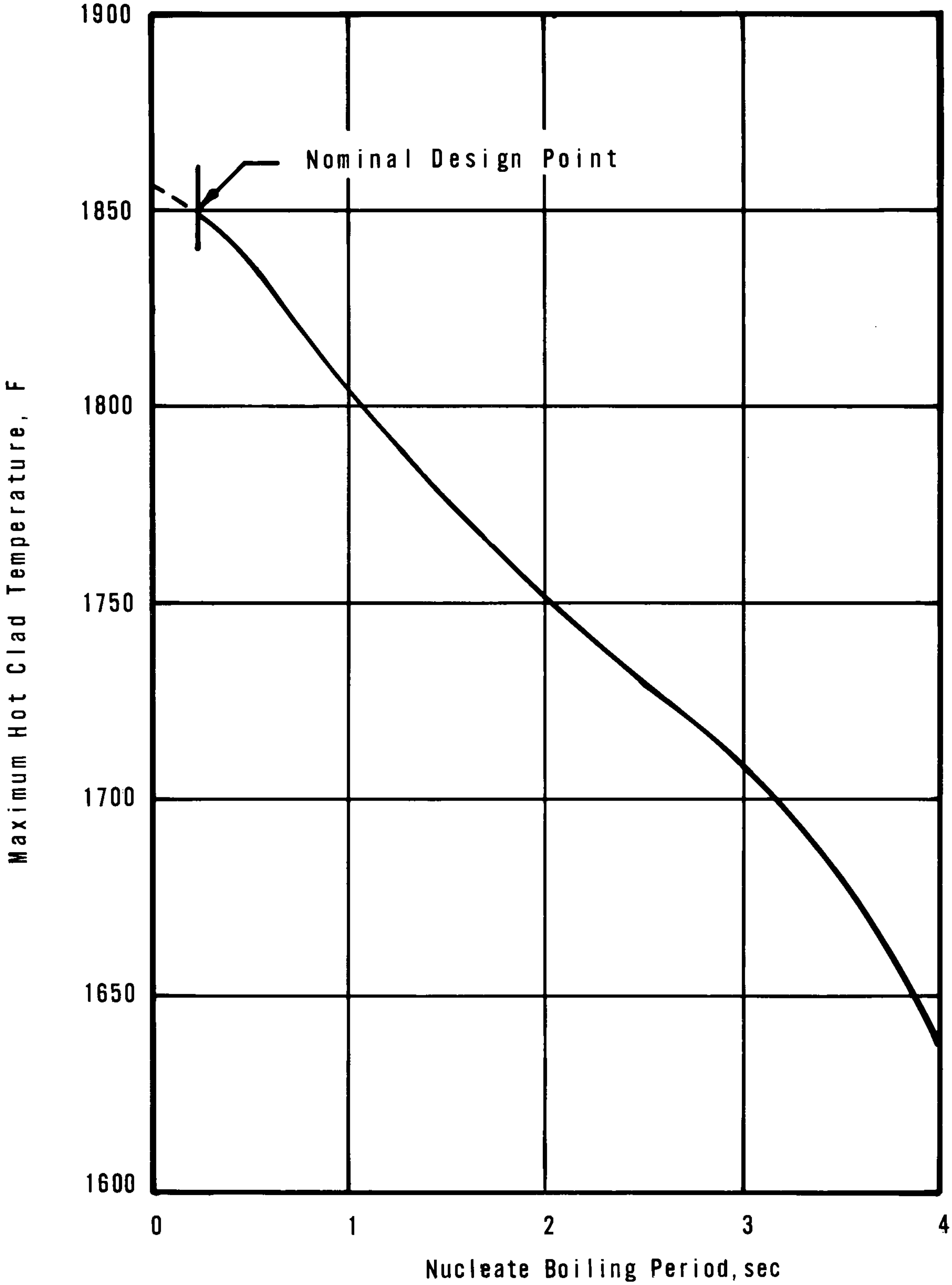
Figure 14 - 40



HOT SPOT CLAD TEMPERATURE VERSUS TIME FOR A 36 - IN. ID, DOUBLE-ENDED, HOT LEG PIPE RUPTURE AND VARIABLE QUENCH COEFFICIENT WITHOUT TRIP.

OCONEE NUCLEAR STATION

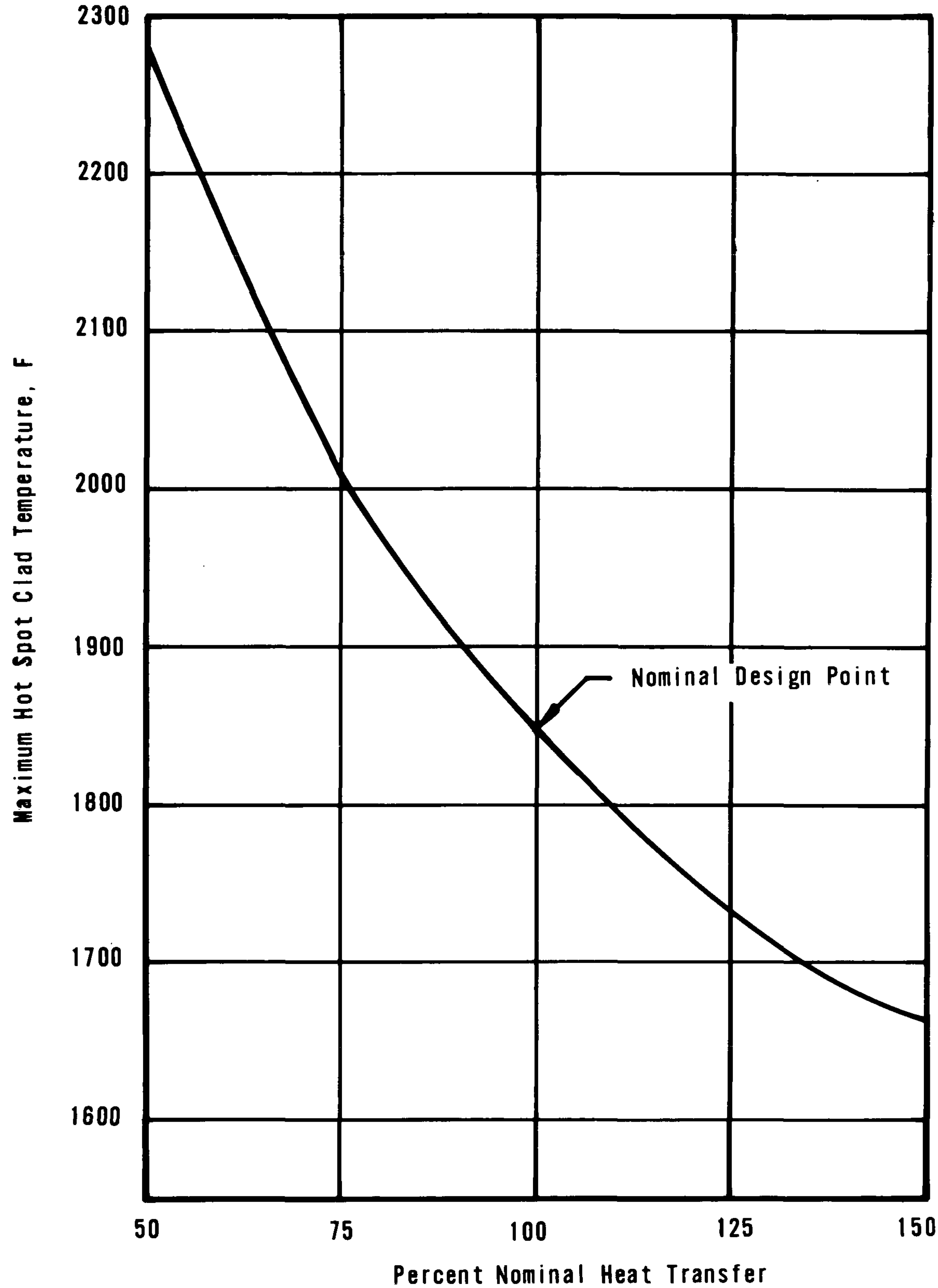
Figure 14 - 41



MAXIMUM HOT SPOT CLAD TEMPERATURE AS A FUNCTION OF TIME TO REACH DNB FOR A 36 - IN. ID, DOUBLE-ENDED HOT LEG PIPE RUPTURE WITHOUT TRIP.

OCONEE NUCLEAR STATION

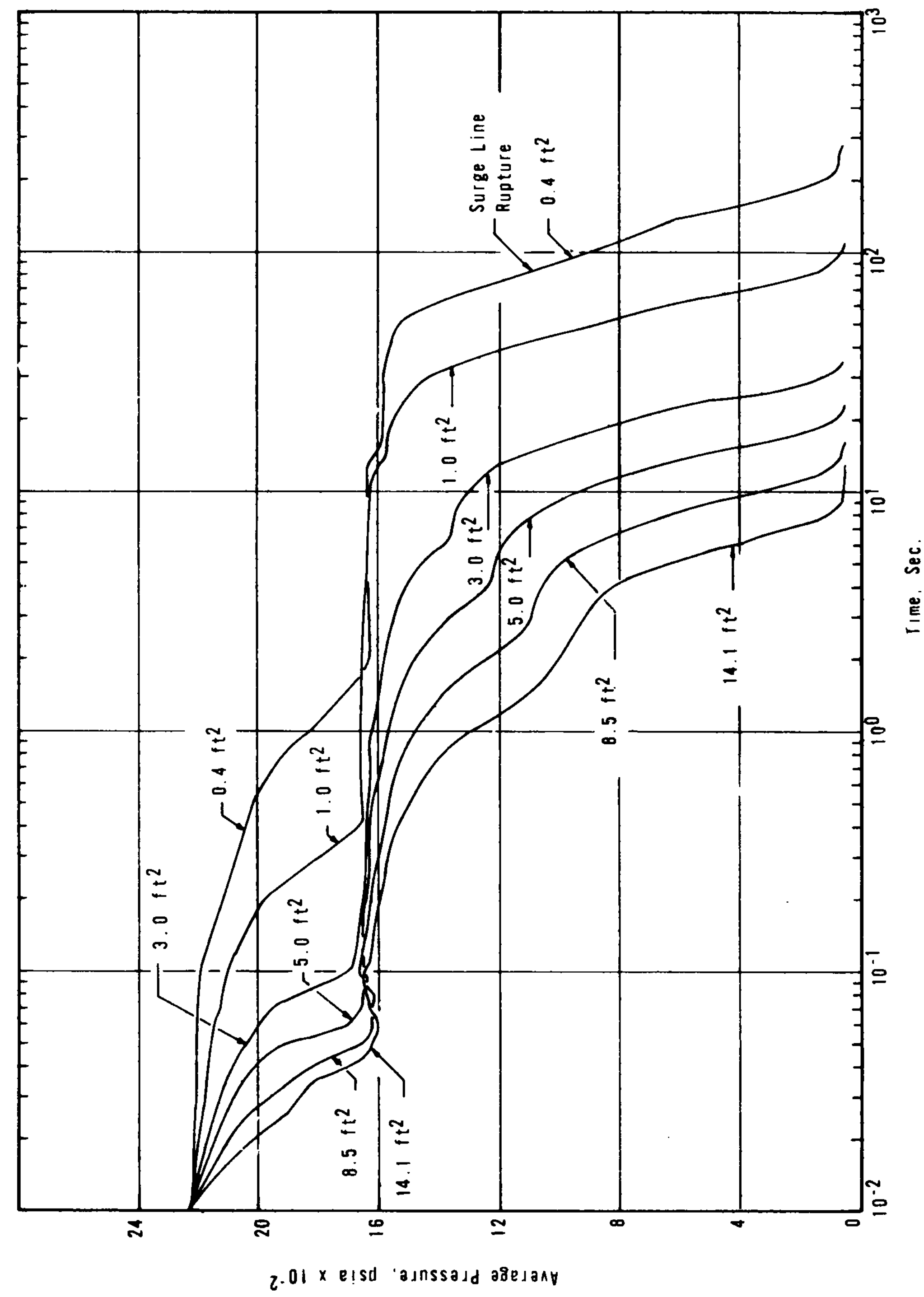
Figure 14 - 42



MAXIMUM HOT SPOT CLAD TEMPERATURE VERSUS VARIABLE HEAT TRANSFER COEFFICIENT AFTER DNB FOR A 36-IN. ID, DOUBLE-ENDED HOT LEG PIPE RUPTURE

OCONEE NUCLEAR STATION

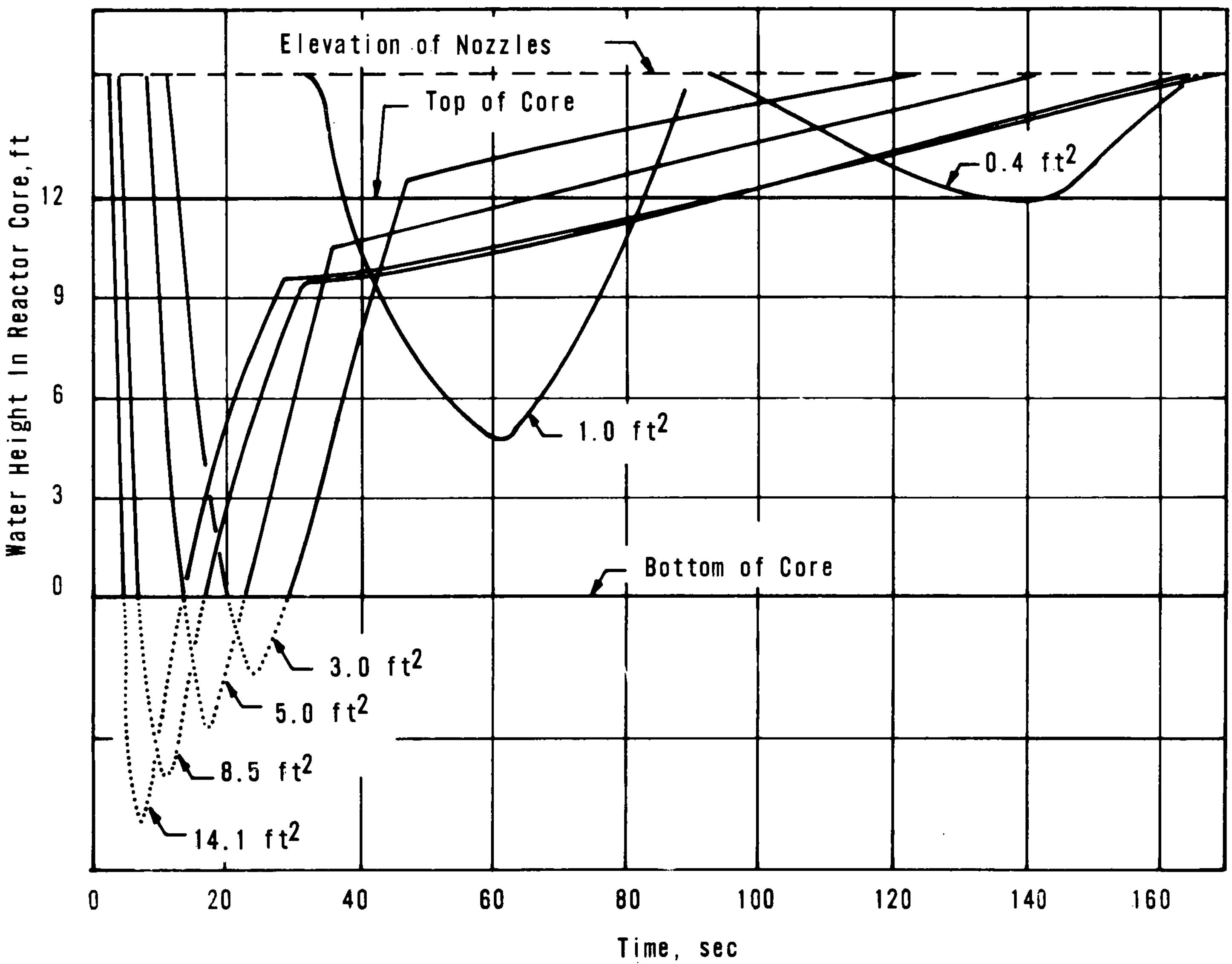
Figure 14 - 43



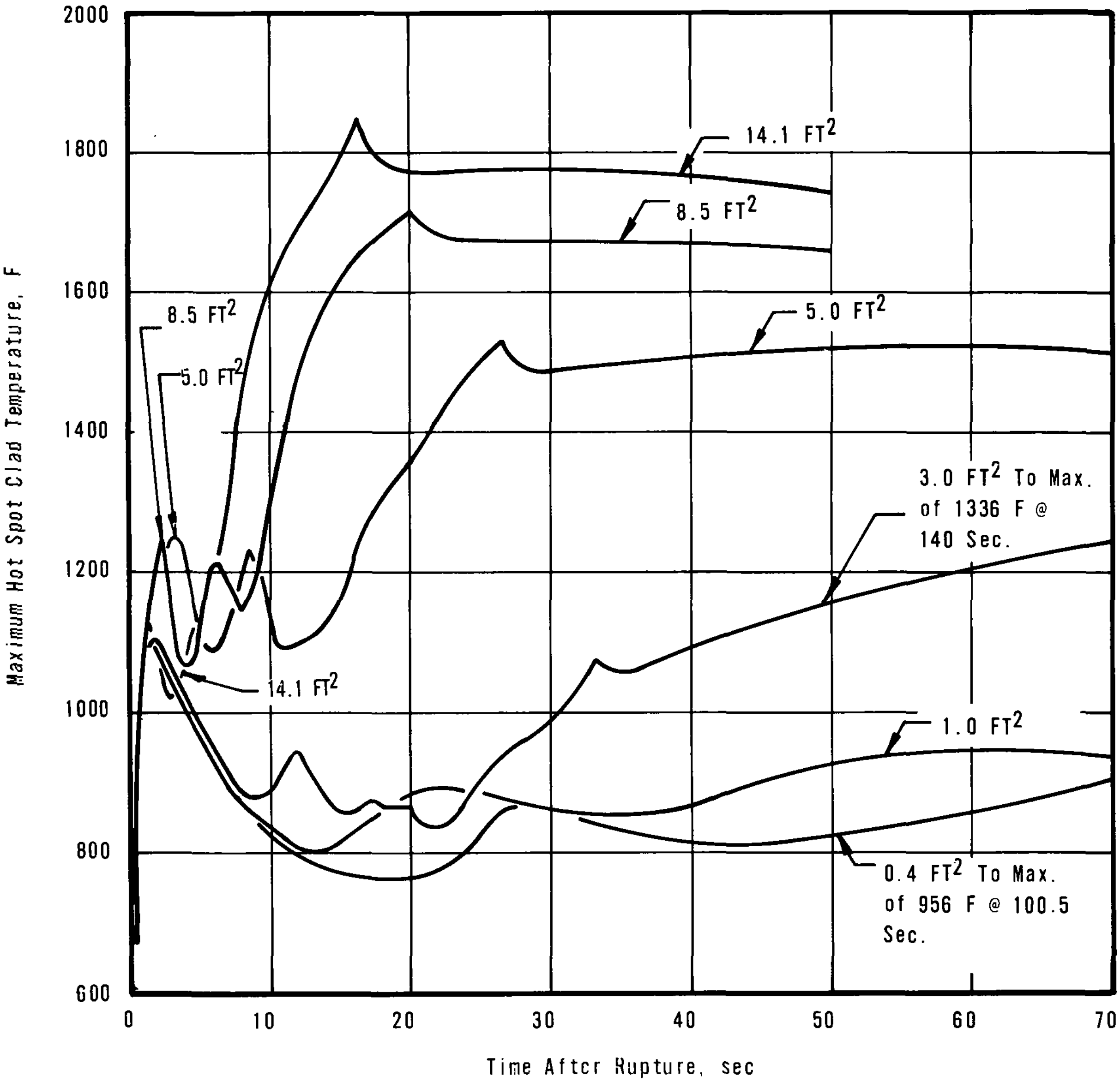
REACTOR COOLANT AVERAGE PRESSURE
FOR THE SPECTRUM OF HOT LEG RUPTURES

OCONEE NUCLEAR STATION

Figure 14 - 44



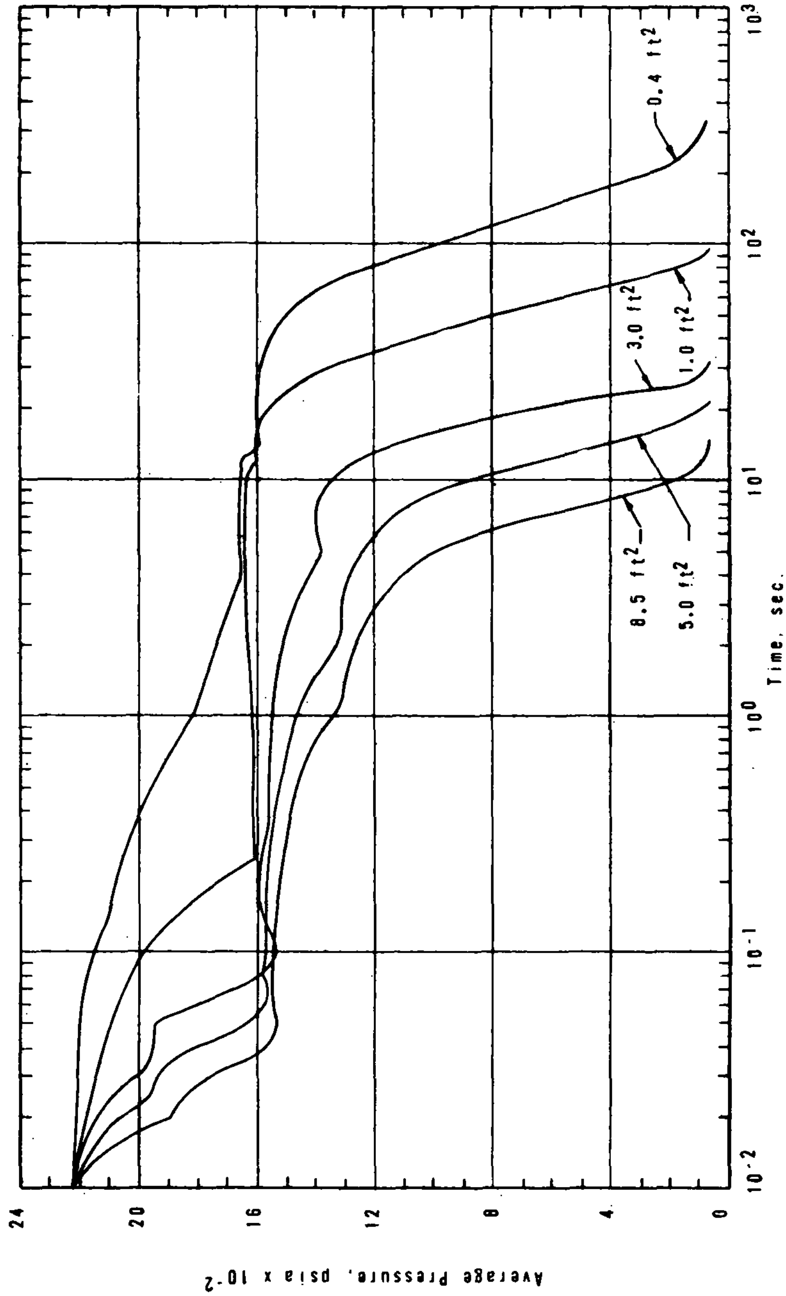
HOT LEG RUPTURES-REACTOR VESSEL WATER HEIGHT VERSUS TIME INCLUDING EFFECTS OF BOILOFF AND INJECTION



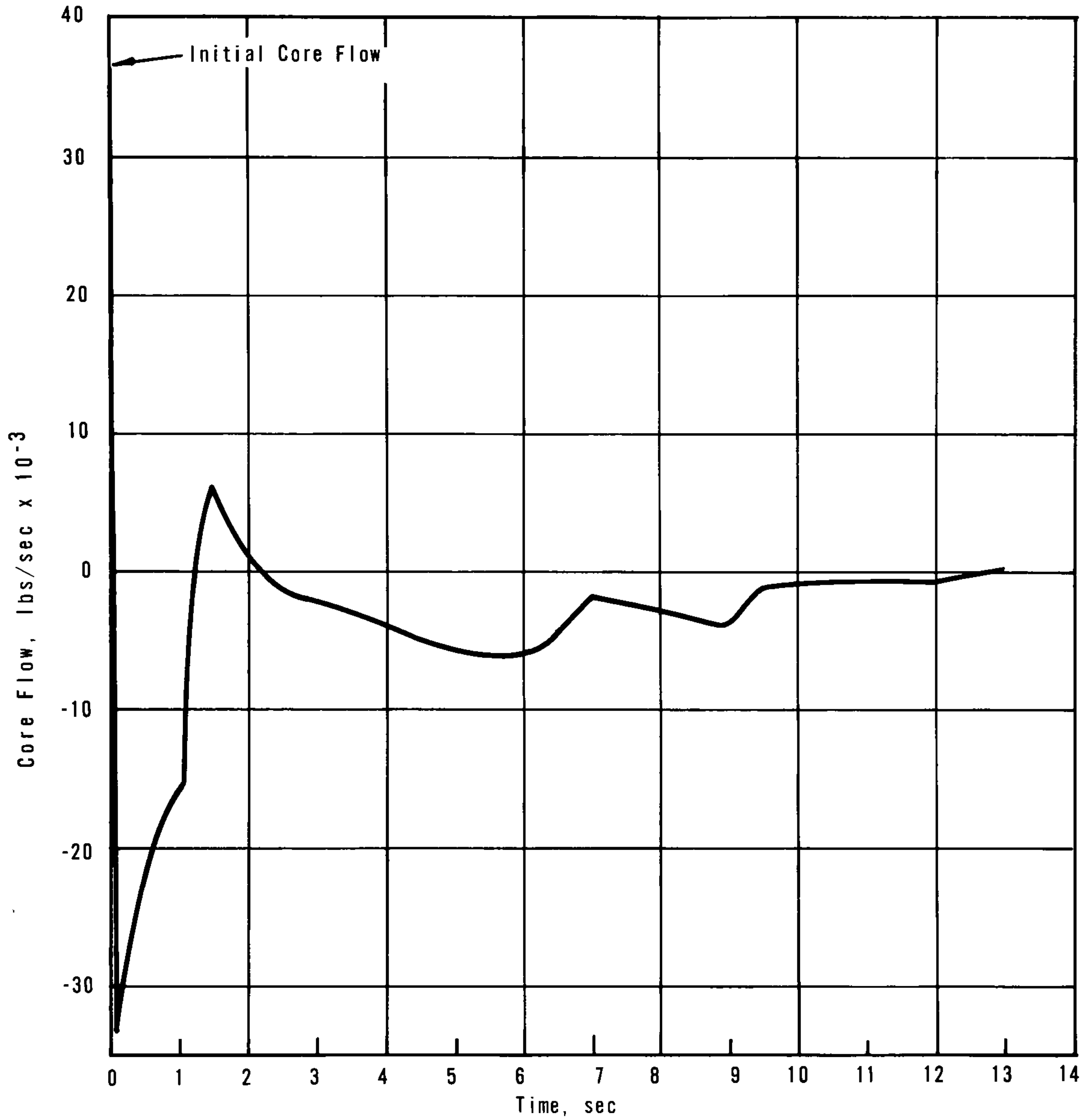
HOT SPOT CLADDING TEMPERATURE VERSUS TIME FOR SPECTRUM OF HOT LEG RUPTURES

OCONEE NUCLEAR STATION

Figure 14 - 46

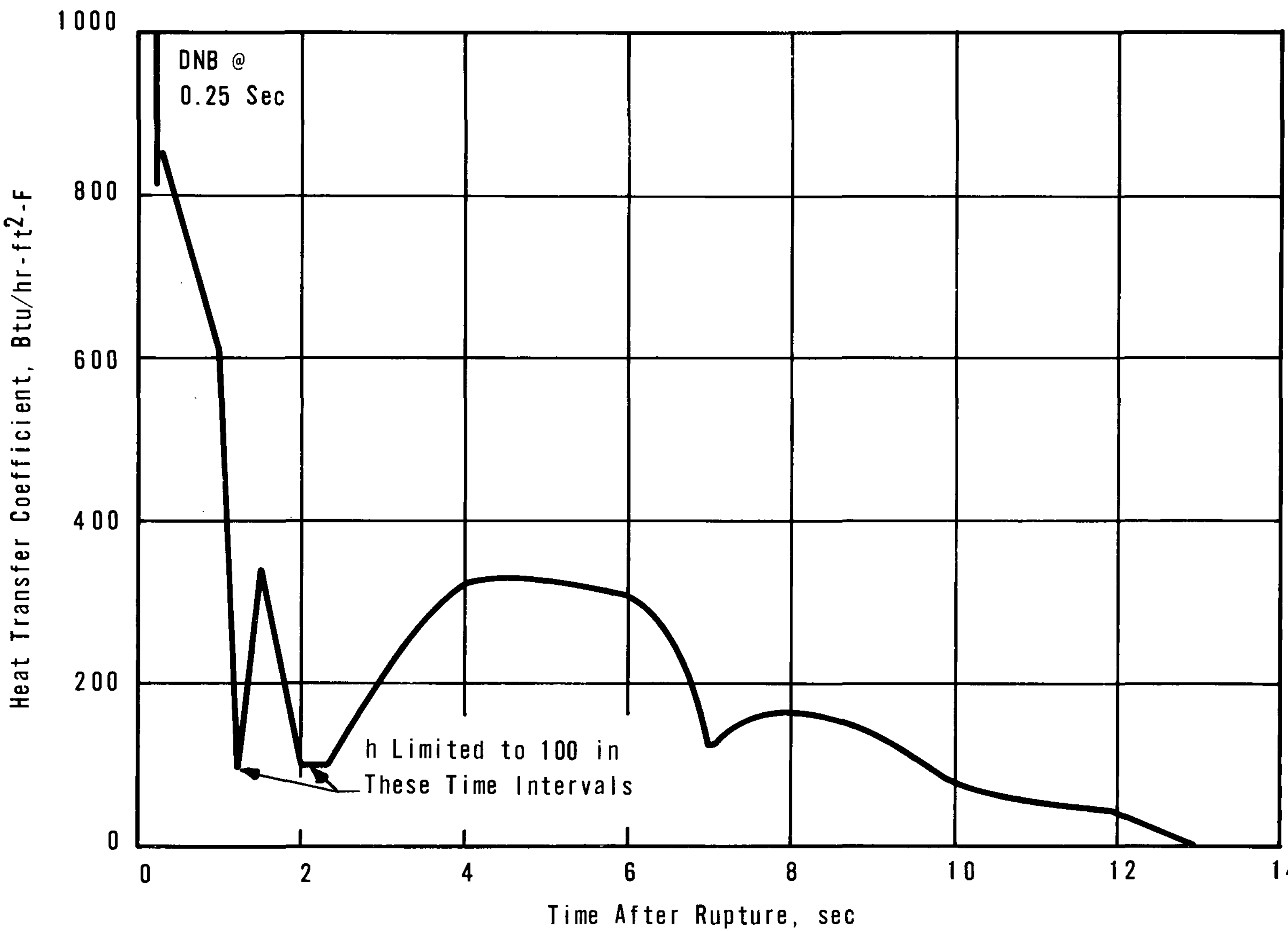


REACTOR COOLANT AVERAGE PRESSURE-
SPECTRUM OF COLD LEG RUPTURE SIZES



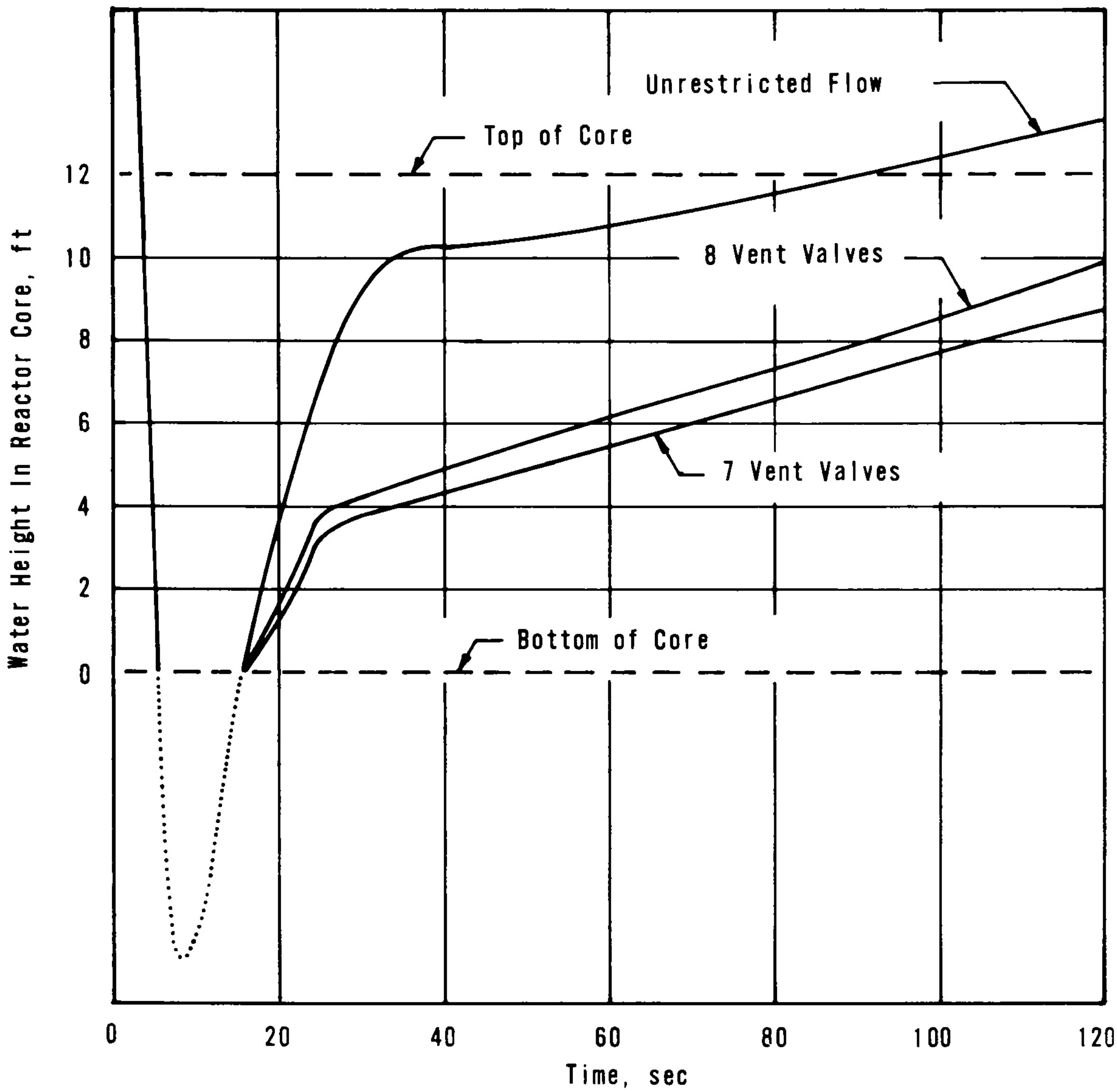
CORE FLOW VERSUS TIME FOR A 28-IN ID, DOUBLE ENDED, COLD LEG PIPE RUPTURE

OCONEE NUCLEAR STATION
Figure 14 - 47A
(New) Rev. 5 5/25/70



HOT CHANNEL CLAD SURFACE HEAT TRANSFER COEFFICIENT AFTER DNB VERSUS TIME FOR A 28-IN ID DOUBLE-ENDED COLD LEG PIPE RUPTURE

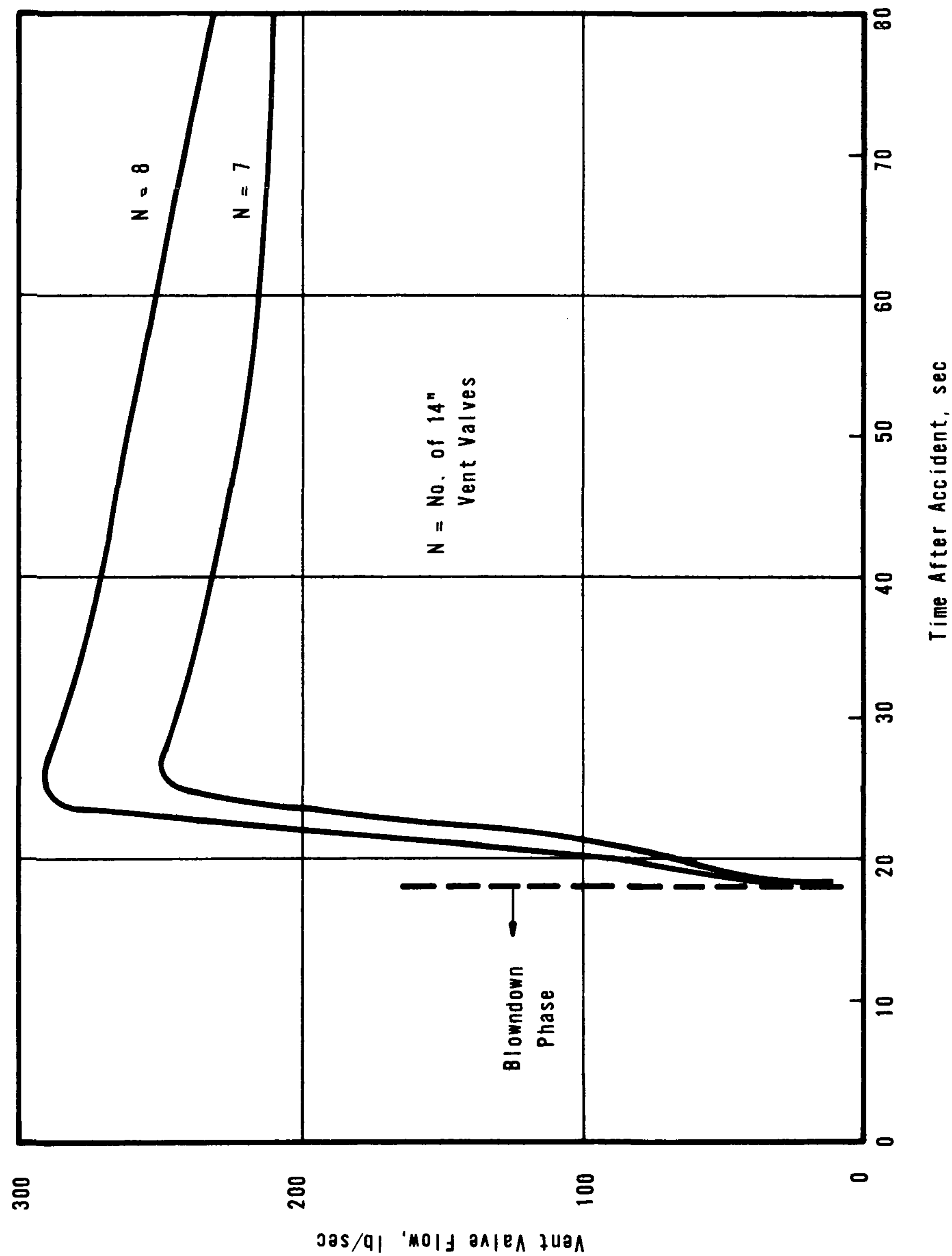
OCONEE NUCLEAR STATION
Figure 14 - 47B
(New) Rev. 5 5/25/70



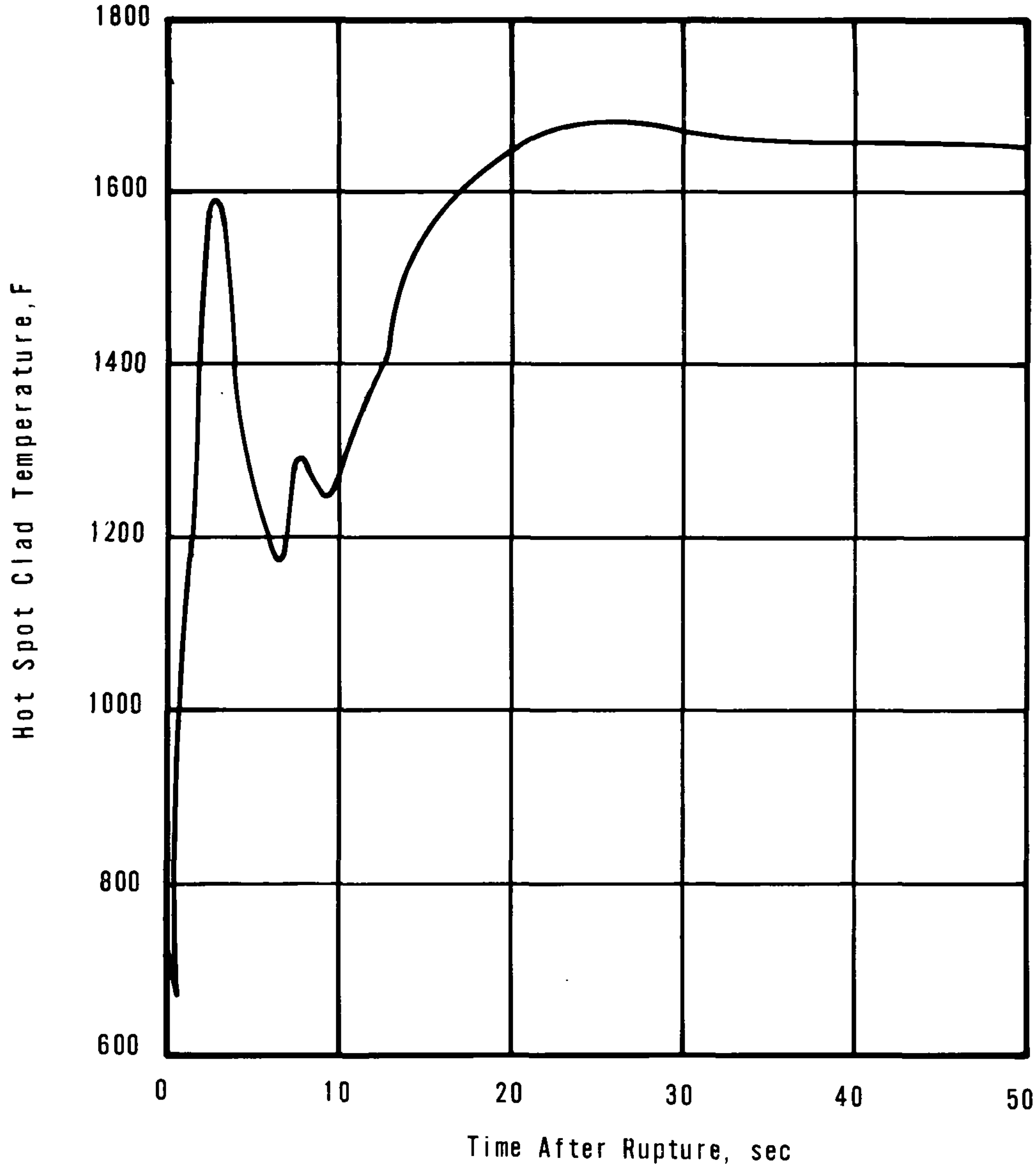
REACTOR VESSEL WATER HEIGHT VERSUS TIME AND NUMBER OF INTERNALS VENT VALVES FOLLOWING A 28-IN. ID, DOUBLE-ENDED, COLD LEG PIPE RUPTURE

OCONEE NUCLEAR STATION

Figure 14 - 48



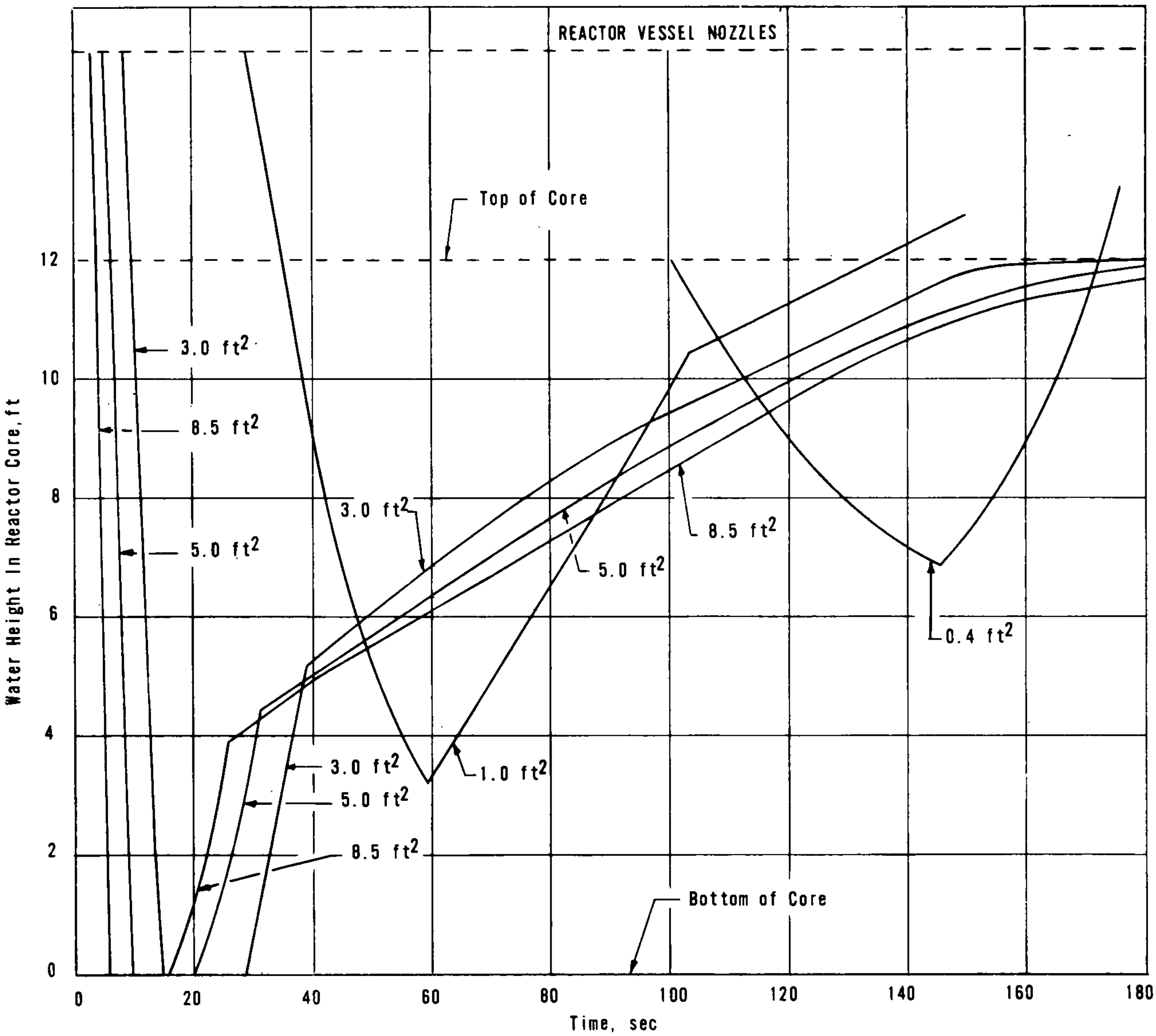
SUPERHEATED STEAM FLOW THROUGH INTERNALS VENT VALVES FOLLOWING A 28-IN ID, DOUBLE-ENDED COLD LEG PIPE RUPTURE



HOT SPOT CLAD TEMPERATURE VERSUS TIME FOR A 8.5 FT^2 DOUBLE-ENDED COLD LEG PIPE RUPTURE WITH 4 SECONDS FLOW DIVERSION

OCONEE NUCLEAR STATION

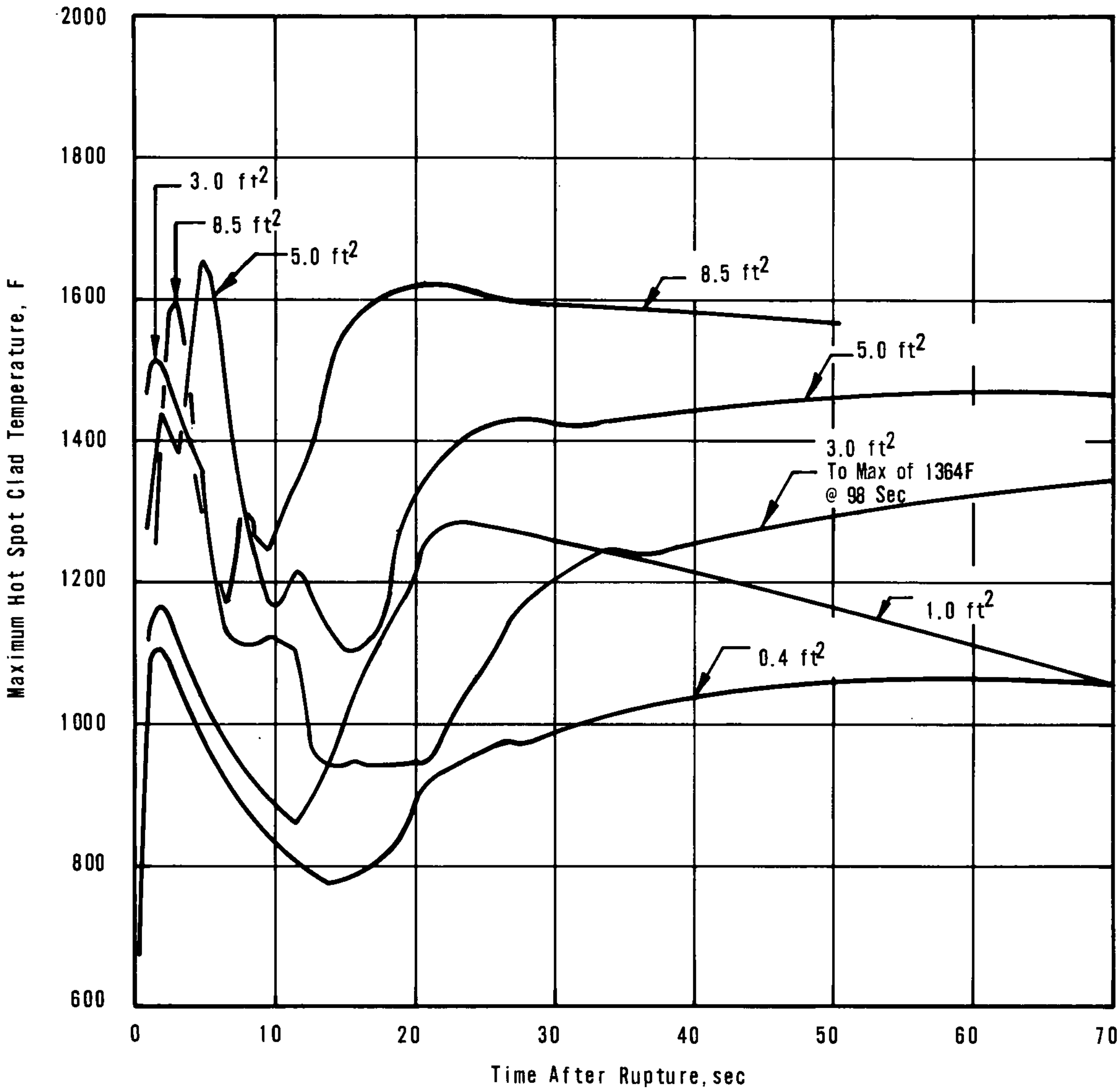
Figure 14 - 49



COLD LEG RUPTURES-REACTOR VESSEL WATER HEIGHT VERSUS TIME INCLUDING EFFECTS OF BOILOFF AND INJECTION

OCONEE NUCLEAR STATION

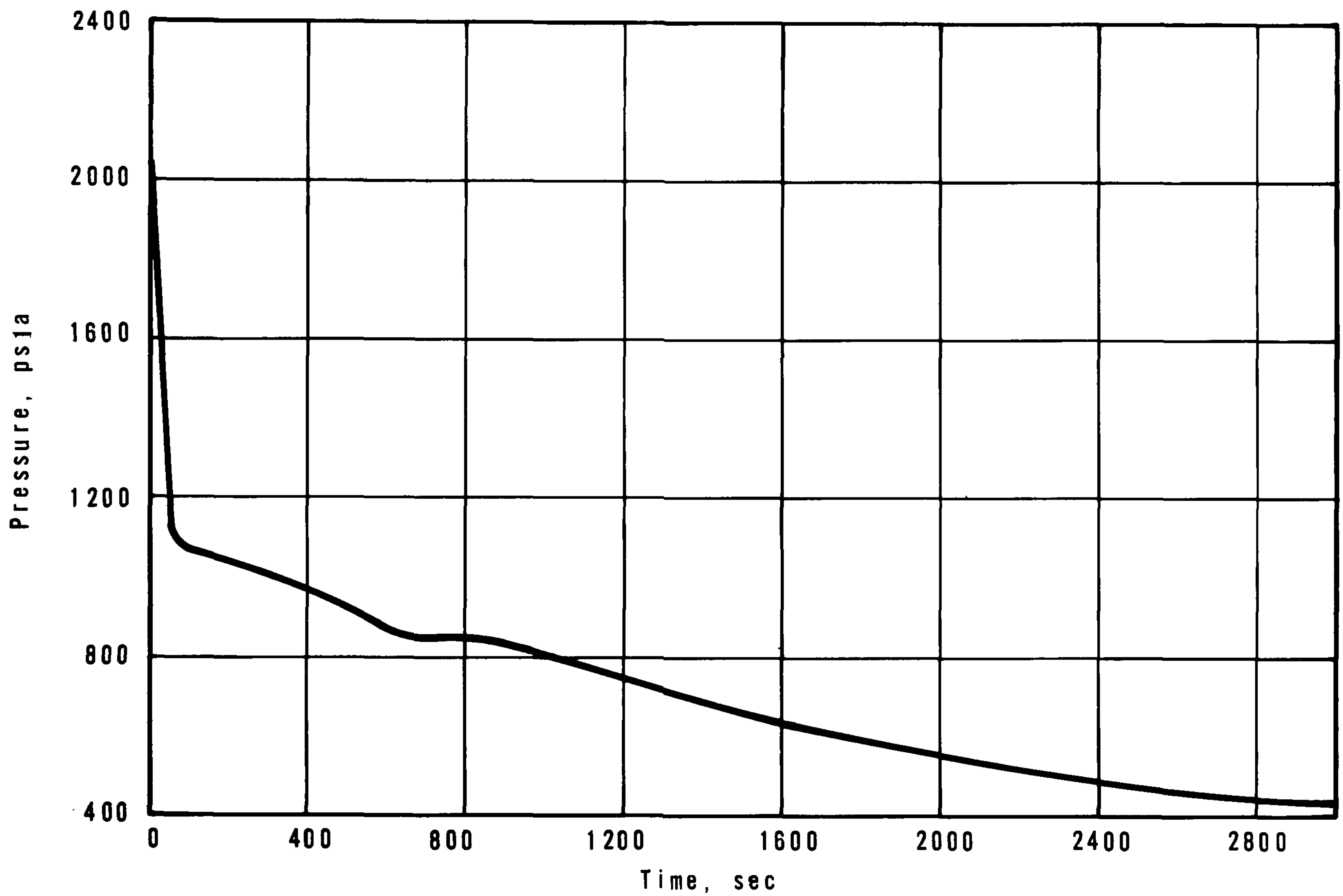
Figure 14 - 50



HOT SPOT CLADDING TEMPERATURE VERSUS
TIME FOR SPECTRUM OF COLD LEG RUPTURES

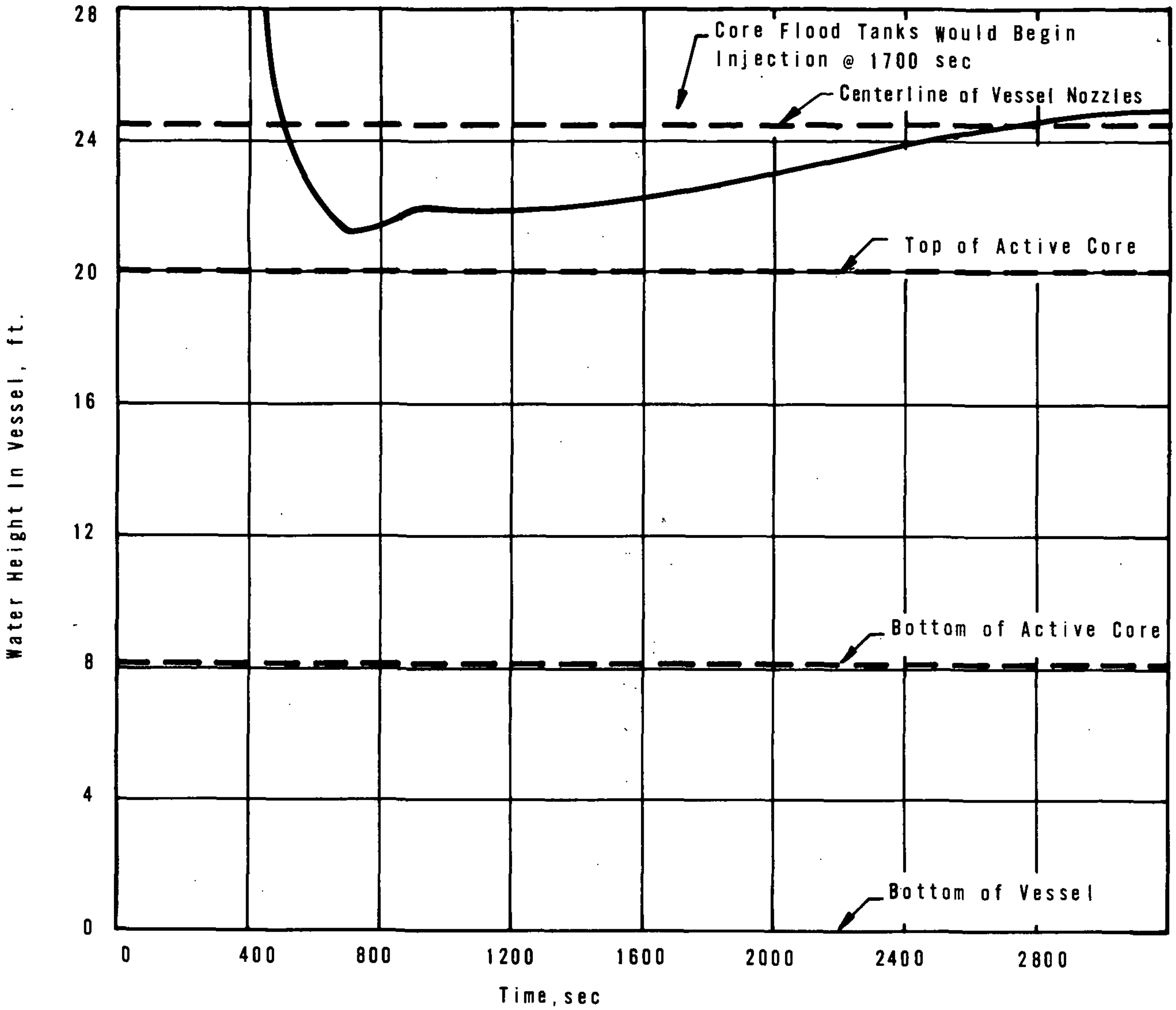
OCONEE NUCLEAR STATION

Figure 14 - 51



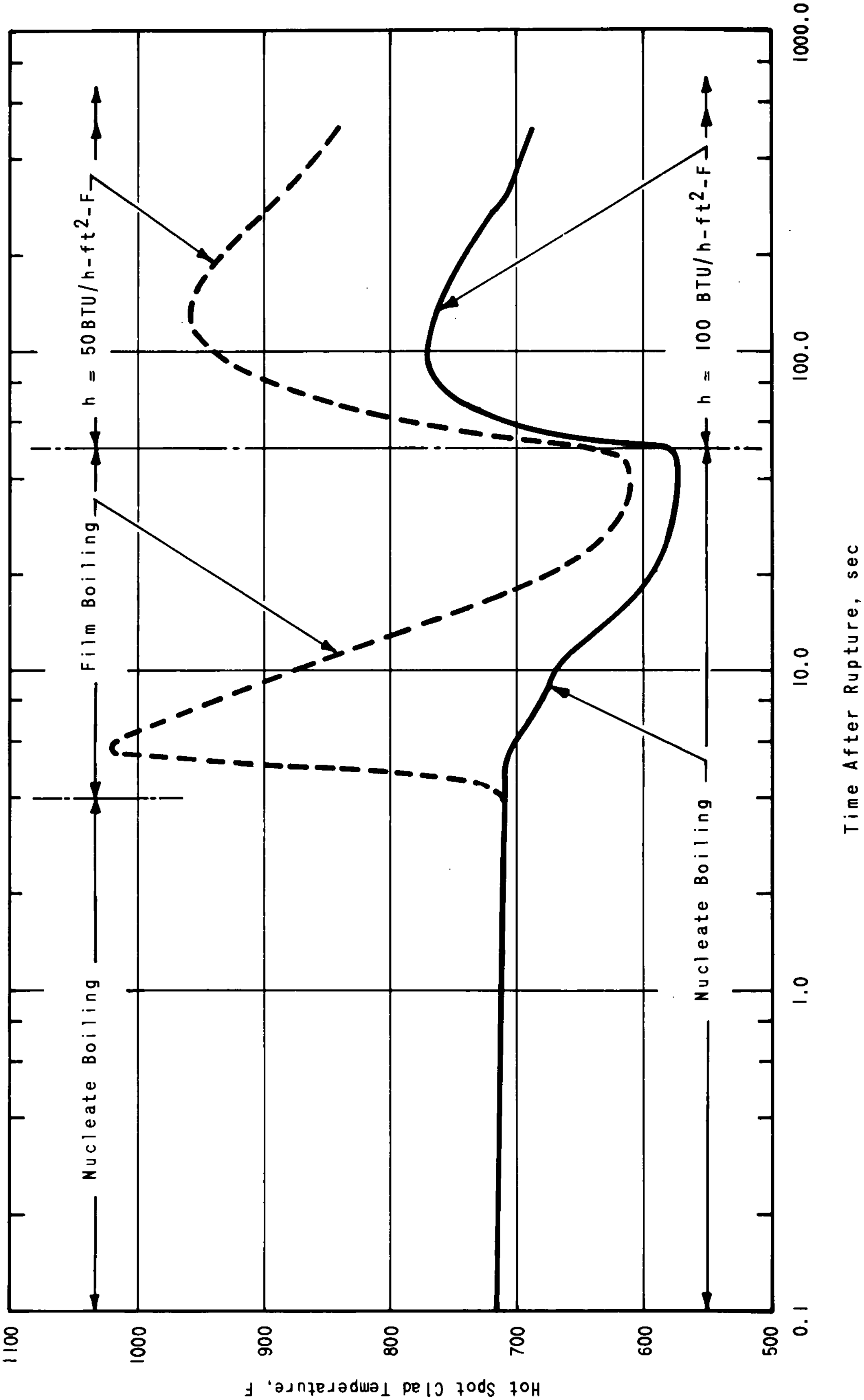
REACTOR COOLANT AVERAGE PRESSURE
VS TIME FOR A 4-IN. ID RUPTURE

FIGURE 14-52



VESSEL WATER HEIGHT VERSUS TIME FOR
A 4-IN, ID RUPTURE

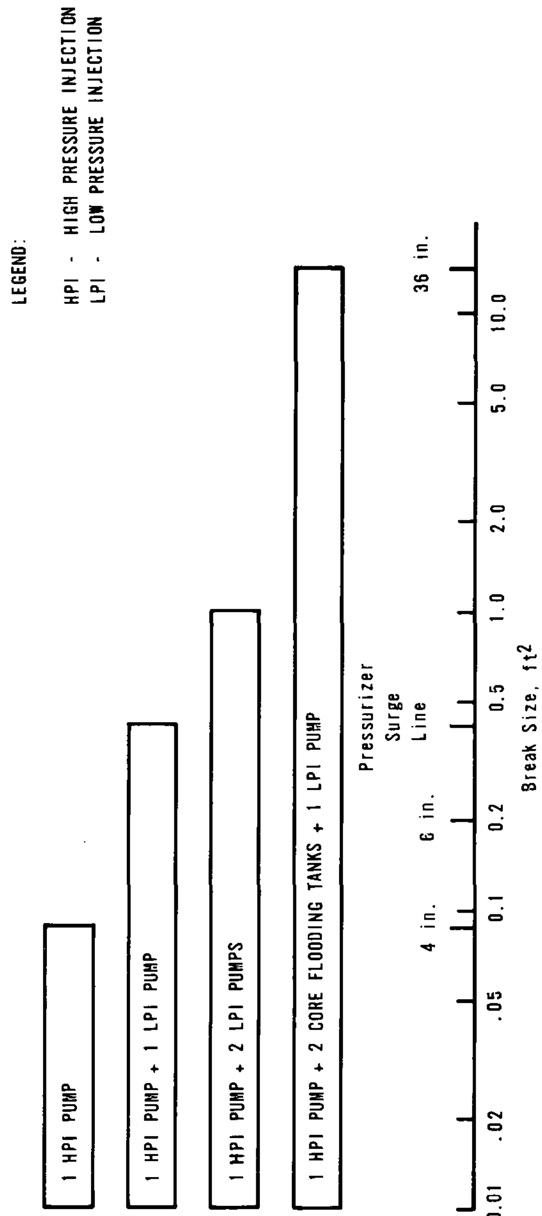
OCONEE NUCLEAR STATION
Figure 14 -53
Rev. 17 12/17/71



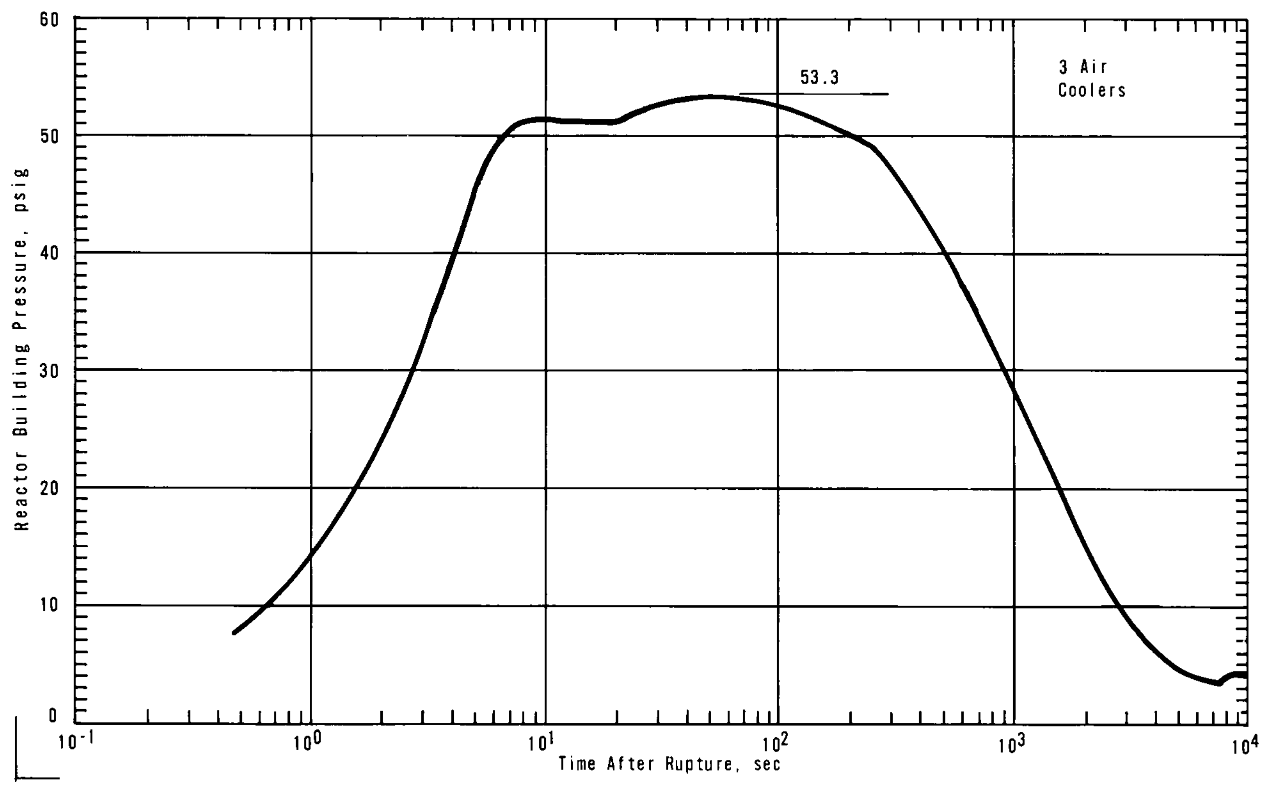
HOT SPOT CLAD TEMPERATURE VERSUS TIME FOR A 4-IN. ID, RUPTURE

OCONEE NUCLEAR STATION

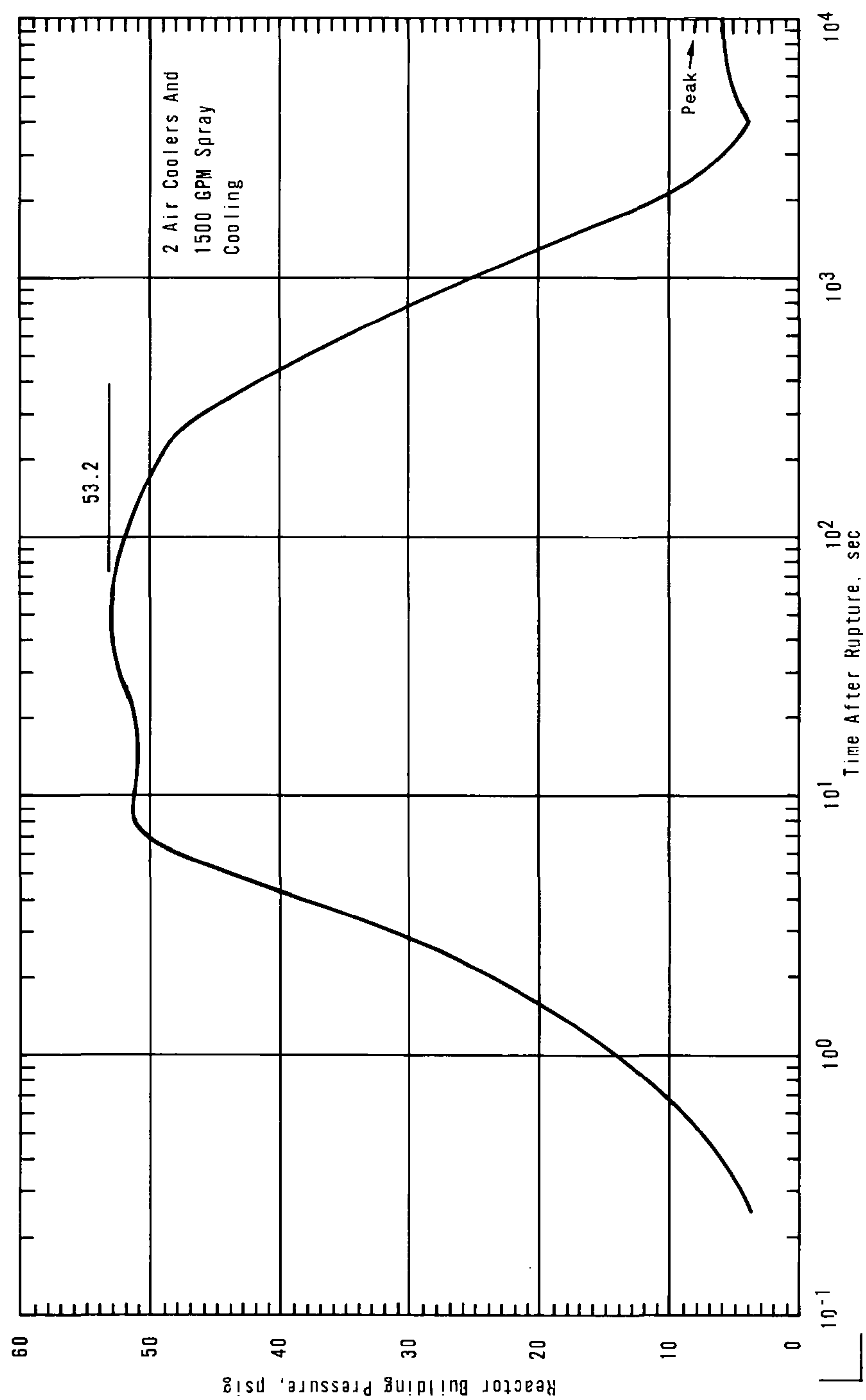
Figure 14 - 54



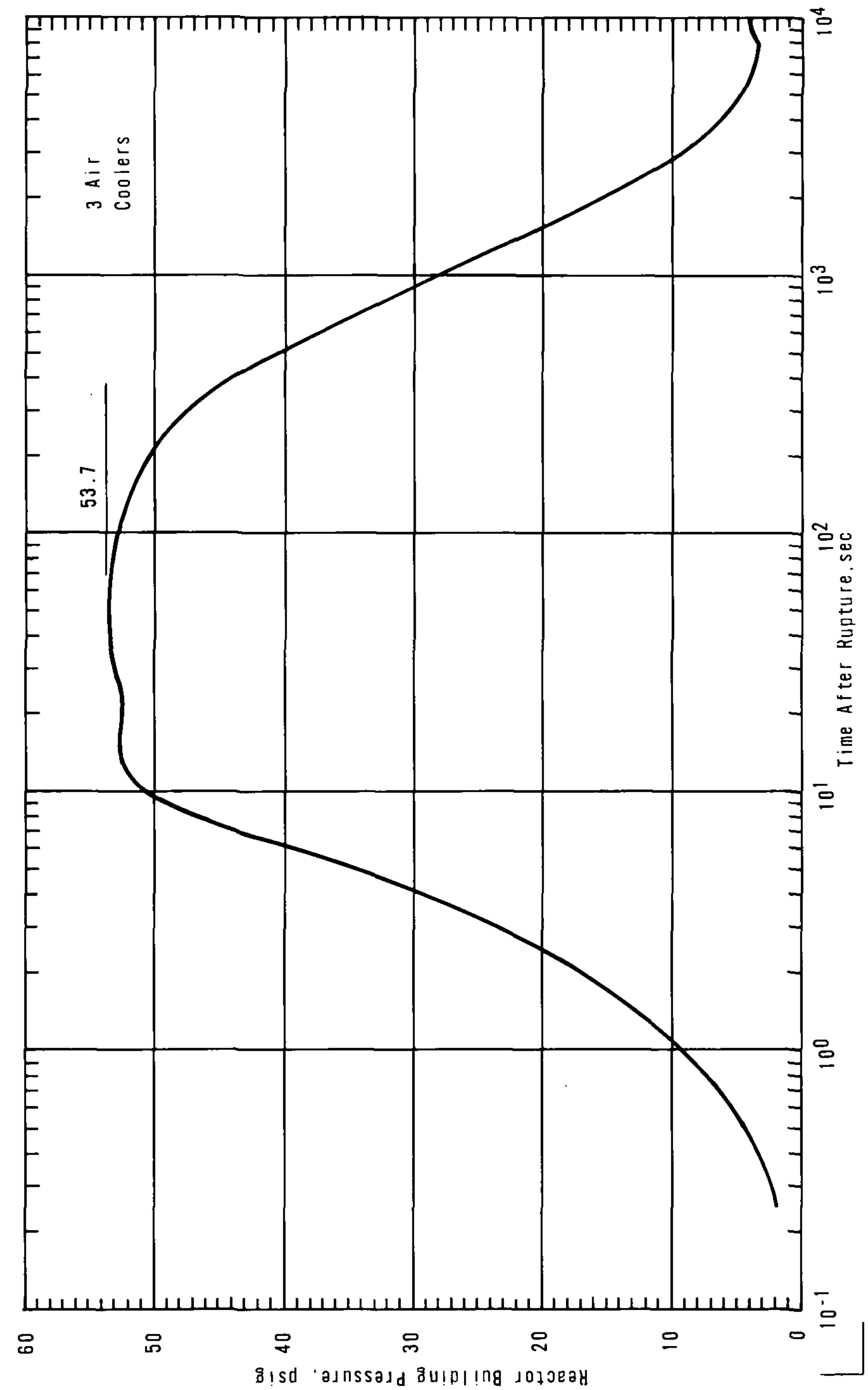
EMERGENCY CORE COOLING SYSTEMS
CAPABILITY TO MEET FUEL CLAD
TEMPERATURE DESIGN LIMIT



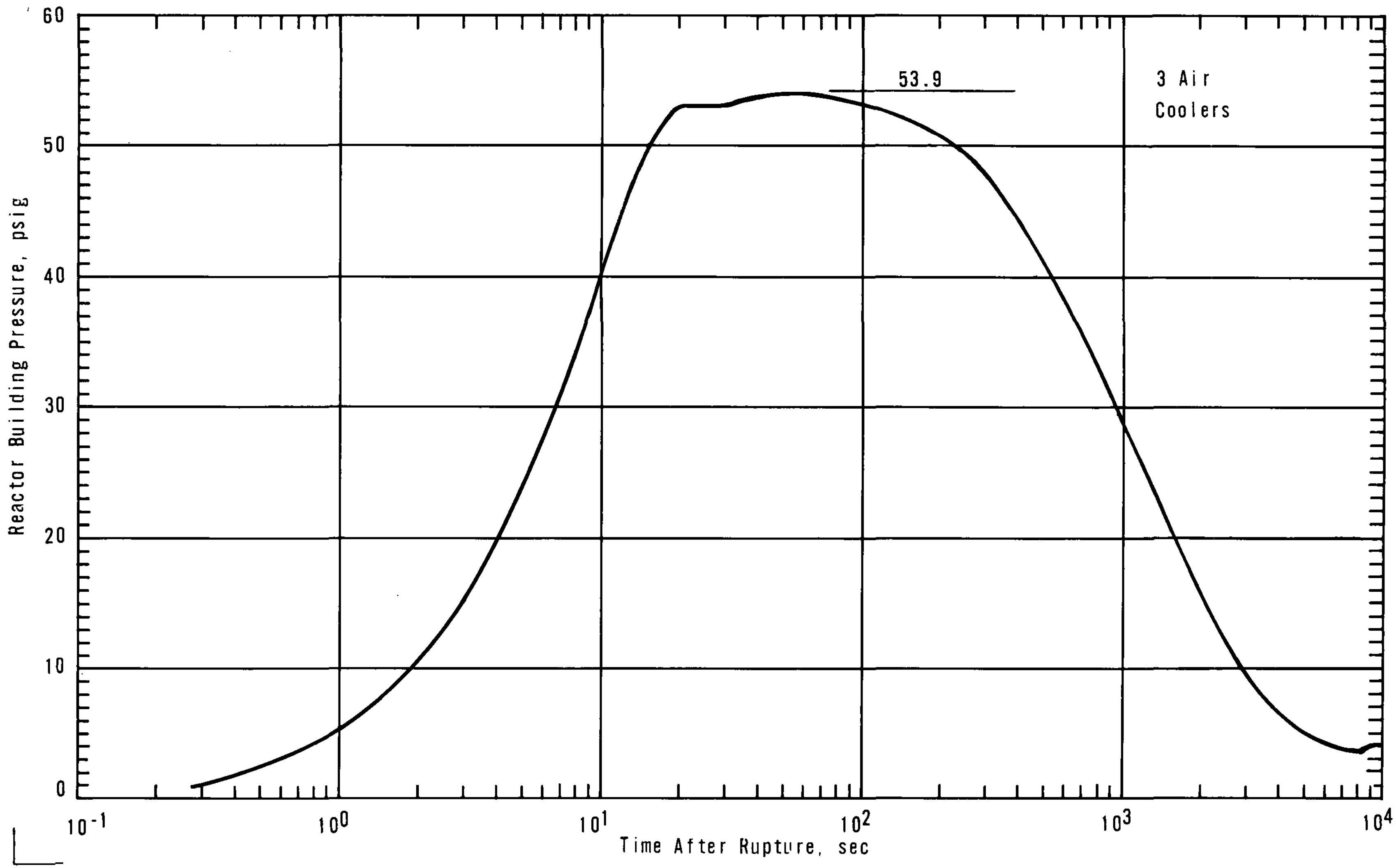
REACTOR BUILDING PRESSURE VERSUS TIME FOR A 36-IN. ID, DOUBLE-ENDED PIPE RUPTURE WITH 3 REACTOR BUILDING AIR COOLERS.



REACTOR BUILDING PRESSURE VERSUS TIME FOR A 36-IN. ID, DOUBLE-ENDED, PIPE RUPTURE WITH 2 AIR COOLERS AND 1500 GPM SPRAYS

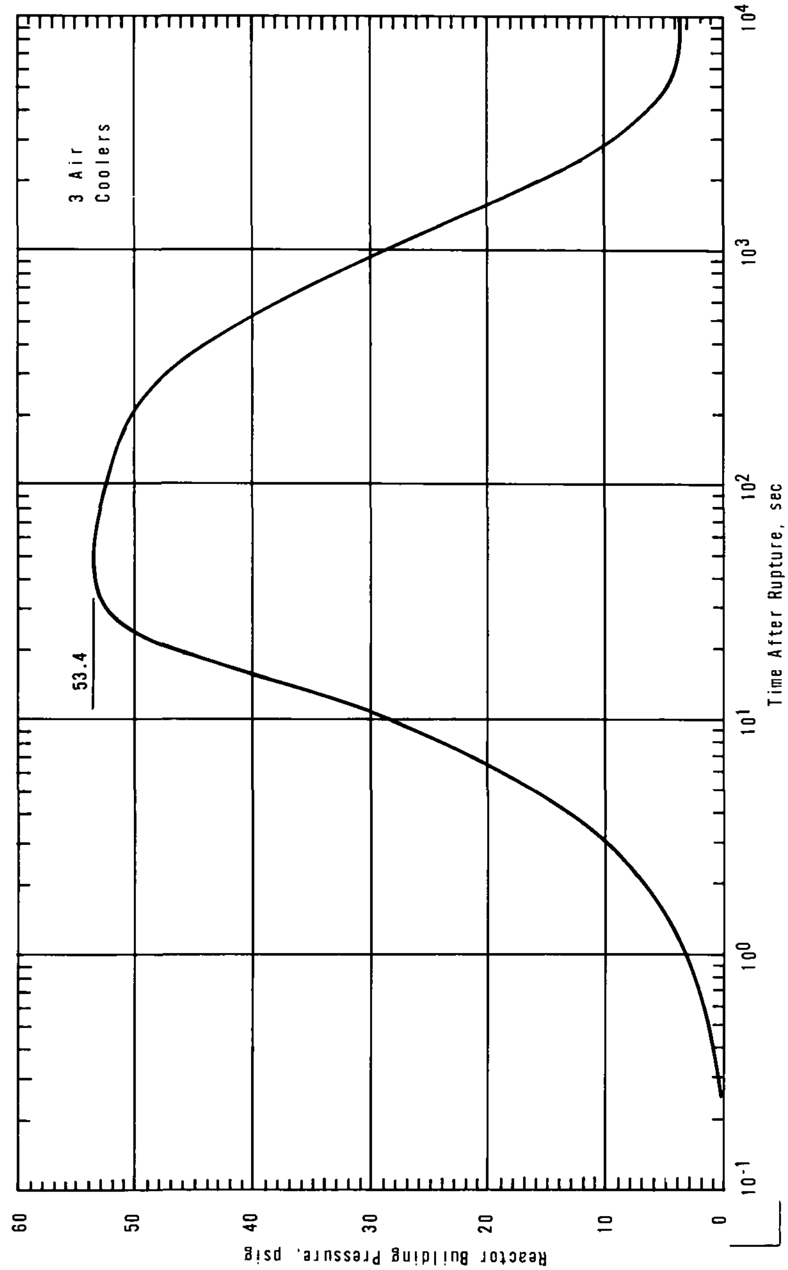


REACTOR BUILDING PRESSURE VERSUS TIME AFTER RUPTURE (8.5 FT^2)



REACTOR BUILDING PRESSURE VERSUS TIME AFTER RUPTURE (5.0 FT^2)

OCONEE NUCLEAR STATION
Figure 14 - 59
Rev. 5 5/25/70

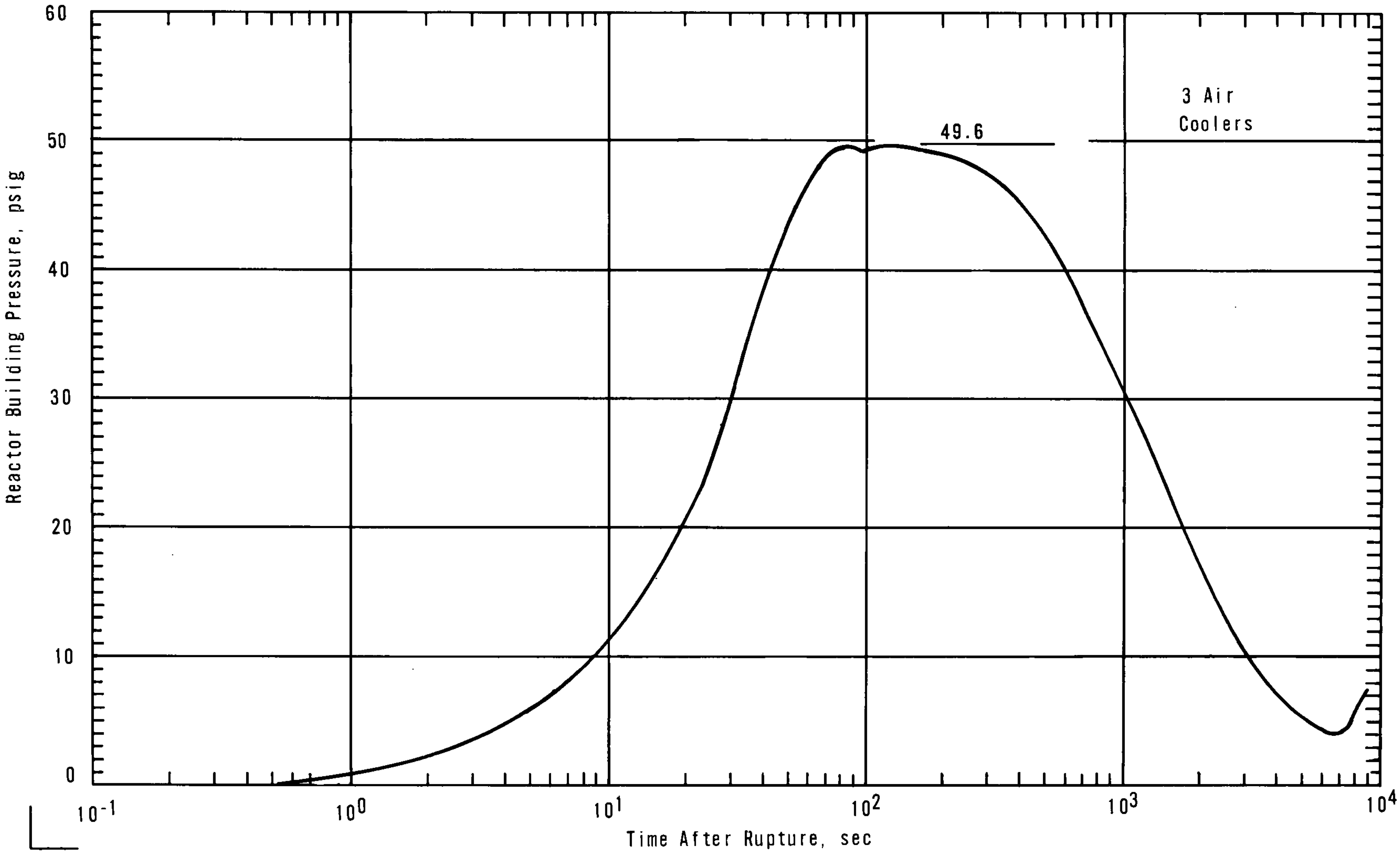


REACTOR BUILDING PRESSURE VERSUS TIME
AFTER RUPTURE (3.0 FT^2)

OCONEE NUCLEAR STATION

Figure 14 - 60

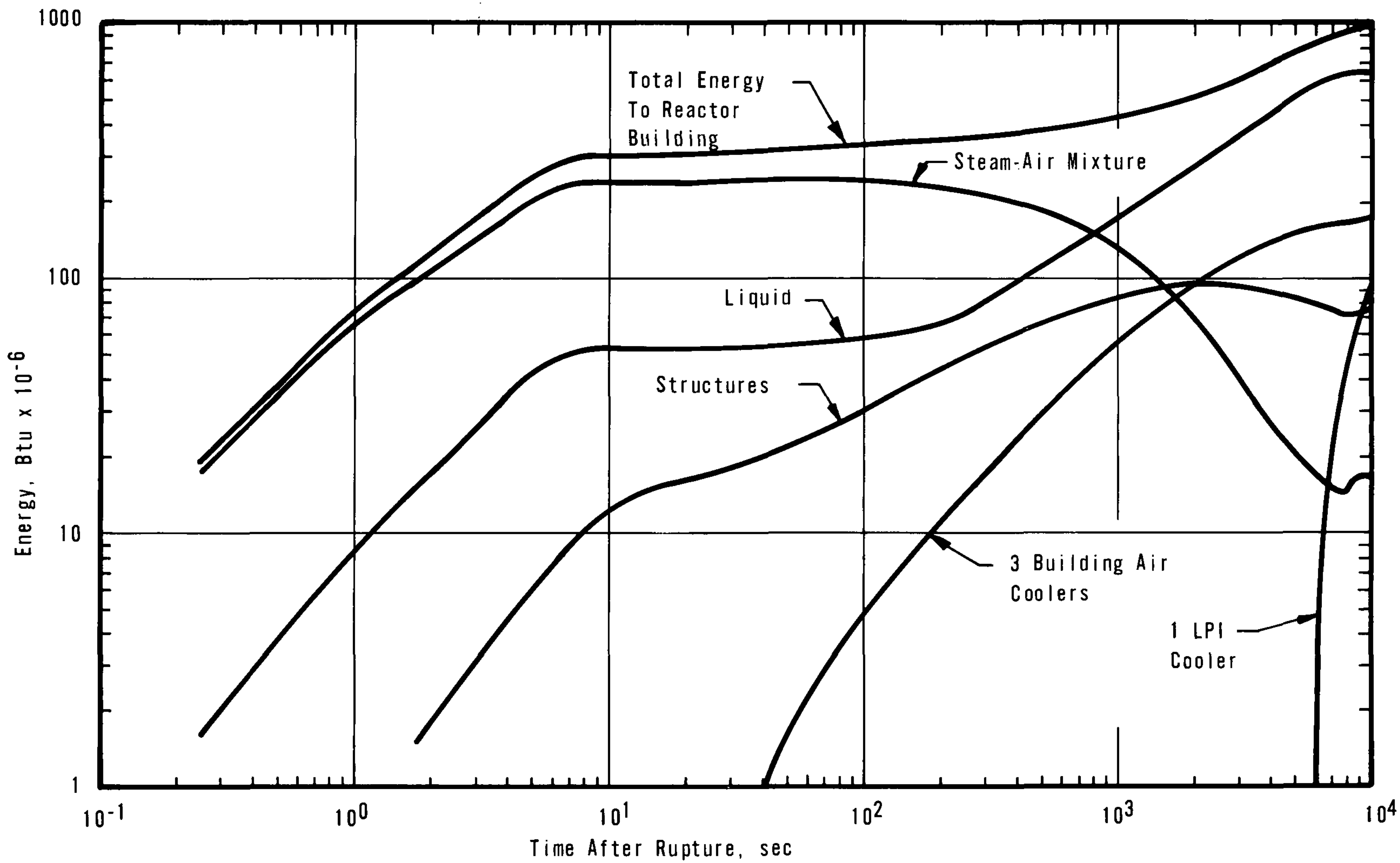
Rev. 5 5/25/70



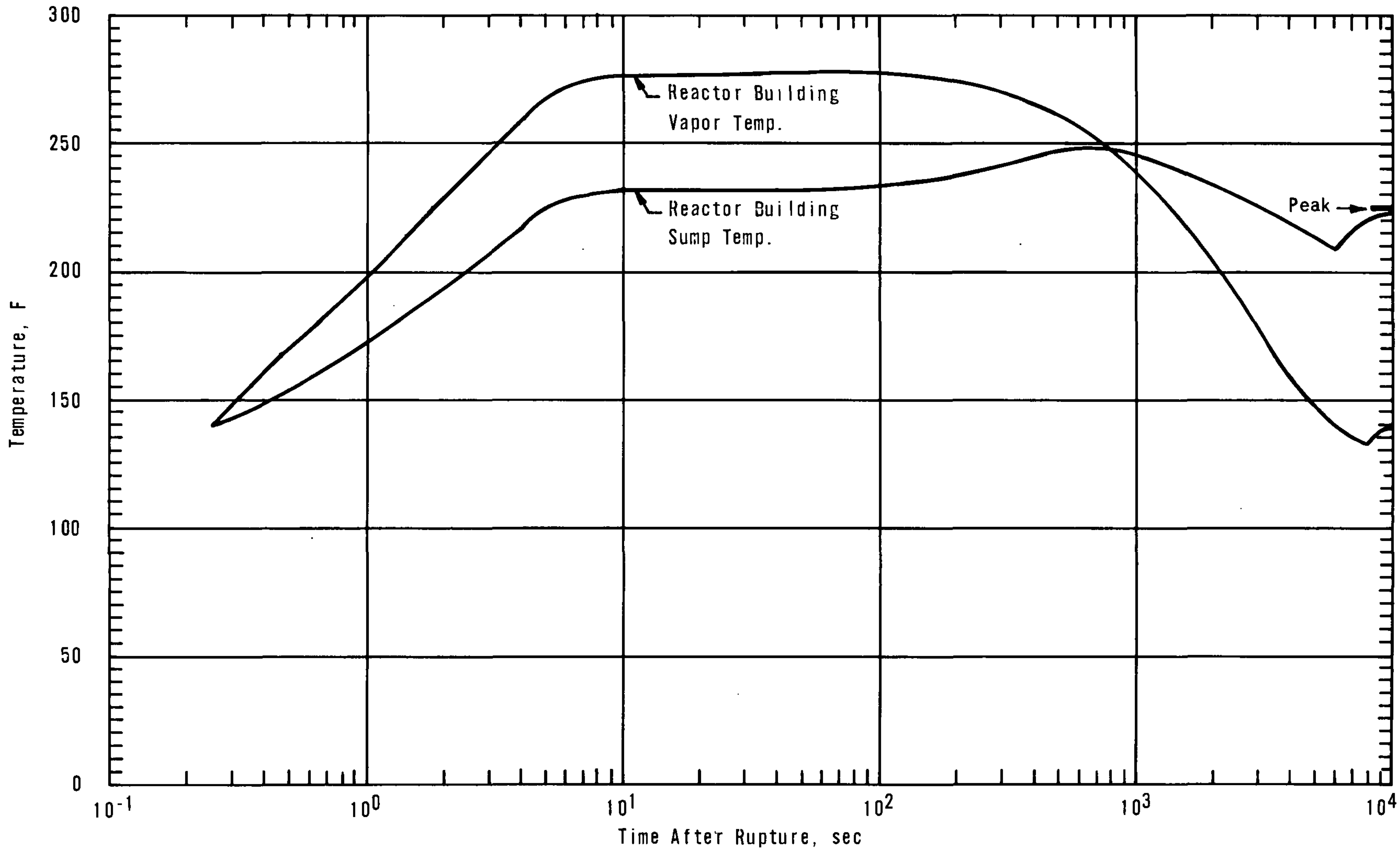
REACTOR BUILDING PRESSURE VERSUS TIME AFTER RUPTURE (1.0 FT^2)

OCONEE NUCLEAR STATION
Figure 14 - 61
Rev. 5 5/25/70

OCONEE NUCLEAR STATION
Figure 14 - 62
Rev. 5 5/25/70



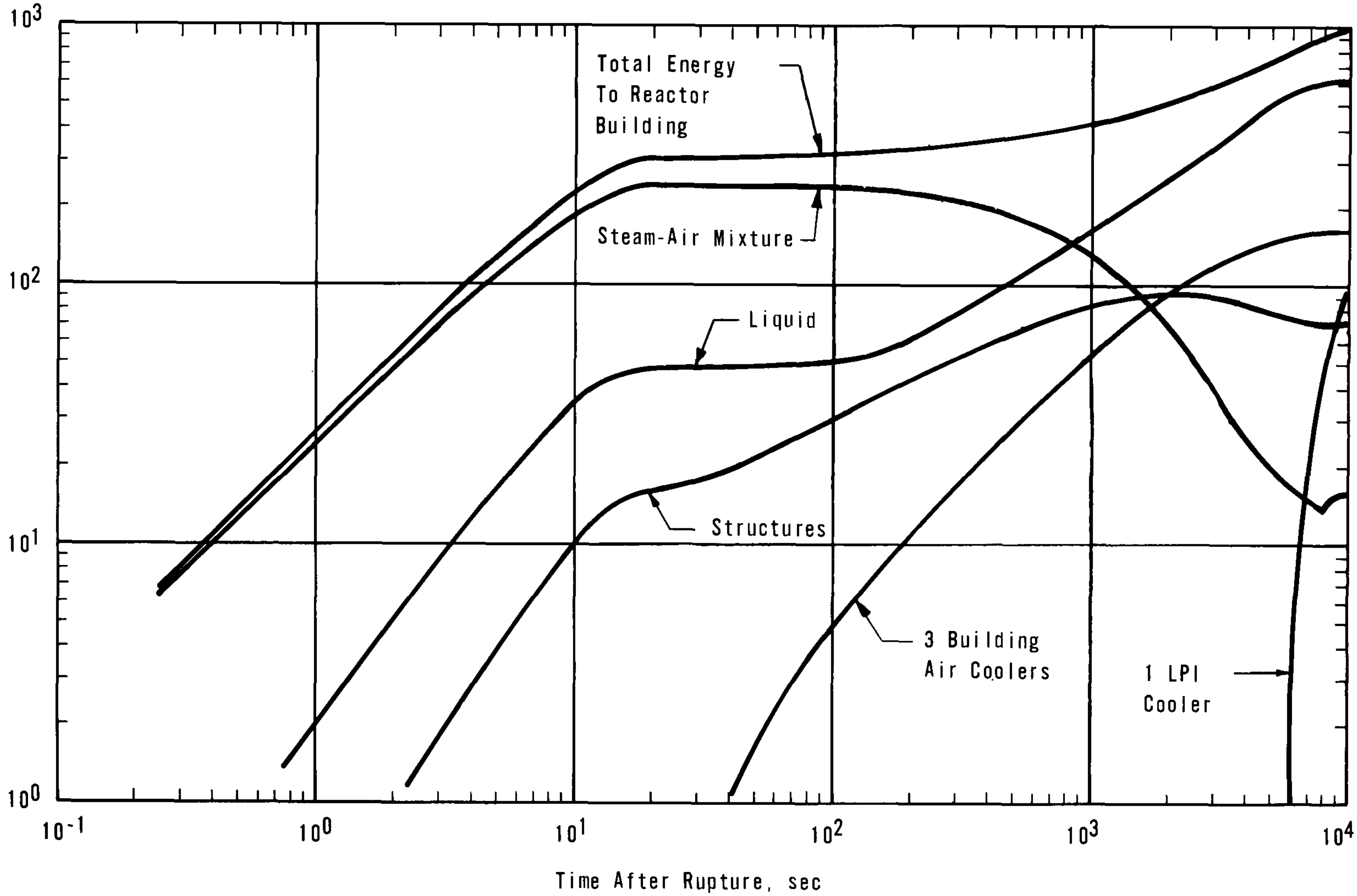
REACTOR BUILDING ENERGY INVENTORY FOR A 36-IN ID, DOUBLE-ENDED, PIPE RUPTURE WITH 3 REACTOR BUILDING AIR COOLERS



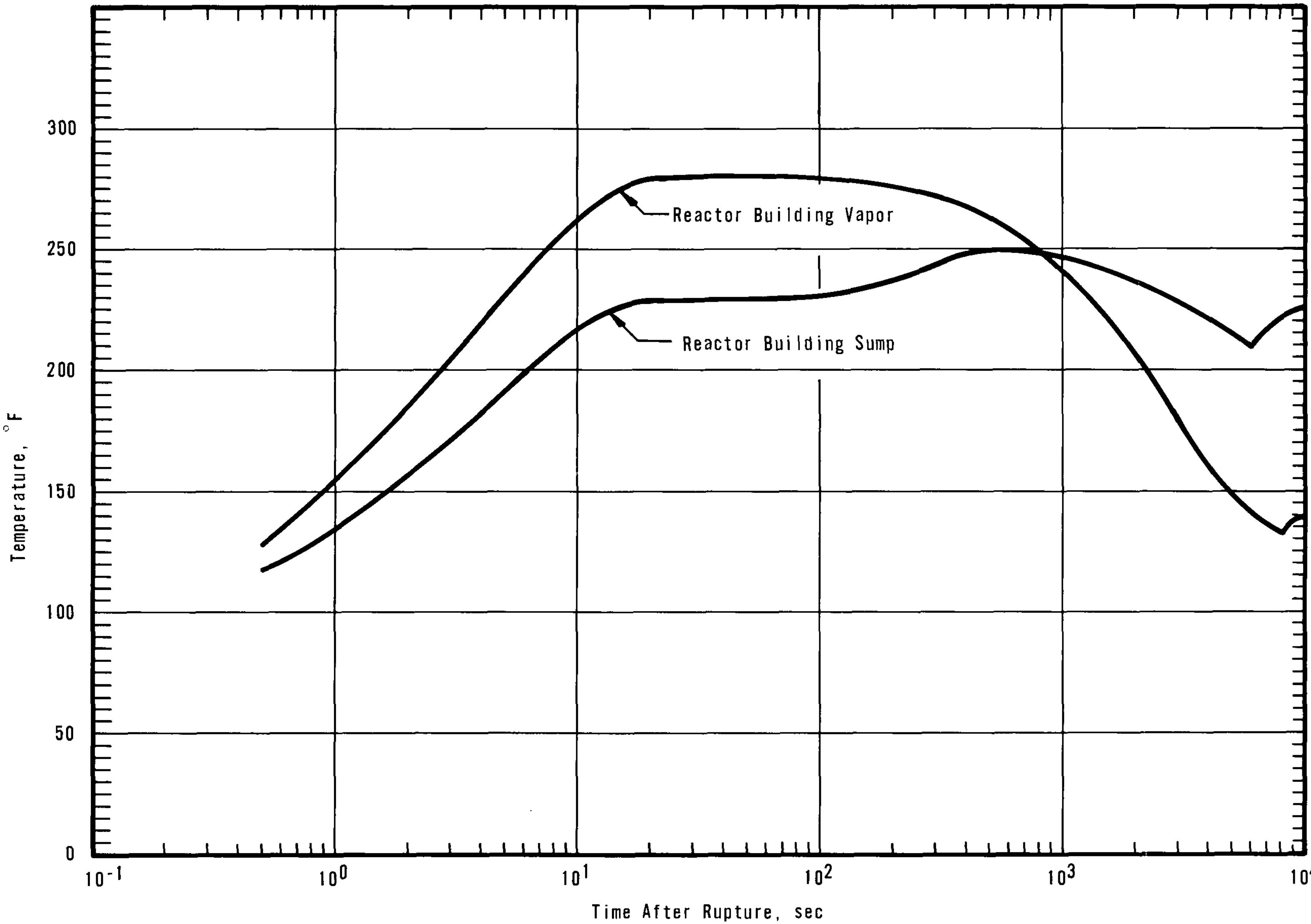
REACTOR BUILDING VAPOR AND SUMP COOLANT TEMPERATURES VERSUS TIME AFTER A 36-IN ID, DOUBLE-ENDED, PIPE RUPTURE WITH 3 REACTOR BUILDING AIR COOLERS

OCONEE NUCLEAR STATION
Figure 14 - 63
Rev. 5 5/25/70

OCONEE NUCLEAR STATION
Figure 14 - 63A
(New) Rev. 5 5/25/70



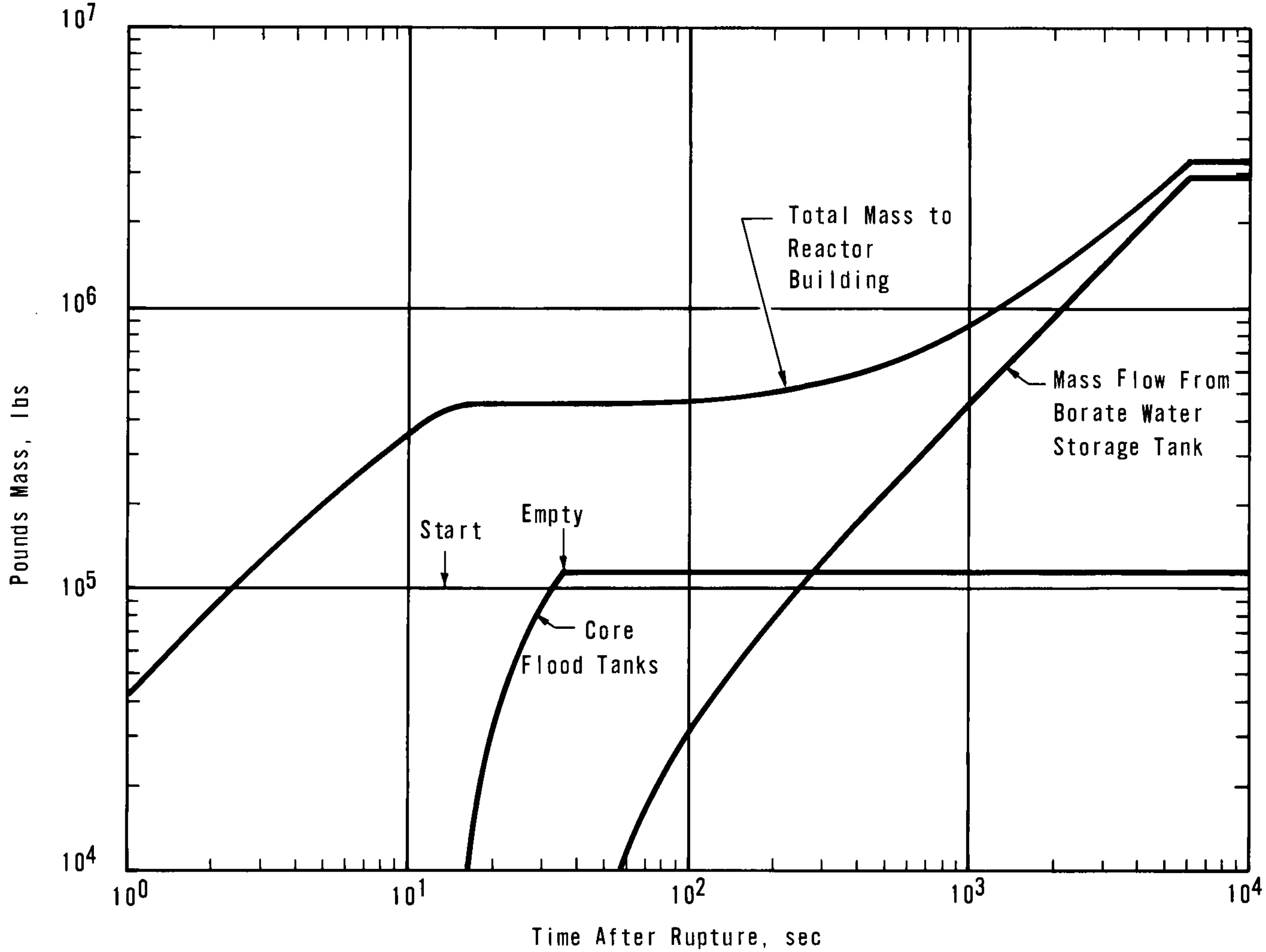
REACTOR BUILDING ENERGY INVENTORY FOR A 5 FT^2 RUPTURE OF THE REACTOR COOLANT SYSTEM PIPING WITH 3 REACTOR BUILDING AIR COOLERS



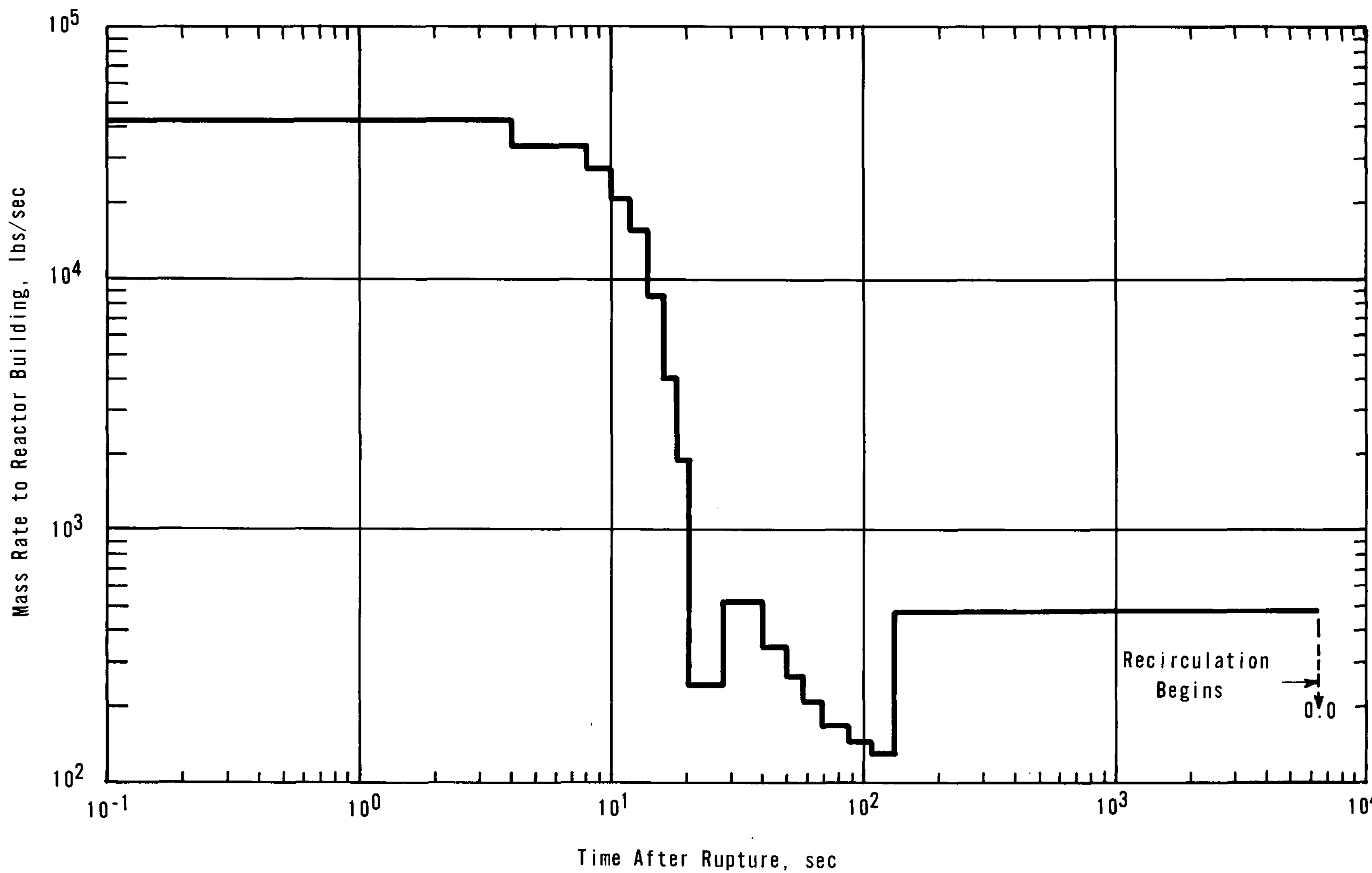
REACTOR BUILDING VAPOR AND SUMP COOLANT TEMPERATURES VERSUS TIME AFTER A 5 FT^2 RUPTURE OF THE REACTOR COOLANT SYSTEM WITH 3 REACTOR BUILDING AIR COOLERS

OCONEE NUCLEAR STATION
Figure 14 - 63B
(New) Rev. 5 5/25/70

OCONEE NUCLEAR STATION
Figure 14 - 63C
(New)Rev. 5 5/25/70



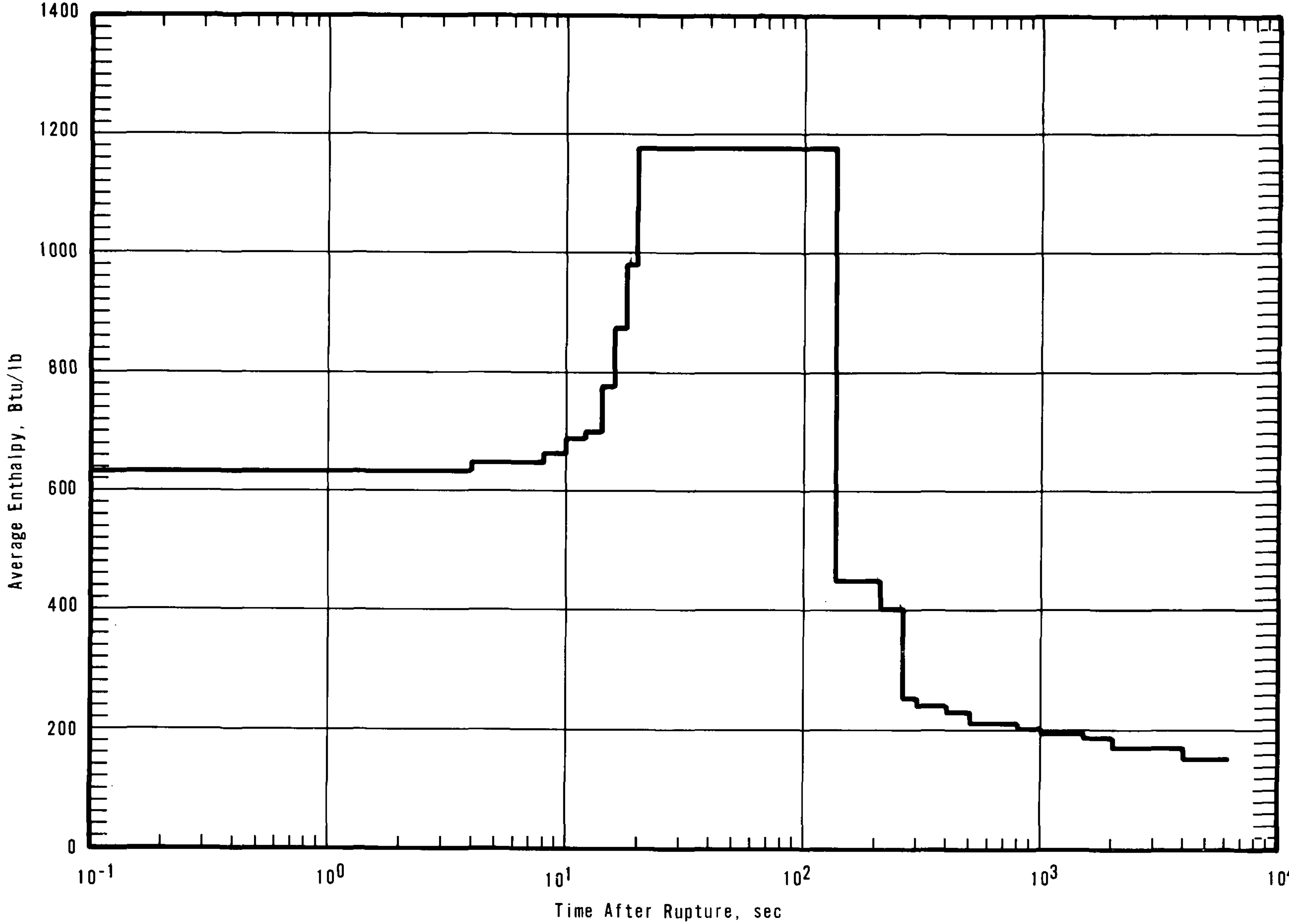
MASS VERSUS TIME RELEASED TO THE REACTOR BUILDING AND LEAVING THE CORE FLOODING TANKS AND THE BORATE WATER STORAGE TANK FOR A 5 FT^2 RUPTURE OF THE REACTOR COOLANT SYSTEM PIPING



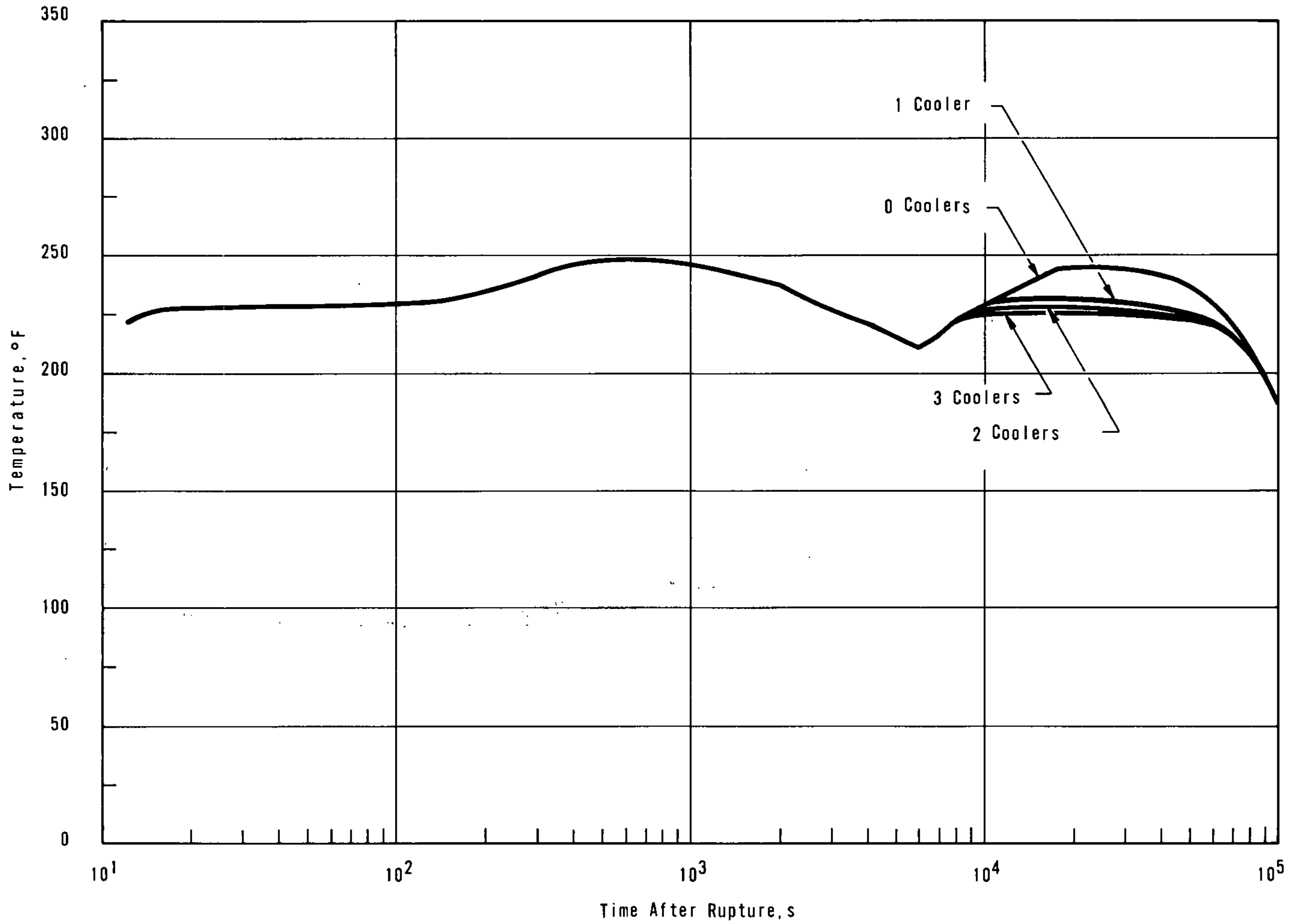
MASS ADDITION RATE TO THE REACTOR BUILDING VERSUS TIME FOR A 5.0 FT^2 RUPTURE OF THE REACTOR COOLANT SYSTEM PIPING

OCONEE NUCLEAR STATION
Figure 14 - 63D
(New) Rev. 5 5/25/70

OCONEE NUCLEAR STATION
Figure 14 - 63E
(New) Rev. 5 5/25/70

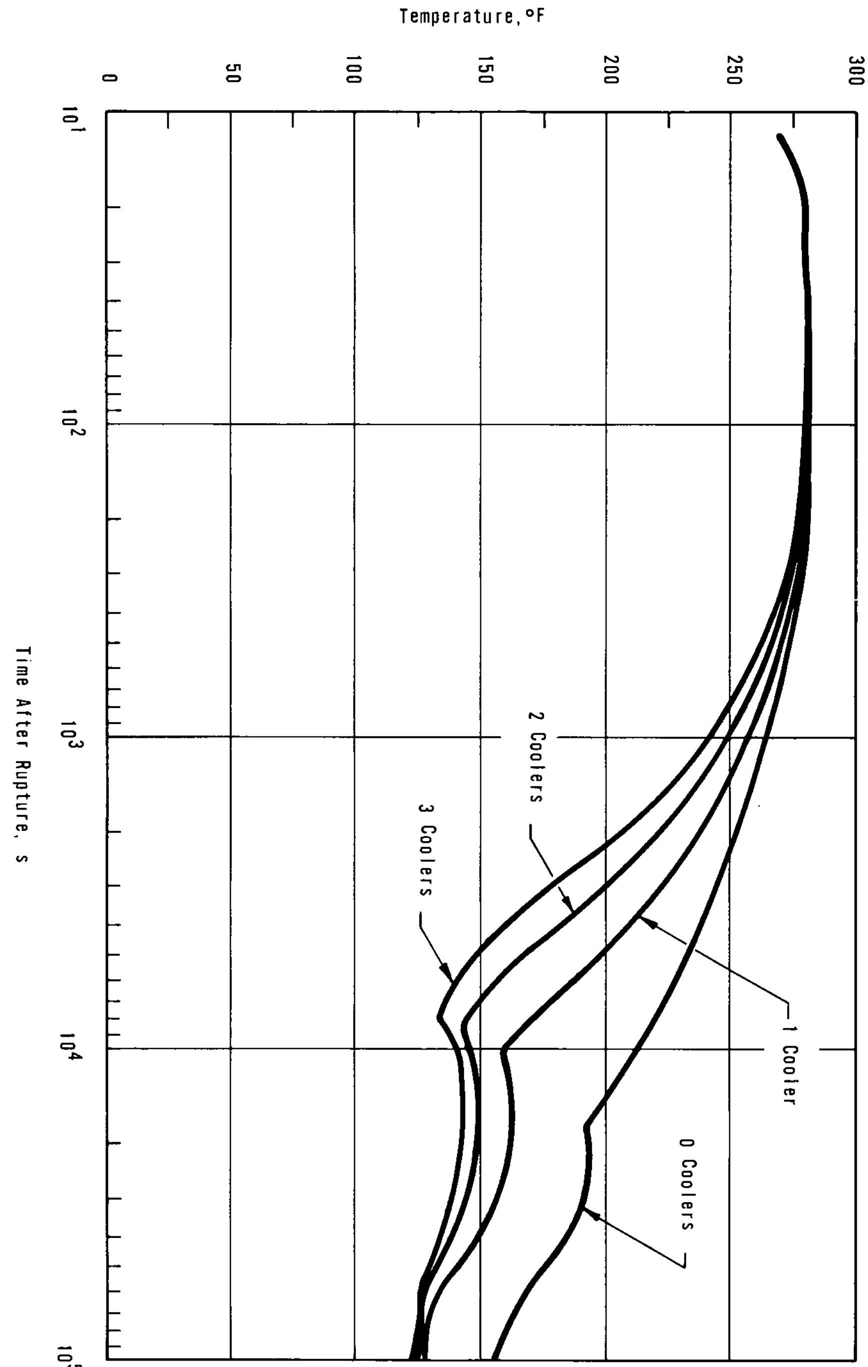


THE ENTHALPY ASSOCIATED WITH THE MASS RELEASED TO THE REACTOR BUILDING AS A FUNCTION OF TIME FOR A 5.0 FT^2 RUPTURE OF THE REACTOR COOLANT SYSTEM PIPING



VARIATION OF REACTOR BUILDING SUMP TEMPERATURE VERSUS TIME FOR DBA 5 FT^2 BREAK WITH 3, 2, 1 OR 0 REACTOR BUILDING AIR COOLERS

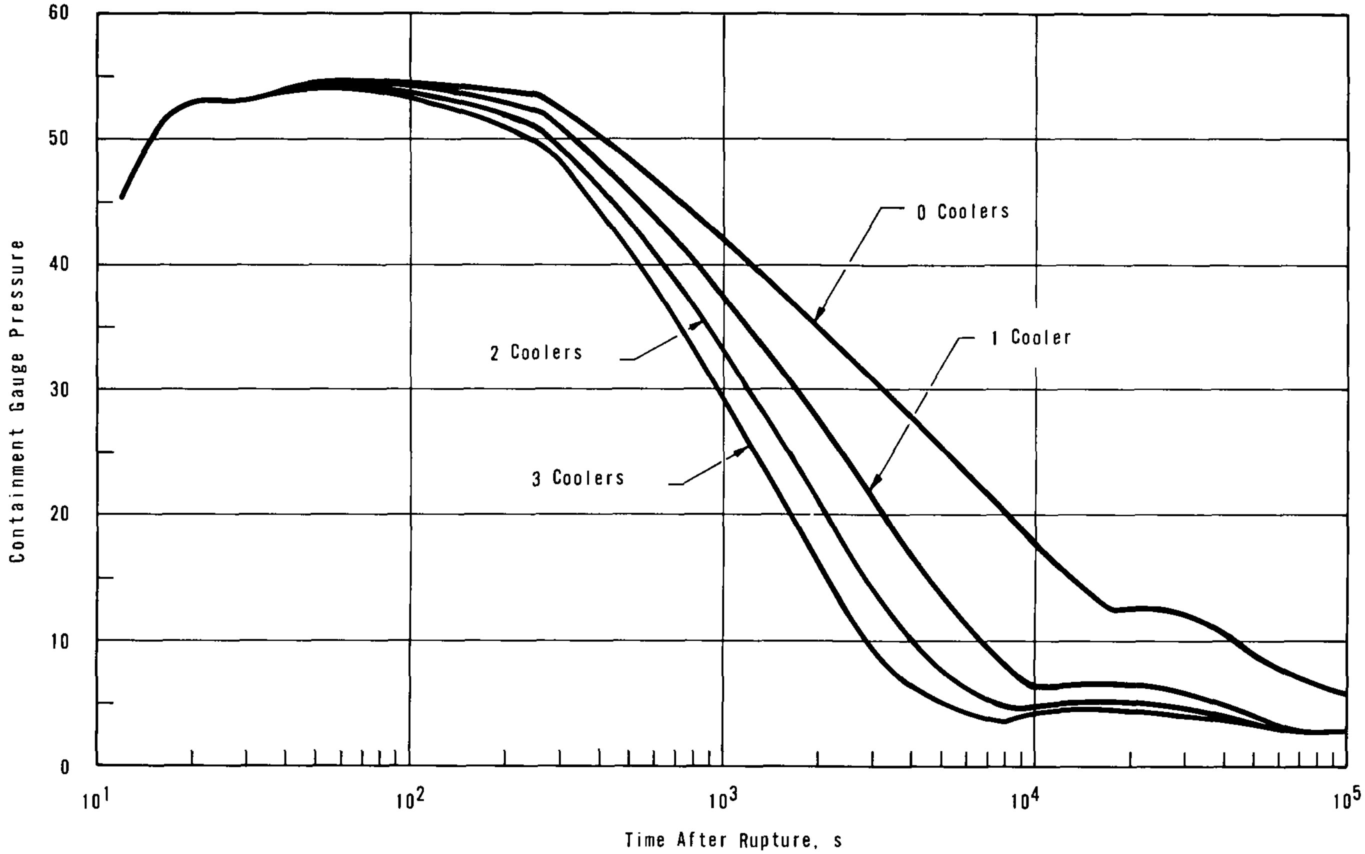
OCONEE NUCLEAR STATION
Figure 14 - 63f
Rev. 19 5/5/72



VARIATION OF REACTOR BUILDING VAPOR TEMPERATURE VERSUS TIME FOR THE DBA 5 FT^2 BREAK WITH 3, 2, 1 OR 0 REACTOR BUILDING AIR COOLERS

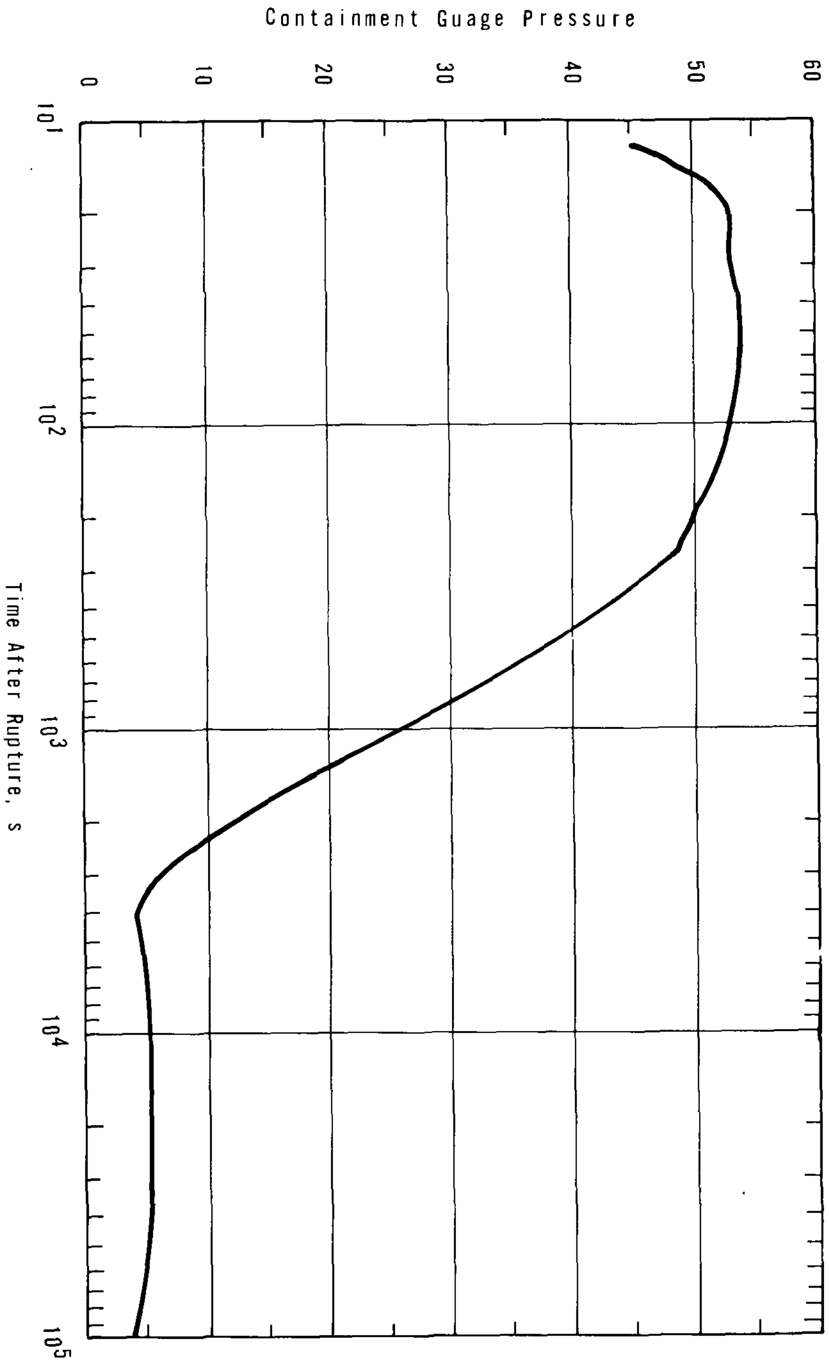
DUKE POWER

OCONEE NUCLEAR STATION
Figure 14 - 63g
Rev. 19 5/5/72



VARIATION OF REACTOR BUILDING PRESSURE VERSUS TIME FOR THE DBA 5 FT^2 BREAK WITH 3,2,1 OR 0 REACTOR BUILDING AIR COOLERS

OCONEE NUCLEAR STATION
Figure 14 - 63h
Rev. 19 5/5/72

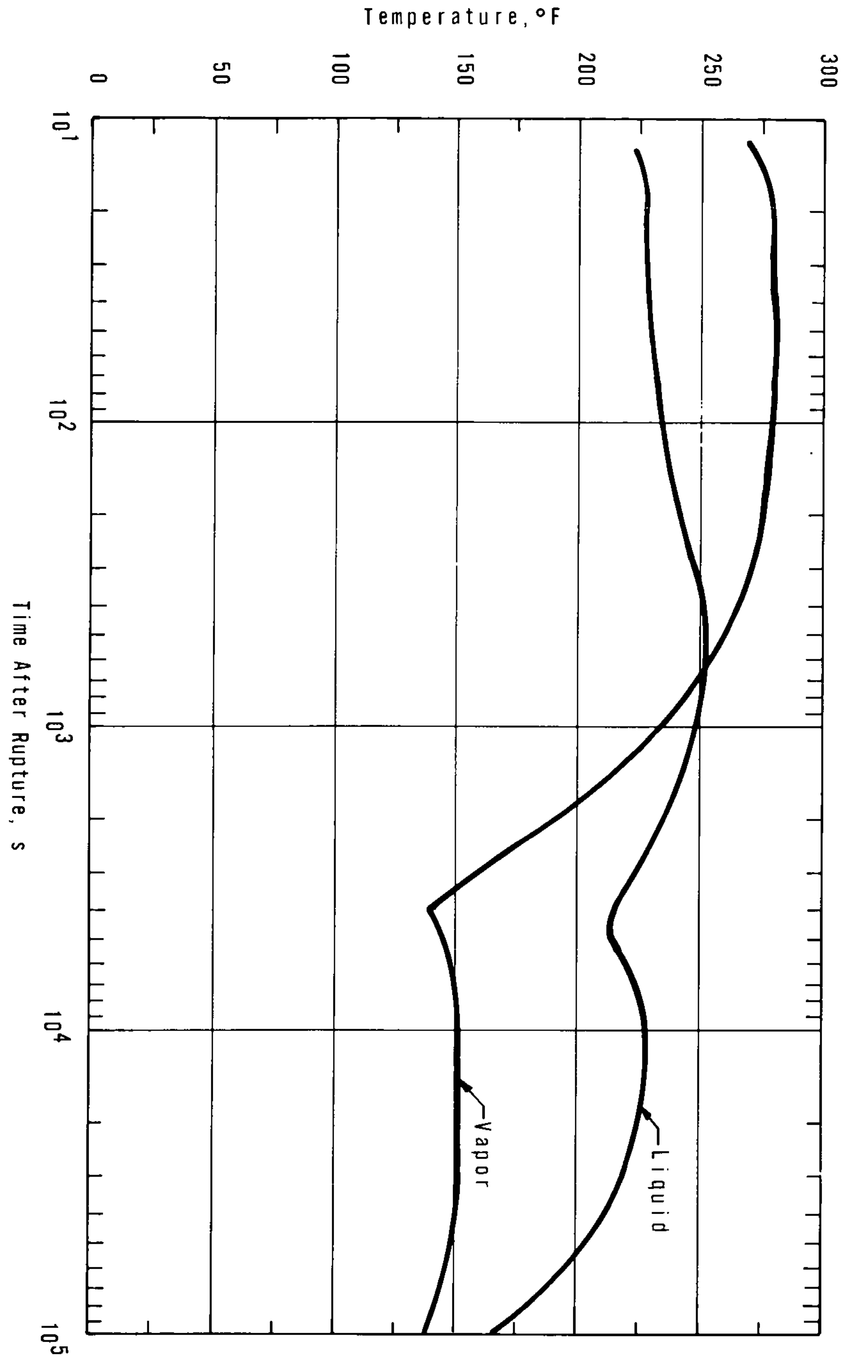


REACTOR BUILDING PRESSURE VERSUS TIME FOR THE DBA 5 FT² BREAK WITH 2 REACTOR BUILDING AIR COOLERS AND 1 1500 GPM SPRAY (1000 GPM SPRAY DURING RECIRCULATORY PERIOD)

OCONEE NUCLEAR STATION

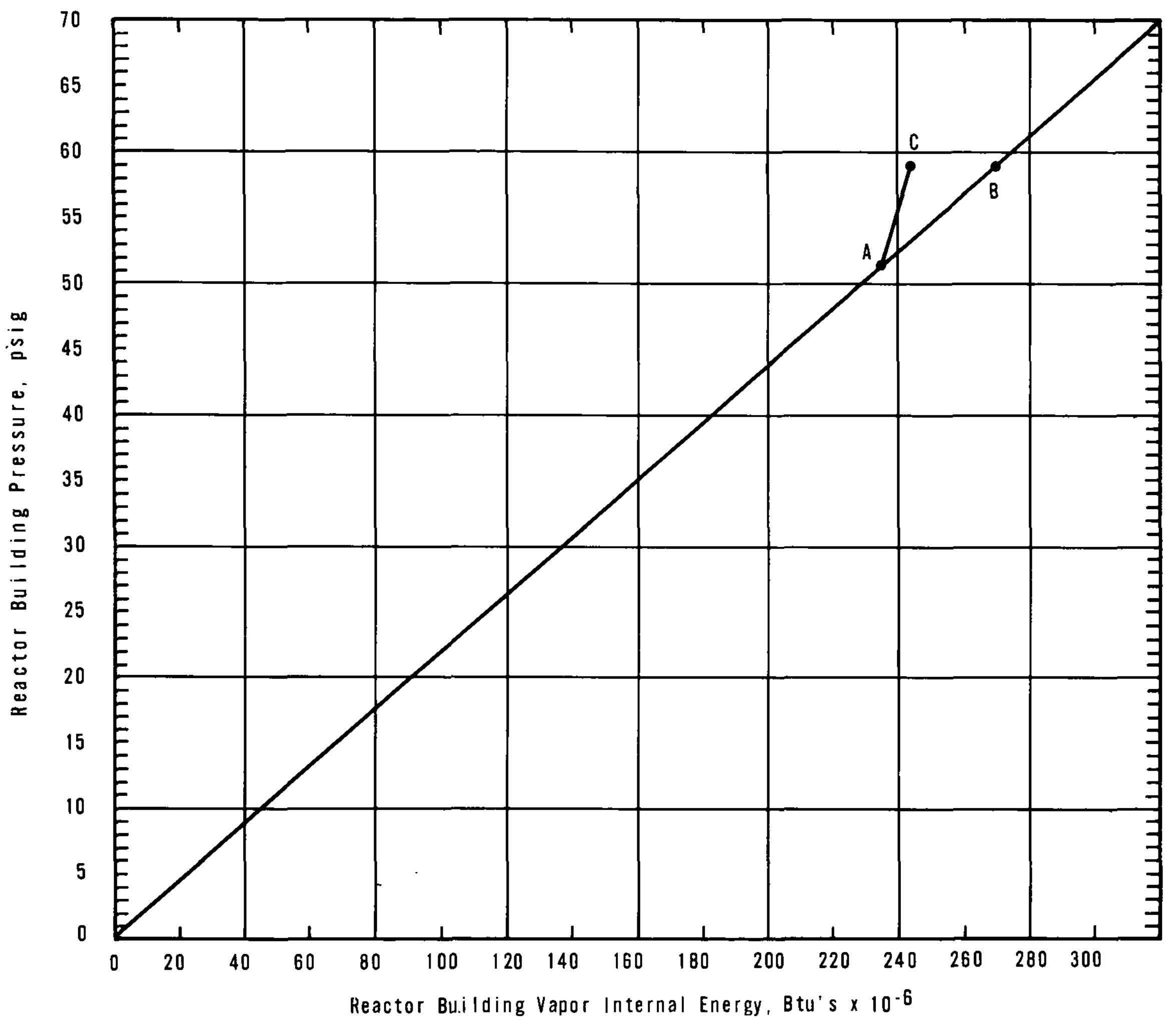
Figure 14 - 63i

Rev. 19 5/5/72

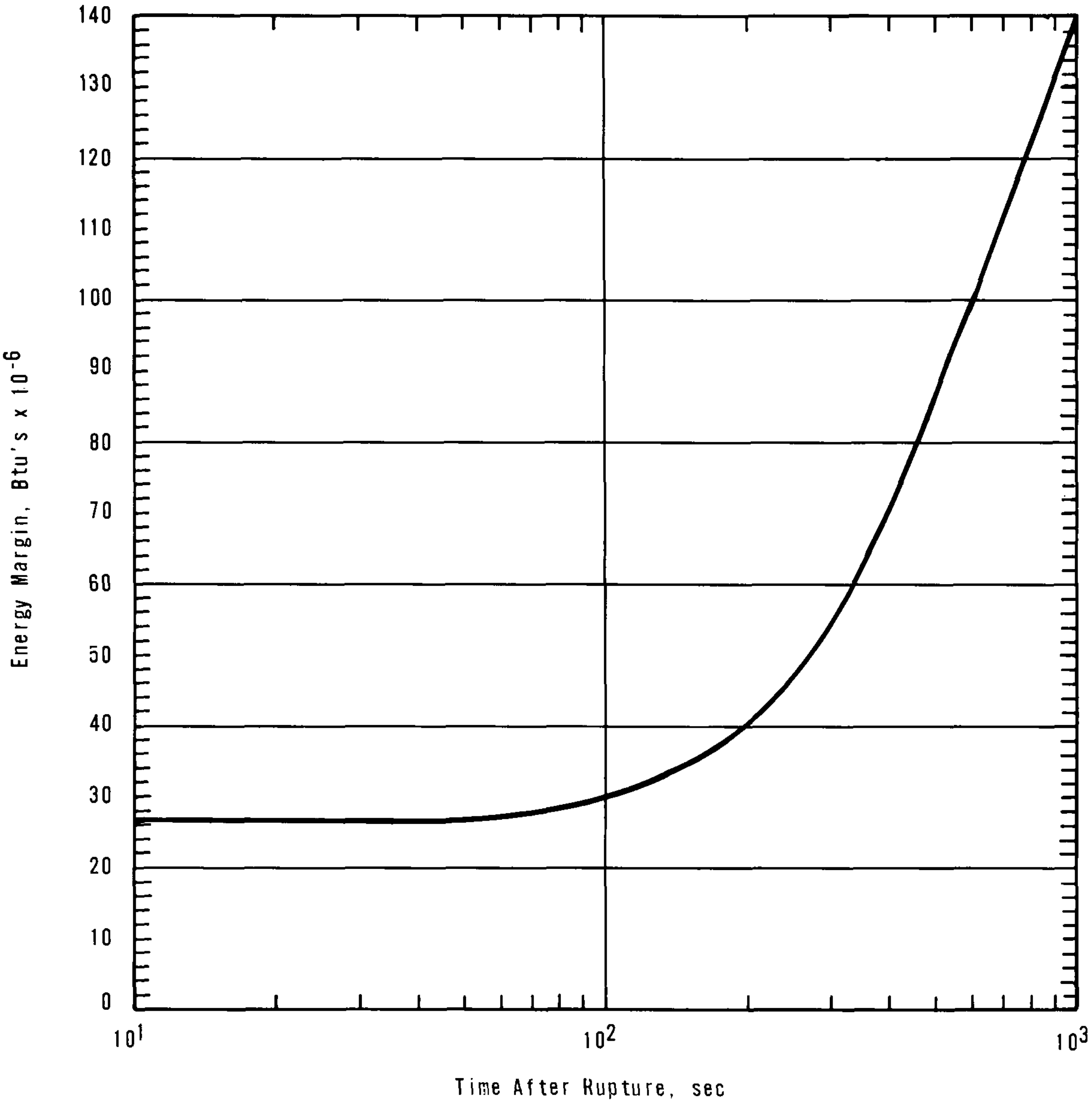


REACTOR BUILDING SUMP AND VAPOR TEMPERATURE VERSUS TIME FOR THE DBA 5 FT^2 BREAK WITH 2 REACTOR BUILDING AIR COOLERS AND 1 1500 GPM SPRAY (1000 GPM SPRAY DURING RECIRCULATORY PERIOD)

OCONEE NUCLEAR STATION
Figure 14 - 63j
Rev. 19 5/5/72



REACTOR BUILDING PRESSURE VERSUS INTERNAL ENERGY OF BUILDING VAPOR REGION

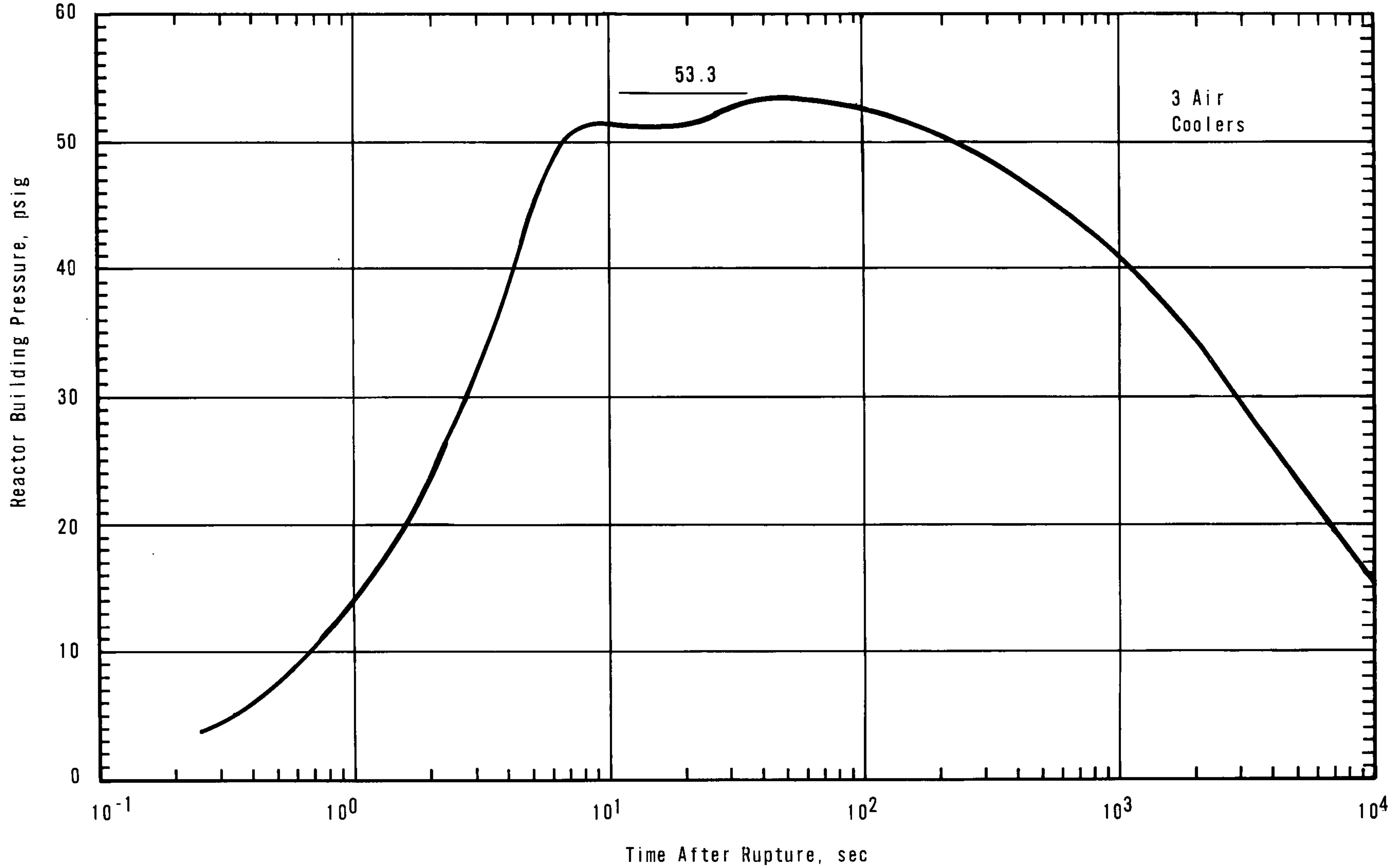


REACTOR BUILDING ENERGY MARGIN VERSUS TIME FOR A 36-IN. ID, DOUBLE-ENDED, PIPE RUPTURE WITH 3 REACTOR BUILDING AIR COOLERS

OCONEE NUCLEAR STATION

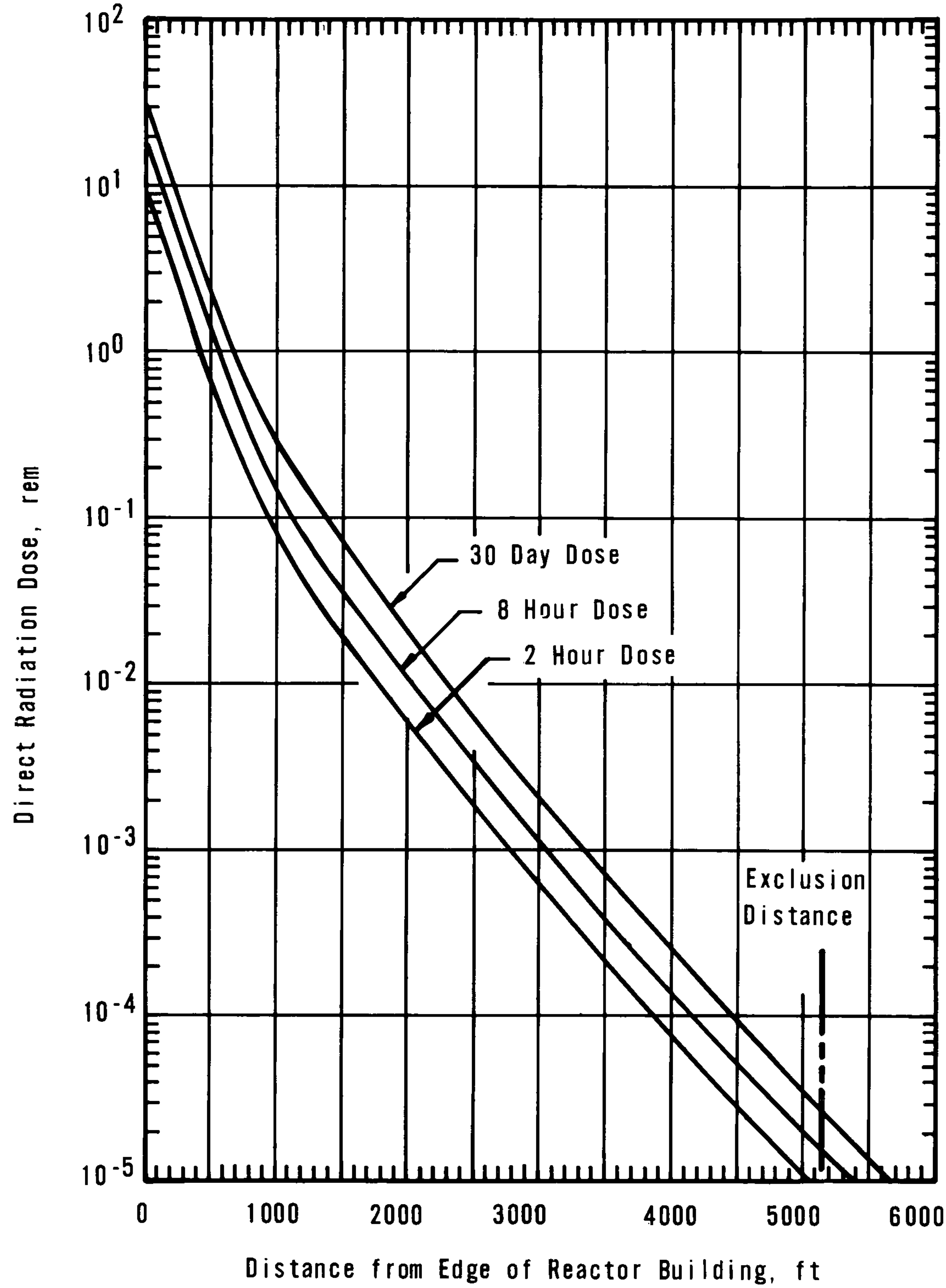
Figure 14 - 65

Rev. 5 5/25/70

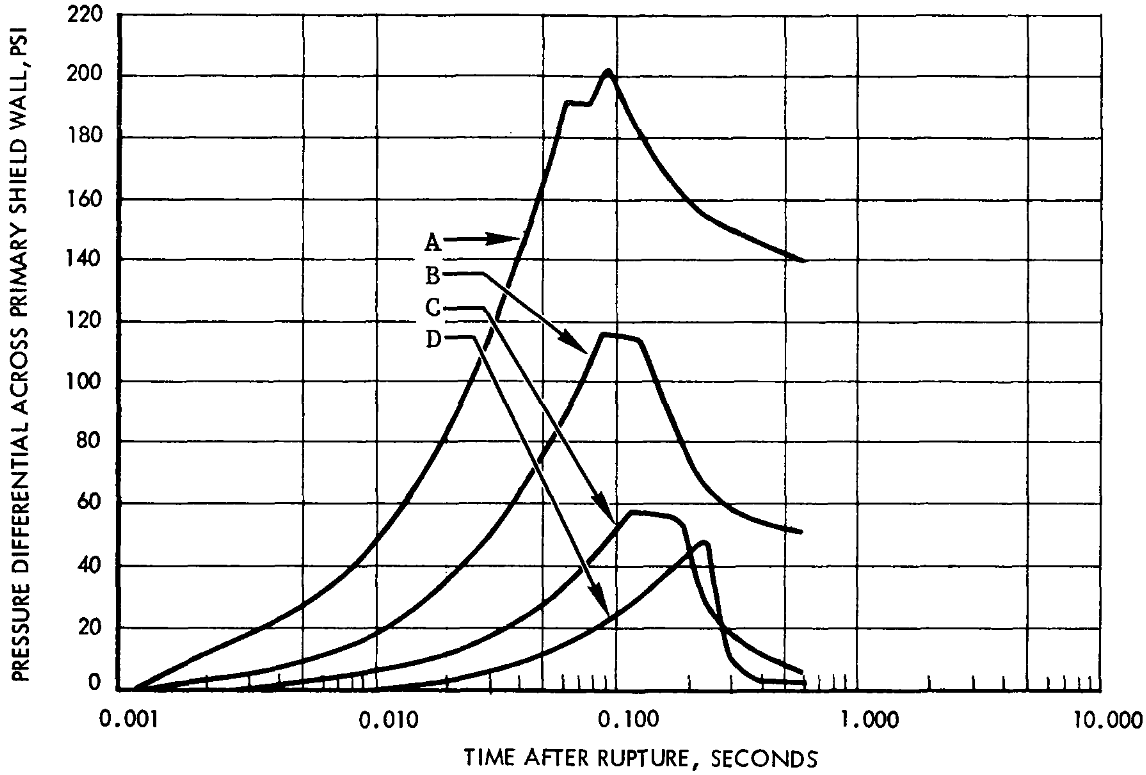


REACTOR BUILDING PRESSURE VERSUS TIME FOR A 36-IN ID, DOUBLE-ENDED, PIPE RUPTURE AND CONTINUOUS STEAM RELEASE WITH 3 REACTOR BUILDING AIR COOLERS

OCONEE NUCLEAR STATION
Figure 14 - 66
Rev. 5 5/25/70

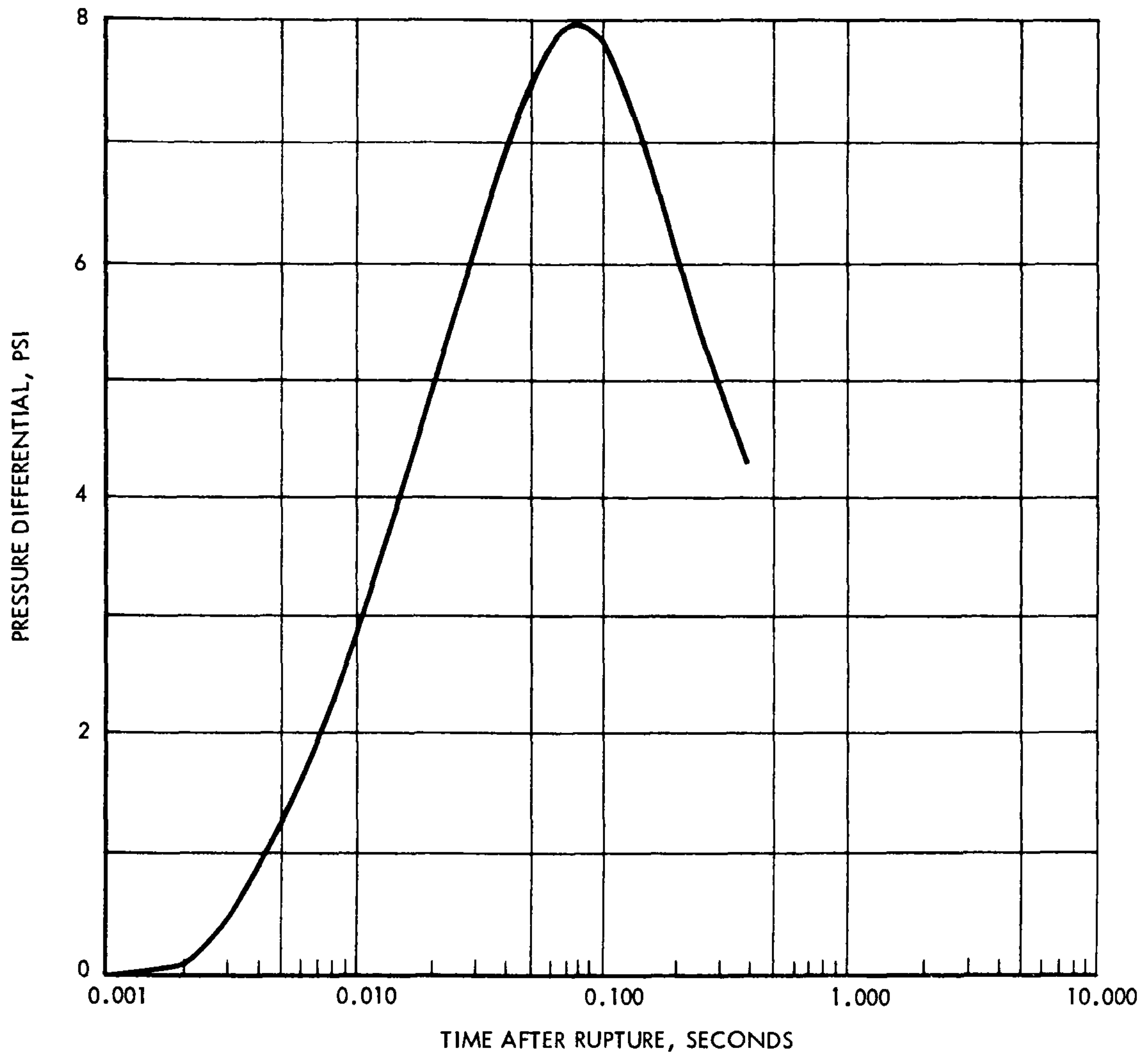


INTEGRATED DIRECT DOSE FOLLOWING MHA WITH 3.75 FT. REACTOR BUILDING WALL THICKNESS

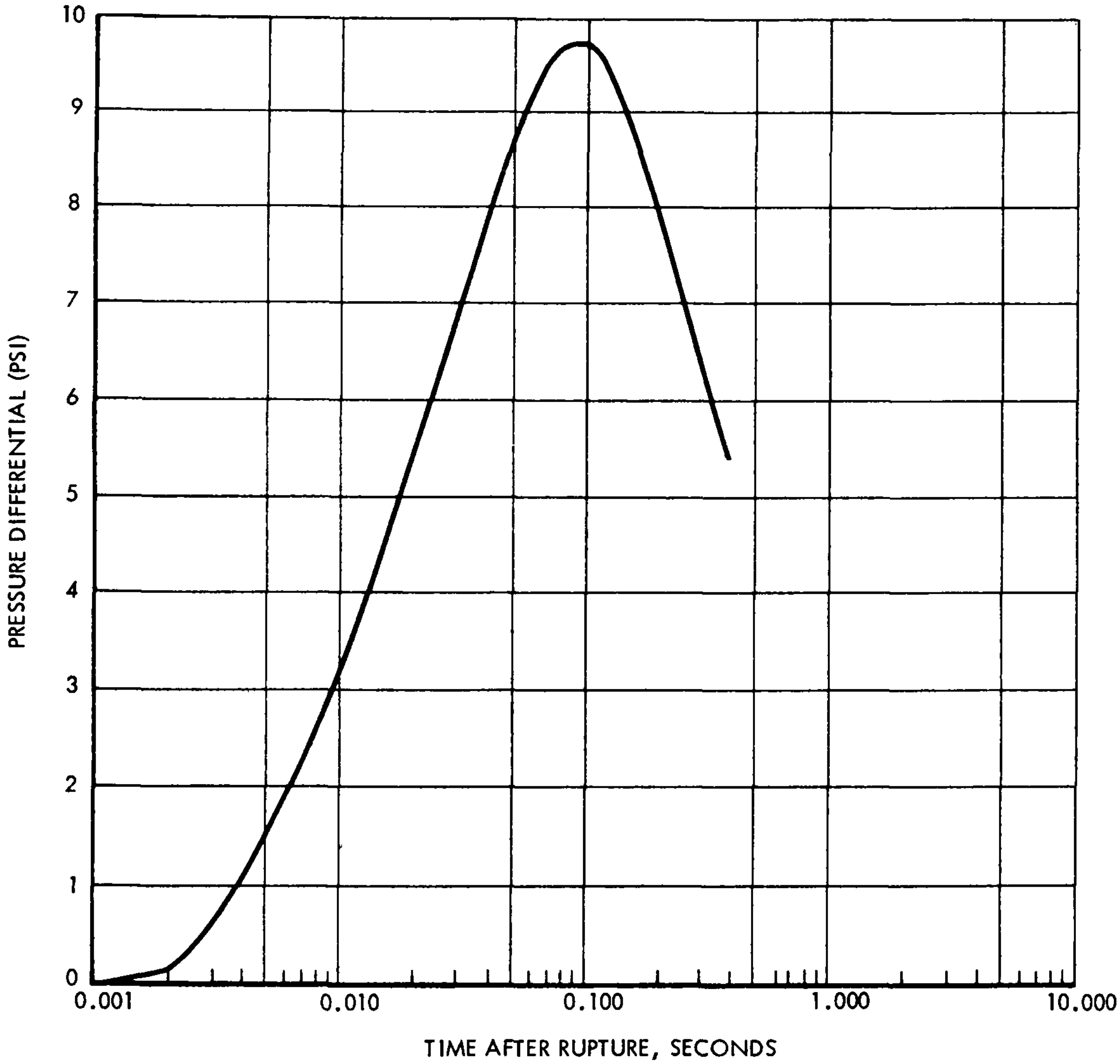


CURVE	BREAK SIZE*	DESCRIPTION
A	8.5 sq. ft.	Corresponds to maximum break size reactor cavity can withstand
B	3 sq. ft.	Corresponds to maximum hot leg break size possible within the reactor cavity
C	1 sq. ft.	1.0 square foot hot leg rupture
D	0.4 sq. ft.	0.4 square foot hot leg rupture
See Figures 14-71 and 14-72 for description of blowdown data used		

Figure 14-68 Pressure Transients for Range of Rupture Sizes Within Reactor Cavity.



Pressure Differential Across West Compartment Walls Versus Time After Double-Ended Hot-Leg Rupture (Case 1-14 Square Feet).

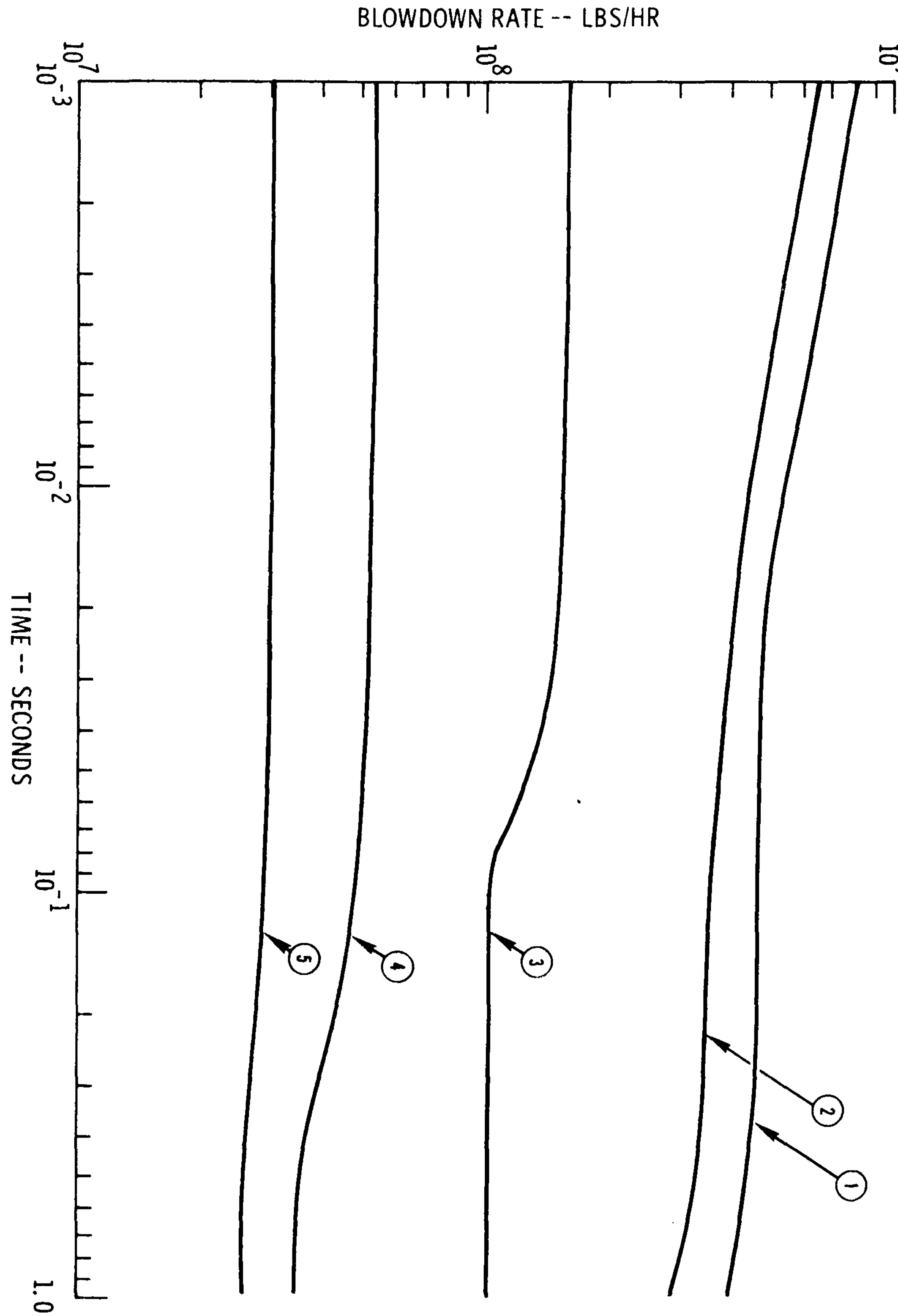


Pressure Differential Across East Compartment Walls Versus Time After Double-Ended Hot-Leg Rupture (Case 1-14 Square Feet).

OCONEE NUCLEAR STATION

Figure 14 - 70

(New) Rev. 5 5/25/70



Mass Flow Rate for Babcock and Wilcox
Blowdown Data

Note: For Key to Blowdown Data see Figure 14-72

OCONEE NUCLEAR STATION

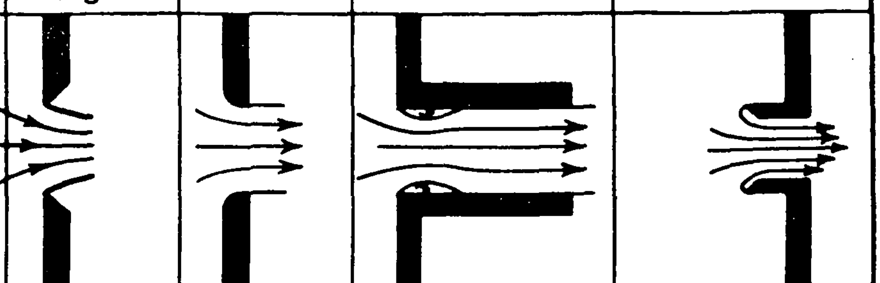
Figure 14 - 71

(New) Rev. 5 5/25/70

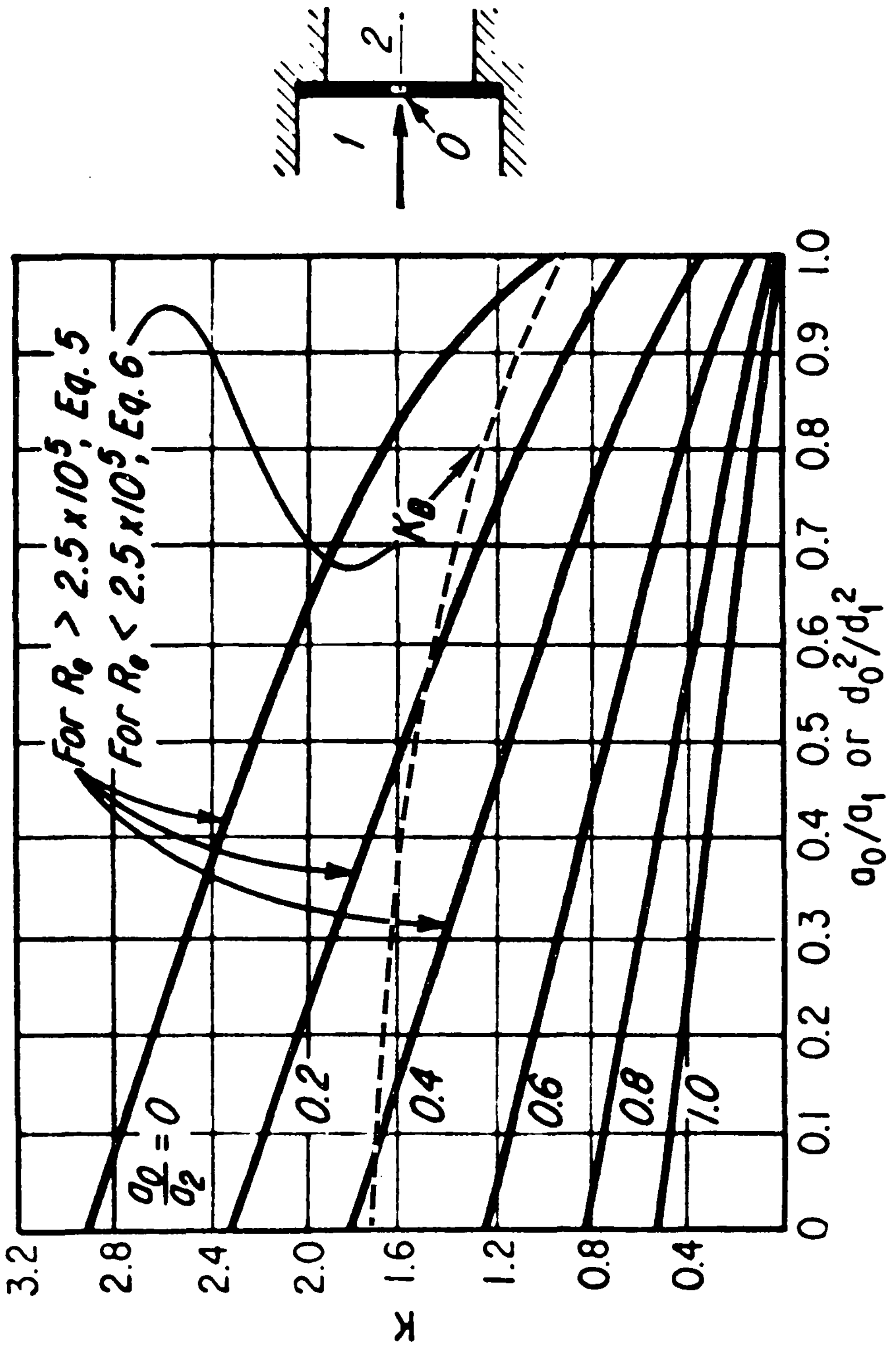
Table 1

KEY TO BLOWDOWN DATA

Case No.	Explanation
1	36-inch double-ended rupture, hot-leg
2	28-inch double-ended rupture, cold-leg
3	3-sq ft rupture in 36-inch hot-leg
4	1-sq ft rupture in 36-inch hot-leg
5	0.4-sq ft rupture in 36-inch hot-leg

	Sharp edged	Rounded	Short tube	Borda
				
C	0.61	0.98	0.80	0.51

Orifice Coefficient Data

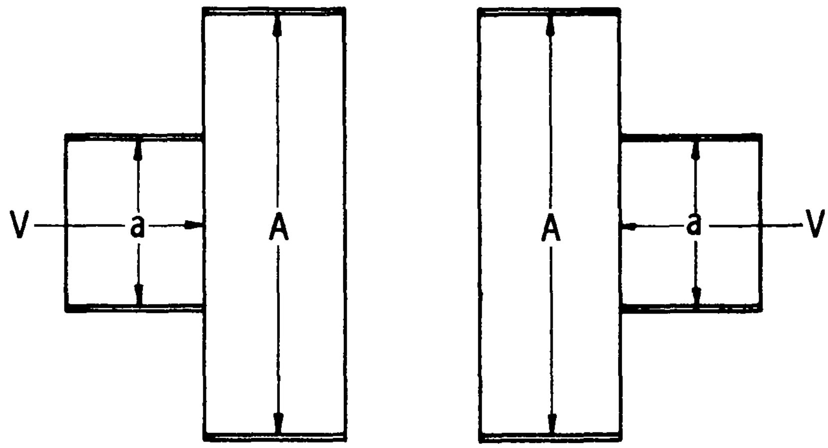


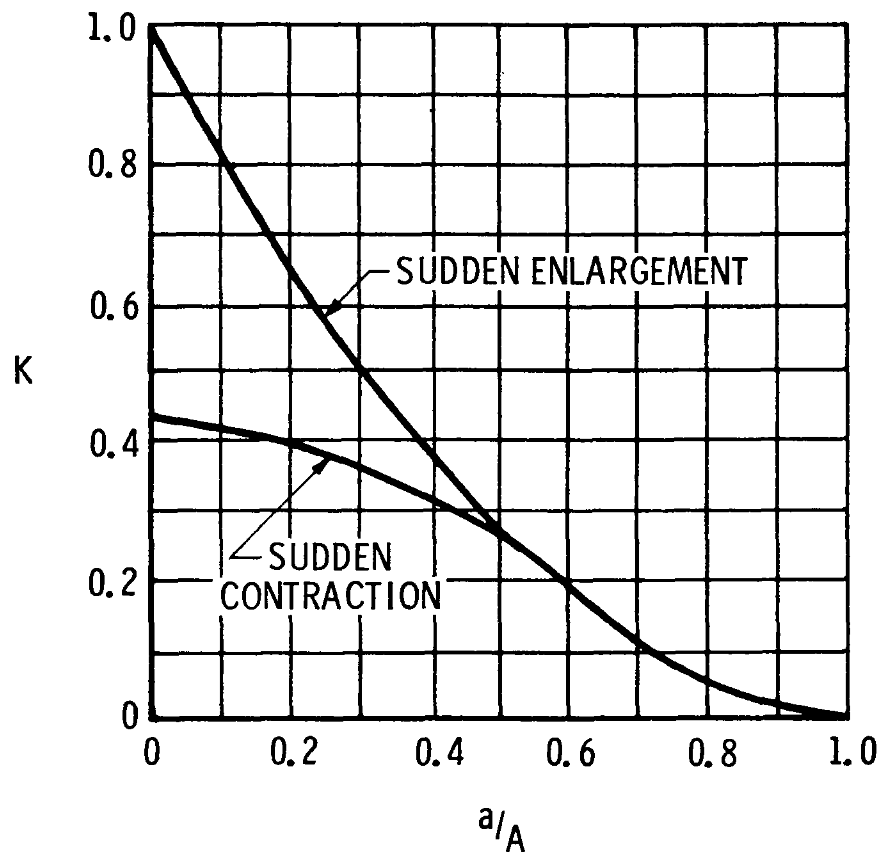
Head Loss Coefficient Data: Pipe
Diameter Change, Orifice

OCONEE NUCLEAR STATION

Figure 14 - 74

(New) Rev. 5 5/25/70





Head Loss Coefficient Data: Sudden Change in Section

APPENDIX 14A

AN EVALUATION OF PURGING
AS A MEANS OF CONTROLLING POST-ACCIDENT
REACTOR BUILDING HYDROGEN CONCENTRATION

Submitted With FSAR Revision No. 4

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Post-Accident Hydrogen Control

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14A AN EVALUATION OF PURGING AS A MEANS OF CONTROLLING POST-ACCIDENT REACTOR BUILDING HYDROGEN CONCENTRATION

14A.1 INTRODUCTION

The purpose of this appendix is to summarize the analyses performed to:

A. Evaluate the hazard caused by hydrogen generation following a loss-of-coolant accident (LOCA).

B. Evaluate the acceptability of controlled post-accident purging as a method for controlling the reactor building hydrogen concentration.

Those processes which determine the hydrogen generation rates, the reactor building hydrogen concentrations and the site doses are graphically described on Figure 14A.1.1. In this appendix the potential for radiolytic hydrogen generation including the dose, or energy deposited in the coolant following the accident, and the basis for the selection of the hydrogen generation constant ("G" value) is analyzed. Since the FSAR adequately analyzes the potential zircaloy-water reaction, this analysis is not presented herein and a 1% zirc-water reaction (greater than 10 times the calculated value) is assumed in the reference case described in subsequent sections. The potential for hydrogen generation from a zinc-boric acid reaction when borated water spray solution contacts galvanized steel in the reactor building at the post-accident temperature is also considered. The analysis shows that the radiolytic hydrogen generation rate plus the hydrogen contributed by the zircaloy and other reactions does not result in unacceptable hydrogen concentrations until 780 hours after the initation of the LOCA. It is this long period of time that allows reactor building purging to be considered as an effective means for controlling the hydrogen concentration. The analysis shows that low purge rates less than twice
18. the containment design leakage rate of .25 v/o per day can adequately control the hydrogen concentration following a loss-of-coolant accident (LOCA).

A primary concern in evaluating the acceptability of purging as a means of controlling the post-accident reactor building hydrogen concentration is the release of radioactive material to the atmosphere and the resulting radiation doses to the public. The analysis shows that incremental increases in doses at the low population boundary (LPZ) are small because of the low purge rates required, and because of the long period of time available before purging is necessary. This allows time for radioactive decay of isotopes in the reactor building atmosphere and for the use of long-term meteorology. These time-affected areas are shown on Figure 14A.1.1.

The fact that some radioactivity would be released by purging under carefully controlled conditions following the improbable event of a LOCA is not in itself unacceptable. Following such an accident it would be necessary at some point in time to open the containment and purge the atmosphere to lower the rare gas activities so that post-accident recovery and cleanup could proceed. This would occur regardless of the proposed hydrogen control-method.

14A.2 SUMMARY AND CONCLUSIONS

The analyses show that hydrogen will be generated in the reactor building following a LOCA, but that the hydrogen concentration can be adequately controlled utilizing reactor building purging. It is demonstrated that incremental dose increases due to purging are small. The basic assumptions and results of these analyses are summarized as follows:

A. The control limit at which purging would begin is conservatively established at 3.5 v/o. The margin between this value and the lower flammability limit of 4.1% allows for measurement error, poor mixing and permits interruption of purging for significant periods of time.

B. The reference conditions for this analysis are: (1) a LOCA, (2) radiolytic hydrogen generation as shown in Figure 14A-3.5, (3) reaction between the boric acid spray solution and zinc in 46,500 ft^2 of galvanized steel, (4) 1% zircaloy-water reaction, and (5) a 90% iodine filtering efficiency when purging.

Purging begins when the hydrogen building concentration reaches 3.5 v/o which is approximately 5 weeks after a LOCA. Starting time is not a critical item because there is another 4 weeks before the hydrogen concentration would reach the lower flammability limit for a hydrogen-air mixture, and a much longer period of time before detonation limits are reached.

The results of the analyses are summarized in Table 14A-2.1 for purge starting times including one week prior to and following the reference case for the LOCA.
6. For comparison, the analysis has been extended to include the reference case for the Maximum Hypothetical Accident (MHA). From the doses shown it is evident that the exact starting time for purging for the reference conditions is not critical. For these conditions the lower flammability limit of hydrogen in air of 4.1 v/o is not exceeded even at the later starting times of 950 hours (40 days) after LOCA or at 790 hours (33 days) after MHA. (Figs. 14A-4.3, 14A-4.6).

The doses resulting from purging are not critically sensitive to the radiolytic hydrogen generation constant ("G" value) used in this analysis as shown in Figures 14A-7.10 and -7.11. The "G" value of 0.45 mols H_2/100 ev used in this analysis is adequately substantiated by existing experimental data.[2,3] However, increasing the "G" value by a factor of two (from 0.45 to 0.9 for the LOCA reference conditions) results in dose increases from purging of only 0.5 rem thyroid and 0.065 rem whole body at the LPZ. In like manner, an increase of 100% in the hydrogen "G" value for the MHA reference case would not result in excessive doses to the thyroid and whole body, 18 rem and 0.090 rem, respectively, when purging is accomplished so as not to exceed a 3.5 volume percent hydrogen concentration.

Doses are similarly shown not to be critically sensitive to the percent metal-water reaction over the range 0 - 10%. For LOCA a 10% metal-water reaction results in thyroid and whole body from purging of only 0.6 rem and 0.066 rem, respectively. The same condition for MHA would result in a 20 rem thyroid and 0.1 rem whole body dose.

It is concluded that purging for control of hydrogen generated in the post-accident environment for the Oconee reactors affords an acceptable and reliable means to prevent excessive hydrogen concentrations.

TABLE 14A-2.1

EFFECT OF PURGE START TIME ON DOSE

		Purge Start Time	Integrated Incremental LPZ Dose(1), rem	
			Thyroid	Whole Body
		950 hrs. (40 days)	0.020	0.003
LOCA	(2)	780 hrs (33 days)	0.050	0.003
		610 hrs (25 days)	0.160	0.004
		790 hrs (33 days)	1.3	0.008
MHA	(2)	620 hrs (26 days)	4	0.016
		450 hrs. (19 days)	9.3	0.040

(1) Without purging, LOCA doses are 5.2 rem thyroid and 0.017 rem whole body
23. and MHA doses are 149 rem thyroid and 0.66 rem whole body (120 days).

(2) Reference case

14A.3 POST-ACCIDENT HYDROGEN GENERATION

14A.3.1 RADIOLYTIC HYDROGEN GENERATION

14A.3.1.1 Sources of Radiation

The radiation sources which contribute to the energy absorbed by the coolant following an accident are shown in Table 14A-3.1. For the LOCA the only significant amount of radiation comes from sources in and near the core. For the MHA there are,in addition to the core sources, the contributions from the gases in the reactor building atmosphere, and the fission products in the coolant water itself. Table 14A-3.2 shows the assumption used in the calculation of the energy deposited in the solution described in the following section.

Figure 14A-3.1 shows the flow path of the post-accident cooling system in the long-term recirculation mode. Following a LOCA, the fluid for the building spray systems and the emergency core cooling system (ECCS) is supplied from the borated water storage tank. After the borated water storage tank is empty, coolant is then circulated from the reactor building sump through the decay heat cooling system to the reactor vessel.

The activity levels of the individual fission product nuclides were determined with B & W's proprietory digital fission product code. This digital code computes the activity of more than 200 fission product nuclides from one or two fissionable materials as a function of reactor operating history. One hundred (100) time steps can be used in the code and at each time step the program will print the individual nuclide activity along with the total gamma source strengths from all nuclides for each of six gamma energy groups.

The activity of the Np-239 and U-239 was obtained using the maximum neutron capture rates that occur in U-238 at any time during a core cycle. Sources from the activated clad and structural materials were calculated assuming saturation activity of these components and using lifetime average neutron fluxes in the core.

14A.3.1.2 Calculation of Absorbed Energy

Table 14A-3.2 summarizes the assumption made in calculating the energy absorbed by the coolant. In a LOCA all of the absorbed energy comes from sources in and near the core with the greatest fraction coming from fission product decay. The energy from fission product decay is about equally divided between gamma rays and beta particles. To determine the energy deposited in the solution by beta particles the fuel pellets were sub-divided into concentric cylindrical source shells, 10 mils in thickness. The amount of beta energy that was transmitted through the fuel cladding was calculated for each of these source shells. These calculations were performed for each fission product nuclide. It was assumed that all of the energy that penetrated the clad was absorbed by the water. The integrated beta energy between 10 and 3,000 hours after the LOCA was only two to three percent of that from the gamma rays. Nearly all of the beta energy is absorbed by the oxide and cladding.

For gamma rays the fission product source strengths were taken from the output of the fission product code and the gamma sources from U-239 and Np-239 were added to the various energy groups to obtain the total core source strengths. Energy deposition rates to the solution at various times were then calculated for each of the six gamma energy groups. Since these sources contribute most of the energy received by the solution, this calculation was checked using three techniques, all based on the assumption of a homogeneous core with a uniform average source distribution.

First, the amount of energy produced per unit volume of core was assumed to equal the amount of energy absorbed per unit volume. The distribution of absorbed energy between the cooling water and the remainder of the core was found by a ratio of the energy absorption coefficient of water to that of the homogenized core for each of the energy groups. In the second technique a receiver point was chosen at the center of the core. The energy absorption rate was calculated at this location with a point kernel integration code. The absorption at all other points in the core was assumed to be the same as that at this point. For the third calculation the core was represented as an infinite homogeneous medium and the flux equations for an infinite slab were used to calculate the absorbed energy. These latter two calculations require the use of energy absorption buildup parameters for the homogenized core. A comparison between the mass and energy absorption coefficients for the core with those of various materials showed that the coefficients for lead matched those of the core quite well. The energy absorption buildup factors for lead were thus incorporated into the flux equations using the Taylor form of the buildup.

At all times following the LOCA the results of the 3 methods were within 5% of one another. The first technique gives the highest answers and these results were used in the analysis of hydrogen generation. The total energy deposited in the solution was obtained by graphical integration of the absorption rate curves. Figure 14A-3.2 shows the energy absorbed by the solution as a function of time following the LOCA.

The results have been presented in terms of the energy absorbed by the solution rather than in dose units. The reason is that dose is a measure of energy absorbed per unit mass of material. Consequently, whenever the volume or mass of the emergency core cooling solution changes, the dose changes. The generation of hydrogen, however, is a function only of the energy absorbed by the solution and does not change simply because the mass of the solution changes.

The curve shown in Figure 14A-3.2 includes all the sources previously mentioned, fission product gammas, fission product betas, Np-239 and U-239, and activated structural materials. Also included is the energy absorbed by the water outside the core from sources inside core. This was determined by computing the spatial variation of the gamma fluxes in the water outside the core and integrating over the water volume. This contributed between 3 and 4 percent to the total energy absorbed. The activated cladding and other core hardware source yields deposited energy which is only a fraction of a percent of that from other sources. The greatest amount of absorbed energy comes from the fission product and Np-239 gamma rays; these contribute approximately 95% of the total energy.

For the MHA, in addition to the core sources, 100% of the noble gases in the reactor building atmosphere, and 50% of the halogens and 1% of the solids in the cooling water must also be considered. The calculation of the energy deposited in the solution from core sources following the MHA is similar to that for the LOCA except that the fission product inventory in the core does not include those nuclides released to the reactor building.

In the reactor building atmosphere the activity levels and beta and gamma yield data for the noble gases were used to compute the individual beta and gamma source strengths for each nuclide. These sources were distributed uniformly throughout the free air volume of the reactor building. For beta particles the energy absorbed per unit volume of air was assumed to be equivalent to that produced per unit volume. For gamma rays the absorbed energy was computed at a point in the center of the reactor building and it was assumed that all other points in the building received this same energy. The gamma flux was calculated with a point kernel integration code. No credit was taken for attenuation by shielding by structures within the reactor building.

For the 50% halogens and 1% solid fission products in the cooling water, it was assumed that all of the energy produced by these sources was absorbed in the water. The absorbed energy from the halogens was computed for each nuclide using the individual beta and gamma yields. The contribution from the 1% solids was obtained by taking 1% of the total fission product decay heat curve and depositing this quantity of energy to the coolant.

Figure 14A-3.3 shows the energy absorbed by the solution following an MHA. As can be seen, initially the energy is controlled by the fission products in the cooling water. However, by 100 hours the sources in the core have taken over and continue to control as time increases. The contribution from the noble gases in the building atmosphere does not show up on the graph - this was insignificant in comparison to the other sources. A comparison of this curve with that for the LOCA Figures 14A-3.2 and 14A-3.3 shows that after 100 hours there is little difference between the two types of accidents since, in both cases, most of the absorbed energy comes from core sources.

Since the core sources, and in particular the fission product gammas, contribute most of the energy to the solution, a comparison has been made between the decay heat gamma sources calculated by B&W with that published by Shure[1]. Figure 14A-3.4 shows the integrated gamma decay heat (fission products plus U-239 and NP-239) between 10 and 3,000 hours decay time following 620 days irradiation time from the two methods. Over the time span of interest for hydrogen generation (100 - 1,000 hours), they are in excellent agreement.

14A.3.1.3 Radiolytic Hydrogen Generation and Resultant H_2 Concentrations

The most significant source of hydrogen in the reactor building results from the radiolytic decomposition of water. The hydrogen generation rates were calculated utilizing the data presented in Figures 14A-3.2 and 14A-3.3 and a hydrogen generation constant ("G" value) equal to 0.45 molecules of hydrogen per 100 ev of energy absorbed by the fluid in the core region.

In the LOCA the integrated energy absorption by the sump solution is small, and this has been lumped with the core energy and a single hydrogen generation constant has been used. Under MHA conditions, when the energy absorption in the sump region is significant, as a result of the assumed sources, a "G" value of 0.3 molecules of hydrogen per 100 ev of energy absorbed was used to establish the sump contribution. Both of these hydrogen generation constants are those reported in the literature. The "G" value of 0.45 is used in the core region and is reported as a conservative upper limit for boiling solutions[2,3]. The "G" value
6. of 0.3 is used for the coolant in the sump which represents a slowly moving fluid and it is based upon published ONRL[4] data.

The radiolytic hydrogen concentration in the reactor building has been calculated as a function of time for LOCA conditions and is shown on Figure 14A-3.5. This curve shows that it is a long period of time after the accident before significant concentrations of several percent are reached. Calculations indicate that following the LOCA the radiolytic generation rates are low (range from 0.5 CFM at 500 hrs to 0.2 CFM at 2000 hrs) and it requires in excess of 6000 hrs. (250 days) to approach flammable concentrations in air.

Figure 14A-3.6 shows the radiolytic hydrogen concentration buildup for the MHA condition. Flammability concentrations under these conditions are approached at 3200 hrs. (133 days). A comparison of these two accident conditions at 1000 hrs shows that the MHA yields approximately 25% by volume more hydrogen than does the LOCA. In this comparison the small difference between the MHA and the LOCA hydrogen inventories demonstrates that conditions resulting from either accident are adaptable to a controlled purging operation.

TABLE 14A-3.1

SOURCES OF RADIATION

LOCA

LOCATION	SOURCE
CORE	100% FISSION PRODUCTS TRANSURANIUM ACTIVATION PRODUCTS (Np-239, U-239) ACTIVATED CLAD AND STRUCTURAL MATERIALS

MHA

LOCATION	SOURCE
REACTOR BUILDING ATMOSPHERE	100% OF NOBLE GASES
EMERGENCY CORE COOLING WATER	50% HALOGENS 1% SOLIDS
CORE	99% SOLID FISSION PRODUCTS 50% HALOGENS TRANSURANIUM ACTIVATION PRODUCTS (Np-239, U-239) ACTIVATED CLAD AND STRUCTURAL MATERIALS

TABLE 14A-3.2

ASSUMPTIONS FOR CALCULATION OF ABSORBED ENERGY

1. The core has operated at 2568 MWt for an average irradiation period of 620 days.

2. The core flooding tanks and borated water storage tanks have been discharged and coolant is circulating through the core and spray system from the sump.

ADDITIONAL ASSUMPTIONS FOR MHA

A. 100% of the gases, 50% of the halogens and 1% of the solids are released instantaneously from the core at the time of the accident.

B. The halogens are removed instantaneously from the reactor building atmosphere to the emergency cooling water.

C. The volume of fluid in each region remains constant.

D. The daughters descending from radioactive materials in a region remain in the same region with their parents, except for the gaseous daughters of the iodine in the cooling water. These were allowed to escape to the reactor building atmosphere.

14A.3.2 SOURCES OF CHEMICAL HYDROGEN

If the only source of hydrogen was from radiolysis of the coolant, the fission products would have decayed to insignificant values before purging would be required. That is, it would take radiolysis 3800 hours under LOCA conditions to build up 3.5 v/o of hydrogen in the building and 2040 hours under MHA conditions. The hydrogen generation rates at these times are less than 10 ft^3/hr and very low purge rates (less than 5 SCFM) would be required. These conditions would not increase the thyroid dose since decay has removed all iodine isotopes and only a small, almost insignificant, incremental whole body dose would be experienced. Thus, purging easily controls radiolytic hydrogen generation alone. Purging operations can also control the hydrogen that is generated from chemical sources within the reactor building. This hydrogen may be evolved from two sources: (1) zirconium-water reaction involving clad material, and (2) from the reaction of zinc within the reactor building with the borated coolant water.

14A.3.2.1 ZIRCALOY-WATER REACTION

The extent of the Zr-water reaction is controlled by the ECCS performance. The emergency core cooling equipment is sized to limit the hot spot fuel temperature to 2300 F or less during a loss-of-coolant accident, thereby limiting the zirconium-water reaction to less than one-tenth of one percent (14.2.2.3.6). Figure 14A-3.7 shows the variation in the hydrogen generation with the percent of Zr-water reaction. A one percent Zr-water reaction consumes only 430 pounds of zirconium, yielding 3,380 ft^3 of hydrogen, and contributes only 0.18 v/o hydrogen to the building hydrogen concentration. The one percent Zr-water reaction is an order of magnitude greater than the percentage metal-water reaction which core cooling analysis predicts. However, an allowance of a 1% metal-water reaction from this source is used in the purging analysis to provide additional design margin.

6. The effectiveness of ECCS has established that the MHA case represents more extreme conditions than can realistically be justified, because the emergency core cooling system prevents TID-14844-type fission product releases and the resulting higher radiolytic hydrogen generation. Therefore, the LOCA represents the accident which is mechanistically described, with the MHA being an upper-limit type approach that reflects a major area of conservatism in the analysis.

14A.3.2.2 ZINC-BORIC ACID REACTION

The second source of chemical hydrogen results from the reaction of boric acid with zinc in the reactor building at high temperature. The reaction produces zinc borate and hydrogen. Each pound of zinc consumed yields 5.5 SCF of hydrogen. The rate of reaction is dependent upon the alloy, the surface area in contact with the solution, the temperature and pH of the solution. If the allowable amount of zinc reaction is complete before the time that purging starts, the corrosion rate has no significance in the overall hydrogen analysis. This analysis is based upon the reaction of 46,500 ft^2 of zinc at a corrosion rate of 200 mg Zn/dm^2-day. For an estimated thickness of zinc on all galvanized surfaces of 2 oz/ft^2, the reaction is complete in 750 hours. Thus, the analysis is corrected for the zinc corrosion rate at all purging starting times prior to 750 hours.

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14A.3.2.3 Miscellaneous Hydrogen Sources

The reactor coolant system initially contains 15-40 cc of dissolved hydrogen per liter of coolant. This hydrogen can be partially released as the coolant is depressurized and cooled. The total amount of dissolved hydrogen is only 472 SCF and this is considered an insignificant source.

14A.4 EVALUATION OF PURGING TO CONTROL HYDROGEN CONCENTRATIONS

14A 4.1 GENERAL CONCEPT DISCUSSION

The controlled purge concept may be utilized for post-accident hydrogen control because hundreds of hours are available for fission product decay before the purging starts. The basic purging approach evaluated herein is to allow the hydrogen concentration to increase to the control limit of 3.5 v/o then to purge at a rate that removes the hydrogen being generated. Therefore, the maximum delay for fission product decay is obtained, and a decrease in purge rate results as hydrogen generation decreases. This method is oriented toward minimizing the radiation dose to the public.

In order to specify purge requirements, those conditions that control the purge starting times and the rate of purge must be established. The starting time is a parameter which has a significant effect on hydrogen purging rate and on the analysis of resulting environmental doses.

The reference conditions that were established to form the basis of this evaluation are as follows:

1. The lower flammability limit of 4.1 v/o of hydrogen in air represents a conservative upper limit on hydrogen concentration permitted in the reactor building.

2. Control limit of 3.5 v/o hydrogen is permitted before purging must start. This provides for a margin between the lower flammability limit and operating limit to account for the accuracy of sampling, for the degree of mixing in the reactor building, and for the capability to interrupt or delay purging for a significant period of time.

3. The chemical hydrogen from a 1% Zr-H_2O reaction plus the hydrogen from metals reacting with the ECCS solution is included.

4. The radiolytic hydrogen generation rate is that described on Figure 14A-3.5.

14A.4.2 HYDROGEN FLAMMABILITY LIMITS

The first condition listed in 14A-4.1 pertained to establishing the hydrogen concentration that would constitute a potential hazard if that concentration were exceeded. The literature establishes that hydrogen concentrations between 4.1 volume percent and 9 volume percent in air may ignite.

The radiolytic coolant decomposition also produces oxygen as the water is consumed. If the oxygen is released from the water to the atmosphere, the oxygen content of the building atmosphere increases after LOCA from 21.0 to 22.1 v/o before the reference purging limit is reached. This increase is not sufficient to significantly shift the flammability limits. For example, the Bureau of Mines[5] reports that the lower limit of concentration for downward flame propagation decreased from 9.45 v/o H_2 in air to 9.15 v/o in pure oxygen. After purging starts, any oxygen liberated will be removed at the rate of liberation.

In the lower flammability range of hydrogen concentrations, the burning is incomplete with a 10% combustion efficiency[6] at 4.1 v/o, and there is a 50% combustion efficiency at 5.6 v/o[7]. Only upward flame propagation is possible at 4.1 v/o, and sideward and upward propagation can occur only above 6.0 v/o. Accordingly, ignition would have to occur at several locations simultaneously to achieve an appreciable energy release while concentrations exist in the lower flammability range of 4.1 to 6.0 v/o. The pressure rise associated with hydrogen burning from concentrations in excess of 4.1 v/o may not be excessive. For example, the energy addition to the saturated air in the reactor building would yield an equilibrium pressure rise of only 2.5 psi for a 4.1 v/o concentration based upon 10% combustion efficiency. At 5.6 v/o hydrogen, where combustion efficiency is only 50%, the energy release would only be 14.7×10^6 Btu's and an equilibrium pressure rise of 14.4 psi would result. Although large and rapid energy releases can be achieved in this flammability range, the combustion, if it should occur, is not a hazard to the integrity of the reactor building. Reaching the 4.1 v/o lower flammability limit does not represent the type of hazard that would destroy the integrity of the building and the conservatism in this limit represents a real margin to the actual upper limit. Although a concentration greater than 4.1 v/o may be acceptable, a lower flammability limit of 4.1 v/o is nevertheless used in this evaluation.

14A.4.3 JUSTIFICATION OF CONTROL LIMIT

The second reference condition was the control limit of 3.5 v/o at which purging would start. The difference between this concentration and the lower flammability limit of 4.1 v/o represents a margin that allows for sampling errors, incomplete mixing of hydrogen, and the ability to interrupt the purge if unfavorable meteorology exists during the early stages of purging. The basis for this margin is as follows:

14A.4.3.1 Mixing of Reactor Building Atmosphere

Mixing in the reactor building atmosphere is expected to be good. The reactor building cooling fans or sprays will introduce considerable turbulence to the building atmosphere to provide good mixing of hydrogen in the early stages of the accident. In addition, all the reactor building volumes are connected by large vent areas (stair wells, elevator shafts, grating) to promote good air circulation.

Figure 14A-4.1 shows the reactor building cross-section. Examining post-accident conditions in the reactor building indicates that the reactor coolant will collect in the bottom of the reactor building and be recirculated to the reactor vessel. Thus, the sources of the hydrogen evolution are: (1) the water in the sump, and (2) the water leaking from the reactor coolant system. These locations are within the unrestricted main volume of the building and will permit the hydrogen to diffuse rapidly and provide a uniform mixture in this area. This rapid mixing occurs because hydrogen has a high diffusion rate and a low generation rate, and is capable of diffusing in all directions. At 780 hrs after the LOCA, the hydrogen generation rate is only 23 SCFH and decreases significantly with time. This results in the maximum addition of 550 ft^3 per day or an increase in concentration of only 0.03 v/o per day. The hydrogen will diffuse very rapidly giving an even distribution under the conditions existing in the reactor building. This situation is not analogous to one where attempts are made to mix streams of gases under dynamic conditions where residence times and mixing distances are critical. In addition, the thermal mixing effects, heating of air above the hot sump water, and possible steam releases from the reactor coolant system will move the hydrogen laden air from the points of generation toward the cold external walls and emergency cooling equipment. Although hydrogen is lighter than air, it will not tend to concentrate in high areas because of the high diffusion rate and because of the open design of the reactor building.

Since the hydrogen is released primarily from the coolant in the sump to the large open areas, the hydrogen must diffuse from the major volumes into those minor volumes which are enclosed. The minor volumes or those not having good communication with the major volumes would be at a lower hydrogen concentration because the hydrogen is diffusing from the higher concentration level to a lower concentration level. Accordingly, pockets, if they exist, will be low concentration pockets rather than high concentration pockets. As the maximum concentration in the major volume will never exceed the 4.1 v/o limit, flammable or explosive mixtures will not exist in the minor volumes which might be considered as pocket areas.

The ability of hydrogen to diffuse rapidly into all volumes is inferred by a recent CSE experiment[8] which measured the spatial concentration of iodine in the various compartments. The tests showed very good mixing in the main chamber and a rapid interchange by diffusion and mixing with the atmosphere of other chambers which had limited communication. The diffusivity of hydrogen is approximately 10 times that of iodine so a more uniform mixture would be expected for hydrogen than for iodine. Also, the higher concentrations would provide greater concentration gradients for better diffusion than was indicated by the CSE tests.

Although it is concluded that a nearly uniform hydrogen concentration will exist, it cannot be conclusively proved that it will; therefore, to be conservative, this analysis has included a 10% mixing margin. This is equivalent to 0.35 v/o hydrogen (6,685 ft^3).

14A.4.3.2 Hydrogen Concentration Measurement Error

Instrumentation used to measure hydrogen concentrations with good accuracy is commercially available. Accuracy can be within $\pm$ 2% of full scale reading, and if a 0-5 v/o meter is used, the maximum error for hydrogen concentration would be $\pm$ 0.1% (2,000 ft^3 of hydrogen). Samples can be obtained from both the main reactor building volume and from the purge line to obtain representative conditions. Thus, sampling and establishing the actual concentration should not be a significant problem.

14A.4.3.3 Time Margin

It may be desirable to stop purging for a few hours because of poor meteorology, to permit equipment maintenance, or for other reasons. The capability to interrupt purging temporarily should exist in terms of margin or control limit overshoot permitted. This margin has been set at 0.15 v/o (2865 ft^3) of hydrogen in the building. This 2865 ft^3 margin will allow purging to be stopped for as long as approximately 125 hours at 780 hours after a LOCA (when the hydrogen generation rate is only 23 ft^3/hr) without increasing the hydrogen volume percentage to more than the lower flammable limit in air.

14A.4.3.4 Total Margin

Adding these margins together, we have a 0.6 v/o (approximately 11,500 ft^3) hydrogen margin. Thus, it is established that a conservative control purging set point of 3.5 v/o provides adequate margin between the control limit and the lower flammable limit.

14A.4.4 REACTOR BUILDING HYDROGEN CONCENTRATIONS WITH CONTROLLED PURGING

14A.4.4.1 During LOCA

Combining the sources of chemical hydrogen with the radiolytic hydrogen from Figure 14A-3.5, the time-dependent hydrogen concentration in the reactor building following a LOCA is shown in Figure 14A-4.2. The hydrogen concentration reaches the purging set point in 780 hours (33 days). The capability to interrupt the purge for periods of unfavorable meteorology is clearly shown on this figure by the dashed line starting from the purge start point. Another 620 hours (26 days) are required without purging before the 4.1 v/o limit is reached. This demonstrates that the low hydrogen generation rates require long periods of time to affect significant concentration changes. The conservatism of the 4.1 v/o lower limit of the flammability range was previously discussed. This curve also shows that it is a very long time before concentrations rise to 5.6 v/o where 50% combustion efficiency exists. Therefore, more than adequate time is available to operate a controlled purge operation without endangering the reactor building integrity due to risk of energy release from rapid combustion.

The purge flow required is a function of the building hydrogen concentration and the hydrogen generation rate. The hydrogen generation rate decreases with time, thereby decreasing the required purge rate. Thus, the amount of purging is minimized to that amount which is necessary to control the hydrogen. This is shown in Figure 14A-4.3 for a LOCA condition which shows rapidly decreasing purge rates. It may be noted that the right-hand scale on this figure gives the purge rate in terms of the building leakage rate. The purge rates are the same order of magnitude as the typical design building leakage rates.

A major factor that affects this analysis is the radiolytic hydrogen generation. The effect of varying the energy deposited in the coolant or the "G" value is best expressed by the change in the time to start the purge as a function of the generation rate. Figure 14A-4.4 shows a plot of the purge start time as a function of the hydrogen generation rate constants. It can be seen that large variations can be tolerated without forcing the purging starting times (a major radiological dose controlling factor) below 400 hours.

14A.4.4.2 During MHA Conditions

The same conditions have also been examined for the MHA. Figure 14A-4.5 shows the higher radiolytic hydrogen generation with the concentration reaching the 3.5 v/o point in 620 hours (26 days) as compared to 780 for LOCA. The time to reach 4.1 v/o is 840 hours as compared to 1400 hours for a LOCA. The initial purge rate required for this accident is higher than for the LOCA because of the earlier start times where hydrogen generation rates are greater. For MHA the hydrogen generation rate is dependent not only on radiolytic decomposition of the coolant but also on the zinc corrosion rate. The MHA requires a purge of 37.5 SCFM as compared to 11 SCFM for the LOCA. However, this required purge rate rapidly decreases with time as shown in Figure 14A-4.6, since all of the zinc is reacted 750 hours after the accident initiation.

The sensitivity of the MHA to the hydrogen generation constant is shown on Figure 14A-4.7. Because the combination value for the generation constant (0.45 for vessel coolant and 0.3 for sump coolant) was used, the results are reported in terms of percent of the nominal value. Because the MHA is a conservative accident, examination of this sensitivity should be toward a reduced "G" value. A 50% reduction would shift the purge start in excess of 1800 hours. This would permit all but the long half-life activities to decay before the purge starts. Examination of higher-than-nominal "G" values shows that "G" must increase considerably before a significant shift occurs. An increase of over 50% would be required to shift the purge starting time to less than 400 hours.

14A.4.5 CONTROLLED PURGING SUMMARY

The control of hydrogen concentrations for both LOCA and MHA conditions is shown to be possible with small purge rates, and significant periods of time for fission product decay are available before starting the purge.

The most important reference case purging parameters for the LOCA and MHA are listed in Table 14A-4.1.

TABLE 14A-4.1

CONTROLLED PURGING PARAMETERS

	LOCA	MHA
Time Purge Starts	780 hours	620 hours
Initial Purge Rate	11 SCFM	37.5 SCFM
Maximum Hydrogen Concentration	3.5 v/o	3.5 v/o
Percent Zr-Water Reaction Considered	1%	1%
Zinc Corrosion		
Rate	200 mg Zn/dm^2-day	200 mg Zn/dm^2-day
Extent*	46,500 ft^2 Zn @ 2 oz/ft^2	46,500 ft^2 Zn @ 2 oz/ft^2

* All zinc coated surfaces in the reactor building.

14A.5 REACTOR BUILDING HYDROGEN PURGE SYSTEM DESCRIPTION

The purge system is composed of a portable purging station and a portion of the penetration room ventilation system. The purge system is operated as necessary to maintain the hydrogen concentration below the control limit. The purge discharge from the Reactor Building is taken from one of the penetration room ventilation system penetrations and discharged to the unit vent. A suction
16. may be taken on the Reactor Building via isolation valves PR-1 PR-6 PR-7 or PR-9 (Figure 6-5) using the existing vent and pressurization connections.

The purge rate is controlled through the use of a portable purging station (Insert, Figure 14A-5.1). The station consists of a purge blower, dehumidifier, filter train, purge flowmeter, sample connection and flowmeter and associated piping and valves. Major component data is listed in Table 14A-5.1.

The blower is a rotary positive type rated at 60 scfm. The dehumidifier consists of two redundant heating elements inserted in a section of ventilation duct. The function of the dehumidifier is to sufficiently increase the temperature of the entering air to assure 70 percent relative humidity entering the filter train with 100 percent saturated air entering the dehumidifier. The purpose of the dehumidifier is to assure optimum charcoal filter efficiency. Heating element control is provided by a thermoswitch. Humidity indication is provided downstream of the heating elements by a humidity readout gage. The filter train provides prefiltration, high efficiency particulate filtra-
15. tion and charcoal filtration. The filter train assembly is identical in design to the waste gas filter train assembly which is rated at 200 scfm, thus conservatively capable of performing the assigned function. Face velocity to the charcoal filter is very low. The charcoal filter is composed of a module, consisting of two inch deep double tray carbon cells. The purge flow to the unit vent is metered using a 0-60 scfm rotometer. The purge sample flow is metered using a 0-12 scfm rotometer. Both of these rotometers have an accuracy of ± two percent of full scale, and each has remote readout capability. The purge discharge rate is controlled by a blower discharge throttling valve. The purge sample activities can be collected, counted and analyzed in the radio-chemistry laboratory. Makeup air to the Reactor Building is supplied by a compressed air system connection to one of the aforementioned existing vent and pressurization connections.

That portion of the penetration room ventilation system piping and valves which is used as a part of the purge system is permanently installed and is designed for seismic loading through the existing vent and pressurization connections. The remainder of the purge system is the portable purging station which is stored in an area where an earthquake will not damage it. Following a LOCA, there is adequate time before purging is required to permit checkout of the portable purging station. The malfunction analysis is given by Table 14A-5.2.

14A.6 SYSTEM OPERATION AND TESTING

It is Duke Power Company's intention to operate all equipment, including the purge system, in such a way so as not to exceed the release limits permitted by 10CFR20 and thus minimize the total dose to the public. The following is the general post-accident procedure.

Following a LOCA, the hydrogen concentration is allowed to increase to the control point of 3.5 v/o hydrogen. Reference 14A.4.1 of this report for a discussion of this evaluation. Upon reaching the control point concentration, the purge system is placed into operation and the hydrogen concentration held at or below the control point. Every opportunity to take advantage of the meteorological conditions during purging will be utilized. Section 14A.4.3.3 of this report discusses the time margin associated with the hydrogen buildup which permits a utilization of the appropriate meteorological conditions. This procedure optimizes the system operation to minimize the total dose to the public.

6. The purge system testing requirements are the same as those described in American Air Filter Testing Procedures ASTT-100-C and ABST-102B (1 September 1969) and 6.4.4 of this report. The charcoal filter is designed to withstand the maximum heat load resulting from the radioactive deposits.

16. Post accident hydrogen concentration in the reactor building is determined using the automatic gas analyzer portion of the Chemical Addition and Sampling System. The gas analyzer utilized is the Model 643E "Condu-Therm" manufactured by the Hays Corporation having a range of 0-5 percent (volume) with an accuracy of $\pm$ 2 percent of fuel scale.

Air samples may be obtained via temporary piping from either the discharge of the hydrogen purge blower or directly from the reactor building as shown on Figure 14A-5.1.

In either case the air sample passes through a desicant filter installed in the sample inlet line near the gas analyzer and through a moisture separator (as
18. shown on Figure 9-3) prior to analysis. During purge operations, additional drying is obtained from dehumidification equipment installed in the portable hydrogen purge system. Analysis error is further reduced in the post-accident situation by introducing dry, bottled air into the hydrogen analyzer as the reference gas.

15. Gas analysis for the determination of hydrogen in air by using the difference of thermal conductivity is based on the fact that the thermal conductivity of hydrogen is greatly different from the other major constituents of air. Relative to air at 100 C, the thermal conductivity of hydrogen is 6.99; oxygen is 1.052; and nitrogen is 0.996. Complex mixtures of gases also can be analyzed by the above method when the component to be measured has enough influence on the mixture thermal conductivity that random changes among the background components are relatively negligible.

Stable background gases expected from the core following a MHA are primarily isotopes of Krypton and Xenon. The percentage of total containment volume represented by these gases range from 1 x 10^{-2} for Xe^{134} to 4 x 10^{-4} for Kr Kr^{84}. Three isotopes of Krypton and four Xenon isotopes are expected, and the percentage volume of the containment represented by all of these gases is approximately 4 x 10^{-2}. Trace amounts of other gases will be present as occur naturally in the atomosphere. Since the background gases are such a minute percentage of the total containment volume, their presence will not have any appreciable effect on the thermal conductivity of the building atmosphere.

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15. The gas analysis equipment is specified to measure oxygen and hydrogen in a background of nitrogen. (There are separate analyzers for the two gases - the oxygen analyzer uses the paramagnetic principle.) Calibration facilities are built in for setting the zero and span calibration and adjustable solid state alarms are provided. Alarms for both analyzers will normally be set at three percent volume.

Section 14A.4.3 of this report addresses the problem of hydrogen pocketing during the post-accident condition.

6. Prior to the purging operation, samples are taken to determine that the discharge will be within acceptable limits (10CFR20). The purge discharge (0-50 scfm) is injected into the 31,000 scfm unit vent effluent stream. A sample is periodically taken from the unit vent effluent during the purging operation. The sample is monitored for particulate, iodine and gaseous activity by unit vent monitors RIA-43, RIA-44, RIA-45 and RIA-46. These instruments are described in 11.1.2.4.2(1) and Table 11-5.

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(Carry-Over)

Table 14A-5.1

Reactor Building Hydrogen Purge System

Major Component Data

Blower	
Number	1
Type	rotary, positive
Rated flow, scfm	60
Rated head, inch WG	22
Dehumidifier	
Heating Elements	
Number	2 (redundant)
Type	Chromolox finned air heating elements. Catalogue Number SEF-10
Rating (each), watts	250
Heating Elements Control	
Number	1
Type	Chromolox thermoswitch, Catalogue Number 1700, immersion type
Filter Train	
Number	1
Type	Composite - prefilter, high efficiency particulate, and charcoal filters. Charcoal bed is a module consisting of two inch double tray carbon cells. Quantity of charcoal is 36 pounds.
Design Efficiency	
Prefilter	80-85 average dust spot with atmospheric dust.
High efficiency particulate	99.97 percent DOP test on 0.3 micron particles.
Charcoal	99 percent removal of radioiodine

Table 14A-5.2

Reactor Building Hydrogen Purge System

Malfunction Analysis

Component	Malfunction	Comments and Consequences
Blower	Blower stops running	Sufficient time is available for repairs.
Dehumidifier thermoswitch	Switch fails to de-energize heating elements and both elements operate to full capacity	With minimum flow of 5 cfm, maximum temperature is 195 F, which is compatible with system design. Heating elements can be manually de-energized.

14A.7 SITE DOSE CALCULATIONS AS A RESULT OF PURGING

As indicated previously, hydrogen generation rates are low, and several hundred hours are required to accumulate hydrogen to significant levels at which controlled purging is necessary. At that time the airborne fission products, such as the noble gases and radio-iodines, have decayed through a number of half-lives. Only Xe-133, Kr-85 and I-131 are available in sufficient quantities to present a potential exposure hazard to the public. The contribution from radioiodine is greatly reduced because of the long time period available after LOCA prior to purging operations, during which the radio-iodine will have decayed through 4 half-lives, or to less than 10% of the original inventory. The particulates will also be removed by various processes which occur within this long period, as discussed in 14A-7.1 below.

14A.7.1 FISSION PRODUCT INVENTORY VS TIME

In a LOCA, 100% of the noble gases and radioiodine in the coolant, and all of that in the fuel rod gap are released. Fifty percent (50%) of the radioiodine is assumed to plate out. A power level of 2568 MWt and 620 days operation at full power with 1% failed fuel is used in calculating the activities. For an MHA, TID-14844 release of fission products is assumed.

As shown in Figures 14A-7.1 and 14A-7.2 for a LOCA, the only isotopes which have sufficient inventory to contribute to the whole body dose after 200 hours are Xe-133, Kr-85 and I-131. Because of higher Xe-133 and I-131 decay rates, Kr-85 becomes the dominant isotope after 500 hours. Activities are in mev/sec so that they are directly comparable in their relative contribution to exposure doses.

For an MHA, Figures 14A-7.3 and 14A-7.4 show that the same isotopes contribute to the whole-body dose, but Xe-133 maintains its dominance for a longer period. There is very little difference between an MHA and a LOCA in the inventory of Kr-85.

The whole body doses calculated are the sums of the contributions of these isotopes. The thyroid doses are based on I-131 only, because after the first few hundred hours, the I-131 concentration is orders of magnitude higher than that of any of the other radioiodine isotopes. These doses are conservatively calculated since, as demonstrated in CSE[8], a condensing steam environment is very effective in removing particulates. Reductions of greater than 10^3 were effected in less than 24 hours. These tests and other sources in the literature[9,10,11] indicate that particulates are easily and effectively removed from the reactor building atmosphere within 100 hours of their release.

14A.7.2 DECAY PRIOR TO RELEASE

As shown previously, the noble gas and iodine inventories are reduced significantly by radioactive decay prior to purging. Decay, along with release, continues to deplete the inventory in the reactor building during the purging period.

Iodine 131 and xenon 133 airborne activity inventories versus time are shown in Figure 14A-7.5.

14A.7.3 DOSE CALCULATIONS

To evaluate hydrogen purging and the doses which result, site meteorological conditions (Section 2.3) are used. Shown in Table 14A-7.1 are Oconee Nuclear Station characteristics which were used in this analysis. The atmospheric dispersion factors are calculated for the 30-day weighted average dispersion conditions at the low population distance. Figures 14A-7.6 through 14A-7.9 show the effect of two months of purging on the thyroid and whole body doses for both LOCA and MHA as a function of purge start time. All doses are integrated over the entire purge period and credit is taken for a 90% reduction in the iodine concentration from purging through a high efficiency filtering system.

Figure 14A-7.6 shows the thyroid dose versus purge start time for a LOCA. The dose for the maximum permissible building leakage rate is indicated by the horizontal line. The Total Dose curve above it indicates the total thyroid dose from both maximum building leakage during and following LOCA and from purging starting at the indicated time, both integrated over essentially an infinite release time. Thus, the space between the curves is indicative of the dose contribution from purging alone. The vertical bar indicates the reference case described in Section 14A-4. The dose increase for this case due to purging is negligible - only 50 mrem at the low population zone. The incremental change in dose would be small even at 400 hours (approximately one-half the reference purge start time) after an accident. A 0.5 rem increase would result from purging at this early time.

Figure 14A-7.7 shows the 120-day time integrated whole body dose with and without purging. Again, the vertical bar marks the reference case showing that the incremental increase in dose due to purging is only 3 mrem at the low population distance. The whole body dose increase is not excessive, less than 7 mrem, even if purging is initiated as early as 400 hours after an LOCA.

Figure 14A-7.8 shows the MHA thyroid doses integrated over an infinite purge period. Again, the horizontal line indicates the dose without purging and the heavy black curve the dose with purging. The dose increase for the reference case, indicated by the vertical bar, is 4 rem. Again, the dose change resulting from purging is only a small fraction of that calculated using the design building leak rate. The increase is only 12 rem at 400 hours.

Figure 14A-7.9 is for the MHA whole body dose. The same coding is used. Again, it is shown that purging does not result in excessive dose increases, approximately 55 mrem, if purging is initiated as soon as 400 hours after an MHA. The incremental dose for the reference case is 16 mrem at the low population distance.

The analysis presented here demonstrates that the doses at the low population distance which result from controlled purging are low, even for extreme changes in purge starting time.

14A.7.4 DOSE SENSITIVITY ANALYSIS

Figure 14A-7.10 shows the sensitivity of the LOCA whole body dose and the thyroid dose to changes in hydrogen "G" value. The reference case is indicated by the vertical bar. The thyroid dose is indicated by the solid line and the corresponding scale is on the left. The whole body dose is indicated by the dashed line and the corresponding scale is on the right. The dose without purging is indicated by the horizontal line. It indicates that a variation in the "G" value by a factor of more than 2 times the reference case does not result in excessive dose change.

Figure 14A-7.11 shows the MHA dose as a function of hydrogen "G" value. An increase in the hydrogen "G" value of 100% from the reference does not result in excessive doses.

Figure 14A-7.12 shows the sensitivity of the LOCA whole body dose and the thyroid dose to changes in percent Zr-water reaction. The reference case is again indicated by the vertical bar. A 10% Zr-water reaction (10 times the percent reaction considered in the reference case) would not result in excessive potential personnel doses.

Figure 14A-7.13 shows the MHA doses as a function of percent Zr-water reaction. Again, a 10% Zr-water reaction dose does not result in excessive potential doses.

Table 14A-7.2 summarizes the analysis inputs for the LOCA and MHA reference cases for the Oconee reactors, and gives the overall incremental doses due to purging for LOCA and MHA.

14A.7.5 THE CALCULATION OF MAXIMUM PERMISSIBLE CONCENTRATIONS

The radioactivity released during purging is further evaluated in terms of maximum permissible concentrations (MPC) as presented in 10 CFR 20. The fractions of an MPC for I-131, Kr-85, and Xe-133 released over a two month purge as a function of purge starting time are listed in Tables 14A-7.3 for LOCA and MHA, respectively. The MPC was taken as 3×10^{-7} μc/ml for Kr-85 and Xe-133, and 1×10^{-8} μc/ml for I-131. This analysis was also based on an average annual meteorology of 8.47×10^{-7} sec/m^3 (Section 2.3) at the low population distance, and the release was averaged over a year. Credit was also taken for a 90% filtering efficiency of the I-131.

For both MHA and LOCA, the purge starting time for the reference case is long enough after the accident that less than 0.1 MPC for iodine or the noble gas is released. Even for purging as early as 400 hours (approximately one-half the reference case purge starting time) after either accident, 10 CFR 20 MPC limits are not exceeded.

TABLE 14A-7.1

OCONEE NUCLEAR STATION CHARACTERISTICS

		CHARACTERISTIC	VALUE
	1.	POWER LEVEL	2568 MWt
	2.	REACTOR BUILDING FREE VOLUME	1.91×10^6 ft^3
18.	3.	BUILDING DESIGN LEAK RATE	0.25v/o Per Day - 1st 24 hrs .125 v/o Per Day - thereafter
	4.	ATMOSPHERIC DISPERSION FACTOR, X/Q	3.70×10^{-7} sec/M^3 @ 6 mile low population distance

TABLE 14A-7.2

CONTROLLED PURGING PARAMETERS

	LOCA	MHA
Time Purge Starts, Hours	780	620
Initial Purge Rate, SCFM	11	37.5
Maximum Hydrogen Concentration, v/o	3.5	3.5
Percent Zr-Water Reaction	1%	1%
Pounds of Zn Reacted	5812.5	5812.5
Iodine Filtering Efficiency, %	90	90

INCREMENTAL LPZ DOSES RESULTING FROM PURGING

	LOCA	MHA
Thyroid Dose, rem	0.05	4.0
Whole Body Dose, rem	0.003	0.016

TABLE 14A-7.3

FRACTIONAL MPC'S AT THE LOW POPULATION BOUNDARY

Purge Start Time (hours)	Fraction of MPC LOCA I-131	LOCA Xe-133	LOCA Kr-85	MHA I-131	MHA Xe-133	MHA Kr-85
400	0.009	0.014	0.020	0.257	0.251	0.027
600	0.004	0.004	0.016	0.094	0.064	0.022
800	0.001	0.001	0.012	0.023	0.010	0.016
1000	-----	-----	0.011	0.010	0.003	0.015

14A.8

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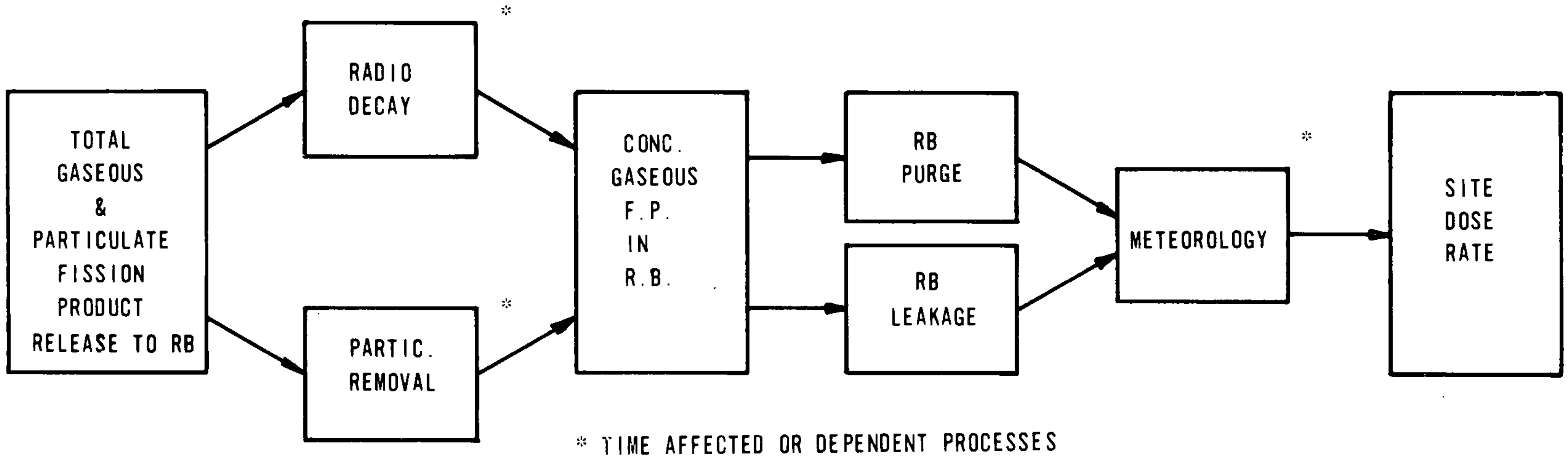
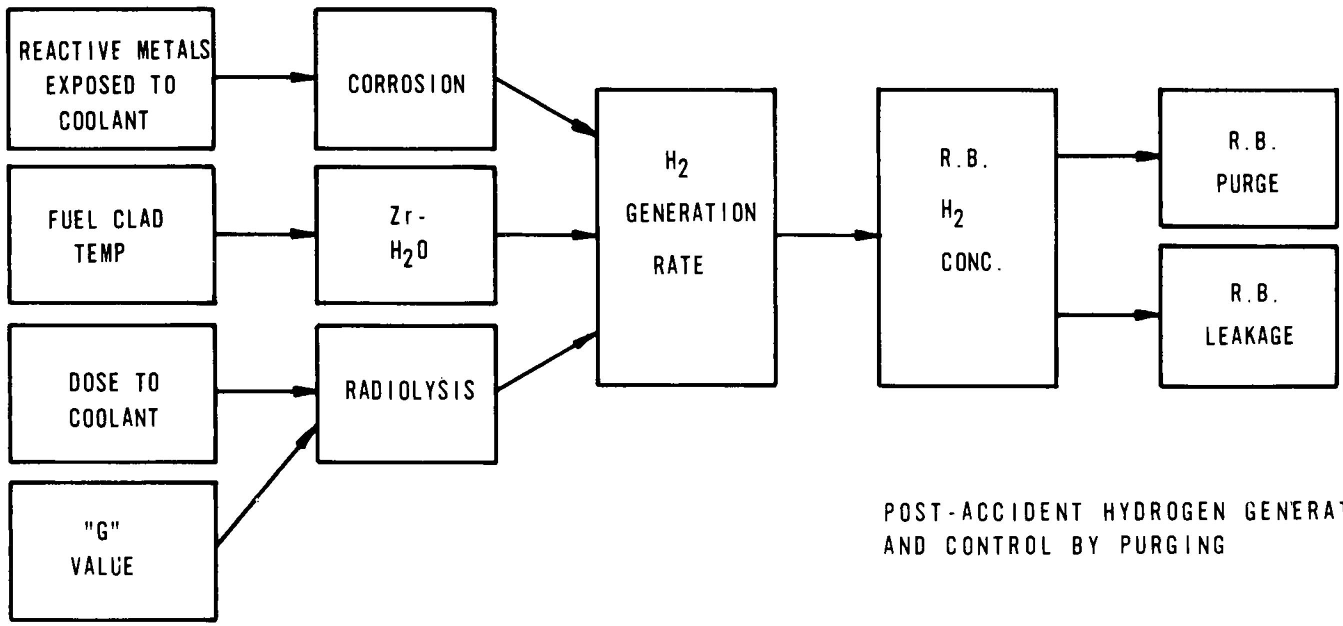
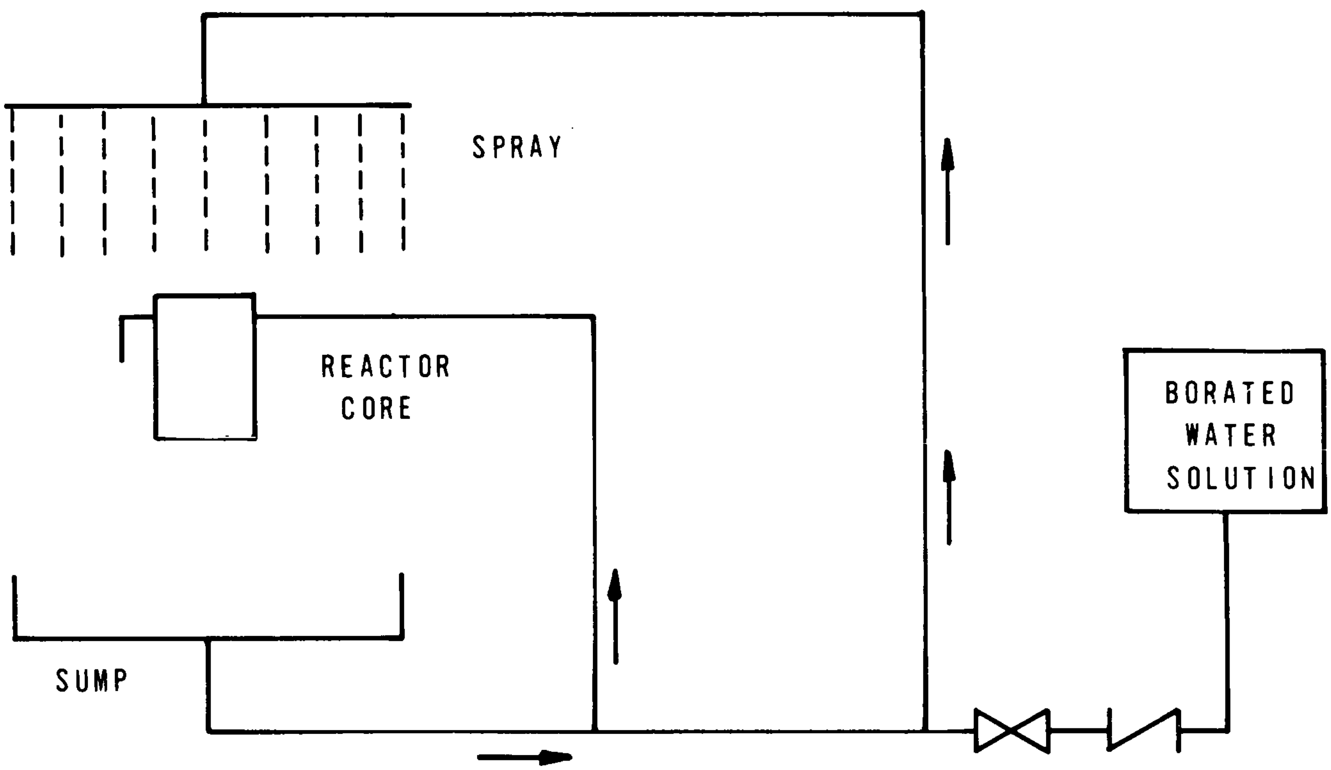


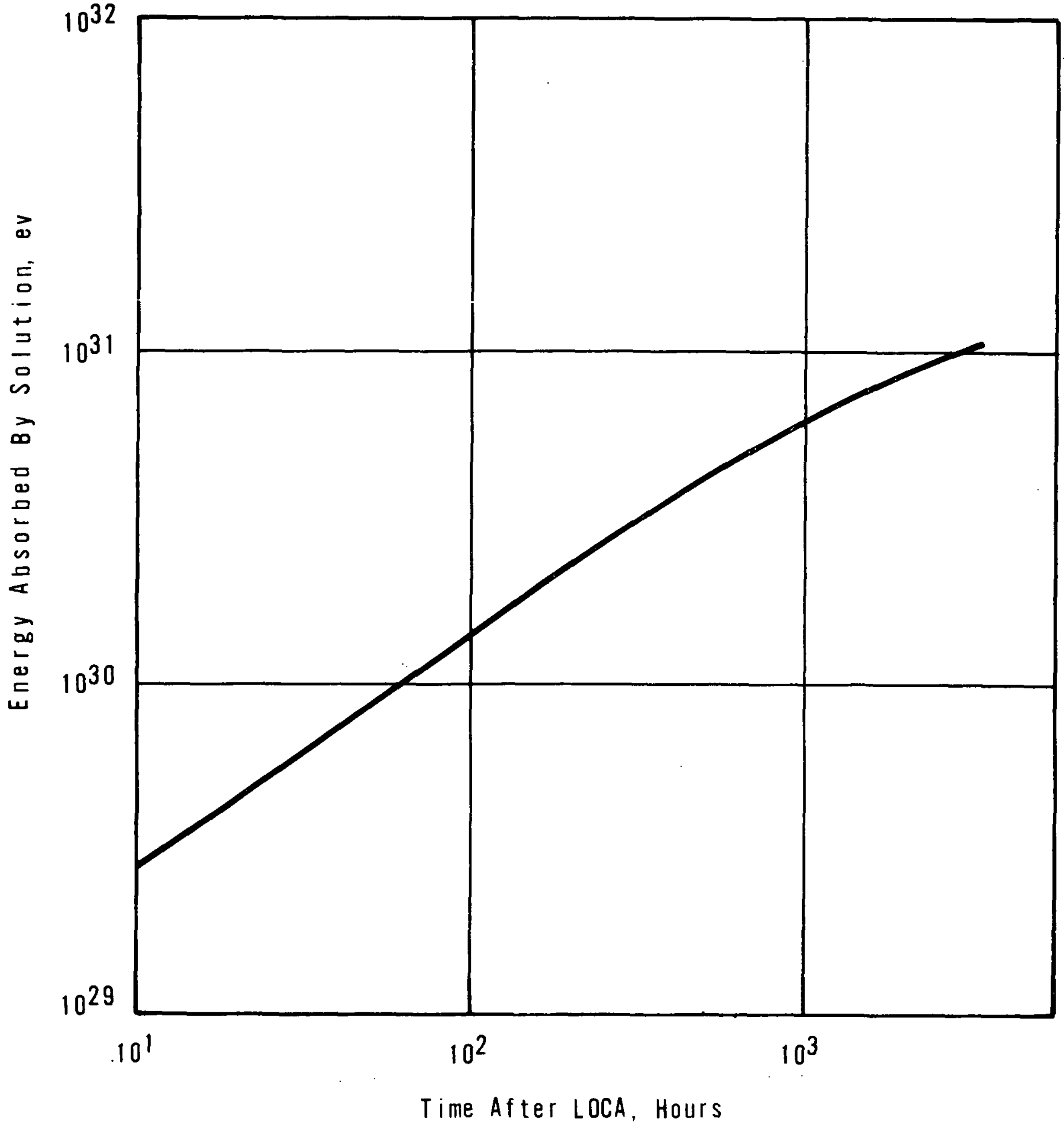
* TIME AFFECTED OR DEPENDENT PROCESSES
WHEN CONSIDERING PURGING



POST-ACCIDENT HYDROGEN GENERATION
AND CONTROL BY PURGING



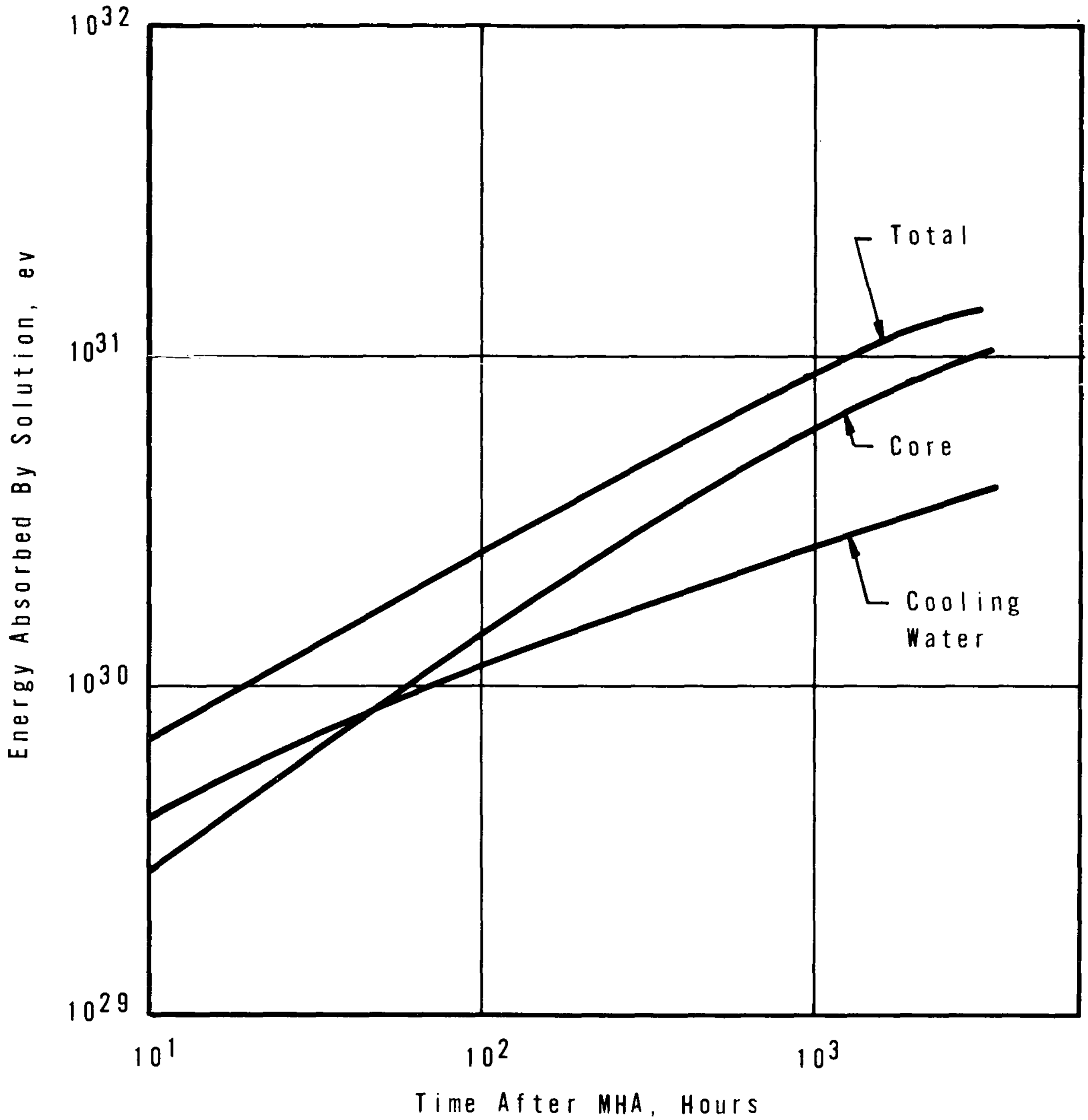
REACTOR BUILDING SPRAY SYSTEM SCHEMATIC



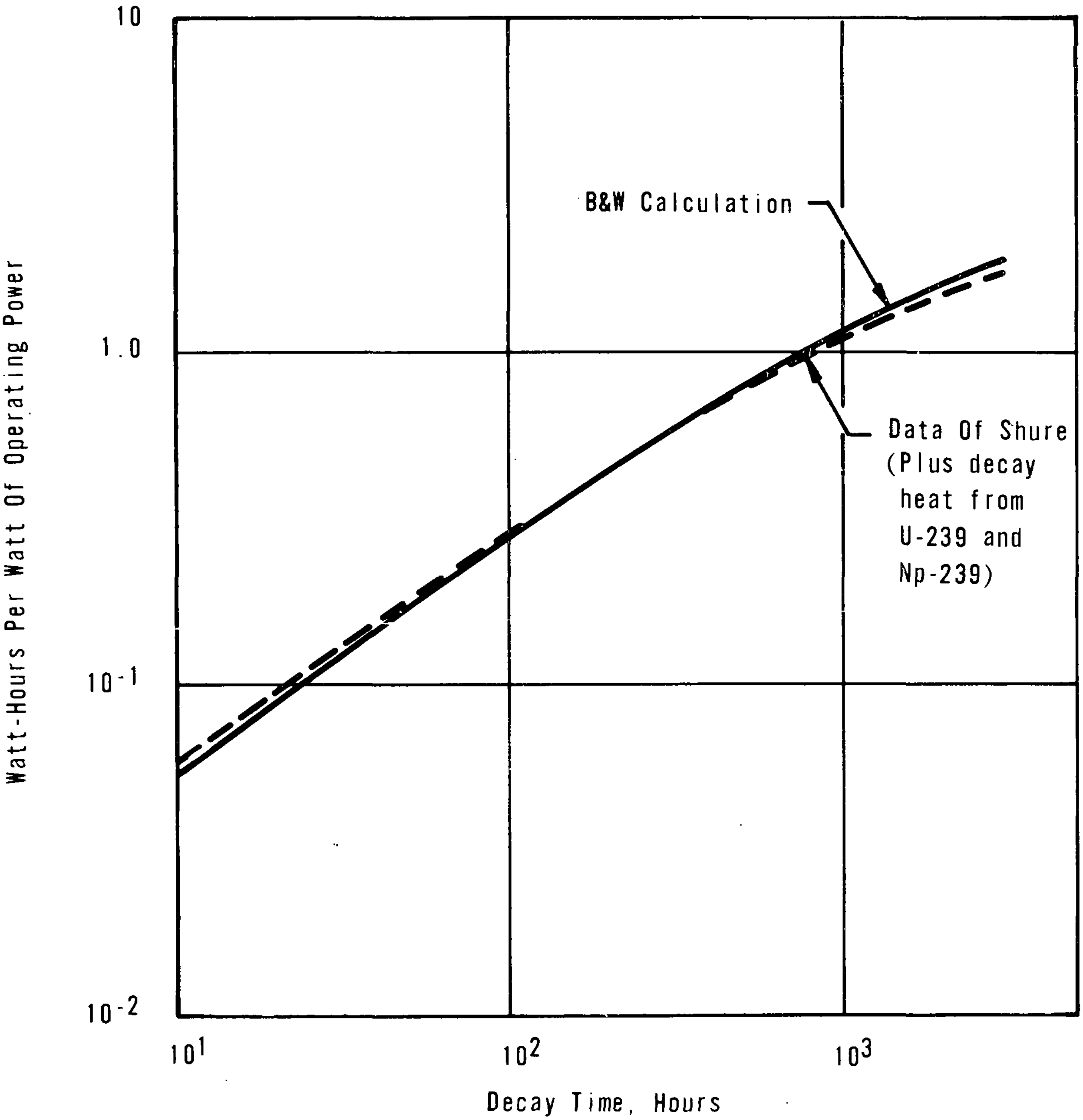
ENERGY ABSORBED BY SOLUTION FOLLOWING LOCA

OCONEE NUCLEAR STATION

Figure 14A - 3.2



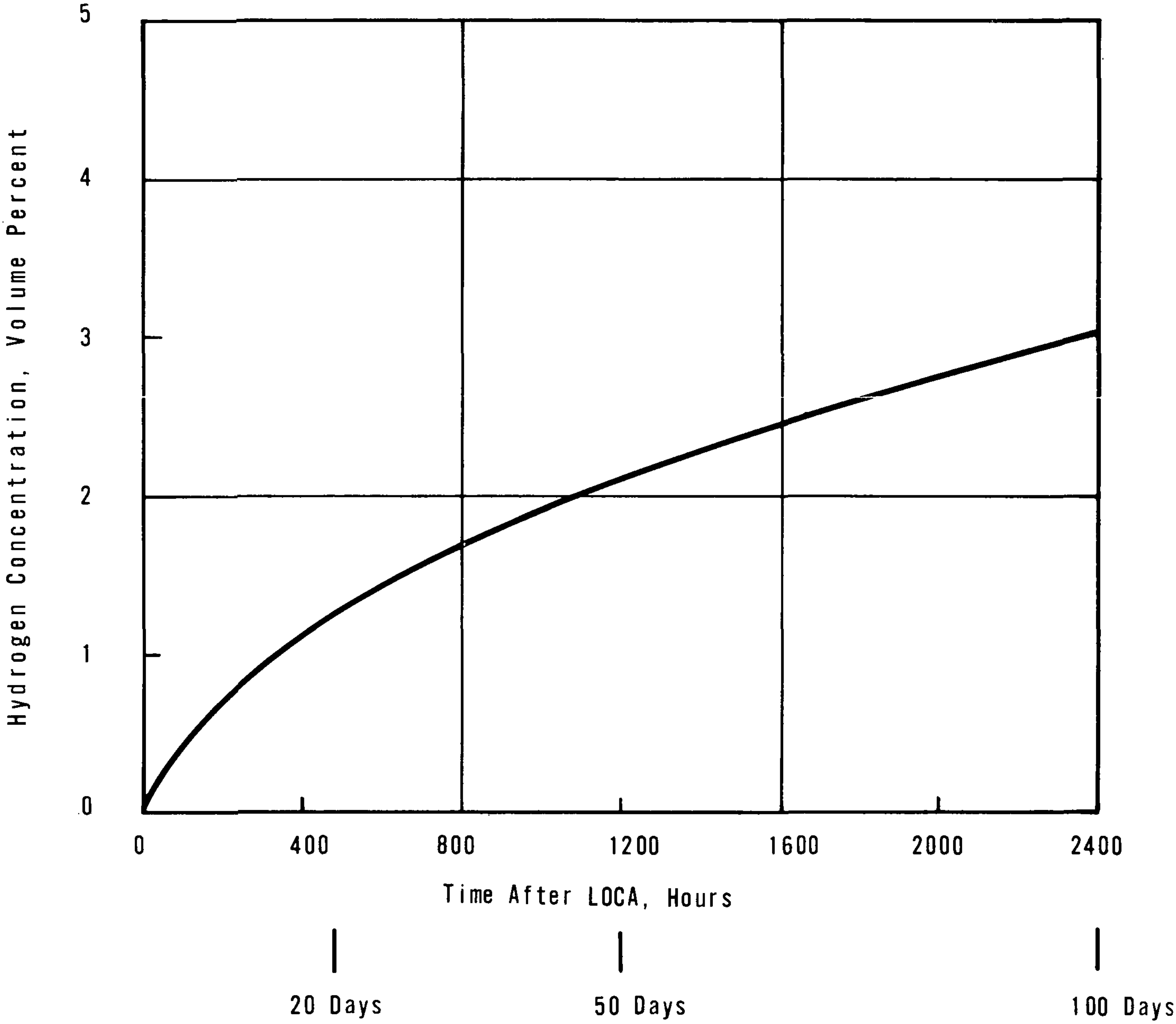
ENERGY ABSORBED BY SOLUTION FOLLOWING MHA



INTEGRATED GAMMA DECAY HEAT-620 DAYS IRRADIATION

OCONEE NUCLEAR STATION

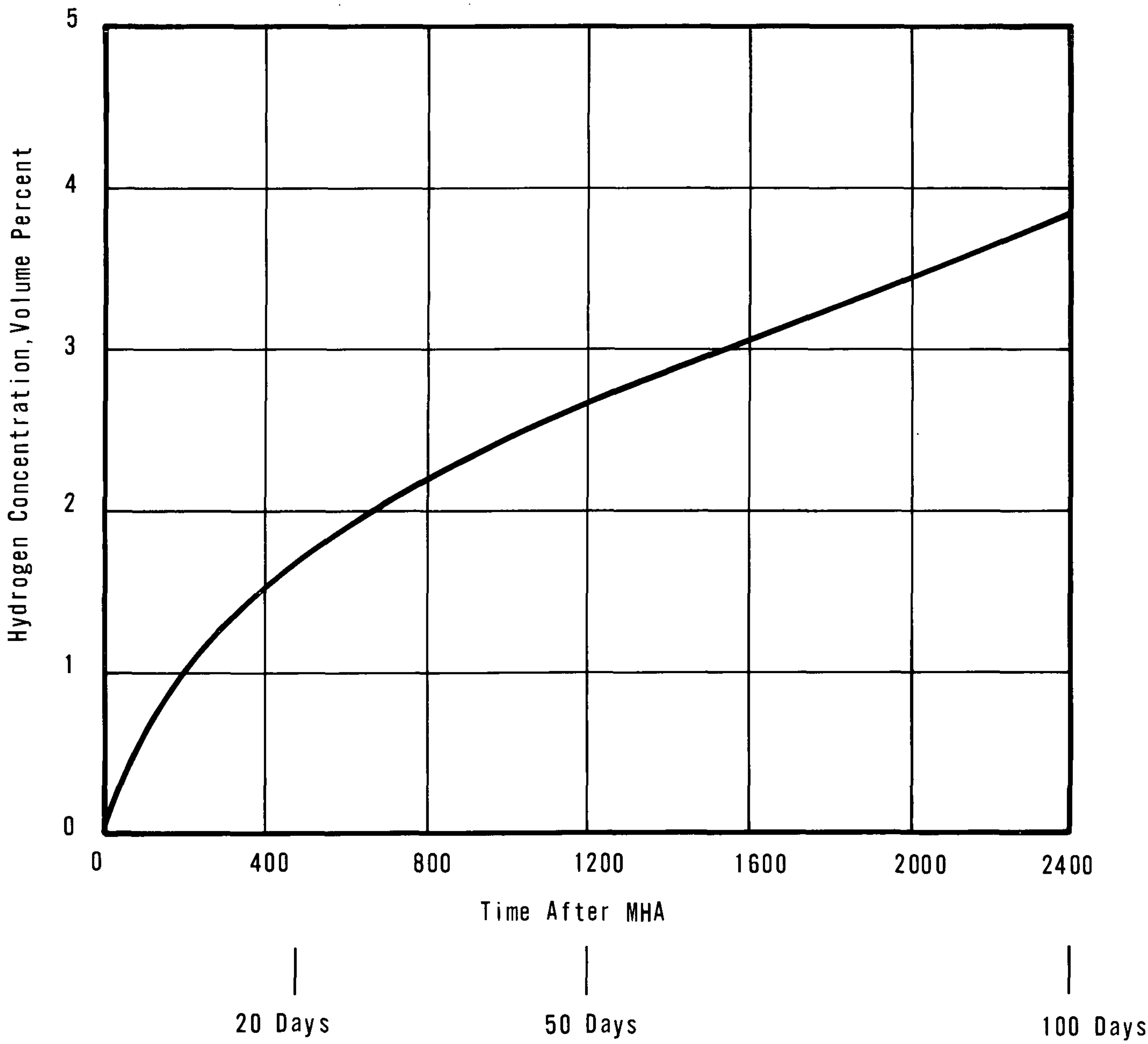
Figure 14A - 3.4



REACTOR BUILDING RADIOLYTIC HYDROGEN
CONCENTRATION FOLLOWING A LOCA

OCONEE NUCLEAR STATION

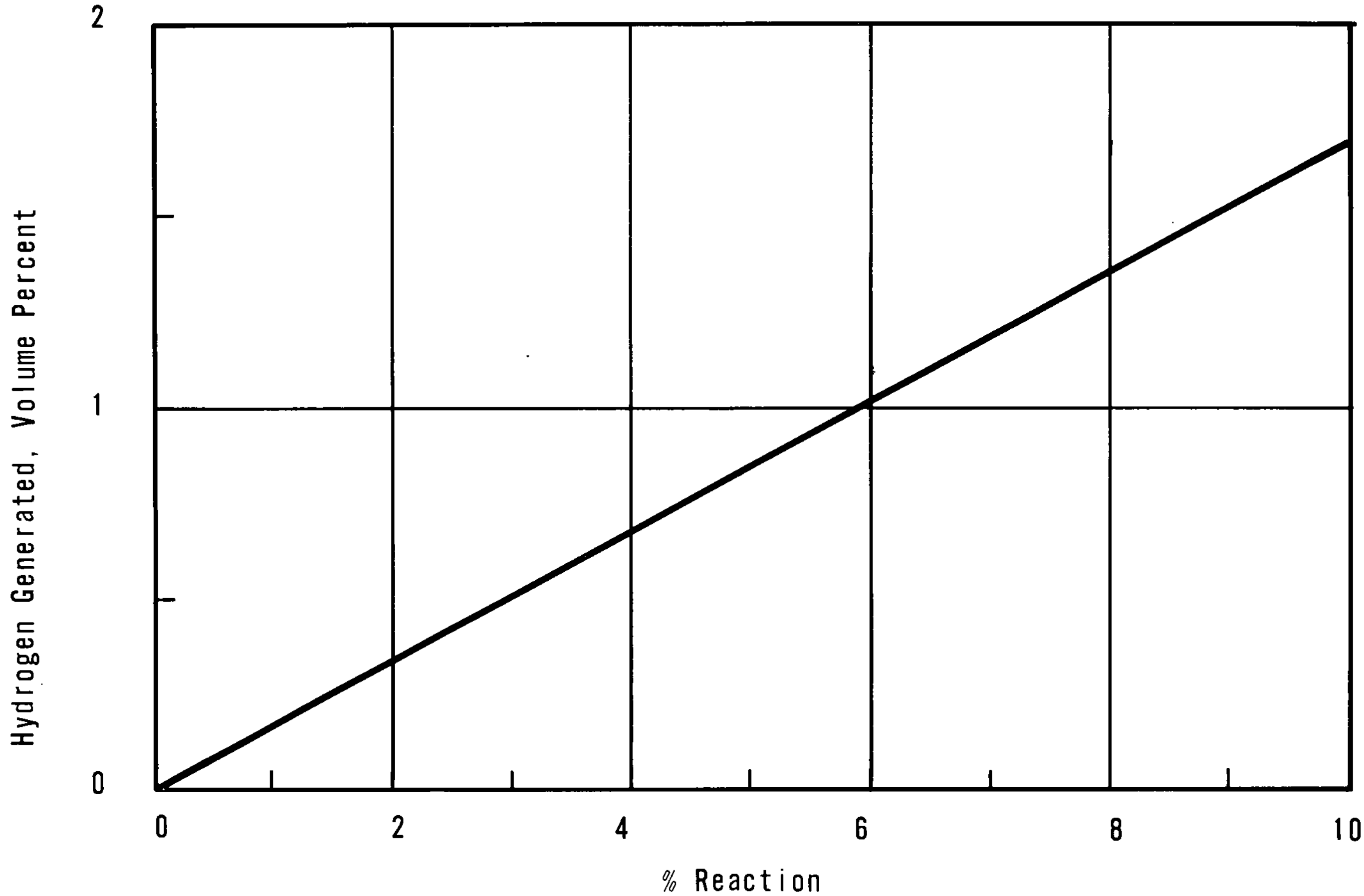
Figure 14A - 3.5



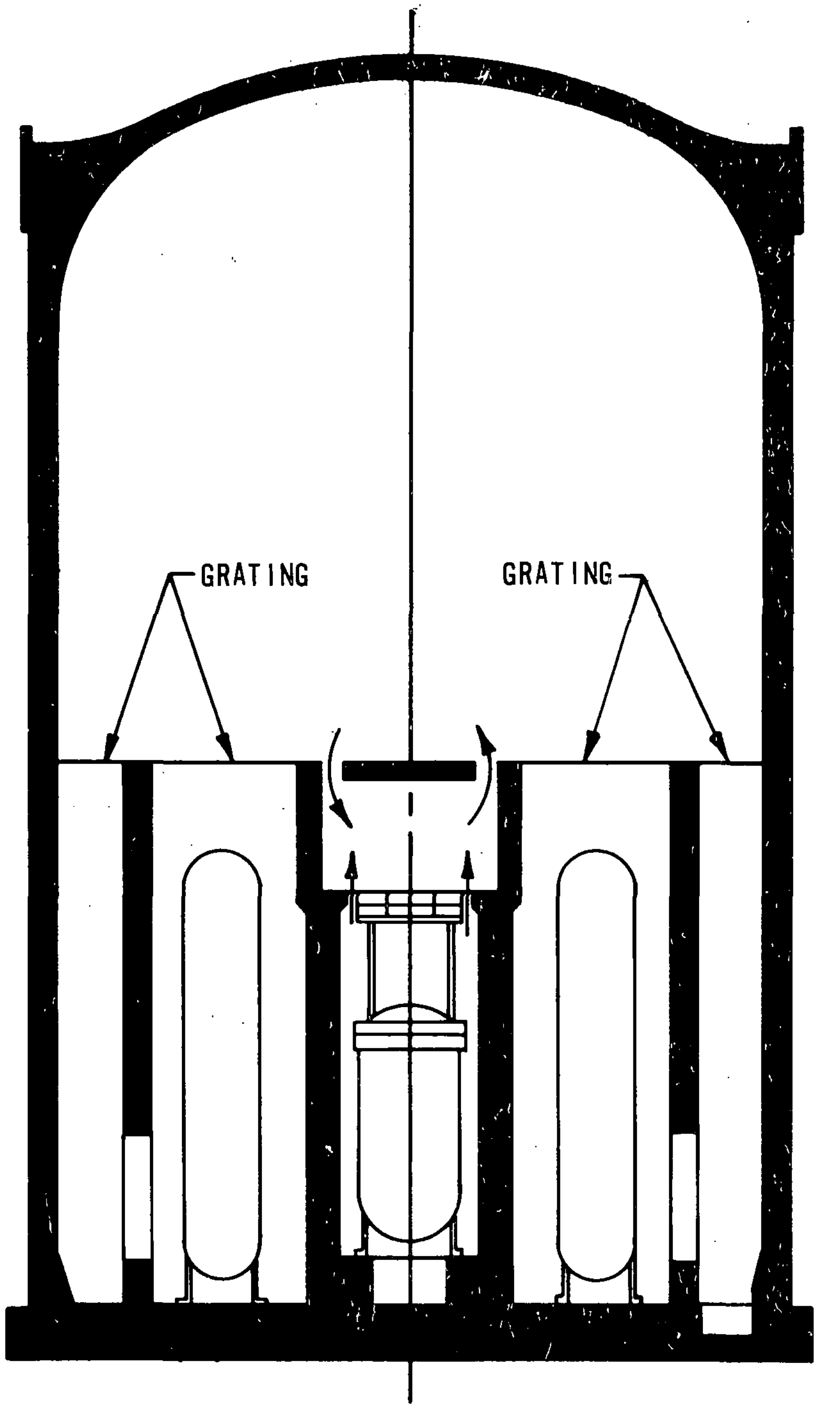
REACTOR BUILDING RADIOLYTIC HYDROGEN
CONCENTRATION FOLLOWING A MHA

OCONEE NUCLEAR STATION

Figure 14A - 3.6

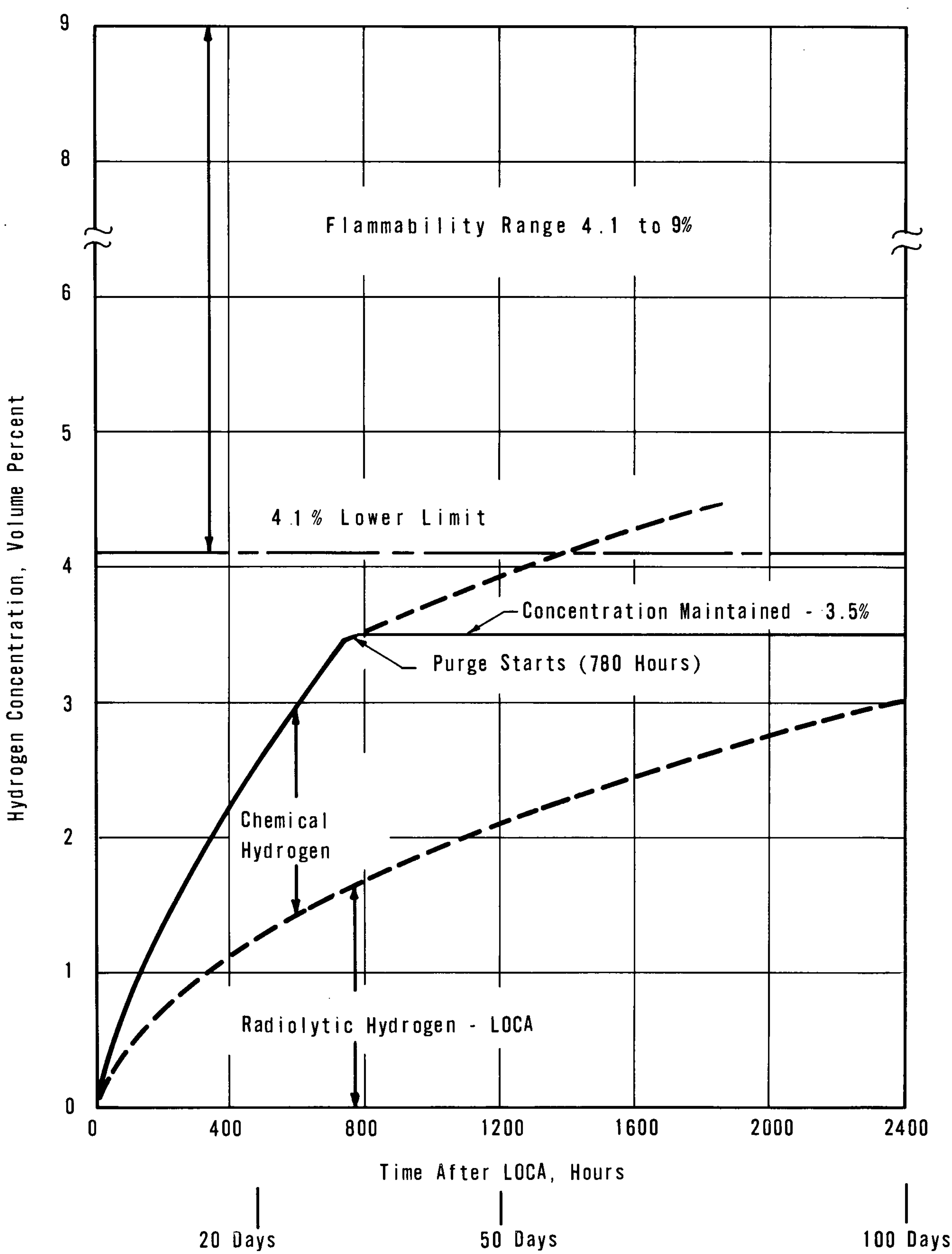


HYDROGEN GENERATION FROM $ZR-H_2O$
REACTION



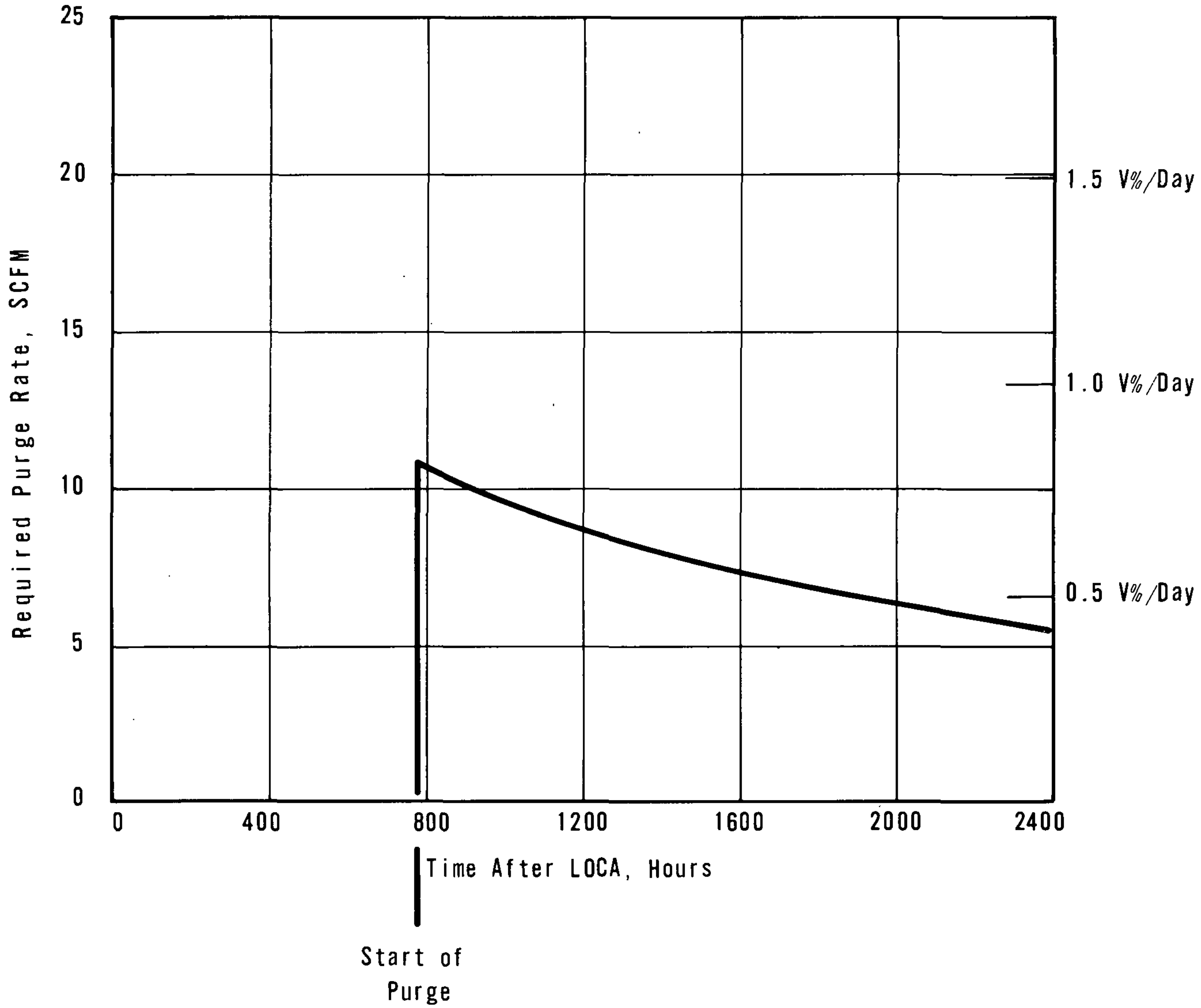
REACTOR BUILDING ARRANGEMENT

Figure 14A-4.1



OCONEE NUCLEAR STATION
Figure 14A - 4.2

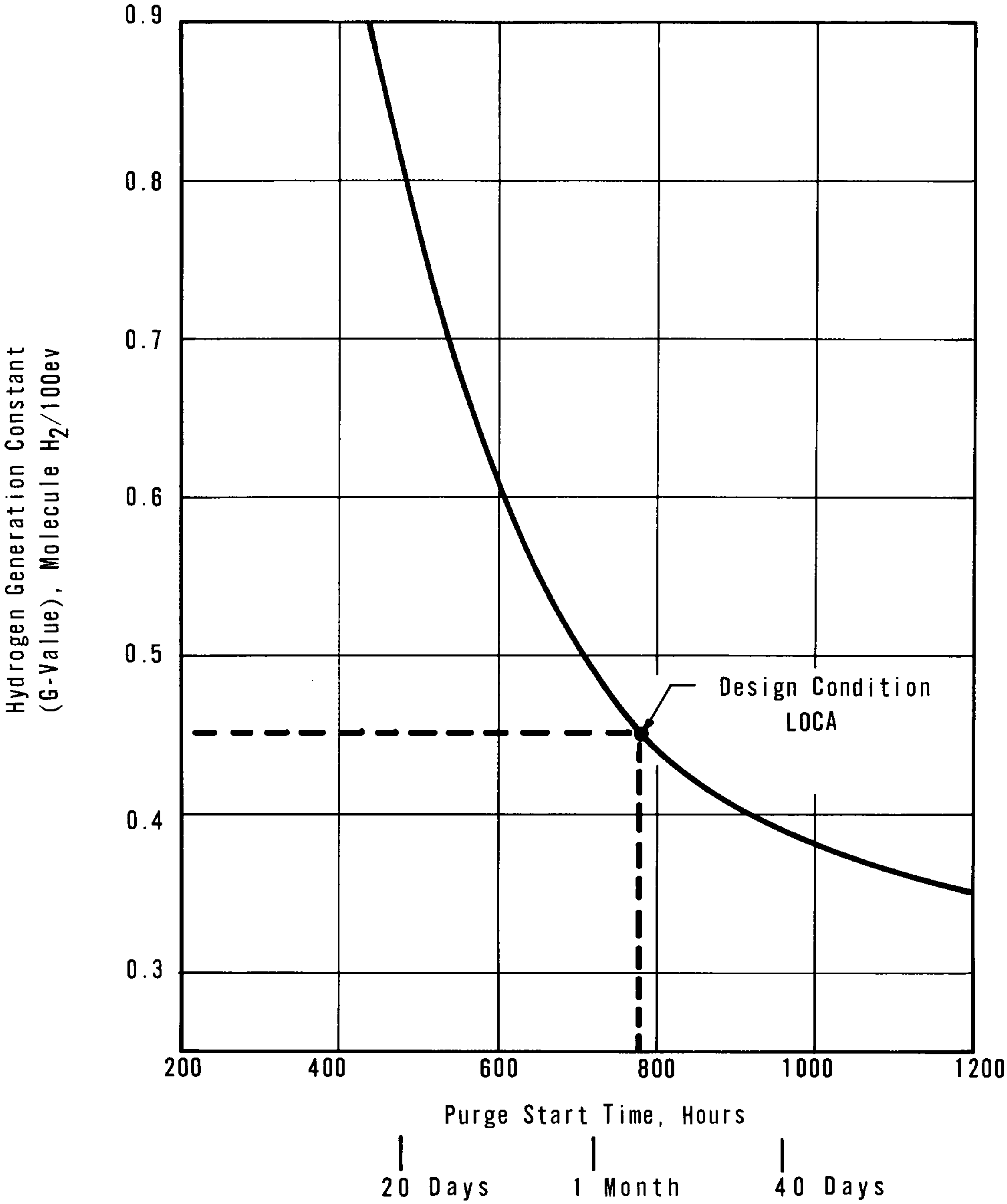
REACTOR BUILDING HYDROGEN CONCENTRATION FOLLOWING A LOCA



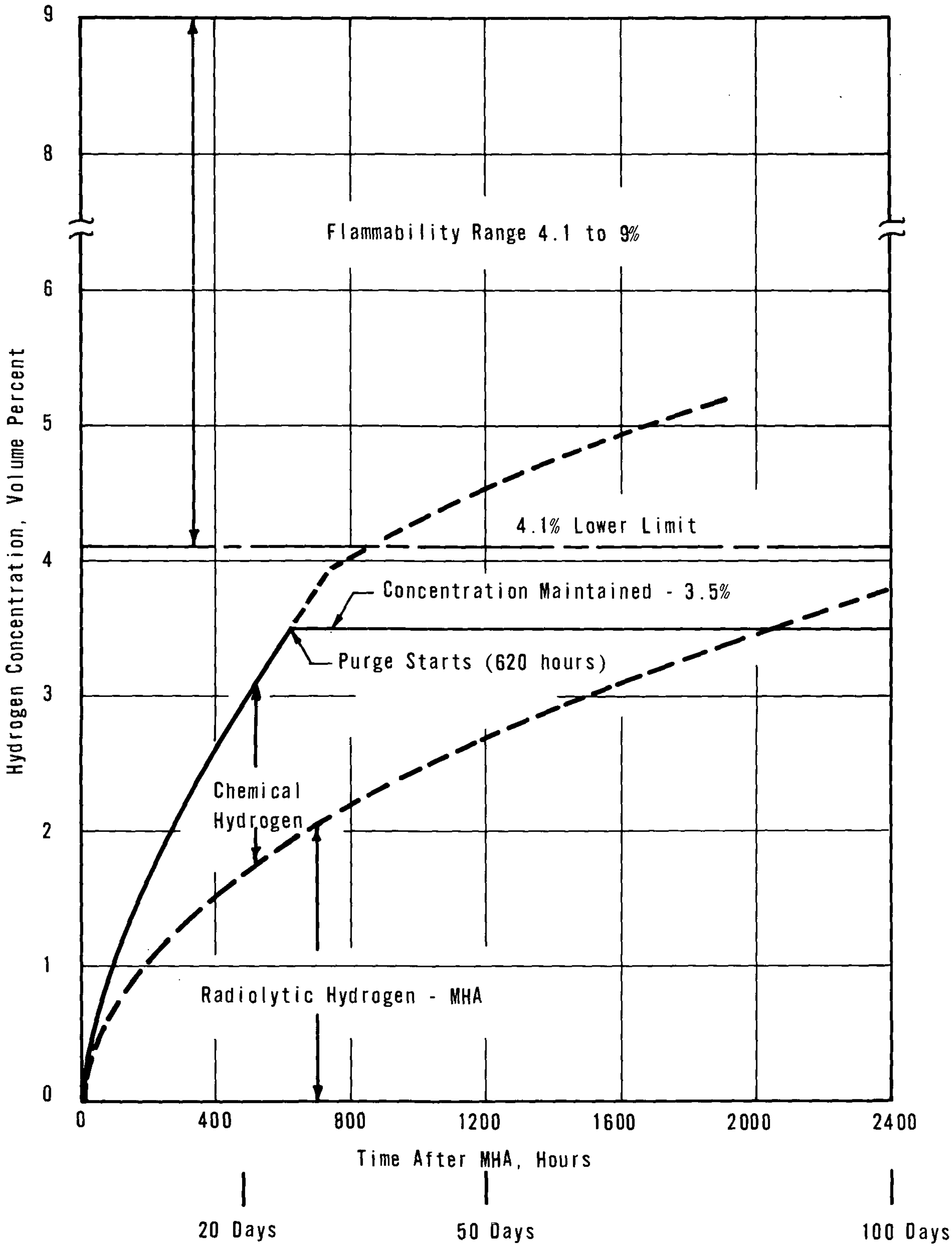
CONTROLLED PURGE FLOW REQUIREMENTS FOLLOWING LOCA TO HOLD BUILDING AT 3.5% HYDROGEN

OCONEE NUCLEAR STATION

Figure 14A - 4.3



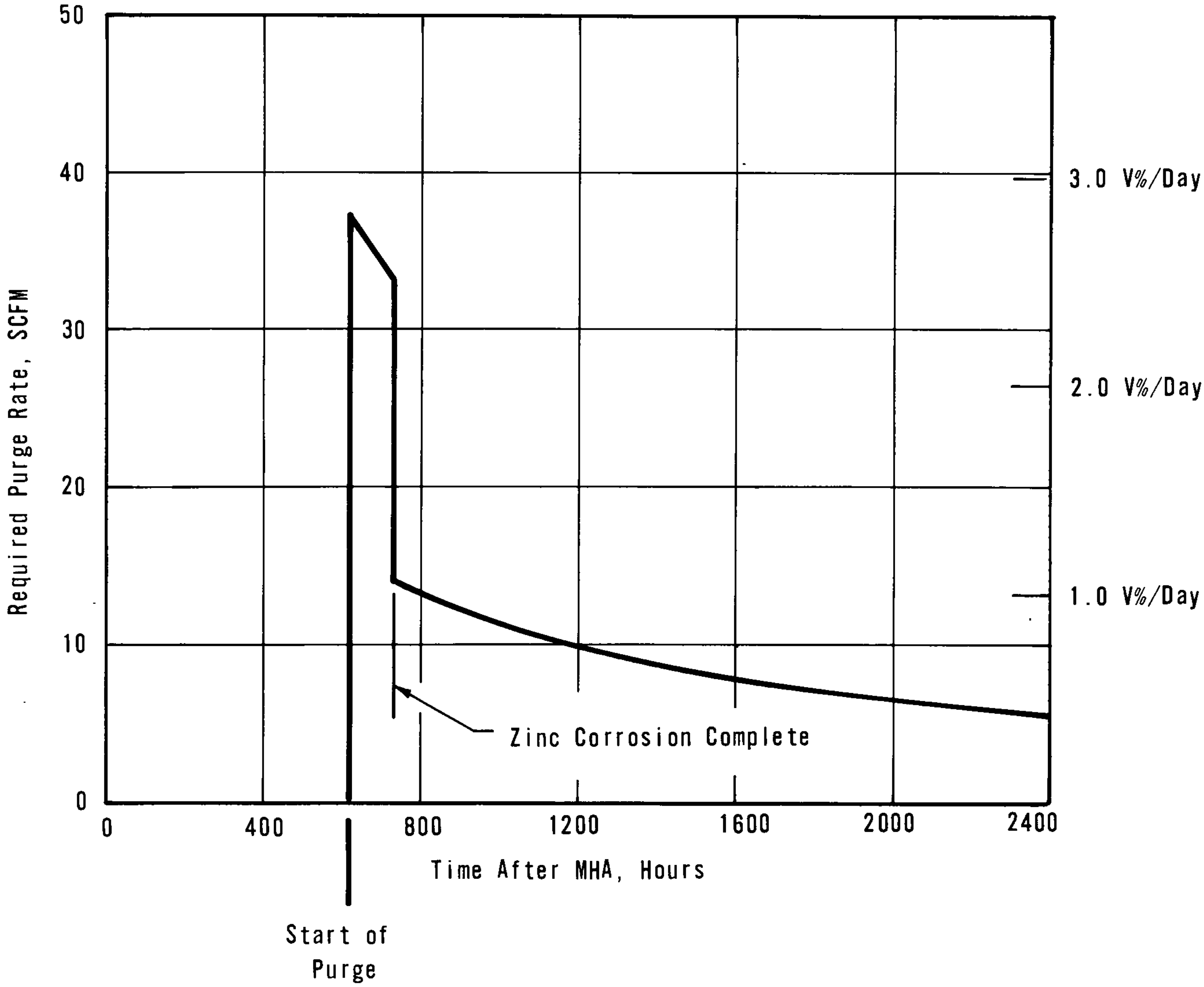
SENSITIVITY OF PURGE START TIMES AS
FUNCTION OF HYDROGEN GENERATION CONSTANT



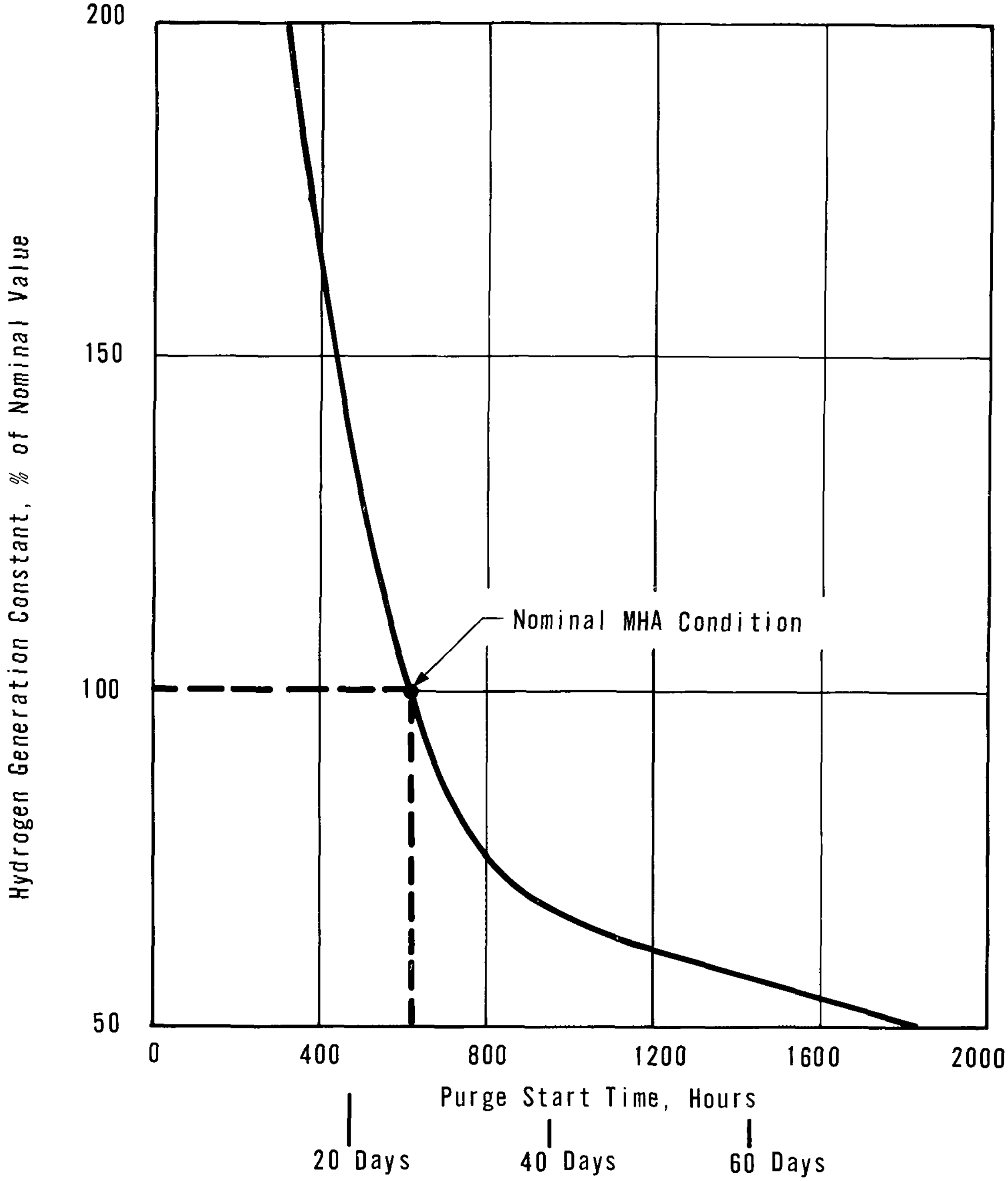
REACTOR BUILDING HYDROGEN CONCENTRATION
FOLLOWING A MHA

OCONEE NUCLEAR STATION

Figure 14A - 4.5



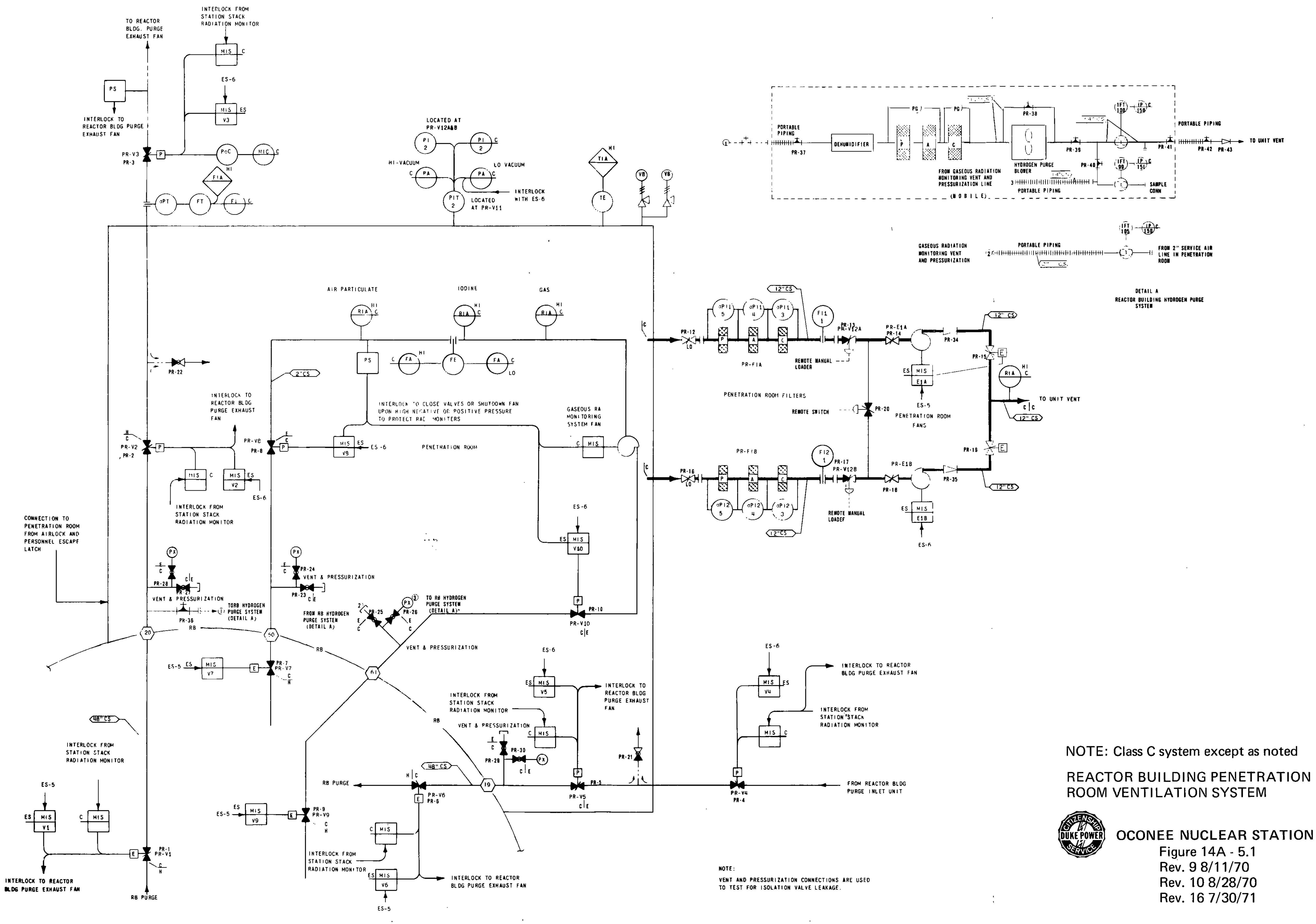
CONTROLLED PURGE FLOW REQUIREMENTS FOLLOWING MHA TO HOLD BUILDING AT 3.5% HYDROGEN



SENSITIVITY OF PURGE START TIME TO THE HYDROGEN GENERATION CONSTANT

OCONEE NUCLEAR STATION

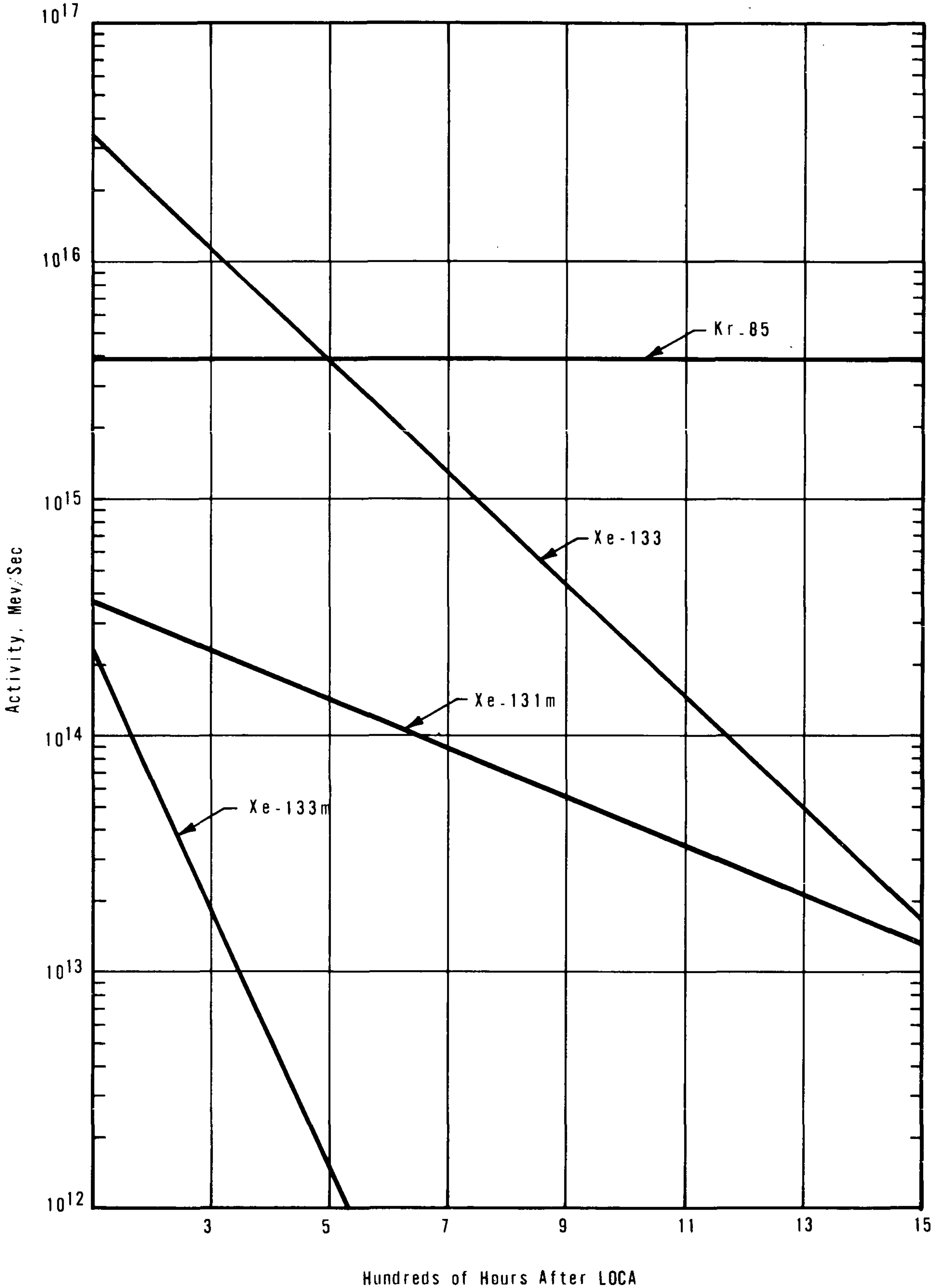
Figure 14A - 4.7



NOTE: Class C system except as noted

REACTOR BUILDING PENETRATION ROOM VENTILATION SYSTEM

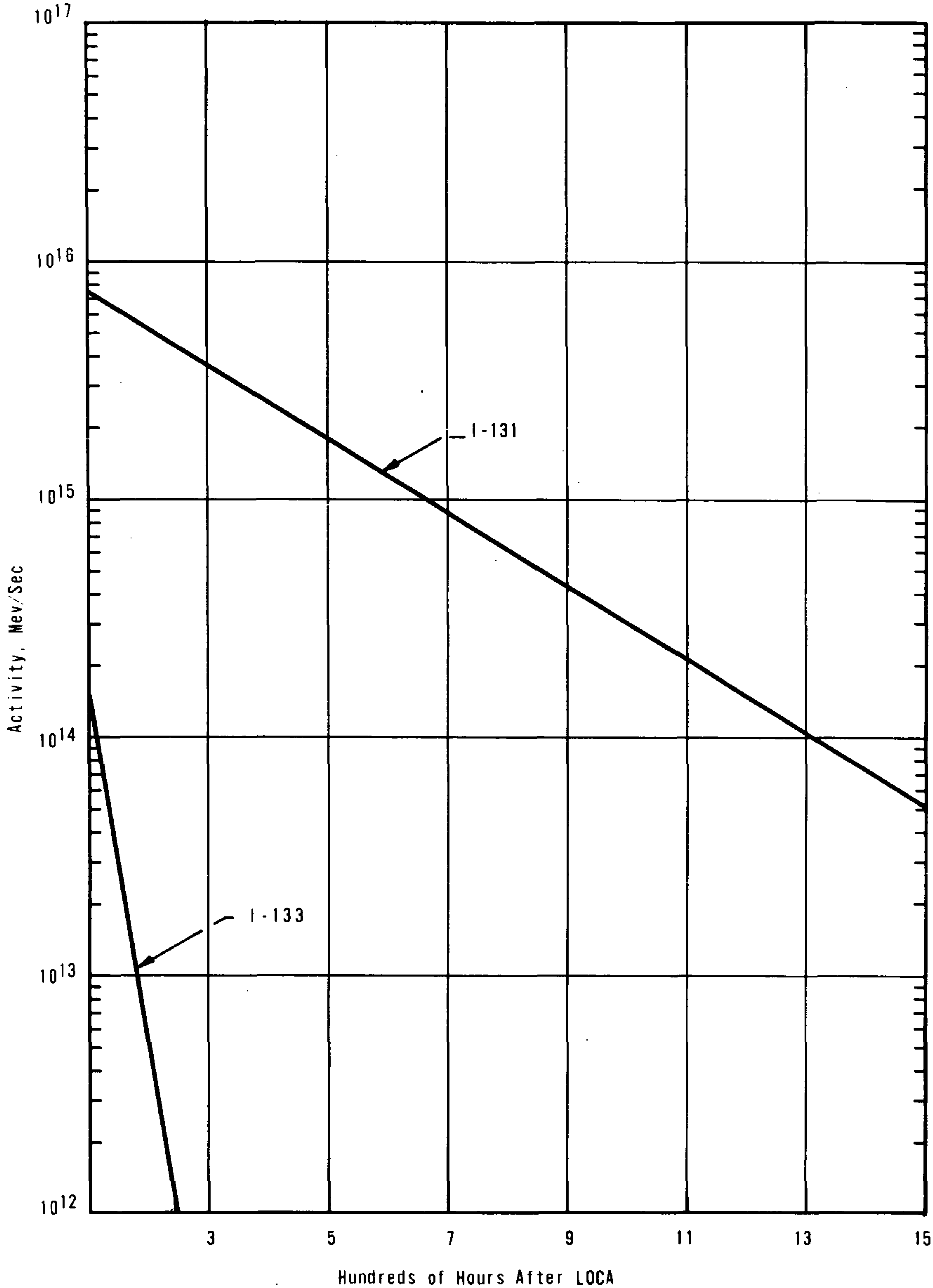
OCONEE NUCLEAR STATION
Figure 14A - 5.1
Rev. 9 8/11/70
Rev. 10 8/28/70
Rev. 16 7/30/71



NOBLE GAS ACTIVITY VS TIME AFTER LOCA

OCONEE NUCLEAR STATION

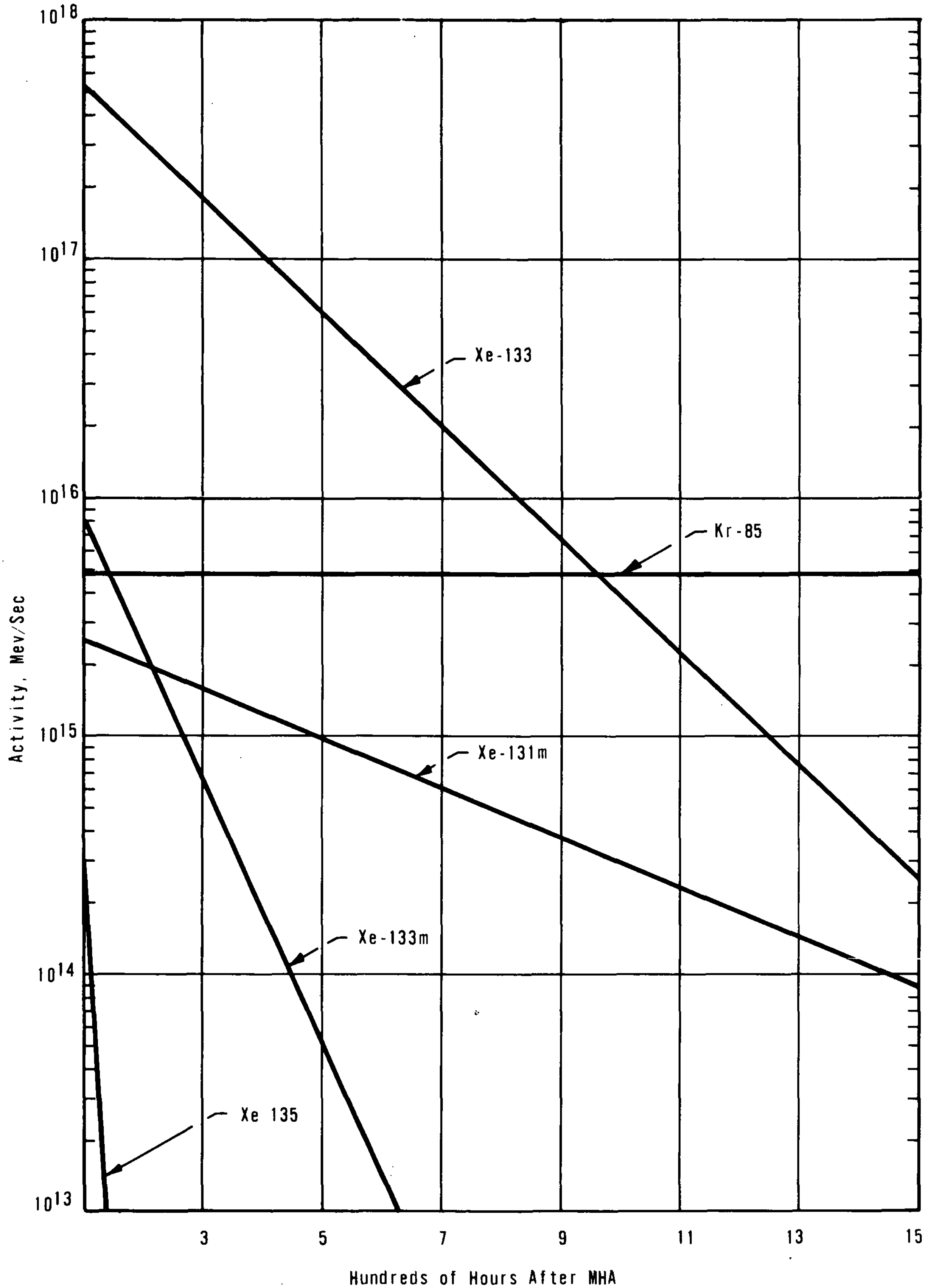
Figure 14A - 7.1



IODINE ACTIVITY VS TIME AFTER LOCA

OCONEE NUCLEAR STATION

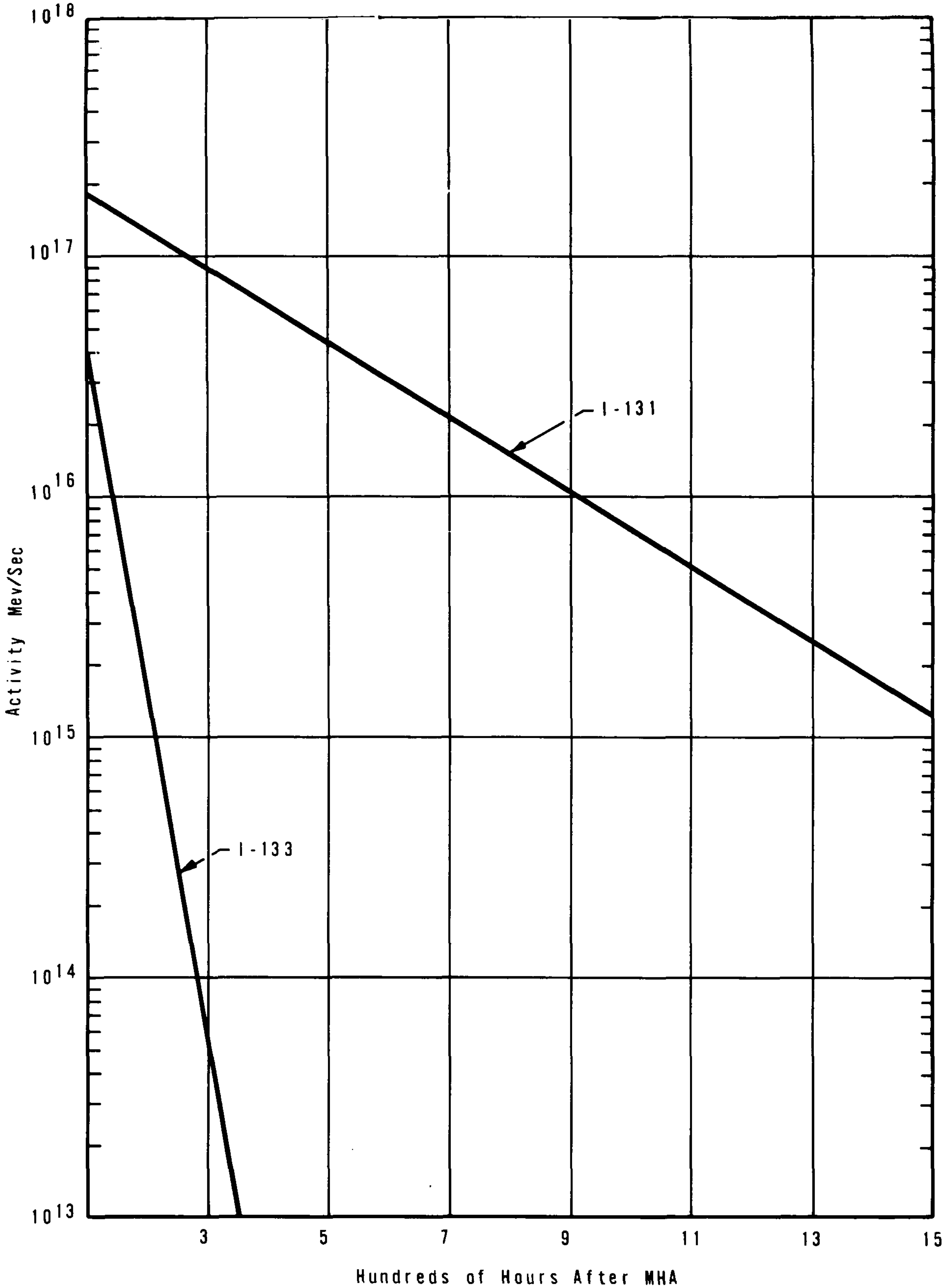
Figure 14A - 7.2



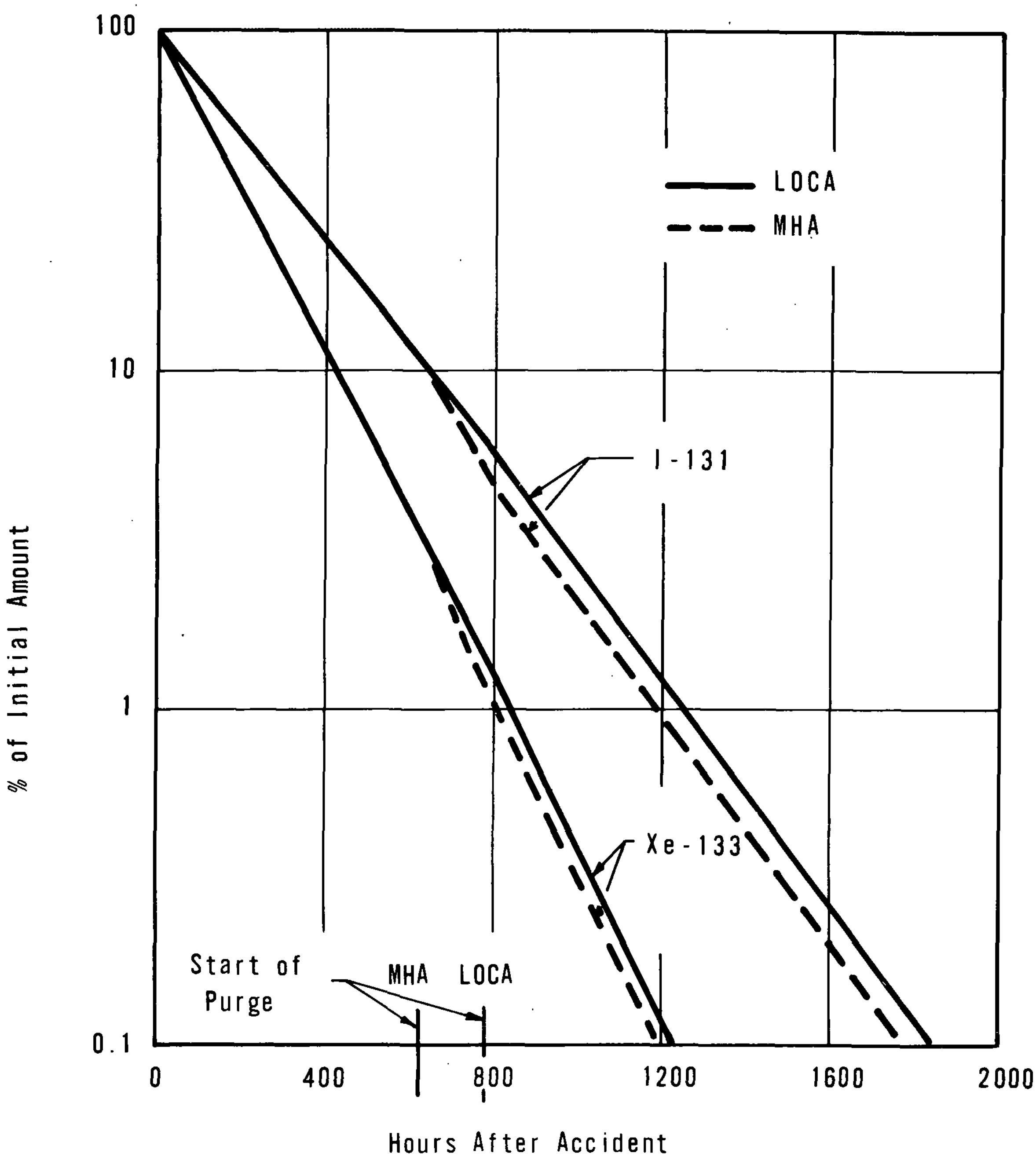
NOBLE GAS ACTIVITY VS TIME AFTER MHA

OCONEE NUCLEAR STATION

Figure 14A - 7.3



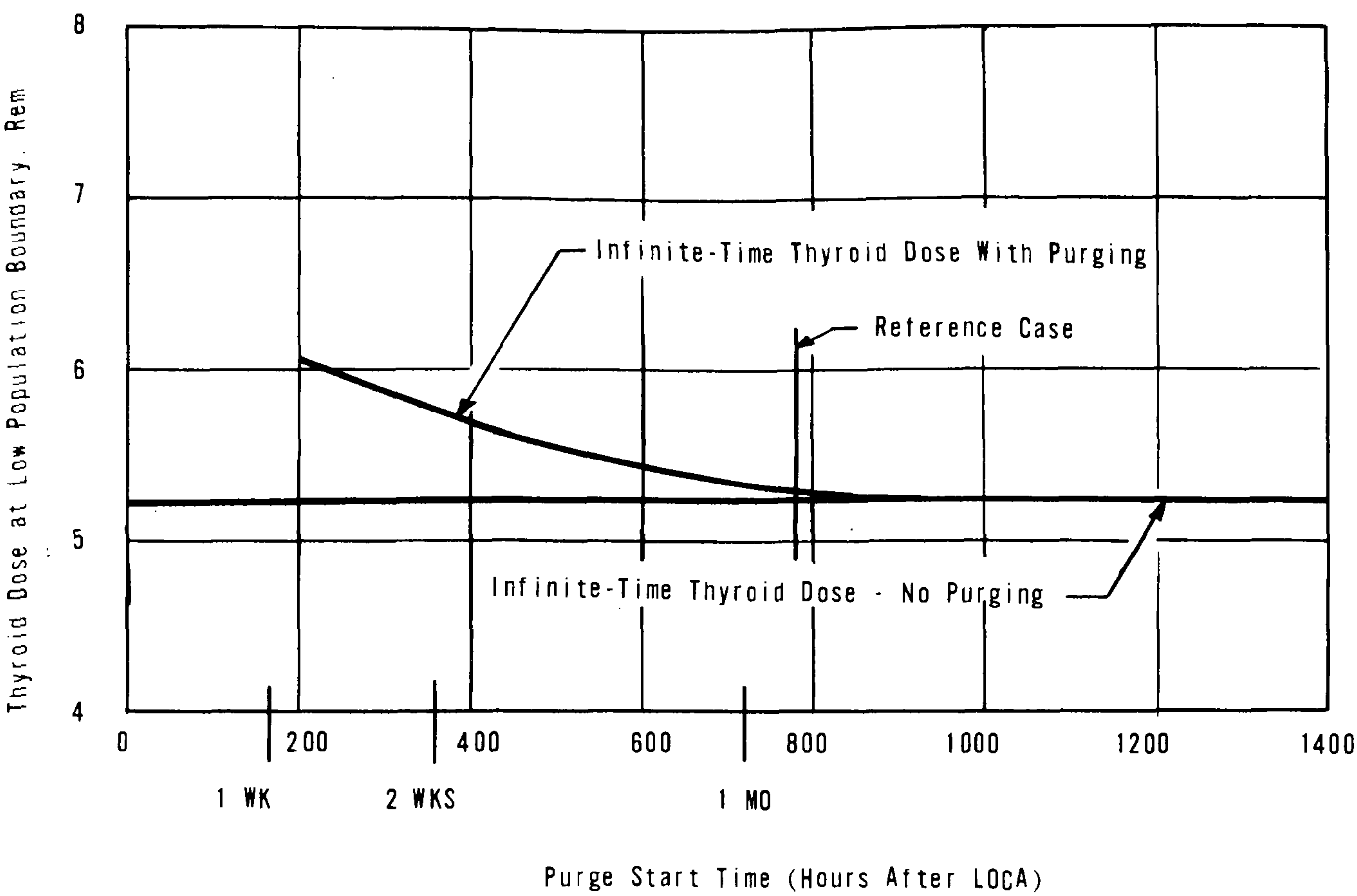
IODINE ACTIVITY VERSUS TIME AFTER MHA



COMBINED EFFECT OF PURGING AND DECAY ON REACTOR BUILDING AIRBORNE ACTIVITY

OCONEE NUCLEAR STATION

Figure 14A - 7.5

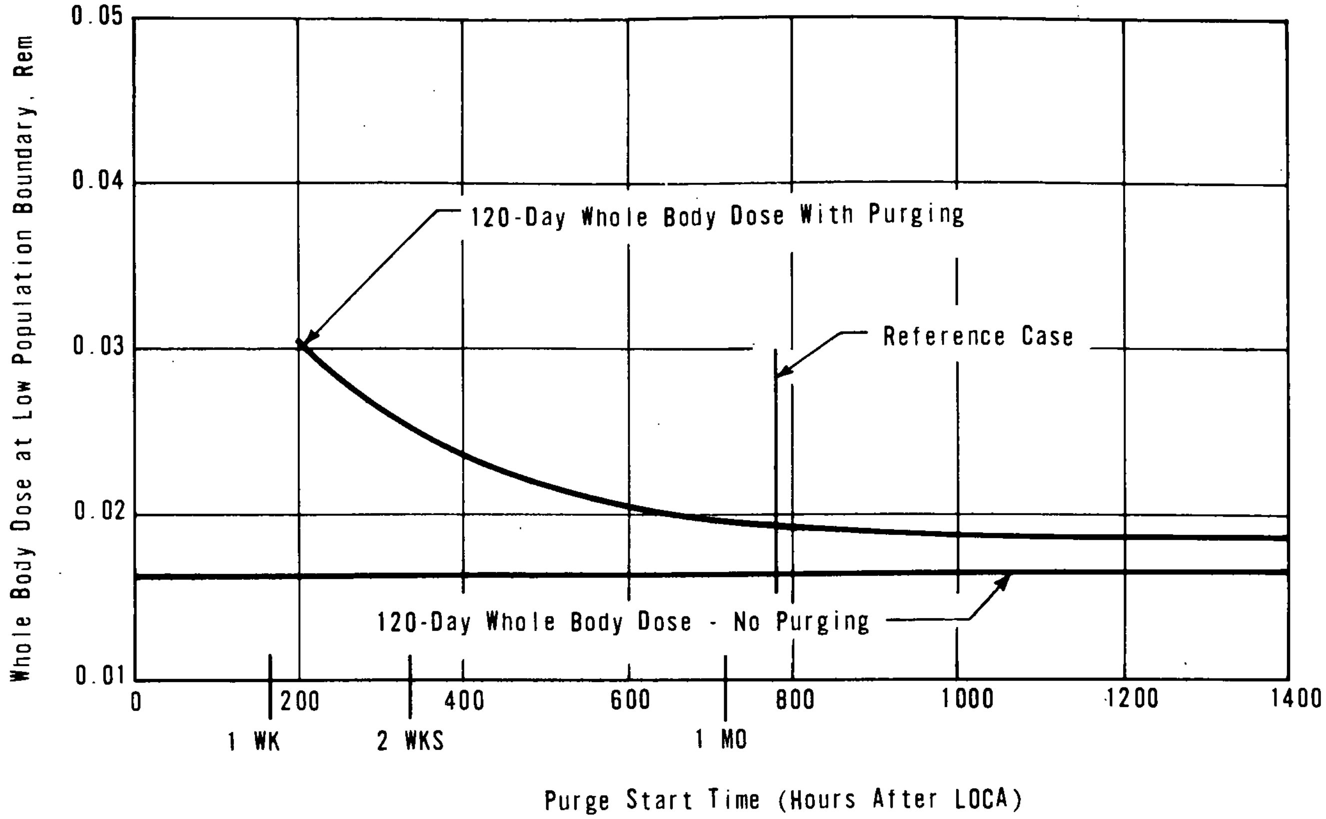


LOCA THYROID DOSE

OCONEE NUCLEAR STATION

Figure 14A - 7.6

Rev. 18 3/10/72

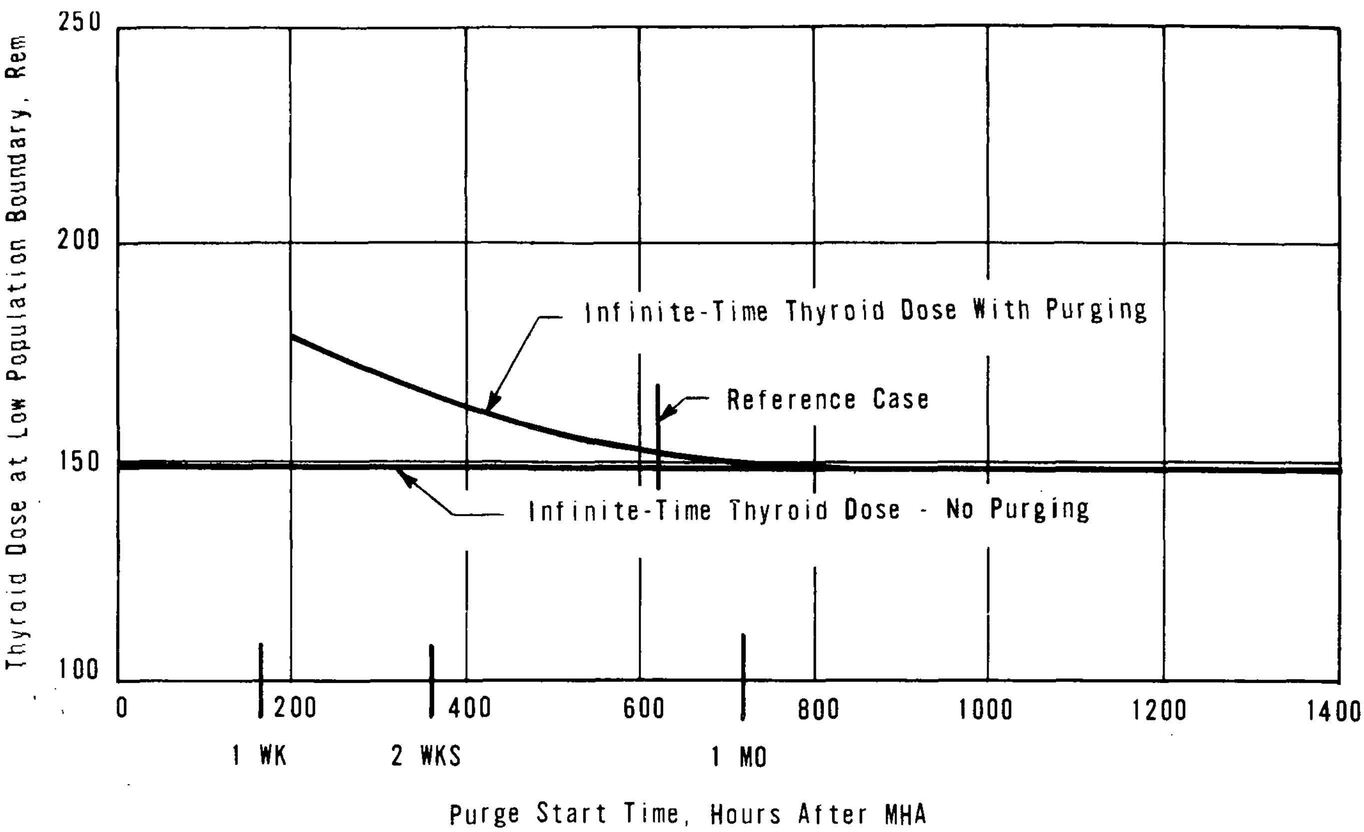


LOCA WHOLE BODY DOSE

OCONEE NUCLEAR STATION

Figure 14A - 7.7

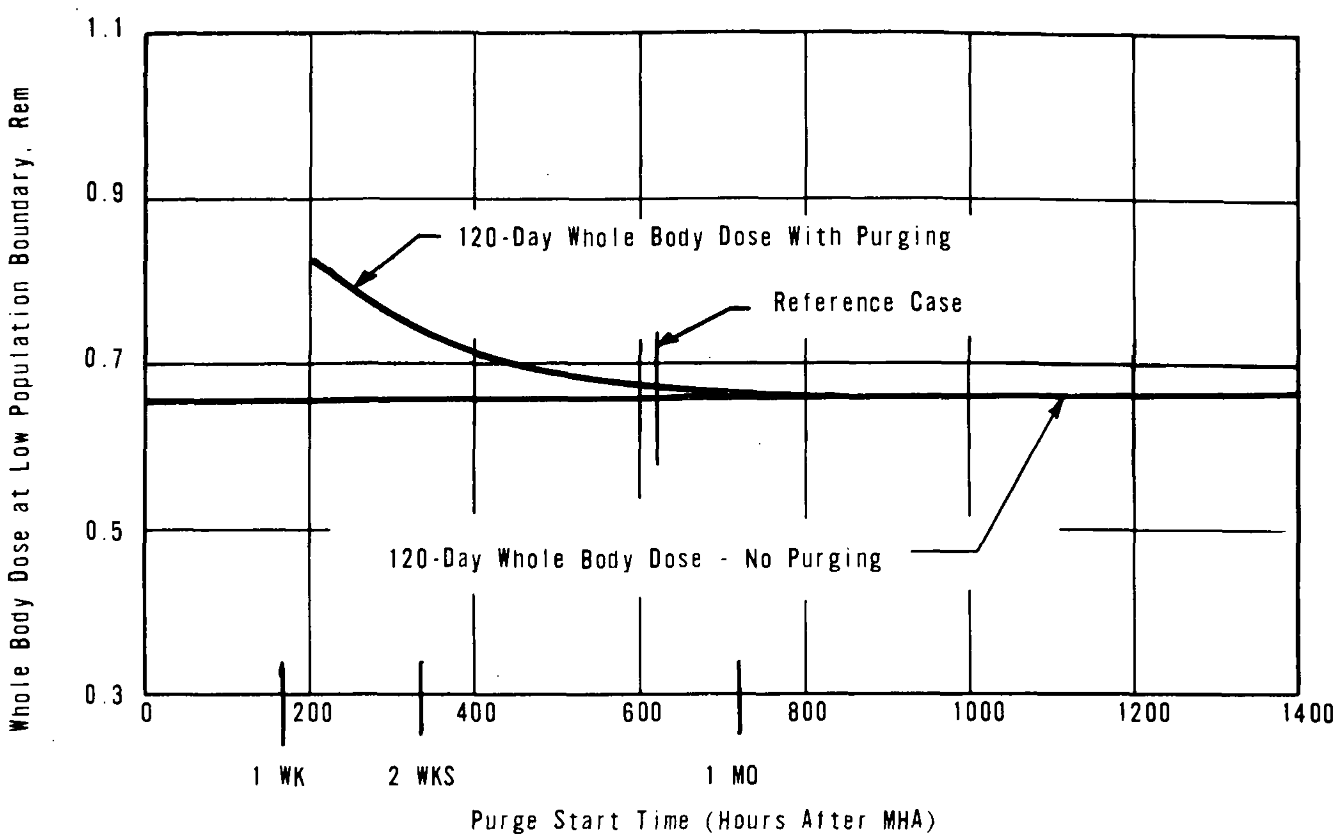
Rev. 18 3/10/72



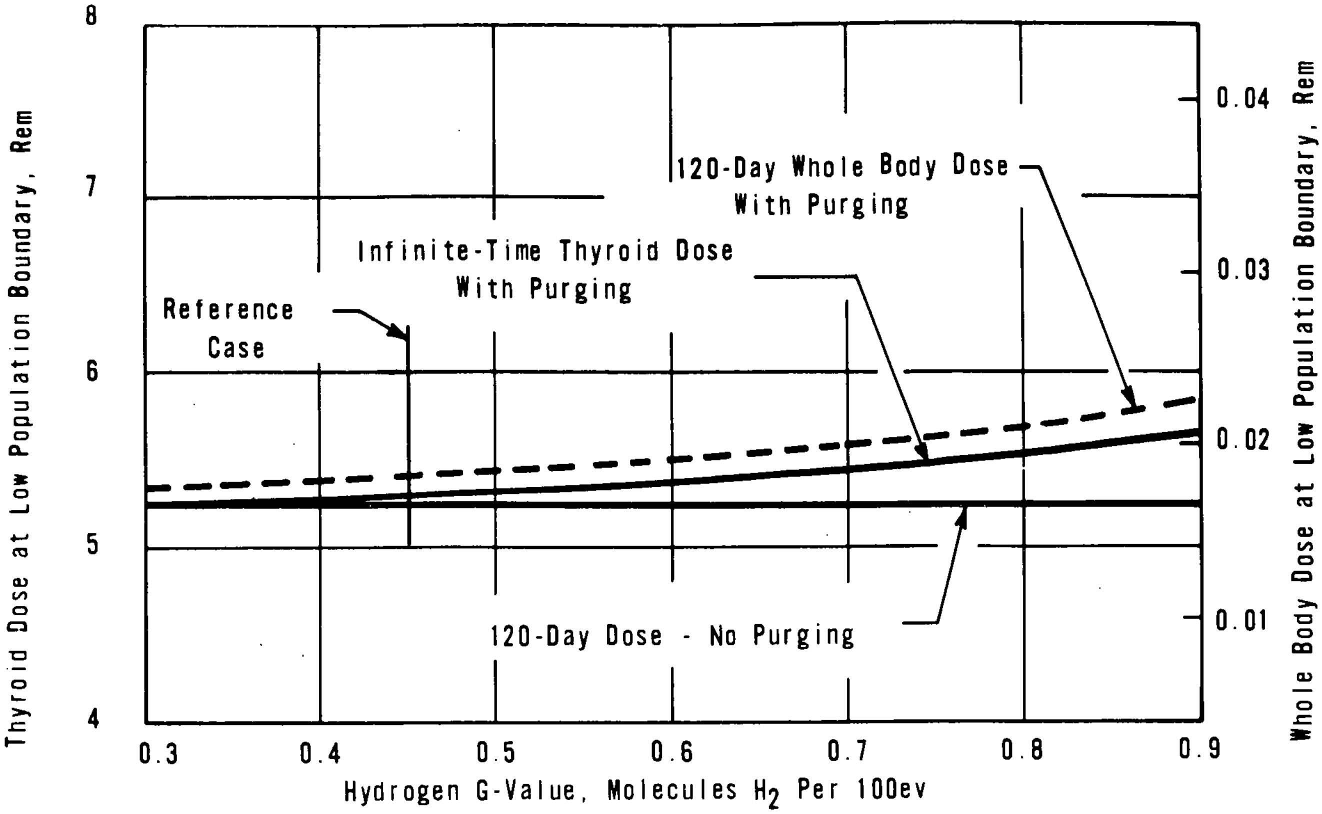
MHA THYROID DOSE

OCONEE NUCLEAR STATION

Figure 14A - 7.8



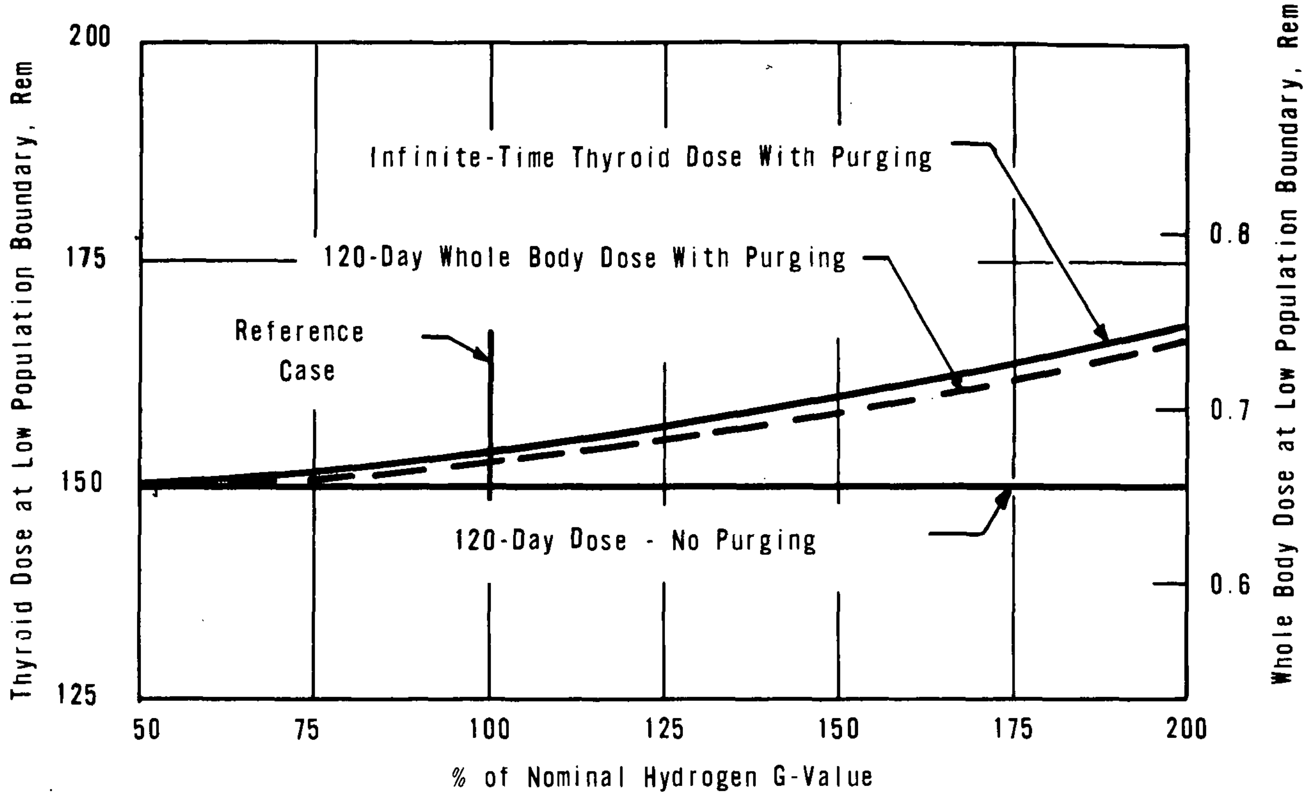
MHA WHOLE BODY DOSE



LOCA DOSE SENSITIVITY TO HYDROGEN G-VALUE

OCONEE NUCLEAR STATION

Figure 14A - 7.10

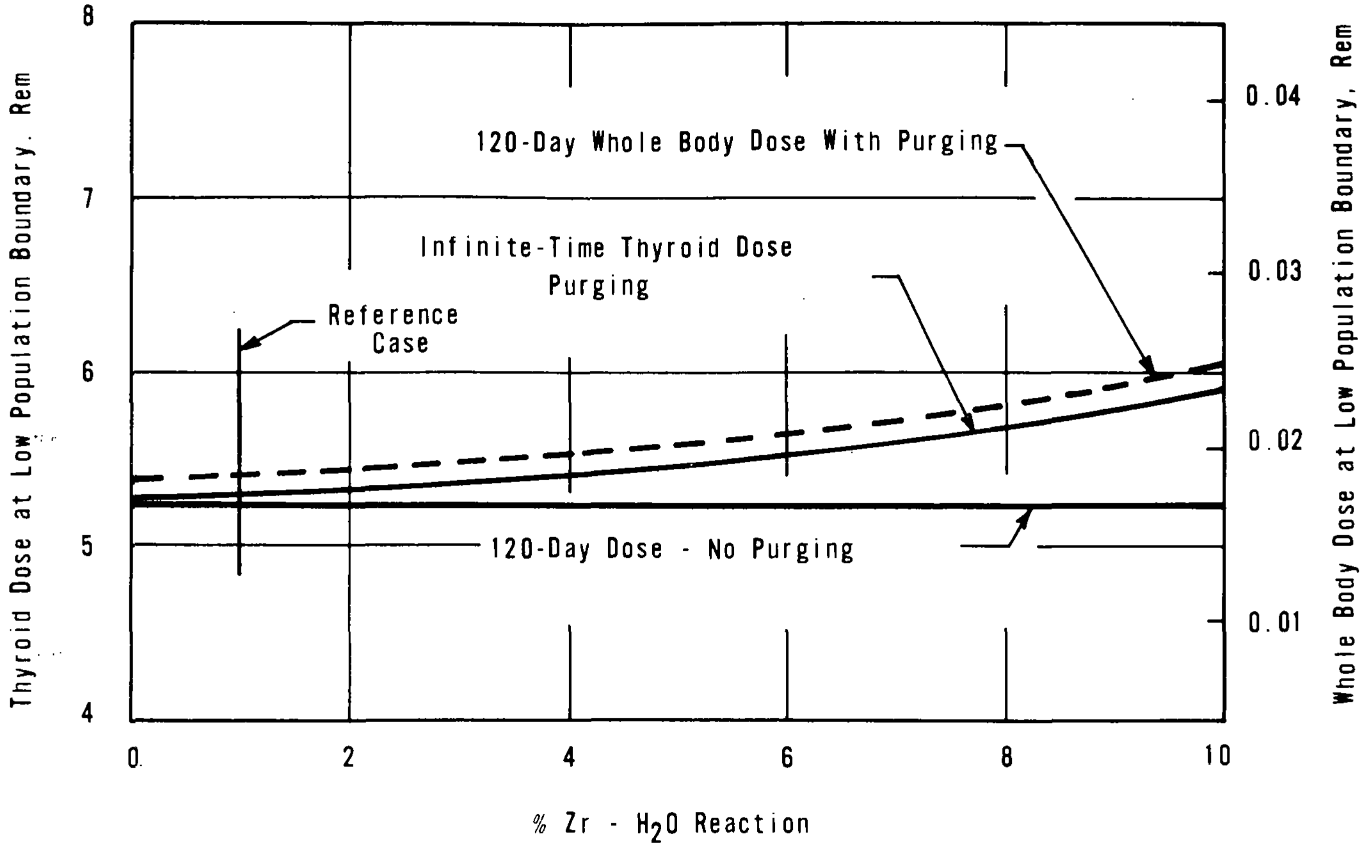


MHA DOSE SENSITIVITY TO HYDROGEN G-VALUE

OCONEE NUCLEAR STATION

Figure 14A - 7.11

Rev. 18 3/10/72

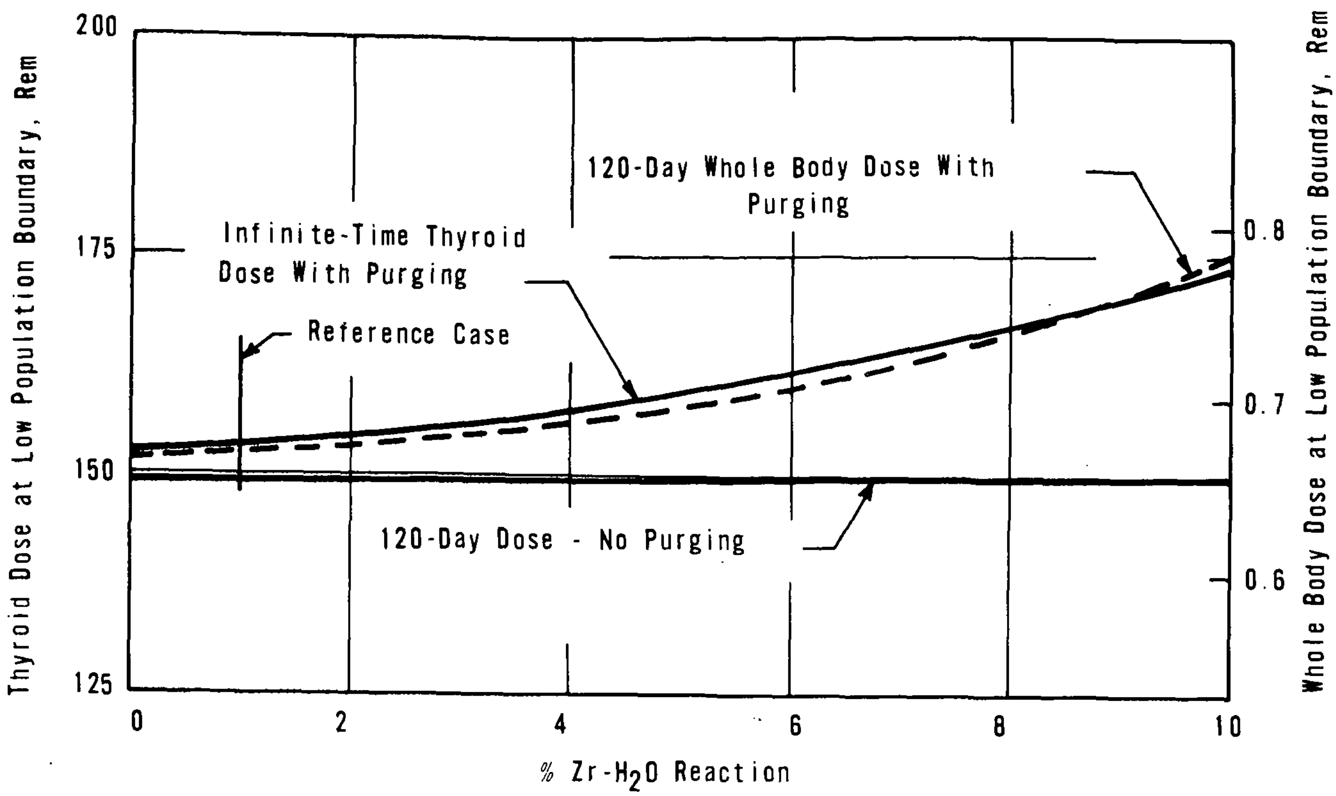


LOCA DOSE SENSITIVITY TO ZR-H_2O REACTION

OCONEE NUCLEAR STATION

Figure 14A - 7.12

Rev. 18 3/10/72



MHA DOSE SENSITIVITY TO ZR-H_2O REACTION

OCONEE NUCLEAR STATION

Figure 14A - 7.13

APPENDIX 14B

OCONEE NUCLEAR STATION

FINAL SAFETY ANALYSIS REPORT

A MULTI-NODE COMPUTER CODE ANALYSIS OF THE LOSS-OF-COOLANT ACCIDENT

Submitted with FSAR Revision No. 13

September 17, 1970

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FSAR APPENDIX 14B

Multinode Computer LOCA Analysis

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14B-ii	Original
14B-1	Original
14B-2	Original
14B-3	Original
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14B-10	Original
14B-11	Original
Fig. 14B-1	Original
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Fig. 14B-9	Original
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14B A MULTI-NODE COMPUTER CODE ANALYSIS OF THE LOSS-OF-COOLANT ACCIDENT

14B.1 INTRODUCTION

A multi-node computer code analysis of the loss-of-coolant accident has been performed using a modified version of the FLASH 2 computer code. The purpose of this report is to show how the code was used and to show that the emergency core cooling systems, as designed, protect the core from gross failure and maintain core cooling after the blowdown.

14B.2 SUMMARY AND CONCLUSIONS

The multi-node analysis of the loss-of-coolant accident was performed using a 27 region model. Results from this analysis show that the core temperature transients are somewhat higher than the ones shown in Section 14 which were based on the 3 region FLASH 1 code. However, the cladding temperature transients are terminated at values below 2300 F. The peak temperature of 2260 F occurred for a 6 ft^2 rupture in the 28-inch cold leg piping. The maximum size break in the hot leg piping, 14.1 ft^2, yielded a temperature of 1730 F. Four cold leg break sizes were examined to ensure that the worst case had been analyzed. In all cases, the core power was determined by assuming that the control rods did not enter the core although there was nothing to impede their insertion. For the larger cold leg breaks, this has little significance because core power is shutdown very rapidly on voids.

The analysis showed that the emergency core cooling system, as designed, terminated the core temperature transient and maintained core cooling after the transient.

14B.3 MULTI-NODE MODEL DESCRIPTION

The basic computer code description is described in the FLASH 2 writeup from the Bettis Atomic Power Laboratory. Several modifications have been made to the code and the major ones are described in 14B.3.1.

For this analysis, the reactor coolant system, the steam generators and the reactor building were represented by 27 control volumes plus the two core flood tanks. This representation is depicted on Figure 14B.1. The 12 control volumes located within the dotted lines represent the volumes in the reactor vessel. Volume 7 represents the downcomer, volume 2 represents the volume below the core, and volume 1 represents the volume above the core. Volume 20 represents the volume between the core and the core barrel. Volumes 17, 18 and 19 represent the volume of the core with the exception of a hot channel which is represented by volumes 21, 22, 23, 26 and 27. Each steam generator is represented by 3 volumes, two within the primary system and 1 on the secondary. These are numbered 4, 5 and 15 on one steam generator and 9, 10 and 16 on the other. Volumes 3 and 8 represent the hot leg piping, volumes 6, 11, 12, 24 and 25 represent the cold leg piping and volume 13 represents the pressurizer. Flow paths between volumes are indicated by circled numbers.

As explained in the FLASH 2 writeup[(1)], the core calculations are performed in the core flow paths. The heat generated in these core flow paths is added into the control volume into which the path is flowing. Referencing Figure 14B-1, the average core heat generation is calculated in paths 1, 2, 3 and 4. The heat is added into volumes 17, 18 and 19 and 1, respectively, if all paths are flowing in the direction from 2 to 1. If the flow in each of these flow paths reverses, the heat from paths 4, 3, 2 and 1 is added into 19, 18, 17 and 2, respectively.

The same principle holds true for the hot channel which is represented by 5 volumes and 6 flow paths. The heat flux in each of these core flow paths is determined from the axial flux shape that has a 1.7 peaking factor, three feet above the bottom of the core. The hot spot in the core is represented by a one-foot section of the hot channel represented by path number 6 which is located between control volumes 21 and 22. All of the reactor heat generation takes place within the fuel. Actually, between 2 and 3 percent of the heat is generated directly in the coolant and this heat is not transferred through the cladding. Therefore, the heat generation at the hot spot which was used in this analysis has a linear heat rate 2.5% higher than the one used in the analysis shown in 14.2.2.3.4.

The representation of the hot channel and hot spot does not allow any mixing between channels. That is, there can be no flow between volumes 17, 18 or 19 and the hot channel. The total hot channel flow does in effect have the same overall pressure drop as the average core flow but the flow in any individual path is based on the pressure in the control volumes to which it is connected.

The power is calculated directly in the code and no iterations are now required between the code and some other reactor kinetics code. Similarly, the hot channel hot spot cladding temperature is calculated directly within the code. This increased capability in the code reduces the possibility of error in going from one code to another.

14B.3.1 MODIFICATIONS MADE TO THE FLASH-2 CODE

Numerous changes have been made to the FLASH 2 code over a period of approximately 2-1/2 years. Of these changes, thirteen stand out as being major modifications. These modifications have improved the programs accuracy and stability. They have also made the program more applicable for use in analyzing the B&W nuclear steam supply system. A brief description of these modifications is given below.

14B.3.1.1 Program Capacity

The original version of FLASH 2 was limited to 20 control volumes and 40 flow paths. The code has been increased in size so that it now can have as many as 40 control volumes and 80 flow paths. This modification provides the capability to perform a more detailed simulation of the loss-of-coolant accident.

14B.3.1.2 Pressure Calculation

The pressure calculation in FLASH 2 involved interpolation in stored tables of water properties. This method was believed to be the source of unstable pressure oscillations. A new method which uses surface fit equations of the 1967 ASME steam tables was programmed into the code. This method reduces many of the instabilities observed in the original version of FLASH 2.

14B.3.1.3 Addition of Core Flood Tanks

The capability to describe the performance of the core flood tanks was incorporated into the program. Flow from each core flood tank is calculated from the conservation of momentum equation and the tank pressures are calculated from the ideal gas laws applied to the nitrogen cover gas.

14B.3.1.4 Addition of Nozzle Height to Pipe Paths

In the original FLASH 2 program, flow paths entering and exiting a control volume were assumed to be at a point. This caused some problems when the mixture height in a control volume approached the elevation of this point. Flow out of the control volume would intermittently switch from steam to mixture as the mixture height oscillated around the point.

The B&W version of the program includes a nozzle height rather than a point. The quality and enthalpy of the fluid entering a flow path is determined by the location of the mixture height relative to the nozzle height. This change provided a smooth transition from the flow of liquid to the flow of steam and prevented pressure oscillations which occurred in the original version of FLASH 2.

14B.3.1.5 Reverse Flow Pressure Drop

When the flow reverses in flow paths which contain expansions and/or contractions, the flow resistance factors are altered. This is done by inputting flow resistance factors for reverse and forward flow in these paths.

14B.3.1.6 Two-Phase Pressure Drop Addition to Core Flow Paths

In order to correspond more closely with two-phase flow pressure drops, the method of calculation of the frictional pressure drop was changed for the core flow paths. Thom's(2) correlation is used for pressures equal to or greater than 500 psia. Below 500 psia, the Martinelli-Nelson(3) correlation is used. In addition, a K-factor correction term was added to take into account changes in the Reynolds number.

14B.3.1.7 Addition of Critical Heat Flux Correlation

The original FLASH 2 program contained a simple three-parameter fit critical heat flux correlation into which the user had to supply 3 constants. In order to ensure that the critical heat fluxes were calculated in a conservative manner, the W-3(4) correlation was added to the program for use with qualities up to 15% and the correlation of Jansen, et al(5) was added for use with qualities between 15% and 45%. Above 45% quality, DNB is assumed to have occurred.

14B.3.1.8 Modification to Film Boiling Heat Transfer

A modified version of the Sieder-Tate correlation as developed by Quinn(6) was added. The justification for the use of this correlation is covered in 14B.3.2.1.

14B.3.1.9 Fuel Pin Conduction Model

The single node fuel pin model in the original FLASH 2 program has been replaced with a 2 node conduction model similar to the one used in the QUENCH model described in 14.2.2.3.3. This model calculates average fuel and clad temperatures as well as the clad surface temperature. All of the nuclear heat is added to the fuel and the metal-water reaction heat is added to the cladding.

14B.3.1.10 Conservation of Energy in the Core

The method of adding heat from the core in the original FLASH 2 code caused a slight energy imbalance over the duration of the blowdown. The method of adding heat to the fluid has been changed so that the heat from any core flow path is added directly into the control volume in the direction of the flow in that path. This change assures conservation of energy.

14B.3.1.11 Reactivity Calculations

The reactivity addition due to changes in the moderator density was calculated in FLASH 2 using a constant reactivity coefficient. This has been replaced with a table of reactivity as a function of the percent of the steady-state core water density. If more than one core path is used (which is the case in the model described in this section), reactivity weighting factors are assigned to these paths. These weighting factors are also used with the Doppler reactivity calculations. The sum of the weighted moderator and Doppler reactivities is used to get the total core reactivity. This modification has eliminated the need for iterating between this code and some reactor kinetics code.

14B.3.1.12 Steam Generator Model

The ability to input feedwater flow as a function of time and pressure has been added. The steam flow does not now have to match the feedwater flow. The primary side has been broken up into two control volumes for each steam generator instead of one. This modification provides for improved secondary and primary fluid conditions and, therefore, results in more accurate heat transfer calculations.

14B.3.1.13 Non-Equilibrium Model

The original FLASH 2 model assumed that the steam and water in a control volume were always in thermodynamic equilibrium. The model was therefore unable to describe the flow of subcooled water through the system in the presence of saturated steam and gave inaccurate predictions of flows during reflooding.

The capability to calculate the mass and energy transfer between the water and steam during flooding has been added. This model goes through separate thermodynamic calculations for the water and the steam in a control volume. This model gives more realistic flows through the system and provides a more realistic simulation of core flooding.

14B.3.2 JUSTIFICATION OF CORE HEAT TRANSFER

In 14B.3.1.8, reference was made to changes which were made to the heat transfer models in FLASH 2. In addition to adding the W-3 correlation to determine the time of departure from nucleate boiling, changes were made in the method of calculating film boiling heat transfer after DNB. The film boiling heat transfer coefficient which was chosen has been checked against all of the applicable data which is available to us. This will be discussed in the following section. B&W has also followed the core flooding tests, FLECHT, very closely and these tests have confirmed that the heat transfer coefficients being used in the program during reflooding are conservative. B&W's evaluation of these tests will be covered in 14B.3.2.2.

14B.3.2.1 Heat Transfer During Blowdown

For nucleate boiling, the cladding surface temperature is calculated using the Jens-Lottes[7] correlation. Departure from nucleate boiling is determined by making use of the W-3 correlation. For post DNB film boiling heat transfer, the correlation used in Section 14 of the FSAR is still being used. The heat transfer coefficient is determined from Quinn's modified form of the Sieder-Tate Equation for dispersed flow film boiling. Once the two-phase mixture gets to steam qualities equal to or greater than one (1), this expression reduces to the Sieder-Tate equation which is then used for heat transfer to the steam.

For dispersed flow film boiling, Quinn's model is a two-step model where energy is first transferred from the wall to the vapor and then from the vapor to the liquid droplets. The form of the equation that is being used is the same as that shown on Page 14-43 of the FSAR. All of the transport properties were evaluated using the 1967 ASME steam tables. Figure 14B-2 shows the conservatism of this correlation by comparing experimentally determined value of the heat transfer coefficient with the calculated values. Data from four different sources are shown on this figure. The data from GEAP-3703[8] were taken over the following range of conditions:

P - 800, 1100, 1400 psia

G - 0.75 - 1.5 x 10^6 lb/hr-ft^2

X - 0.17 x 0.85

de - 0.240 in.

Data from GEAP-5563[9] were obtained from a two-rod test section over the following range of conditions:

P - 600, 1000, 1400 psia

G - 0.5 - 1.95 x 10^6 lb/hr-ft^2

X - 0.2 - 1.0

de - 0.406 in.

The data from AECL-3281[10] was taken over the range of:

P - 600, 1000, 1200 psia

G - 1.0 - 3.0 x 10^6 lb/hr-ft^2

X - 0.15 x 0.60

de - 0.160 in.

Also shown on Figure 14B-2 is the comparison based on the limited bundle data of Matzner, et al[11]. These data covered the range of:

P - 500, 750, 1000, 1200 psia

G - 0.5 - 1.0 x 10^6 lb/hr-ft^2

X - 0.35 - 0.7

de - 0.336 in.

An examination of the figure shows that the experimentally determined values for the heat transfer coefficient are on the average 1.5 to 2 times the calculated values.

14B.3.2.2 Heat Transfer During Reflooding

The objective of the PWR Full Length Emergency Cooling Heat Transfer (FLECHT) test program[12] was to obtain experimental data for use in evaluating the heat transfer capabilities of pressurized water reactor emergency core cooling systems during reflooding following a simulated loss-of-coolant accident. The FLECHT results to date can be summarized as follows:

1. Water is entrained in the steam which is generated near the bottom entrance and is carried up through the core.

2. Higher initial cladding temperatures result in less of of a temperature rise. The first Group III test (Run 8874 performed on July 21, 1970) started at a midplane temperature of 2320 F, and the peak temperature was 2460 F. This is the highest temperature run to date.

3. Higher flooding rates reduced the temperature rise.

4. Increased subcooling increased the temperature rise slightly.

5. Increased test pressure reduced the temperature rise.

6. Flow blockage resulted in improved heat transfer immediately in the vicinity of the blockage. This improvement is a result of improved heat transfer from the steam to the suspended droplets due to the droplet breakup.

7. 2000 ppm boric acid improved the heat transfer coefficient.

8. Fallback of water droplets reduced the temperature slightly.

9. The responsibility between test runs has been excellent.

Group I data has been correlated to give a better understanding of the system parameters on the ability to cool the core with bottom flooding. The first expression relates the maximum midplane temperature rise to the system parameters.

$$\Delta T_{peak} = T_{peak} - T_o$$

$$= \frac{1.43 \times 10^9 \times Q^{1.65} \times \Delta T_{sub}^{0.066}}{p^{0.15} \times V^{1.07} \times T_o^{2.0}}$$

where

Q = maximum linear heat rate, kw/ft

p = operating pressure, psia

T_o = midplane temperature at start of reflooding, F

T_{peak} = midplane peak temperature, F

V = flooding rate, in/sec

ΔT_{sub} = subcooling of flooding water, F

The second correlation illustrates the effect of the same system parameters on the time to temperature turn around at the midplane.

$$\Delta t_{peak} = t_{peak} - t_o$$

$$= \frac{8.84 \times 10^9 \times Q^{0.59} \times \Delta T_{sub}^{0.052}}{p^{0.41}\ V^{1.4}\ T^{2.3}}$$

where

t_o = time at which reflooding water started entering the core, sec.

t_{peak} = time of peak cladding temperature, sec.

The correlation for time to midplane quench is

$$\Delta t_{quench} = t_{quench} - t_o$$

$$= \frac{1080 \times Q^{0.544}\ T_o^{0.15}}{p^{0.26} \times V^{0.70} \times \Delta T_{sub}^{0.32}}$$

where t_{quench} is the time at which midplane quench occurred. The standard deviations about the mean measured to predicted ratio are:

Parameter	σ	Mean
ΔT_{rise}	17.9%	1.01
Δt_{peak}	29.8%	1.04
Δt_{quench}	13.3%	1.00

The flow blockage results were unaffected in that the simulated flow blockage improved the heat transfer immediately downstream of the blockage. The flow blockage was prodiced by installing a 3/8 inch-thick stainless steel plate 1 inch below the midplane thermocouples. The flow blockage effect on the temperature at the midplane of the hottest rod is given in Figure 14B-3. A dramatic rise in the heat transfer coefficient at the inner rod was noted for the blocked cases. This improved heat transfer could be due to the breakup of droplets passing through the plate and to increased turbulence in this region due to contraction and expansion. It should be noted that there was no apparent change in the heat transfer for the outer channels. Also, there was no apparent change in the heat transfer for the blocked channels at the four and eight-foot elevations. Further, INC tests indicate that the use of a flat plate is a satisfactory method for simulating clad swelling effects when the swelling occurs near one plane in the test assembly.

Several variable flow tests have been run, however, and run 8874 is of particular interest to us since it used flooding rates similar to the ones used in the analysis in Section 14 of the FSAR. The test is summarized in Table 14B.1. The failure detection system installed for this test (power monitors) indicated that five heater rods failed at the following times during the test:

Rod	Time, sec
4B	9.6
5B	18.5
4C	19.3
5C	29.3
5D	55.5

Only one of the failed rods was instrumented with thermocouples and its midplane thermocouple failed at approximately 12-14 seconds after flooding. The behavior of this rod appeared to be normal up to the time of thermocouple failure. Its peak temperature was 2330 F in 4 seconds and this dropped to 2149 F at 11 seconds.

The cooldown rate after turnaround, approximately 3-10 seconds after flooding started, was generally consistent among all thermocouples. The initial portion of the test was not affected by the five heater rod failures, thus the test results are considered to be valid for at least 18 seconds. Although the complete analysis of the data taken for run 8874 is not available, it indicates that the heat transfer coefficients for the predicted flooding rates are better than the values used in the analysis which was presented in Section 14 and in this work.

14B.4 CORE COOLING ANALYSIS

The Babcock & Wilcox version of the FLASH 2 code was used to analyze the Oconee reactor coolant system's behavior following an assumed rupture in the hot and cold leg piping. The peak clad temperature was calculated by assuming design tolerances between the fuel and its cladding. In addition, no credit was taken for rod insertion and a positive moderator coefficient was used to determine reactor shutdown. The results from this analysis are presented below.

14B.4.1 HOT LEG RUPTURES

The largest hot leg pipe rupture occurs for the double-ended rupture of the
16. 36-inch diameter pipe. This results in an area of 14.1 ft^2. This break results in the fastest blowdown time and, as stated in the FSAR, is the break that was used for sizing the core flood tanks.

The analysis shows that the flow through the core is very good early in the blowdown and this results in good heat transfer. The peak clad temperature is 1730 F and this transient curve is shown on Figure 14B-4.

14B.4.2 COLD LEG RUPTURES

The largest cold leg pipe break results in an area of 8.55 ft^2. For this break the core flow reverses and goes into an oscillatory state which lasts for several seconds. During these oscillations, the minimum heat transfer coefficient is assumed to be 10 Btu/hr-ft^2-F even though larger values could be justified. The peak clad temperature for this break is 2220 F as shown on Figure 14B-5. Average core flows for the cold leg break are shown in Figures 14B-7 thru 14B-10.

For all size breaks, this analysis assumed a discharge coefficient of 1.0 on the leak flow. Actually, blowdown testing at Idaho has shown that the discharge coefficient is actually somewhere between 0.4 and 1.0 depending on the state of the fluid and geometric factors concerning the break and its location. Therefore, since the cold leg break seems to be the controlling break in this analysis, two other break sizes were run. These were the 4.276 ft^2 (cross sectional area of the pipe) and a 6 ft^2 break. Both of these cases were run with a discharge coefficient of 1.0. These two runs could be looked upon as a 8.55 ft^2 break with a discharge coefficient of 0.5 and a 8.55 ft^2 break with a discharge coefficient of 0.7. For purposes of this analysis, the former method was chosen. Figure 14B-6 shows the peak clad temperature as a function of the break area. Two curves are shown on this figure. The upper curve is the one that assumes that the pin with design tolerances happens to be located at the hottest spot in the core. The lower curve is the curve which is obtained if the nominal fuel-clad gap tolerance were used on the hot pin. Even in the case with design tolerances, the peak clad temperature is held below a value of 2300 F and is shown to be quickly cooled down. The peak clad temperature is shown to occur for a 6 ft^2 rupture at a value of 2260 F.

Table 14B.1

Preliminary Data for the Hot Midplane of Run 8874

Flooding rate, inches/sec	6.0 1.0	$0 < t < 8$ $t > 8$
Peak power, KW/ft	1.24	
Pressure, psia	64	
Inlet Coolant Temperature, F	141	
Initial Cladding Temperature, F	2320	
Peak Cladding Temperature 1st Peak, F	2360	
Turnaround time 1st peak, sec	3	
Minimum Cladding Temp. Between Peaks, F	2030	
Minimum time, sec	11	
Peak Cladding Temperature 2nd Peak, sec	2075	
Turnaround time 2nd peak, sec	23	
Quench temperature, F	approx. 1125	
Quench time, sec	approx. 200	

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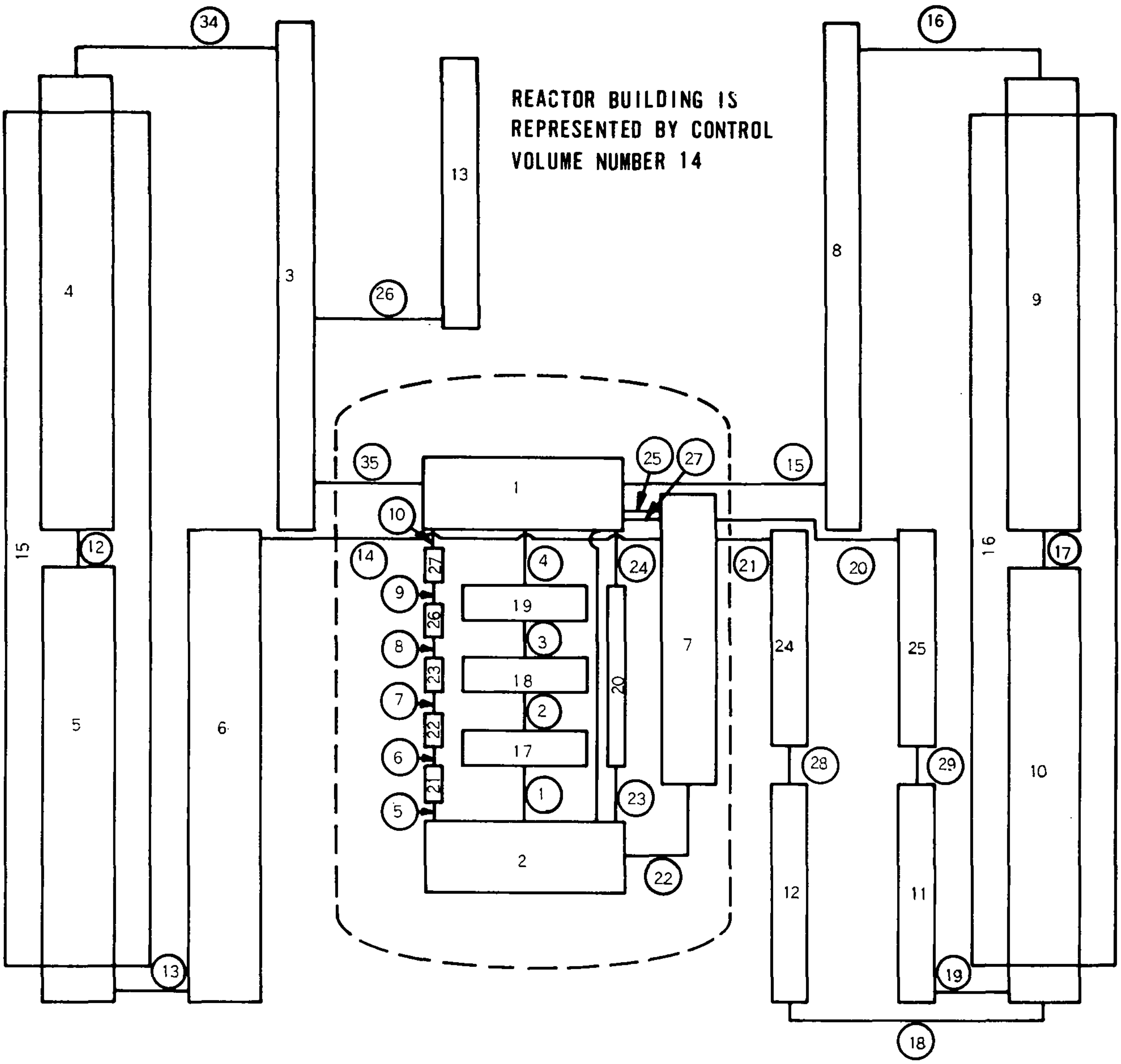
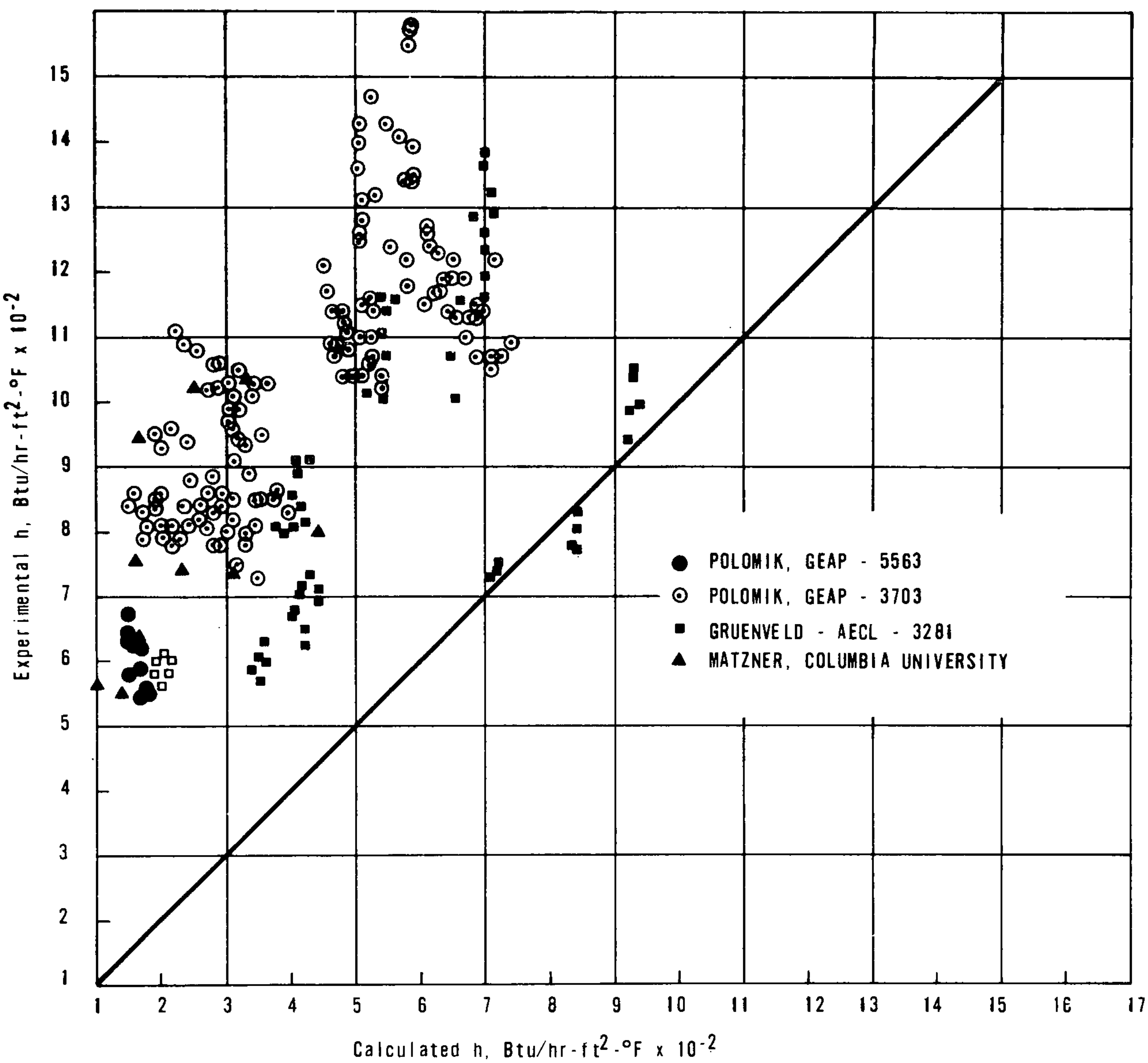
(12) Cermak, J. O. et. al., PWR Full Length Emergency Cooling Heat Transfer (FLECHT) Group I Test Report, WCAP-7435, January, 1970.



MULTINODE REPRESENTATION OF THE
REACTOR COOLANT SYSTEM

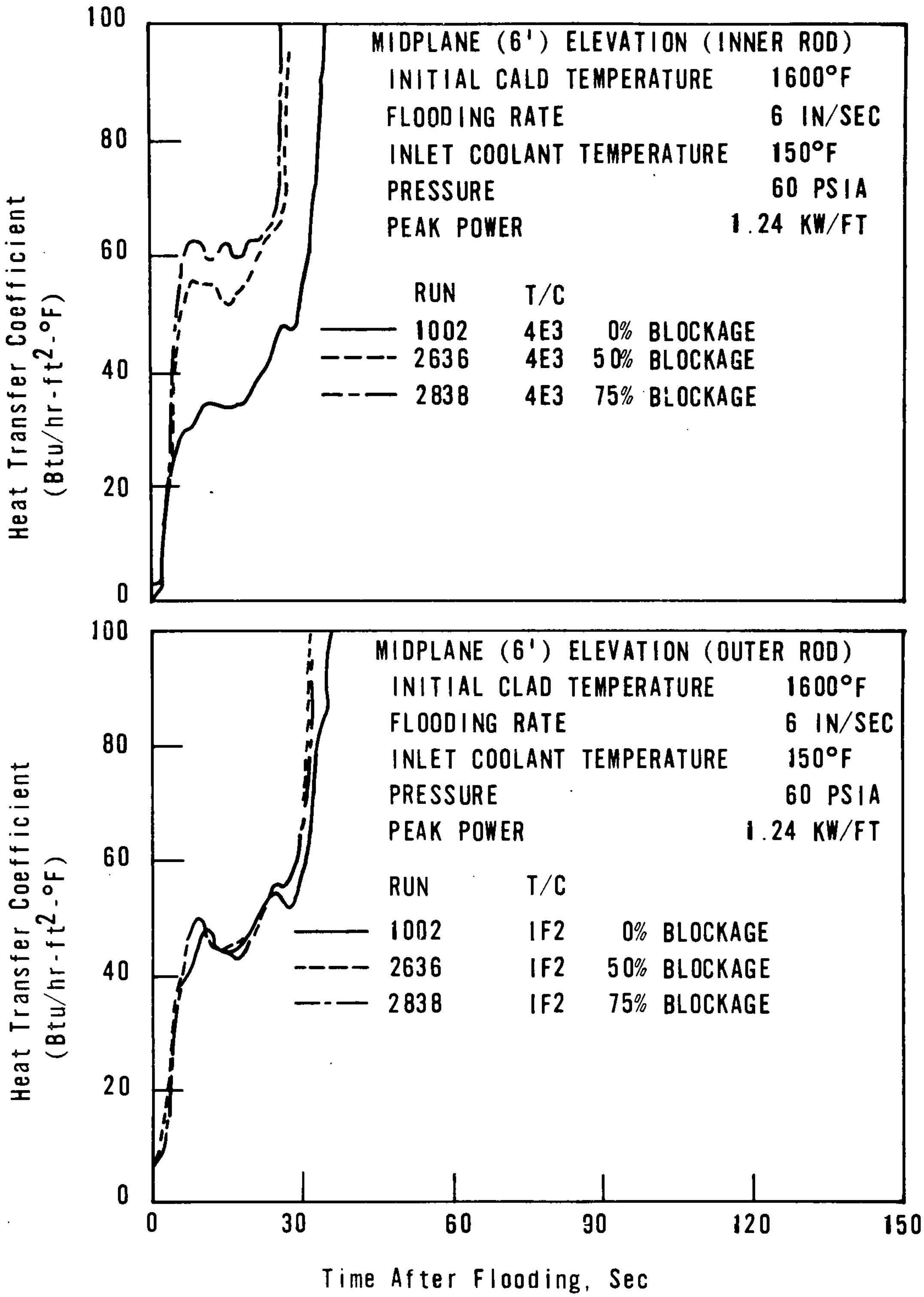
DUKE POWER
CITIZENSHIP SERVICE

OCONEE NUCLEAR STATION
Figure 14B-1



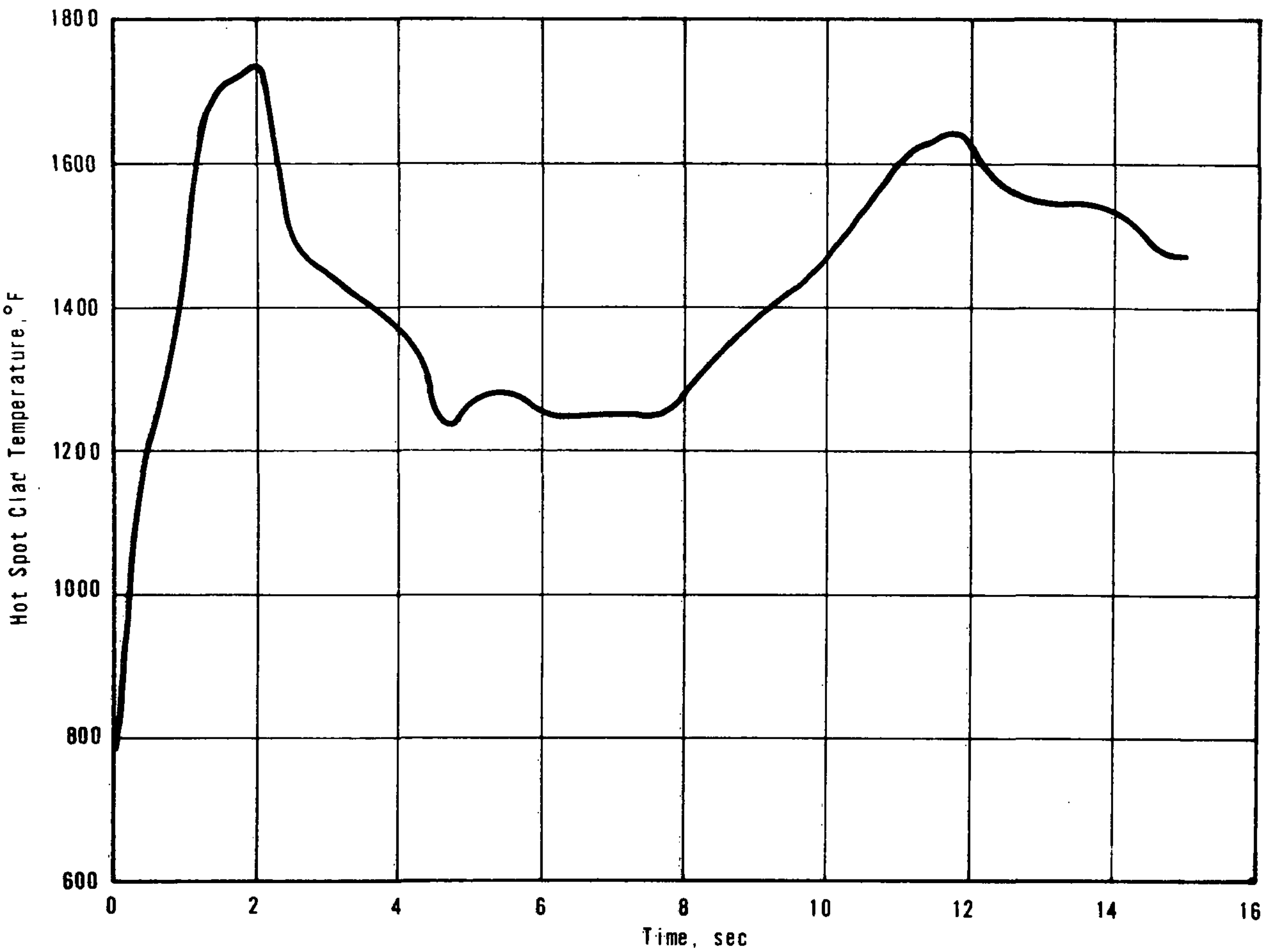
COMPARISON OF EXPERIMENTAL DETERMINED VALUES AT THE HEAT TRANSFER COEFFICIENT WITH THE CALCULATED VALUES

OCONEE NUCLEAR STATION
Figure 14B-2



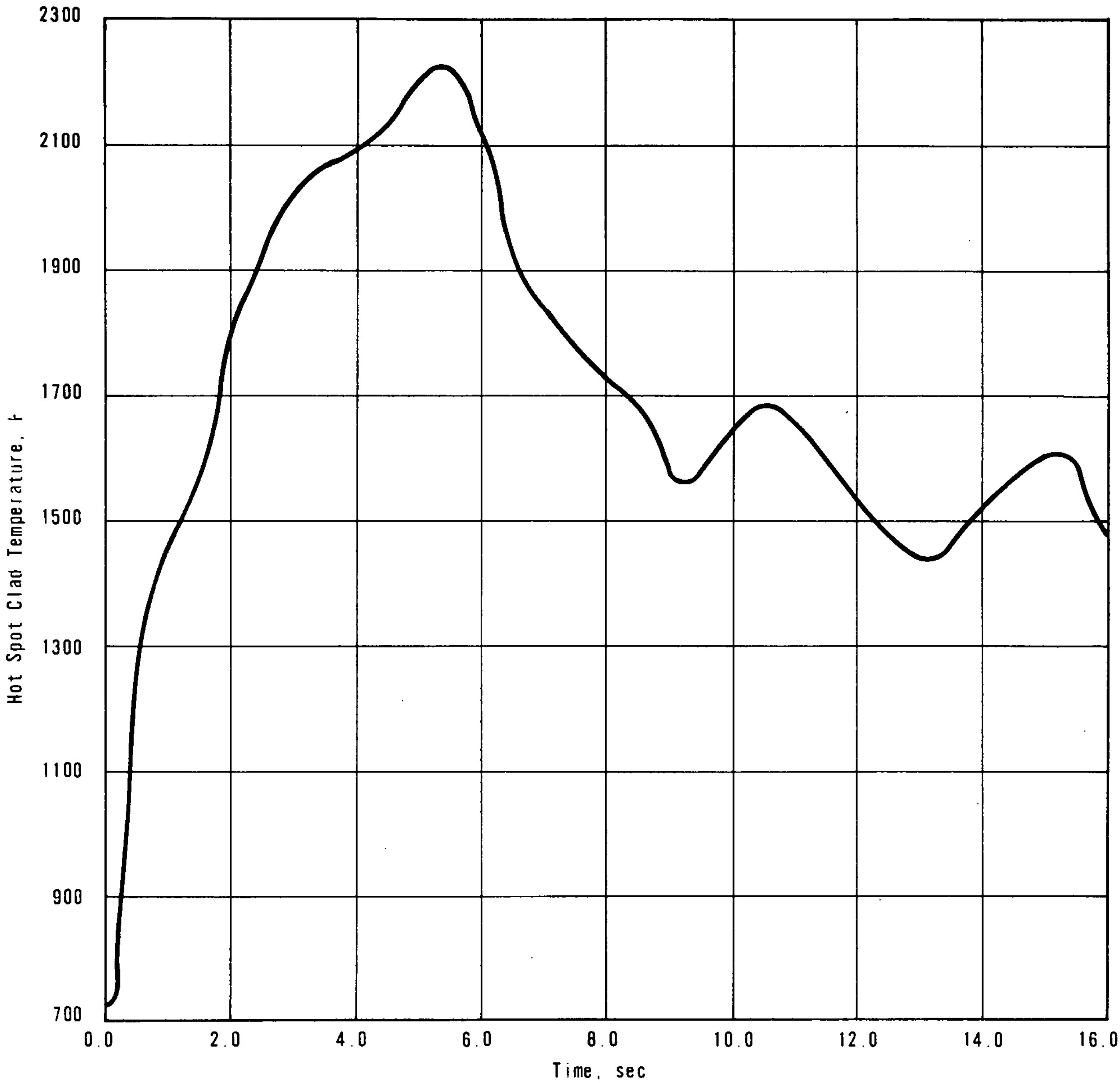
EFFECT OF FLOW BLOCKAGE ON INNER AND OUTER RODS AT 6 FT ELEVATION

OCONEE NUCLEAR STATION
Figure 14B-3



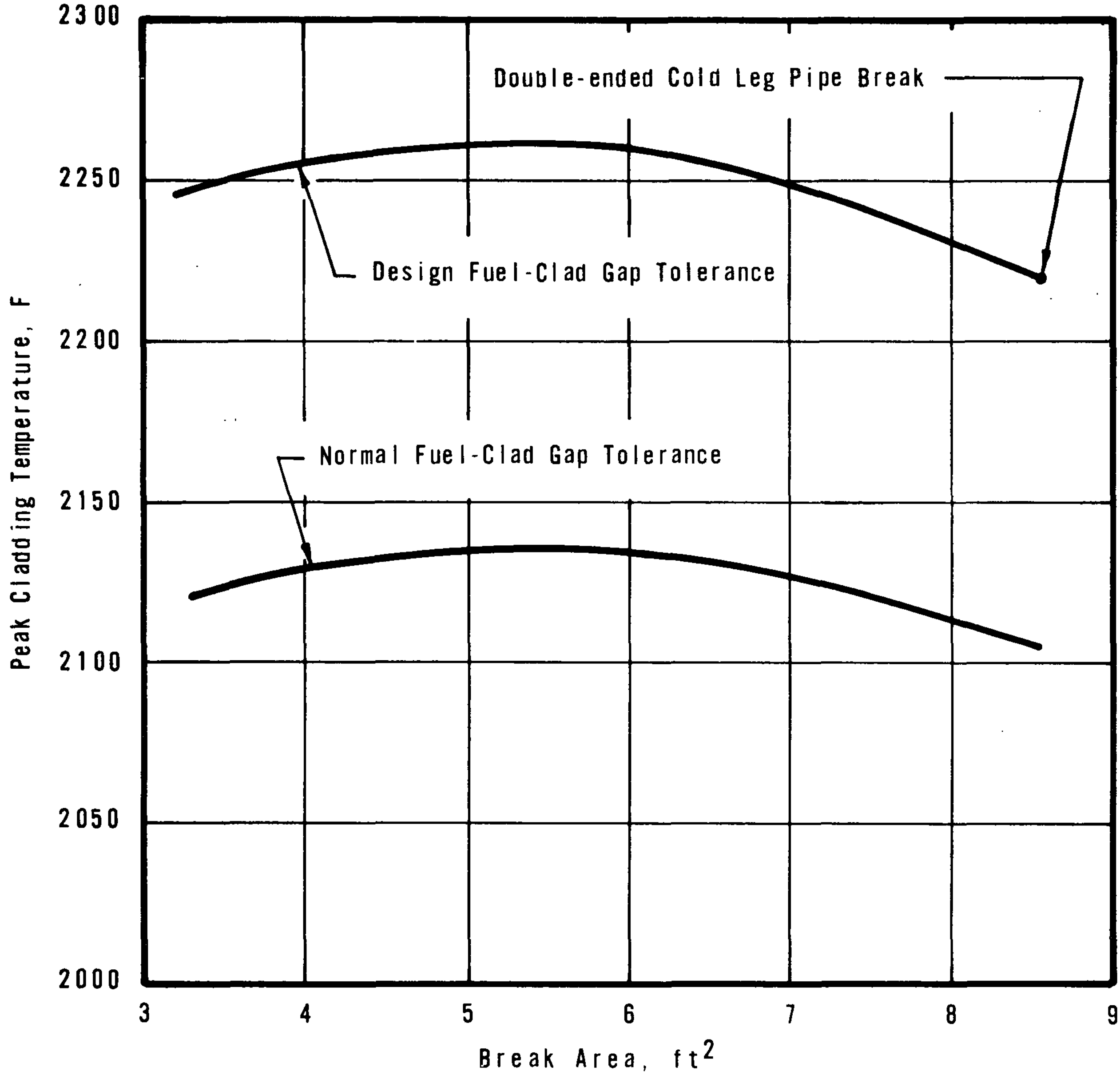
HOT SPOT CLADDING TEMPERATURE
14.1 FT^2 HOT LEG BREAK

OCONEE NUCLEAR STATION
Figure 14B-4



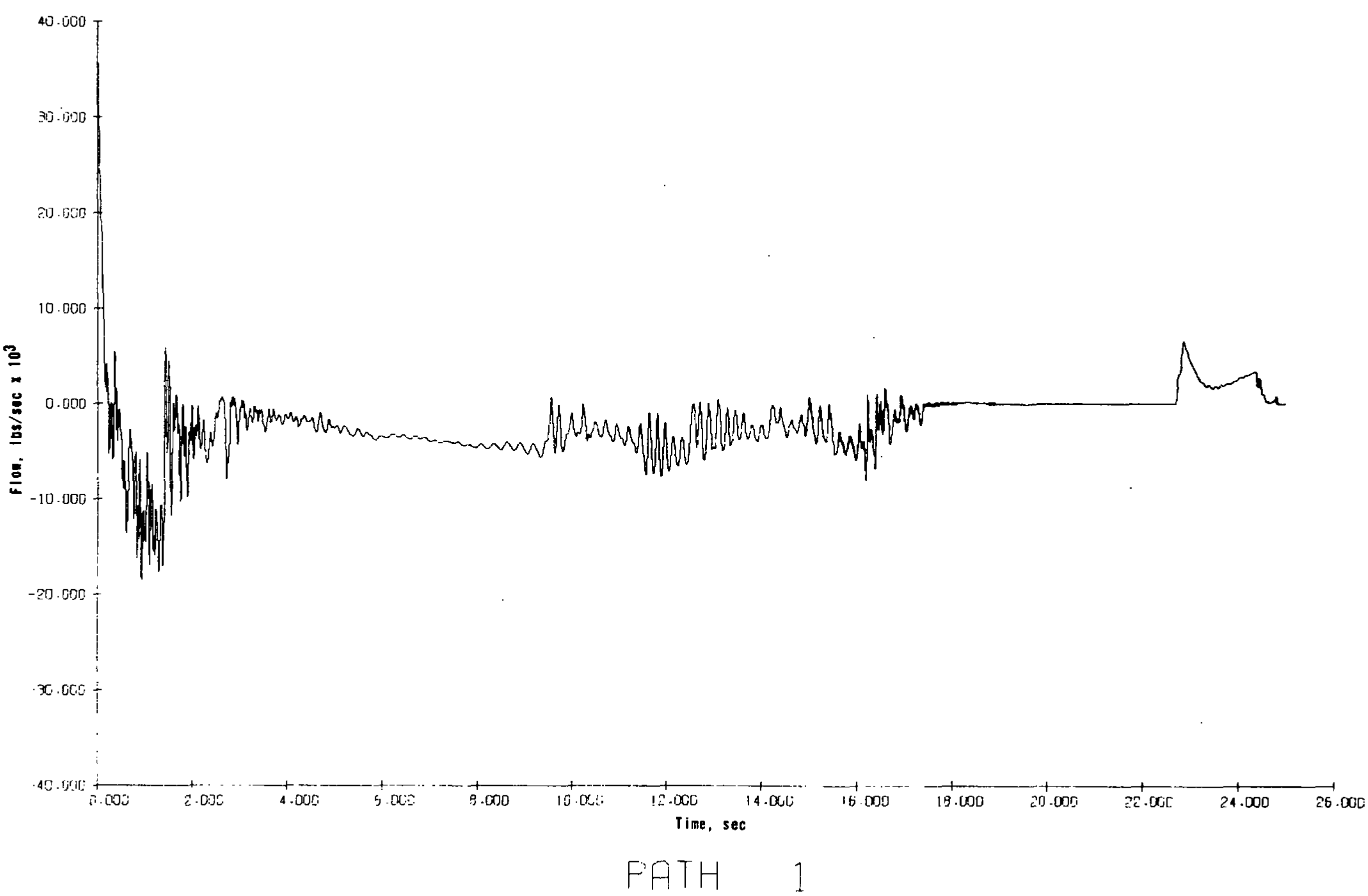
HOT SPOT CLADDING TEMPERATURE
8.5 FT^2 COLD LEG BREAK

OCONEE NUCLEAR STATION
Figure 14B-5



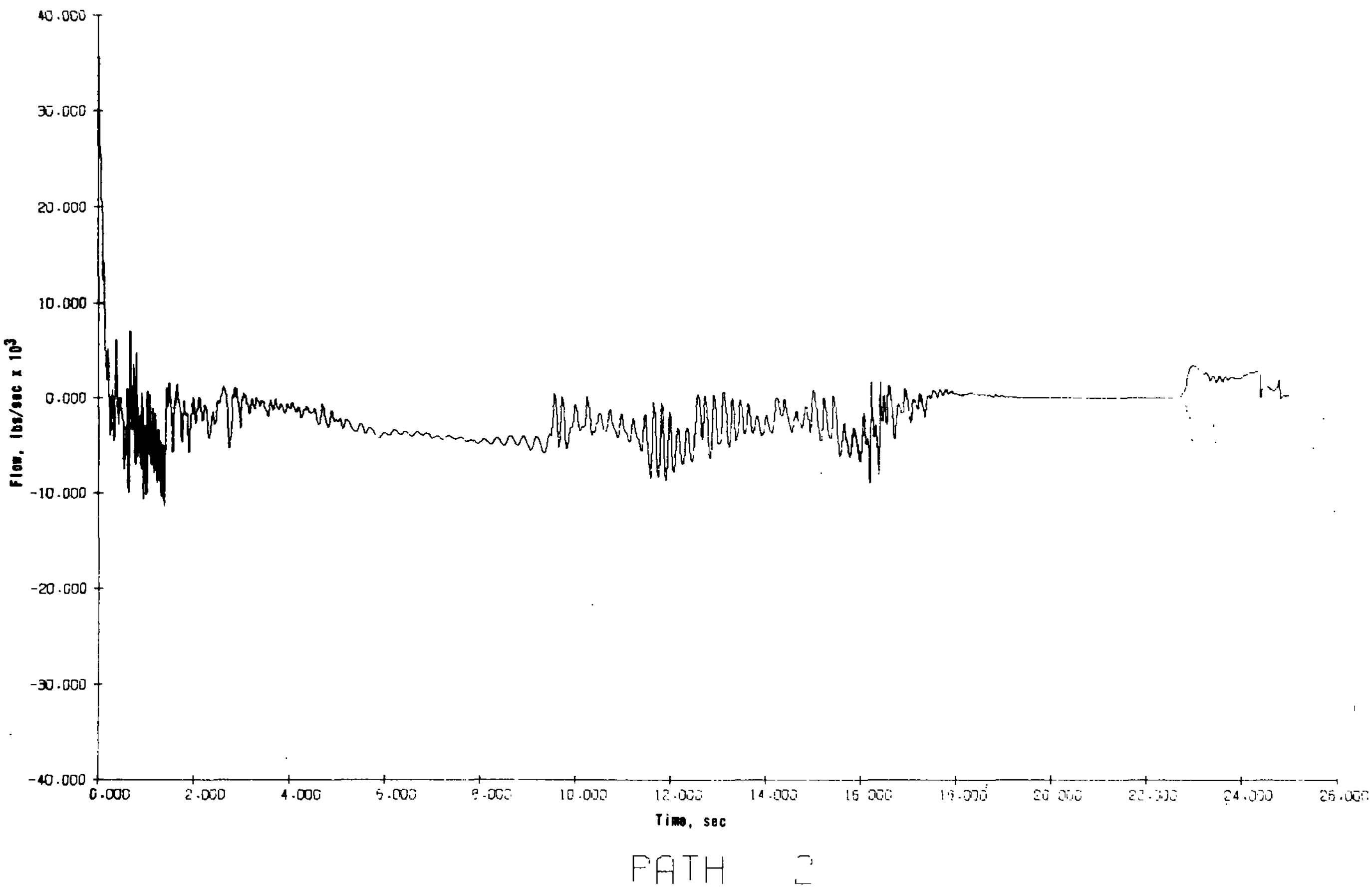
MAXIMUM CLADDING TEMPERATURE AS A FUNCTION OF BREAK SIZE IN THE COLD LEG PIPING

OCONEE NUCLEAR STATION
Figure 14B-6



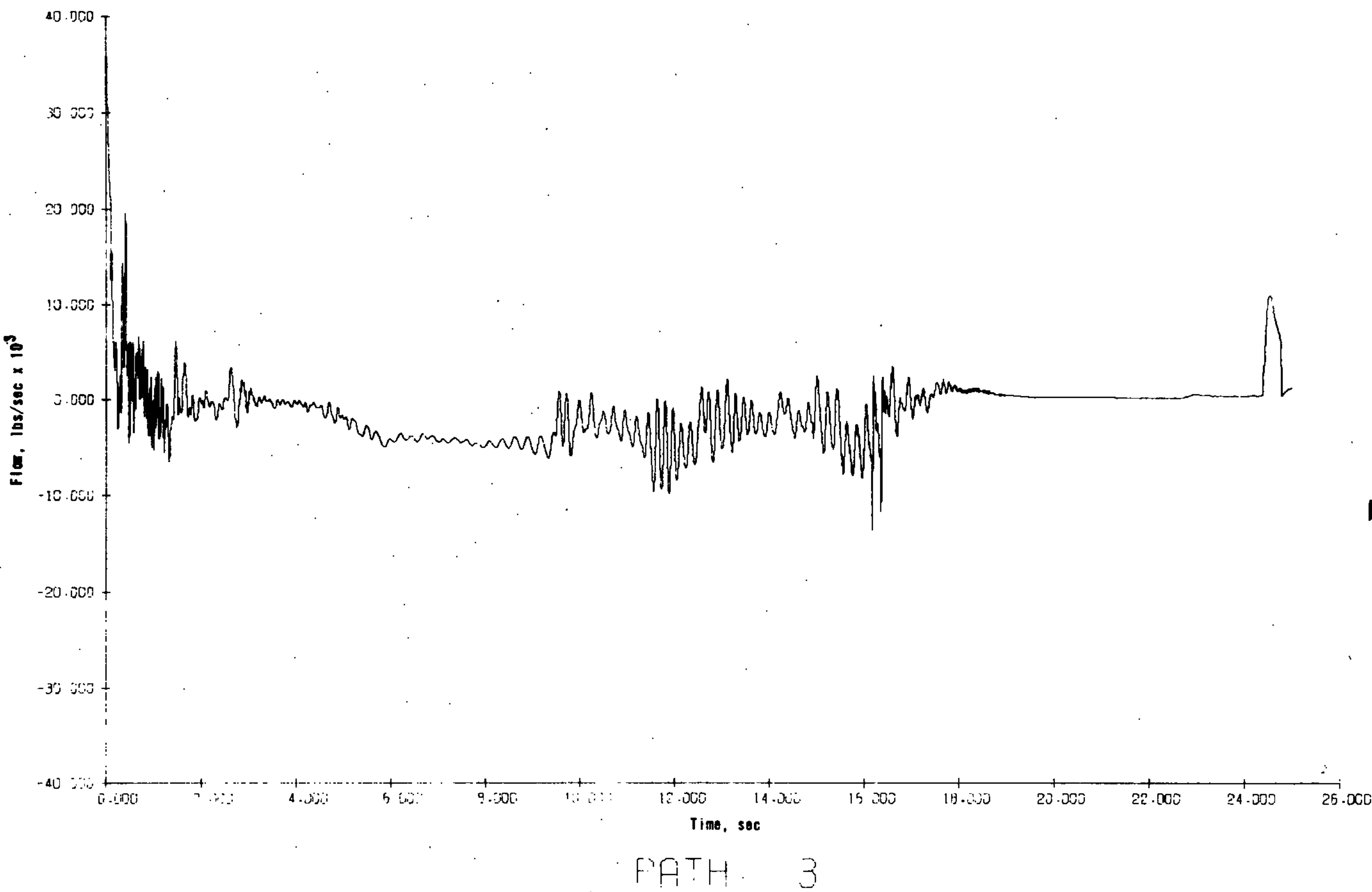
AVERAGE CORE FLOW (FLOW PATH 1)
8.5 FT^2 COLD LEG BREAK

OCONEE NUCLEAR STATION
Figure 14B-7



AVERAGE CORE FLOW (FLOW PATH 2)
8.5 FT^2 COLD LEG BREAK

OCONEE NUCLEAR STATION
Figure 14B-8



AVERAGE CORE FLOW (FLOW PATH 3)
8.5 FT^2 COLD LEG BREAK

OCONEE NUCLEAR STATION
Figure 14B-9

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