APPENDIX E

REQUIREMENTS RESULTING FROM TMI-2 ACCIDENT

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DRAWINGS CITED IN THIS APPENDIX*

*The listed drawings are included as "General References" only; i.e., refer to the drawings to obtain additional detail or to obtain background information. These drawings are not part of the UFSAR. They are controlled by the Controlled Documents Program.

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E.0 LIST OF TMI ACTION PLAN REQUIREMENTS

WHICH ARE NOT APPLICABLE TO BYRON/BRAIDWOOD STATIONS

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II.K.1.21	Auto SG Anticipatory Reactor Trip
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II.K.1.23	RV Level, Procedures

Orders on B&W Plants:

II.K.2.2	Procedures to Control AFW Independent of ICS
II.K.2.9	FMEA on ICS
II.K.2.10	Safety-Grade Anticipatory Trip
II.K.2.14	Lift Frequency of PORV and SVs
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I.C.8	Pilot NRC Review of Selected Emergency Procedures for NTOLs
III.A.l.l	Emergency Preparedness (Short-Term)

APPENDIX E

REQUIREMENTS RESULTING FROM TMI-2 ACCIDENT

E.1 SHIFT TECHNICAL ADVISOR (I.A.1.1)

POSITION:

A qualified individual, trained as a Shift Technical Advisor (STA), is provided on each shift at all times when a nuclear unit is in operational modes 1-4. This individual meets the requirements set forth in the Commission Policy Statement on Engineering Expertise on Shift which states: this person is either an on-shift senior reactor operator (SRO) qualified as an STA, or a qualified individual trained as an STA, in accordance with an approved STA training program. The person who is fulfilling the role of STA shall remain within 10 minutes of the control room at all times.

The selection criteria to be used in identifying future STA candidates are discussed in Table E.1-1. These criteria are developed in a way that the educational and management qualifications of each prospective candidate are evaluated in light of the guidance, provided by the NRC, for acceptance in the STA program. Minor deficiencies identified during this screening process will not disqualify a candidate if it is clear that training provided in the STA program will resolve those deficiencies. Other factors not specifically addressed in the NRC criteria such as naval reactors training or other nuclear industry experience, may also contribute to a prospective candidate's qualifications.

The minimum shift crew composition is provided in Table 13.1-1.

The STA Training Program is described in Subsection 13.2.1.

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TABLE E.1-1

STA SELECTION CRITERIA

The STA Program is based upon certain trainee prerequisites. These are necessary both as job prerequisites and as background to assure success in training. The limited rate at which STAs are qualified makes rigid, prescribed selection criteria unnecessary and undesirable, since there is adequate opportunity for case-by-case consideration. There is still the need, however, to lay out general guidelines by which the case-by-case selection process is conducted.

STA SELECTION GUIDELINES

1. The candidate must possess a technical degree; that is, a degree in an engineering or science field. Examples of acceptable fields are: Biology, Chemistry, Computer Science, Environmental Science, Mathematics, Physics, Chemical Engineering, Civil Engineering, Electrical Engineering, Mechanical Engineering, Nuclear Engineering.

2. For candidates with degrees other than Mechanical Engineering or Nuclear Engineering, a careful study of courses taken will be made to identify demonstrated competence in college level mathematics, physics, and chemistry. To identify such competence, consideration will be given to the grade received and the reputation of the college or university for strength in the scientific or engineering field.

3. For candidates with considerable company experience, evaluation of work performed or training completed at the company may be utilized in conjunction with or in lieu of evaluation of past academic performance to identify demonstrated competence in college level mathematics, physics, or chemistry.

E.2 SHIFT SUPERVISOR ADMINISTRATIVE DUTIES (I.A.1.2)

POSITION:

The administrative duties of the shift supervisor (i.e., Shift Manager position at Exelon Generation Company Midwest ROG plants) are controlled by station administrative procedures. The administrative procedures are such that the administrative duties assigned to Shift Manager personnel do not detract from their primary responsibility of assuring the safe operation.

Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant will be delegated to other operating personnel not on duty in the control room.

E.3 SHIFT MANNING (I.A.1.3)

POSITION:

Minimum shift crew composition is in accordance with Table 13.1-1 and meets the minimum licensed operator staffing requirements of 10 CFR 50.54(m)(2).

Staffing is sufficient to provide 24 hr/day, 7 day/week coverage, as well as time for vacation, sickness, and requalification training.

Overtime limitations are consistent with the requirements of 10 CFR 26, Subpart I, "Managing Fatigue," effective October 1, 2009. 10 CFR 26, Subpart I, contains requirements for managing fatigue and controlling work hours. Under 10 CFR 26, Subpart I, work hour restrictions include not only work hour limitations for rolling 24-hour, 48-hour, and 7-day periods, but also include a required minimum break between work periods and varying required minimum days off. Additionally, Subpart I confines the use of waivers (deviations from restrictions) to situations where overtime is necessary to mitigate or prevent a condition adverse to safety or necessary to maintain the security of the facility. Subpart I also addresses reporting requirements. The work hour controls scope includes operating and maintenance personnel, as well as those directing operating and maintenance personnel, performing work on risk-significant equipment, health physics and chemistry personnel who are a part of the on-site minimum shift complement, the fire brigade leader or advisor, and security personnel.

E.4 IMMEDIATE UPGRADING OF RO AND SRO TRAINING AND QUALIFICATIONS (I.A.2.1)

The original UFSAR commitments to NUREG 0737 section I.A.2.1 concerning Immediate Upgrading of RO and SRO Training and Qualification have been superseded by subsequent license amendments. The revised requirements are based on licensed operator training programs being accredited by INPO, a revision to 10 CFR Part 55, and adoption of a systems approach to training as required by 10 CFR 50.120. Current education and experience eligibility requirements for an operator license are addressed in the Quality Assurance Topical Report, NO-AA-10.

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E.5 ADMINISTRATION OF TRAINING PROGRAMS FOR LICENSED OPERATORS (I.A.2.3)

POSITION:

Byron and Braidwood Stations Training Department Instructors who teach systems, integrated response or transient courses to licensed operators, or licensed operator candidates, will be enrolled in senior reactor operator training or retraining programs, or will hold a senior reactor operator license certification. Members of other organizations, members of the training staff or guest lecturers, who conduct courses at the licensed operator level will be under the cognizance of an Instructor who possesses a senior reactor operators license or a senior reactor operator license certification.

E.6 REVISE SCOPE AND CRITERIA FOR LICENSING EXAMINATIONS - SIMULATOR EXAMS (PART 3) (I.A.3.1)

POSITION:

Byron and Braidwood Stations include as part of the licensing effort, a simulator examination. Braidwood operators normally train and test on the PWR simulator located at the Production Training Center at Braidwood, Illinois. Byron operators normally train and test on the simulator located at the Byron site.

E.7 EVALUATION OF ORGANIZATION AND MANAGEMENT IMPROVEMENTS OF NEAR-TERM OPERATING LICENSE APPLICANTS (I.B.1.2)

The contents and requirements for this section have been relocated to the Exelon Quality Assurance Topical Report (QATR).

E.8 GUIDANCE FOR THE EVALUATION AND DEVELOPMENT OF PROCEDURES FOR TRANSIENTS AND ACCIDENTS (I.C.1)

POSITION:

In accordance with a Byron Unit 1 operating license condition, a Procedures Generation Package was submitted to the NRC. Subsequently, Emergency Operating Procedures were developed in accordance with the Westinghouse Owners Group Emergency Operating Procedure Guidelines Revision 1. Emergency Operating Procedures exist for both Byron and Braidwood Stations.

E.9 SHIFT AND RELIEF TURNOVER PROCEDURES (I.C.2)

POSITION:

The requirements for shift and relief turnover are included in each Station's Administrative Procedures.

E.10 SHIFT SUPERVISOR RESPONSIBILITIES (I.C.3)

POSITION:

The "Statement of Position Responsibilities" issued on November 30, 1979, by the Nuclear Division - Manager of Operations indicates that the basic function of the shift supervisor (i.e., the Shift Manager position at Exelon Generation Company plants) is "the responsibility for the safe and efficient operation of the station." This position was further emphasized in Vice-President instruction No. 1-0-17 dated October 22, 1979, which clearly indicated that the prime concern of Commonwealth Edison management with regard to operation of its generating plants is to ensure the health and safety of the public and plant personnel.

The Shift Manager shall be responsible for directing the control room command function and the daily operations of the facility. These responsibilities are also included in the Station Administrative Procedures.

E.11 CONTROL ROOM ACCESS (I.C.4)

POSITION:

Administrative Procedures are in place to govern control room access during normal and emergency plant conditions and address the requirements of NUREG - 0578 dated July 25, 1979, Section 2.2.2.a.

E.12 PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE TO PLANT STAFF (I.C.5)

POSITION:

The Operating Experience (OPEX) Review is performed by designated personnel in accordance with administrative procedures. These personnel perform a review in conjunction with others, as appropriate, to fulfill the functions listed below. It is the responsibility of the OPEX staff (i.e., corporate and site) to disseminate information from a variety of sources for operating information that may be pertinent to plant safety with feedback to plant staff.

Specific responsibilities include:

1. The corporate and site OPEX function is to review information from a variety of sources for operating information that may be pertinent to plant safety. Typical sources of operating information include NRC documents, INPO documents, vendor documents, and Architect Engineer information. This information is transmitted in accordance with administrative procedures. Incorporation into training and retraining programs is in accordance with applicable station administrative procedures.

2. Information that, in the judgment of the OPEX staff may be pertinent to plant safety, is transmitted to appropriate personnel via the OPEX program. Subject Matter Experts (SMEs) are responsible for detailed review of OPEX documents associated with their discipline or area of expertise. The SME recommends actions necessary for incorporating OPEX information into procedures, operating instructions, etc.

3. The OPEX staff may designate the applicability of the operating experience to the appropriate organization. The OPEX staff may also initiate actions directing affected personnel to perform a review of specific OPEX documents to identify appropriate actions. Reviews are performed in accordance with administrative procedures. In the case of INPO significant operating experience reports (SOER) affecting several (or all) plants in the fleet, the Corporate Functional Area Manager is responsible for reviewing responses for adequacy and the uniformity of action taken throughout the fleet.

4. Program effectiveness reviews are performed periodically to assure recommendations and corrective actions have been incorporated as appropriate. The fleet OPEX coordinator is responsible for coordinating effectiveness reviews.

5. The OPEX function of the station Regulatory Assurance Department is to review information from a variety of sources. Information viewed pertinent to plant safety will be forwarded for review by station personnel.

6. Audits of the OPEX program will be conducted by Nuclear Oversight.

E.13 VERIFY CORRECT PERFORMANCE OF OPERATING ACTIVITIES (I.C.6.)

POSITION:

Administrative procedures to govern verification of correct performance of operating activities incorporate the guidelines as stated in NUREG-0737 dated October 31, 1980. Consistent on the item position, persons, not licensed, who complete a training qualification program will be considered "qualified" to implement action to take equipment out-of-service and return it to service. Upon return to service, following maintenance on, or modification to a system/component, verifying the operable condition of that system/component may be accomplished by either special testing or by adherence to surveillance testing or general inspection. An independent verification for all components that provide a safety function or important function as identified by the SRO shall be required after taking safety-related equipment out of service, following return to service of safety-related equipment, following maintenance modifications on a safety-related equipment, during performance of safety-related surveillances, following placement/removal of temporary alterations, and after alignment changes have been made. The methods of independent verification are verification of alignment by a second qualified operator or by functional checks. Independent verification by a second qualified individual may include observation of control room indicators, observation of position indicators or status lights, or field verification of equipment alignment.

In accordance with NUREG-0737 (I.C.6.), independent verification requirements may be waived for cases where significant radiation exposure is expected.

E.14 NSSS VENDOR REVIEW OF PROCEDURES (I.C.7)

POSITION:

NSSS preoperation and startup test procedures written for Byron/ Braidwood Stations were reviewed by Westinghouse and Sargent & Lundy for verification of scope and adequacy as directed by CECo's Project Engineering Department. Westinghouse was involved in Emergency Procedures as outlined in I.C.1, in addition to a review of operating procedures written by the station. Procedures generated by the station as a result of Task I.C.1 were reviewed by Westinghouse and Sargent & Lundy as requested.

E.15 PILOT MONITORING OF SELECTED EMERGENCY PROCEDURES FOR NTOL (I.C.8)

POSITION:

Byron/Braidwood Station will not write interim emergency procedures that would necessitate NRC review. Due to Byron/Braidwood schedules and the WOG schedule on Task I.C.1, writing interim emergency procedures that would require review by NRC and revision after Task I.C.1 is approved is not necessary. Review of emergency procedures by NRC will be covered under item I.C.9 (from NUREG-0660).

E.16 CONTROL ROOM DESIGN REVIEWS (I.D.1)

POSITION:

The Applicant submitted the preliminary control room design review report for the Byron Station on November 12, 1981 with letter dated November 12, 1981 to H. R. Denton from T. R. Tramm. Supplements I and II to the report were submitted for staff review on May 9, 1983 (letter from E. D. Swartz to H. R. Denton).

A detailed design review has been completed for Byron/Braidwood Stations. The final Summary report on this review was submitted in a letter dated November 26, 1986, from K. A. Ainger to H. R. Denton.

E.17 PLANT SAFETY PARAMETERS DISPLAY CONSOLE (I.D.2)

POSITION:

The Byron/Braidwood design includes a plant safety parameter display system. This system is described below.

Design Basis

The safety parameter display system (SPDS) is provided in response to NUREG-0696. That document addresses the requirements related to emergency response facilities and states that a display system is to be installed in existing control rooms "...to help operating personnel in the control room make quick assessments of plant safety status." The Byron/Braidwood SPDS design is intended to meet this requirement and fulfills the further intent stated in NUREG-0696 that the display system should serve "...to concentrate a minimum set of plant parameters from which plant safety status can be assessed. The selection of parameters is based on the function of enhancing the operator's capability to assess plant status in a timely manner without surveying the entire control room."

As indicated in NUREG-0696, the safety parameter display system is intended to provide a display of plant parameters from which the safety status of operations may be assessed in the control room, TSC and EOF. The primary function of the SPDS is to help control room operating personnel make quick assessments of plant status. Duplication of the SPDS displays in the TSC is intended to improve the exchange of information between these facilities and the control room, and assist in the decision making process.

NUREG-0696 indicates that SPDS should be responsive to transient and accident sequences and should be sufficient to indicate plant status. A display format consisting of a minimum set of parameters or derived variables from which the overall plant status may be determined, should be provided.

These parameters, or derived variables, are individual plant parameters or are composed of a number of parameters or derived variables giving an overall system status.

The intent of the SPDS is to meet two specific operational goals:

a. Prevention - To provide a display system which will permit the operator to monitor the state of the plant process and detect any abnormalities for which corrective action might be taken to terminate the event prior to the initiation of automatic reactor trip and/or safeguards actuation.

b. Mitigation - In the case of events which the operator does not detect or cannot terminate, the goal is to enable the operator to assess the safety status of the plant and verify proper safeguard functions to mitigate the consequences of the event.

In addition, the SPDS display is located so it is convenient to the control room operators. This system will be continuously available to display information so that the plant safety status can be readily and reliably assessed by control room personnel who are responsible for the prevention and mitigation of off-normal plant conditions.

The SPDS is driven by the plant process computer. It does meet requirements of the single-failure criteria and will not be qualified to meet Class 1E requirements. The SPDS is suitably isolated from electrical or electronic interference with equipment and sensors used for safety systems. The SPDS is not seismically qualified, and additional seismically qualified indication is not required for the sole purpose of being a backup for SPDS. The overall availability of SPDS is approximately equal to plant process computer availability. The Byron and Braidwood simulators are provided with SPDS software that is compatible with the computer for the simulators.

The isolation device between the safety-related heated junction thermocouple/core exit thermocouple monitoring system and the non-safety-related SPDS is fiber optic (glass) cable.

The fiber optic cable is used to transmit data from the safety-related system to the non-safety-related system. Data transmission is accomplished by non-electrical light pulses. Since this data link is non-electrical fiber optic cable, no electrical connections exist between the HJTC/CET monitoring system and the SPDS. Because there are no electrical connections between the safety-related and non-safety-related systems, specific testing of the effect of postulated electrical faults is not required.

The isolation devices have been seismically and environmentally qualified as Class 1E components of the HJTC/CET monitoring system. The qualification methodology and results are documented in Combustion Engineering Report 2382-ICE-3310, Rev. 0, "Qualification Summary Report for the Heated Junction Thermocouple/Core Exit Thermocouple Monitor System for Commonwealth Edison Company Byron Station Units 1 and 2" dated January 31, 1984. Since the equipment is located in a mild environment, the requirements of 10 CFR 50.49 do not apply.

The inherent characteristics of data transmission by fiber optic cables (non-electrical) are such as to preclude any electrical interference. Thus specific measures to protect the HJTC/CET monitoring system from electrical interference are not required.

Parameters Monitored

NUREG-0696 is the basis for the selection of parameters for the SPDS displays. That document requires important plant functions be monitored in each plant operating mode to inform control room operating personnel of overall plant status.

Plant functions considered in the selection of the Byron/ Braidwood SPDS variables included the following:

Reactivity Control
Reactor Core Cooling
Reactor Coolant System Integrity
Reactor Coolant System Inventory Control
Containment Activity Level
Containment Integrity
Secondary System Status.

A comparison of the Byron/Braidwood SPDS functions with those identified in NUREG-0696, NUREG-0737, Supplement 1, and the Westinghouse ERG Criteria Safety Functions is provided in Table E.17-3.

The operational modes for Byron/Braidwood are listed below. Parameters selected for the SPDS displays provide the operator the ability to monitor plant status over the entire range of operating modes.

	Reactivity Condition, K_{eff}	% Rated Thermal Power	Average Coolant Temperature
1. Power Operation	≥0.99	>5%	N/A
2. Startup	≥0.99	≤5%	N/A
3. Hot Standby	<0.99	N/A	≥350° F
4. Hot Shutdown	<0.99	N/A	<350° F >200° F
5. Cold Shutdown	<0.99	N/A	≤200° F
6. Refueling	N/A	N/A	N/A

The SPDS display is available on color graphics displays in the control room and technical support center for key plant parameters. Within the frame of each display, a visual indicator is provided to indicate the overall status of the SPDS. The system takes its input from several sources for each parameter and determines which sensors are valid. If any SPDS indication is in alarm the visual indicator turns red and the operator can quickly access the full SPDS display to assess the safety status of operations.

The number of parameters on each display has been kept to a minimum so information can be readily interpreted by the operator. Also, each parameter consists of only one variable, with the exception of power mismatch and net charging/letdown flow, to assist the operator in recognizing the problem area independent of supplementary displays. The parameters have all been normalized to aid the operator in recognizing an off-normal condition.

The variables presented on the SPDS displays are considered adequate to cover incidents of moderate frequency, infrequent incidents or limiting faults. Use of the power mismatch parameter will allow the operator to assess, more quickly, an increase or decrease in heat removal by the secondary system.

Increases and decreases in reactor coolant inventory are signaled by monitoring the net charging/letdown flow rate and pressurizer level. Off-normal reactivity changes are quickly observed by monitoring the power mismatch and T_{avg} parameters. In addition to the SPDS displays, the operator will have at his disposal other parameters for display to assist in assessing off-normal conditions.

The SPDS is an output from the plant process computer. Any information contained within the process computer data base is available for display at the operator's console on some type of video display device or printer.

The video displays include the wide and narrow range iconic displays for SPDS, the incore thermocouple display, pressure/ temperature curves for heatup and cooldown. Although these displays are non-SPDS, their availability is equivalent to the SPDS displays.

The operator's control room interface with the process computer will allow trend display of all process computer point ID's for at least up to 48 hours at update intervals under 1 minute.

The listing in Table E.17-1 identifies each parameter displayed on the SPDS displays and how it relates to monitoring plant functions.

Some parameters are not included on the SPDS displays. Those parameters, the rationale for their exclusion from SPDS, and the locations of that data are discussed below:

1. Neutron Flux - On the narrow range display, neutron flux is displayed in percent on the power mismatch spoke. Nuclear power is compared to turbine power to provide the operator indication of reactivity balance in Mode 1. Power mismatch in conjunction with T_{avg} provide the operator with indication of primary and secondary power balance. Power mismatch is the initial input to the rod control system to initiate rod motion to affect T_{avg}.

   On the wide range display, startup rate is displayed. During startup, the change in flux level is more indicative of reactivity balance than the absolute value of the flux. The startup rate is displayed in the same format (decades per minute) whether indication is generated by source range or intermediate range instruction.

   In addition, flux is recorded on the main control board as well as indicated on the NIS panel and main control board. The upper and lower detector currents are individually recorded on the main control board. Alarms associated with flux deviation and axial offset alert the operator of impending trends. The manner in which neutron flux is displayed on the iconic provides the operator with the data necessary to ascertain the core reactivity status.

2. Hot Leg and Cold Leg Temperatures - On the main control board, hot leg and cold leg temperatures are indicated on recorders for trending of long term events. The immediate indications provided to operators are T_{avg} and incore thermocouple temperatures. T_{avg} is available when reactor coolant pumps are running, and core exit thermocouple temperatures are available for all modes of operation. These temperatures combined with subcooling indications provide the operator with sufficient indication to rapidly determine heat removal capabilities.

3. Steam Pressure - In Mode 1, heat removal by the secondary system is verified by monitoring steam generator

narrow range level. Steam generator water level is provided as a positive indicator to the operator for heat removal capability. Steam pressure was not used because of the potential for rendering misleading information on heat removal capabilities. Narrow range steam generator level in conjunction with main steam line and air ejector radiation levels provide indication of steamline break events. Likewise, wide range steam generator level in conjunction with radiation levels and containment pressure provide indication of steamline breaks being inside or outside containment.

4. RHR Flow - Heat removal capability is verified by using the main control board indication as a backup to the SPDS. Core exit temperatures and RCS pressure provide primary indication to the operator. In the RHR mode, these indications can direct the operator to the main control board to verify proper RHR operation. Our intent was to keep the SPDS parameters to a minimum and use the SPDS indication to direct operators to areas of the main control board where control system manipulations can be made to affect parameters monitored, and thereby maintain the critical safety functions.

5. Steam line radiation is monitored on both the wide and narrow range displays.

6. Containment isolation is not monitored by the SPDS. This function is verified by visual inspection of the main control board containment isolation status panel by the operator prior to blocking automatic safety injection actuation. Displaying this function on the SPDS display would not provide any additional significant information.

7. Containment Hydrogen Concentration - This parameter would not provide useful information during normal (Mode 1) operation. Hydrogen generation does not become a concern until greater than twenty-four hours after an accident. Hydrogen concentration is displayed on the main control board and at the local panel. The fact that the control room operator cannot affect the concentration from the control room was considered in determining the usefulness of hydrogen concentration on SPDS displays.

DATA VALIDATION

The data which is displayed on the SPDS is validated. The quality or validity of the data is also displayed to the operator.

There are four steps in data validation: quality setting, quality checking, quality carrying, and deviation checking among redundant points. Quality and the alarm conditions determine the status of a point. The primary method of displaying status is with a color change. The specifics change with the format of the information being displayed.

The first step in establishing the status is accomplished in the scanning subsystem of the Plant Process Computer. Each input is assigned a point quality, representing one of these states (GOOD, SUSPECT, POOR, MANUALLY ENTERED, DAS (Data Acquisition System) BAD, FAILED RANGE CHECK, or BAD). For example, if a point is out of sensor limits, it will be marked FAILED RANGE CHECK. If the value is manually entered, it will be marked MANUALLY ENTERED. If a point can not be scanned due to an Input/Output card failure, it will be marked DAS BAD.

The second step is a quality check of the inputs to the SPDS algorithms. In those cases where redundant inputs are available, poor and bad values are removed from the calculations. If all points of one parameter are removed, the parameter is set to an alarm limit and marked bad.

The third step logically combines the quality of the inputs used in a calculation. A suspect quality combined with a good quality yields a suspect quality. In arithmetic operations, this carrying of quality occurs automatically. The resulting quality is actually contained in the value. If the calculation is not arithmetic, the algorithm duplicates this logic.

The fourth step is a maximum deviation check on redundant points. Each point outside of a specified range (generally 5%) is rejected. Points are rejected one at a time and a new average calculated. If only one point remains, the average is set to one of the alarm limits and marked bad.

Four formats of information are on the SPDS displays: analog values, digital tic marks, dynamic text, and the distorted octagon (iconic). Bad or poor quality values used in drawing the iconic distort the octagon to an alarm limit. The octagon will always be yellow. Tic marks can be displayed in two different colors. The default is cyan, and alarm is red. Numeric display default is white and alarm is yellow.

The SPDS validation and verification program has been conducted in accordance with Commonwealth Edison's response to NUREG-0737, Supplement 1.

The parameters displayed on the SPDS iconic displays are the same parameters evaluated in the UFSAR; no additional unreviewed safety issues should result. During the performance of the Detailed Control Room Design Review (E.16), in accordance with the schedule for implementation of NUREG-0737, Supplement 1 the SPDS was again reviewed with respect to the introduction of unreviewed safety issues.

The Technical Specifications address limiting conditions for operation of equipment whose satisfactory operation, when called upon, is required for safe operation of the plant. The SPDS does not function in this capacity and, therefore, is not addressed in the Technical Specifications.

Location of Displays

Three SPDS video display devices are installed for each station as follows:

Unit 1 Main Control Board
Unit 2 Main Control Board
Technical Support Center

Iconic Display

Two iconic displays are included in the SPDS. The iconic format consists of eight parameters. Each parameter is represented by a spoke of a geometric figure.

For those iconic vectors which display more than one parameter, the parameter which is furthest from the reference value will be the parameter whose active value is displayed.

Figure E.17-1 is an example of a basic iconic display with the elements labeled. Each spoke in the figure represents a normalized scale for a specific parameter. Reference points for all parameters are drawn as fixed points on the spokes such that when dotted lines connecting them are drawn, the result is a regular geometric pattern. Deviations in active parameters from the reference parameters are represented as points plotted on the normalized scales drawn between fixed reference and high and low limit points. A positive deviation results in a point drawn between the reference point and the high limit and conversely a negative deviation results in a point drawn between the reference and the low limit point. Solid lines are drawn connecting the active points to form a second geometric figure.

The narrow range (NAR RNG) and wide range (WID RNG) iconic displays for Byron/Braidwood are presented in Figures E.17-2 and E.17-3. The display labeled NAR RNG is specifically devoted to normal plant power operation. The display labeled WID RNG is devoted to the full range of plant operations from shutdown to full power operation. The operator may select either display for viewing at his discretion. Table E.17-2 contains a description of each of the spokes which comprise the iconic display.

TABLE E.17-1

PARAMETERS ASSOCIATED WITH PLANT FUNCTION MONITORING

Reactivity Control

Power Mismatch
T_{avg}
Startup Rate
Core Exit Temperature

Reactor Core Cooling

Core Exit Temperature
NR SG Level
WR SG Level

Reactor Coolant System Integrity

NR SG Level
WR SG Level
WR RCS Pressure
Pressurizer Level
Pressurizer Pressure
Net Charging/Letdown Flow Rate

Reactor Coolant System Inventory Control

Net Charging/Letdown Flow Rate
Pressurizer Level
Containment Floor Drain Sump Level

Containment Activity Level

Containment Activity
Containment Floor Drain Sump Level

Containment Integrity

Containment Temperature
Containment Pressure

Secondary System Status

NR SG Level
WR SG Level
Power Mismatch
T_{avg}

TABLE E.17-2

ICONIC DISPLAY DESCRIPTION

Analog Spokes

1. T_{avg}

The value and loop number for the T_{avg} which corresponds to the maximum deviation (positive or negative) from the reference.

a. Active value - maximum deviation of the loop T_{avg} with respect to the reference value.

b. Reference value - plant input (T_{ref}) obtained from turbine power.

c. High limit - system constant equal to the Technical Specification DNB limit for T_{avg}.

d. Low limit - system constant equal to the Technical Specification minimum temperature for criticality.

2. Power Mismatch

Nuclear to Turbine Power Mismatch:

a. Active value - nuclear power minus turbine power.

b. Reference value - system constant equal to zero.

c. High limit - system constant equal to 9% (equivalent to high flux reactor trip with turbine power equal to 100%).

d. Low limit - system constant equal to -10%.

3. Narrow Range Steam Generator Level

The value and loop number for the steam generator whose narrow range level indication corresponds to the maximum deviation (positive or negative) from the reference value.

a. Active value - steam generator narrow range level that corresponds to the maximum deviation from the reference value.

TABLE E.17-2 (Cont'd)

b. Reference value - steam generator narrow range reference level.

c. High limit - system constant equal to high steam generator level reactor trip setpoint.

d. Low limit - system constant equal to low steam generator level reactor trip setpoint.

4. Net Charging Flow

The value of net charging flow is calculated by the difference in charging pump flow - letdown flow and total RCP seal return flow.

a. Active value - charging pump flow minus letdown flow and RCP seal return flow.

b. Reference value - system constant equal to zero.

c. High limit - system constant equal to the maximum makeup flow.

d. Low limit - system constant equal to the maximum letdown flow.

5. Pressurizer Level

a. Active value - pressurizer level average.

b. Reference value - pressurizer level reference as input from the plant (T_{avg}).

c. High limit - 100% for wide range and RX TRIP for narrow range.

d. Low limit - 0% for wide range and heater interlock letdown isolation level for narrow range.

6. Pressurizer Pressure

a. Active value - pressurizer pressure average.

b. Reference value - system constant equal to normal operating pressure (2235 psig).

c. High limit - PORV setpoint.

TABLE E.17-2 (Cont'd)

d. Low limit - pressurizer minimum pressure post-reactor trip (2000 psig).

7. <u>Core Exit Temperature</u>

The value from the RCS pressure-temperature curve for maximum incore thermocouple.

a. Active value - the average of the ten highest incore thermocouple temperatures.

b. Reference value - reactor core exit reference temperature.

c. High limit - RCS saturation temperature.

d. Low limit - RCS low temperature limit.

8. <u>Startup Rate</u>

a. Active value - nuclear flux rate of change.

b. Reference value - system constant equal to zero.

c. High limit - system constant equal to 2 DPM.

d. Low limit - system constant equal to -0.5 DPM.

9. <u>Containment Pressure</u>

a. Active value - containment average pressure.

b. Reference value - system constant equal to the normal containment pressure.

c. High limit - system constant equal to 3.4 psig (HI-1 setpoint).

d. Low limit - system constant equal to 0.5 psig.

10. <u>Steam Generator Wide Range Level</u>

The values and loop number for the wide range steam generator level which corresponds to the maximum deviation.

a. Active value - maximum deviation of the loop steam generator wide range levels with respect to the reference value.

TABLE E.17-2 (Cont'd)

b. Reference value - system constant equivalent to the design reference of the narrow range steam generator level span.

c. High limit - system constant equal to the top of narrow range span.

d. Low limit - system constant equal to the bottom of narrow range span.

11. Reactor Coolant System Pressure

a. Active value - reactor coolant system pressure average.

b. Reference value - reactor coolant system pressure reference.

c. High limit - reactor coolant system pressure high limit.

d. Low limit - reactor coolant system saturation pressure.

12. Containment Conditions

The vector length is calculated using the parameter which corresponds to the maximum deviation from the reference value.

a. Active value - containment average air temperature or containment floor drain sump average level.

b. Reference value - system constant equal to the normal containment air temperature or normal containment floor drain sump level.

c. High limit - for containment air temperature a system constant equal to 120°F. For containment floor drain sump level a system constant equal to HI-HI level.

d. Low limit - for both containment air temperature and containment floor drain sump level a system constant equal to the low instrumentation limit.

Digital Spokes

1. Reactor Vessel Level

The vector length is calculated by checking for the

TABLE E.17-2 (Cont'd)

existence of a low level condition from any one of the eight discrete vessel level sensors.

a. Active value - input from vessel level sensors and reactor coolant pump status.

b. Reference value - the parameter is maintained at its maximum level except in mode 6.

c. High limit - equal to reference value.

d. Low limit - represents any void in the vessel head.

2. Radiation

The vector length is calculated by determining existence of a high radiation condition from any one of the radiation monitors.

a. Active value - input from radiation monitors.

b. Reference value - system constant equal to normal activity levels.

c. High limit - system constant equal to alarm setpoint.

TABLE E.17-3

COMPARISON OF SPDS SAFETY FUNCTIONS AND PARAMETERS WITH NUREG-0696, NUREG-0737, SUPPLEMENT 1 AND WESTINGHOUSE CRITICAL SAFETY FUNCTIONS

NUREG-0696 NUREG-0737, SUPPLEMENT 1	WESTINGHOUSE ERG* CRITICAL SAFETY FUNCTIONS	B/B CRITICAL SAFETY FUNCTION	B/B SPDS PARAMETERS MONITORED
1. Reactivity Control	Sub-criticality	Reactivity Control	** A. SUR B. Power Mismatch C. Core Exit Temp D. T_{avg}
2. Reactor Core Cooling and Heat Removal	Core Cooling	Reactor Core Cooling	** A. Core Exit Temp ** B. Subcooling C. NR S/G Level D. WR S/G Level
3. --	Heat Sink	Secondary System Status	** A. NR S/G Level B. WR S/G Level C. Power Mismatch D. T_{avg}
4. RCS Integrity	Integrity	RCS Integrity	***A. T_{avg} ** B. WR RCS Press C. Przr Level D. Net Chg/Letdown Flow E. Containment Floor Drain Sump Level
5. Containment Conditions	--	Containment Integrity	** A. Containment Pressure B. Containment Temperature
6. Radio-activity Control	Containment Conditions	Containment Activity Level	** A. Containment Rad Levels ** B. Containment Floor Drain Sump Level

TABLE E.17-3 (Cont'd)

NUREG-0696 NUREG-0737, SUPPLEMENT 1	WESTINGHOUSE ERG* CRITICAL SAFETY FUNCTIONS	B/B CRITICAL SAFETY FUNCTION	B/B SPDS PARAMETERS MONITORED
7. --	Inventory	RCS Inventory Control	** A. Pressurizer Level B. RVLIS C. Net Charging/ Letdown Flow D. Containment Floor Drain Sump Level

*ERG - Emergency Recovery Guidelines
**Indicates primary indication utilized in Westinghouse ERGs.
***T_{avg} indication utilized as opposed to T_c.

E.18 TRAINING DURING PREOPERATION AND LOW POWER TESTING PROGRAM (I.G.1)

POSITION:

As noted in the SER, Section 14, the NRC concluded that the initial plant test program was acceptable. Training on natural circulation, plant response and characteristics is conducted during initial operator license training utilizing the Byron and Braidwood simulators.

Natural circulation testing was performed at Diablo Canyon; the NRC has reviewed the results and found them to be acceptable. The NRC has reviewed the similarities between Byron/Braidwood and Diablo Canyon and has accepted the analysis for natural circulation testing as representative of the prototype test run at Diablo Canyon.

E.19 REACTOR COOLANT SYSTEM VENTS (II.B.1)

POSITION:

The reactor coolant system vent (RCSV) line is located at the top of the reactor integrated head. This 0.5-inch (Byron) or 1.0-inch (Braidwood) diameter schedule 160 line contains four safety grade solenoid-operated valves which are powered by emergency buses. Braidwood Unit 2 has ½-inch piping at the inlet and outlet of Valve Assembly 2RC014B/D. Being located at a high point permits this line to vent the reactor coolant system normally connected to the reactor pressure vessel. The RCSV is remotely operated and monitored from the main control room. Since the RCSV line is either a 0.5-inch pipe (Byron) or 1.0-inch pipe (Braidwood), it is smaller than the size for which a LOCA analysis would be required.

The RCSV line was designed and installed as ASME Section III, Class 1 piping to applicable codes. Final positioning of the discharge of the RCSV minimizes possible impingement on equipment or obstructions. (See Figure E.19-1, RCSV ISOMETRIC DRAWING.)

The RCSV system meets the requirements of 10 CFR 50.44(c)(3)(iii). Seismic and environmentally (IEEE 323-1974) qualified ASME Section III Class 1 solenoid-operated valves (1(2) RC014A-D) are installed in parallel sets of two, supplied by redundant emergency buses. Positive indication of valve position is provided, from valve operator limit switches, to the control switch lights in the main control room. In addition, surface mounted resistance temperature detectors with main control room alarms are provided downstream of the solenoid-operated valves for leak detection.

These valves are designed to pass steam, steam/water, water, and noncondensible gases. The RCS vents directly to the containment.

Station procedures for use of the RCSV system have been written utilizing guidelines developed by the Westinghouse Owners' Group. Human factor considerations have been taken into account in writing these procedures.

Technical Requirements Manual 3.4.e ensures operability of the RCSV system.

E.20 PLANT SHIELDING (II.B.2)

POSITION:

A radiation and shielding design review was conducted for Byron/Braidwood Stations (B/B) using the guidance provided in NUREG-0737. Reference 9 describes in detail the source terms used for radiation qualification. Inside the containment, the source term is based on the release of 100% of the noble gases and 50% of the halogens to the containment atmosphere. Systems containing liquid sources are analyzed using a source term consisting of 50% of the core inventory of halogens and 1% of the core inventory of fission solids.

Note that the liquid source term does not contain noble gases as specified in this clarification. This is not done because it is inconsistent with a severe accident, and since the fission solids are the dominant nuclide group for long term doses (one year period), there is no effect on the evaluation of the radiological impact.

Two accident scenarios were considered:

a. Line Break Accident - a loss-of-coolant accident initiated by a major break in a primary coolant pipe; and

b. No Line Break Accident - an accident during which all activity released from the fuel remains in the primary coolant.

The postaccident radiation environment was determined by (1) analyzing each system operating following an accident to establish pathways for fission products out of containment; (2) investigating process streams in the auxiliary building to identify contaminated equipment and associated activity levels; (3) calculating the radiation field due to each source with computer codes ISOSHLD (Ref. 3), QAD (Ref. 4), and G^3 (Ref. 5); and, (4) superimposing the effects of all sources to obtain the maximum expected dose rate throughout the plant. The radiation environment was evaluated 1 hour, 1 day, and 1 week following the reactor shutdown that precedes fuel failure.

The potentially contaminated systems considered in identifying postaccident radiation sources included (1) the emergency core cooling system, which consists of all or parts of the safety injection, residual heat removal, and chemical and volume control systems; (2) the containment spray and hydrogen recombiner systems (Hydrogen Recombiners have been abandoned at Braidwood), which assist in maintaining containment integrity; (3) the control room, technical support center, and auxiliary building HVAC systems, portions of which are designed to remove airborne fission products; (4) the high radiation sampling system; (5) those parts of the chemical and volume control and the boron recycle systems associated with charging, letdown, and seal water; (6) the shutdown cooling portion of the residual heat removal (RHR) system; and, (7) those parts of the gaseous waste processing system and the liquid radwaste system which would normally operate in conjunction with the other systems under consideration. The large amount of equipment in the auxiliary building operating following an accident produces elevated dose rates in many areas.

The locations of postaccident vital areas and areas essential for access to vital areas (stairways) are indicated in Figure E.20-1. Also shown in this figure are the sources which may be contaminated with fission products following an accident. Figures E.20-2, E.20-3, and E.20-4 show the radiation environment expected to exist 1 hour, 1 day, and 1 week following the line break accident, and Figures E.20-5, E.20-6, and E.20-7 illustrate the same points in time following a no line break accident. It should be noted that the radiation zones are due to activity released from the fuel because of the accident occurring in Unit 1 only, and do not reflect normally existing radiation levels (see Chapter 12.0). The following addresses each sheet of the zone maps (Figures E.20-2 through E.20-7) and describes the sources contributing to the elevated dose rate levels following each accident:

a. Plant Vicinity (Sheet 1)

   The only source affecting areas outside the plant is the airborne activity inside containment following the line break accident. This activity decays very quickly, as illustrated by comparing Figures E.20-2 and E.20-3. Since there is no significant airborne activity inside containment following the no line break accident, areas outside the plant building do not experience elevated dose rate levels.

b. Main Floor, El. 451 ft 0 in. (Sheet 2)

   The airborne activity inside containment is the primary cause of high radiation zones on these maps. In the containment purge rooms very high radiation zones are caused by the purge duct penetrations, which provide a pathway directly into containment. The pooling effect occurs because the penetrations are 11 feet off the floor, whereas the dose rates were calculated 6 feet off the floor. Radiation from the containment also affects the turbine floor with both direct and air scattered radiation. The dose rate zones in the turbine building are short lived, but the purge room will experience high radiation levels for a long period following an accident. Note that the containment is a significant source only after the line break accident.

The only other source located on this figure is the auxiliary building exhaust filter unit, which becomes contaminated by containment leakage following a line break accident and by pump seal leakage following a no line break accident.

c. Mezzanine Floor, El. 426 ft 0 in. (Sheet 3)

Direct radiation from containment also affects this level of the plant following a line break accident, but existing walls limit the effects to areas adjacent to containment and portions of the turbine building. The highest dose rates due to this radiation occur in the vicinity of the personnel and equipment hatch, which is a less effective shield than the containment wall. Following the no line break case this elevation is not affected until letdown via the volume control tank is established. Then the tank cubicle and valve aisle become high radiation zones.

Additionally, when the primary coolant is degassed, the waste gas compressor and the lines connecting it to the volume control tank and the waste gas decay tanks become radiation sources.

d. Grade Floor, El. 401 ft 0 in. (Sheet 4)

Direct radiation from containment again generates dose rates in the turbine building and parts of the auxiliary building adjacent to containment following a line break accident. However, in the two vertical pipe chases next to the fuel transfer tube the dose rate is dominated by containment spray system piping. At 1 day following the accident (Figure E.20-3, Sheet 4), the hydrogen recombiner is put into operation and remains a radiation source for the rest of the accident. If there is no line break, the only abnormal radiation levels at 1 hour are caused by letdown equipment located on the elevation below. Once letdown to the volume control tank is established, pipes in the vertical pipe chase leading to the tank cubicle become contaminated and affect the area around the pipe chase. The line connecting the volume control tank to the waste gas compressors runs through the Unit 2 chiller pump room and causes elevated dose rate levels when the primary coolant is degassed. The sample room is the only high radiation area on all maps, and at 1 hour it limits access to the stairway bordering it to the south.

e. Basement Floor, El. 383 ft 0 in. (Sheet 5)

This figure shows the highest elevation on which ECCS equipment is located, and following a major line break accident the radiation zones are dominated by the RHR heat exchangers and the ECCS piping in the pipe chases adjacent to containment. Radiation zones 1 hour following an accident involving no line break are caused by the parts of the CVCS contaminated by letdown via the letdown heat exchanger to the recycle holdup tank. Note that the tank and its associated piping, which are located on the elevation below, are strong enough radiation sources to affect the area immediately above them. By 1 day following the accident, normal letdown and RHR cooling have been established, contaminating the RHR and seal water heat exchangers. Also by this time the degassing of the primary coolant has contaminated the lines to the waste gas system. The other major source on this elevation is the high radiation sample system drain tank room, and it becomes contaminated 1 day following both accidents.

f. Basement Floor, El. 364 ft 0 in. (Sheet 6)

This elevation is the primary ECCS equipment floor and contains the RHR heat exchangers, charging and safety injection pumps, and associated piping. All this equipment, and the floor drain collection tanks, are contaminated within 1 hour of a line break accident and remain contaminated throughout the accident. One hour following a no line break accident contaminated equipment on this elevation includes the recycle holdup tank and piping associated with coolant letdown. By 1 day the ECCS equipment and floor drain tank are also contaminated. The chemical drain tank, which is connected to the sample system, becomes contaminated 1 day after both accidents.

g. Basement Floor, El. 346 ft 0 in. (Sheet 7)

This elevation contains four major sources, the RHR pumps, the containment spray pumps, the recycle holdup tanks, and the waste gas decay tanks. In the sub-basement, elevation 330 feet 0 inch, are the floor drain collection sumps, which are connected to the floor drain tanks on elevation 364 feet 0 inch by a single line. Following a line break accident, the RHR and containment spray equipment is contaminated, and, since a pump seal rupture is assumed to occur during this time period, the floor drain collection sump and the pipe connecting it to the collection tank are also contaminated. Following a no line break accident the recycle holdup tank is the only major source at 1 hour, but after RHR cooling is initiated, pump leak occurs, and the resulting zones include the effects of the collection sump and its associated piping. One day after this accident, degassing of the primary coolant causes the waste gas decay tank to become a major source.

h. Miscellaneous Floors (Sheet 8)

The only sources which affect these floors, other than direct radiation from containment, are the control room intake filters located on elevation 463 feet 4 1/2 inches. These become contaminated following a line break accident.

i. HVAC Equipment Floors (Sheet 9)

These floors contain the auxiliary building exhaust filters, which are contaminated by containment leakage following a line break accident and by equipment leakage following an accident without a line break. Additionally, the containment purge penetrations cause high radiation levels on elevation 467 feet 4 inches.

j. Fuel Handling Building (Sheet 10)

There are no postaccident sources located in the fuel handling building, but the pipe chase adjacent to containment contains both ECCS and CVCS piping, and therefore, is contaminated in all accident phases.

k. Radwaste/Service Complex/TSC (Sheet 11)

There are no postaccident sources located in the radwaste/service complex, but if a line break accident occurs in Unit 2, direct radiation from containment will affect the radiation environment in this area during the first day following the accident. Also following a line break accident, the intake filter for the TSC becomes contaminated within 1 hour of the accident.

In general, the shielding design review shows that personnel occupancy in the vital areas will not be unduly restricted by postaccident radiation fields. A significant radiological conclusion is that the "less than 15 mr/hr" criterion is met at B/B for plant areas requiring extended or continuous occupancy (the control room, the technical support center, and the radwaste control room where the remote shutdown panels are located). Additionally, application of General Design Criterion 19 accident limit of 5 rem whole body (or equivalent) for areas requiring infrequent access indicates that adequate occupancy times are available for typical operator actions in the remaining vital areas. Postaccident dose rates from contained sources are shown in Table E.20-1 for select vital areas. Table E.20-2 gives the postaccident doses for personnel transit between selected plant control centers and other postaccident vital areas. This table gives a conservative estimate of the travel dose due to contained sources in the auxiliary building and airborne activity in containment.

Additional vital areas where infrequent, short-term, entry may be required to perform actions necessary for accident mitigation such as manual valve operation or motor control center breaker operation are depicted in Figure E.20-1. Vital activities involving infrequent occupancy meeting the GDC 19 requirements include:

a. Auxiliary Building el. 401 general area (<15 mR/hr; Approx. 15 mR total) - Startup of the Hydrogen Monitor

b. Auxiliary Building el. 364 general area (<15 mR/hr; Approx. 120 mR total) - Post-accident realignment of CC system

c. Auxiliary Building el. 346 general area (<15 mR/hr; Approx. 15 mR total) - Post-accident alignment of the 0/1/2SX007 valves

Byron License Amendment 147 and Braidwood License Amendment 140 revised design bases post-accident dose analyses based on Alternative Source Term (AST) methodology. For the accidents analyzed using AST, Control Room post-accident dose rates from contained sources were increased for conservatism, to maintain margin due to the increased assumed containment leakage. However, in all practicality, they will remain approximately the same based on the release timing and isotopic mix attributes of the AST source term. The overall effect of these doses is insignificant relative to the total post-accident dose to the operators. In addition, the AST License Amendments did not impact the design of plant shielding.

The methodology used to postulate values for postaccident radiation levels and application of these values in qualification of equipment was completed in accordance with NUREG-0588 and NUREG-0737 requirements. NRC Staff review concluded that these values were acceptable as used (SER, Supplement 5, Section 3.11.3.3.6).

Evaluations of the environmental qualifications of essential equipment, which demonstrates that the equipment will not be unduly degraded by postaccident radiation fields, is documented in the equipment qualification documentation packages for the affected equipment.

TABLE E.20-1

POSTACCIDENT DOSE RATES FROM CONTAINED SOURCES

(in rem/hr)

LOCATION	1-HOUR	1-DAY	1-WEEK
Control Room	<0.001	<0.001	<0.001
Remote Shutdown Panels	<0.015	<0.015	<0.015
Technical Support Center	<0.001	<0.001	<0.001
Primary Sample Room	<0.1	<0.1	<0.1
Laboratories	<0.015	<0.015	<0.015
Hydrogen Recombiner Control Panel (Hydrogen Recombiners have been abandoned at Braidwood)			
Unit 1	<0.1	<0.1	<0.1
Unit 2	<0.015	<0.015	<0.015
Pathway to TSC	<1.0	<0.015	<0.015
Pathway to Remote Shutdown Panel	<0.015	<0.015	<0.015

TABLE E.20-2

POSTACCIDENT DOSES FOR ESSENTIAL POSTACCIDENT PATHS[3]

(DOSES ARE IN MILLIREMS FOR SELECTED POSTACCIDENT TIMES)

	WALK[1]			RUN[2]		
PATHWAY	1 Hour	1 Day	1 Week	1 Hour	1 Day	1 Week
TSC to CR	5.77	0.33	0.02	2.89	0.17	0.01
RSCP to CR	0.42	0.42	0.03	0.25	0.25	0.02
LABS to CR	0.02	0.02	(4)	0.01	0.01	(4)
HRSS to LABS	1.39	0.21	0.01	0.80	0.12	0.01
HRCP-1 to CR (Hydrogen Recombiners have been abandoned at Braidwood)	0.29	0.29	0.02	0.16	0.16	0.01
HRCP-2 to CR (Hydrogen Recombiners have been abandoned at Braidwood)	0.01	0.01	(4)	0.01	0.01	(4)

CR - Control Room, TSC - Technical Support Center, RSCP - Remote Shutdown Control Panel, LABS - Laboratories, HRSS - High Radiation Sample System, HRCP-1 (2) - Hydrogen Recombiner Control Panel, Unit 1 (Unit 2).

Note:
1. Walking Speeds: 300 ft/min horizontal, 50 ft/min upstairs, 90 ft/min downstairs.
2. Running Speeds: 600 ft/min horizontal, 80 ft/min upstairs, 120 ft/min downstairs.
3. Sources: All contained sources in the auxiliary building and airborne activity in containment based on NUREG-0737. Dose due to plume is not included.
4. Dose is less than 0.01 millirems.

E.21 POSTACCIDENT SAMPLING (II.B.3)

Braidwood License Amendment No. 121 and Byron License Amendment No. 126 approves the elimination of the requirement to have and maintain the Postaccident Sampling system. The following items were committed to as part of License Amendment 121 and 126:

1. Exelon has developed contingency plans for obtaining and analyzing highly radioactive samples of the reactor coolant, containment sump, and containment atmosphere. The contingency plans will be contained in the Braidwood Station and Byron Station Chemistry procedures and implemented with the implementation of the License amendment. Establishment of contingency plans is considered a regulatory commitment.

2. The capability for classifying fuel damage events at the Alert level threshold will be established at a level of core damage associated with radioactivity levels of 300 μCi/gm dose equivalent iodine. This capability will be described in our emergency plan and emergency plan implementing procedures and implemented with the implementation of the License amendment. The capability for classifying fuel damage events is considered a regulatory commitment.

3. Exelon has established the capability to monitor radioactive iodines that have been released offsite to the environs. This capability is described in our emergency plan and emergency plan implementing procedures. The capability to monitor radioactive iodines is considered a regulatory commitment.

Pages E.21-1a through E.21-15 have been deleted intentionally.

E.21.1 References

1. "Issuance of Amendments for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2 – To Eliminate Requirements for Post Accident Sampling System," dated December 27, 2001.

E.22 TRAINING FOR MITIGATING CORE DAMAGE (II.B.4)

POSITION:

The Byron Station Training Department has presented to all licensed operators, including the station manager, the Mitigating Core Damage Training Course as conducted by the staff of the Westinghouse Training Center at Zion, Illinois, prior to writing the NRC license exam.

Braidwood licensed operators and the station manager have received Mitigating Core Damage Training at the Production Training Center.

At Byron and Braidwood Stations, replacement licensed operators receive Mitigating Core Damage Training in accordance with NUREG-0737 requirements as part of initial license training.

Managers and technicians in the Instrument Maintenance Departments receive training on the incore thermocouple system and the use of this system during abnormal containment environments.

Managers and technicians in the Radiation Protection and Chemistry Departments receive training on radio-chemistry makeup, radiation levels, and gas generation resulting from postulated class 9 events.

E.23 RELIEF AND SAFETY VALVE TEST REQUIREMENTS (II.D.1)

POSITION:

By letter dated April 1, 1982, D. P. Hoffman (Consumers Power) transmitted the Safety and Relief Valve Test Report for the EPRI PWR Safety and Relief Valve Test Program. This report summarizes all the operability test data collected on relief and safety valves. Byron/Braidwood plants have Copes-Vulcan Model D-100-160 3-inch air-operated globe relief valves (316SS l/stellite clad plug and 17-4PH cage) and Crosby Model HP-BP-86, size 6M6 safety valves. Specific results in Sections 3.5 and 4.6 of the EPRI safety and relief valves test report are applicable to Commonwealth Edison plants. Final evaluation of the data indicates that the relief and safety valves will perform their intended functions for all expected fluid inlet conditions. Commonwealth Edison submitted the plant specific final evaluation confirming the adequacy of the relief and safety valves, and other plant specific data for all relief and safety valve inlet conditions, by letter from T. R. Tramm dated October 26, 1982.

Regarding verification of block valve functionability, this topic was discussed between the PWR utilities and the NRC staff. Commonwealth Edison concurs with the final conclusions reached between the PWR owners and the NRC staff.

E.24 DIRECT INDICATION OF RELIEF AND SAFETY VALVE POSITION (II.D.3)

POSITION:

The reactor coolant system motor-operated pressurizer relief isolation valves (lRY8000A and B), pressurizer power-operated relief valves (lRY455A and lRY456) and the pressurizer safety relief valves (lRY8010A through C) are provided with positive position indication in the main control room.

A discussion of the clarification items as listed in NUREG-0737 follows:

1. The basic requirement is to provide the operator with unambiguous indication of valve position (open or closed) so that appropriate operator actions can be taken.

   For the motor-operated pressurizer relief isolation valves (lRY8000A and B), valve operator mounted limit switches provide positive position indication at the main control room valve control switch indicating lights. For the pressurizer power operated relief valves (lRY455A and lRY456), externally mounted limit switches provide positive position indication at the main control room valve control switch indicating lights. For the pressurizer safety relief valves (lRY8010A through C), valve mounted reed switches provide positive position indication at the main control room valve position indicating lights.

2. The valve position should be indicated in the control room. An alarm should be provided in conjunction with this indication.

   All valve position indications are located in the main control room. Main control room annunciator alarms are also provided for all valves.

3. The valve position indication may be safety grade. If the position indication is not safety grade, a reliable single-channel direct indication powered from a vital instrument bus may be provided if backup methods of determining valve position are available and are discussed in the emergency procedures as an aid to operator diagnosis of an action.

   All valve position indications are safety grade.

4. The valve position indication should be seismically qualified consistent with the component or system to which it is attached.

Reference the response to clarification item 5 below.

5. The position indication should be qualified for its appropriate environment (any transient or accident which would cause the relief or safety valve to lift) and in accordance with Commission Order, May 23, 1980 (CLI-80-21).

   For the motor-operated pressurizer relief isolation valves (RY8000A and B) the position limit switches will be qualified per Reference 1 of Subsection 3.11.11. For the pressurizer power-operated relief valves (RY455A and RY456) the position limit switches will also be qualified per Reference 1 of Subsection 3.11.11. For the pressurizer safety relief valves (RY8010A through C) the position limit switches will be qualified to IEEE 323-1974. All position indication in the main control room will be qualified to IEEE 323-1974.

6. It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human factor analysis should be performed taking into consideration:

   a. the use of this information by an operator during both normal and abnormal plant conditions,

   b. integration into emergency procedures,

   c. integration into operator training, and

   d. other alarms during emergency and need for prioritization of alarms.

The review of the main control room position indications for the reactor coolant system relief and safety valves is included in the Byron/Braidwood Station Control Room Design Review for NUREG-0700.

E.25 AUXILIARY FEEDWATER SYSTEM EVALUATION (II.E.1.1)

POSITION:

The following analysis of the Byron/Braidwood auxiliary feedwater system was done:

1. A point-by-point review of the auxiliary feedwater system design against Subsection 10.4.9 of the Standard Review Plan and Branch Technical Position ASB 10-1;

2. A reliability study discussed in NUREG-0611;

3. A point-by-point review of the auxiliary feedwater system design; technical specifications and operating procedures against the generic short-term and long-term requirements discussed in the March 10, 1980 letter (Ref. 8);

4. A design basis evaluation of the auxiliary feedwater system; and,

5. The Byron/Braidwood auxiliary feedwater system (AFWS) design has been evaluated to, and found to meet, the requirements of Item II.E.1.1. The B/B AFWS Technical Specification was generated with consideration of Generic Letter 83-37, and meets requirements for ensuring operability. Therefore, this item is considered closed.

E.26 AUXILIARY FEEDWATER SYSTEM AUTOMATIC INITIATION AND FLOW INDICATION (II.E.1.2)

POSITION:

Part 1: Auxiliary Feedwater System Automatic Initiation

Automatic initiation of the auxiliary feedwater system is part of the engineered safety features actuation system (ESFAS) described in Section 7.3. The conformance of the ESFAS to the requirements of IEEE 279-1971 is detailed in Table 7.3-4. Other supporting information for the auxiliary feedwater system are found as follows:

a. system design description - Subsection 10.4.9,

b. piping and instrument diagrams – Drawing M-37,

c. test procedures - Chapter 14.0,

d. Technical Specifications, and

e. schematic and logic diagrams - submitted to NRC-ICSB during FSAR review period.

Part 2: Auxiliary Feedwater System Flowrate Indication

Auxiliary feedwater flowrate indication is provided in the main control room for each of the redundant train auxiliary feedwater flow paths to each steam generator. In addition, a wide range level indication is provided in the main control room for each steam generator. The auxiliary feedwater flowrate indication instrumentation is a Class 1E design with ESF power supplies.

Additional details of the auxiliary feedwater flow instrumentation can be found in Subsections 7.3.1.1.6.e and 10.4.9.2.1.4.

E.27 EMERGENCY POWER SUPPLY FOR PRESSURIZER HEATERS (II.E.3.1)

POSITION:

As outlined in Chapter 8.0 of the UFSAR, the Byron/Braidwood distribution system is designed with Class 1E qualified breakers between the 4160-volt ESF buses and the 4160-volt non-safety-related buses. They can be closed to provide emergency power to the pressurizer heaters by manual operator action. For Byron/ Braidwood, one bank of backup heaters from each redundant power supply can be connected to maintain natural circulation after loss of offsite power. The circuits are designed to automatically shed from the emergency power sources upon the occurrence of a safety injection actuation signal. The requirement to manually load the pressurizer heater on to the emergency power sources has been included in the Station's emergency procedures.

E.28 DEDICATED HYDROGEN PENETRATIONS (II.E.4.1)

POSITION:

Two permanently installed hydrogen recombiners are provided at the Byron/Braidwood Stations. The suction, discharge, and cross-tie piping and valves are ASME III, Class 2 and utilize penetrations dedicated for this system. At Braidwood, the hydrogen recombiner suction, discharge, and cross-tie piping and valves, with the exception of the containment penetration piping and isolation valves, have been abandoned in place. The containment purge system is safety-related and utilizes dedicated penetrations per code requirements for the purge system. Redundancy and single failure requirements of General Design Criteria 54 and 56 of 10 CFR 50, Appendix A are met.

E.29 CONTAINMENT ISOLATION DEPENDABILITY (II.E.4.2)

POSITION:

The containment isolation system for Byron/Braidwood is an automatic, redundant, safety grade system which has diverse parameters for actuation. The system functional diagrams are provided in Drawing series 108D685.

The following parameters are monitored for the initiation of containment isolation:

a. automatic safety injection,

b. containment pressure,

c. steamline pressure, and

d. pressurizer pressure.

Table 6.2-58 provides a summary of containment isolation signals and identifies the essential and nonessential systems that provide a possible open path out of the primary containment through Class B penetrations. Such systems are either automatically isolated by isolation signals, by check valves that would prevent flow out of the containment, by manual valves that are normally closed during reactor operation, or as in the case of instrument lines, by a closed piping system. All systems not required for hot shutdown are automatically isolated by the containment isolation signal. All systems penetrating the containment were designed to the requirements of General Design Criteria 54, 55, 56, and 57.

The individual control circuits are designed to prevent automatic loss of containment isolation due to the resetting of the isolation signals. Deliberate operator action is required to open the containment isolation valves after resetting the actuating signal. Each containment isolation valve must be opened individually.

The containment isolation setpoint pressure is 3.4 psig. This value is used in all analyses of the capability of the containment to withstand and contain the results of postulated line breaks. Operating plant experience indicates that use of this setpoint pressure will not result in unnecessary isolation signals. Analytical results show that the containment pressure and offsite releases will stay well below limits and that safety systems will work properly with this setpoint.

A high radiation signal, separate from the containment isolation signal, will close the containment purge and vent isolation valves. Area radiation detectors RE-AR011 and RE-AR012 are interlocked with containment purge isolation valves VQ001A and B and VQ002A and B and containment mini-purge isolation valves

VQ003, VQ004A and B, and VQ005A, B, and C. Upon detection of high radiation levels, containment ventilation isolation signal will be initiated and the above-mentioned valves that are open will be closed. It should be noted that the containment ventilation isolation signal is separate from either the Phase A or Phase B containment isolation signal as shown in Table 6.2-58.

In order to support the containment isolation function, the 48-inch valves of the normal containment purge system are required to be maintained closed in all plant modes other than those of cold shutdown or refueling. These valves are under administrative control per ANSI N271-1976. These valves are verified to be closed at least once every 31 days by checking position indication in the control room.

The normal containment purge valves are locked closed by the administrative procedure of interrupting power to the valve at the circuit breaker; i.e., the circuit breaker is racked out (open) and the breaker is tagged as out of service. Inadvertent operation of the purge valves requires violation of procedures prohibiting both the operation of tagged-out equipment and the containment purge system. Tagging out at the breaker is considered equivalent to a mechanical lock because in both instances positive action is used to prevent the valve from receiving power and an administrative procedure is required to return the breaker to service. At Braidwood, valves VQ001A/B and VQ002A/B have exterior mechanical stops mounted to the valve. These valve stops are used as an additional method of locking the valves closed.

Valves VQ003, VQ004A/B, and VQ005A/B/C are equipped with an operator capable of closing the valves in 5 seconds for containment isolation (see Table 6.2-58). These 8-inch post-LOCA purge and miniflow purge valves meet the guidance of Branch Technical Position CSB 6-4.

E.30 ADDITIONAL ACCIDENT-MONITORING INSTRUMENTATION (II.F.1) POSITION:

1. Noble Gas Effluent Monitor (II.F.1-1)

a. Auxiliary Building Vent Stack

Two General Atomic Company wide-range monitors are installed on the auxiliary building vent stacks (final release points), one monitor per stack. The monitor has a range for radioactive gas concentration of 1 x 20^{-7} μCi/cc to 1 x 10^{+5} μCi/cc. The monitor is designed to meet 1E requirements and is qualified to IEEE 323-1974. The wide-range gas monitor meets the requirements of Table II.F.1-1 of NUREG-0737. The monitor includes the following: one set of isokinetic nozzles and one 3/4 inch, heat traced sample line for both the low and mid/high range operating conditions at the constant flow rate of 1.67 scfm; an auxiliary pump skid with automatic isokinetic flow control connected to the 3/4 inch sample line near the flow splitter; sample rack (reference discussion of II.F.1-2); sample conditioner, operating only at high range conditions to filter out large concentrations of radioiodines and particulate; wide-range gas detectors assembly, consisting of three radioactive gas detectors, a low-range detector and mid/high range detectors. Each monitor system has a microprocessor which utilizes digital processing techniques to analyze data and control monitor functions. Control room readouts include an RM-23 remote display module for all monitor parameters.

1. The calibration techniques and procedures including the energy dependence of the detectors is provided to meet the requirements of NUREG-0737.

2. The monitors receive power from ESF buses.

3. Postaccident plant release rate calculations can be made using station procedure and vent stack monitor readings. Mid- and high-range WRGM detector readings and stack flow rates are obtained from control room displays and used in the following equation to determine the noble gas release rate.

Detector Reading (μCi/cc) x Stack Flow (cc/sec) x Release Rate Correction Factor = μCi/sec

where, the Detector Reading - the mid or high range noble gas detector reading based on the detector's response to a pseudonoble gas (0.8 MeV discrete gamma and 1.68 MeV maximum beta).

and, the Release Rate Correction Factor = corrects the pseudonoble gas release rate to the actual noble gas mix release rate as a function of time after shutdown.

b. Main Steamline

Two General Atomic Company RD-10B detectors are provided for each of the four main steamlines upstream of the safety and relief valves. The range of the monitor is 1 x 10^{-1} mR/hr to 1 x 10^{4} mR/hr. The monitor is designed to meet Class 1E requirements and is qualified to IEEE 323-1974. The monitors are mounted external to the main steamline piping and corrections are made for the loss of low energy gammas.

The detectors are connected to local mounted microprocessors that collect and store data. Main control room mounted remote readout modules are connected directly to the microprocessors to provide information to the operator during and following an accident. Conversion of detector readings to release rate (μCi/sec) is accomplished through the use of a station procedure.

2. <u>Sampling and Analysis of Plant Effluents (II.F.1-2)</u>

The General Atomic Company wide range gas monitor includes a sampling rack for collection of the auxiliary building vent stack particulate and radioiodine samples. Filter holders and valves are provided to allow grab sample collection for isotopic analyses in the station's counting rooms. The sampling rack is shielded to minimize personnel exposure. The sampling media will be analyzed by a gamma ray spectrometer which utilizes a Ge(Li) detector. Filter cartridges will be reverse blown with air to purge interfering noble gases.

The sampling system is designed such that radiation exposures are within the requirements of GDC 19 as stated in NUREG-0737. The sample media used for both iodine and particulates sample collection meets the requirements for effective adsorption and retention as stated in NUREG-0737.

<u>Exception to II.F.1-2 Criteria</u>

Table II.F.1-2 of NUREG-0737 indicates that an iodine source term of 100 μCi/cc and a sample time of 30 minutes are to be used as the design basis shield envelope for postaccident grab sampling of vent stack effluent. The Licensee asserts that these criteria are unnecessarily conservative and has verified that the existing equipment and procedures are adequate to keep personnel exposures during sampling well within the limits imposed by GDC 19 (5 rem whole body and 75 rem extremity), assuming release of 1.5% of the core iodine inventory.

The Licensee has performed a source term calculation based on a loss of all a-c power scenario. In this scenario, 100% of the noble gas core inventory and 1.5% of the iodine core inventory is released 33 hours after the event occurs. The duration of the release is 4 hours. At the time of the release, it is assumed that one auxiliary building vent fan is operable and that the charcoal filters are 10% efficient in filtering iodines. The calculated vent stack effluent iodine source term is 1.83 μCi/cc (Reference 1). The calculated dose rate from the General Atomic wide range gas monitor grab sample assembly, assuming a 1-minute grab sample and a monitor flow rate of 0.06 cfm, is 9.37 mR/hr at 1 foot.

Table E.30-1 gives a summary of the dose accumulated in the process of going to the General Atomic wide range gas monitor, obtaining the grab sample, and transporting the samples back to the laboratory for analysis. The whole body dose accumulated by an individual in performing this task is about 638.7 mrem. Since two individuals are needed to perform this task, the total dose for performing the task is 1.2774 rem whole body. The extremity dose to individuals collecting the sample is not expected to exceed 2 rem.

Reference 1: Letter from D. H. Smith (CECo) to H. R. Denton, dated 10-24-84.

3. Containment High-Range Radiation Monitor (II.F.1-3)

In accordance with NUREG-0737 the following required documentation items as listed in the NUREG are listed with the response for the Byron/Braidwood Stations:

1. The description of or name of manufacturer and model number of the monitors;

   General Atomic Company Model RD-23 high range radiation detector, Model RM-80 microprocessor and RM-23 remote display unit.

2. Verification that the monitors meet the specifications of Table II.F.1.3;

   REQUIREMENT - The capability to detect and measure the radiation level within the reactor containment during and following an accident.

   Comply.

RANGE - $1R/hr$ to 10^7 (gamma only).

Comply.

RESPONSE - 60 KeV to 3 MeV photons, with linear energy response ± 20% for photons of 0.1 MeV to 3 MeV. Instruments must be accurate enough to provide usable information.

Comply.

REDUNDANT - A minimum of two physically separated monitors (i.e., monitoring widely separated spaces within containment).

Comply.

DESIGN AND QUALIFICATION - Category 1 instruments as described in Appendix A except as listed below.

It is assumed Category 1 refers to Regulatory Guide 1.97. The design of the high range containment monitors complies with the Category 1 instrument requirements of Regulatory Guide 1.97.

SPECIAL CALIBRATION - In situ calibration by electronic signal substitution is acceptable for all range decades above 10R/hr. In situ calibration for at least one decade below 10R/hr shall be by means of calibrated radiation source. The original laboratory calibration is not an acceptable position due to the possible differences after in situ installation. For high-range calibration, no adequate sources exist, so an alternate was provided.

A General Atomic Company RT-11 portable calibration source will be used for the first decade requirement. Additionally, electronic operability check is provided by means of an internal current source corresponding to $10^5R/hr$.

SPECIAL ENVIRONMENTAL QUALIFICATION - Calibration and type-test representative specimens of detectors at sufficient points to demonstrate linearity through all scales up to $10^6R/hr$. Prior to initial use, calibration of coil detector for at least one point per decade of range between 1 R/hr and 10^3 R/hr will be certified.

The radiation detectors have been tested over a range of 43.5 KeV to 4.5 MeV and at 5.17×10^6 R/hr. Sufficient tests have been performed to demonstrate linearity. Calibration prior to initial use are performed by the manufacturer at 200 R/hr, 2000 R/hr, and 20,000 R/hr to satisfy special environmental calibration requirements.

3. Verification that the monitors will be operable on June 15, 1984.

   The requirement, as stated in Enclosure 2 of NUREG-0737, of the implementation four months prior to issuance of an operating license have been met.

4. A plant layout drawing showing the location of the monitors.

   Figure E.30-1 shows the locations of the detectors in both the Unit 1 and Unit 2 containments.

4. Containment Pressure Monitor (II.F.1-4)

For containment pressure, the requirements of NUREG-0737 and Regulatory Guide 1.97 are met by pressure transmitters with diaphragm seals.

The existing pressure measurement instrumentation consists of four channels, with each channel having a range of 0 to 50 psig. The four channels provide inputs to the reactor protection system. Additional pressure measurement instruments are provided as follows:

a. Qualification to IEEE 323-1974 and IEEE 344-1975.

b. The range of each channel is 5 psia to 150 psig.

c. Two channels are provided with one transmitter/ diaphragm seal in each.

d. Continuous display and recording are provided in the main control room.

5. Containment Water Level Monitor (II.F.1-5)

For Byron and Braidwood containment wide range water level (PC006 and PC007), the requirements of NUREG-0737 and Regulatory Guide 1.97 are met by differential pressure transmitters with diaphragm seals. For Byron Unit 1 and Braidwood Unit 1 narrow range containment water level (Containment Floor Drain Sump Level Instruments - PC002 and PC003), the requirements of NUREG-0737 and Regulatory Guide 1.97 are met by differential pressure transmitters with diaphragm seals. For Byron Unit 2 narrow range containment water level, the requirements of NUREG- 0737 and Regulatory Guide 1.97 are met by float-type resistance level transmitters with signal conditioners. For Braidwood Unit 2 narrow range containment water level, the requirements of NUREG-0737 and Regulatory Guide 1.97 are met by a thermal dispersion type level measurement system.

Two channels of level measurement are provided for each application of containment level and containment floor drain sump level as follows:

a. Qualified to IEEE 323-1974 and IEEE 344-1975.

b. Containment wide range water level is measured from the bottom of containment to the equivalent level of 600,000 gallons of water. (PC006 and PC007).

c. The low reference point for containment water level and containment floor drain sump level is as near containment bottom and containment floor drain sump bottom; respectively, as physically possible.

6. Containment Hydrogen Monitor (II.F.1-6)

The requirement to have a continuous indication of containment hydrogen concentration available in the control room is met with redundant Teledyne Analytical Instruments Model 225CM monitoring units. Their capability covers a split range of 0-10% and 0-30% hydrogen concentration by volume (dry analysis) over a pressure regime of -5 psig (9.7 psia) to +50 psig. The monitors are IEEE 323-1974 qualified. The units have an accuracy of ±2.5% of full scale.

The hydrogen monitors are located at auxiliary building elevation 401 feet. Samples are piped from containment penetrations to the monitors. The mechanical piping penetrations used for the hydrogen monitoring system are as follows:

Penetration	Byron	Braidwood
1PC-12, 2PC-12	Train A discg.	spare
1PC-31, 2PC-31	Train B discg.	spare
1PC-36, 2PC-36	Train B suction	Train B suction/discg.
1PC-45, 2PC-45	Train A suction	Train A suction/discg.

Additional information concerning the mechanical penetration's elevations an azimuths are listed in Table 3.8-1. The portions of the hydrogen monitoring piping system which form the containment atmosphere isolation barrier are designated Seismic Category I, Quality Group B. The remainder of the system outside the containment is Seismic Category I, Quality Group B up to the hydrogen monitoring instrumentation. Piping internal to the instrumentation is classified as ANSI B31.1. The piping from the containment to the first isolation valve will be designed to the requirements of SRP 3.6.2.

Operation of the hydrogen monitors is independent of the hydrogen recombiner (Hydrogen Recombiners have been abandoned at Braidwood) since both systems used separate piping and containment penetrations and are not dependent upon the other to operate in any way. The hydrogen monitoring system consists of two independent, physically separated and redundant subsystems and, thus, meets the single-failure criteria. Separate piping penetrations of the containment are utilized by each train of this system. Each train's hydrogen monitor discharge containment isolation valve (PS230A/B) and one of two series inlet containment isolation valves (PS228A/229B) are powered from separate 1E sources. The second inlet containment isolation valve (PS228B/229A) is powered from the alternate power train. Isolation Valves PS228B and PS229A are designed to fail open on loss of power. Thus, failure of one of the 1E electric power sources will disable only one train of the hydrogen monitoring system.

Byron/Braidwood stations meet the requirements for continuous indication in the main control room with IEEE 323-1974 qualified indicators. The monitors may be controlled from the control room.

Technical Specification Amendment Nos. 143 and 137 for Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2, respectively, approved the removal of the hydrogen recombiners and the containment hydrogen monitors from the Technical Specifications. The Technical Specification Amendments are based on a revision to 10 CFR 50.44, "Combustible gas control for nuclear power reactors," which eliminated the design basis loss-of-coolant hydrogen release since it was determined not to be risk significant. With the elimination of the design basis loss-of-coolant hydrogen release, hydrogen monitors are no longer required to mitigate design basis accidents. However, because the hydrogen monitors are required to diagnose the course of beyond design basis accidents a regulatory commitment was made to maintain a hydrogen monitoring system capable of diagnosing the course of beyond design basis accidents.

The hydrogen monitoring system was originally designed to meet the requirements of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Category 1 instruments. Regulatory Guide 1.97 Category 1 is intended for key variables that most directly indicate the accomplishment of a safety function for design basis accident events and provides for full qualification, redundancy, and continuous real-time display and requires onsite (standby) power. Based on the revision to 10 CFR 50.44, the hydrogen monitors have been reclassified as Regulatory Guide 1.97 Category 3 instruments. Category 3 is the least stringent in that it provides for high quality commercial grade equipment that requires only offsite power. Category 3 instruments do not require seismic qualification or redundancy. The design of the hydrogen monitoring instrumentation was based on the original Regulatory Guide 1.97 Category 1 classification that exceeds the design requirements for Regulatory Guide 1.97 Category 3 instrumentation. Therefore, no changes are required for the existing instrumentation as a result of the change in Regulatory Guide 1.97 classification.

SAMPLE CONDITIONING

The Model 225CM monitoring system is designed to monitor containment gas for percentage by volume of hydrogen (dry analysis). The operating range is -5 to +50 psig, 40°F to 445°F and relative humidity from 10 to 100%. A sample of the containment atmosphere will be taken at or near one of the containment penetrations and another approximately 180 degrees away on the other side of the containment (approximately 135 degrees away for Byron Unit 2 only). The samples taken are representative of the containment atmosphere due to the mixing system effects, which is discussed in Subsection 6.2.5.2.3. Radioactive sample gas is drawn from the containment vessel by means of a sample pump into the analysis unit precooler where it is lowered from temperatures as high as 445°F to ambient temperature of the analyzing unit. A solid state self-regulating thermoelectric cooler further reduces the gas temperature to below analysis unit ambient; after which 0.4 scfh of sample gas is directed to the sample measuring cell maintained at 170°F.

After the gas passes through the cell, it is returned to the containment via a pressure regulating network which maintains pressure above containment assuring return of the sample gas. Any condensation formed in either of the coolers is gravity drained to a water trap which is automatically purged back to the containment with the aid of the pressure regulating network.

CALIBRATION

Instrument calibration is performed by actuating the appropriate solenoid valve directing zero or span gas with a known concentration through a flow controller and into the cell.

GAS MEASUREMENTS - GENERAL DISCUSSION

Analysis is accomplished by using the well established principle of thermal conductivity measurement of gases. This technique utilizes two pairs of self-heating filaments fixed in the center of separate cavities inside the analyzing cell housing. One of the cavities in the cell, which is the reference, is connected to a nonflowing desiccated air cannister, while the other cavity is connected to the containment sample source. The filament temperature is determined by the amount of heat conducted by the presence of gas inside the cavity. Thermal conductivity varies with gas species, thereby causing the filament temperature to change as the gas in the sample cavity changes. By using two pairs of filaments in separate cavities and connecting them in an electrical bridge, the differences in thermal conductivity of the gases in the separate cavities may be determined electrically.

Electrical zero is set first by introducing the same quality air as that of the reference cavity into the sample cavity, then adjusting the electrical bridge to balance, resulting in a zero output. As different gases are introduced to the sample cavity, the bridge will become unbalanced and electrical output will amplify with increasing difference in thermal conductivities of the gases used.

Although this technique is nonspecific, it is an extremely reliable technique when the gases or gas mixtures are known, and the variation in composite thermal conductivity can be determined.

HYDROGEN MEASUREMENT

The measurement of containment hydrogen concentration is accomplished by using a thermal conductivity measurement cell with the containment sample flowing through the sample section of the measuring cell. The containment hydrogen concentration is indicated by the difference measured between the sample and reference sides of the cell. The measurement of hydrogen in the presence of nitrogen and oxygen is possible because the thermal conductivity of hydrogen is approximately seven times higher than nitrogen and oxygen, which have nearly the same thermal conductivities.

Teledyne Analytical Instruments hydrogen analysis is a dry basis analysis which will provide a higher hydrogen concentration than actual concentrations during accident conditions, when steam is present. To correct to a wet or actual hydrogen concentration, an isolated analyzer output from each monitor, as well as containment pressure and temperature measurements will be inputted to the plant computer. A program in the plant computer will calculate the actual hydrogen concentration using these inputs and established gas law equations. The calculated actual hydrogen concentration will be available in the main control room at the computer console.

CONTROLS (per monitor)

Calibration, zero and span controls and lights, range selector control switch and a digital readout are located on the local control cabinet. A master off, standby power on, and associated indication lights are located in the main control room.

OUTPUTS (per monitor)

In addition to high hydrogen, and instrument failure alarms, a 4 to 20 mA current output from each analyzer provides the signal which feeds the main control room indicator and process computer. Isolation of the analog output is accomplished prior to the termination at the plant computer. The two alarm outputs are displayed and alarmed on the plant annunciator in the main control room.

The reference and span gas bottles are installed on a seismically mounted bottle rack, and are sized for 100 days of continuous unattended operation during post-LOCA events.

MONITOR ACTUATION AND OPERATION

The operation of these monitors requires 10 to 20 minutes of warmup time for stabilization of the sample chamber. Byron/Braidwood normally maintain these monitors in the STANDBY mode. The STANDBY mode will keep the system energized except for the pump and allow the system to be functional in 4 minutes. Actuation and control of the hydrogen monitors will be from the main control room or at the local panel.

NUREG-0737 originally required that the system be functional in 30 minutes after initiation of a safety injection signal. This requirement has been revised to 90 minutes based on Technical Specification Amendment Nos. 143 and 137 for Byron Station, Units 1 and 2 and Braidwood Station, Units 1 and 2, respectively. The Technical Specification Amendments are based on a revision to 10 CFR 50.44 which eliminated the design basis loss-of-coolant hydrogen release since it was determined not to be risk significant. The 10 CFR 50.44 rule change identifies a change in the hydrogen monitor initiation time from 30 minutes to 90 minutes. The 90 minutes is based on the time needed to get the monitors running in a manner that still meets the goal of monitoring hydrogen levels and allowing sufficient tie for other operator actions based on severe accident emergency operating procedures since hydrogen monitoring is not a near term need.

TABLE E.30-1

TIME MOTION STUDY FOR VENT STACK EFFLUENT GRAB SAMPLE

AREA	TIME SPENT IN AREA	DOSE RATE IN AREA*	DOSE ACCUMULATED IN AREA
OSC to 451' Aux. Bldg.	3 minutes	1 R/hr	50 mR
451' to 426' Aux. Bldg.	1 minute	15 mR/hr	.25 mR
426' to 451' Aux. Bldg.	0.5 minute	10 R/hr	83.5 mR
451' to 477' Aux. Bldg	1.5 minutes	10 R/hr	250 mR
At Monitor	4 minutes	70 mR/hr**	4.7 mR
477' to 451' Aux. Bldg.	1.5 minutes	10 R/hr	250 mR
451' to Lab	1 minute	15 mR/hr	.25 mR
		TOTAL DOSE	638.7 mR

* Dose rates along the path to and from the monitor are based on values specified in the postaccident radiation zone maps (for a major line break inside containment) in Appendix E.20.

** Includes dose rate at 1 foot from grab sample filter assembly, dose rates from sample lines, and dose rates from the high range continuous sample casks.

E.31 INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING (II.F.2)

Position:

Since early 1980, Combustion Engineering, Inc., has been conducting an evaluation of response characteristics of instrumentation under conditions of inadequate core cooling (ICC). An outline of this evaluation was discussed with the NRC staff at a meeting in Bethesda, Maryland, on May 28, 1980. Commonwealth Edison Company evaluated these results and decided to implement a three-element ICC detection instrumentation system which was detailed in the B/B Inadequate Core Cooling Detection System Summary Status Report, April 1982 (letter from T. R. Tramm to H. R. Denton dated June 7, 1982). This section of Appendix E provides a status summary of the ICC detection instrumentation system as of November 1984.

The technique which has been recommended by Combustion Engineering, Inc., is use of two heated junction thermocouple (HJTC) instruments with sensors at several axial locations in the reactor vessel above the fuel alignment plate. The design objective of this system is to provide a measurement of the water inventory and its trend in the reactor vessel above the fuel alignment plate. The details of this design activity were discussed with the NRC Staff at meetings in Bethesda, Maryland, on May 28, 1980 and March 4, 1981. Commonwealth Edison Company has installed the HJTC system as a component of an instrumentation system for monitoring inadequate core cooling as described in the following subsections.

The following discussion provides a status of inadequate core cooling activities, including Byron/Braidwood specific design/hardware implementation efforts. Also provided is a design description of the inadequate core cooling detection system, which includes the heated junction thermocouple system, the core exit thermocouple system (CETs) and the subcooled margin monitor.

E.31.1 INADEQUATE CORE COOLING DETECTION SYSTEM STATUS SUMMARY

E.31.1.1 Summary of Activities

This status summary, provided in response to the requirements in Section II.F.2 of NUREG-0737 (Reference 1), describes design and development activities conducted to define and implement a system of instrumentation being used to detect inadequate core cooling (ICC). This summary also provides information specific to Byron/Braidwood Units 1 and 2 in order to demonstrate the applicability of the generic design and development to Byron/Braidwood Units 1 and 2.

Results of initial studies by the C-E Owners Group are documented in reports CEN-117 (Reference 2) and CEN-125 (Reference 3). All studies have been based on the requirement to indicate the approach to, the existence of, and the recovery from ICC.

The ICC system selected was based specifically on the results presented in CEN-185 (Reference 5). The basis for the instruments selected is summarized below.

E.31.1.2 <u>Bases for ICC Instrument Selection</u>

The ICC instrumentation sensor package described herein is designed to:

a. provide the operator with an advance warning of the approach to ICC and

b. cover the full range of ICC from normal operation to complete core uncovery.

The ICC detection system employs this sensor package and displays, trends, and logs the sensor outputs, enables the reactor operator to monitor system conditions associated with the approach to and the recovery from ICC.

E.31.1.3 <u>Description of ICC Event Progression</u>

The instrument sensor package for ICC detection provides the reactor operator with a continuous indication of the thermal-hydraulic state within the reactor pressure vessel (RPV) during the progression of an event leading to and away from ICC. The progression towards and away from ICC can be divided into intervals based on physical processes occurring within the RPV. These are characterized as follows:

<u>Intervals Associated with the Approach to ICC</u>

Interval 1 Loss of fluid subcooling prior to the first occurrence of saturation conditions in the coolant.

Interval 2 Decreasing coolant inventory within the upper plenum (from the top of the vessel to the top of the active fuel).

Interval 3 Increasing core exit temperature produced by uncovering of the core resulting from the drop in level of the mixture of vapor bubbles and liquid below the top of the active fuel.

<u>Intervals Associated with Recovery from ICC</u>

Interval 4a Decreasing core exit temperature resulting from the rising of the mixture level within the core.

Interval 4b Increasing inventory above the fuel.

Interval 4cEstablishment of saturation conditions followed by an increase in fluid subcooling.

These intervals encompass all possible coolant states associated with any ICC event progression. Intervals 1 through 3 refer to fluid situations that occur during the approach to ICC. Intervals 4a, 4b, and 4c refer to fluid situations which occur during the recovery from ICC.

In order to provide indication during the entire progression of an event, an ICC instrument system should consist of instruments which provide at least one appropriate indicator for each of the physical intervals described above.

Applying this description of the "approach to" and "recovery from" ICC to ICC instrument selection:

a. provides assurance that the selected ICC system detects the entire progression.

b. demonstrates the extent of instrument diversity or redundancy which is possible with the available instruments.

Furthermore, by defining the ICC progression on a physical basis, the general labels of approach to, and recovery from ICC can now be associated with specific physically measurable processes (see Subsections E.31.3.1 and E.31.3.3).

The inadequate core cooling instrument sensor package consists of (1) reactor coolant loop wide range pressure sensors, (2) reactor vessel level monitors employing the HJTC concept, and (3) core exit thermocouples. The signals from the temperature and pressure sensors can be combined to indicate the loss of subcooling and occurrence of saturation (Interval 1) and the achievement of a subcooled condition following core recovery (Interval 4c). The reactor vessel level monitors provide information to the operator on the decreasing liquid inventory in the reactor pressure vessel (RPV) regions above the fuel alignment plate, as well as the increasing RPV liquid inventory above the fuel alignment plate following core recovery (Intervals 2 and 4b). The core exit thermocouples monitor the increasing steam temperatures associated with core and the decreasing steam temperatures associated with core recovery (Intervals 3 and 4a).

E.31.1.3.1 Advanced Warning of the Approach to ICC

The ICC instrumentation provides the operator with an advance warning of the approach to ICC by providing indications of:

a. the loss of subcooling and occurrence of saturation (Interval 1) with a subcooled margin monitor receiving input from primary system temperature and pressure sensors.

b. the loss of inventory in the RPV (Interval 2) with the RVLIS.

c. the increasing core coolant exit temperature (Interval 3) with CETs.

It should be noted that the RVLIS measures inventory (collapsed liquid level) rather than two-phase level. This measurement provides the operator with an advanced indication of the coolant level should conditions arise to cause the two-phase froth to collapse via system overpressurization, or the loss of operating reactor coolant pumps.

E.31.1.3.2 Application of the ICC Detection Instruments

Following an event leading to ICC, the ICC detection instruments will provide information to the reactor operator so that he may:

a. verify that the core heat removal safety function is being met.

b. establish the potential for fission product release.

ICC instrumentation indications will be used to support the operator in helping to verify that the core heat removal safety function is being met. ICC instrumentation indications available to the operator are: (1) a decreasing core exit steam superheat, (2) an increasing inventory above the fuel alignment plate, or (3) an increasing subcooling in the RPV or RCS piping.

The operator is informed about the progression of an event by both static and trend displays. The trending of ICC information enables the operator to quickly assess the success of automatically or manually performed mitigating actions. A chart indicating the ICC instrumentation trending during the various ICC progression intervals associated with the approach to and recovery from ICC is presented in Table E.31-1.

E.31.1.3.3 Instrument Range

In the ICC instrumentation sensor package, saturation temperature and water inventory are used as indicators for the approach to and recovery from ICC when there is water inventory above the fuel alignment plate. These measurements characterize Intervals 1, 2, 4b, and 4c of the ICC progression.

When the two-phase level is below the fuel alignment plate, the measurement of core exit fluid temperature represents a direct indication of the approach to and recovery from ICC (Intervals 3 and 4a). Therefore, the ICC sensor package is sufficient to provide information to the reactor operator on the entire progression of an event with the potential of resulting in ICC.

E.31.1.4 Summary of Sensor Evaluations

Several sensors have been evaluated for use in an ICC detection system. Significant conclusions about each instrument are given below. The descriptions do not necessarily describe the actual ICC detection system implementation for the Byron and Braidwood Stations.

E.31.1.4.1 Subcooled Margin Monitor

A subcooled margin monitor (SMM), using inputs from incore thermocouple in the upper intervals, and from the wide range reactor coolant system pressure sensors, is adequate to detect the initial occurrence of saturation during LOCA events and during loss of heat sink events.

However, the usefulness of the SMM can be significantly increased by using the signals from the fluid temperature measurements from the HJTCS and the signals from selected core exit thermocouples and by modifying the SMM to calculate and display degrees superheat (up to about 1800°F) in addition to degrees subcooling. The core exit thermocouples respond to the coolant temperature at the core exit, and their signal indicates superheat after the coolant level drops below the top of the core, thus, the core exit thermocouples provide an approximate indication of the depth of core uncovering.

With this implementation, the SMM can be used for detection of the approach to ICC, namely Interval 1 (loss of subcooling), Interval 3 (core uncovering), Interval 4b (core recovery) and Interval 4c (establishment of saturation conditions). Even with the modifications, the SMM will not be capable of indicating the existence of Interval 2 when the coolant is at saturation conditions and the level is between the top of the vessel and the top of the core.

The recovery interval may occur at low system pressure and temperature. Since the errors in the existing SMM calculations increase with lower temperature and pressure, required subcooling margins need to be revised for this situation.

E.31.1.4.2 Resistance Temperature Detectors (RTDs)

The RTDs are adequate for sensing the initial occurrence of saturation. Narrow range RTDs are located in the hot and cold leg manifolds, and the wide range RTDs are located in the hot and cold legs of the reactor coolant piping. Either of the narrow or wide range RTDs are sufficient to sense saturation for events initiated at power. The wide range RTDs are sufficient to sense saturation for events initiated from zero power or shutdown conditions.

The RTD range is not adequate for ICC indications during core uncovering. For depressurization LOCA events, the core may uncover at low pressure, when the saturation temperature is below the lower limit of the RTD. Initial superheat of the steam will therefore not be detected by the RTD. As the uncovering proceeds, the superheated steam temperature may quickly exceed the upper limit of the RTD range. In order to be useful during the core uncovery interval, the range of RTD would have to be increased to cover a temperature range from 100°F to 1800°F.

E.31.1.4.3 Heated Junction Thermocouple System (HJTCS)

The HJTCS is being designed to show the liquid inventory of the mixture of liquid and vapor coolant above the core. It is an instrument which shows the approach to ICC and is the only one which functions in Interval 2, namely the period from the initial occurrence of saturation conditions until the start of core uncovering and Interval 4b, the period when inventory is increasing above the fuel alignment plate.

E.31.1.4.4 Core Exit Thermocouples

The core exit thermocouples are adequate to show the approach to ICC after core uncovering for the events analyzed, provided that the signal processing and display does not add substantial time delay to the thermal delay at the thermocouple junction. As mentioned above, the core exit thermocouples respond to the coolant temperature at the core exit and indicate superheat after the core is no longer completely covered by coolant.

Except for a time delay, depending on event, the trend of the change in superheat corresponds to the trend of core uncovering as well as to the accompanying trend of the change in cladding temperature.

E.31.2 SYSTEM FUNCTIONAL DESCRIPTION

In the following sections, a functional description of the instruments of the ICC Detection System is given and the function of the instrument is related to the ICC intervals which were described in Subsection E.31.1.

E.31.2.1 Subcooling and Saturation

The parameters measured to detect subcooling and saturation are the RCS coolant temperature and pressure. The measurement range extends from the shutdown cooling conditions up to saturation conditions at the pressurizer safety valve setpoint. The response time needs to be such that the operator obtains adequate information during those events which proceed slowly enough for him to observe and to act upon the information.

The information which is derived from the reactor vessel temperature and pressure measurements is the amount of subcooling during the initial approach to saturation conditions and the occurrence of saturation during Interval 1. During Interval 4, the reestablishment of subcooled conditions is obtained.

E.31.2.2 Coolant Level Measurement in Reactor Vessel

The reactor coolant system is at saturation conditions until sufficient coolant is lost to lower the two-phase level to the top of the active core. During this interval, there are no existing instruments which would measure directly the coolant inventory loss. A reactor vessel level monitoring system provides a direct measurement during this period. The parameter which is measured is the collapsed liquid level above the fuel alignment plate. The collapsed level represents the amount of liquid mass which is in the reactor vessel above the core. Measurement of the collapsed water level was selected in preference to measuring two-phase level, because it is a direct indication of the water inventory while the two-phase level is determined by water inventory and void fraction.

The collapsed level is obtained over the same temperature and pressure range as the saturation measurements, thereby encompassing all operating and accident conditions where it must function. Also, it is intended to function during Interval 4, the recovery interval. Therefore, it must survive the high steam temperature which may occur during the preceding core uncovering interval.

The level range extends from the top of the vessel down to the top of the fuel alignment plate and includes two major regions above the upper core alignment plate: the upper plenum and the upper head. The response time is short enough to track the level during small break LOCA events. The resolution is sufficient to show the initial level drop, the key locations near the hot leg elevation, and the lowest levels just above the alignment plate. This provides the operator with adequate indication to track the progression during Intervals 2 and 4 and to detect the consequences of mitigating actions or the functionability of automatic equipment.

E.31.2.3 Fuel Cladding Heatup

The overall intent of ICC detection is understood to be the detection of the potential for fission product release from the reactor fuel. The parameter which is most directly related to the potential for fission product release is the cladding temperature rather than the uncovering of the core by coolant.

Since clad temperature is not directly measured, a parameter to which cladding temperature may be related is measured. This parameter is the fluid temperature at the core exit. After the core becomes uncovered, the fluid leaving the core is superheated steam and the amount of superheat is related to the fuel length exposed and to the cladding temperature.

The temperature of the superheated steam leaving the core will be measured by the core exit thermocouples. The time behavior of the superheat temperature is, with the exception of an acceptably small time delay, similar to the time behavior of the cladding temperature. Thus, from the observation of the steam superheat, the behavior of the cladding temperature can be inferred. Observation of the cladding temperature trends during an accident is considered to be of more value to the operator than information on the absolute value of the cladding temperature.

The core exit steam temperature is measured with the thermocouples located at an elevation a few inches above the fuel alignment plate. Generic calculations of a similar installation for representative uncovering events show that the thermocouples respond sufficiently fast to the increasing steam temperature.

The required temperature range of the thermocouples extends from the lowest saturation temperature at which uncovering may occur up to the maximum core average exit temperature which occurs when the peak clad temperature reaches 2200°F. The required thermocouple range is, therefore, 200°F to about 1800°F, which is the approximate upper service temperature limit. Thermocouples are expected to function with reduced accuracy at even higher temperatures, so the range for processing the thermocouple output extends to about 2300°F.

E.31.3 SYSTEM DESIGN DESCRIPTION

The following sensors have been selected as the basic instruments to meet the functional requirements described in Subsection E.31.2.

a. the subcooled margin monitor (SMM) (Reference 1);

b. the heated junction thermocouple (HJTC) system (Reference 2); and

c. the core exit thermocouple system (CETS).

The conceptual design of each ICC instrument is described in this subsection which addresses:

a. sensor design and

b. signal processing and display design

Figure E.31-1 is the functional diagram for the ICC instrument systems. The HJTC and CET instrument systems consist of two safety-grade channels from sensors through signal processing equipment. The outputs of processing equipment systems feeding the primary display are isolated to separate safety-grade and non-safety-grade systems. Channelized safety-grade backup displays are included for the two instrument systems. The SMM instrument system consists of various sensor inputs to main control board indicators and to the process computer. The generation and display of the primary SMM is implemented by the process computer and is non-safety-grade. A backup SMM consists of main control board indication coupled with a procedure for operator determination of the subcooled margin. The following subsections present details of the design.

E.31.3.1 Sensor Design

E.31.3.1.1 Subcooled Margins Monitoring System

The subcooled margin monitor design configuration being implemented is detailed in Subsection E.31.3.2.4.1. This includes the representative core exit thermocouple (CET) temperature (Subsections E.31.3.1.3 and E.31.3.2.3). The sensor inputs to the SMM are:

	Input	Range
a.	Reactor Coolant Loop Pressure (Wide Range, RC Hot Legs A & C)	0-3000 psig
b.	Representative CET Temperature (from CET processing)	200-2300°F

E.31.3.1.2 Heated Junction Thermocouple (HJTC) System

The HJTC system measures reactor coolant liquid inventory with discrete HJTC sensors located at different levels within a separator tube ranging from the top of the core to the reactor vessel head. The basic principle of system operation is the detection of a temperature difference between adjacent heated and unheated thermocouples.

As pictured in Figure E.31-2, the HJTC sensor consists of a Chromel-Alumel thermocouple near a heater (or heated junction) and another Chromel-Alumel thermocouple positioned away from the heater (or unheated junction). In a fluid with relatively good heat transfer properties, the temperature difference between the adjacent thermocouples is very small. In a fluid with relatively poor heat transfer properties, the temperature difference between the thermocouples is large.

Two design features ensure proper operation under all thermal-hydraulic conditions. First, each HJTC is shielded to avoid overcooling due to direct water contact during two phase fluid conditions. The HJTC with the splash shield is referred to as the HJTC sensor (see Figure E.31-2). Second, a string of HJTC sensors is enclosed in a tube that separates the liquid and gas phases that surround it.

The separator tube acts like an internal sight glass and creates a single phase water level inside it when surrounded by a two-phase mixture. This effect is shown schematically in Figure E.31-3. The HJTC/LLP (liquid level probe) probe assembly with separator tube is shown in Figure E.31-4. The height of the water level inside the separator tube is equal to the height of the collapsed water level (water inventory) in the two-phase mixture surrounding the separator tube. Tests performed at C-E during the Phase II portion of the HJTC test program have demonstrated that a single phase water level is created inside the separator tube when immersed in a two-phase mixture and that the HJTC sensors, which are placed inside the separator tube, measure that level.

The reactor vessel configuration consists of two major regions above the upper core alignment plate: the upper plenum and the upper head (see Figure E.31-5). The upper plenum is the region between the fuel alignment plate and the upper internals support plate. The upper head is the region between the upper internals support plate and the top of the reactor vessel. These two regions have only limited hydraulic communication through leakage paths around the control rod drive shafts at the top of the RCCA guide thimbles. Thus, during loss-of-inventory or volume reduction accidents, the loss of water inventory proceeds essentially independently in the two regions. Based on analyses presented in Reference 8, the upper plenum is expected to drain faster than the upper head.

Two HJTC or HJTC/LLP probe assemblies are installed. These probe assemblies are similar to System 80 (C-E plants) probe assemblies. There are eight heated/unheated thermocouple pairs (sensors) in each probe assembly.

The HJTC probe assemblies are designed to measure the collapsed water level in the upper head independently from the collapsed water level in the upper plenum. Level monitoring is accomplished by use of a split probe assembly which creates two functionally separate probe sections, one in the upper head and the other in the upper plenum. A divider disk is located inside the separator tube of the probe at the upper internals support plate elevation to divide the two probe segments hydraulically. Flow holes at the top and bottom of each separator tube section allow the collapsed water level in each region to be formed and measure inside the two separator tube sections.

The axial locations of the eight HJTC sensors are shown in Figure E.31-6. The HJTC sensor arrangement has two sensors located in the upper head and six sensors in the upper plenum. Only two sensors are placed in the upper head because once the water level falls below the top of the RCCA guide thimbles, the upper head inventory no longer communicates with the upper plenum and reactor core. For the upper head to drain, water must exit through orifice holes located around the periphery of the upper internals support plate and drain into the downcomer annulus (reactor inlet region).

One of the upper head sensors is located as close as practical to the top of the reactor vessel. This provides the reactor operator with information about the loss of reactor coolant inventory from the upper head region as early as possible.

The second upper head sensor is located just above the upper internals support plate which separates the upper head region from the upper plenum region. This sensor indicates to the operator when the coolant inventory in the upper head has been fully depleted.

The remaining six sensors are utilized to provide more detail on coolant inventory in the upper plenum which is in direct communication with the reactor core. One sensor is located as close as practical to the upper internals support plate. This sensor indicates the formation of a void space as early as possible and that loss of coolant inventory in the upper plenum has begun.

A second sensor is located halfway between the upper internals support plate and the top of the hot leg. This location results in a minimum distance between sensors in the upper portion of the upper plenum of 39 inches, thus maximizing the continuity of indication of the reactor coolant inventory change in the region. This sensor indication on Byron Unit 1 Train A is defeated to support a procedurally controlled temporary configuration change under EC# 626390. Unit 1 Train B configuration is normal and is as stated in this description.

A third sensor is located at the top of the hot leg elevation. This location is chosen to provide information about the natural circulation capability of the plant. If the water inventory falls below this elevation, the loss of natural circulation becomes imminent.

A fourth sensor is located at the midpoint of the reactor vessel hot leg elevation and a fifth sensor is located at the bottom of the hot leg. These sensors provide the operator with more detailed information in this region. More detail at these elevations may be important to the operator because when the water level drops below the bottom of the hot leg, communication between the liquid inventory in the reactor coolant system piping and the reactor vessel ceases and the water inventory in the reactor vessel may drop more rapidly than before.

The sixth sensor is located as close as practical to the fuel alignment plate. This sensor tells the operator that loss of coolant inventory has proceeded to the point where core uncovering may become imminent. Subsequent ICC monitoring should be done using the core exit thermocouple indications.

For recovery from an inadequate core cooling incident, the sensors at the various locations provide confirmation that the operator actions or the automatic safety systems are working.

E.31.3.1.2.1 HJTC Performance with Reactor Coolant Pumps Off

A sample HJTC response for the Byron/Braidwood units following a representative 4-inch diameter (0.09 ft^2) pipe break is shown in Figure E.31-7. In the figure the collapsed liquid level in the upper head and upper plenum is shown. In addition, the time is shown when a particular HJTC sensor becomes uncovered. The collapsed level data used in the analysis presented in Figure E.31-7 was obtained from mixture level data from Reference 8, and void fraction information for the upper plenum for this case was provided by Westinghouse. The 0.09 ft^2 break transient presented shows that the liquid inventory in the upper plenum drains completely several minutes before the upper head drains. The liquid inventory in the upper head drains quickly (in less than 100 seconds) to the elevation of the top of the RCCA guide thimbles and then drains slowly through the orifice holes around the periphery of the upper internals support plate.

E.31.3.1.2.2 HJTC Performance with Reactor Coolant Pumps On

The overall effect of reactor coolant pump (RCP) operation is to circulate more vigorously and homogenize more uniformly (relative to no RCPs in operation) the two-phase mixture which is produced by flashing and/or boiling of the coolant. The capability of the separator tube to create a single-phase water level inside it is not affected by the RCP operation. Also, due to the very small flow in the upper head region of the reactor vessel (as will be discussed in more detail in the following paragraph), RCP operation will have no significant effect on the two-phase response in this region. Therefore, the HJTC response in the upper head region is expected to be the same with or without the RCPs in operation.

The expected normal operating, single-phase liquid, flow pattern with RCPs running in the reactor vessel is shown in Figure E.31-8. The bulk of the coolant flows down in the downcomer, flows through the core, enters the upper plenum through the fuel alignment plate and leaves the upper plenum through the hot leg pipes. A small bypass flow from the inlet annulus enters the upper head through the orifice holes around the periphery of the upper internals support plate. The bypass flow travels down through the RCCA guide thimbles and exits into the upper plenum above the fuel alignment plate through the open section of the RCCA guide thimbles. The split of the total coolant flow into a main portion flowing through the core and a small bypass flowing through the upper head results in a difference in pressure in the upper head (P_1) relative to the pressure in the upper plenum (P_2).

In addition, it is expected that there is an axial variation (beyond the one due to changing elevation heads) in the static pressure within the upper plenum region with the RCPs running. High velocity flow jets through the holes in the fuel alignment plate (FAP) result in a reduced static pressure, P_3, immediately downstream of the FAP, relative to the pressure, P_2, in the bulk of the upper plenum. Recovery of the static pressure occurs as the flow velocity decreases farther away from the FAP. Thus, there is a greater static pressure at the top of the upper plenum (just below the upper internals support plate) than at the bottom (just above the FAP).

In general, the pressure difference between upper head and upper plenum as well as the axial pressure variation in the upper plenum will decrease as the void fraction in the primary coolant system increases. This is due to degradation of the pump performance that occurs as the void fraction increases and results in a decrease in mass flow rate. Thus, as water inventory is lost during a small break LOCA with the RCPs running and system voiding increases, the effect of the RCPs on the indicated collapsed level is expected to decrease. The effect of the RCPs is also smaller if not all RCPs are running.

The HJTC indication in the upper plenum is expected to be affected by the axial pressure variation when the RCPs are running. The HJTC measurement in the upper plenum may be biased toward giving a lower indication of the collapsed level than actually exists when the pumps are running. This is because the static pressure at the top of the upper plenum is greater than the static pressure at the bottom of the upper plenum due to the pressure recovery downstream of the holes in the FAP. Thus, the greater static pressure at the top holes of the upper plenum separator tube depresses the collapsed water level inside the separator tube relative to the collapsed level in the upper plenum. This would result in a lower than actual indicated water level. Therefore, based on present information, it will be recommended that the reactor operator disregard the indicated level from the lower portion of the probe (upper plenum) when the RCPs are running.

Because of the split probe design, the HJTC measurement in the upper head is not affected by the operation of the RCPs. Sensors in this region will correctly indicate voiding in the upper head even with the reactor coolant pumps running. Display information is processed and displayed separately for these two regions.

E.31.3.1.3 Core Exit Thermocouple (CET) System

The design of the Byron/Braidwood Units 1 and 2 in-core instrumentation (ICI) system includes 65 Type K (Chromel-Alumel) thermocouples. The thermocouples are installed into guide tubes which penetrate the reactor vessel head and terminate at the exit flow end of selected fuel assemblies.

The available core exit thermocouples (CET) monitor the temperature of the reactor coolant as it exits the fuel assemblies. The core locations of the thermocouples are shown in Figure E.31-9.

UFSAR Subsection 4.4.6.1 and Section 7.7 describe the present design of the CET system. The basic design of the CET system will not change for the final ICC detection system, however, modifications will be performed to upgrade the CET to meet environmental qualification requirements. (See Subsection E.31.5.)

The CETs have a usable temperature range from 200° F to 2300°F (Reference 4) although accuracy is reduced at temperatures above 1800°F.

The signal processing and display for the CET portion of the ICC detection instrumentation is described in Subsection E.31.3.2.4.3.

E.31.3.2 <u>Signal Processing and Display Equipment Design</u>

The processing and display hardware depicted in Figure E.31-1 includes two subsystems of hardware: a qualified, safety-related subsystem of ICC instrumentation and an unqualified, non-safety subsystem of ICC instrumentation. The equipment subsystems process and display the ICC detection sensor inputs as well as sensor inputs to meet other NRC requirements. The back-up displays for reactor level and core exit temperature are safety-grade while the primary displays are non-safety-grade. Human factors engineering reviews are applied to both types of display. The design objective for the equipment is to address NUREG-0737, Item II.F.2.

E.31.3.2.1 <u>Backup Displays</u>

As depicted in Figure E.31-1, the backup displays for reactor vessel level and core exit temperature are driven by a two channel system. Both the HJTC and CET systems use microprocessor based designs for the signal processing function in conjunction with main control room digital indications. Each channel will accept and process ICC input signals, and provide outputs to the channel related indicator and the plant process computer. The backup displays are designed to give information to the operator in the remote chance that the primary display becomes inoperable and to confirm primary display information. Specific display descriptions for each ICC detection instrument are included in Subsection E.31.3.2.4.

E.31.3.2.2 <u>Primary Displays</u>

The primary displays for ICC detection are generated by the plant process computer using isolated outputs from the HJTC and CET processor cabinets and NSSS protection system cabinets (for reactor coolant loop pressures). The main control room primary displays for ICC detection are part of the safety parameter display system (SPDS). A complete description of the SPDS is included in Section E.17, in response to Item I.D.2 of NUREG-0737. A description of specific ICC displays is included in Subsection E.31.3.2.4.

E.31.3.2.3 Cabling Systems

The in-containment cabling system for the CETs and HJTCs uses environmentally qualified cabling and Class 1E connectors. Qualified containment penetrations route the CET and HJTC signals through the containment wall to the auxiliary building.

Separation of the two CET/HJTC channels is initiated below the missile shield and maintained to the signal processing equipment in accordance with the requirements of Regulatory Guide 1.75. Subsection E.31.5 discusses the qualification testing of the cabling.

The SMM inputs are routed from the sensors to the processing equipment via existing safety-grade cabling, containment penetrations and signal isolation hardware.

E.31.3.2.4 Processing and Display Description

The following subsections describe the processing and display for each of the ICC detection instruments.

E.31.3.2.4.1 Subcooled Margin Monitor

The SMM functions performed by the process computer are as follows:

a. Calculate the subcooled margin.

   The saturation temperature is calculated from the reactor coolant loop pressure inputs (wide range). The saturation pressure is calculated from the average of the ten hottest core exit thermocouples. The temperature sub-cooled margin is the difference between the saturation temperature and the hottest temperature input noted above.

b. Process all outputs for display.

   The SMM routine processes the temperature and pressure inputs over the following ranges: CET temperatures from 200°F to 2300°F, reactor coolant loop pressure from 0 to 3000 psig. The saturation temperature is calculated by the process computer from a saturation curve.

The following information is presented on the primary display:

a. Temperature margin to saturation; and

b. Graphic display of pressure-temperature conditions.

Also available on supporting lower level graphics is a trend of temperature margin to saturation.

Additional information regarding the primary display, safety parameter display systems (SPDS) is included in Section E.17.

Backup displays are not provided for the SMM, however a procedure has been developed for operator use, utilizing the main control board displays for reactor coolant wide range pressure and the average of the ten hottest CETs to determine the subcooling margin.

E.31.3.2.4.2 Heated Junction Thermocouples - Reactor Level

The processing equipment for the HJTC performs the following functions:

a. Determine if liquid inventory exists at the HJTC positions.

   The heated and unheated thermocouples in the HJTC are connected in such a way that absolute and differential temperature signals are available. This is shown in Figure E.31-10. When water surrounds the thermocouples, their temperature and voltage output are approximately equal. $V_{(A-C)}$ on Figure E.31-10 is, therefore, approximately zero. In the absence of liquid, the thermocouple temperatures and output voltages become unequal, causing $V_{(A-C)}$ to rise. When $V_{(A-C)}$ of the individual HJTC rises above predetermined setpoint, liquid inventory does not exist at this HJTC position;

b. Process all inputs and calculated outputs for display;

c. Provide an alarm output to the plant annunciator system when the HJTC detects the absence of liquid level;

d. Provide control of heater power for proper HJTC output signal level. Figure E.31-11 shows a single channel design which includes the heater power controller; and

e. Provide an input to the process computer for percent liquid inventory level above the fuel alignment plate. This output is an isolated signal.

The following information is presented on the primary display:

a. Liquid level inventory above the fuel alignment plate (normal or low).

Also available on supporting lower level graphics are trends of liquid level inventory for reactor head and plenum.

Additional information regarding the primary display, safety parameter display system (SPDS) is included in Section E.17.

The following information is presented on the backup HJTC display:

a. Percent liquid inventory level above the fuel alignment plate for each reactor head and plenum area derived from the eight discrete HJTC positions;

b. Unheated junction temperature at eight positions; and

c. Heated junction temperature at eight positions.

E.31.3.2.4.3 Core Exit Thermocouple System

The processing equipment for the CETS will perform the following functions:

a. Process all core exit thermocouple inputs. Processing of up to 33 CET inputs will be performed by Channel A and up to 32 CET inputs by Channel B to the backup displays;

b. Calculate the average of the hottest reading CETs, for Train A and for Train B and provide outputs to the respective backup displays; and

c. Provide data link outputs to the process computer for all available thermocouple inputs. These outputs are isolated signals.

These functions are intended to meet the design requirements of NUREG-0737, II.F.2, Attachment 1.

The primary display provides information on core exit temperature (the average of the ten highest CET temperatures).

Also available on supporting lower level graphics are trends of core exit temperatures and a spatially oriented core map indicating the temperature at each of the CET locations.

Additional information regarding the primary display and safety parameter display system (SPDS) can be found in Section E.17.

The following information is available on the backup displays:

a. Selectable temperatures of up to 65 core exit thermocouples. Channel A display includes the available CET individual temperatures and the average of the ten hottest CETs; and

b. Channel B display includes the available CET individual temperatures and the average of the ten hottest CETs.

E.31.4 SYSTEM VERIFICATION TESTING

This section describes tests and operational experience with ICC instruments.

E.31.4.1 Pressure Sensors

The reactor coolant wide range pressure sensors are standard NSSS instruments which have well known responses. No special verification tests have been performed nor are planned for the future. These sensors provide basic pressure inputs which are considered adequate for use in the SMM and other additional display functions.

E.31.4.2 HJTC System Sensors and Processing

An extensive test program has been performed to demonstrate that the HJTC system will operate as intended. Three separate test phases were undertaken for the program, which are:

Phase I - Proof of Principle Testing;
Phase II - Design Development Testing; and
Phase III - Prototype Testing.

The Phase I testing consisted of a series of five tests performed at C-E and ORNL test facilities. These tests demonstrated the feasibility of using the HJTC as a level sensing device and provided information necessary to the development of a preliminary RVLMS probe assembly.

Phase II was a design verification test series for the probe assembly. The objective of these tests was to simulate the thermal-hydraulic conditions surrounding the HJTC probe assembly that might exist in a PWR and verify the HJTC probe assembly performance under these conditions.

The Phase II HJTC probe assembly consisted of three HJTC sensors with splash shields installed inside a 12-foot long separator tube. Single phase, two-phase and depressurization transient tests were run. The tests covered a pressure range from atmospheric to 1450 psig, with blowdown tests initiated at 1875 psig. The two-phase mixture void fraction varied from 0 to 0.52. The collapsed level change rate tested varied from 0.5 to 3.0 inches/second for drain tests and 0.2 to 1.0 inches/second for refill tests. For comparison, the representative liquid level plot for Byron/Braidwood presented in Figure E.31-7 shows a peak drain rate of about 1.3 inches/second for a 0.09 ft^2 break.

The results of the Phase II test series show that the separator tube is capable of creating a collapsed water level that can be detected by the HJTC sensor when the probe is immersed in a two-phase mixture. The separator tube produces a region of all liquid below a region of nearly dry steam. The HJTC sensor responds to the movement of this steam/water interface past the sensor elevation. Good agreement is obtained between the water level indicated by the HJTC sensors and the collapsed water level measured independently by a DP cell.

In conclusion, the Phase II tests demonstrate that the HJTC probe assembly functions correctly to measure the collapsed water level under thermal-hydraulic conditions which the probe might be exposed to in a PWR. These tests, therefore, verify the performance of the HJTC probe assembly as an instrument to measure the water inventory in the upper head or upper plenum of a reactor.

The Phase III test series was developed to verify the final design of the HJTC probe assembly and associated electronics. A complete prototype HJTC system was tested under normal and accident thermal-hydraulic conditions that the probe may be exposed to in a PWR. Since the thermal-hydraulic performance of the probe assembly was verified by the Phase II tests, these tests concentrated on the performance of the integrated HJTC system, consisting of the probe assembly, signal processor and sensor heater power control.

The Phase III HJTC probe assembly consisted of eight HJTC sensors inside a 12-foot long separator tube. Single phase, two-phase, blowdown, and repressurization transients were run. The tests covered a pressure range from about 50 to 2000 psig and two-phase void fractions from 0 to 0.62.

The probe assembly thermal-hydraulic performance, which had been verified in the Phase II tests, was reconfirmed in these tests. That is, the collapsed water level is formed and measured inside the separator tube while a two-phase mixture exists outside.

The Phase III tests also show that the signal processor generates an uncovered or covered signal when the sensor ΔT (which is the temperature difference between heated and unheated thermocouples) reaches a setpoint value. When this occurs for each sensor, the percent level display changes to show the new collapsed water level. The sensor heater power control system successfully limits the maximum temperature and ΔT by reducing the power supplied to the heaters. The heater power control system maximizes the power supplied to the sensor heaters (to minimize the response time) while preventing damage to the sensor due to high heated junction temperatures. Even at low pressure where the heater power is reduced, the HJTC system still provides a good indication of the collapsed water level.

The integrated HJTC system was tested in the Phase III tests under simulated PWR thermal-hydraulic conditions and performed very well. Based on the analyses presented in Reference 8, the representative thermal-hydraulic parameter ranges for Byron/Braidwood were adequately covered by the Phase III as well as by the previous Phase II tests. Thus, the test results are applicable to Byron/Braidwood. Furthermore, it can be concluded that the integrated HJTC system will indicate to the reactor operator the status and trend of the water inventory in the reactor vessel during an accident.

E.31.4.3 Core Exit Thermocouples (CETs)

No verification testing of the CETs is planned. A study at ORNL was performed to test the response of CETs under simulated accident conditions (Reference 4). This test showed that the instruments remained functional up to 2300°F. This test along with previous reactor operating experience verifies the response of CETs.

E.31.5 SYSTEM QUALIFICATION

The plant equipment qualification program includes the reactor vessel level and core exit thermocouple portions of the ICC detection system. The subcooled margin monitor is non-safety grade and therefore, is not included in the qualification program.

E.31.5.1 Core Exit Thermocouples (CETs)

The core exit thermocouples are qualified to the requirements of IEEE 323-1974 and 344-1975 and in accordance with the Westinghouse approach to qualification as outlined in WCAP-8587. The mineral-insulated cable, which connects the thermocouples to the integral reference junction (IRJ) assembly, and connectors are qualified to the requirements of IEEE 323-1974 and 344-1975, and NUREG-0588, Revision 1.

The qualification of other core exit thermocouple system components including the interconnecting cables from the IRJ assemblies and the cable splicing junction boxes to containment electrical penetrations, the electrical penetrations themselves and interconnecting cables from the containment electrical penetrations to the processor cabinets is described in Sections 3.10 and 3.11 for BOP equipment. The processor and display qualifications are discussed in the following subsection.

E.31.5.2 Reactor Vessel Level

The in-vessel heated junction thermocouple probes have been qualified generically by Combustion Engineering Inc. The NRC approved this qualification generically in NUREG/CR-2627. Environmental qualification of this system is site specific and is documented in the EQ binders. Plant specific seismic qualification of the probes to safe shutdown conditions verify that the plant seismic conditions do not exceed the stress criteria and thereby established seismic qualification.

The out-of-vessel instrumentation for reactor vessel level including the mineral insulated cable, connectors, processor cabinets and main control board displays (for reactor vessel level and core exit temperatures) are environmentally qualified in accordance with IEEE 323-1974. The equipment is also qualified to IEEE 344-1975. CEN-99(s), "Seismic Qualification of NSSS Supplied Instrumentation Equipment," Combustion Engineering, Inc. (August 1978) describes the methods used to meet the criteria of the standard for the heated junction thermocouple system. The processor cabinets and main control board displays are located in the mild zone and the qualification documents are prepared commensurate with the schedule for mild zone qualification completion.

The qualification of other reactor vessel level system components including the interconnecting cables and containment electrical penetrations is described in Sections 3.10 and 3.11.

E.31.5.3 Primary Display

Inputs from the ICC detection system to the computer which drives the primary display are isolated by isolation devices qualified to Class 1E criteria. Therefore, the primary display is not designed as a Class 1E system, but is designed for high reliability; thus, it is not qualified environmentally or seismically to Class 1E requirements nor does it meet the single failure criteria of NUREG-0737, Appendix B, Item 2. Postaccident maintenance accessibility is included in the design. The quality assurance provisions of Appendix B, Item 5 do not apply to the primary display.

Verification and validation of the SPDS software for the primary ICC display will be performed. Additional information regarding the SPDS can be found in Section E.17.

E.31.6 OPERATING INSTRUCTIONS

The Byron Station emergency operating procedures for the use of information from the ICC instrumentation system have been developed taking into account recommendations from the Westinghouse Owners Group Generic Procedures and have been approved for use. The Byron operator training program has been modified to include material associated with the use of the ICC instrumentation system.

E.31.7 REFERENCES

1. NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November, 1980.

2. CEN-117, "Inadequate Core Cooling - A Response to NRC IE Bulletin 79-06C, Item 5 for Combustion Engineering Nuclear Steam Supply Systems," Combustion Engineering, October, 1979.

3. CEN-125, "Input for Response to NRC Lessons Learned Requirements for Combustion Engineering Nuclear Steam Supply Systems," Combustion Engineering, December, 1979.

4. Anderson, R. L., Banda, L. A., and Cain, D. G., "Incore Thermocouple Performance Under Simulated Accident Conditions," IEEE Nuclear Science Symposium, Vol. 28, No. 1 Page 773, Figure 81.

5. CEN-185, "Documentation of Inadequate Core Cooling Instrumentation for Combustion Engineering Nuclear Steam Supply Systems," Combustion Engineering, September, 1981.

6. CEN-185, Supplement 1, "HJTC Phase 1 Test Report," Combustion Engineering, November, 1981.

7. CEN-185P, Supplement 2-P, "HJTC Phase 2 Test Report," Combustion Engineering, November, 1981.

8. WCAP 9601, Vol. III, Section 4.1, "Plots for Case B, Mixture Level In-Core and Mixture Level in Upper Head."

9. NRC Generic Letter No. 82-28, "Inadequate Core Cooling Instrumentation System," dated December 10, 1982.

TABLE E.31-1

ICC STATUS AS AVAILABLE TO THE OPERATOR FROM ICC INSTRUMENTATION TRENDING

I. APPROACHING AN ICC CONDITION

INTERVAL	SUBCOOLING MEASURED BY SMM	WATER INVENTORY MEASURED BY HJTC PROBE	COOLANT SUPERHEAT MEASURED BY CET
1	Decreasing	Constant	Constant
2	Constant	Decreasing	Constant
3	Constant	Constant	Increasing

II. RECEDING FROM AN ICC CONDITION

INTERVAL	SUBCOOLING MEASURED BY SMM	WATER INVENTORY MEASURED BY HJTC PROBE	COOLANT SUPERHEAT MEASURED BY CET
4a	Constant	Constant	Decreasing
4b	Constant	Increasing	Constant
4c	Increasing	Constant	Constant

E.32 EMERGENCY POWER FOR PRESSURIZER EQUIPMENT (II.G.1)

POSITION:

The motor-operated pressurizer relief isolation valves and the solenoid air-operated pressurizer power relief valves are qualified Class 1E devices per Reference 1 of UFSAR Subsection 3.11.11. Therefore, their motive and control power is supplied from qualified emergency buses at all times. In addition, the source supplying the relief isolation valves is different from the source supplying the power relief valves. A Category I air supply is provided for the PORVs. The pressurizer level indication on Byron/Braidwood is designed for postaccident monitoring, and therefore, the instrument channels are always powered from the vital instrument buses.

E.33 ASSURANCE OF PROPER ESF FUNCTIONING (II.K.1.5)

POSITION:

Valve positioning requirements, positive controls, and test and maintenance procedures associated with ESF systems have been reviewed. Motor-operated valves in safety systems are normally maintained in a configuration such as to require the least number of valve automatic movements upon system actuation. System initiation logic is such that valves automatically move to the required position when required. The position of vital manual ECCS valves is controlled by the use and documentation of locks on valve handwheels.

Byron/Braidwood Stations are equipped with an ESF status panel which continuously monitors the ESF systems for any deviation which would indicate the system is not available. Typical parameters monitored include:

a. valve position,

b. power available to motor-operated valves,

c. initiation logic power available,

d. power sources (including emergency diesels) available, and

e. breaker status.

Alarms are provided on a system level if the system is not in a ready mode.

Surveillance and testing procedures for ESF systems include checklists to ensure that individual systems return to ready status upon completion of testing.

When ESF equipment is removed from service for maintenance, the Equipment Outage Procedure requires documentation of removal and return to service. Functional tests of equipment returned to service are required by this procedure to ensure operability.

In accordance with general station practices, all ESF systems are verified to be lined-up in accordance with approved mechanical and electrical checklist prior to fuel load.

E.34 SAFETY-RELATED SYSTEM OPERABILITY STATUS ASSURANCE (II.K.1.10)

POSITION:

During the preoperational test program at Byron/Braidwood (B/B), the vital bus independence verification preoperational test (see UFSAR Table 14.2-14) was conducted to verify the independence and redundancy of each of the two ESF division power supplies. As outlined in Regulatory Guide 1.40, the test conducted for both energized and deenergized conditions of the division not under test.

The PWR Standardized Technical Specification and the B/B Technical Specification clearly state that performance of a surveillance requirement within the specified time interval shall constitute compliance with operability requirements for limiting conditions for operations (LCO) and associated Action Statements, unless otherwise required by the specification. This section of the Technical Specification eliminates the requirements of previous Technical Specifications to test redundant components when a component was determined to be inoperable.

The ESF systems at B/B as described in the various chapters of the UFSAR are designed to be electrically and mechanically independent and redundant where required. Based on the design redundancy of station systems and the above information, the B/B station procedures require operability testing of redundant components remaining in service as a part of a redundant system only when specifically required by the B/B Technical Specifications.

E.35 TRIP PRESSURIZER LOW-LEVEL BISTABLES (II.K.1.17)

POSITION:

Byron/Braidwood Stations do not utilize a low level coincident pressurizer trip. Therefore, this item is not applicable to Byron/Braidwood Stations.

E.43 THERMAL MECHANICAL REPORT - EFFECT OF HIGH-PRESSURE INJECTION ON VESSEL INTEGRITY FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENT WITH NO AUXILIARY FEEDWATER (II.K.2.13)

POSITION:

To completely address the NRC requirements for detailed analysis for the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater, a program was to be completed and documentation sent to the NRC by January 1, 1982. This program consisted of analysis for generic Westinghouse PWR plant groupings. This generic work was performed by the Westinghouse Owner's Group and WCAP 10019 was issued. This item is also referenced in Generic Letter 83-37.

Following completion of this generic program, additional plant specific analyses, if required, were to be provided. A schedule for the plant-specific analysis was determined based on the results of the generic analysis.

The plant-specific analysis was documented in Babcock and Wilcox Report No. 77-1159832-00, "Pressurized Thermal Shock Evaluation in Accordance with 10 CFR 50.61 for Reactor Vessels in Byron Units 1 and 2 and Braidwood Units 1 and 2," dated January 13, 1986. This was submitted to the NRC in a January 17, 1986 letter from G. L. Alexander to H. R. Denton. Subsection 5.3 of the UFSAR also reflects this analysis.

E.47 POTENTIAL FOR VOIDING IN THE REACTOR COOLANT SYSTEM DURING TRANSIENTS (II.K.2.17)

POSITION:

Westinghouse (in support of the Westinghouse Owners' Group) has performed a study which addresses the potential for void formation in Westinghouse designed nuclear steam supply systems during natural circulation cooldown/depressurization transients. This study has been submitted to the NRC by the Westinghouse Owners' Group (Reference 1) and is applicable to the Byron/Braidwood units.

In addition, the Westinghouse Owners' Group has developed a natural circulation cooldown guideline that takes the results of the study into account so as to preclude void formation in the upper head region during natural circulation cooldown/depressurization transients, and specifies those conditions under which upper head voiding may occur. These Westinghouse Owners' Group generic guidelines have been submitted to the NRC (Reference 2). Commonwealth Edison considered the generic guidance developed by the Westinghouse Owners' Group (augmented as appropriate with plant specific considerations) in the development of Byron/Braidwood plant specific operating procedures.

The potential for the formation of voids in the reactor coolant system (RCS) has been evaluated. Any noncondensibles such as hydrogen that are generated in the RCS may be removed by the reactor coolant system vent or the pressurizer vents. The reactor vessel head vent system meets the requirements of 10 CFR 50.44(c)(3)(iii) as described in E.19. However, the pressurizer vent system need not be designed to the requirements set forth in 10 CFR 50.44(c)(3)(iii) since noncondensible gas accumulation in the pressurizer will not cause a loss of the core cooling function. Two sets of events were examined in this regard. The first of these are the design basis events.

The quantity of hydrogen generated due to reaction of 1.5% zirconium cladding is approximately 5100 scf. Should this entire volume of gas enter the pressurizer (neglecting hydrogen loss through the break or entrapment in the reactor vessel head or steam generator tubes), pressurizer venting would not be required to maintain adequate core cooling.

In addition, ultraconservative analyses for the second set, beyond design basis events have shown that the Byron/Braidwood reactor coolant system with the elevated steam generators maintained at their safety valve setpoint have sufficient capacity to maintain the primary system in the safe shutdown condition with significant hydrogen (i.e., the amount corresponding to as much as 20% of the zircaloy clad oxidized by a zirconium water reaction) trapped in the primary system by using only the reflux mode of heat transfer to the steam generator secondary side. (This corresponds to about 400 pounds of hydrogen in the primary system.) More realistically, by using the steam generator atmospheric steam dump valves to depressurize the secondary side of the steam generators, the plant could easily be maintained in the safe shutdown condition with hydrogen corresponding to up to 50% of the cladding oxidized (i.e., about 1000 pounds of hydrogen) trapped in the primary system.

The general scenario considered in each case is one in which the ECCS is unavailable and core uncovery occurs for an extended time period. This allows core damage to occur and generates significant hydrogen through a zirconium water reaction. Some of this hydrogen would be bottled up in the primary system, and the remainder would leak out of the break or stuck open valve which initiated the event. The scenario further specifies that recovery has occurred, i.e., core cooling water is made available later in the event for injection into the primary system. Two criteria must be met for the system to be maintained in a safe condition. The first is that the makeup flow into the primary system must be greater than or equal to the rate of primary system inventory loss, and the second is that there must be a means available for decay heat removal.

The thermodynamic condition at which the primary system stabilizes is related to both these criteria. The means available for decay heat removal are energy removal by the break itself, or heat removal through the secondary side of the steam generator. The rate of these energy removal mechanisms is directly proportional to the pressure and temperature of the primary system. The primary system will stabilize at a condition where decay heat input is equal to the energy removal.

In general, the most limiting condition for recovery is one in which the break occurs by a valve sticking open, the ECCS is unavailable, at that time core damage and hydrogen generation results, ECCS recovery and core reflooding occurs, and the valve reseats. This provides for loss of the breakflow energy removal mechanism, reduction in the effectiveness of steam generator heat removal, and trapping of some hydrogen in the primary system. (Scenarios involving a physical break or a continuously stuck open valve are less severe since the "break" provides a hydrogen vent and a means of sweeping hydrogen and steam out of the primary system. For such cases, ECCS recovery is the only criteria for core cooling and the pressurizer vent system has no role to play.)

The condition of the primary system at this point in the transient is a recovered core (i.e., water covered) with hydrogen present in the system (thus decreasing the effective heat transfer by the steam generator). The primary system will attempt to stabilize at a pressure where decay heat can be removed. If this pressure is below the safety valve setpoint, no inventory loss will occur and no makeup flow is necessary. The mode for heat removal from the primary system would be a "reflux mode" of heat transfer in which boiling in the core generates steam which condenses on the steam generator tubes and the resulting condensate then flows back into the core. It is this situation for which the analyses was performed whose results were previously discussed. Clearly, in this case, the worst case assumption involves the postulation that all the hydrogen is located in the steam generators. Any hydrogen located in the reactor vessel could be removed by the reactor vessel head vent and any hydrogen located in the pressurizer would have no effect other than to enhance the heat removal capability of the steam generators.

These results were obtained under conservative analysis of the Byron/Braidwood Stations. In order to bound the problem, two possible configurations of the hydrogen in the primary system were considered. The first assumed that the hydrogen formed a bubble in the top of the steam generator U-tubes causing a flow blockage and a reduction in the available heat transfer area. The second configuration considered was that the hydrogen formed a homogeneous mixture with the steam in the steam generator tubes.

The first situation is illustrated in Figure E.47-1. Masses of hydrogen corresponding to various amounts of zirconium water reactions have been assumed to form a discrete bubble in the steam generator. These amounts of hydrogen obviously represent some fraction of the total hydrogen which could have been generated in some postulated severe accident. This fraction is probably far less than one, and in fact hydrogen masses corresponding to 50% zirconium water reaction may be actually physically unrealizable given the fact that as the accident occurs, steam and hydrogen relief will occur thus limiting the amount of hydrogen present in the primary system at the recovery time. Thus, the assumption that all of the hydrogen is present in the steam generator represents a significant conservatism. The bubble is at pressure equilibrium with the primary system, but in thermal equilibrium with the secondary side. It is also conservatively assumed that this bubble blocks the loop flow and thus the only mode of heat transfer is the reflux method illustrated in Figure E.47-1 and thus only half of the tube surface is available for condensation.

The method of analysis for the case of a separate hydrogen bubble forming in the tops of the steam generator U-tubes was as follows. First the volume of the steam generator taken up by the hydrogen bubble was determined based on the ideal gas law and the assumption that the hydrogen bubble temperature was the same as the secondary side saturation temperature. (This methodology is consistent with assuming no heat transfer from the bubble.) The required heat transfer coefficient was then determined based on the heat transfer area available in the steam generator, and a conservative decay heat at 1000 seconds. (Note that calculations indicate that for this limiting case using a decay heat value corresponding to 1000 seconds is very conservative.)

The analysis was performed for a range of primary system pressures from 2500 psia to 400 psia, for a range of percent cladding reaction, and for the case of a pressurized secondary side (steam generator at its safety valve setpoint) and a depressurized secondary side. The required heat transfer coefficient for the two cases, pressurized secondary side and depressurized secondary side, is plotted in Figures E.47-2 and E.47-3 respectively, as a function of primary system pressure at which the system could stabilize and the percent clad reacted. These graphs can be used to identify the pressure at which the primary system stabilizes following an accident which generates a given amount of hydrogen, once the expected heat transfer coefficient has been quantified. If this pressure is below the primary system's safety valve setpoint (i.e., about 2500 psia) the plant will not continue to lose inventory and the plant is in a safe stable condition.

Figure E.47-2 shows that if the secondary side is at its safety valve setpoint (i.e., about 1200 psia) for 5%, 10%, and 20% clad reaction, the primary system will stabilize at 2500 psia or less given that the heat transfer coefficient from the primary or secondary side is 28, 35, or 80 Btu/hr/ft^2/°F or greater respectively. Similarly, Figure E.47-3 shows that if the secondary side has been depressurized to about atmospheric conditions (i.e., by using the safety grade atmospheric dump system) then for zirconium water reaction of 5%, 10%, 20%, 30%, and 50%, the primary system will stabilize at 2500 psia or less given that the heat transfer coefficient from primary to secondary is 5.6, 6.0, 7.0, 8.0, and 12.0 Btu/hr/ft^2/°F or greater respectively. These required heat transfer coefficients are significantly below those expected for this situation.

The overall heat transfer coefficient is set by three separate heat transfer resistances. The boiling heat transfer coefficient on the secondary side, the conduction through the tube, and the condensation on the secondary side. The boiling and conduction coefficients can be calculated rather straightforwardly and yield values of approximately 2000 and 3000 Btu/hr/ft^2/°F, respectively. The final component, the condensation heat transfer coefficient inside the steam generator tubes, can be calculated by performing a force balance at the inlet of the steam generator tubes to determine the water film thickness. This water film thickness can then be used to calculate the film heat transfer coefficient by assuming that the heat transfer coefficient can be represented by the thermal conductivity of water divided by the film thickness. Using the film heat transfer coefficient, along with the steam generator tube conductivity and boiling heat transfer coefficient on the secondary side, an expected overall heat transfer coefficient was estimated. This expected heat transfer coefficient was then compared with the required heat transfer coefficients. This method is conservative since the force balance was performed at the steam generator tube inlet thus giving the thickest film layer and greatest resistance to heat transfer. The condensation coefficient is calculated to be approximately 420 Btu/hr/ft^2/°F and the overall heat transfer coefficient about 310 Btu/hr/ft^2/°F.

The results of this analysis indicate that 20% clad reaction (about 400 pounds hydrogen) could be tolerated assuming a pressurized secondary side (steam generator safety valve setpoint) and up to 50% clad reaction (about 1000 pounds hydrogen) could be tolerated assuming a depressurized secondary side. Final points worth noting are that the range of the required heat transfer (i.e., from 6 to 80 Btu/hr/ft^2/°F) are in the range of those expected for free and forced convection, and are significantly below those expected for condensation heat transfer. The calculation performed to quantify the condensation heat transfer coefficient was probably not even necessary since from a judgmental point of view it should be obvious that the expected heat transfer coefficient will be greater than that required. It should also be noted that the ability to maintain counter-current flow in the steam generator U-tubes was verified using the Wallis flooding criteria. The conclusion being that counter-current flow can be maintained for the primary system pressures of interest (i.e., 2500 to 400 psia), and the decay heat level used.

The second situation studied is illustrated in Figure E.47-4. All of the hydrogen which has been assumed to be generated, is residing as a homogeneous mixture in thermal equilibrium with the steam in the steam generator and occupying the total tube volume.

The method of analysis for the case of a homogeneous mixture of steam-hydrogen in the steam generator was to first determine the steam-hydrogen mixture temperature based on the ideal gas law and Dalton's law of partial pressures. Based on the mixture temperature, the secondary side saturation temperature, and decay heat at 1000 seconds, a required heat transfer coefficient was determined for a range of primary system pressures (i.e., 2500 to 400 psia) and for a pressurized and unpressurized primary side. The required heat transfer coefficient for the two cases, pressurized and depressurized secondary sides, is plotted in Figures E.47-5 and E.47-6, respectively, as a function of primary system pressure and percent clad reacted. From the plots, it can be seen that a smaller heat transfer coefficient is required to remove the decay heat than those calculated for the case of a hydrogen bubble in the tops of the steam generator U-tubes. This is due to the larger heat transfer area available (essentially the entire steam generator) used in this portion of the analysis. The required heat transfer coefficients determined were compared to an adjusted stagnant Tagami correlation and to work performed by Sparrow, Minkowycz, and Saddy (Reference 3) for heat transfer in the presence of noncondensibles. This comparison should be conservative since the empirical data from Tagami and the analytical work of Sparrow, Minkowycz, and Saddy were developed for low pressure steam-air mixture. Steam diffusion through a hydrogen boundary layer is expected to be faster than diffusion through an air boundary layer and increases in system pressure will also increase the rate of mass transfer. The results of this portion of the analysis support the conclusions of the first portion of the analysis.

It should be noted that one significant conservatism in this analysis is that no credit was taken for the safety grade reactor head vent which provides a noncondensible gas vent, and a means to remove decay heat. Another conservatism places all the hydrogen in the steam generator. This is equivalent to saying that the Byron/Braidwood design can accommodate far more than the 50% zirconium-water reaction without a loss of core cooling. Clearly, the reactor vessel head vent extends this margin and is the vent associated with the flow configuration of interest for Byron/Braidwood.

In summary for design basis events the PORVs are not required as noncondensible vents. For beyond design basis events the Byron/Braidwood plant specific primary system design provides a significant capability for core cooling even given significant quantities of noncondensible gases are trapped in the primary system. The PORVs would not be required for removing noncondensibles given a wide spectrum of beyond design basis events.

References

1. Letter OG-57, dated April 20, 1981, R. W. Jurgensen (Chairman, Westinghouse Owners' Group) to P. S. Check (NRC).

2. Letter OG-64, dated November 30, 1981, R. W. Jurgensen (Chairman, Westinghouse Owners' Group) to D. G. Eisenhut (NRC).

3. Sparrow, Minkowycz, and Saddy, International Journal of Heat and Mass Transfer, Volume 10, pp. 1829-1845.

E.48 BENCHMARK ANALYSIS OF SEQUENTIAL AUXILIARY FEEDWATER FLOW (II.K.2.19)

POSITION:

Subsequent to the issuance of NUREG-0737 and as documented in the reference below, the NRC has completed a generic review of this subject and concluded that the concerns expressed in Item II.K.2.19 are not applicable to NSSS with inverted U-tube steam generators such as those designed by Westinghouse. Therefore, this item is not applicable to Byron/Braidwood and no further action is necessary.

Letter dated June 26, 1981, S. A. Varga (NRC) to J. S. Abel (Commonwealth Edison Company).

E.49 INSTALLATION AND TESTING OF AUTOMATIC POWER OPERATED RELIEF VALVE ISOLATION SYSTEM (II.K.3.1)

POSITION:

Westinghouse (in support of the Westinghouse Owners' Group) has evaluated the necessity of incorporating an automatic pressurizer PORV isolation system. This evaluation is documented in WCAP-9804, "Probabilistic Analysis and Operational Data in Response to NUREG-0737 Item II.K.3.2 for Westinghouse NSSS Plants," which concluded that such a system should not be required.

E.50 REPORT ON OVERALL SAFETY EFFECT OF POWER-OPERATED RELIEF VALVE ISOLATION SYSTEM (II.K.3.2)

POSITION:

The Westinghouse Owners' Group letter OG-52 dated March 13, 1981 (R. Jurgensen letter to J. R. Miller) was submitted recommending against the installation of automatic PORV isolation (Item II.K.3.1). Commonwealth Edison concurs with that recommendation. Therefore, in our judgment, no further action is required at this time.

E.51 REPORTING SV and RV FAILURES and CHALLENGES (II.K.3.3)

POSITION:

The Byron/Braidwood Technical Specifications require all pressurizer safety valves to be operable during modes 1, 2, and 3. Therefore, failure of any safety valves to close would be reported to the NRC through the normal licensee event reporting process.

E.52 AUTOMATIC TRIP OF REACTOR COOLANT PUMPS DURING LOSS-OF COOLANT ACCIDENT (II.K.3.5)

POSITION:

Response to this item was made by the Westinghouse Owners' Group in the June 15, 1981 letter from R. W. Jurgensen to P. S. Check. Additional submittals concerning this item were transmitted under letters dated August 22, 1985, and September 18, 1985, from G. Alexander to H. R. Denton.

E.54 PROPORTIONAL INTEGRAL DERIVATIVE CONTROLLER MODIFICATION (II.K.3.9)

POSITION:

Westinghouse recommended modification to the proportional integral derivative controller which was implemented at Byron and Braidwood Stations.

E.55 ANTICIPATORY TRIP MODIFICATION (II.K.3.10)

POSITION:

The setpoint of the anticipatory reactor trip on turbine trip is modified from 10% (P-7) power to 30% (P-8) power.

The impact of this setpoint modification on the results of the accident analysis presented in the UFSAR was evaluated. The modification was determined not to significantly impact any accident analysis results since this is an anticipatory trip and is not relied upon to mitigate the consequences of an accident. The anticipatory trip is active as required by Technical Specifications. There are no plans for any additional revisions to the setpoint.

E.56 JUSTIFICATION FOR USE OF CERTAIN PORVS (II.K.3.11)

POSITION:

None of the Commonwealth Edison plants use the type of PORVs that are the subject of this item.

E.57 CONFIRM EXISTENCE OF ANTICIPATORY REACTOR TRIP UPON TURBINE TRIP (II.K.3.12)

POSITION:

The reactor protection system contains an anticipatory reactor trip on turbine trip. This trip is described in UFSAR Subsection 7.2.1.1.2(f) and is required by the Technical Specifications. Therefore, no modification is proposed or required.

E.61 REPORT ON OUTAGES OF EMERGENCY CORE COOLING SYSTEMS LICENSEE REPORT AND PROPOSED TECHNICAL SPECIFICATION CHANGES (II.K.3.17)

POSITION:

The Staff reviewed the ECCS data provided by the Licensees and determined that no changes in the Technical Specifications are required.

E.66 EFFECT OF LOSS OF ALTERNATING-CURRENT POWER ON PUMP SEALS (II.K.3.25)

POSITION:

Byron/Braidwood Stations are equipped with emergency diesel generators which start on loss of offsite power. The service water and component cooling pumps automatically restart within 37 seconds following loss of offsite power. Therefore, the loss of offsite power has no effect on reactor coolant pump seals as emergency diesel generators rapidly restore ac power.

E.69 SMALL BREAK ANALYSIS METHODS (II.K.3.30)

POSITION:

The Westinghouse Owners' Group has authorized a program that will demonstrate the conservatism of the present WFLASH small break LOCA analyses. This program will demonstrate on a generic basis, that the new small break LOCA model, NOTRUMP, developed to respond to NUREG-0737, Section II.K.3.30 predicts lower calculated peak clad temperatures than WFLASH. Compliance with NUREG-0737, Section II.K.3.31 is required within one year of receiving the Safety Evaluation Report (SER) on the new small break model developed for NUREG-0737, Section II.K.3.30. The NRC approved the use of NOTRUMP on May 21, 1985 (reference: Letter from B.J. Youngblood (NRC) to D. L. Farrar (CECo), dated May 24, 1985). Therefore, Item II.K.3.30 is closed.

E.70 PLANT SPECIFIC CALCULATIONS TO SHOW COMPLIANCE WITH 10 CFR 50.46 (II.K.3.31)

POSITION:

Item II.K.3.30 was closed by the NRC on May 21, 1985 (Reference: Letter from B. J. Youngblood (NRC) to D. L. Farrar (CECo), dated May 24, 1985).

The response to close item II.K.3.31 was transmitted by a letter dated December 24, 1986, from S. C. Hunsader to H. R. Denton. This letter documented Commonwealth Edison's intent to satisfy this item by reference to Topical Report WCAP-11145. The staff's approval of this usage was transmitted in a letter dated February 2, 1987, from L. N. Olshan to D. L. Farrar.

E.75 UPGRADE EMERGENCY RESPONSE FACILITY (III.A.1.2)

POSITION:

The design of the Emergency Operating Facility and Technical Support Center is consistent with the guidelines of NUREG-0696. The instrumentation is being developed within the guidelines of Regulatory Guide 1.97. The information in this response includes the Exelon Nuclear Standardized Radiological Emergency Plan information. A description of communications between the Control Room, TSC, and EOF is included.

The site specific annex to the Exelon Nuclear Standardized Radiological Emergency Plan has been completed and individuals assigned to the specific positions of the emergency plan have been identified. Details of equipment and procedures pertaining to its operation have been submitted for review.

E.75.1 EOF AND TSC DESCRIPTION

The following facilities and systems are provided at Byron and Braidwood Stations to improve response to emergency situations.

The Technical Support Center (TSC) is located in the turbine building addition above the condensate polisher at elevation 435 feet. This is above the 100-year flood plain. The TSC has sufficient area for 25 people, 5 of which will be considered to represent the NRC and one will be considered to represent the State of Illinois, and is approximately 2-minute walking distance from the main control room. Also provided in the protected area of the TSC facility is an NRC office, records storage room, and equipment rooms for HVAC, electrical, and communications equipment. It is located within the site security boundary. Drawing A-831 shows the arrangement of the TSC.

The TSC is designed to be habitable to the same degree as the control room for postulated accident conditions, except that the equipment is not Seismic Category I qualified, redundant, instrumented as in the control room, or automatically activated. The TSC ventilation system is designed to limit the introduction of potential radioactive contaminants into the supply air by the use of high-efficiency particulate air (HEPA) and charcoal filters.

Radiation monitoring equipment is provided to continuously monitor radiation dose rates and airborne radioactivity concentrations inside the TSC while it is in use during an emergency.

TSC personnel will be protected from radiological hazards, including direct radiation and airborne radioactivity, from in-plant sources under accident conditions.

The TSC has dedicated communications with the Main Control Room (MCR) Operational Support Center (OSC), and Emergency Operations Facility (EOF).

The TSC has displays selected from Regulatory Guide 1.97, including meteorological variables of Regulatory Guide 1.23. The technical data displays assist in the detailed analysis and diagnosis of abnormal plant conditions and any significant release of radioactivity to the environment. The TSC has interactive color graphics displays and hard copy printouts. The accuracy of the data displayed is consistent with that needed to perform TSC functions.

Records that pertain to the as-built conditions and layout of structures, systems, and components are available to TSC personnel.

There is no impact on plant operations when the TSC is activated.

The TSC data is processed by the plant process computer with qualified isolators employed to separate 1E and non-1E circuits as required.

The data system serves the TSC and EOF using the same data base.

The Safety Parameter Display System (SPDS) parameters can be displayed in the TSC, EOF, and main control room. However, the iconic graphic displays are only available in the TSC and main control room.

Trend displays, time history, and dynamic displays are utilized to perform the task assigned to the TSC.

The EOF is located at the Exelon Nuclear Corporate Offices in Warrenville, Illinois. The EOF is more than 10 miles from Byron and Braidwood Stations.

The Operational Support Centers have been designed to satisfy the OSC requirements and will be fully described later. The Operational Support Centers are equipped with communications for the main control room and the TSC.

E.75.2 Power Supply Provisions

E.75.2.1 TSC

The power supply to the TSC is designed with sufficient alternate backup power sources to provide a very high degree of reliability to meet the operational unavailability requirements, while not degrading the safety-related power sources. The TSC power supply provides service to the HVAC and lighting at the TSC.

The power supply to the instrumentation data system equipment (existing process plant computer) is designed with the normal power supply being AC and the power backup being DC. This maintains continuity of TSC functions with a very high degree of reliability.

E.75.2.2 EOF

The design of the EOF power supply has been completed, and meets the high degree of reliability required according to NUREG-0696.

E.75.3 HVAC Provisions

The following descriptions apply to the TSC heating, ventilating, and air conditioning systems.

E.75.3.1 Emergency Operations Facility (EOF) HVAC System

The Emergency Operations Facility HVAC system serves the monitor room, records room, NRC office, meeting room, communications room, news media room, and various offices and toilet rooms.

E.75.3.1.1 Design Bases

The EOF HVAC system is designed to maintain comfort conditions, provide environmental conditions conducive to habitability in the EOF area to meet the requirements of NUREG-0696.

E.75.3.1.2 Safety Design Basis

The EOF HVAC system is non-safety-related; therefore, there is no safety design basis. Since the EOF is located greater than 10 miles from the Stations, HVAC system isolation with HEPA filtration is not required. The EOF is in leased office space and meets local codes for industrial office space.

E.75.3.1.3 (Deleted)

E.75.3.1.4 <u>Safety Evaluation</u>

The EOF HVAC system is non-safety-related.

E.75.3.1.5 <u>(Deleted)</u>

E.75.3.2 <u>Technical Support Center (TSC) HVAC System</u>

The Technical Support Center HVAC system is designed to serve the monitor room, computer room, electrical equipment room, kitchen, toilets, and other offices in the technical support center. The HVAC equipment room is shielded off from TSC boundary and is provided with a separate ventilation system.

E.75.3.2.1 Design Bases

The TSC HVAC system is a non-safety-related system and is designed to the habitability requirements in conformance to NUREG-0696.

E.75.3.2.2 Safety Design Basis

a. The system is designed for 10,000 cfm capacity with 9,100 recirculation air and 900 cfm makeup air.

b. The makeup air of 900 cfm is adequate to keep the TSC areas under 1/8 inch w.g. positive pressure with respect to surroundings.

c. On high radiation detection 900 cfm of makeup air and 1,100 cfm of recirculation air is routed through a 2,000 cfm filter unit with 4 inch bed tray type charcoal adsorbers. The filter unit is designed such that the postaccident integrated dose to personnel and TSC is less than the requirements specified in NRC General Design Criterion 19.

d. The system monitors products of combustion in makeup air intake. Ionization trips are alarmed in the local HVAC panel and the control room.

e. NUREG-0696 does not require system redundancy and hence, the system is not designed with redundancy. The system components are of comparable quality of those provided in the control room HVAC system.

f. The TSC is provided with permanently installed area radiation monitors in the TSC monitor room and in the health physics office, and a continuous air monitor on the TSC ventilation system.

E.75.3.2.2 Safety Design Basis

a. The system is designed for 10,000 cfm capacity with 9,100 recirculation air and 900 cfm makeup air.

b. The makeup air of 900 cfm is adequate to keep the TSC areas under 1/8 inch w.g. positive pressure with respect to surroundings.

c. On high radiation detection 900 cfm of makeup air and 1,100 cfm of recirculation air is routed through a 2,000 cfm filter unit with 4 inch bed tray type charcoal adsorbers. The filter unit is designed such that the postaccident integrated dose to personnel and TSC is less than the requirements specified in NRC General Design Criterion 19.

d. Upon notification of an offsite chlorine accident, the TSC HVAC system is manually isolated and is then operated with 100% recirculated air. Simultaneously, all exhaust fans are shut down.

e. The system monitors products of combustion in makeup air intake. Ionization trips are alarmed in the local HVAC panel and the control room.

f. NUREG-0696 does not require system redundancy and hence, the system is not designed with redundancy. The system components are of comparable quality of those provided in the control room HVAC system.

g. The TSC is provided with permanently installed area radiation monitors in the TSC monitor room and in the health physics office, and a continuous air monitor on the TSC ventilation system.

E.75.3.2.3 Power Generation Design Basis

The TSC HVAC system is designed to provide a controlled environment of 75° ± 2°F and a relative humidity of 45% ± 5% except in the electrical equipment room where the maximum design temperature is 104°F during summer. For winter, the system is designed to maintain 68° ± 2°F and a relative humidity of 35% ± 5%.

E.75.3.2.4 System Description

The TSC HVAC system is shown in Drawing M-94 Sheets 2 and 3. The capacities of principal components are listed in Table E.75-1.

a. The TSC HVAC system is comprised of an air handling unit with a direct expansion cooling system, a makeup filter unit, ductwork, and associated accessories. HVAC equipment room air is exhausted outdoors (1000 cfm) by a separate fan, which induces air supply through the intake louvers. During normal operation of the TSC HVAC system, the minimum outside air quantity is induced through an outside air intake to a mixed air plenum where it is mixed with return air from all spaces.

   The air handling unit fan discharges this mixed air through a DX cooling coil into main supply duct. The supply duct branches off to four main zones, each zone consisting of areas shown in Drawing M-94 Sheets 2 and 3. The branch duct to each zone is provided with electric reheat coils and steam humidifiers to further control the space environment to the design requirements. Air from all spaces, except from the HVAC equipment room, toilets, and the janitor room, is returned through ductwork to mixed air plenums. The toilets and janitor room exhaust air of 500 cfm is discharged by a separate fan. Makeup air compensates the loss of exhaust air and exfiltration air to keep the TSC under a minimum of 1/8 inch positive pressure.

b. An ionization detector is located in the intake duct.

   The tap for the portable radiation monitor is located in the supply duct.

c. Low leakage pneumatic operated butterfly isolation dampers are provided with fail positions such that the makeup air will be routed through makeup air filter unit on loss of air or high radiation detection.

d. The makeup air filter is designed for 2,000 cfm (900 cfm makeup air and 1,100 cfm return air). The unit consists of a centrifugal fan, prefilter, electric heating coil, upstream HEPA filter, two 2 inch bed tray type charcoal adsorbers, downstream HEPA filter in series. All the components and controls are designed to meet ANSI 509-80 requirements.

e. The controls and instrumentation are either pneumatic or electric. A local control panel for various functions to operate the HVAC system is provided and located within the TSC boundary. Important functions such as high radiation detection and ionization detection are annunciated in the HVAC panel. Common trouble alarm is provided in the main control room.

f. Water deluge valves for charcoal fire protection are provided. These are connected to station fire protection system.

E.75.3.2.5 Safety Evaluation

The TSC HVAC system is non-safety-related.

E.75.3.2.6 Inspection and Testing

All equipment is factory inspected and tested in accordance with the applicable equipment specification and codes. System ductwork and erection of equipment are inspected during various construction stages. Component demonstration tests are performed on all mechanical components and the system is balanced for the design airflows and system operating pressures. Controls, interlocks, and safety devices on each system are checked, adjusted and tested to ensure the proper sequence of operation. The equipment manufacturer's recommendations and station practices are considered in determining required maintenance.

E.75.4 EMERGENCY FACILITIES AND EQUIPMENT

E.75.4.1 Emergency Control Centers

E.75.4.1.1 Station Control Room

The nuclear station control room shall be the initial onsite center of emergency control. Control room personnel must evaluate and effect control over the initial aspects of an emergency and initiate activities necessary for coping with the initial phases of an emergency until such time that support centers can be activated. These activities shall include:

a. Continuous evaluation of the magnitude and potential consequences of an incident;

b. Initial corrective actions; and

c. Notification of appropriate individuals.

Support centers provided are an onsite Technical Support Center, an onsite Operational Support Center and an Emergency Operations Facility (EOF).

E.75.4.1.2 Technical Support Center (TSC)

Each Exelon Generation Company nuclear generating station has established a Technical Support Center (TSC) for use during emergency situations by plant management, technical, and engineering support personnel. When activated during an emergency, the TSC shall be staffed by sufficient personnel to:

a. support the control room command and control function;

b. assess the plant status and potential offsite impact; and

c. coordinate emergency response actions.

Staffing of the TSC shall be as directed by the Emergency Director. Reporting initially to the TSC for the Alert, Site Area and General Emergency shall be all Directors of the station group. The Shift Manager when acting as initial Station Director would not report to the onsite TSC. Other personnel may augment the TSC staff. The organization of the TSC staff is provided in the Exelon Nuclear Standardized Radiological Emergency Plan Figure B-1b.

E.75.4.1.3 Operational Support Center (OSC)

Each Exelon Generation Company nuclear generating station has established an operational support center (OSC). The OSC is the location to which operations support personnel will report during an emergency and from which they will be dispatched for assignments or duties in support of emergency operations. Personnel who may report to the OSC include:

a. operating personnel not assigned to the control room,

b. maintenance personnel,

c. radiation protection technicians, and

d. chemistry technicians.

A designated individual from the station staff is in charge of the activities of the OSC.

A limited inventory of supplies will be kept in the OSC. This inventory will include respirators, protective clothing, flashlights, and portable survey instruments.

E.75.4.1.4 Deleted

E.75.4.1.5 Emergency Operations Facility (EOF)

The Emergency Operations Facility (EOF) is the location that provides for the management of overall emergency response, the coordination of radiological and environmental assessments, the determination of recommended public protective actions, the management of recovery operations, and the coordination of emergency response activities with federal, state, and local agencies. The EOF functions under a Manager of Emergency Operations and is activated for all Alert, Site Area and General Emergency situations. Activation for other emergency conditions is optional. The EOF is located in Warrenville, Illinois which is greater than 10 miles from Byron and Braidwood Station. There will be four major groups of emergency response personnel functioning at the EOF. They are: (1) the technical support group; (2) the advisory support group; (3) protective measures group and (4) the emergency news center. These functions are outlined in detail in the Exelon Nuclear Standardized Radiological Emergency Plan.

Technical support group functions under the direction of the Technical Support Manager and serve at the EOF for direction of all recovery operations.

Advisory support group provide administrative services to the EOF and notification to responsible authorities.

Protective Measures Group is under the direction of the Protective Measures Director and function to evaluate emergency situations that affect the public.

Emergency news center operate from the EOF and Joint Information Center (JIC) which is under the direction of the Public Information Manager and functions as the single-point contact for disseminating information to the public.

E.75.4.1.6 Emergency Classification Levels

Table E.75-3 defines emergency classification levels. The emergency action levels associated with the classification levels are detailed in the Exelon Nuclear Standardized Radiological Emergency Plan station specific annexes.

E.75.4.2 Communication Systems

Extensive and reliable communication systems are installed at Exelon Generation Company generating stations, electric operations, corporate headquarters, and region load dispatching offices. These systems include the use of normal and special telephone lines, radio, and satellite radio/telephone capability.

For the purposes of emergency communications, the system is addressed in terms of functional areas as described in the following sections.

E.75.4.2.1 Nuclear Accident Reporting System (NARS)

The Nuclear Accident Reporting System is a dedicated telephone voice communication system that has been installed for the purpose of notifying State and local authorities of declared nuclear emergencies. This system links together the station control room, technical support center, electric operations, the EOF, and state and local authorities as appropriate.

The State of Illinois Emergency Management Agency, in cooperation with Exelon Generation Company, is responsible for the development and execution of all steps necessary to ensure continuous operation of the NARS.

E.75.4.2.2 Dedicated Emergency Response Facility Communications Systems

The Company has established several dedicated communication systems that ensure reliable and timely exchange of information necessary to provide effective command and control over any emergency response. These systems include a telephone link between the Control room, the TSC and the EOF at each nuclear station.

The communication systems are described in detail in the Exelon Nuclear Standardized Radiological Emergency Plan.

E.75.4.2.3 Environmental Assessment Communications

A satellite radio/telephone communication system has been installed to allow coordinated environmental monitoring and assessment during an emergency.

The system consists of the necessary hardware to allow communication between the Control Room, the TSC, the EOF, and mobile units in Exelon Generation Company vehicles.

E.75.4.2.4 NRC Communications

There exists a dedicated telephone, Emergency Notification System (ENS), between each nuclear station's Control Room and the NRC, with an extension of that line in the Technical Support Center. There also exists a separate dedicated telephone, health physics network (HPN), between the NRC and the TSC at each nuclear station. The actual configuration of these systems may vary from station to station. Installation and use of the NRC phones is under the direction of the NRC.

E.75.4.3 Assessment Facilities

E.75.4.3.1 Onsite Systems, Instrumentation, and Equipment

Each nuclear station is equipped with instrumentation for seismic monitoring, radiation monitoring, fire protection and meteorological monitoring. The actual instrumentation varies somewhat from site to site and thus will not be described in this generic plan. Descriptions of the above equipment will appear in each site specific annex.

With regard to the Company's meteorological monitoring program, there has been a quality assurance program since 1976. The program was adopted from 10 CFR 50, Appendix B. However, since the meteorological facilities are not composed of structures, systems, and components that prevent or mitigate the consequences of postulated accidents and are thus not "safety-related", not all aspects of 10 CFR 50, Appendix B apply. Those aspects of quality assurance germane to supplying good meteorological information for a nuclear power station were adopted into the meteorological quality assurance program.

E.75.4.3.2 Safety Parameter Display System (SPDS) and Point History

The safety parameter display system (SPDS) and point history provide a display of plant parameters from which the safety status of operation may be assessed in the Control Room, TSC, and the EOF for each nuclear station. The primary function of the SPDS and point history is to help operating personnel in the control room make quick assessments of plant safety status. Duplication of the SPDS and point history displays in the onsite TSC and the EOF promotes the exchange of information between these facilities and the control room and assist management in the decision making process.

E.75.4.3.3 Offsite Dose Calculation System (ODCS)

The offsite dose calculation system (ODCS) estimates the environmental impact of unplanned airborne releases of radioactive material from nuclear stations.

The objective of the Exelon Generation Company ODCS is to provide quick and reasonably accurate estimates of radiation dose equivalent to persons living offsite.

E.75.4.3.4 EOF and TSC Monitored Parameters

A list of parameters monitored at the emergency offsite facility and technical support center is indicated in Table E.75-2.

E.75.4.4 Protective Facilities and Equipment

Each nuclear station has chosen locations to serve as both onsite assembly areas and offsite evacuation assembly areas. The specific locations of these areas are shown in each site specific annex.

E.75.4.5 First Aid and Medical Facilities

Each nuclear station maintains onsite first aid supplies and equipment necessary for the treatment of contaminated or injured persons. Medical treatment given to injured persons is of a "first aid" nature. Injured persons may be transported to a local clinic or hospital, if required.

E.75.4.6 Damage Control Equipment and Supplies

The onsite storeroom of each nuclear station maintains a supply of parts and equipment for normal plant maintenance. These parts, supplies, and equipment are available for damage control use as necessary. When an emergency condition exists at one station, additional supplies can be obtained from other stations and from division resources upon request.

E.75.4.7 Facilities and Equipment for Offsite Monitoring

Exelon Generation Company has contracted with a company to conduct an extensive offsite environmental monitoring program to provide data on measurable levels of radiation and radioactive materials in the environs. The program includes: fixed continuous air samplers; routine sampling of river water; routine sampling of milk; routine sampling of fish; and a fixed environmental dosimetry monitoring network. The environmental dosimetry program consists of the following elements at each nuclear station:

a. A nearsite ring of dosimeters covering the 16 meteorological sectors.

b. A 16 sector ring of dosimeters placed in a zone about 5 miles from the plant.

c. Environmental dosimeters placed at each of the normal fixed air sampler locations (typically about 8-15 air samplers per nuclear station).

Each nuclear station maintains a supply of emergency equipment and supplies which may be used for environmental monitoring. The actual equipment may vary somewhat from site to site and thus the specific listing of equipment appears in Station Emergency Plan Implementing Procedures.

TABLE E.75-1

TECHNICAL SUPPORT CENTER HVAC SYSTEM EQUIPMENT PARAMETERS

NAME OF EQUIPMENT		NUMBER, TYPE, QUANTITY AND NOMINAL CAPACITY
A.	TSC HVAC Supply Air System	
	Type	Blow-through packaged
	Following are the components:	
1.	Supply Air Fan	0VV23C
	Type	Centrifugal
	Quantity	1
	Drive	Belt
	Capacity (cfm)	10,000
	External Static Pressure (in. of water)	4.0
2.	Supply Air Filters	0VV18F
	Type	High Efficiency
	Quantity	1
	Capacity (cfm)	10,000
	Pressure Drop:	
	Clean (in. of water)	0.3
	Dirty (in. of water)	1.2
	Efficiency (% by ASHRAE 52 Test Std.)	80
	Media	
3.	Cooling Coil	0VV18A
	Type	Direct Expansion
	Quantity	1
	Cooling Capacity (Btu/hr)	300,000
	Air Quantity (cfm)	10,000

TABLE E.75-1 (Cont'd)

NAME OF EQUIPMENT		NUMBER, TYPE, QUANTITY AND NOMINAL CAPACITY			
B.	Heating Coils	0VV20A	0VV21A	0VV22A	0VV23A
	Type	Elec	Elec	Elec	Elec
	Quantity	1	1	1	1
	Capacity (kW)	15	10	25	10
	Air Quantity (cfm)	2350	1085	5015	1150
C.	Humidifiers	0VV17M	0VV18M	0VV19M	0VV20M
	Type	Steam	Steam	Steam	Steam
	Quantity	1	1	1	1
	Capacity (lb/hr)	4	8	5	15
	Air Quantity (cfm)	1150	2350	1085	5015
D.	Humidifier Steam Generator	0VV18B			
	Type	Electric			
	Quantity	1			
	Capacity (lb/hr)	45			
	Operating Pressure (psig)	10			

TABLE E.75-1 (Cont'd)

NAME OF EQUIPMENT		NUMBER, TYPE, QUANTITY AND NOMINAL CAPACITY			
B.	Heating Coils	0VV20A	0VV21A	0VV22A	0VV23A
	Type	Elec	Elec	Elec	Elec
	Quantity	1	1	1	1
	Capacity (kW)	15	10	25	10
	Air Quantity (cfm)	2350	1250	4850	1150
C.	Humidifiers	0VV17M	0VV18M	0VV19M	0VV20M
	Type	Steam	Steam	Steam	Steam
	Quantity	1	1	1	1
	Capacity (lb/hr)	4	8	5	15
	Air Quantity (cfm)	1150	2350	1250	4850
D.	Humidifier Steam Generator	0VV18B			
	Type	Electric			
	Quantity	1			
	Capacity (lb/hr)	45			
	Operating Pressure (psig)	10			

TABLE E.75-1 (Cont'd)

NAME OF EQUIPMENT		NUMBER, TYPE, QUANTITY AND NOMINAL CAPACITY
E.	Toilet Room Exhaust Fan	0VV27C
	Type	Wall Exhaust Fan
	Quantity	1
	Drive	Direct
	Capacity (cfm)	500
	External Static Pressure (in. of water)	0.5
F.	HVAC Equipment Room Exhaust Fan	OVV28C
	Type	Wall Exhaust Fan
	Drive	Direct
	Quantity	1
	Capacity (cfm)	1000
	External Static Pressure (in. of water)	0.25
G.	TSC Emergency Makeup Air Filter Unit	0VV19S
	Type	Packaged
	Following are the components:	
	1. Makeup Air Supply Fan	0VV25C
	Type	Centrifugal
	Quantity	1
	Drive	Direct
	Capacity (cfm)	2000
	External Static Pressure (in. of water)	3.0
	2. Pre-Filters	0VV19F
	Type	High Efficiency
	Quantity	1
	Capacity (cfm)	2000
	Pressure Drop:	
	Clean (in. of water)	0.3
	Dirty (in. of water)	1.0
	Efficiency (% by ASHRAE 52 Test Std.)	80
	Media	Glass Fiber

TABLE E.75-1 (Cont'd)

NAME OF EQUIPMENT	NUMBER, TYPE, QUANTITY AND NOMINAL CAPACITY
3. Heating Coil	0VV19A
Type	Electric
Quantity	1
Capacity (kW)	9
Air Quantity (cfm)	2000
4. HEPA Filters	0VV20F, 0VV23F
Type	Nuclear Grade
Quantity	2
Capacity (cfm)	2000
Pressure Drop:	
Clean (in. of water)	1.0
Dirty (in. of water)	2.0
Efficiency (% minimum 0.3 micron and larger)	99.97
Media	Glass Fiber waterproof and fire retardant
5. Charcoal Adsorbers	0VV21F 0VV22F
Type	Tray
Quantity	2
Capacity (cfm)	2000
Pressure Drop (in. of water)	1.2
H. Condensing Unit	0VV30C
Type	Reciprocating
Quantity	1
Type of Refrigerant	R-22
Type of Compressor Drive	Direct
Type of Condensers	Air Cooled
Maximum Net Total Cooling (Tons)	25

TABLE E.75-1 (Cont'd)

NAME OF EQUIPMENT		NUMBER, TYPE, QUANTITY AND NOMINAL CAPACITY	
I.	Process cooling equipment for computer and communication room	0VV20S	0VV21S
	Type	Package	Package
	Quantity	1	1
	Type of refrigerant	R-410A	R410A
	Compressor Type	Digital Scroll	Digital Scroll
	Type of Condenser	Air cooled remote	Air cooled remote
	Maximum Net Total Cooling (Tons)	4.9	4.9
J.	Process cooling equipment for computer and communication room	0VV63S	
	Type	Package	
	Quantity	1	
	Type of Refrigerant	R-22	
	Compressor Type/Drive	Centrifugal/Direct	
	Type of Condenser	Air cooled remote	
	Maximum Net Total Cooling (Tons)	10.9	

TABLE E.75-1 (Cont'd)

NAME OF EQUIPMENT		NUMBER, TYPE, QUANTITY AND NOMINAL CAPACITY	
I.	Process cooling equipment for computer and communication room	0VV20S	0VV21S
	Type	Self-Contained Package	Self-Contained Package
	Quantity	1	1
	Type of refrigerant	R-22	R-22
	Type of Condenser	Air cooled remote	Air cooled remote
	Maximum Net Total Cooling (Tons)	5	5

TABLE E.75-2

EOF AND TSC MONITORED PARAMETERS

A. PLANT SYSTEM PARAMETERS

Core Exit Temperature

Reactor Coolant System

1) RCS hot leg temperature
2) RCS cold leg temperature
3) RCS pressure
4) Pressurizer level

Secondary System

1) Steam line pressure
2) Steam generator level - narrow range
3) Steam generator level - wide range
4) Auxiliary feedwater recirculation flow

ECCS

1) High head safety injection pump flow
2) Medium head safety injection pump flow
3) Low head safety injection pump flow

Containment Parameters

1) Containment pressure
2) Containment sump water level
3) Containment water level
4) Hydrogen concentration

B. RADIATION PARAMETERS

Steam Generator Blowdown

Condenser Air Ejector

Containment Radiation - High Range Area Monitor
Effluent Radioactivity

1) Noble gases
2) Halogens and particulates

TABLE E.75-2 (Cont'd)

TSC Data

1) TSC radiation level
2) TSC airborne radioactivity concentration

C. Meteorological Data

Temperature, differential temperature, dew point, wind speed, and wind direction parameters available from the meteorological tower.

TABLE E.75-3

EMERGENCY ACTION LEVELS

CONDITION	DESCRIPTION
Unusual Event	Events are in progress or have occurred that indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.
Alert	Events are in progress or have occurred that involve an actual or potential substantial degradation of the level of safety of the plant. Any radioactive material releases are expected to be limited to small fractions of the EPA protective action guideline exposure levels.
Site Area Emergency	Events are in progress or have occurred that involve actual or likely major failures of the plant functions needed for protection of the public. Any radioactive material releases are not expected to exceed EPA protective action guideline exposure levels except beyond the site boundary.
General Emergency	Events are in progress or have occurred that involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Releases can be reasonably expected to exceed EPA protective action guideline exposure levels for more than the immediate site area.

TABLE E.75-3 (Cont'd)

CONDITION	DESCRIPTION
Recovery	That period when the emergency phase is over and actions are being taken to return the situation to a non-emergency state. The plant is under control and no potential for further degradation of the plant or the environment is believed to exist.

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E.76 IMPROVING LICENSEE EMERGENCY PREPAREDNESS - LONG TERM (III.A.2)

POSITION:

Byron/Braidwood Stations radiological emergency response plans comply with 10 CFR 50 Appendix E. The Exelon Nuclear Standardized Radiological Emergency Plan (E-Plan) has been implemented and a site specific annex for each station has been approved and implemented. The E-Plan portion applicable to Byron/Braidwood Stations is consistent with NUREG-0654. Byron Station conducted its first emergency plan exercise in November 1983 and Braidwood Station conducted its first emergency plan exercise in November 1985. Exercises have been conducted on an annual basis since. The offsite dose calculation system has been developed and implemented, consistent with the guidelines of NUREG-0654, Appendix 2.

E.77 PRIMARY COOLANT SOURCES OUTSIDE CONTAINMENT (III.D.1.1)

POSITION:

Integrated Leak Testing

Per Technical Specification 5.5.2, integrated leak tests will be performed at least once per 18 months on each system or portions of systems, which could potentially contain highly radioactive liquids or gases. Station surveillances and procedures will be used to:

a. Monitor the leak testing of piping so that the appropriate lines are examined at the required intervals.

b. Direct leak test examinations such that systems are tested at approximate operating pressures or higher.

c. Align systems such that all piping tested is properly pressurized.

d. Identify lines which contain gases that require pressure decay, and/or metered makeup testing.

e. Quantify results of leakage examinations.

f. Initiate corrective action.

Leakage observed during the performance of inservice tests will be documented and a work request generated to repair leakage.

Systems to be Tested

The following piping systems outside containment would or could contain highly radioactive fluids and gases during or following a serious transient or accident. Portions of systems which will not be included in the integrated leak tests are identified.

a. Chemical and volume control (CV)

The chemical mixing tank and associated piping are not included because this portion of the system is protected by check valves and normally closed valves.

The boric acid addition portion of the system is not included because it is protected by check valves and normally closed valves which prevent back leakage into this portion of the system.

The resin fill tank and associated piping is not included because this portion of the system is normally valved out and would not be required to operate during an accident.

b. Containment spray (CS)

The spray additive tank and its associated piping are excluded because this portion of the system only supplies uncontaminated NaOH to the spray eductors. When all NaOH has been supplied to the eductors, check valves prevent leakage from the containment spray system into this portion of the system.

The 6-inch recirculation line which goes to the refueling water storage tank has not been included because it is normally isolated from the system and is only used to test the containment spray pumps in the recirculation mode.

c. Radioactive waste gas (GW)

The drain lines from the gas decay tanks have not been included because they are normally isolated from the system during plant operation. If any drain line were to be open, the inlet valve on that tank would also be closed, preventing any highly radioactive gases from entering this portion of the system.

The relief lines from the gas decay tanks have been excluded because these lines vent directly to the plant vent. The relief valves are set at 150 psi while the automatic system will divert gas to the standby tank at 110 psi. It is unlikely these lines would become highly radioactive.

d. Off-gas system (OG)

The calibration gas lines to the hydrogen analyzer are not included because these lines carry clean bottled gas to the analyzer for calibration purposes.

The portion of the off-gas system associated with the steam system has been excluded. The flow bypasses the off-gas filter unit, including charcoal adsorbers and HEPA filters, under all operating conditions. The condition has been analyzed and the results show that the release would be below the allowable limits. In addition, the secondary steam portion of the off-gas system must not exceed low pressure (1 psig) and low flow (20-40 scfm) characteristics in order for the steam jet air ejectors to function properly. Such flow characteristics indicate the potential for major leakage is small.

e. Residual heat removal (RH)

f. Safety injection (SI)

The refueling water storage tank and its associated piping is not included because it provides relatively clean 2300 ppm borated water to the safety injection pumps, containment spray pumps, and residual heat removal pumps during an accident. When the level in the tank reaches its low level, suction is transferred to the recirculation sump and the RWST becomes isolated from the system which prevents highly contaminated water from entering it.

The accumulator fill lines are excluded because they are used to fill the accumulators and would not be in use during a serious accident and could not become contaminated with highly radioactive water.

The leakoff lines from the recirculation line isolation valve cans have not been included because there is little potential for these lines to become highly radioactive.

g. Fuel pool cooling (FC)

h. Process sampling (PS)

The pressurizer steam and liquid sample lines, reactor coolant sample lines, and residual heat exchanger sample lines are included. Also, the chemical and volume control system demineralizer outlet sample line and the letdown heat exchangers sample line are included. These sample lines will be tested up to the liquid sample panel.

The remainder of the PS system will be excluded. This system is normally isolated and sampling is intermittent. In addition, all piping is 1 inch or less with most piping in the 3/8 inch to 3/4 inch range. Piping lengths are minimized for sampling considerations. The sample panels, where the greatest probability of leakage exists, are kept at a negative pressure. Also, the system is frequently operated by the Chemistry Department during normal plant operation. Considering all the above system characteristics, it would be unlikely that a loss of flow would go undetected.

Integrated Test Leak Acceptance Criteria

The initial system leak monitoring data will be taken after fuel load, during the startup testing. Leakage rates observed during this period provide a better baseline than those taken prior to fuel loading. Leak rates observed during preoperational testing are not necessarily representative of operating leak rates because of continuing adjustments to valve packing and seals, valve seat lapping, and the opening and closing of various mechanical joints. This will assure that initial leak monitoring will accurately indicate leak rates under actual operating conditions.

Other Leak Testing

In addition to this integrated leak test program, all Class 1, 2, and 3 systems will be leak tested at prescribed intervals, in accordance with the requirements of Section XI of the ASME Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Power Plant Components, as described in Byron/ Braidwood Station's "Inservice Inspection and Testing Program Plan." Therefore, Class 1, 2, and 3 portions of systems excluded from this leakage program, will be leak tested through the ISI Program.

The piping and components which make up the containment penetrations are tested at prescribed intervals as part of the 10 CFR 50, Appendix J leakage testing program for Type A and Type B testing, (Type C testing is in accordance with the Technical Specifications).

Prior to fuel load, all systems or portions of systems constructed in accordance with ASME Section III are hydrostatically tested to 125% of the system's design pressure. In the case of gaseous systems, a pneumatic type pressure decay test at 125% of system's design pressure is performed. All systems in this program are tested prior to initial plant startup via the pre-operational test program. During these tests, system walkdowns are conducted by the system test engineer and deficiencies are generated for leaking and defective components. In addition to the individual system tests, integrated type tests such as integrated hot functional and emergency core cooling full flow tests are conducted. During these integrated tests, additional system walkdowns are conducted for vibrational testing and inspection of piping thermal expansion. Deficiencies are generated during these walkdowns also.

E.78 IN-PLANT RADIATION MONITORING (IODINE) (III.D.3.3)

POSITION:

Instruments on site include air samplers designed for long term, continuous duty for in-plant or environmental outplant sampling where the collection of air samples is required. Calibration of these samplers is certified and traceable to the National Institute of Standards and Technology. There are an adequate number of air samplers available to support the emergency plan, normal operations, and outage work.

Byron/Braidwood Stations will have the capability to purge samples of any entrapped noble gases using nitrogen or air which is free from noble gases.

Byron/Braidwood Stations have acquired a transportable data acquisition and analysis system (DAAS) for analyzing samples that cannot be counted and analyzed in the normal station counting room because of background problems. The DAAS is a system provided in addition to counting equipment specified in Table 12.5-2.

An auxiliary counting room location has been identified in the turbine building. The location is the turbine building, 401 level, inside the secondary sample room.

It is expected that a sample can be obtained, purged, and analyzed for iodine content within a 2-hour time frame. Chemistry personnel have been trained on the procedures for iodine determination in areas under accident conditions.

E.79 CONTROL ROOM HABITABILITY REQUIREMENTS (III.D.3.4)

POSITION:

The control room HVAC system layout and functional design includes protection of the control room from radioactive and toxic gases. The system's operation is fully described in Section 6.4.

Subsection 2.2.3 reports the results of the evaluation of potential accidents involving hazardous materials (non-radioactive) including gaseous fuels, liquified gases, explosives, and toxic chemicals. The evaluation included an inventory of industries with hazardous materials within 10 miles of the station, including mobile transportation quantities.

E.80 TRAINING AND QUALIFICATION OF OPERATIONS PERSONNEL (I.A.2.2 from NUREG-0660)

POSITION:

A task analysis has been performed and is the basis for the following training programs:

1. Senior Reactor Operator/Shift Supervisor*
2. Reactor Operator
3. Licensed Operator Requalification
4. Non-Licensed Operator
5. Instrument and Control Technician
6. Electrical Maintenance Personnel
7. Mechanical Maintenance Personnel
8. Radiation Protection Technician
9. Chemistry Technician
10. Unit Supervisor/STA
11. Engineering Staff and Managers

Each of the programs is fully accredited by INPO.

For further information regarding these programs, refer to Section 13.2.

*Corresponds to the Shift Manager position at Exelon Generation Company Plants.

E.81 LONG TERM PLAN FOR UPGRADING OF PROCEDURES (I.C.9 FROM NUREG-0660)

POSITION:

Byron/Braidwood emergency operating procedures comply with the approved Westinghouse Owners' Group Guidelines (Item I.C.l). The emergency operating procedures are available for review by NRC per Item I.C.9 guidelines.

E.82 REFERENCES

1. Letter from B. J. Youngblood (NRC) to J.S. Abel (Commonwealth Edison) dated June 11, 1981, regarding NUREG-0737 item I.C.l.

2. Letter from D. G. Eisenhut (NRC) to R. W. Jurgensen (Commonwealth Edison) dated May 28, 1981.

3. R. L. Engel, J. Greenborg, and M. M. Hendrickson, "ISOSHLD -- A Computer Code for General-Purpose Isotope Shielding Analysis," BNWL-236, Pacific Northwest Laboratory, Richland, Washington, June 1966; Supplement 1, March 1967; Supplement 2, April 1969.

4. R. E. Malenfant, "QAD -- A Series of Point-Kernel General Purpose Shielding Programs," LA-3573, Los Alamos Scientific Laboratory, April 5, 1967.

5. R. E. Malenfant, "G^3 -- A General-Purpose Gamma-Ray Scattering Program," LA-5176, Los Alamos Scientific Laboratory, June 1973.

6. Letter from R. C. Youngdahl (Consumers Power) to H. R. Denton (NRC) dated July 1, 1981.

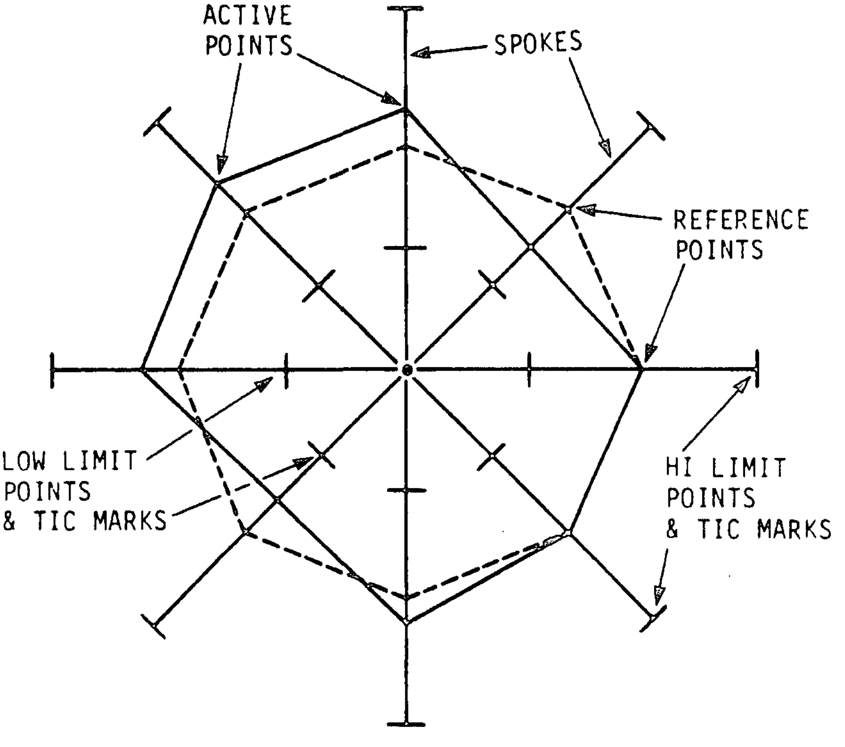
7. Letter from R. C. Youngdahl (Consumers Power) to D. G. Eisenhut (NRC) dated December 15, 1980.

8. Letter from D. F. Ross, Jr. (NRC) to all pending operating license applicants of nuclear steam supply systems designed by Westinghouse and Combustion Engineering, dated March 10, 1980.

9. "Byron Station Units 1 and 2, Braidwood Station Units 1 and 2, Environmental Qualification of Electrical Equipment, NRC Docket Nos. 50-454, 50-455, 50-456, 50-457," letter from T. R. Tramm, Commonwealth Edison Company to Harold R. Denton, Director, Office of Nuclear Reactor Regulation, USNRC, dated June 17, 1982, with attachments.

10. Letter from Stuart A. Richards (NRC) to Oliver D. Kingsley (NGG) dated February 9, 1999 approving a central EOF for all ComEd Nuclear Stations.

Figure E.7-1 has been deleted intentionally.



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.17-1

BASIC ICONIC DISPLAY

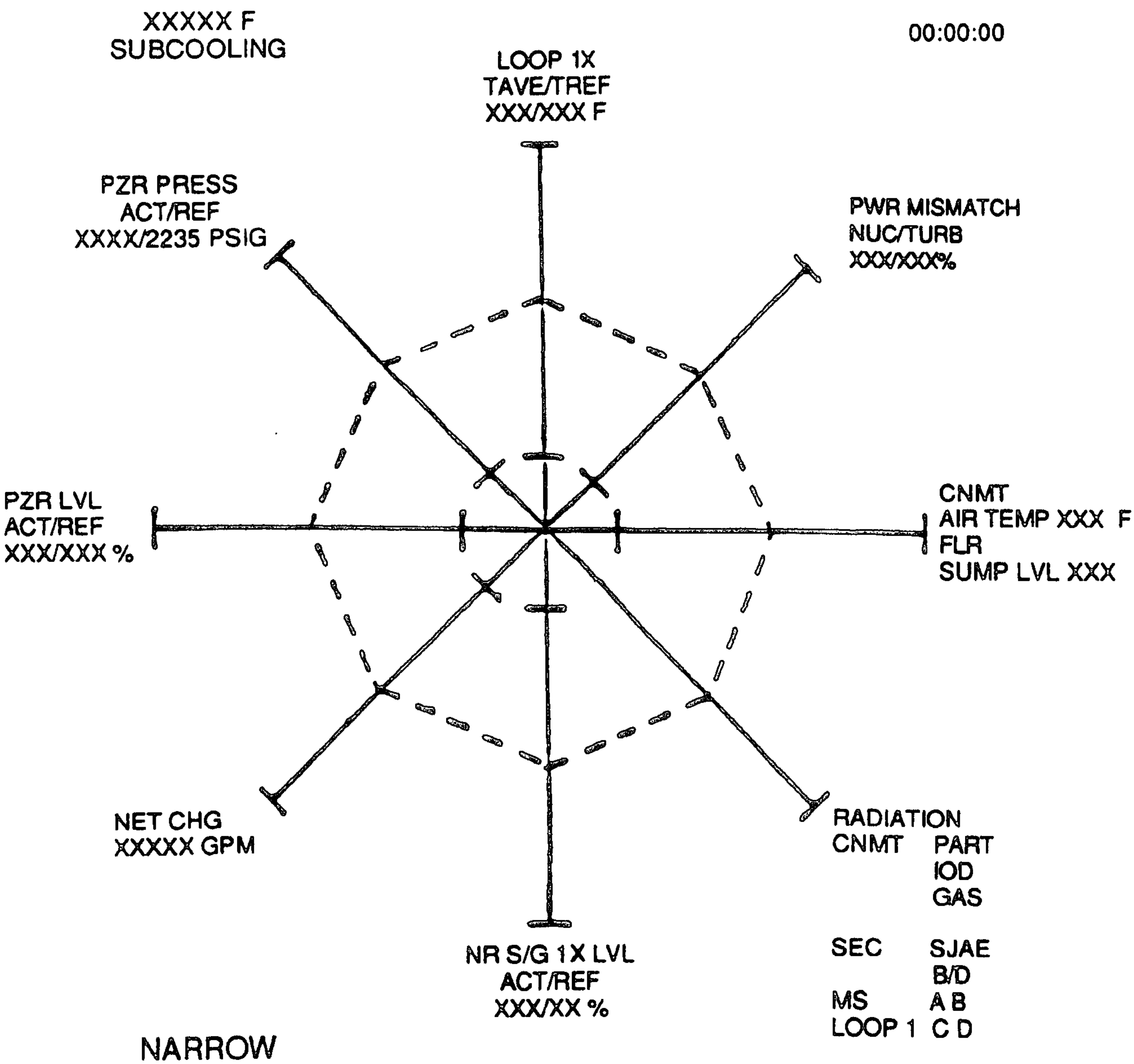


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FIGURE E.17-2

NARROW RANGE DISPLAY

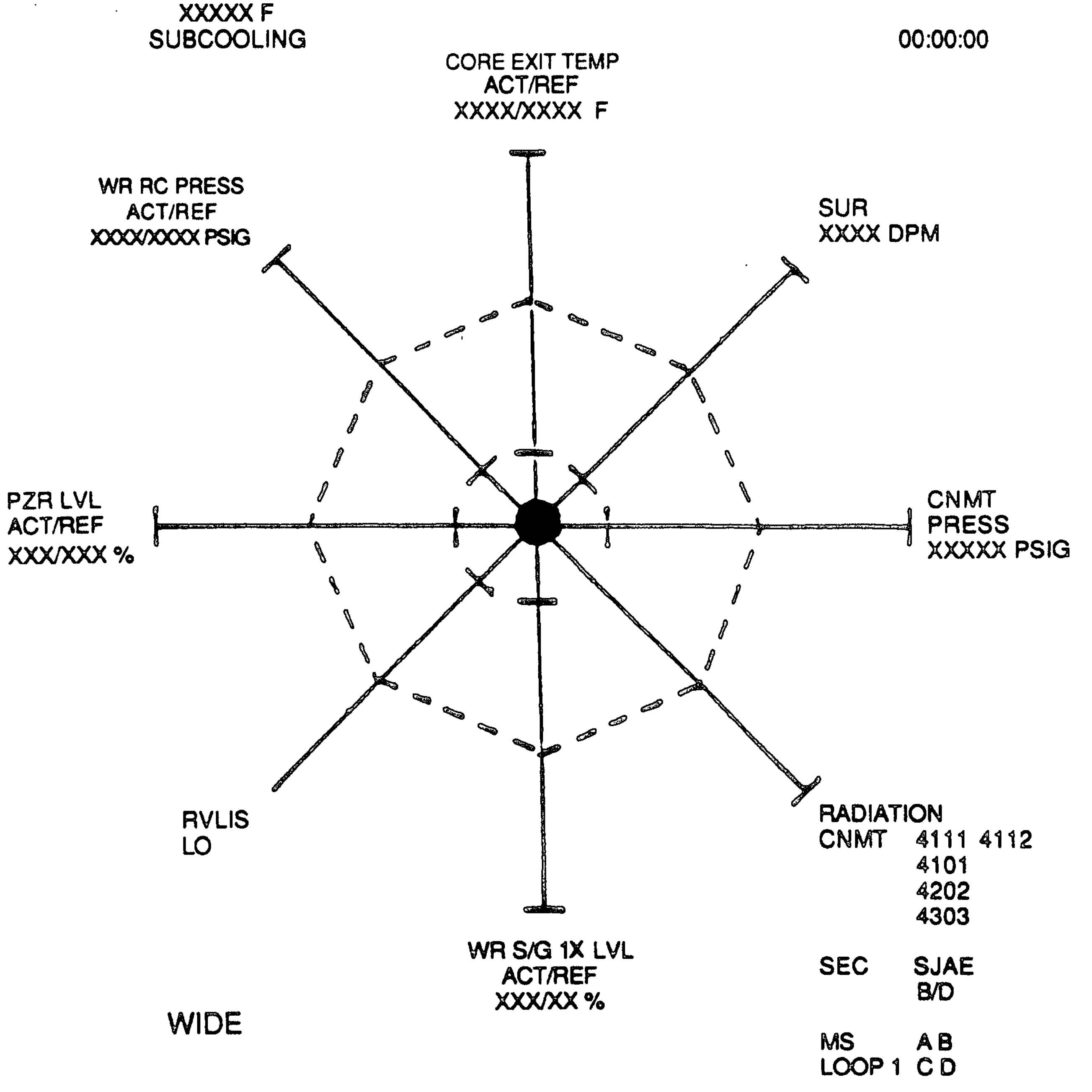

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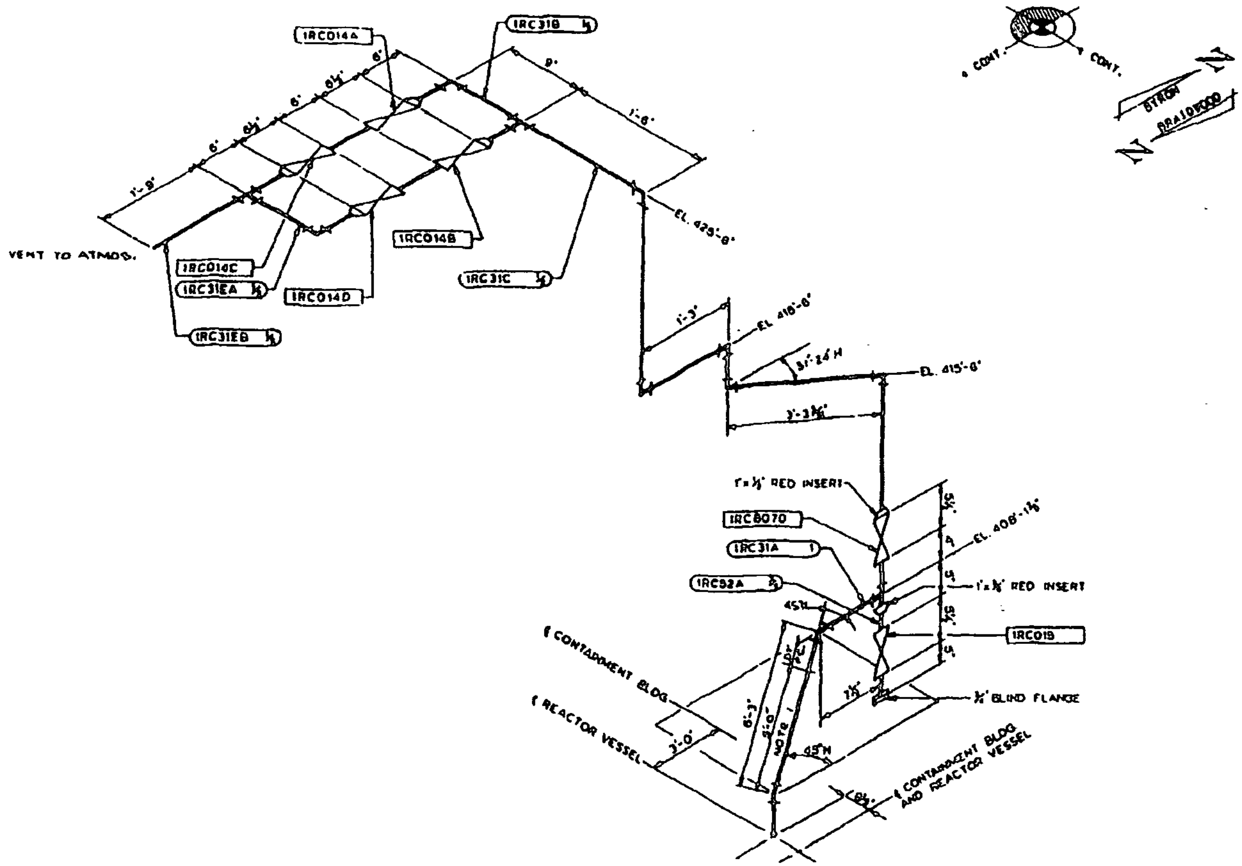
FIGURE E.17-3

WIDE RANGE DISPLAY



NOTE:
CONFIGURATION SHOWN IS APPROXIMATE
PLANT/UNIT SPECIFIC DIFFERENCES EXIST,
INCLUDING LINE SIZES AND NUMBERS.

BYRON/BRAIDWOOD STATION
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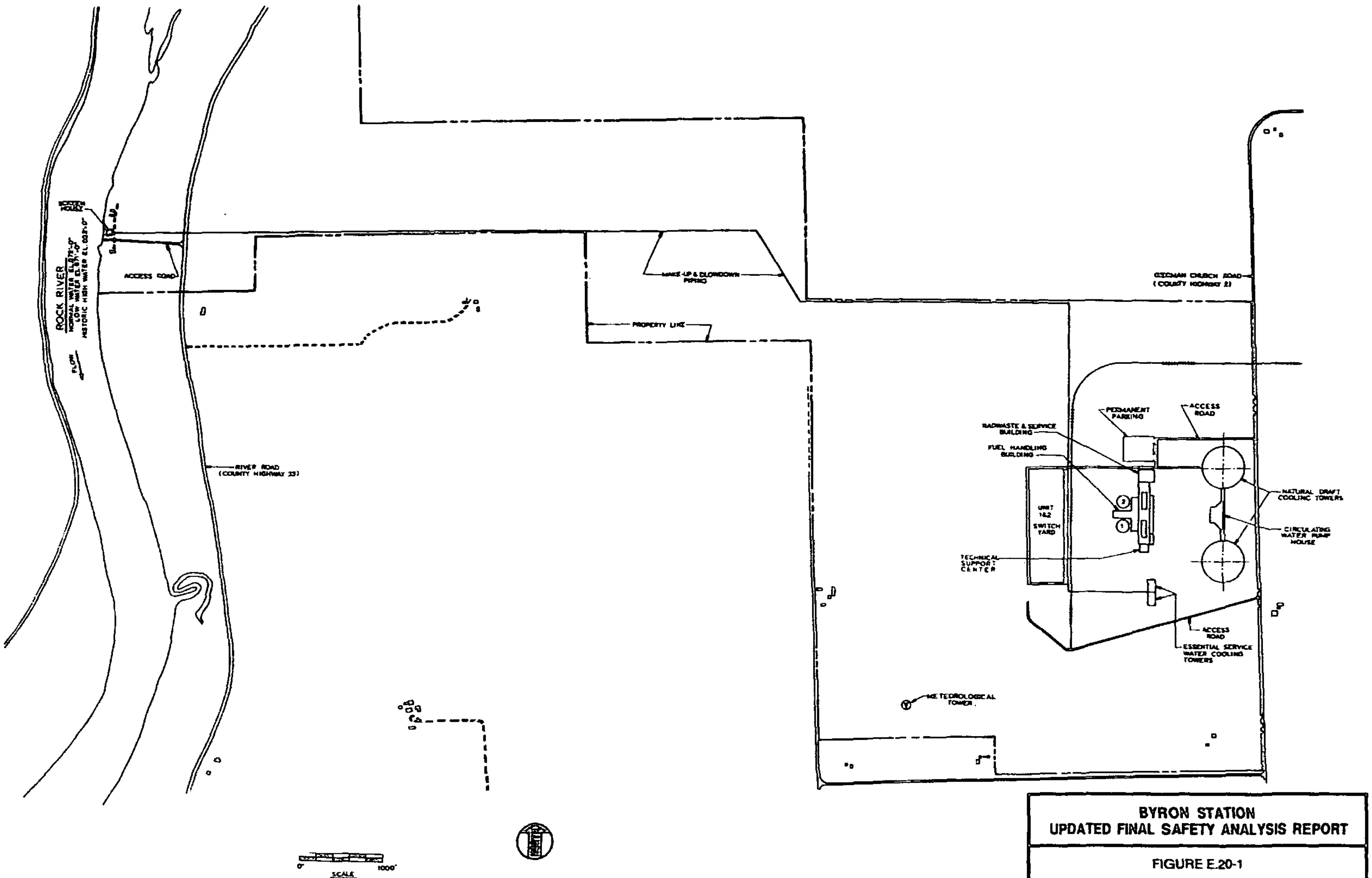
FIGURE E.19-1

REACTOR VESSEL, HEAD VENT SYSTEM
CONTAINMENT BUILDING



PROPERTY PLAT

BYRON STATION
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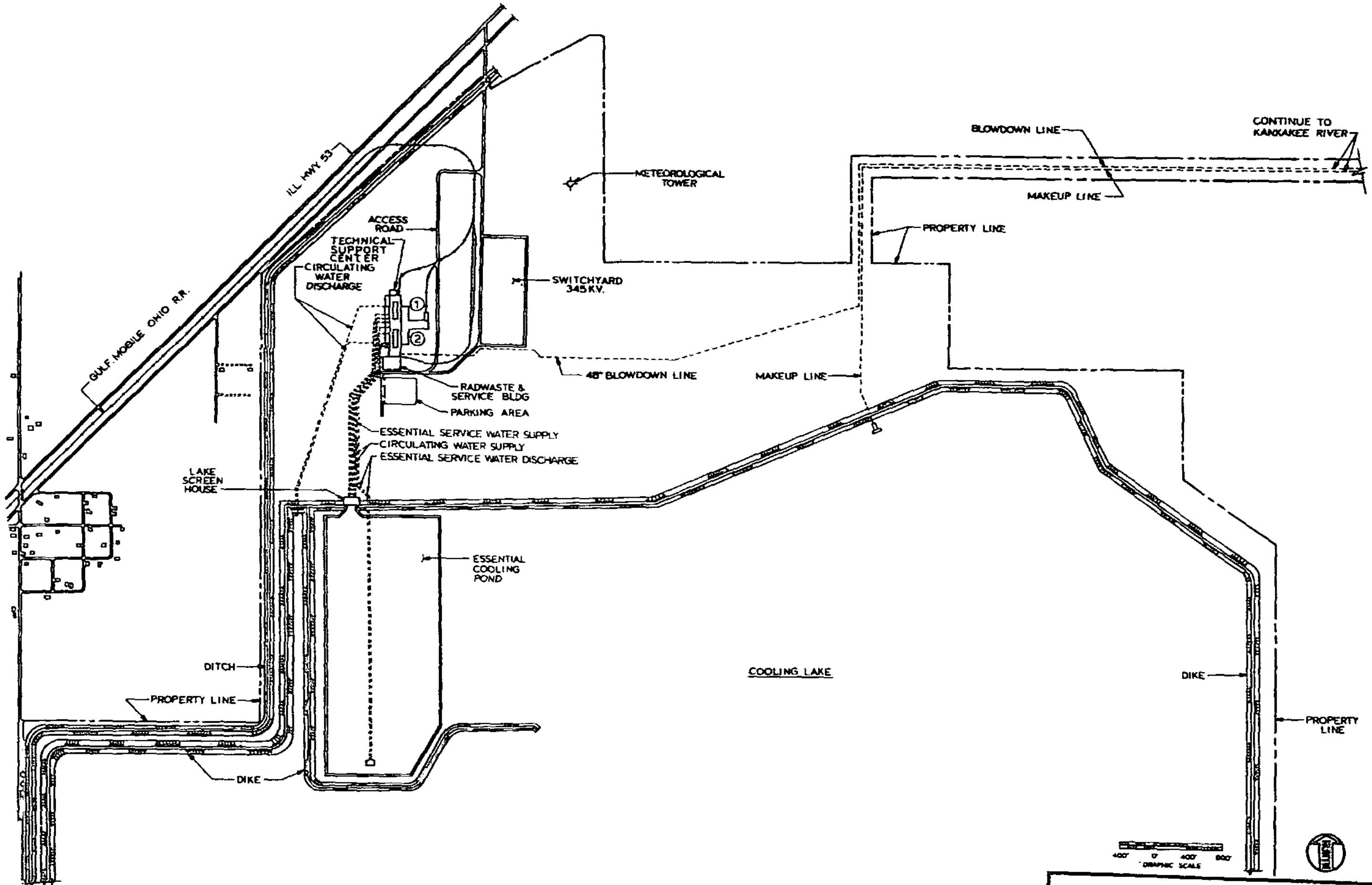
FIGURE E.20-1

POSTACCIDENT VITAL AREAS
(SHEET 1 OF 12)



BRAIDWOOD STATION
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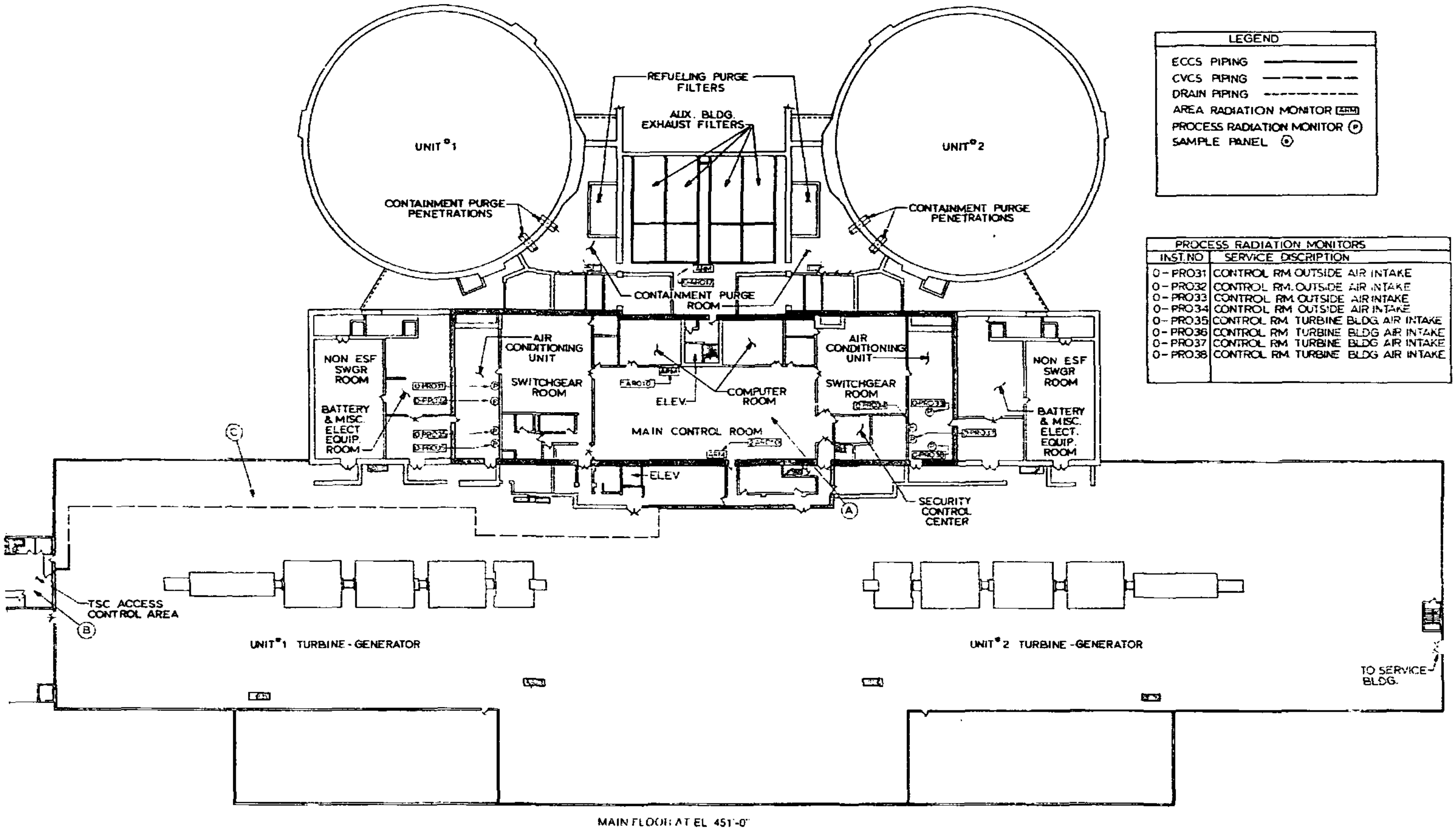
FIGURE E.20-1

POSTACCIDENT VITAL AREAS
(SHEET 2 OF 12)



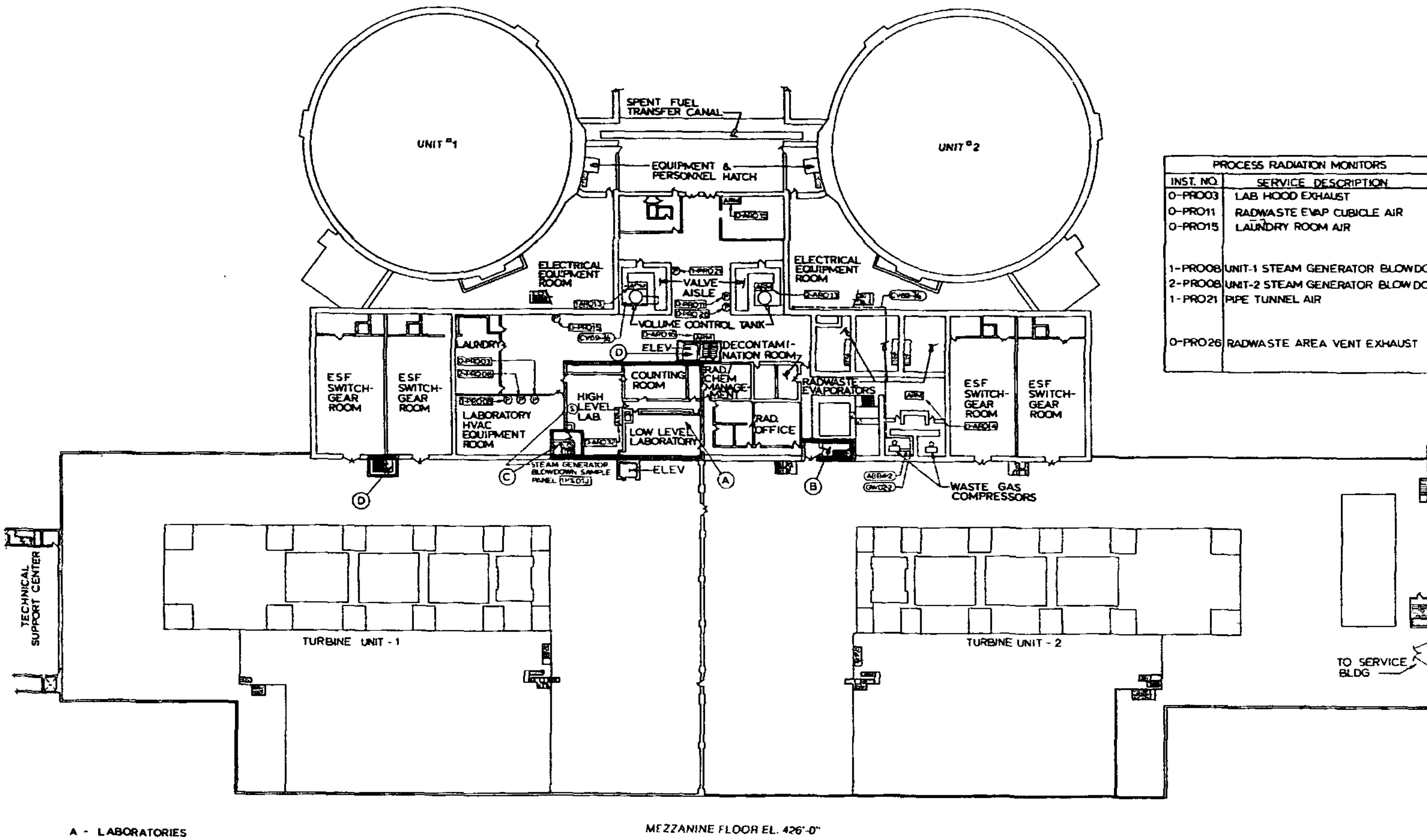
PROCESS RADIATION MONITORS	
INST. NO	SERVICE DISCRIPTION
0 - PRO31	CONTROL RM OUTSIDE AIR INTAKE
0 - PRO32	CONTROL RM. OUTSIDE AIR INTAKE
0 - PRO33	CONTROL RM OUTSIDE AIR INTAKE
0 - PRO34	CONTROL RM OUTSIDE AIR INTAKE
0 - PRO35	CONTROL RM TURBINE BLDG AIR INTAKE
0 - PRO36	CONTROL RM TURBINE BLDG AIR INTAKE
0 - PRO37	CONTROL RM TURBINE BLDG AIR INTAKE
0 - PRO38	CONTROL RM TURBINE BLDG AIR INTAKE

A - CONTROL-ROOM
B - TECHNICAL SUPPORT CENTER
C - PATHWAY TO TECHNICAL SUPPORT CENTER

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-1

POSTACCIDENT VITAL AREAS
(SHEET 3 OF 12)



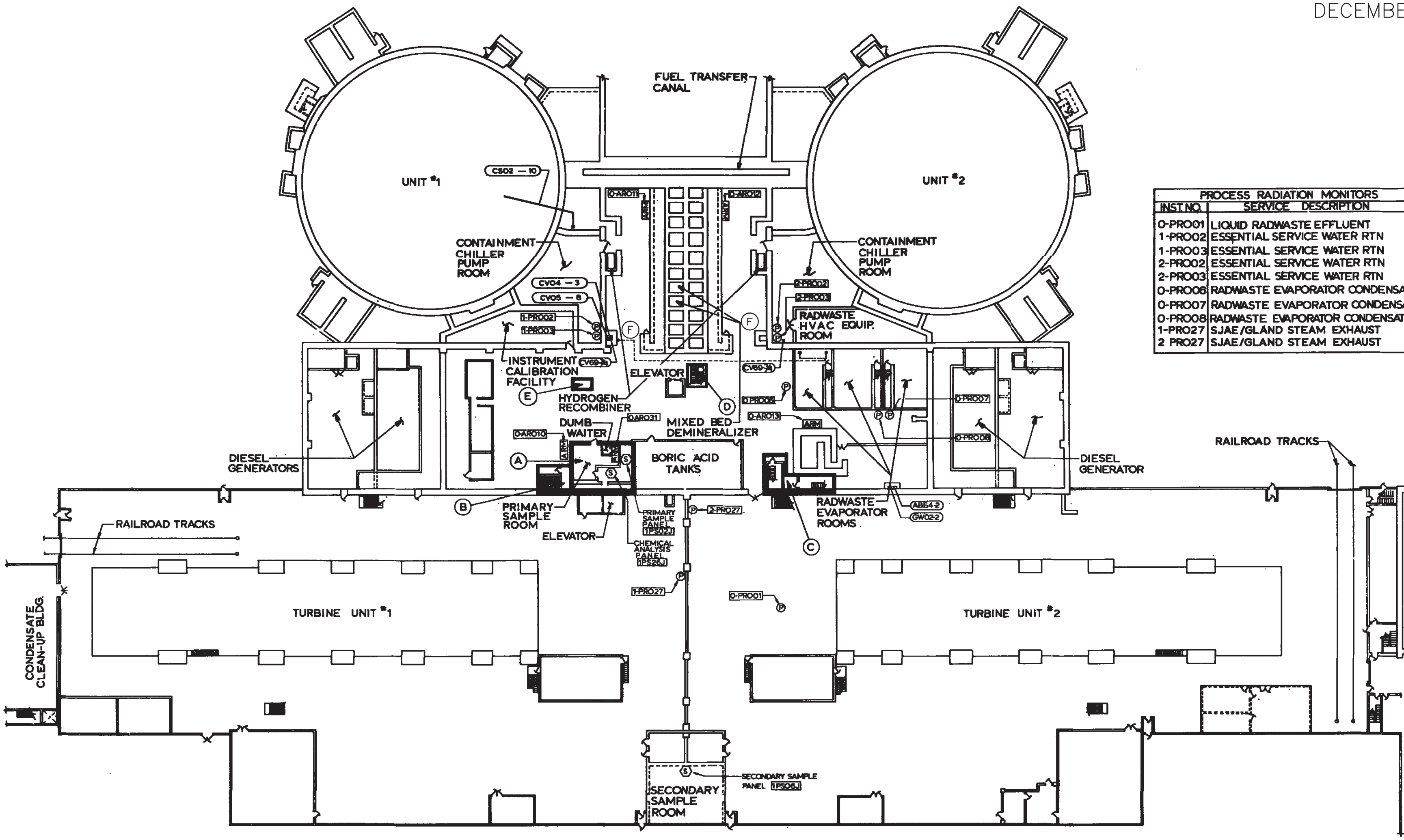
PROCESS RADIATION MONITORS	
INST. NO.	SERVICE DESCRIPTION
0-PR003	LAB HOOD EXHAUST
0-PR011	RADWASTE EVAP CUBICLE AIR
0-PR015	LAUNDRY ROOM AIR
1-PR008	UNIT-1 STEAM GENERATOR BLOWDOWN
2-PR008	UNIT-2 STEAM GENERATOR BLOWDOWN
1-PR021	PIPE TUNNEL AIR
0-PR026	RADWASTE AREA VENT EXHAUST

A - LABORATORIES
B - ACCESS TO REMOTE CONTROL PANEL
C - ACCESS TO PRIMARY SAMPLE ROOM
D - ALTERNATE ACCESS ROUTES

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-1

POSTACCIDENT VITAL AREAS
(SHEET 4 OF 12)



GRADE FLOOR AT EL. 401'-0"

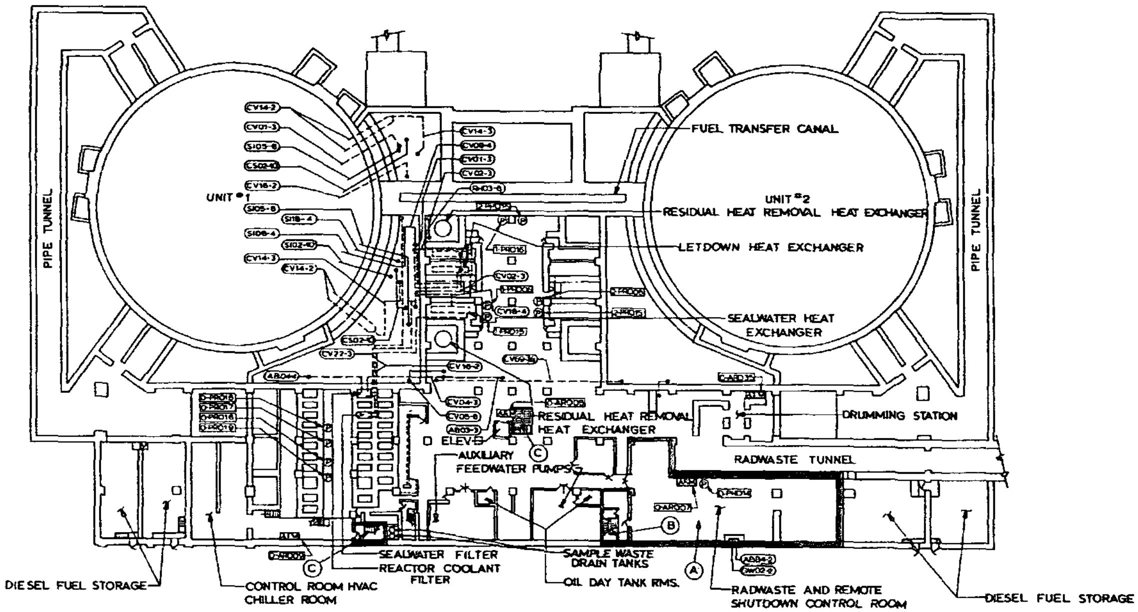
PROCESS RADIATION MONITORS	
INST NO.	SERVICE DESCRIPTION
0-PR001	LIQUID RADWASTE EFFLUENT
1-PR002	ESSENTIAL SERVICE WATER RTN
1-PR003	ESSENTIAL SERVICE WATER RTN
2-PR002	ESSENTIAL SERVICE WATER RTN
2-PR003	ESSENTIAL SERVICE WATER RTN
0-PR006	RADWASTE EVAPORATOR CONDENSATE RTN
0-PR007	RADWASTE EVAPORATOR CONDENSATE RTN
0-PR008	RADWASTE EVAPORATOR CONDENSATE RTN
1-PR027	SJAE/GLAND STEAM EXHAUST
2 PR027	SJAE/GLAND STEAM EXHAUST

A - PRIMARY SAMPLE ROOM
B - ACCESS TO PRIMARY SAMPLE ROOM
C - ACCESS TO REMOTE SHUTDOWN PANEL
D - ALTERNATE ACCESS ROUTE
E - HYDROGEN RECOMBINER CONTROL PANEL UNIT #1
F — HYDROGEN MONITOR PANEL

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-1

POSTACCIDENT VITAL AREAS
(SHEET 5 OF 12)



PROCESS RADIATION MONITORS	
INST NO.	SERVICE DESCRIPTION
1-PR006	GROSS FAILED FUEL
2-PR006	GROSS FAILED FUEL
0-PR014	DRUM STORAGE TUNNEL AIR
1-PR015	RHR HEAT EXCH 1A AIR
2-PR015	RHR HEAT EXCH 2A AIR
1-PR016	RHR HEAT EXCH 1B AIR
2-PR016	RHR HEAT EXCH 2B AIR
0-PR016	BLOWDOWN AFTER FILTER OUTLET
0-PR017	BLOWDOWN AFTER FILTER OUTLET
0-PR018	BLOWDOWN AFTER FILTER OUTLET
0-PR019	BLOWDOWN AFTER FILTER OUTLET

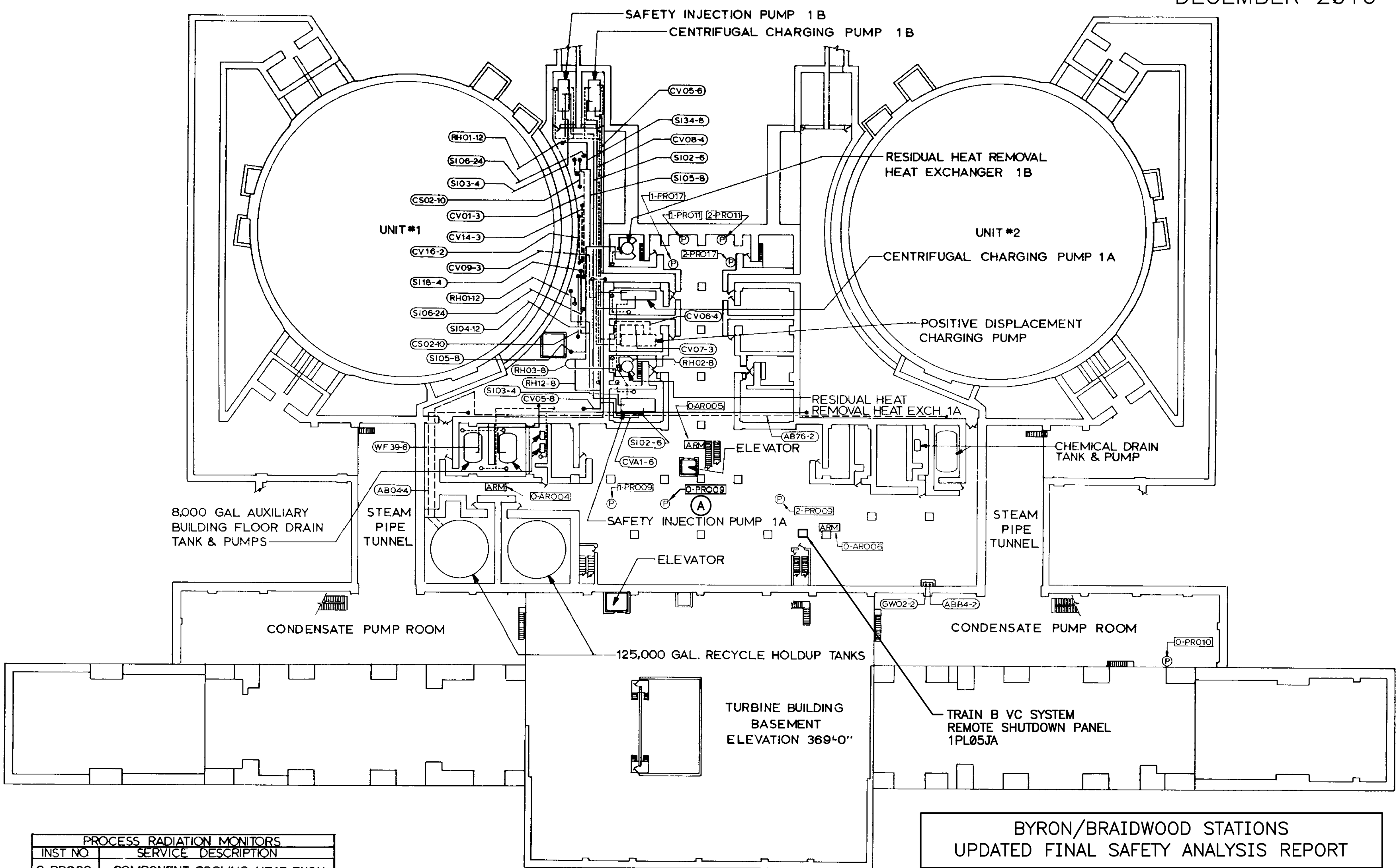
FLOOR PLAN EL. 383'-0"

A - REMOTE SHUTDOWN CONTROL ROOM
B - ACCESS TO REMOTE SHUTDOWN CONTROL ROOM
C - ALTERNATE ACCESS ROUTES

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-1

POSTACCIDENT VITAL AREAS
(SHEET 6 OF 12)



PROCESS RADIATION MONITORS	
INST NO	SERVICE DESCRIPTION
0-PR009	COMPONENT COOLING HEAT EXCH.
1-PR009	COMPONENT COOLING HEAT EXCH.
2-PR009	COMPONENT COOLING HEAT EXCH.
0-PR010	STATION BLOWDOWN
1-PR011	UNIT 1 CONTAINMENT ATMOSPHERE
2-PR011	UNIT 2 CONTAINMENT ATMOSPHERE
1-PR017	CHARGING PUMP CUBICLE AIR
2-PR017	CHARGING PUMP CUBICLE AIR

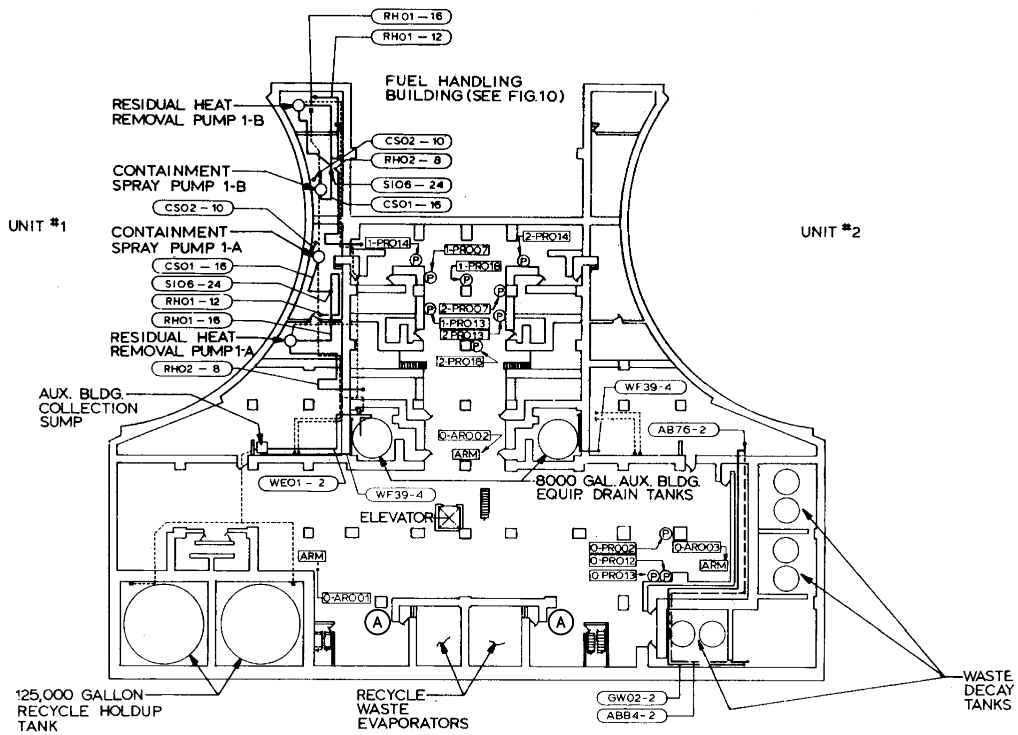
A – CCW SYSTEM

BASEMENT FLOOR EL. 364'-0"

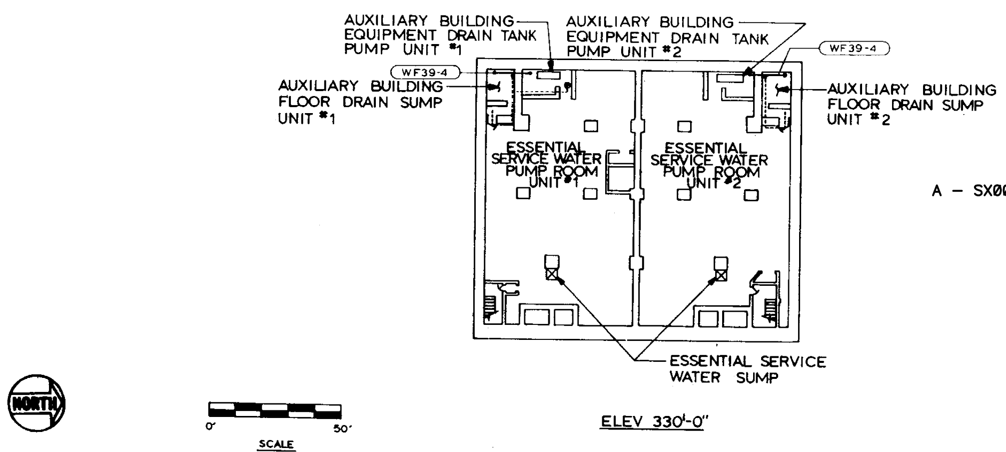
BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20–1

POST ACCIDENT VITAL AREAS
(SHEET 7 OF 12)



FLOOR PLAN EL. 346'-0"



ELEV 330'-0"

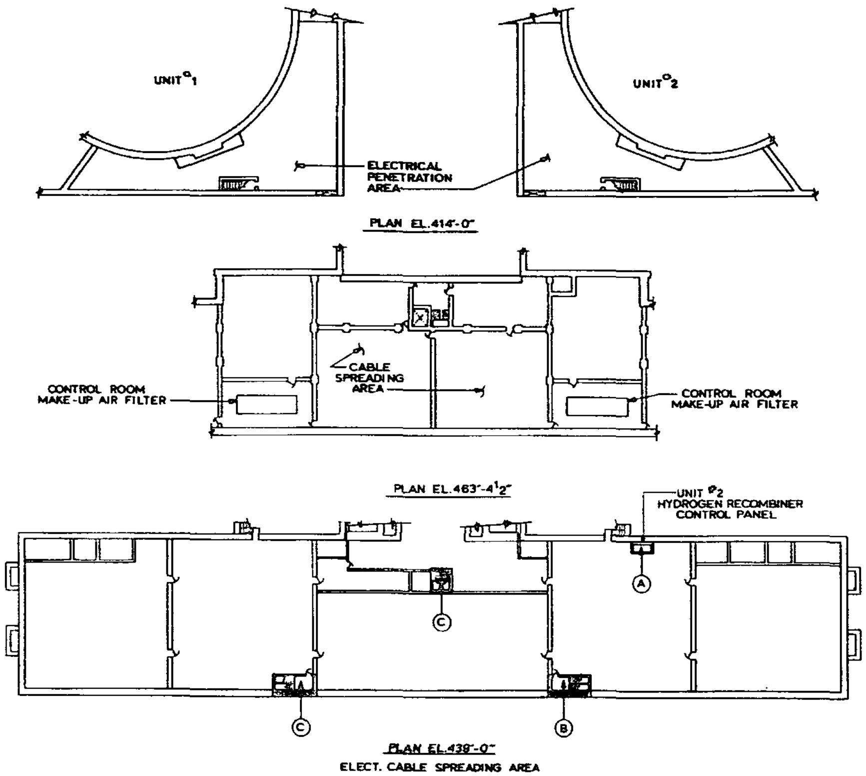
A — SX007 VALVES

PROCESS RADIATION MONITORS	
INST. NO	SERVICE DESCRIPTION
1-PR007	BTRS CHILLER SURGE TANK
2-PR007	BTRS CHILLER SURGE TANK
0-PR012	RECYLE EVAPORATOR CUBICLE AIR
0-PR013	GAS DECAY TANK CUBICLE AIR
1-PR013	RHR/CS PUMP CUBICLE AIR
2-PR013	RHR/CS PUMP CUBICLE AIR
1-PR014	RHR/CS PUMP CUBICLE AIR
2-PR014	RHR/CS PUMP CUBICLE AIR
1-PR018	CHARGING PUMP CUBICLE AIR
2-PR018	CHARGING PUMP CUBICLE AIR
0-PR002	WASTE GAS EFFLUENT

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-1

POSTACCIDENT VITAL AREAS
(SHEET 8 OF 12)

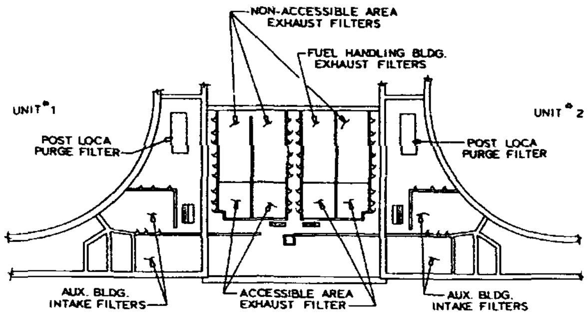


A - HYDROGEN RECOMBINER CONTROL PANEL UNIT #2
B - ACCESS ROUTE
C - ALTERNATE ACCESS ROUTES

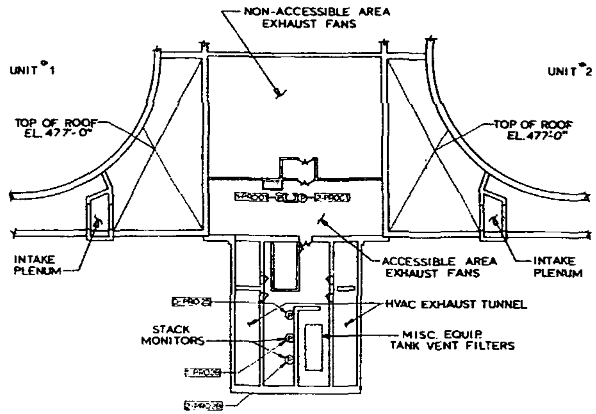
BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-1

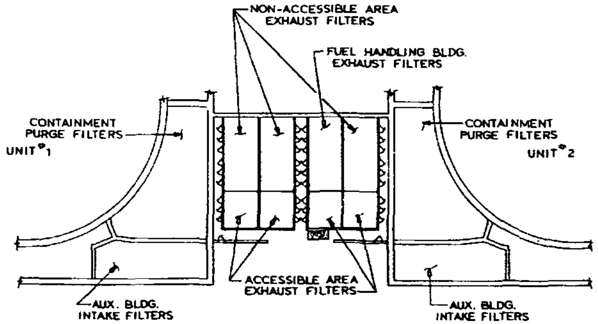
POSTACCIDENT VITAL AREAS
(SHEET 9 OF 12)



PLAN AT EL 467'4"



PLAN AT EL 475'6"



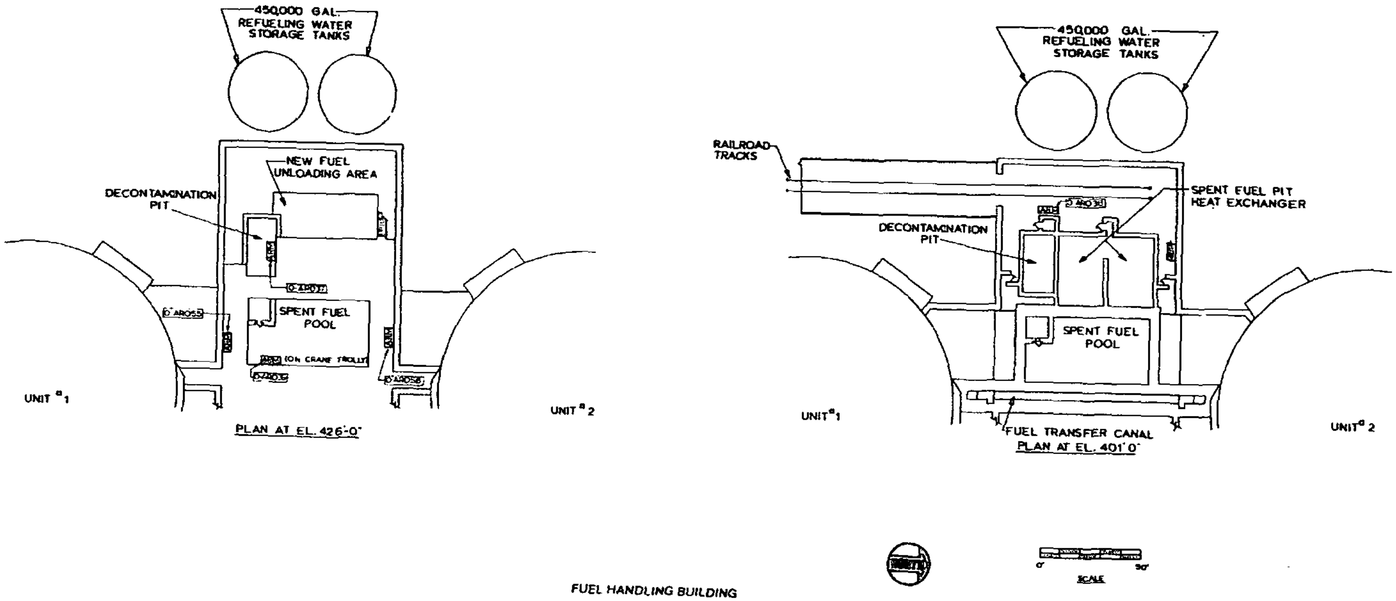
PLAN AT EL 459'-2"

PROCESS RADIATION MONITORS	
INST NO	SERVICE DESCRIPTION
1-PR001	CONTAINMENT PURGE EFFLUENT
2-PR001	CONTAINMENT PURGE EFFLUENT
0-PR025	MISC TANK VENT EFFLUENT
1-PR028	AUX BLDG VENT STACK EFFLUENT
2-PR028	AUX BLDG VENT STACK EFFLUENT

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-1

POSTACCIDENT VITAL AREAS
(SHEET 10 OF 12)

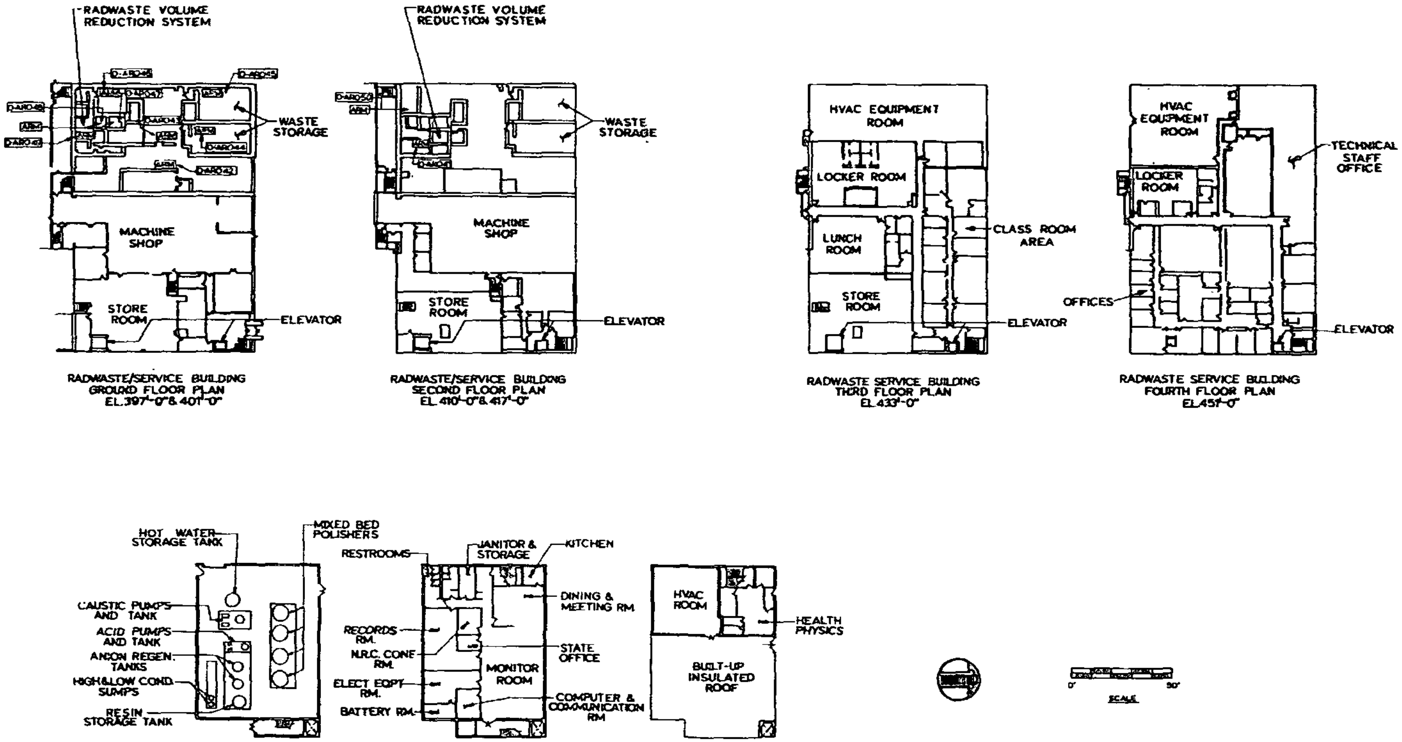


FUEL HANDLING BUILDING

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-1

POSTACCIDENT VITAL AREAS
(SHEET 11 OF 12)



RADWASTE/SERVICE BUILDING GROUND FLOOR PLAN EL.397'-0" & 401'-0"

RADWASTE/SERVICE BUILDING SECOND FLOOR PLAN EL.410'-0" & 417'-0"

RADWASTE SERVICE BUILDING THIRD FLOOR PLAN EL.433'-0"

RADWASTE SERVICE BUILDING FOURTH FLOOR PLAN EL.451'-0"

CONDENSATE CLEANUP BLDG. GROUND FLOOR PLAN EL.401'-0"

TECHNICAL SUPPORT CENTER SECOND FLOOR PLAN EL.435'-0"

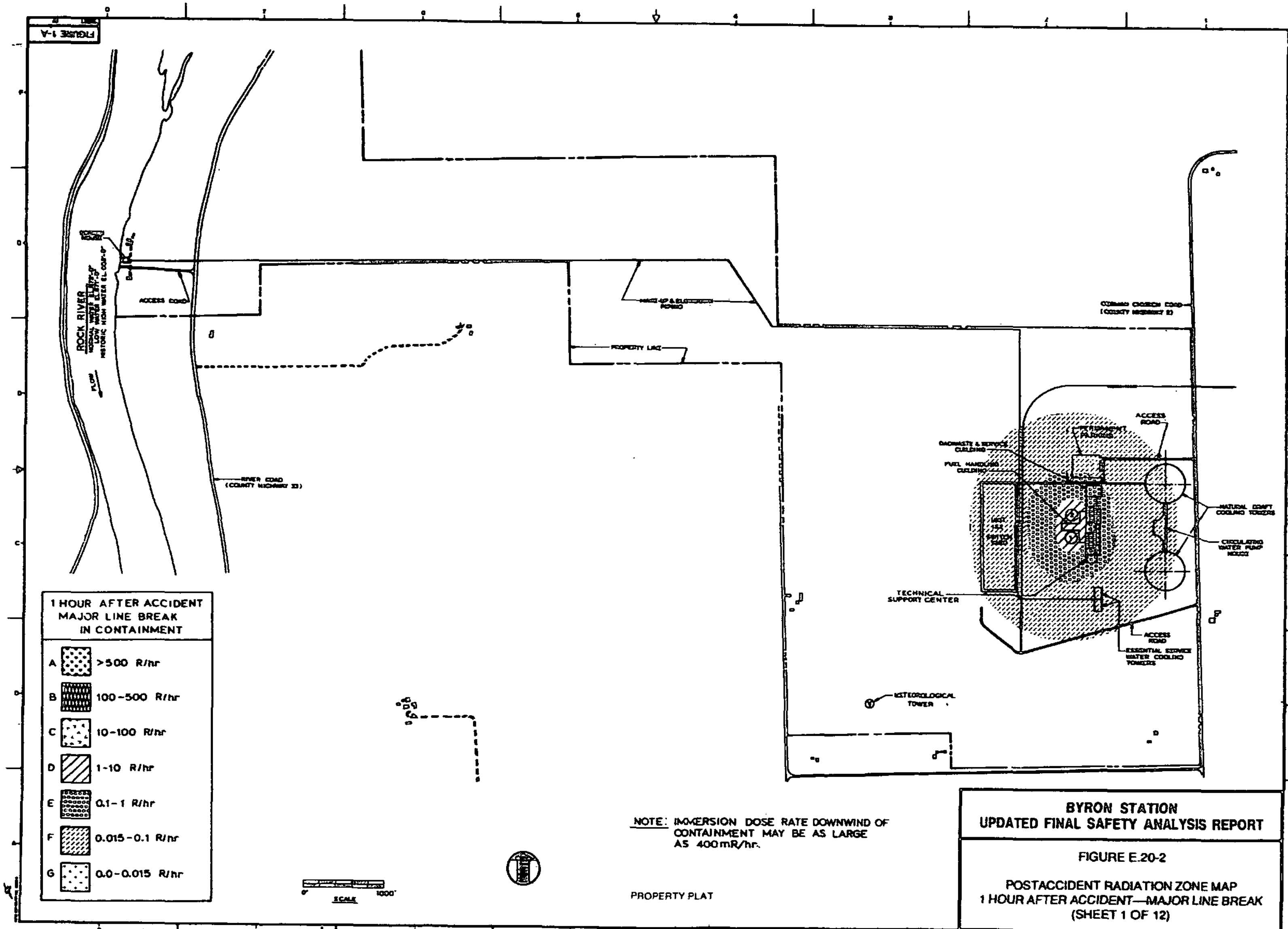
TECHNICAL SUPPORT CENTER ROOF PLAN EL. 451'-0"

RADWASTE/SERVICE BUILDING/TSC

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-1

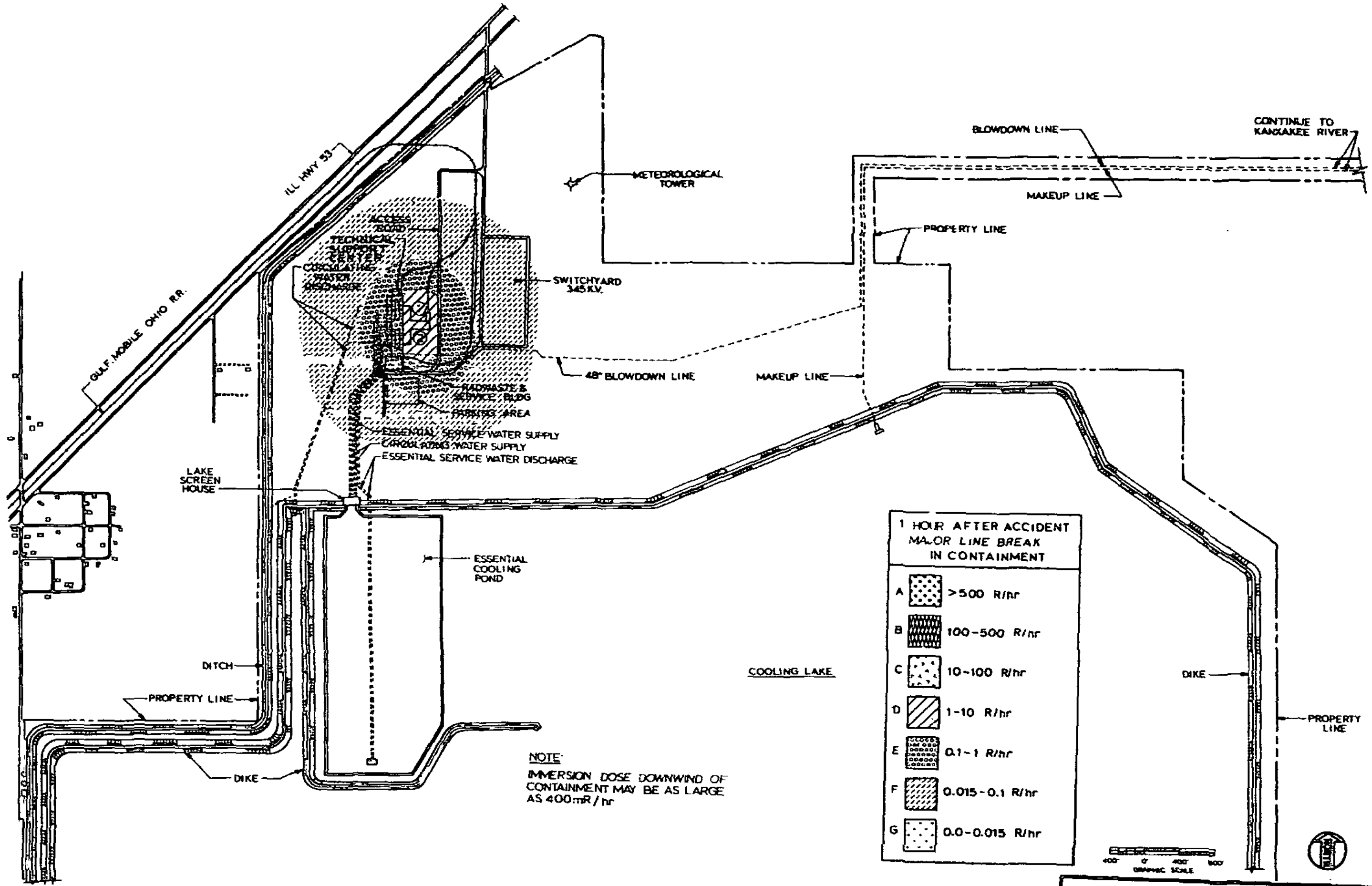
POSTACCIDENT VITAL AREAS
(SHEET 12 OF 12)



BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-2

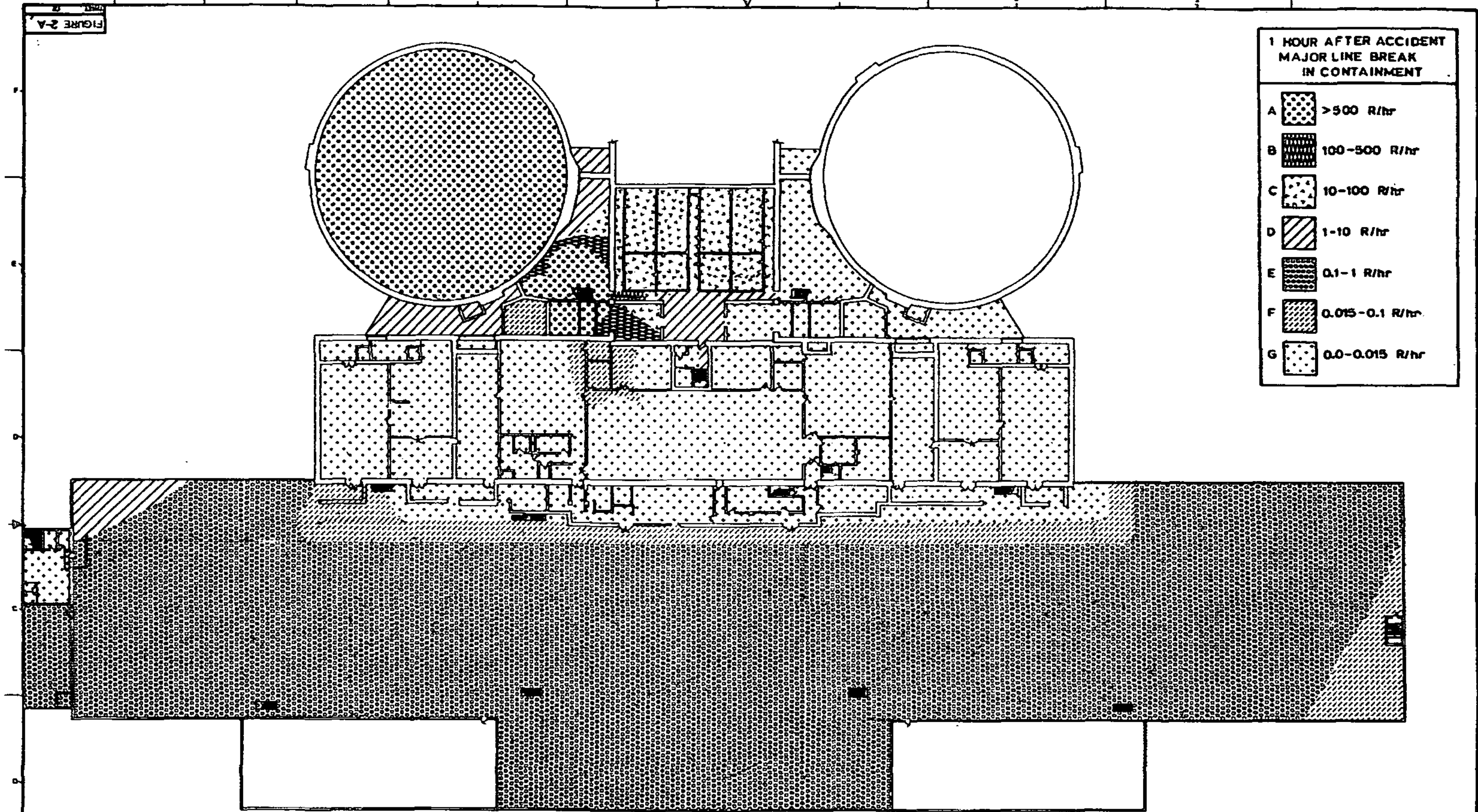
POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 1 OF 12)



BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-2

POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 2 OF 12)



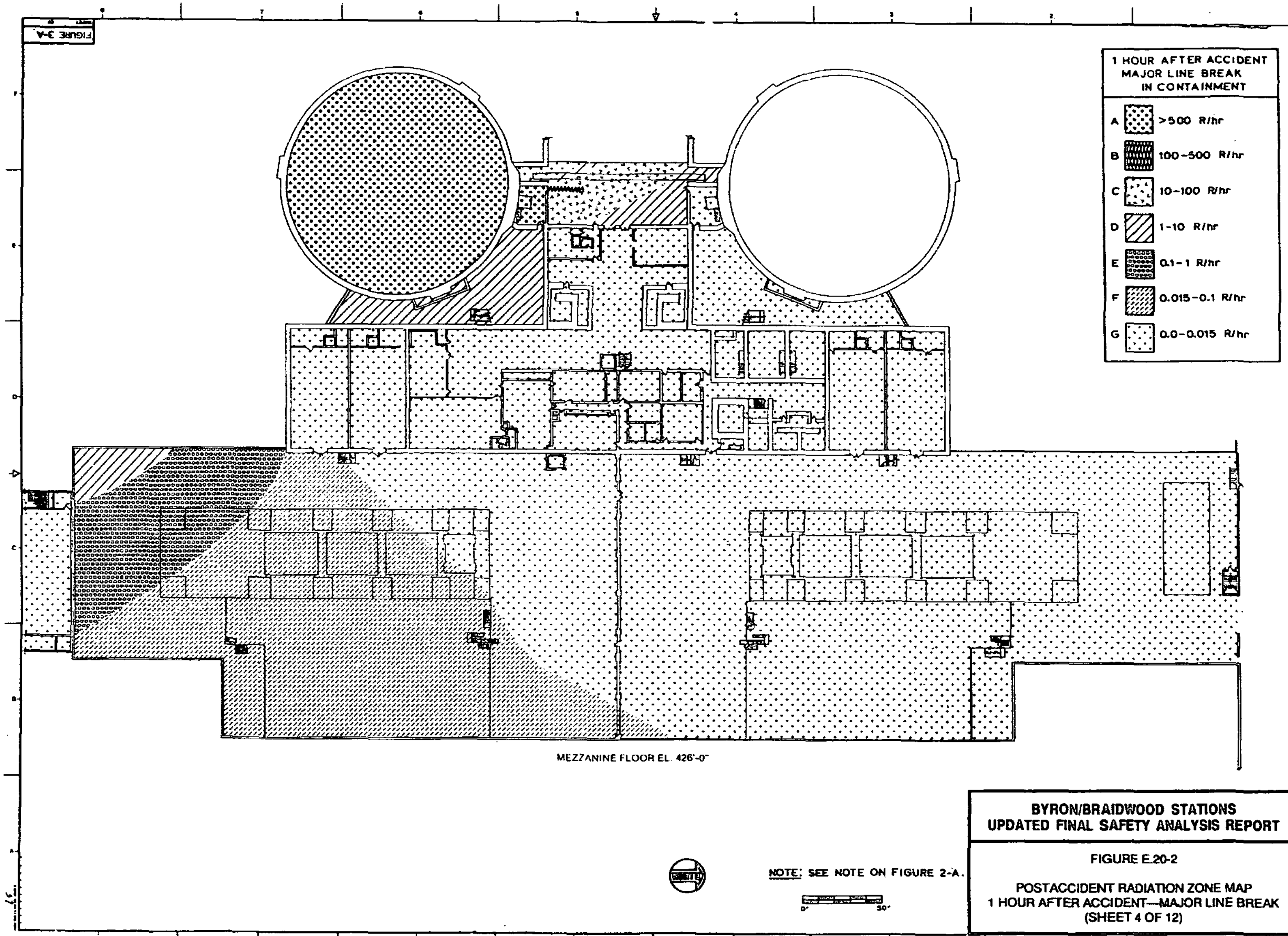
NOTE: 1. THESE RADIATION ZONE MAPS ASSUME THAT THE ACCIDENT HAS OCCURRED IN UNIT 1. IF THE ACCIDENT OCCURS IN UNIT 2, THE RADIATION ZONES AROUND UNIT 2 CONTAINMENT WILL BE THE MIRROR IMAGE OF THE ZONES SHOWN HERE.

2. AIRBORNE RADIATION MAY CAUSE AN AN ADDITIONAL WHOLE BODY DOSE RATE AS LARGE AS 80 mR/hr IN THE CONTROL ROOM AND T.S.C. & 160mR/hr IN OTHER AREAS OF THE PLANT. LOCALLY HIGHER AIRBORNE DOSE RATES WILL OCCUR IF PRIMARY COOLANT IS SPILLED.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-2

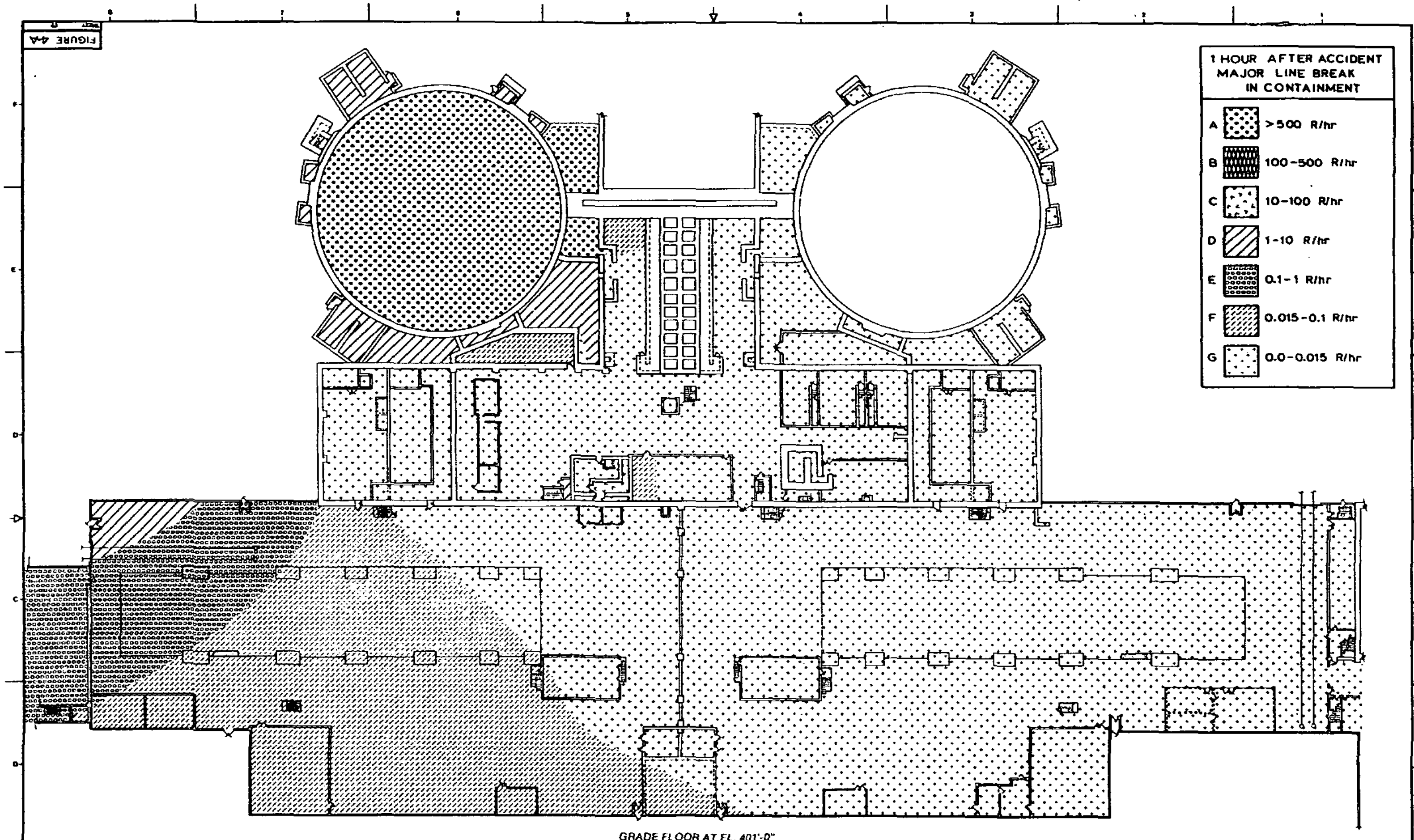
POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 3 OF 12)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-2

POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 4 OF 12)

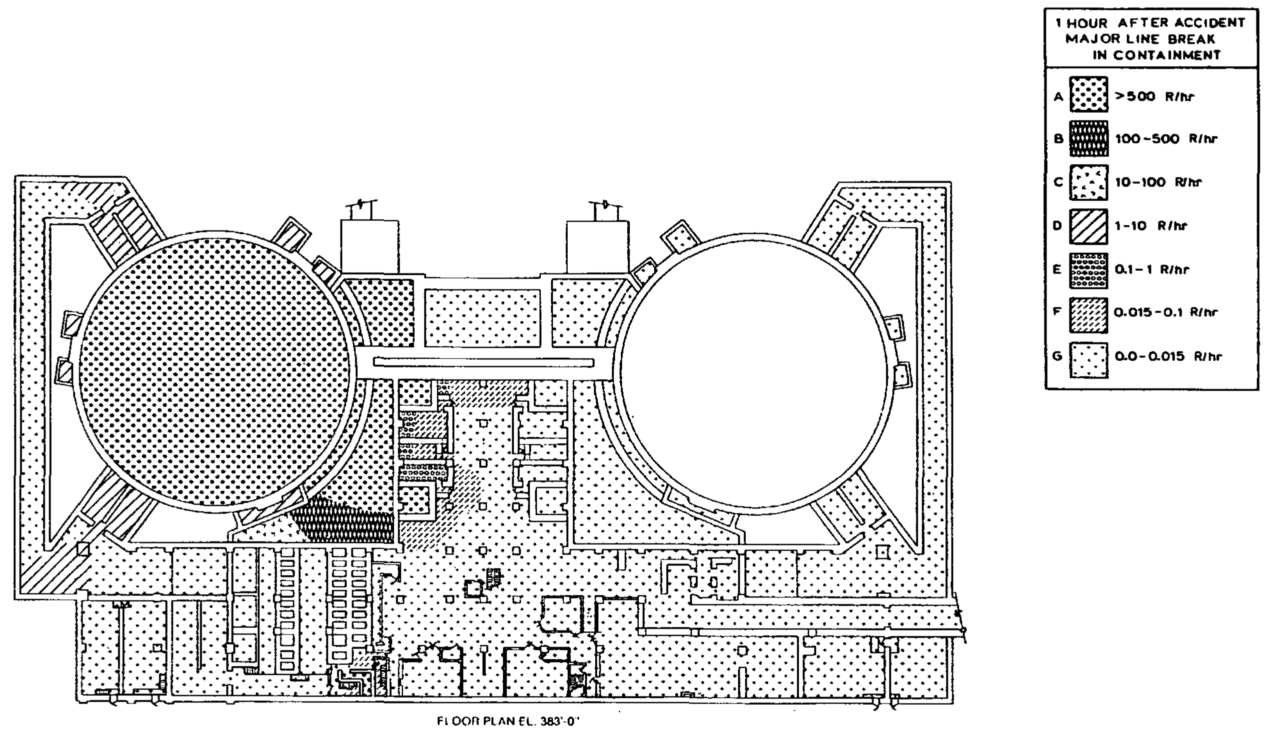


NOTE: SEE NOTE ON FIGURE 2-A.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-2

POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 5 OF 12)

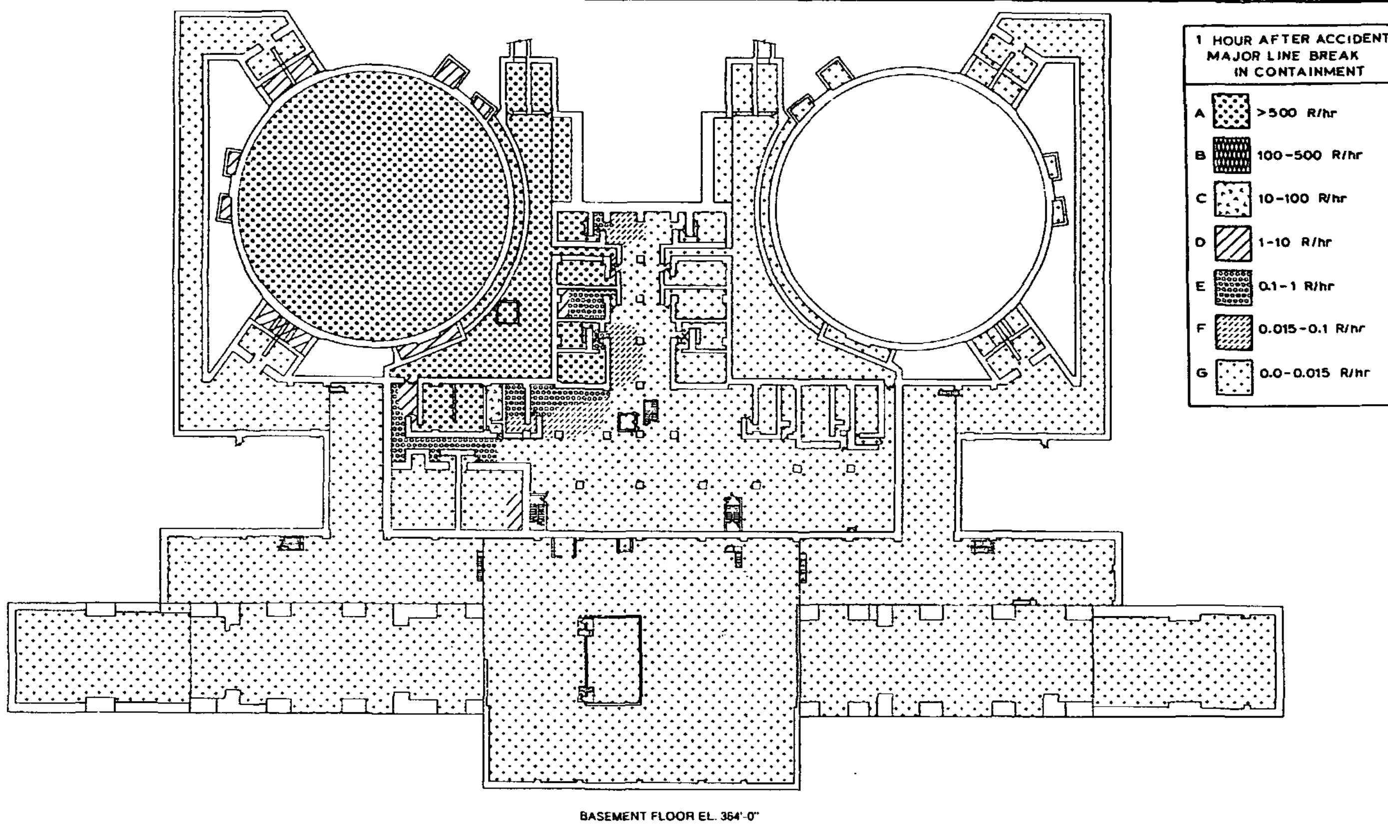


NOTE: SEE NOTE ON FIGURE 2-A.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-2

POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 6 OF 12)

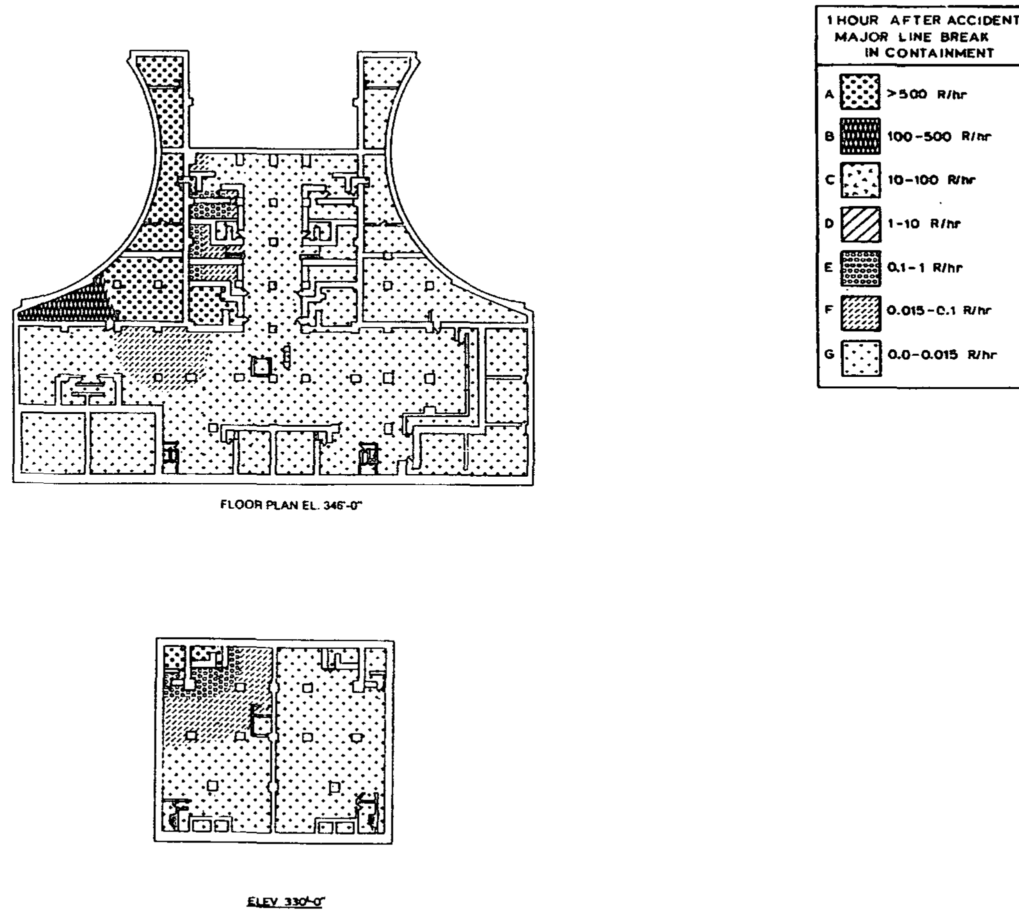


NOTE: SEE NOTE ON FIGURE 2-A.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-2

POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 7 OF 12)

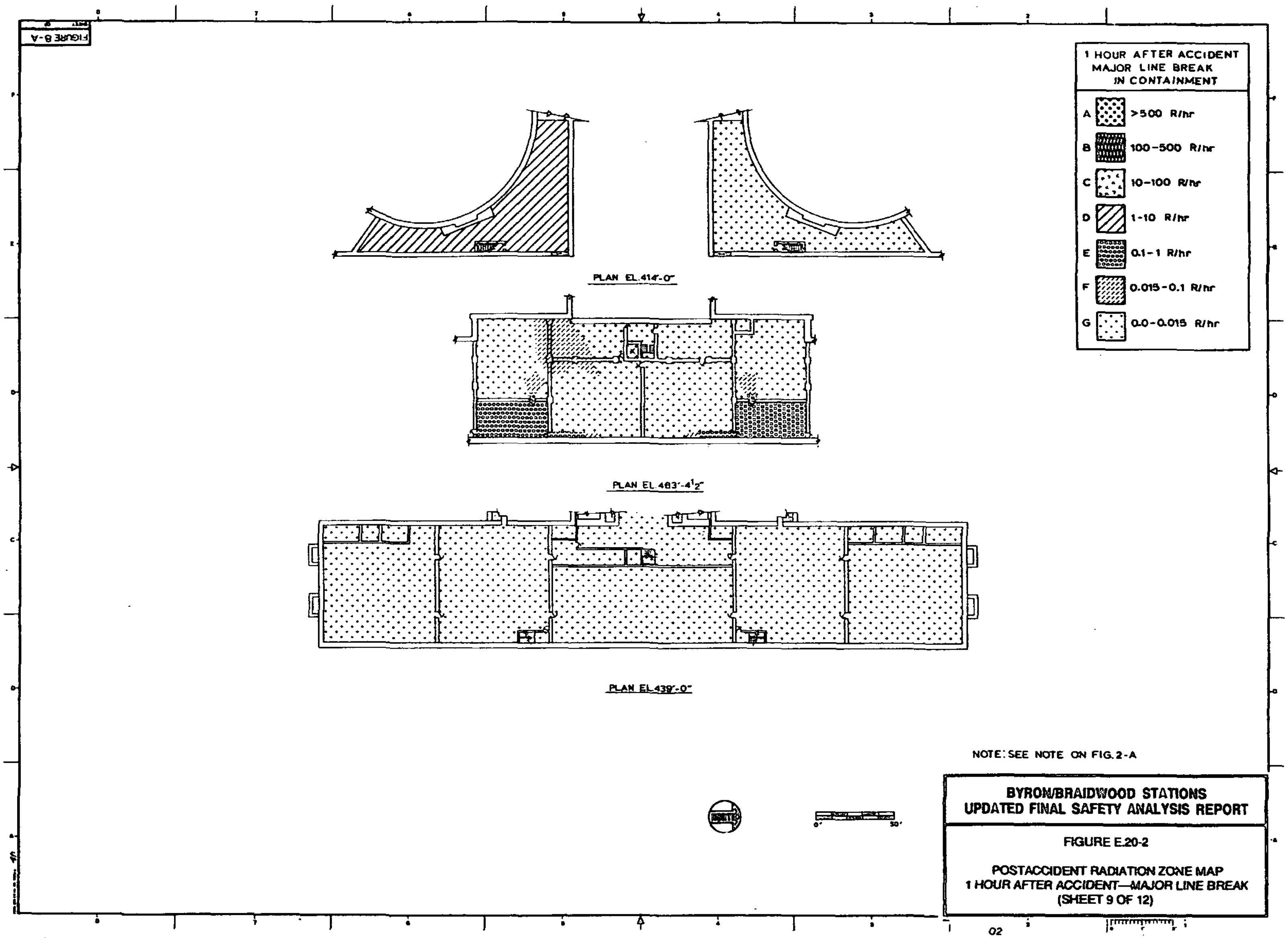


NOTE: SEE NOTE ON FIG. 2-A

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-2

POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 8 OF 12)

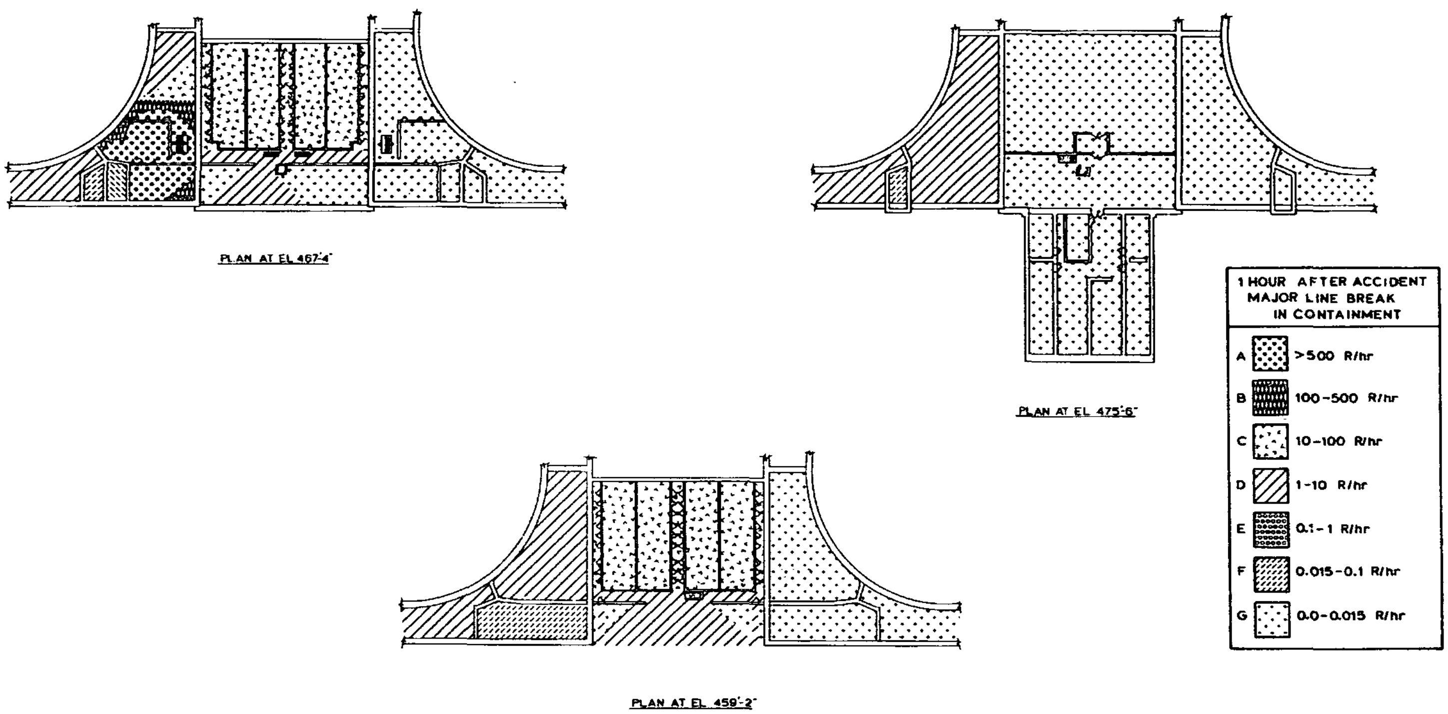


NOTE: SEE NOTE ON FIG. 2-A

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-2

POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 9 OF 12)

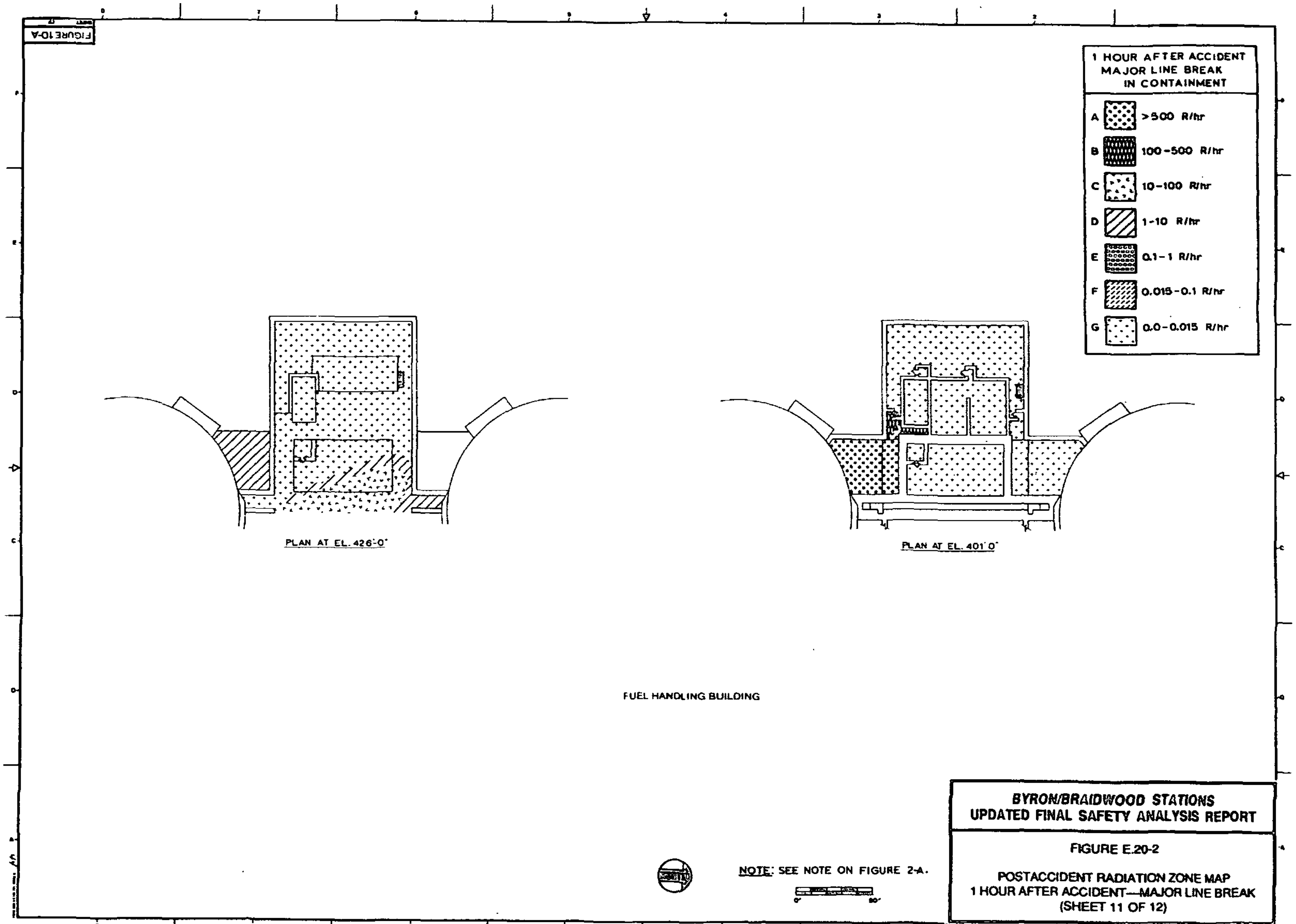


NOTE: SEE NOTE ON FIGURE 2-A.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-2

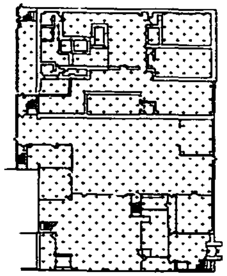
POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 10 OF 12)



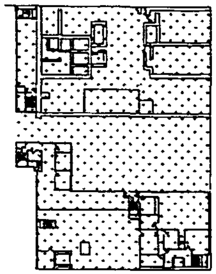
BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-2

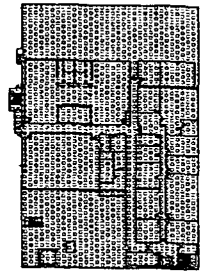
POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 11 OF 12)



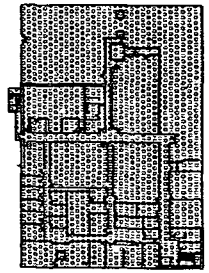
GROUND FLOOR PLAN
EL. 397'-0" & 401'-0"



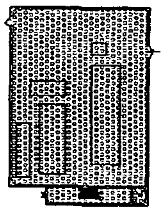
SECOND FLOOR PLAN
EL. 410'-0" & 417'-0"



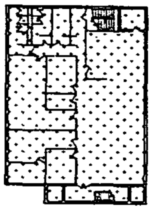
THIRD FLOOR PLAN
EL. 433'-0"



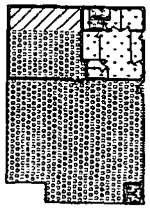
FOURTH FLOOR PLAN
EL. 451'-0"



CONDENSATE CLEAN-UP BLDG.
EL. 401'-0"



TECHNICAL SUPPORT CENTER
EL. 435'-0"



TECHNICAL SUPPORT CENTER
EL. 451'-0"

RADWASTE/SERVICE BUILDING/TSC

1 HOUR AFTER ACCIDENT MAJOR LINE BREAK IN CONTAINMENT	
A	>500 R/hr
B	100-500 R/hr
C	10-100 R/hr
D	1-10 R/hr
E	0.1-1 R/hr
F	0.015-0.1 R/hr
G	0.0-0.015 R/hr

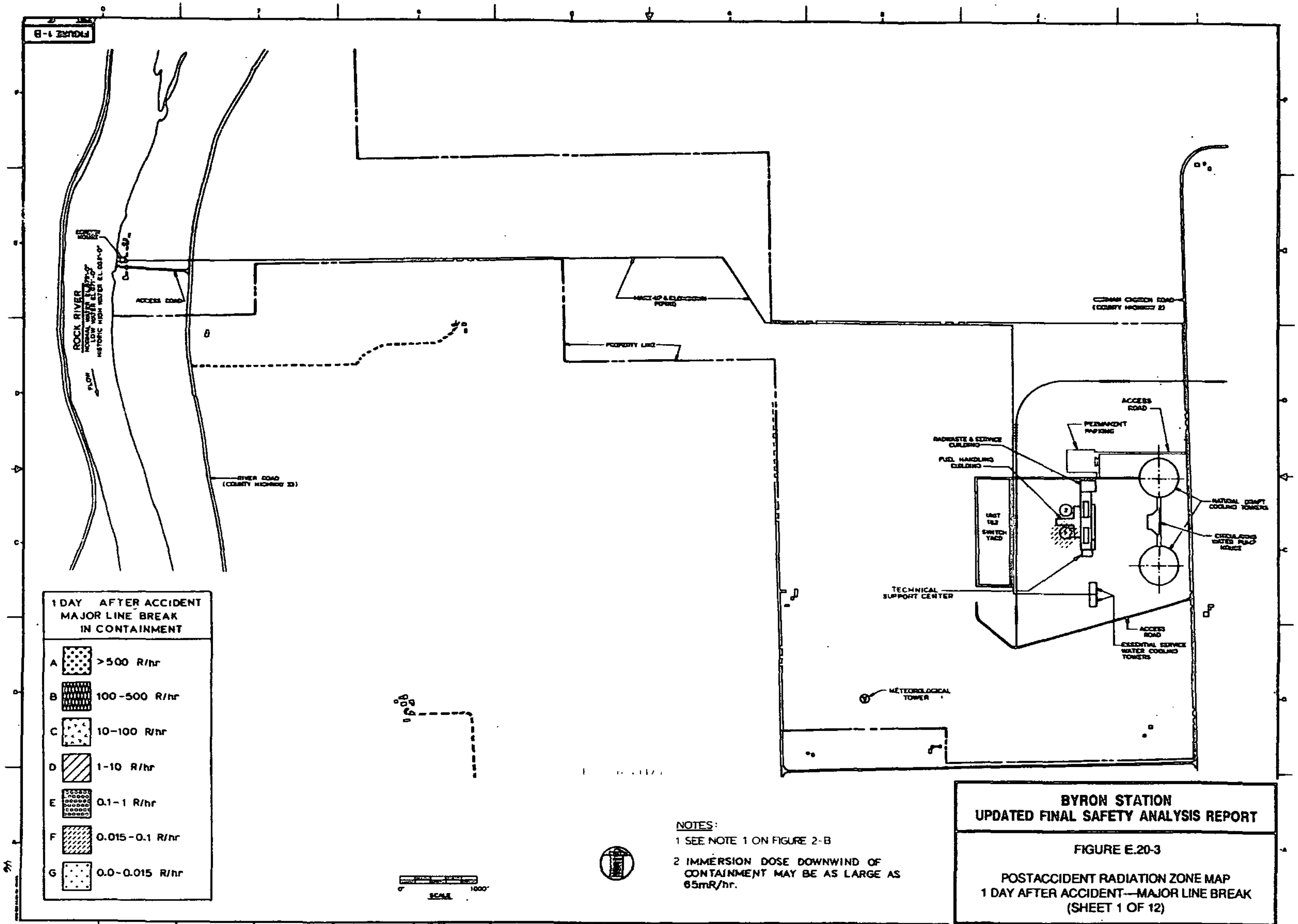
NOTE 1. FOR THE RADWASTE/SERVICE BLDG. THIS RADIATION ZONE MAP ASSUMES THAT THE ACCIDENT HAS OCCURRED IN UNIT #2. IF THE ACCIDENT OCCURS IN UNIT #1, THEN ALL ZONES ARE LESS THAN 15mR/hr

2. SEE NOTE #2 ON FIGURE 2-A

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-2

POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 12 OF 12)



NOTES:

1 SEE NOTE 1 ON FIGURE 2-B

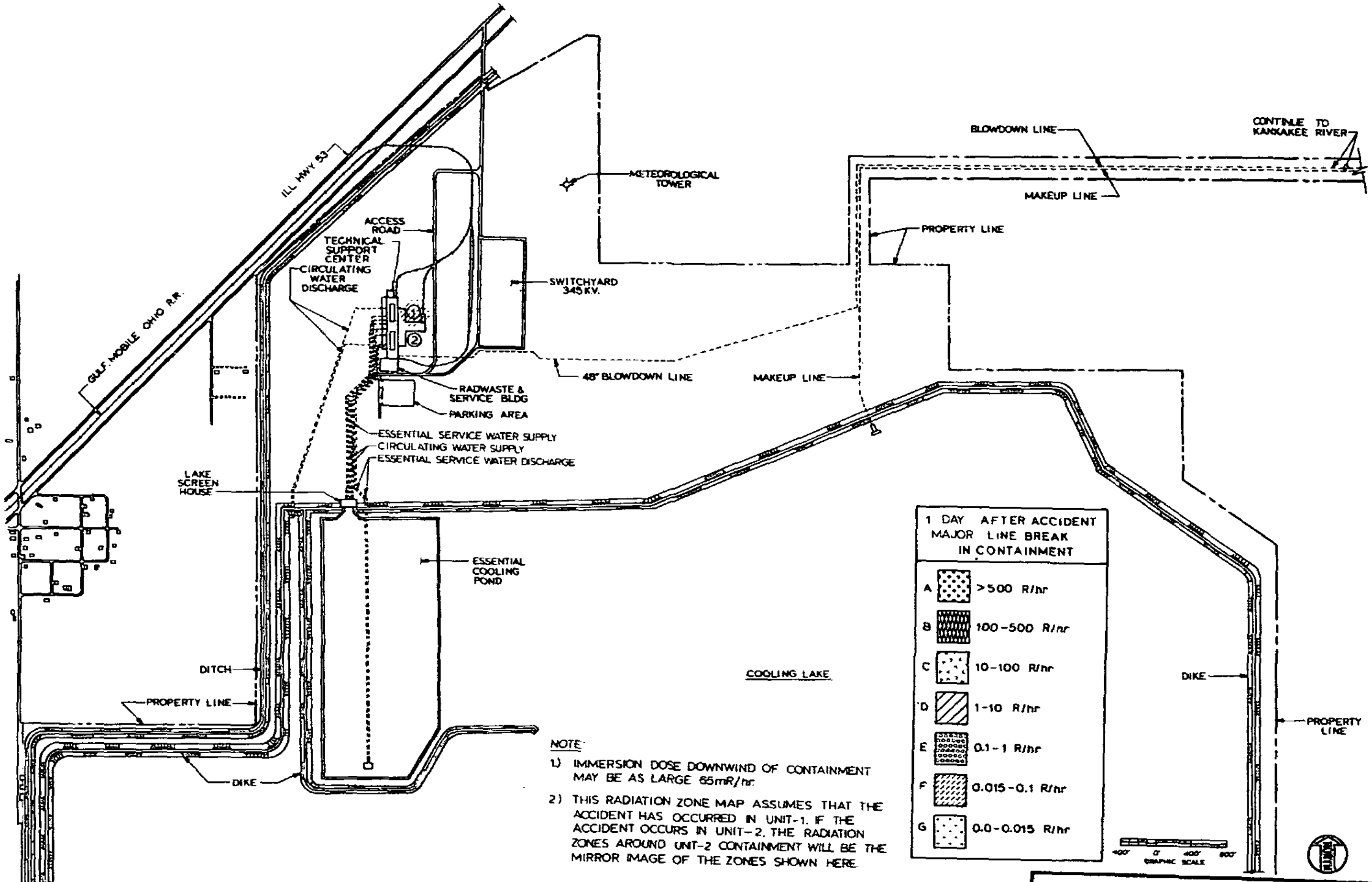
2 IMMERSION DOSE DOWNWIND OF CONTAINMENT MAY BE AS LARGE AS 65mR/hr.

BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-3

POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 1 OF 12)

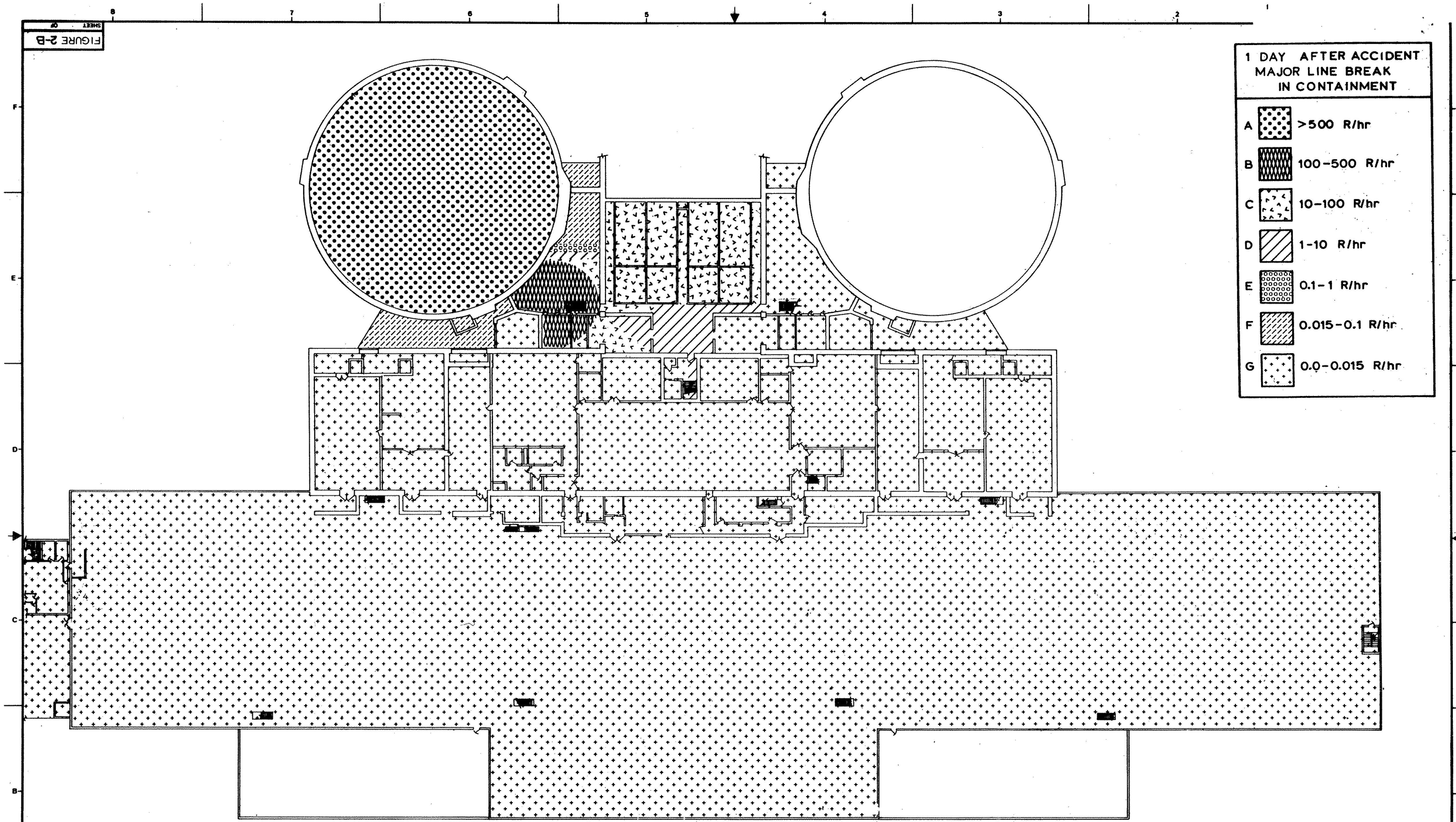
02



BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-3

POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 2 OF 12)



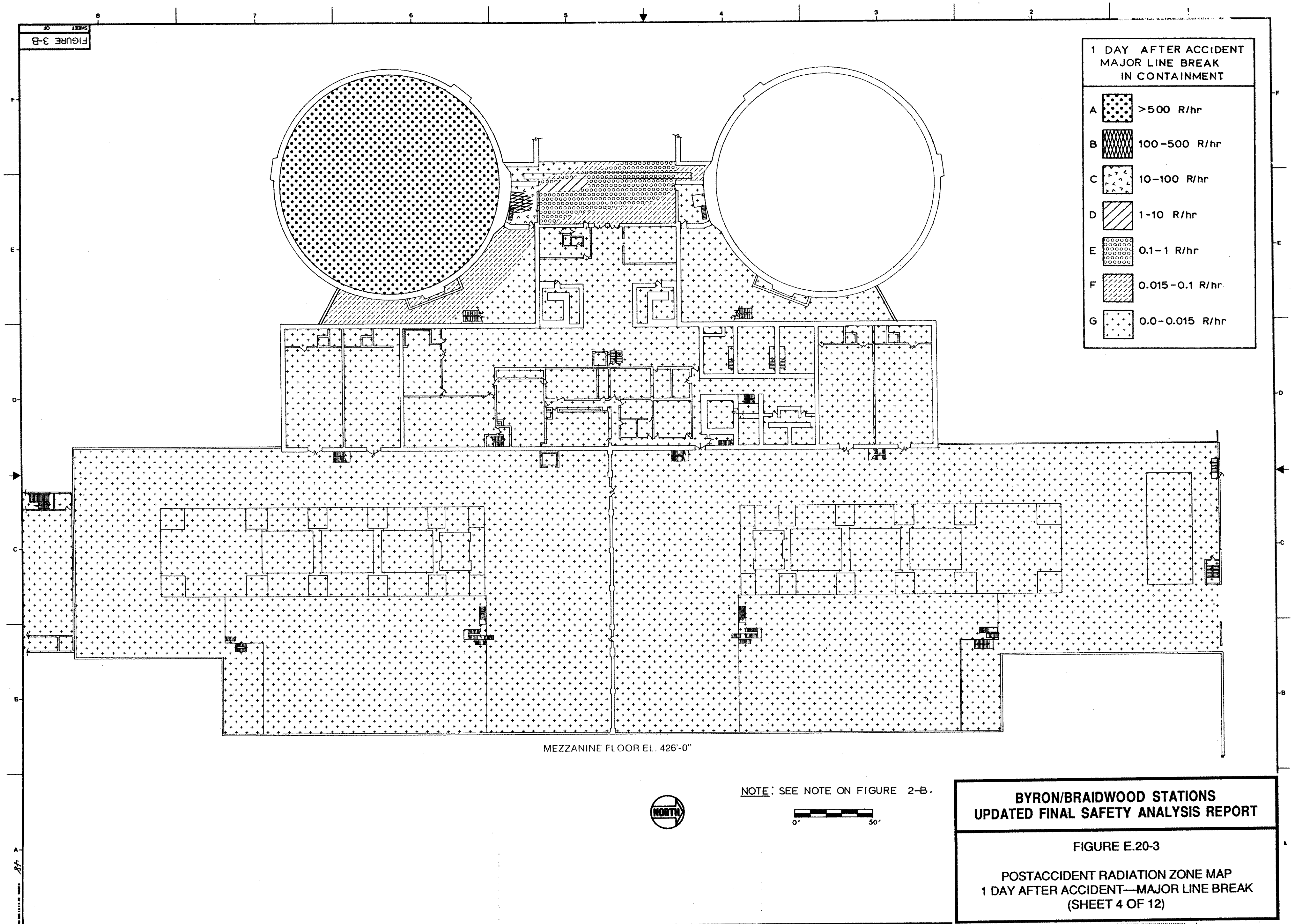
NOTE: 1. THESE RADIATION ZONE MAPS ASSUME THAT THE ACCIDENT HAS OCCURRED IN UNIT 1. IF THE ACCIDENT OCCURS IN UNIT 2, THE RADIATION ZONES AROUND UNIT 2 CONTAINMENT WILL BE THE MIRROR IMAGE OF THE ZONES SHOWN HERE.

2. AIRBORNE RADIATION MAY CAUSE AN ADDITIONAL WHOLE BODY DOSE RATE AS LARGE AS 20mR/hr. IN THE CONTROL RM. & T.S.C. & 17mR/hr IN OTHER AREAS OF THE PLANT. LOCALLY HIGHER AIRBORNE DOSE RATES WILL OCCUR IF PRIMARY COOLANT IS SPILLED.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-3

POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 3 OF 12)



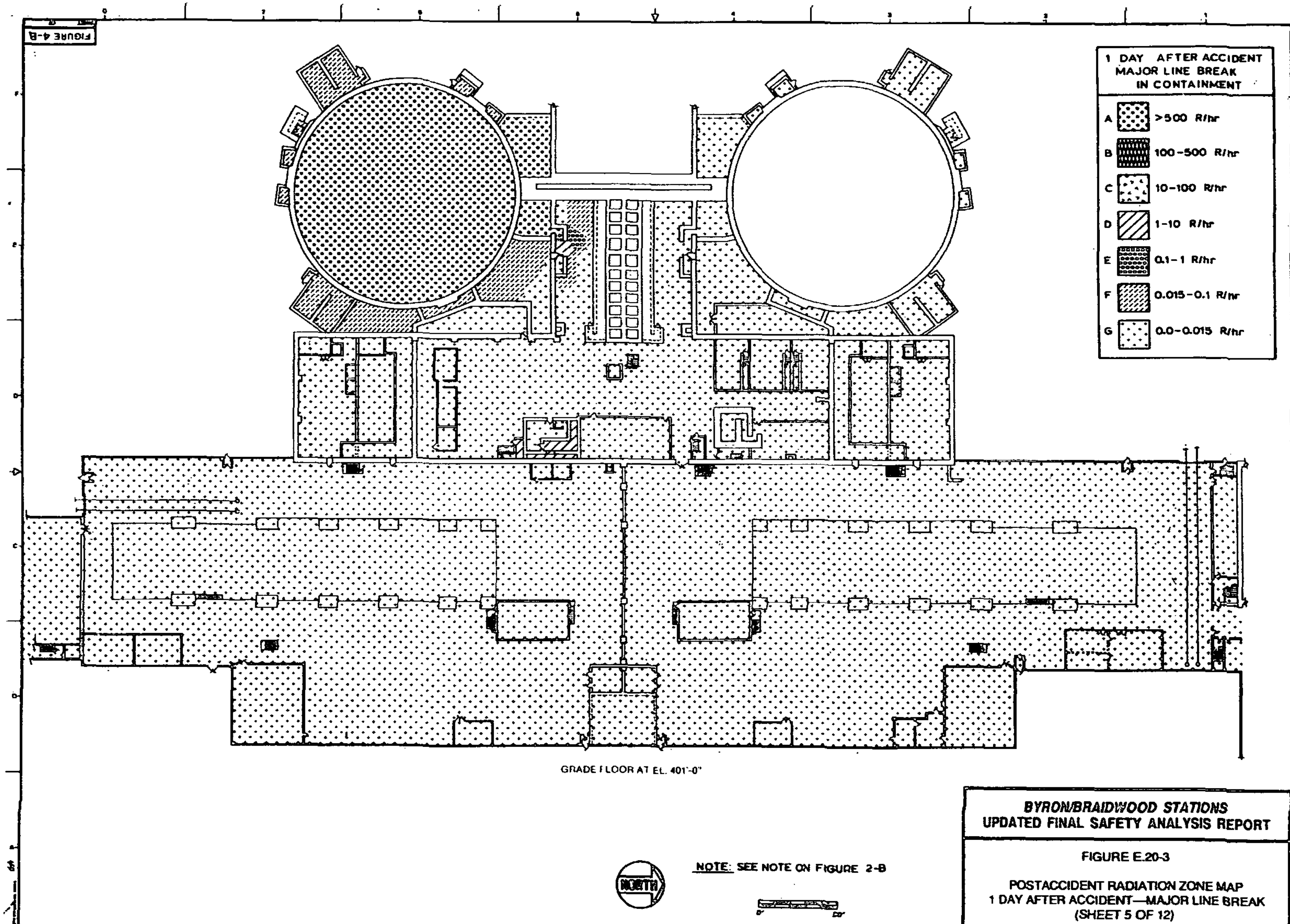
NOTE: SEE NOTE ON FIGURE 2-B.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-3

POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 4 OF 12)

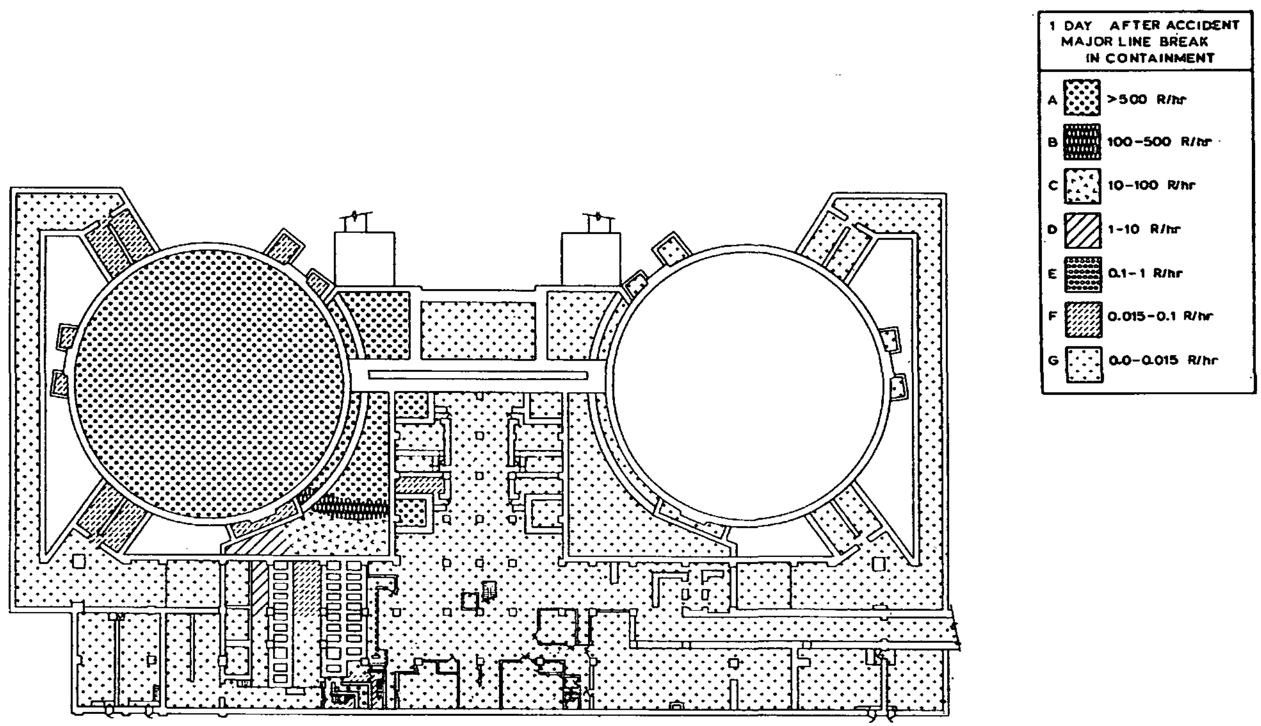
02



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-3

POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 5 OF 12)



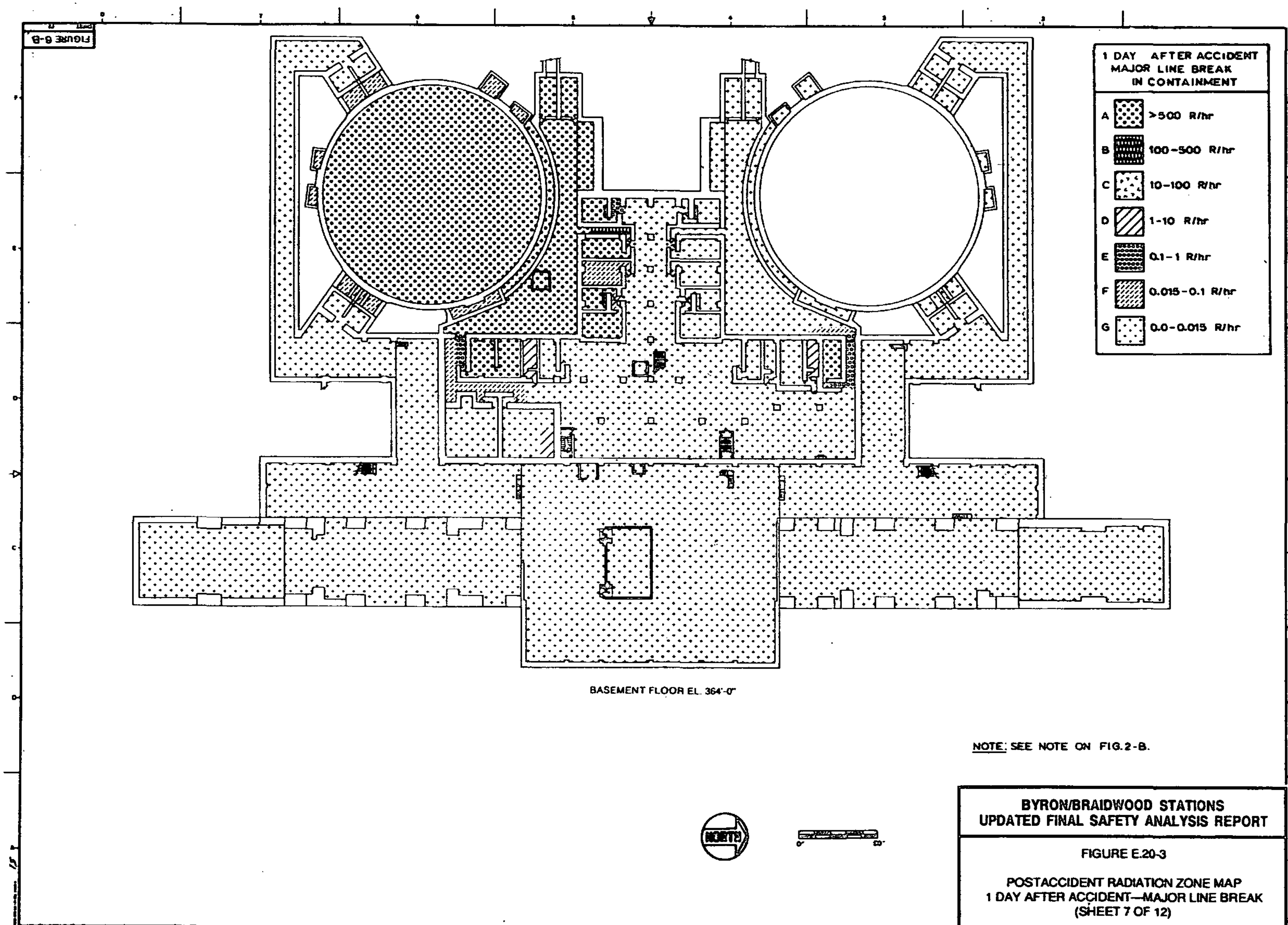
FLOOR PLAN EL. 383'-0"

NOTE: SEE NOTE ON FIGURE 2-B.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-3

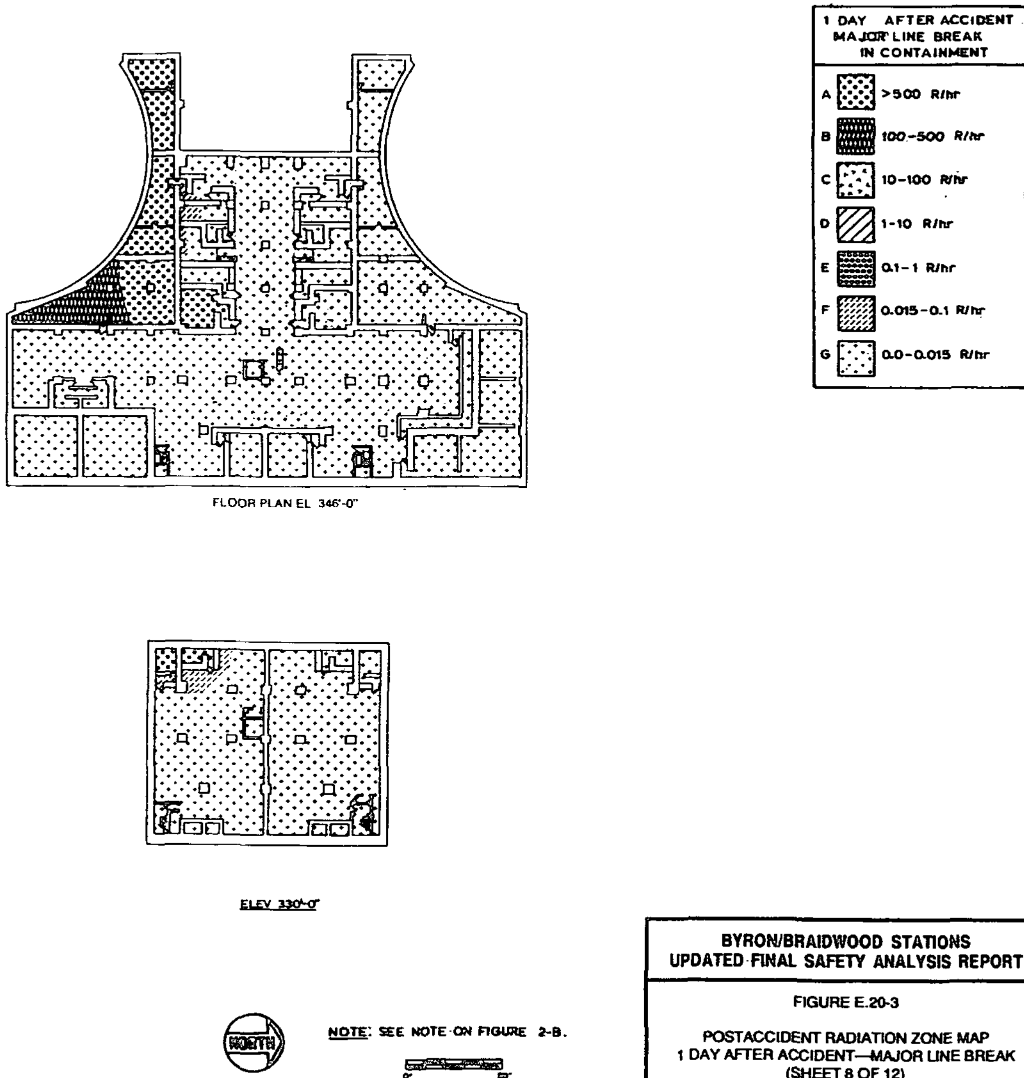
POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 6 OF 12)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-3

POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 7 OF 12)

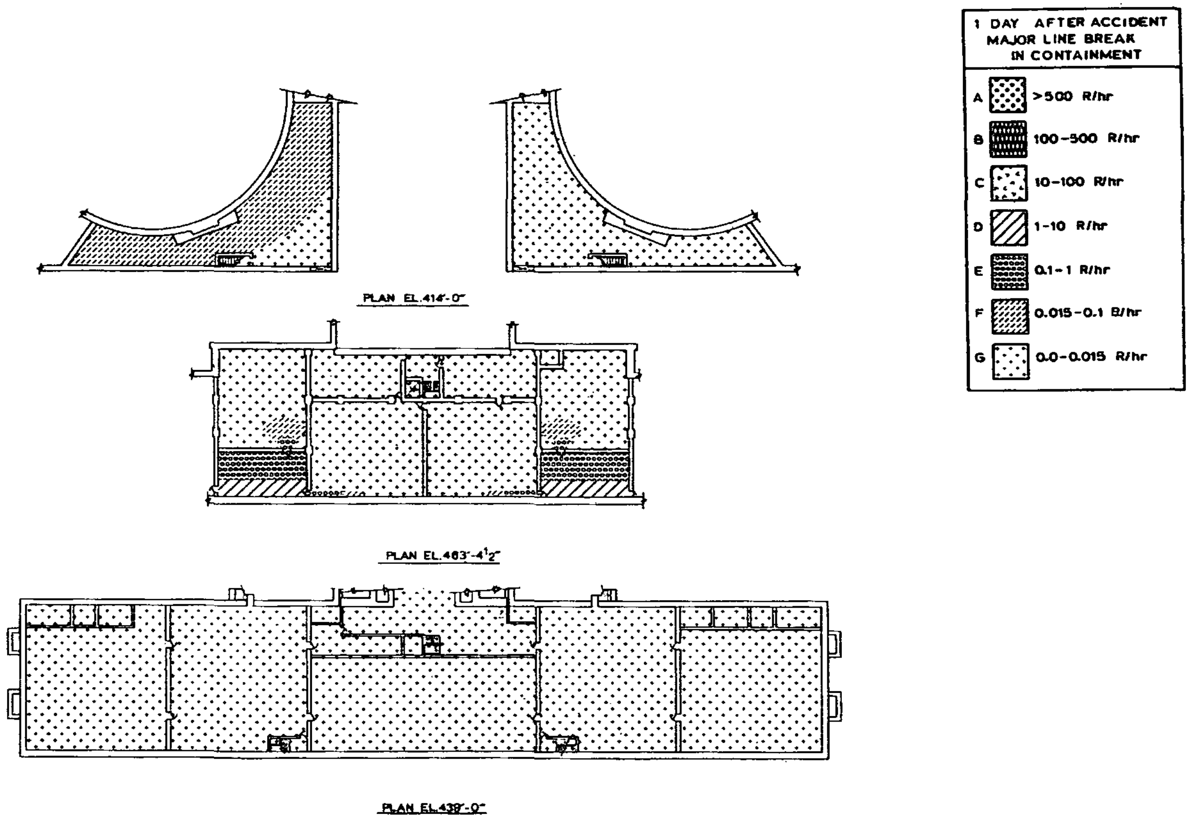


NOTE: SEE NOTE ON FIGURE 2-B.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-3

POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 8 OF 12)

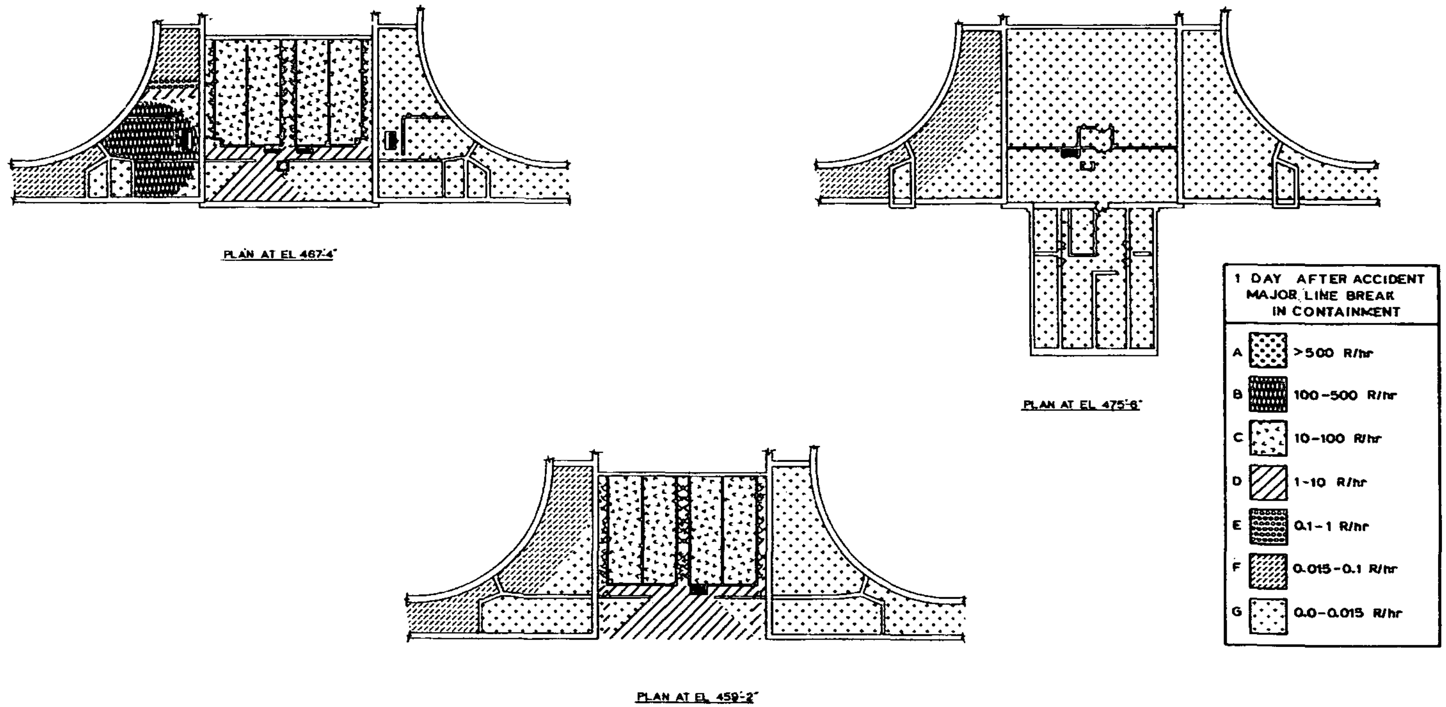


NOTE: SEE NOTE ON FIGURE 2-B

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-3

POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 9 OF 12)

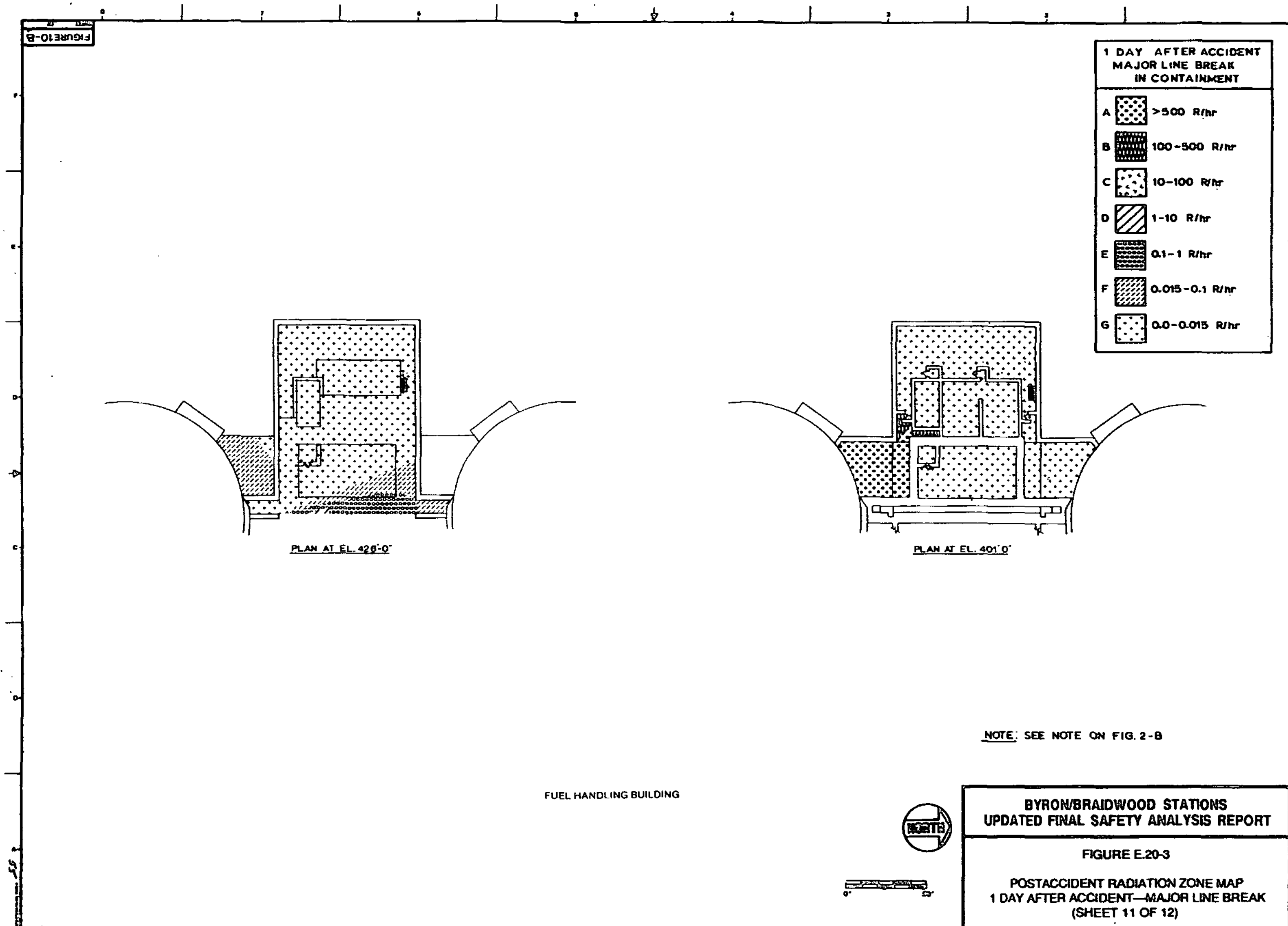


NOTE: SEE NOTE ON FIGURE 2 B.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-3

POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 10 OF 12)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-3

POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 11 OF 12)

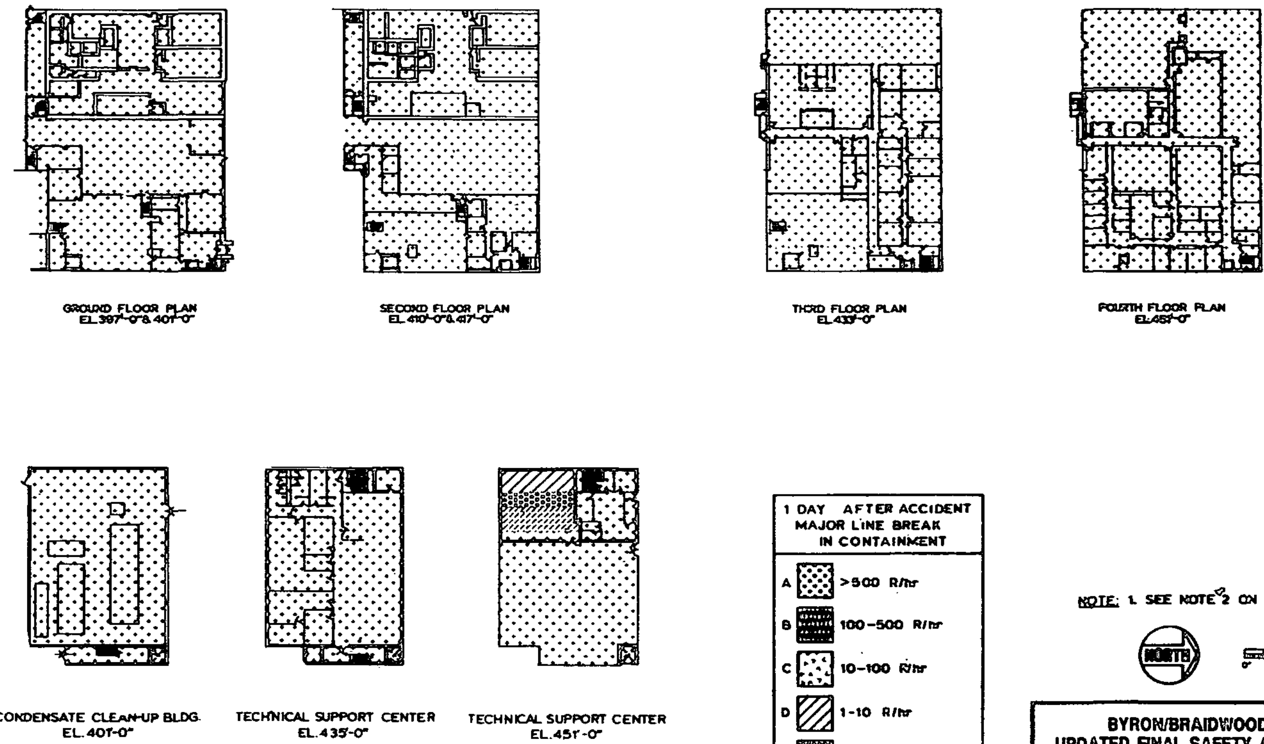


FIGURE 11-B

1 DAY AFTER ACCIDENT MAJOR LINE BREAK IN CONTAINMENT

A	>500 R/hr
B	100-500 R/hr
C	10-100 R/hr
D	1-10 R/hr
E	0.1-1 R/hr
F	0.015-0.1 R/hr
G	0.0-0.015 R/hr

NOTE: 1. SEE NOTE 2 ON FIGURE 2-B

NORTH

0' 50'

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-3

POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 12 OF 12)

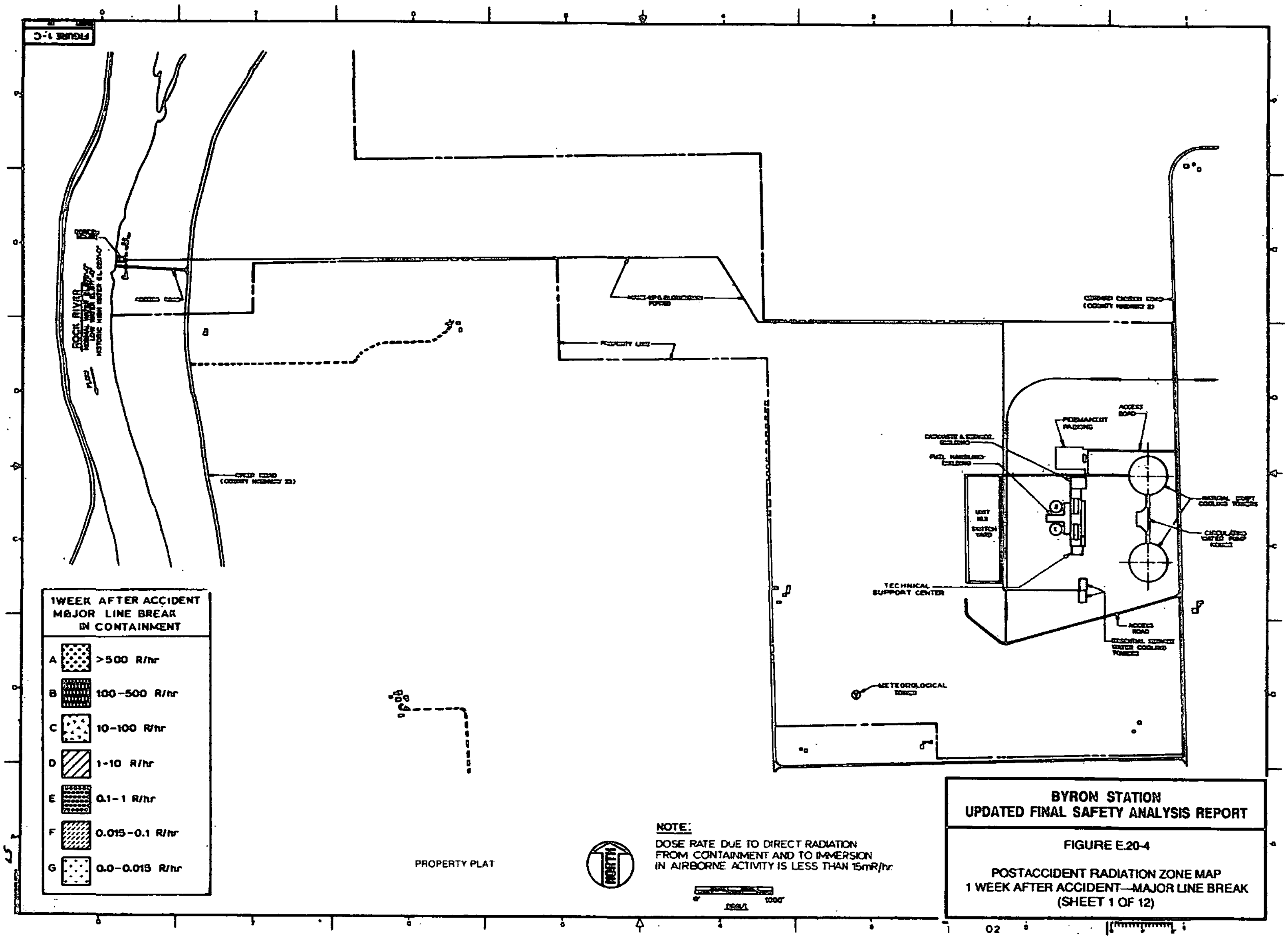
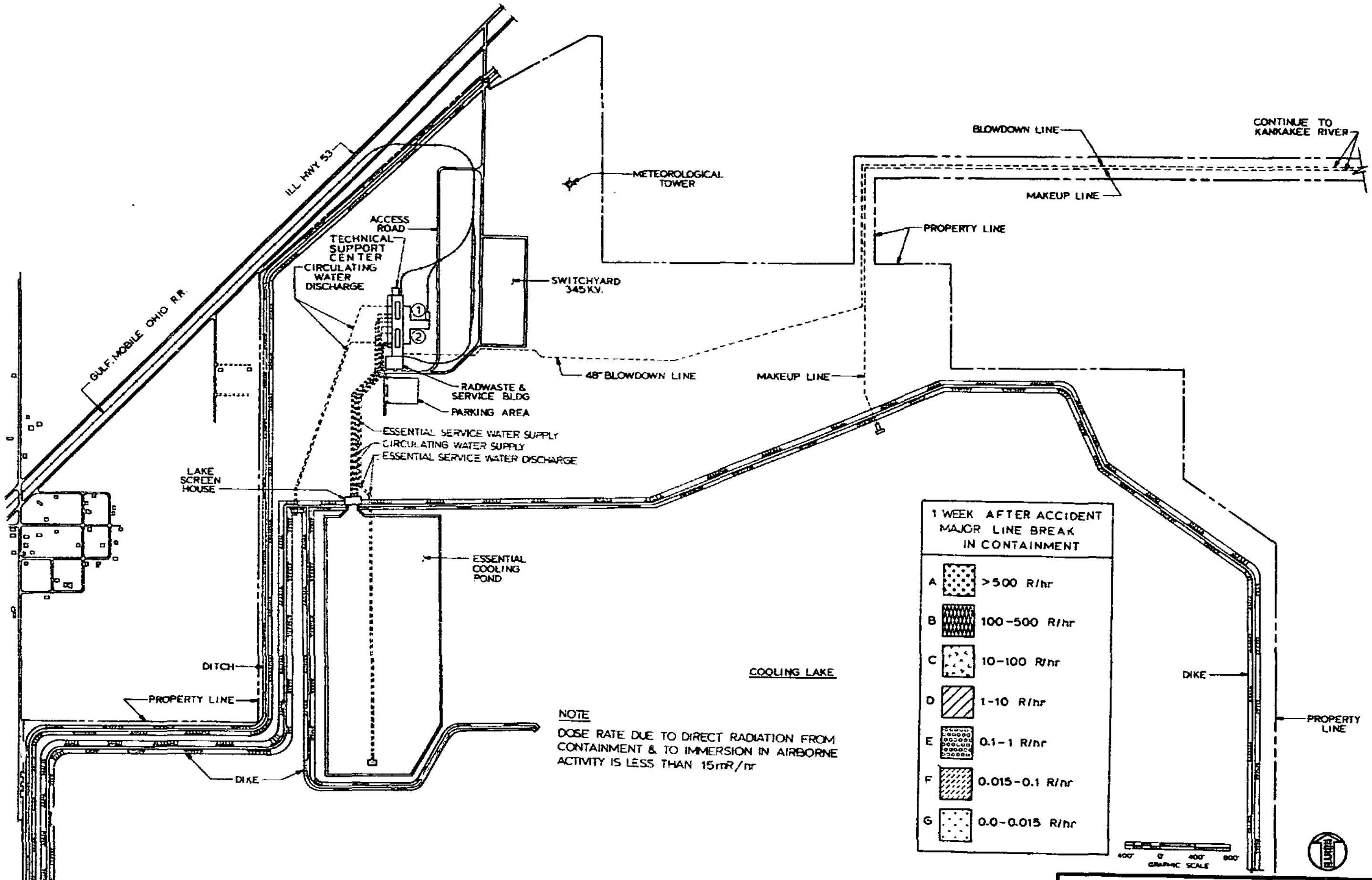


FIGURE 1-C
ROCK RIVER
TECHNICAL SUPPORT CENTER
METEOROLOGICAL TOWER
1WEEK AFTER ACCIDENT
MAJOR LINE BREAK
IN CONTAINMENT
A >500 R/hr
B 100-500 R/hr
C 10-100 R/hr
D 1-10 R/hr
E 0.1-1 R/hr
F 0.015-0.1 R/hr
G 0.0-0.015 R/hr
PROPERTY PLAT
NORTH
NOTE:
DOSE RATE DUE TO DIRECT RADIATION FROM CONTAINMENT AND TO IMMERSION IN AIRBORNE ACTIVITY IS LESS THAN 15mR/hr.
0' 1000'
BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT
FIGURE E.20-4
POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 1 OF 12)



BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-4

POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 2 OF 12)

REVISION 3
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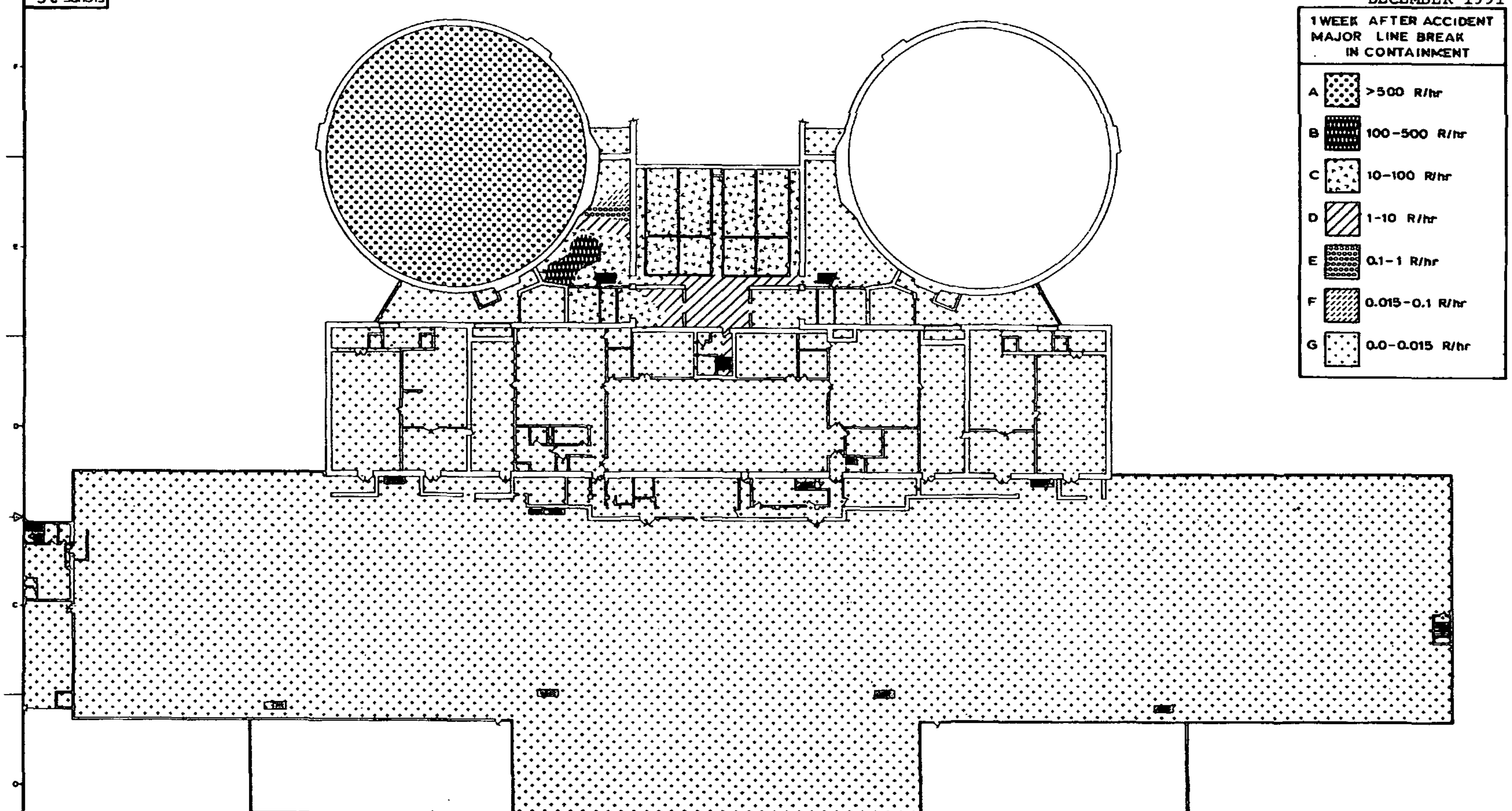


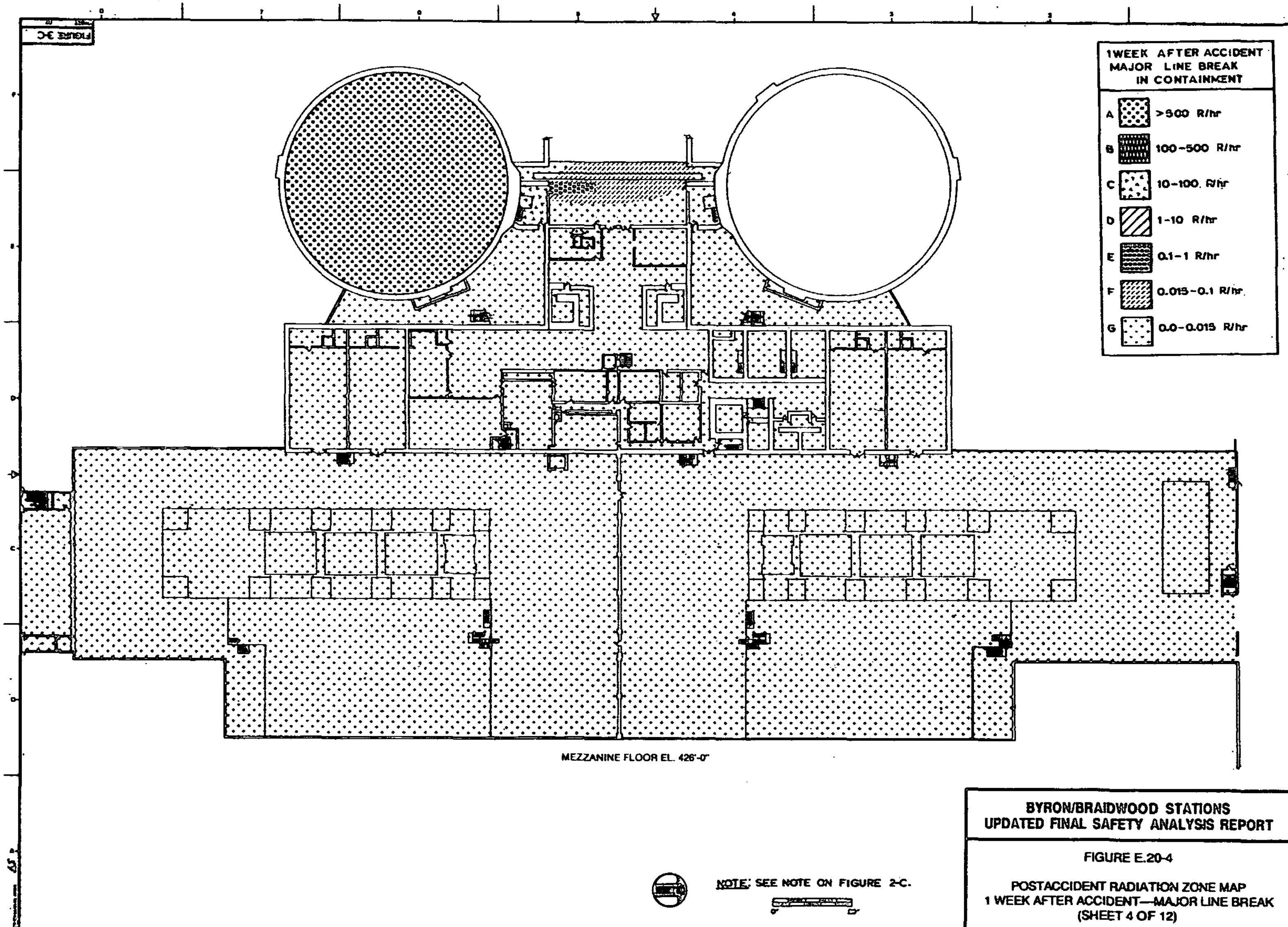
NOTE: 1. THESE RADIATION ZONE MAPS ASSUME THAT THE ACCIDENT HAS OCCURRED IN UNIT # 1. IF THE ACCIDENT OCCURS IN UNIT # 2, THE RADIATION ZONES AROUND UNIT # 2 CONTAINMENT WILL BE THE MIRROR IMAGE OF THE ZONES SHOWN HERE.

2. DOSE RATE DUE TO AIRBORNE RADIATION IS LESS THAN 15 mR/hr IN ALL AREAS OF THE PLANT. LOCALLY HIGHER AIRBORNE DOSE RATES WILL OCCUR IF PRIMARY COOLANT IS SPILLED

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-4

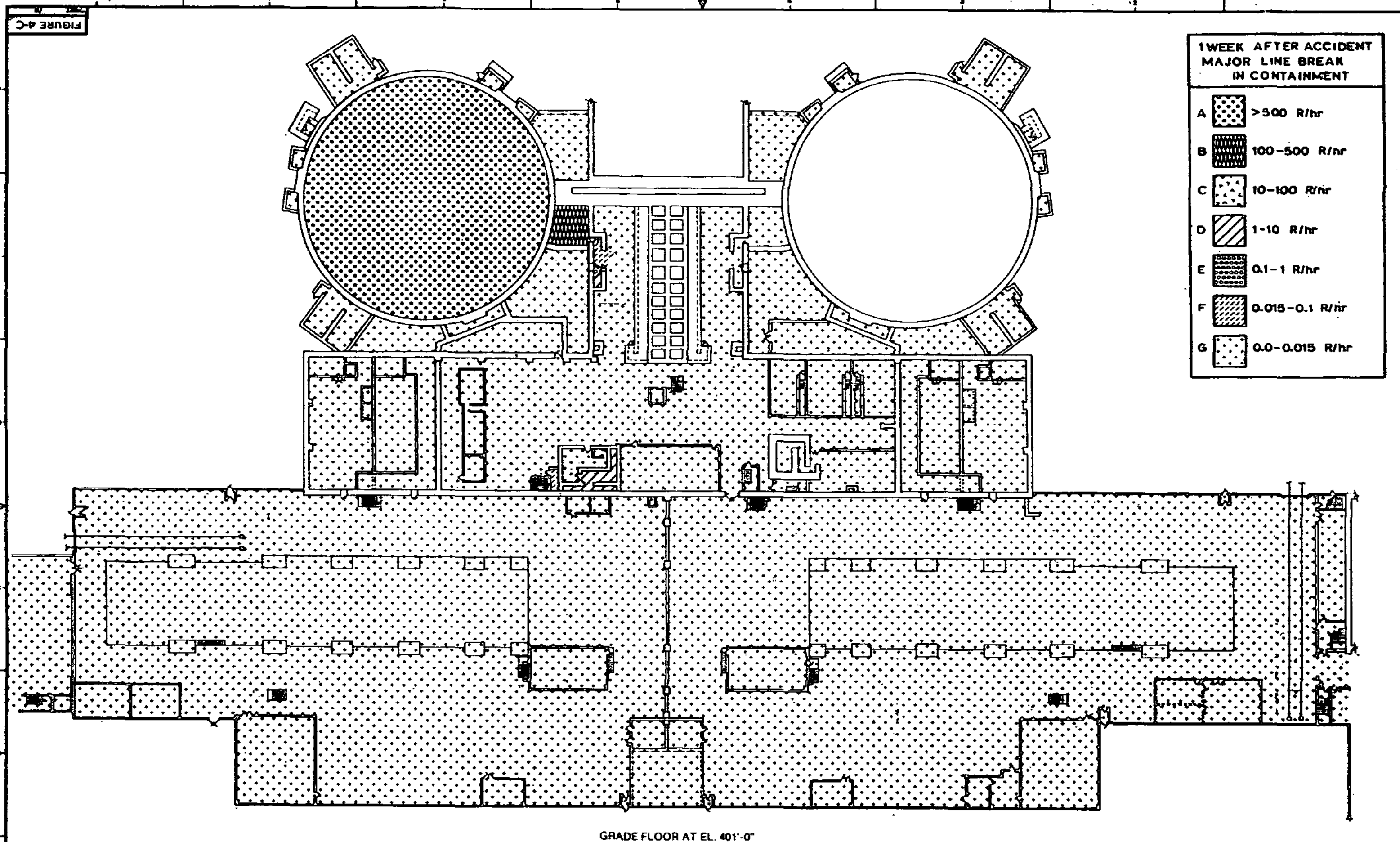
POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 3 OF 12)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-4

POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 4 OF 12)



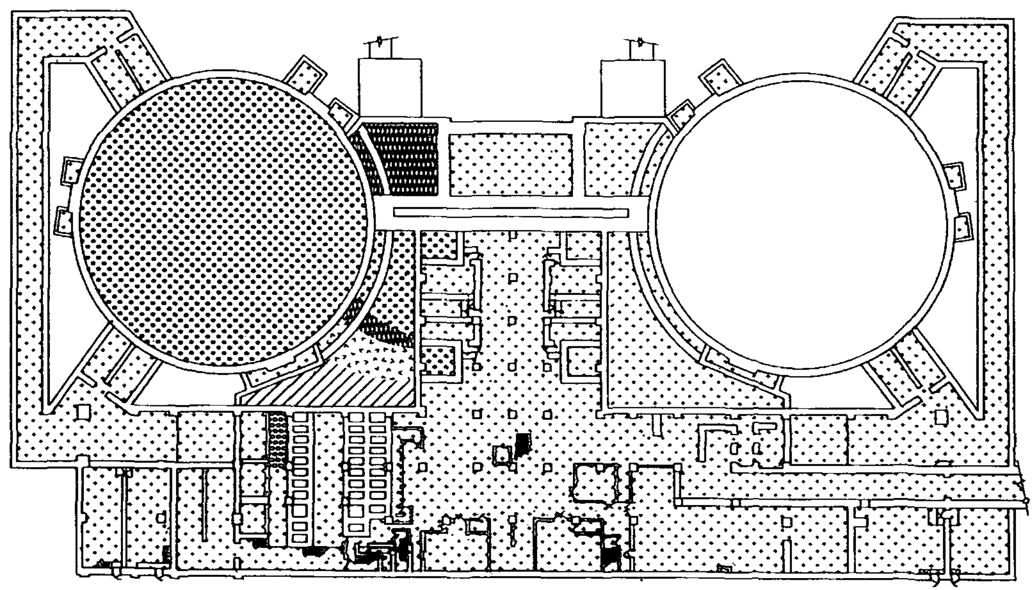
NOTE: SEE NOTE ON FIGURE 2-C.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-4

POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 5 OF 12)

REVISION 3
DECEMBER 1991



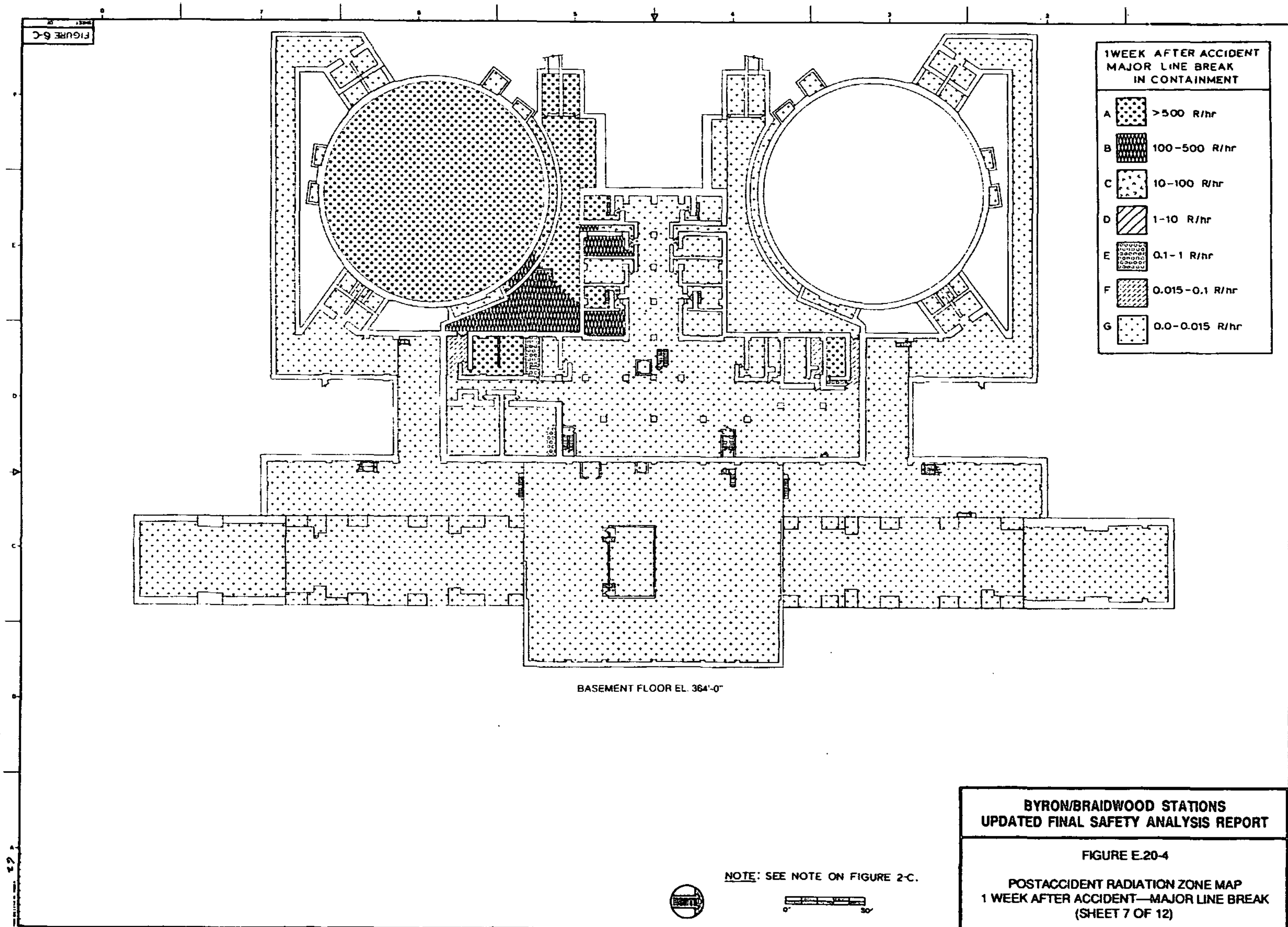
FLOOR PLAN EL. 383'-0"

NOTE: SEE NOTE ON FIGURE 2-C.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-4

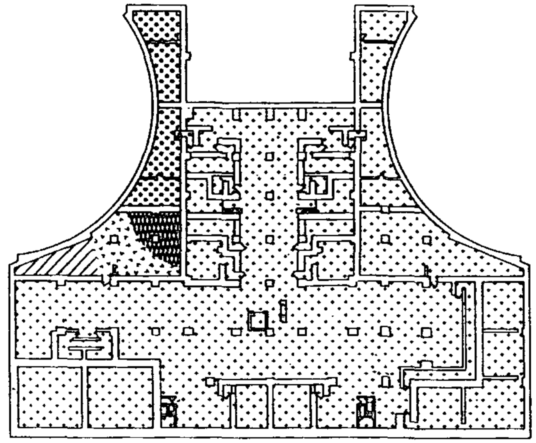
POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 6 OF 12)



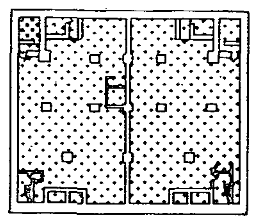
BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-4

POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 7 OF 12)



FLOOR PLAN EL. 346'-0"



ELEV 330'-0"

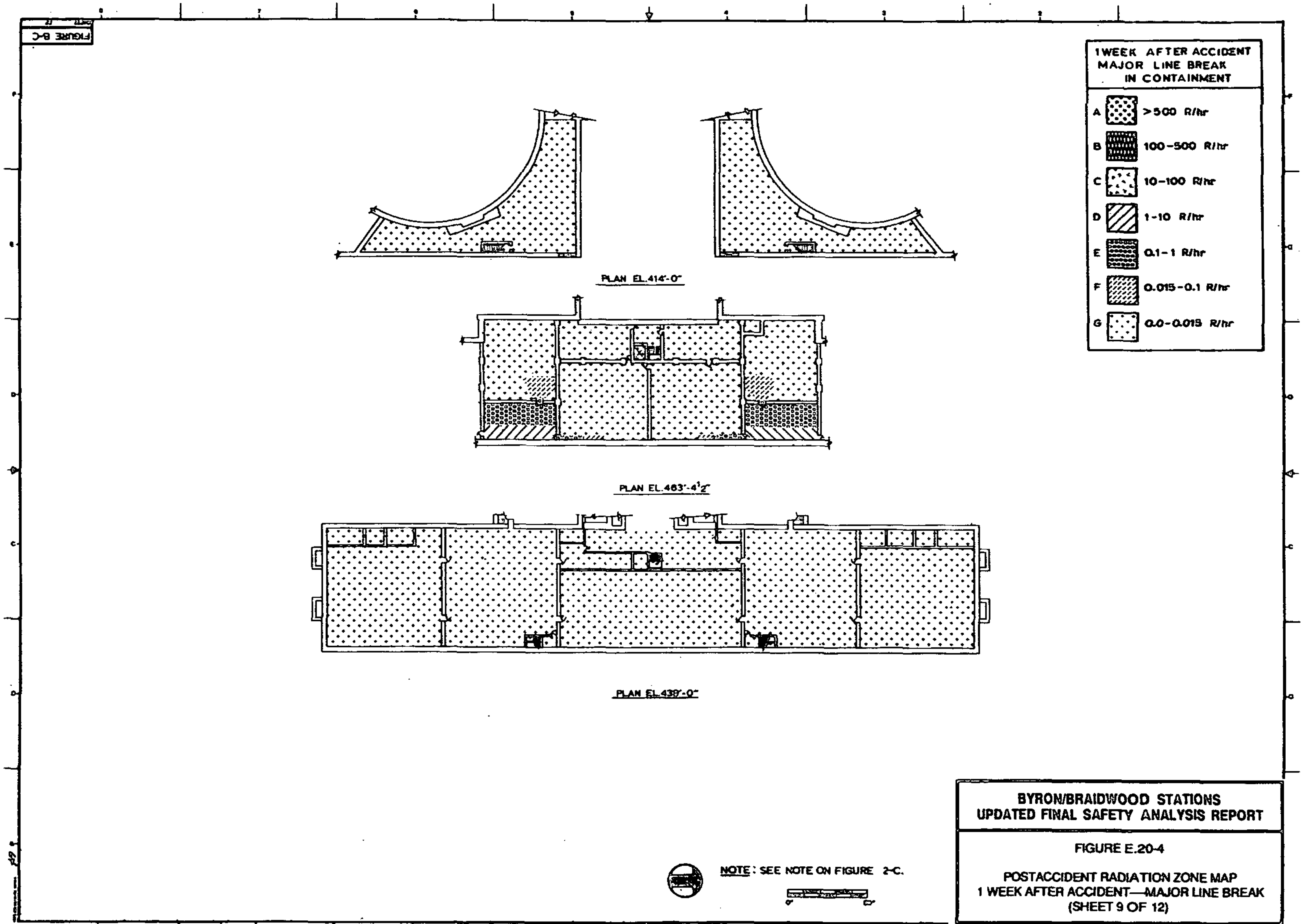
1WEEK AFTER ACCIDENT MAJOR LINE BREAK IN CONTAINMENT	
A	>500 R/hr
B	100-500 R/hr
C	10-100 R/hr
D	1-10 R/hr
E	0.1-1 R/hr
F	0.015-0.1 R/hr
G	0.0-0.015 R/hr

NOTE: SEE NOTE ON FIGURE 2-C.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-4

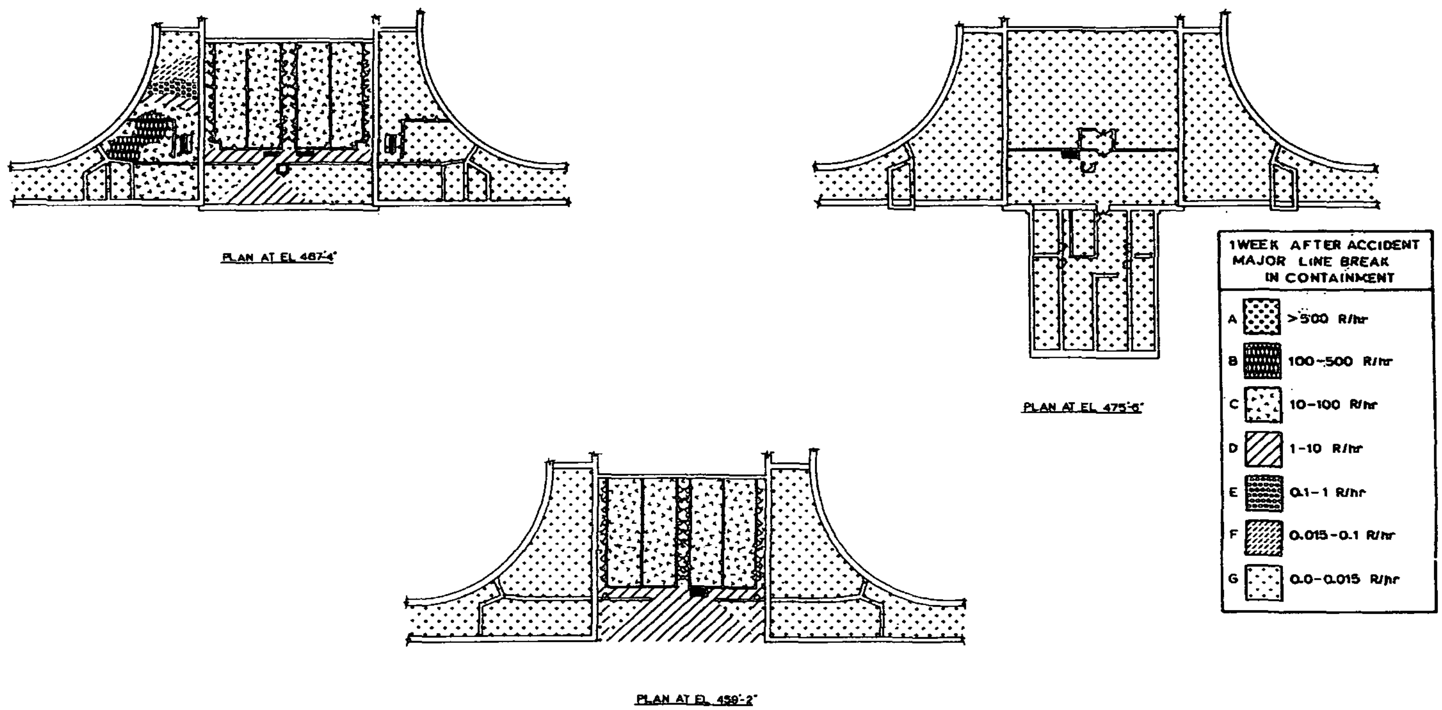
POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 8 OF 12)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-4

POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 9 OF 12)

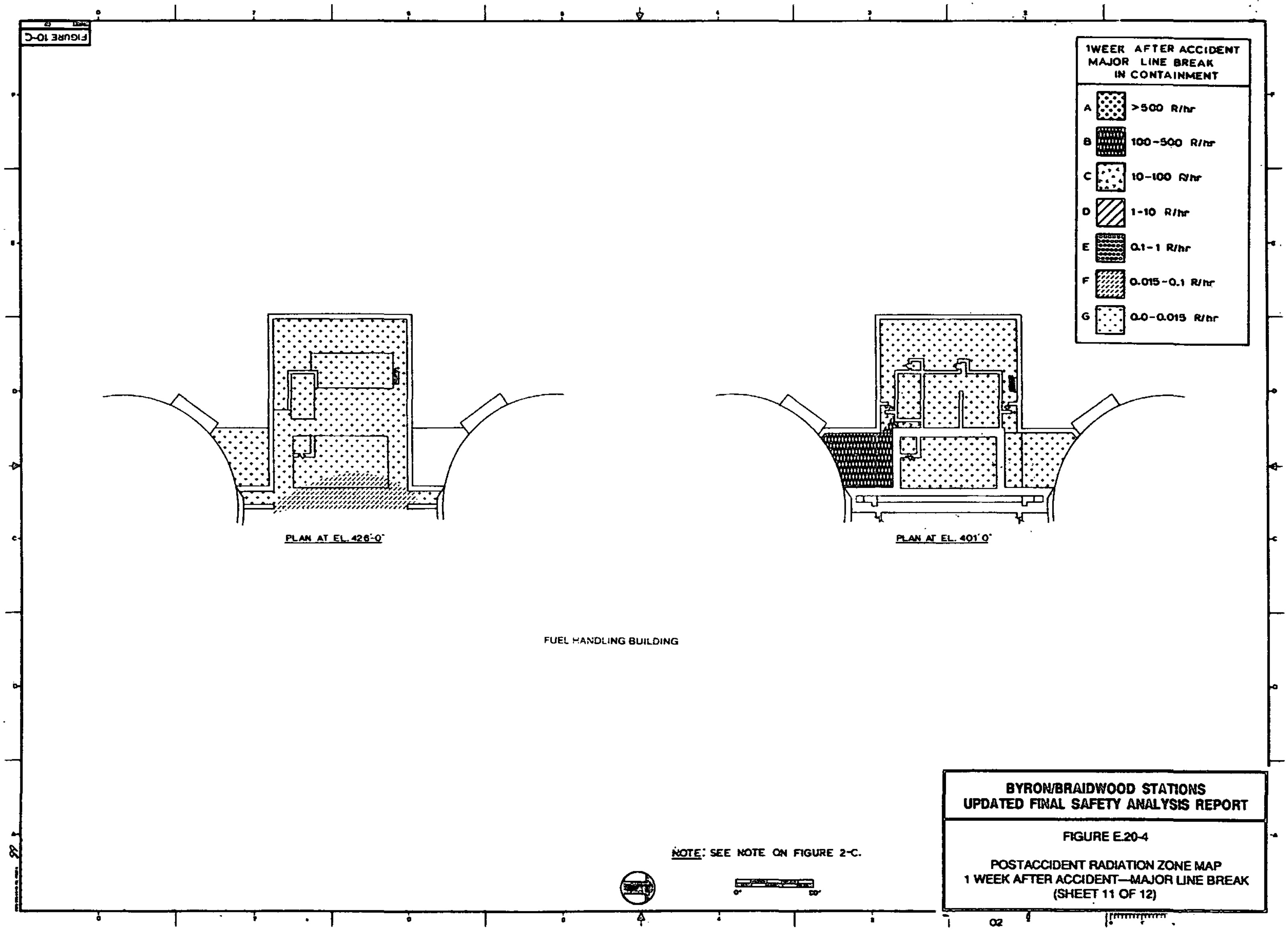


NOTE: SEE NOTE ON FIGURE 2-C.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-4

POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 10 OF 12)

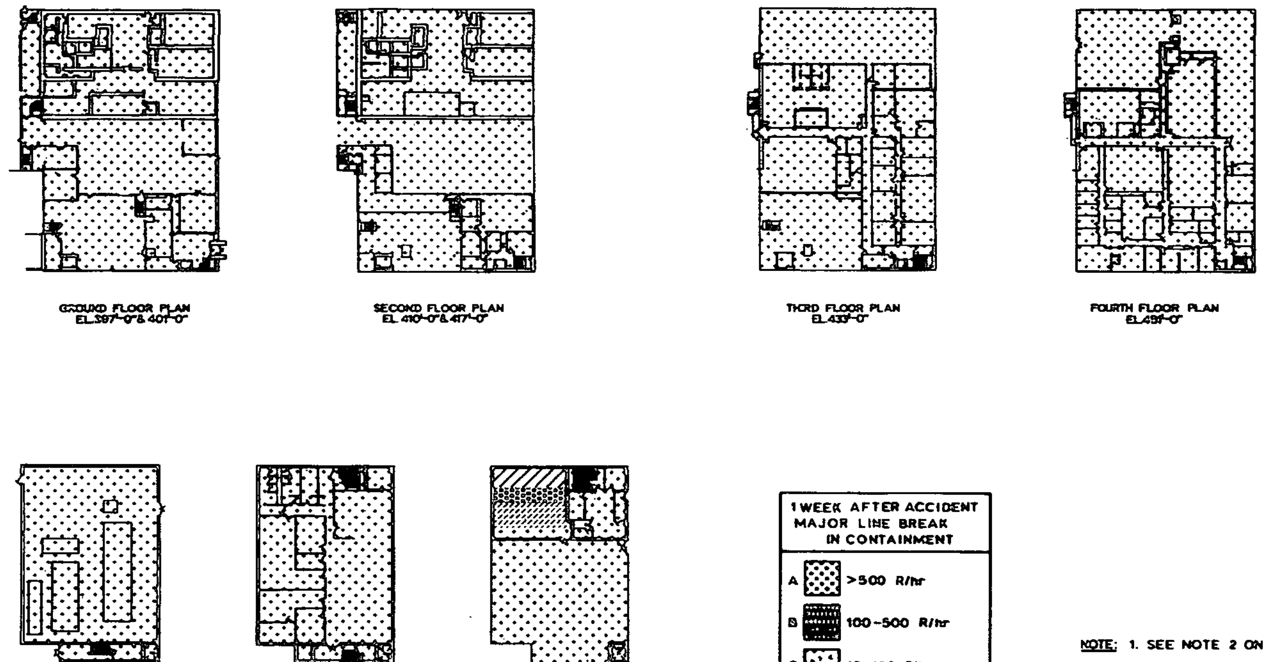


NOTE: SEE NOTE ON FIGURE 2-C.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-4

POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 11 OF 12)



1 WEEK AFTER ACCIDENT MAJOR LINE BREAK IN CONTAINMENT

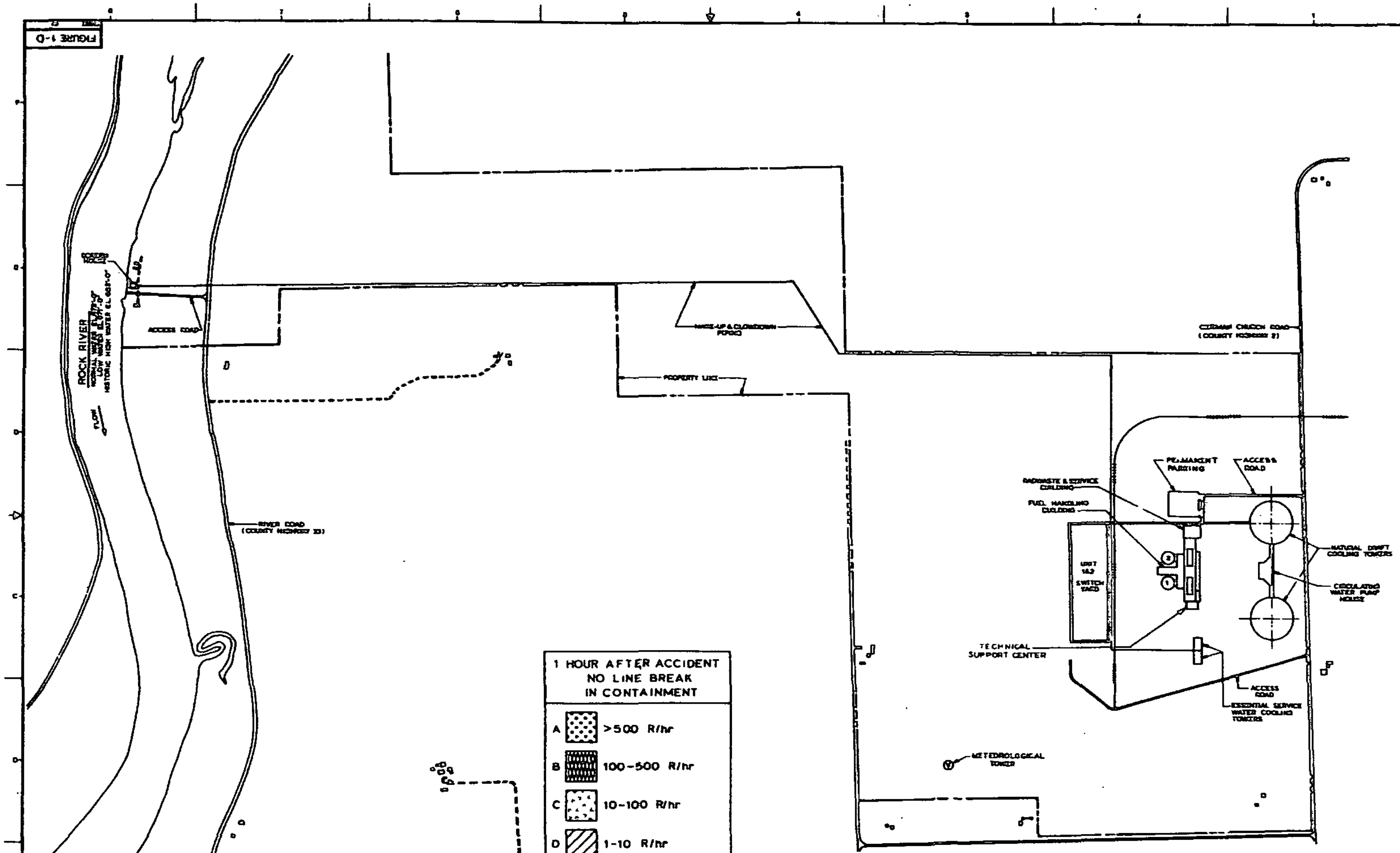
A	>500 R/hr
B	100-500 R/hr
C	10-100 R/hr
D	1-10 R/hr
E	0.1-1 R/hr
F	0.015-0.1 R/hr
G	0.0-0.015 R/hr

NOTE: 1. SEE NOTE 2 ON FIGURE 2-C

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-4

POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—MAJOR LINE BREAK
(SHEET 12 OF 12)



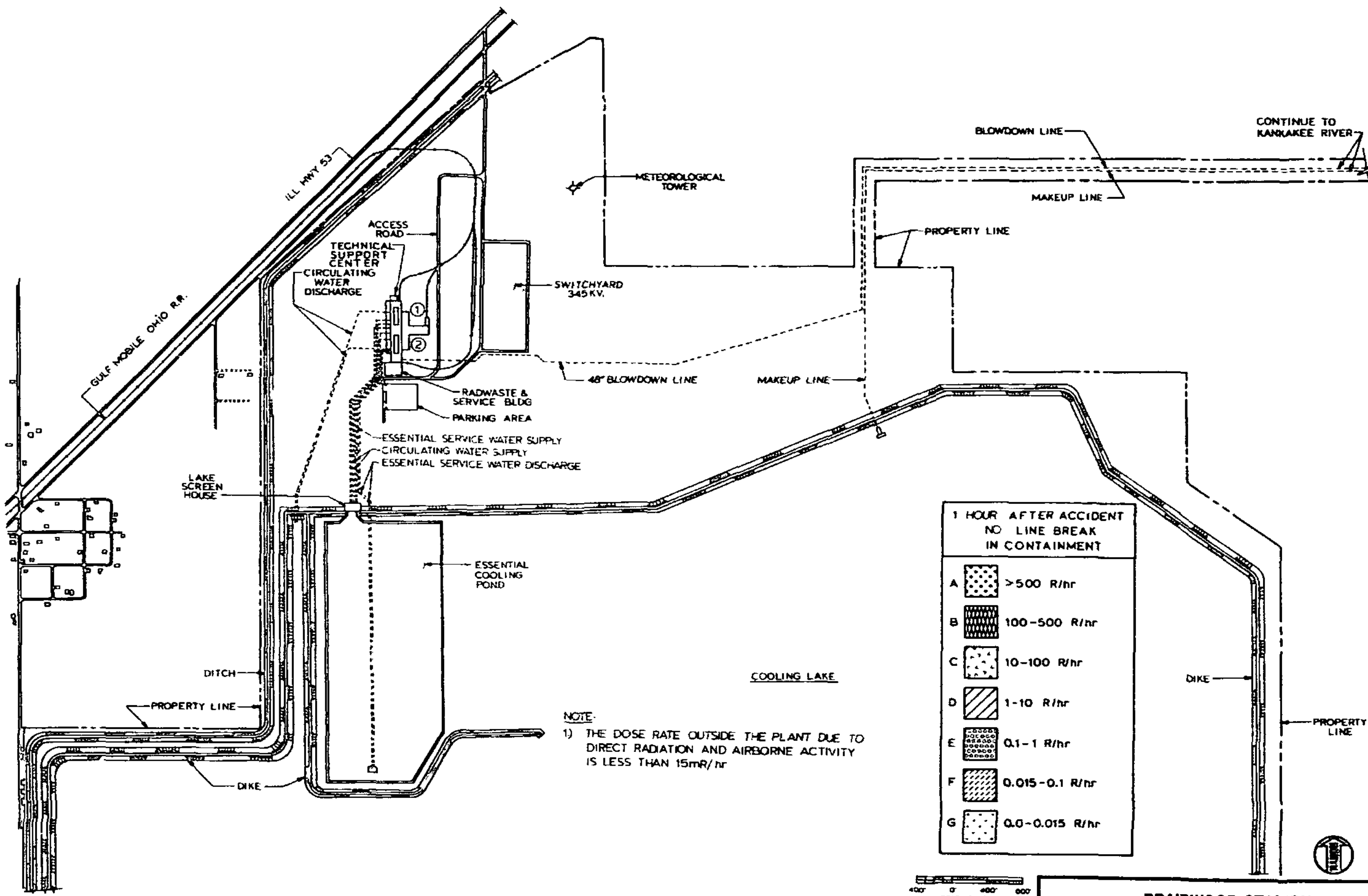
	1 HOUR AFTER ACCIDENT NO LINE BREAK IN CONTAINMENT
A	>500 R/hr
B	100–500 R/hr
C	10–100 R/hr
D	1–10 R/hr
E	0.1–1 R/hr
F	0.015–0.1 R/hr
G	0.0–0.015 R/hr

NOTE: THE DOSE RATE OUTSIDE THE PLANT DUE TO DIRECT RADIATION AND AIRBORNE ACTIVITY IS LESS THAN 15mR/hr

BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-5

POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—NO LINE BREAK
(SHEET 1 OF 12)

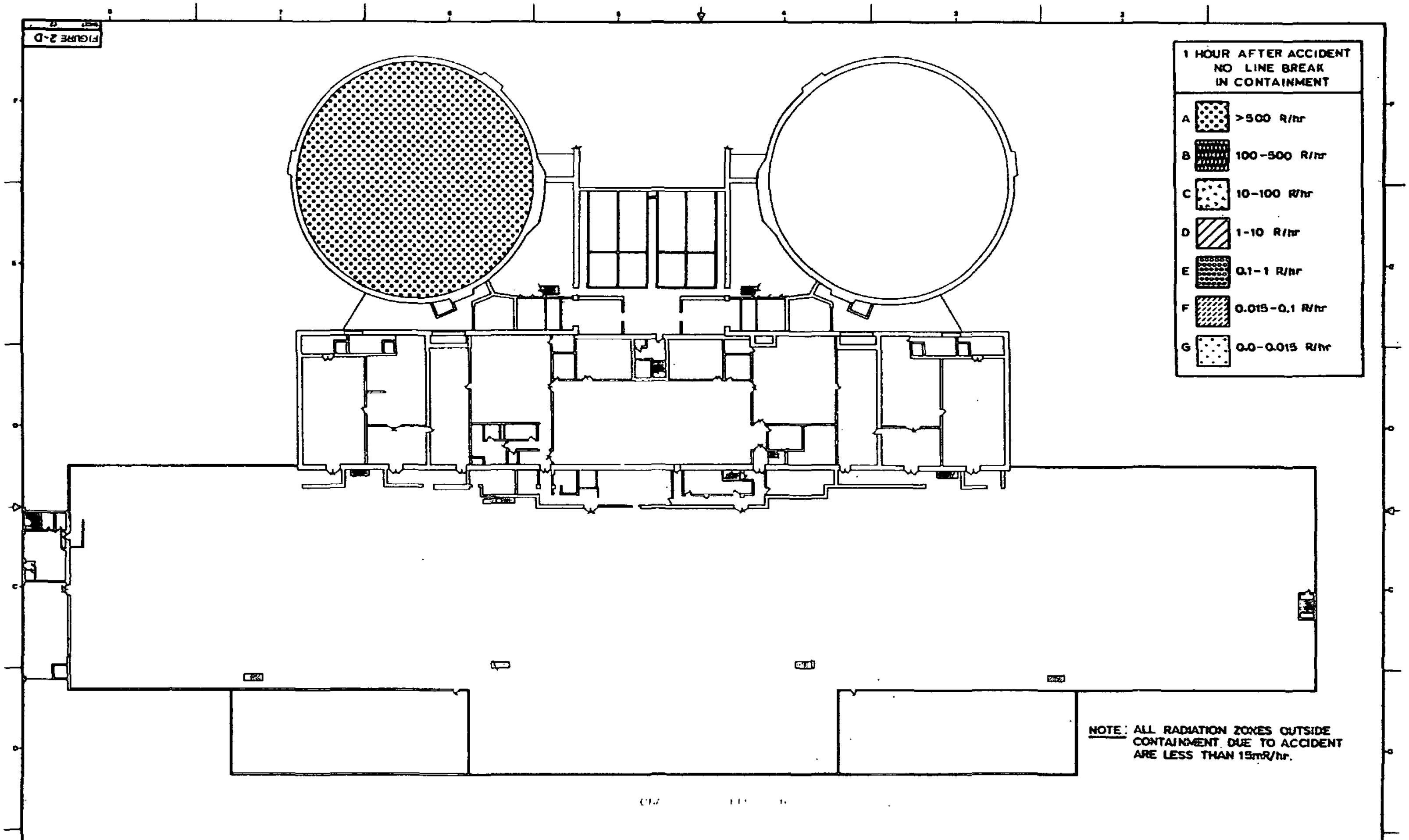


NOTE:
1) THE DOSE RATE OUTSIDE THE PLANT DUE TO DIRECT RADIATION AND AIRBORNE ACTIVITY IS LESS THAN 15mR/hr

BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-5

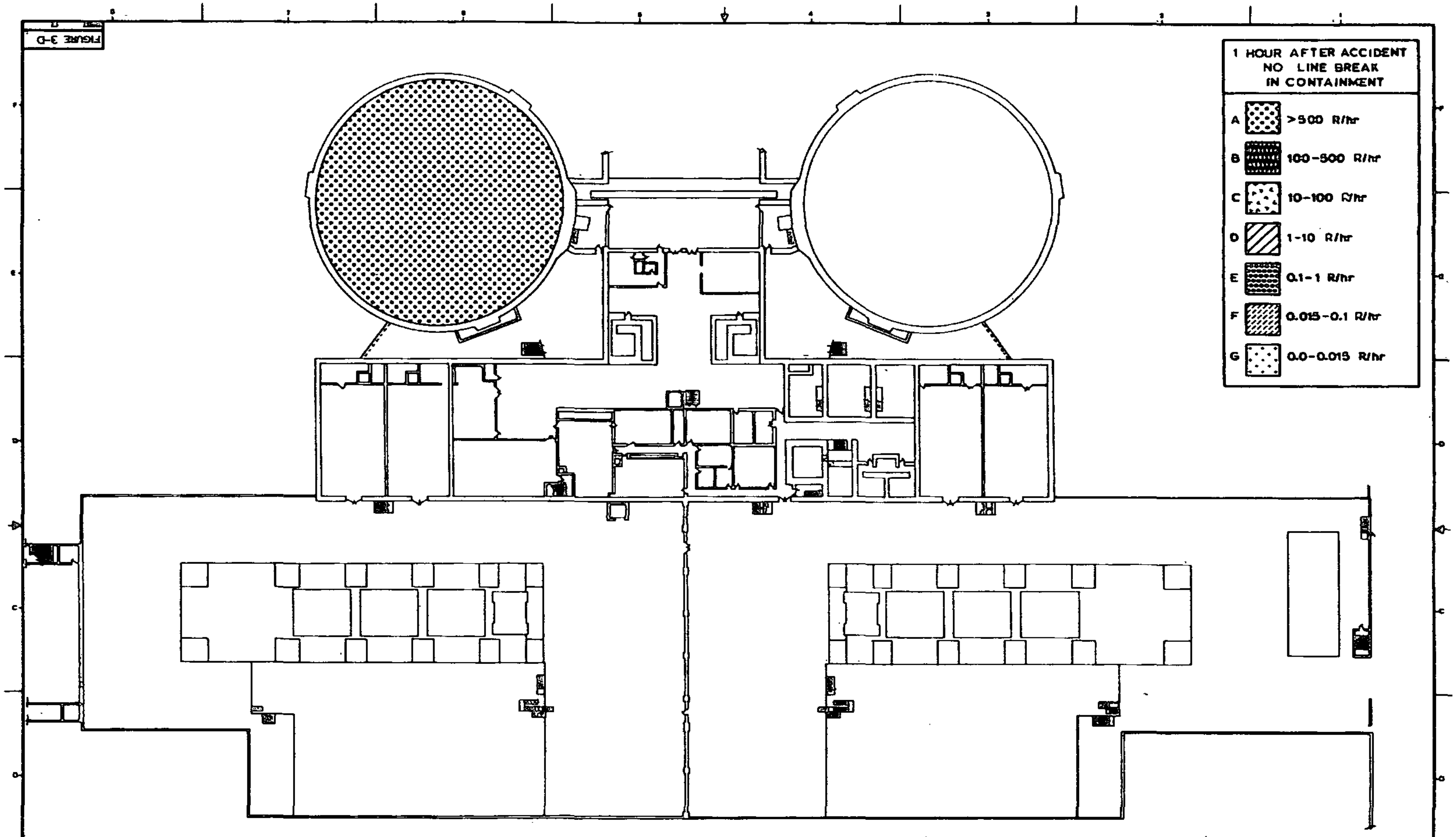
POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—NO LINE BREAK
(SHEET 2 OF 12)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-5

POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—NO LINE BREAK
(SHEET 3 OF 12)

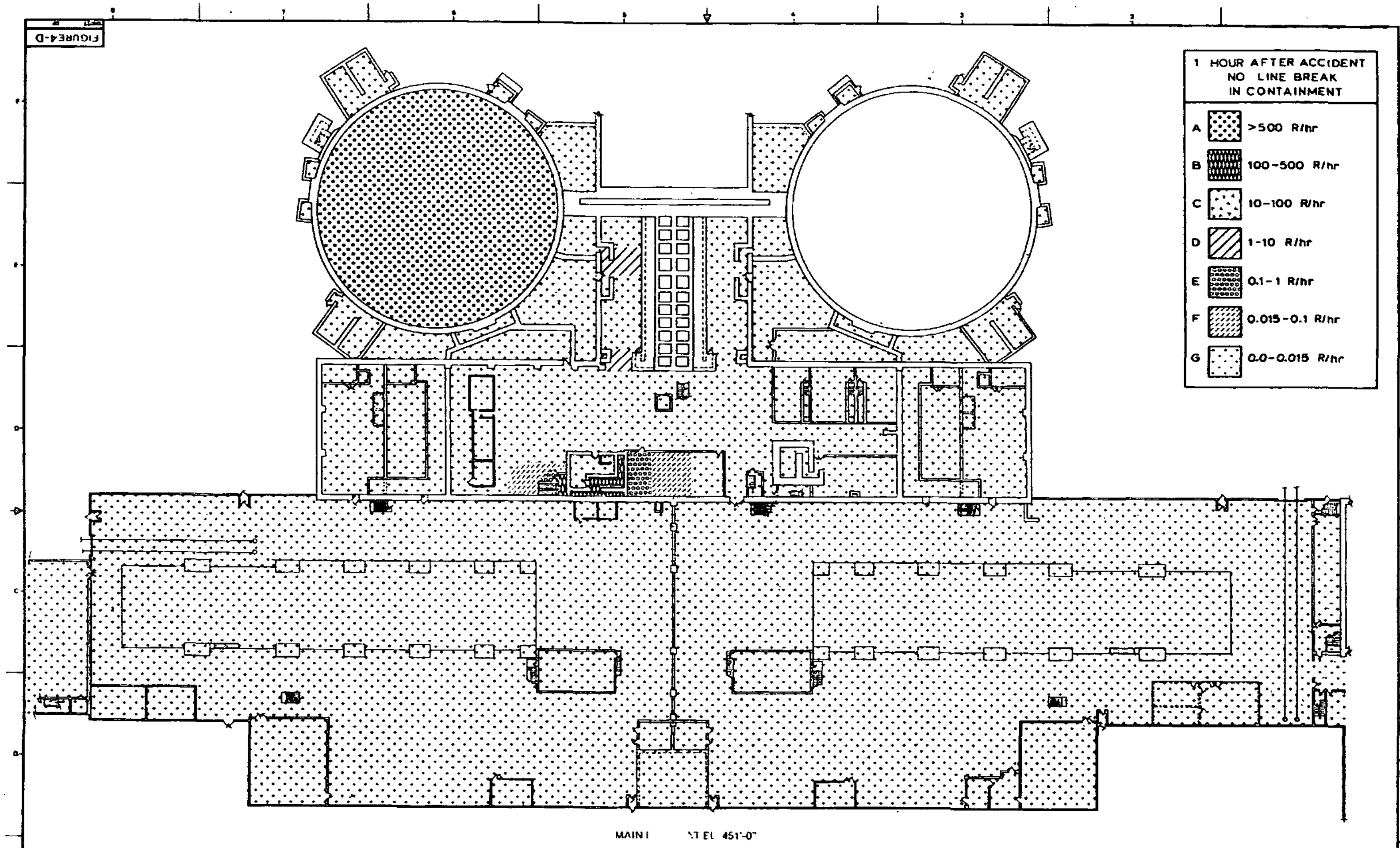


NOTE: ALL RADIATION ZONES OUTSIDE CONTAINMENT DUE TO THE ACCIDENT ARE LESS THAN 15mR/hr.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-5

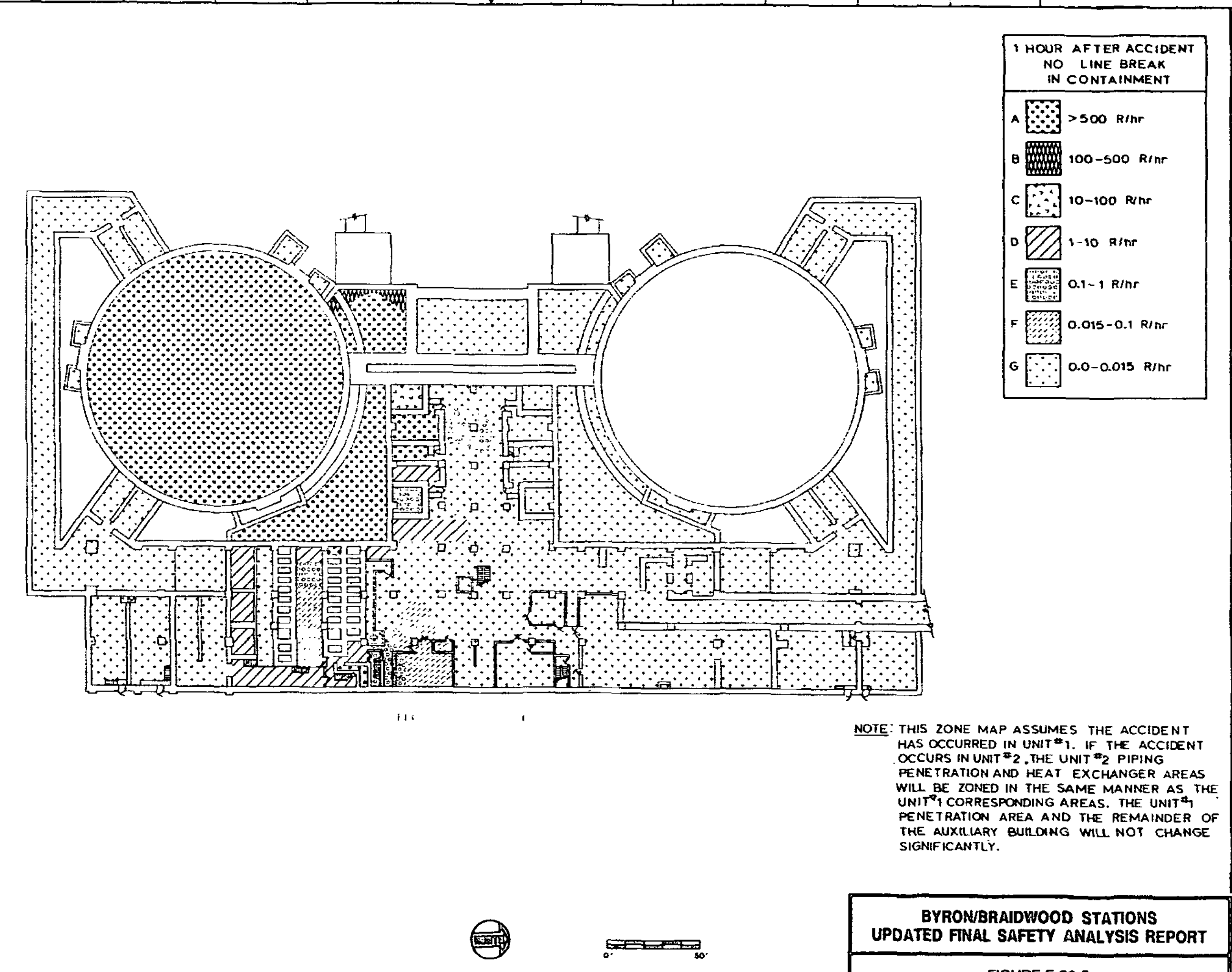
POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—NO LINE BREAK
(SHEET 4 OF 12)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-5

POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—NO LINE BREAK
(SHEET 5 OF 12)

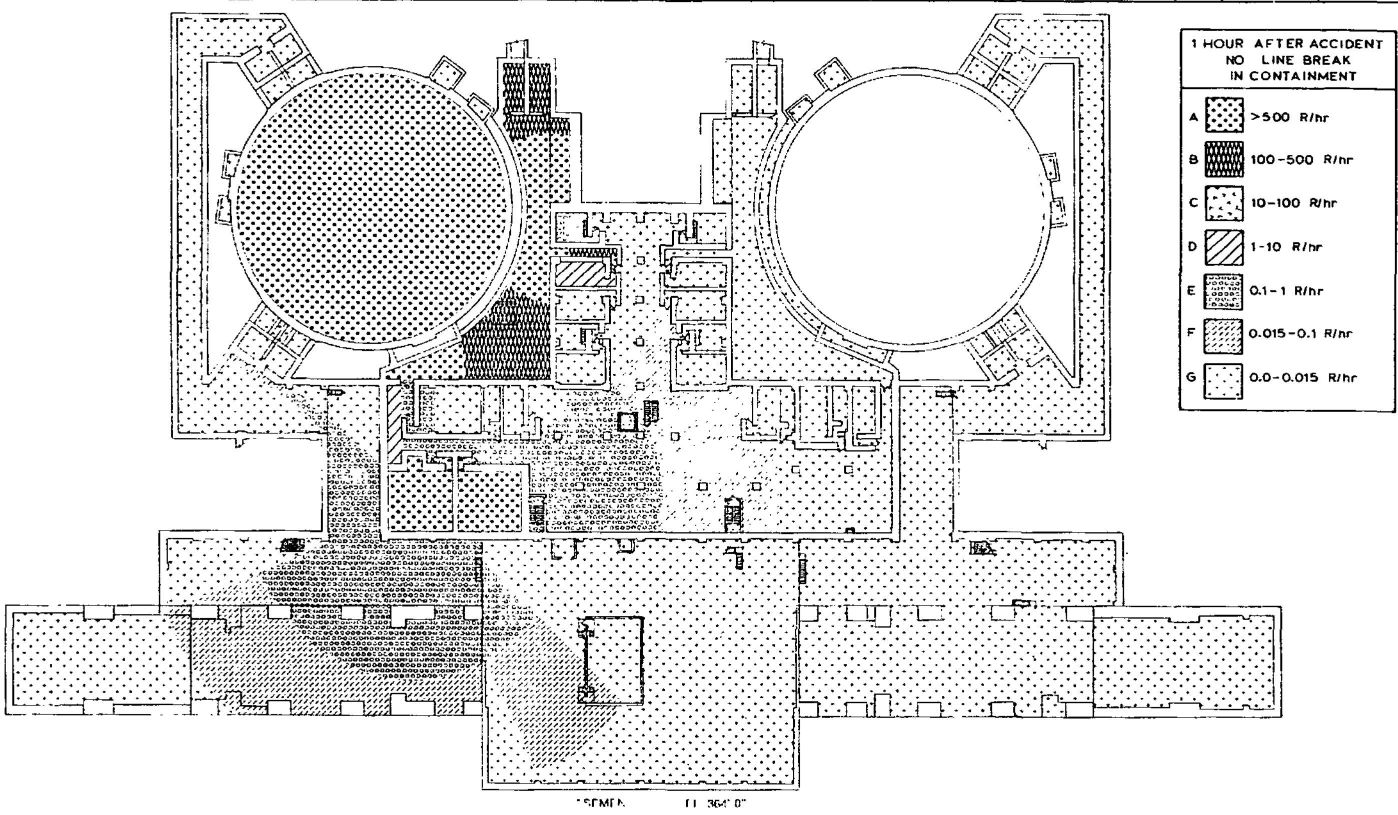


NOTE: THIS ZONE MAP ASSUMES THE ACCIDENT HAS OCCURRED IN UNIT #1. IF THE ACCIDENT OCCURS IN UNIT #2, THE UNIT #2 PIPING PENETRATION AND HEAT EXCHANGER AREAS WILL BE ZONED IN THE SAME MANNER AS THE UNIT #1 CORRESPONDING AREAS. THE UNIT #1 PENETRATION AREA AND THE REMAINDER OF THE AUXILIARY BUILDING WILL NOT CHANGE SIGNIFICANTLY.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-5

POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—NO LINE BREAK
(SHEET 6 OF 12)

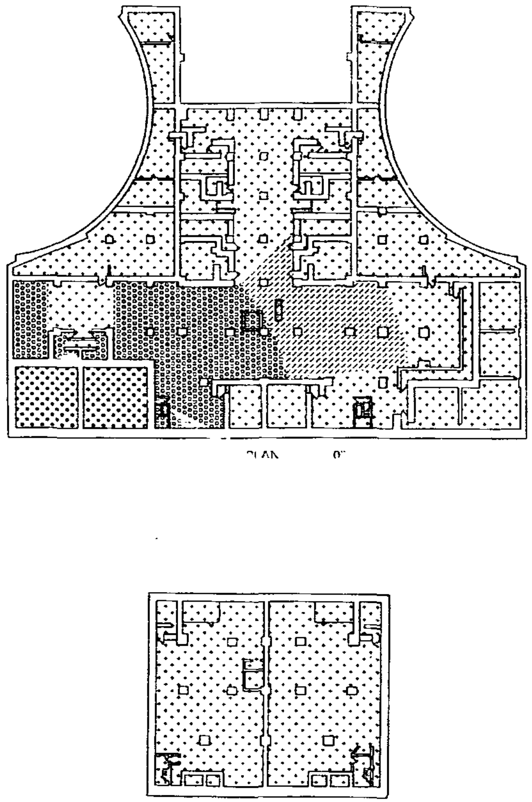


NOTE: SEE NOTE ON FIGURE 5-D.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-5

POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—NO LINE BREAK
(SHEET 7 OF 12)



ELEV 330'-0"

1 HOUR AFTER ACCIDENT NO LINE BREAK IN CONTAINMENT		
A		>500 R/hr
B		100-500 R/hr
C		10-100 R/hr
D		1-10 R/hr
E		0.1-1 R/hr
F		0.015-0.1 R/hr
G		0.0-0.015 R/hr

NOTE: SEE NOTE ON FIGURE 5-D

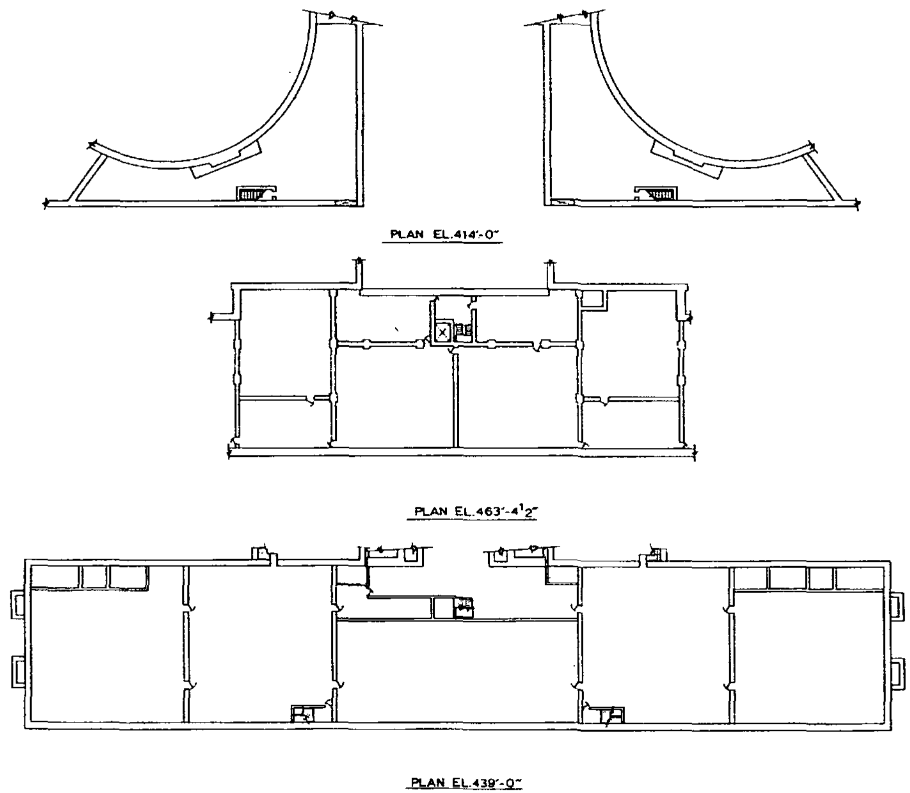
BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-5

POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—NO LINE BREAK
(SHEET 8 OF 12)

REVISION 3
DECEMBER 1991

1 HOUR AFTER ACCIDENT NO LINE BREAK IN CONTAINMENT	
A	>500 R/hr
B	100-500 R/hr
C	10-100 R/hr
D	1-10 R/hr
E	0.1-1 R/hr
F	0.015-0.1 R/hr
G	0.0-0.015 R/hr

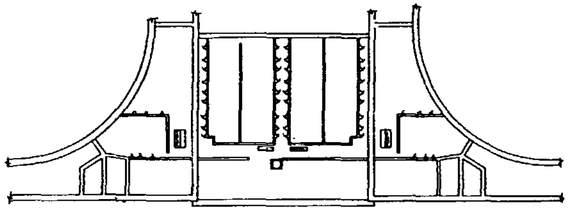


NOTE: ALL RADIATION ZONES DUE TO THE ACCIDENT ARE LESS THAN 15mR/hr.

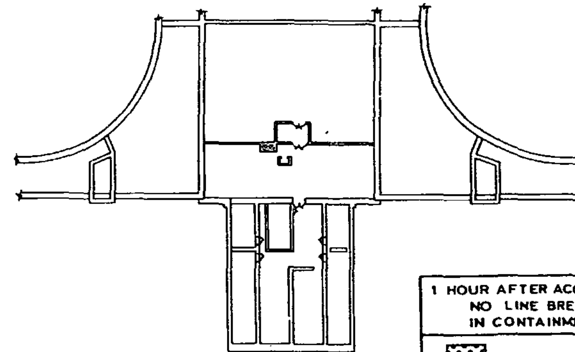
BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-5

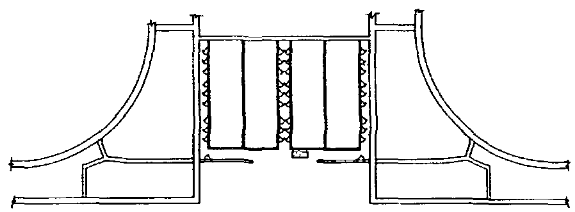
POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—NO LINE BREAK
(SHEET 9 OF 12)



PLAN AT EL 467'-4"



PLAN AT EL 475'-6"



PLAN AT EL 459'-2"

1 HOUR AFTER ACCIDENT NO LINE BREAK IN CONTAINMENT	
A	>500 R/hr
B	100-500 R/hr
C	10-100 R/hr
D	1-10 R/hr
E	0.1-1 R/hr
F	0.015-0.1 R/hr
G	0.0-0.015 R/hr

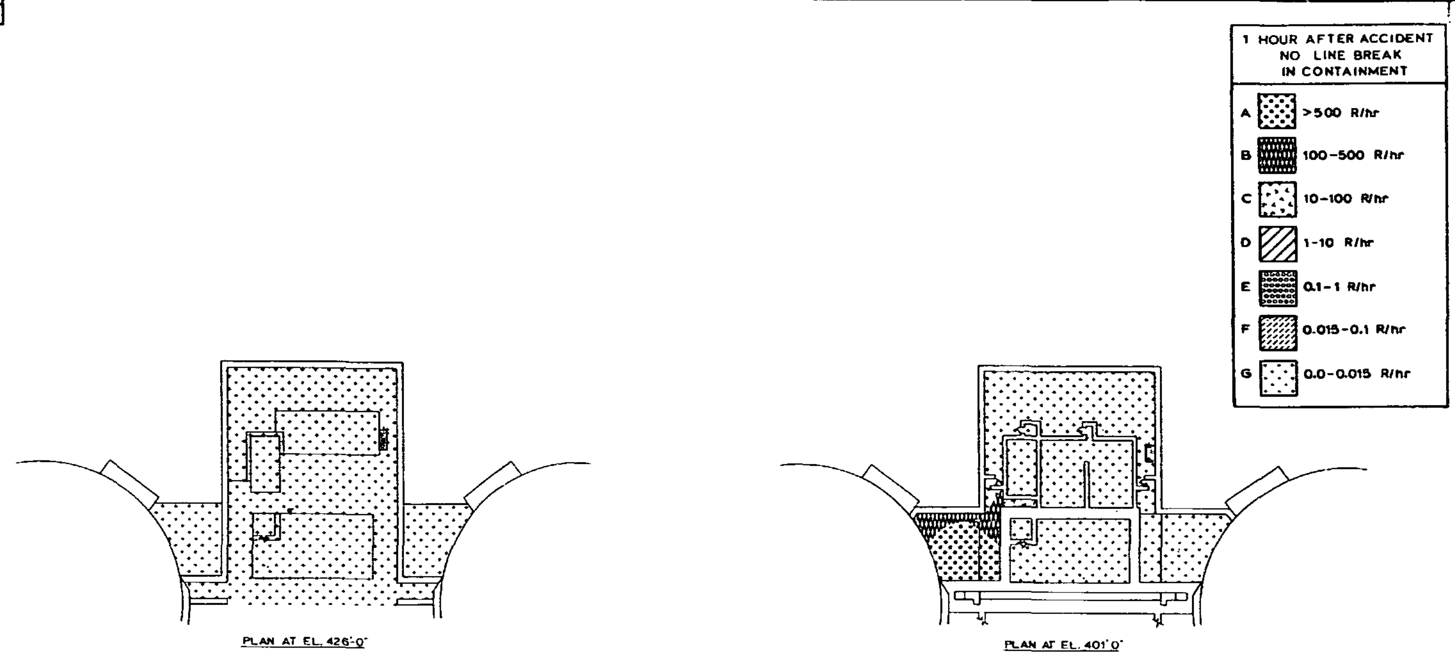
NOTE: ALL RADIATION ZONES DUE TO THE ACCIDENT ARE LESS THAN 15mR/hr.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-5

POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—NO LINE BREAK
(SHEET 10 OF 12)

02

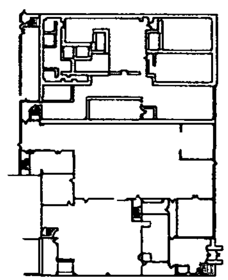


NOTE: THESE RADIATION ZONE MAPS ASSUME THAT THE ACCIDENT HAS OCCURRED IN UNIT # 1. IF THE ACCIDENT OCCURS IN UNIT # 2, THE RADIATION ZONES AROUND UNIT # 2 CONTAINMENT WILL BE THE MIRROR IMAGE OF THE ZONES SHOWN.

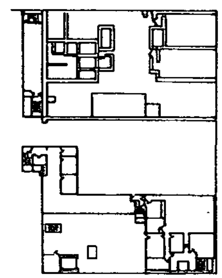
BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-5

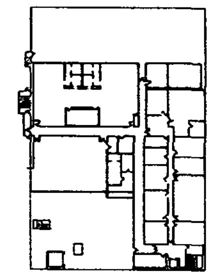
POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—NO LINE BREAK
(SHEET 11 OF 12)



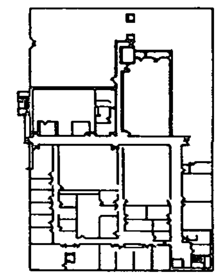
GROUND FLOOR PLAN
EL 397'-0" & 401'-0"



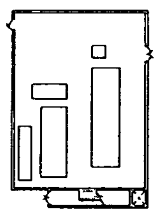
SECOND FLOOR PLAN
EL 410'-0" & 417'-0"



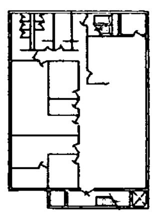
THIRD FLOOR PLAN
EL 433'-0"



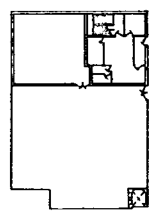
FOURTH FLOOR PLAN
EL 451'-0"



CONDENSATE CLEAN-UP BLDG.
EL 401'-0"



TECHNICAL SUPPORT CENTER
EL. 435'-0"



TECHNICAL SUPPORT CENTER
EL. 451'-0"

0' 30'

1 HOUR AFTER ACCIDENT NO LINE BREAK IN CONTAINMENT

A	>500 R/hr
B	100-500 R/hr
C	10-100 R/hr
D	1-10 R/hr
E	0.1-1 R/hr
F	0.015-0.1 R/hr
G	0.0-0.015 R/hr

NOTE: ALL RADIATION ZONES DUE TO THE ACCIDENT ARE LESS THAN 15mR/hr.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-5

POSTACCIDENT RADIATION ZONE MAP
1 HOUR AFTER ACCIDENT—NO LINE BREAK
(SHEET 12 OF 12)

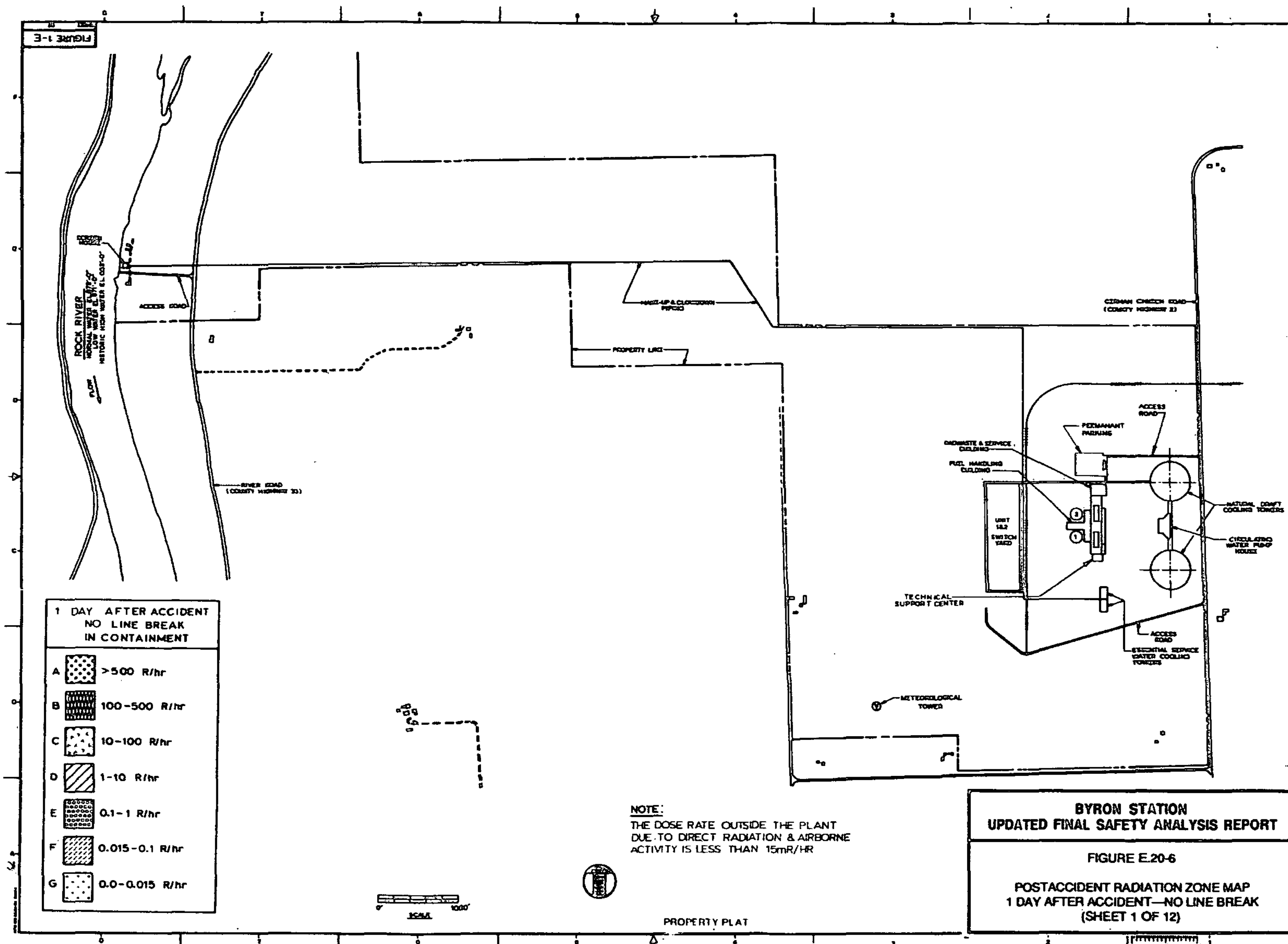
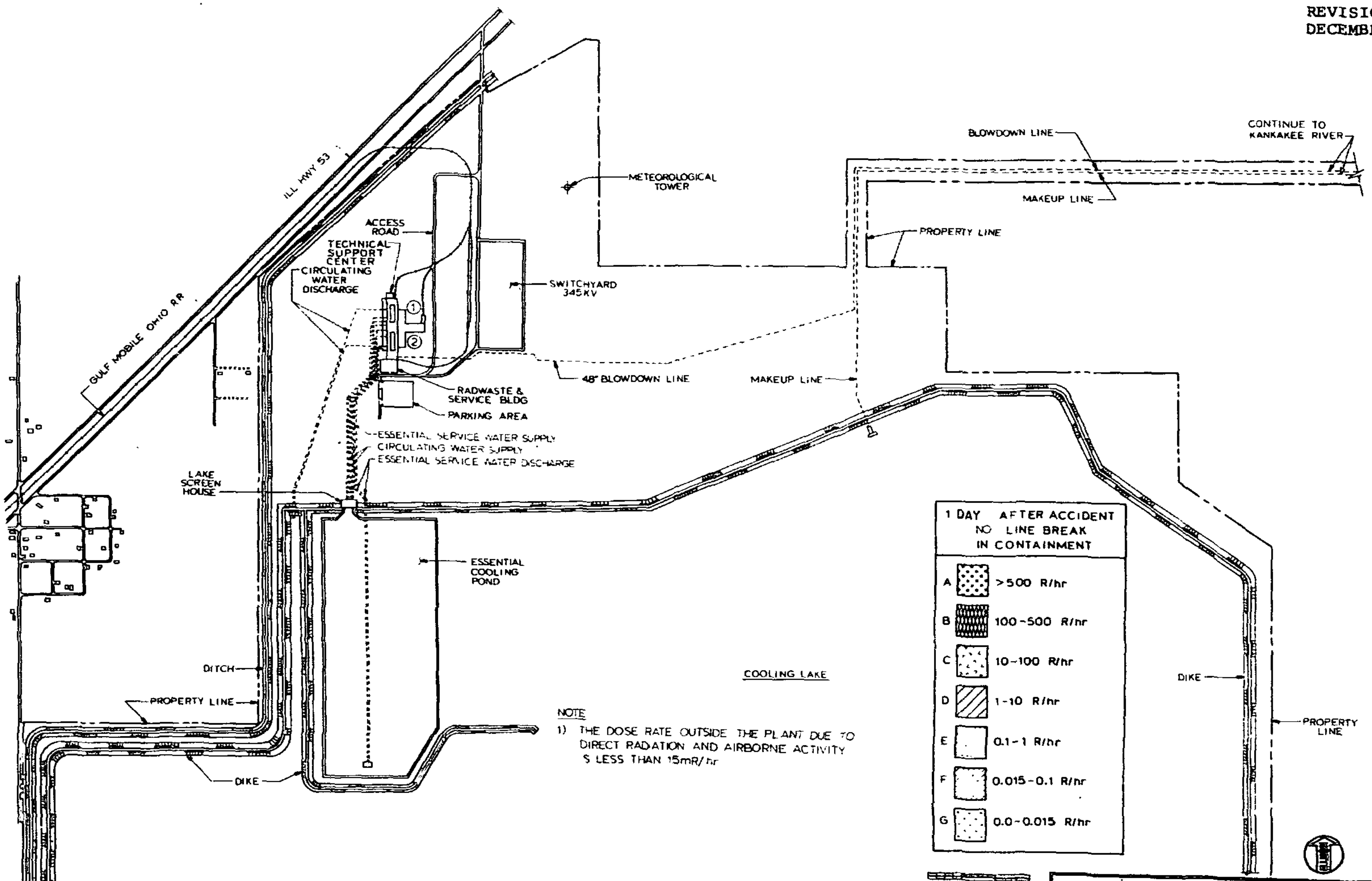


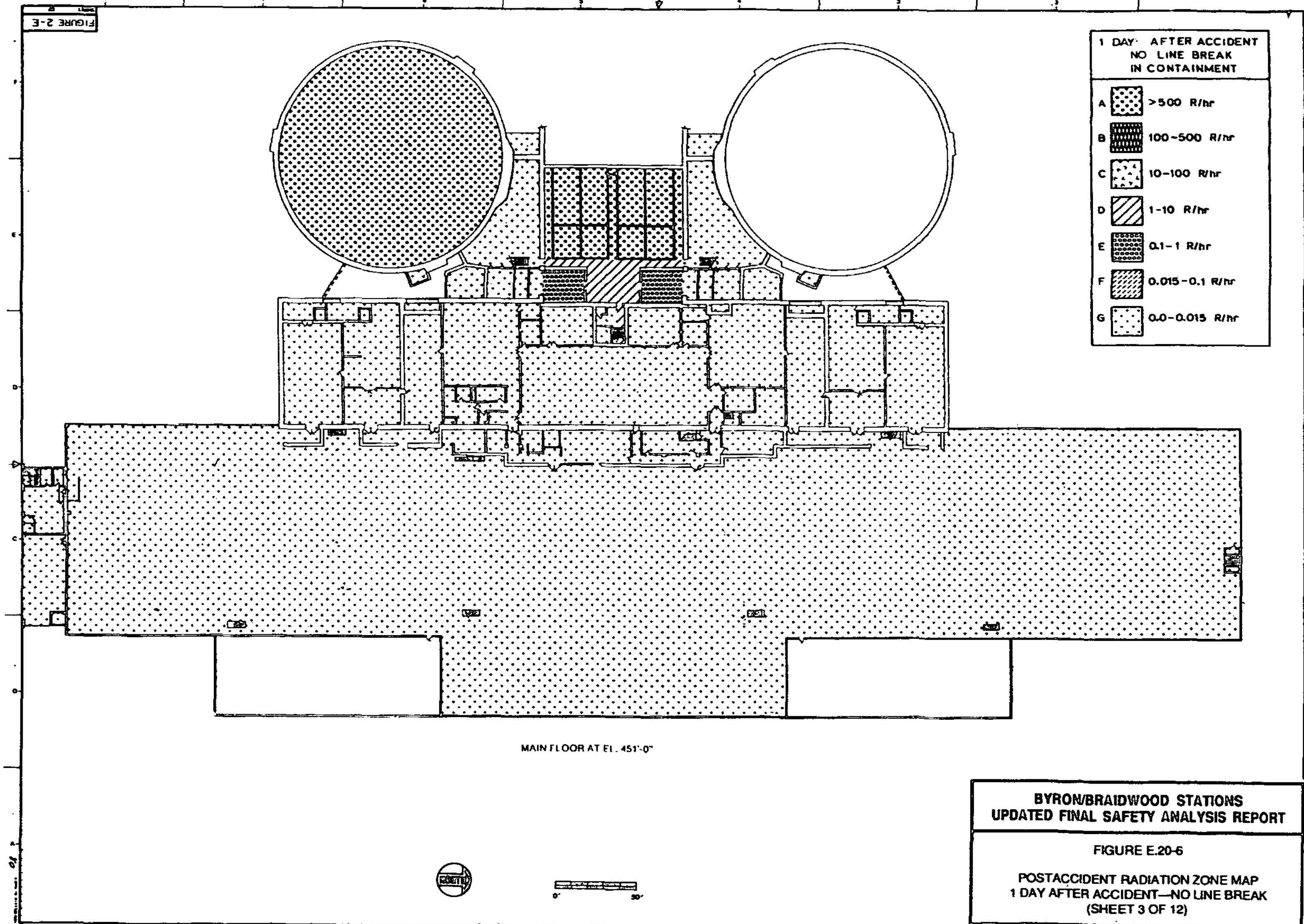
FIGURE E.20-6 POSTACCIDENT RADIATION ZONE MAP 1 DAY AFTER ACCIDENT—NO LINE BREAK (SHEET 1 OF 12)



BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-6

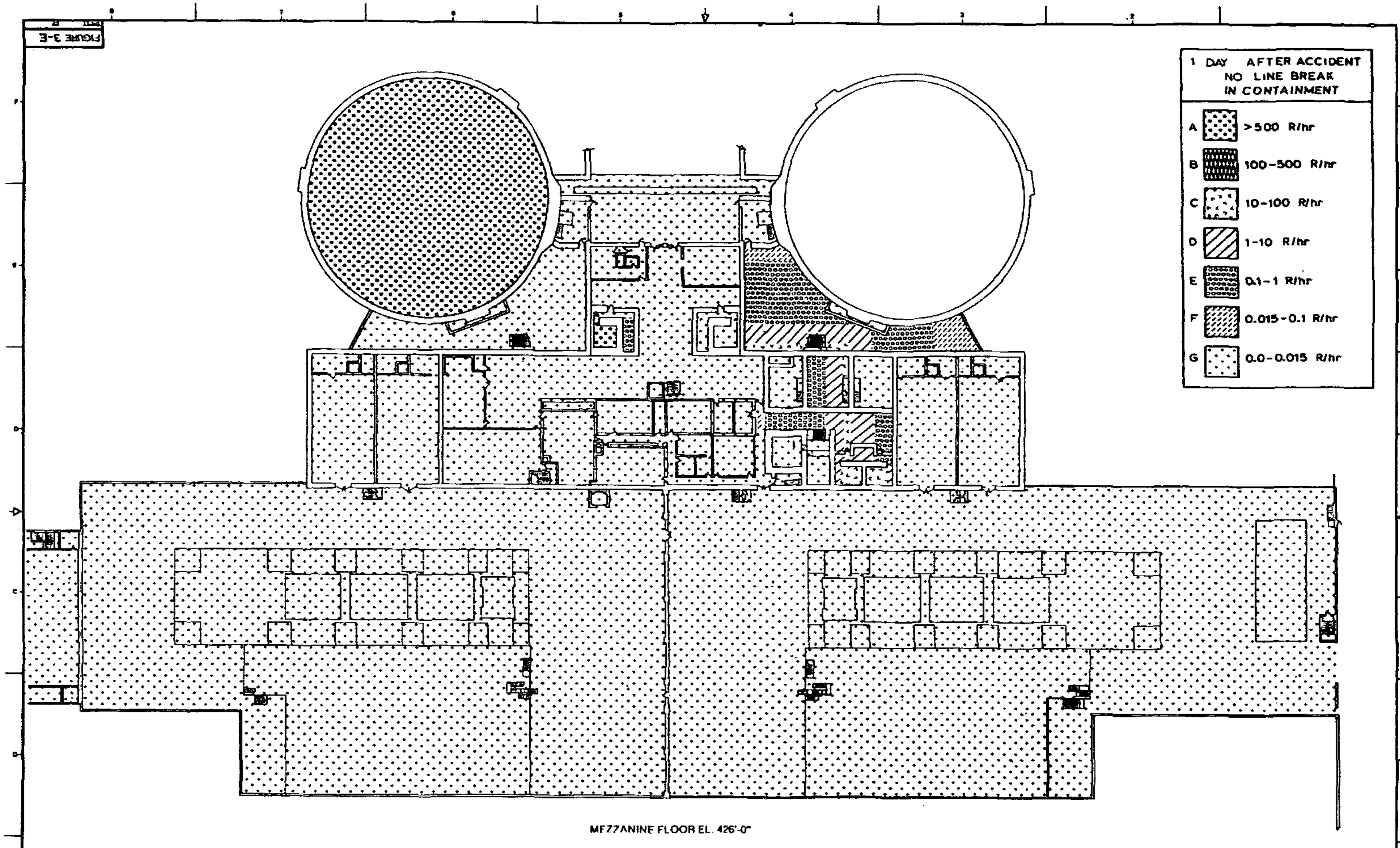
POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—NO LINE BREAK
(SHEET 2 OF 12)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-6

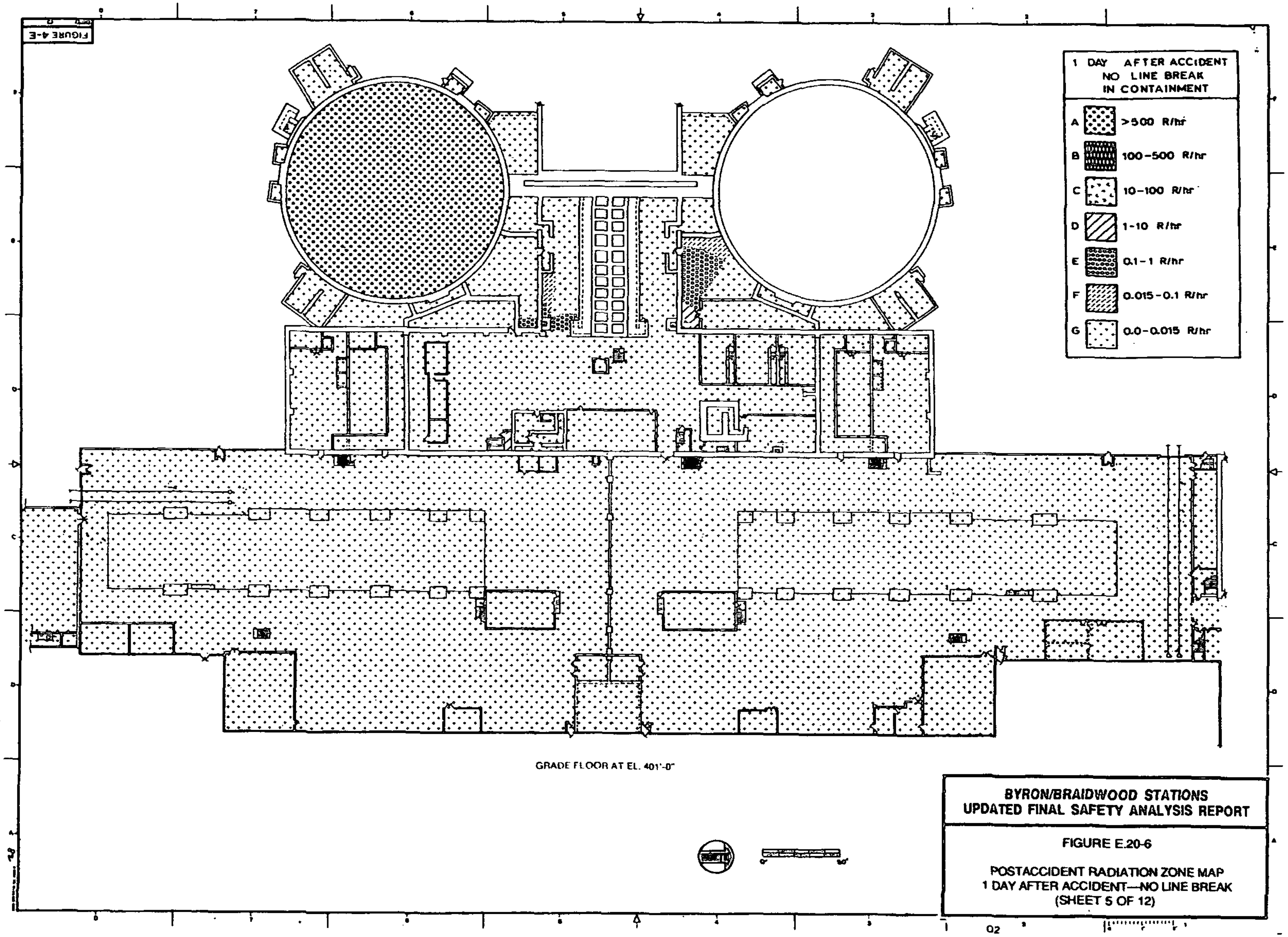
POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—NO LINE BREAK
(SHEET 3 OF 12)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-6

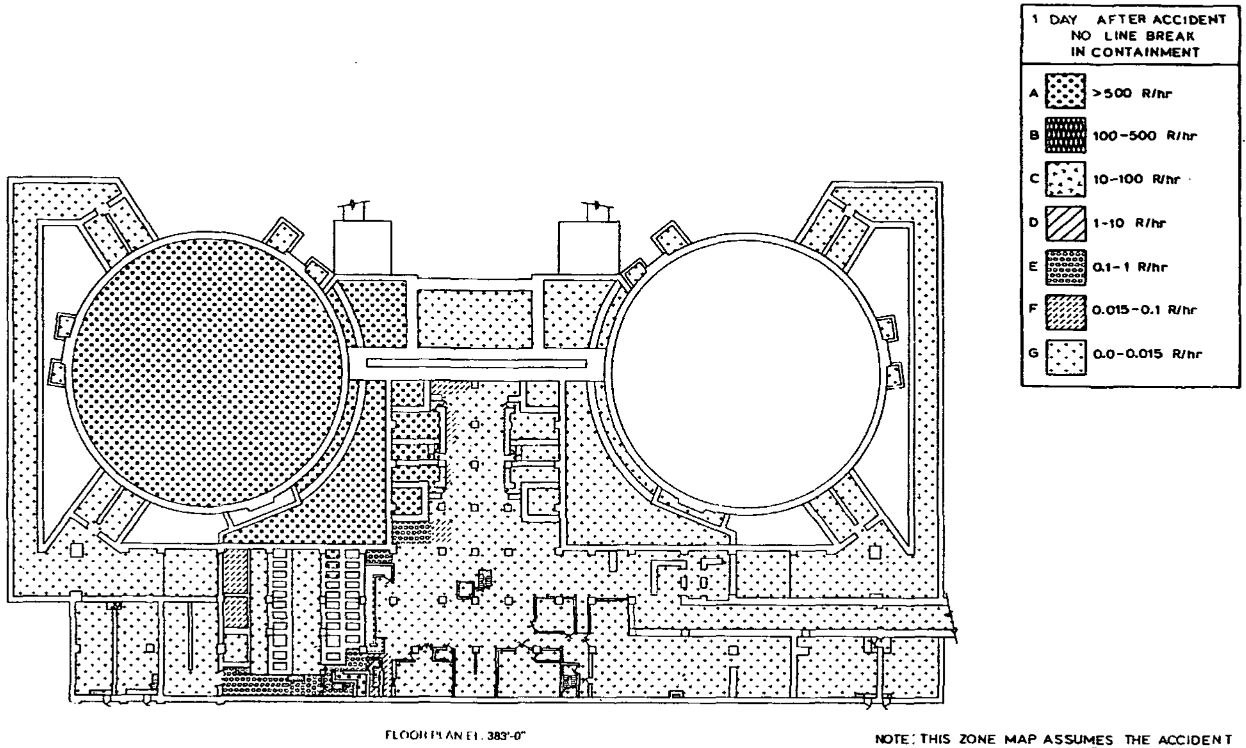
POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—NO LINE BREAK
(SHEET 4 OF 12)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-6

POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—NO LINE BREAK
(SHEET 5 OF 12)

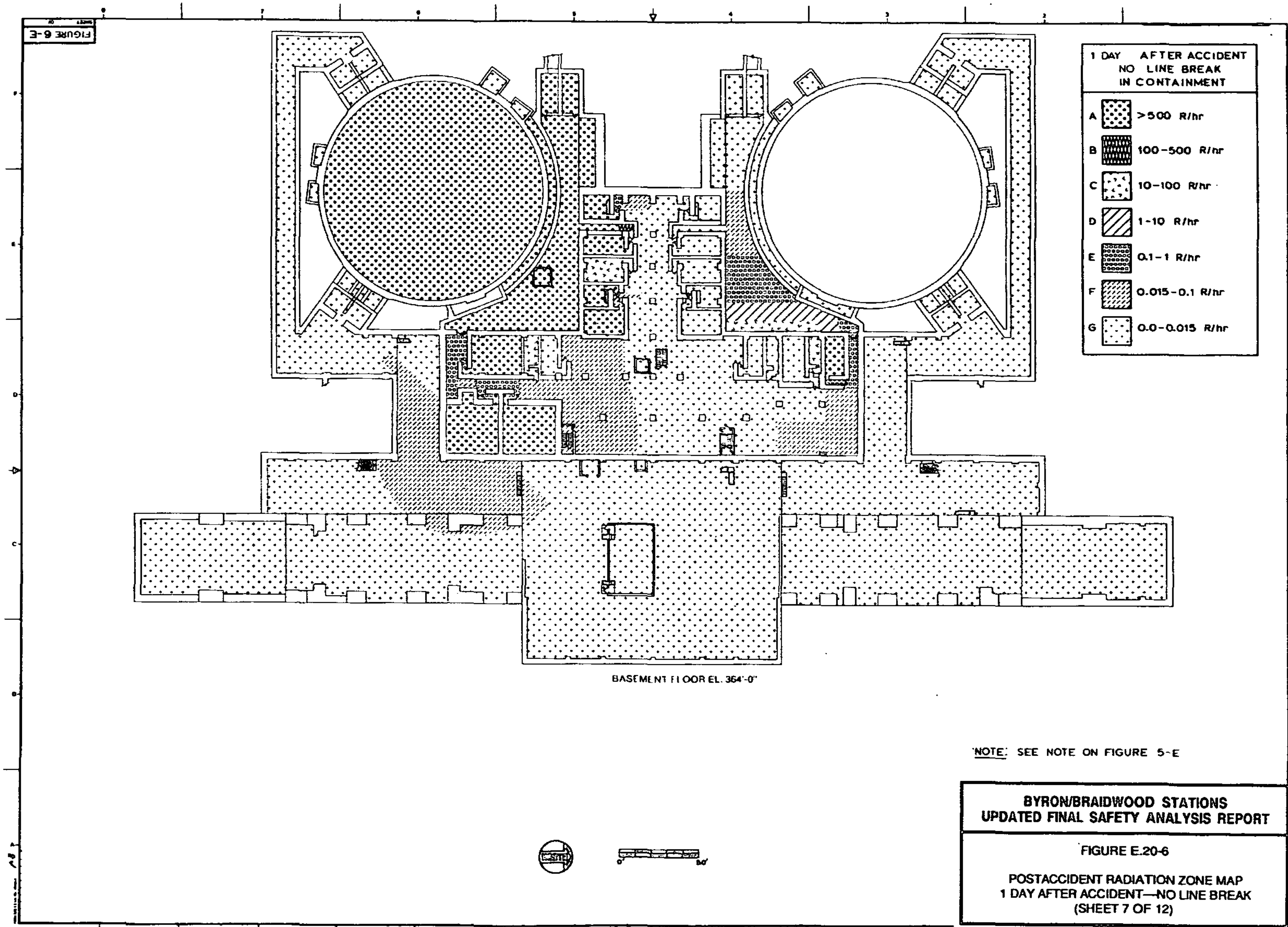


NOTE: THIS ZONE MAP ASSUMES THE ACCIDENT HAS OCCURRED IN UNIT #1. IF THE ACCIDENT OCCURS IN UNIT #2, THE UNIT #2 PIPING PENETRATION AND HEAT EXCHANGER AREAS WILL BE ZONED IN THE SAME MANNER AS THE UNIT #1 CORRESPONDING AREAS. THE UNIT #1 PENETRATION AREA AND THE REMAINDER OF THE AUXILIARY BUILDING WILL NOT CHANGE SIGNIFICANTLY.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-6

POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—NO LINE BREAK
(SHEET 6 OF 12)



NOTE: SEE NOTE ON FIGURE 5-E

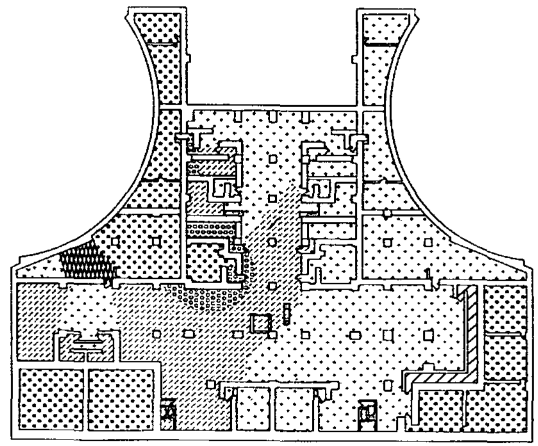
BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-6

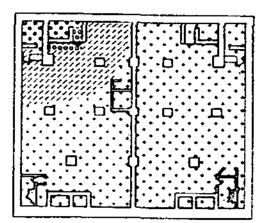
POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—NO LINE BREAK
(SHEET 7 OF 12)

FIGURE 7-E

1 DAY AFTER ACCIDENT NO LINE BREAK IN CONTAINMENT	
A	>500 R/hr
B	100-500 R/hr
C	10-100 R/hr
D	1-10 R/hr
E	0.1-1 R/hr
F	0.015-0.1 R/hr
G	0.0-0.015 R/hr



FLOOR PLAN EL. 346'-0"



ELEV 330'-0"

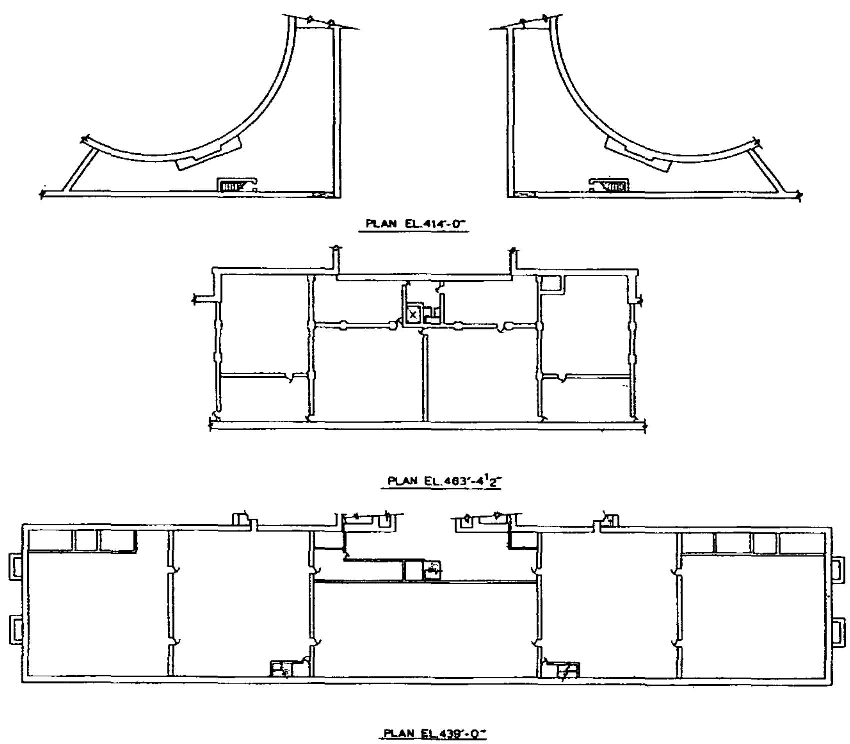
NOTE: SEE NOTE ON FIGURE 5-E

0' 50'

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-6

POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—NO LINE BREAK
(SHEET 8 OF 12)



1 DAY AFTER ACCIDENT NO LINE BREAK IN CONTAINMENT	
A	>500 R/hr
B	100–500 R/hr
C	10–100 R/hr
D	1–10 R/hr
E	0.1–1 R/hr
F	0.015–0.1 R/hr
G	0.0–0.015 R/hr

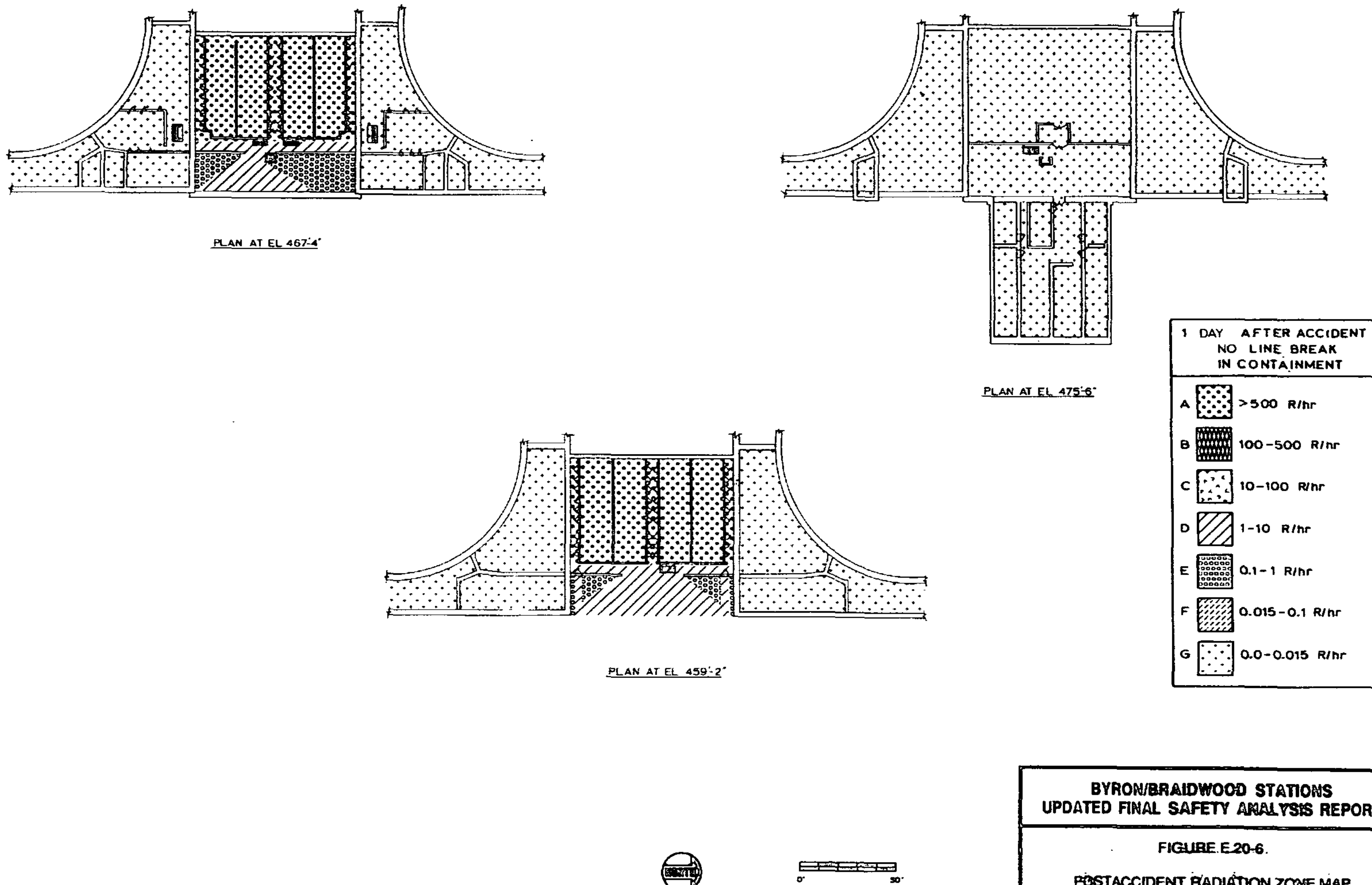
NOTE: ALL DOSE RATE ZONES ON THIS FIGURE ARE LESS THAN 15 mR/hr.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-6

POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—NO LINE BREAK
(SHEET 9 OF 12)

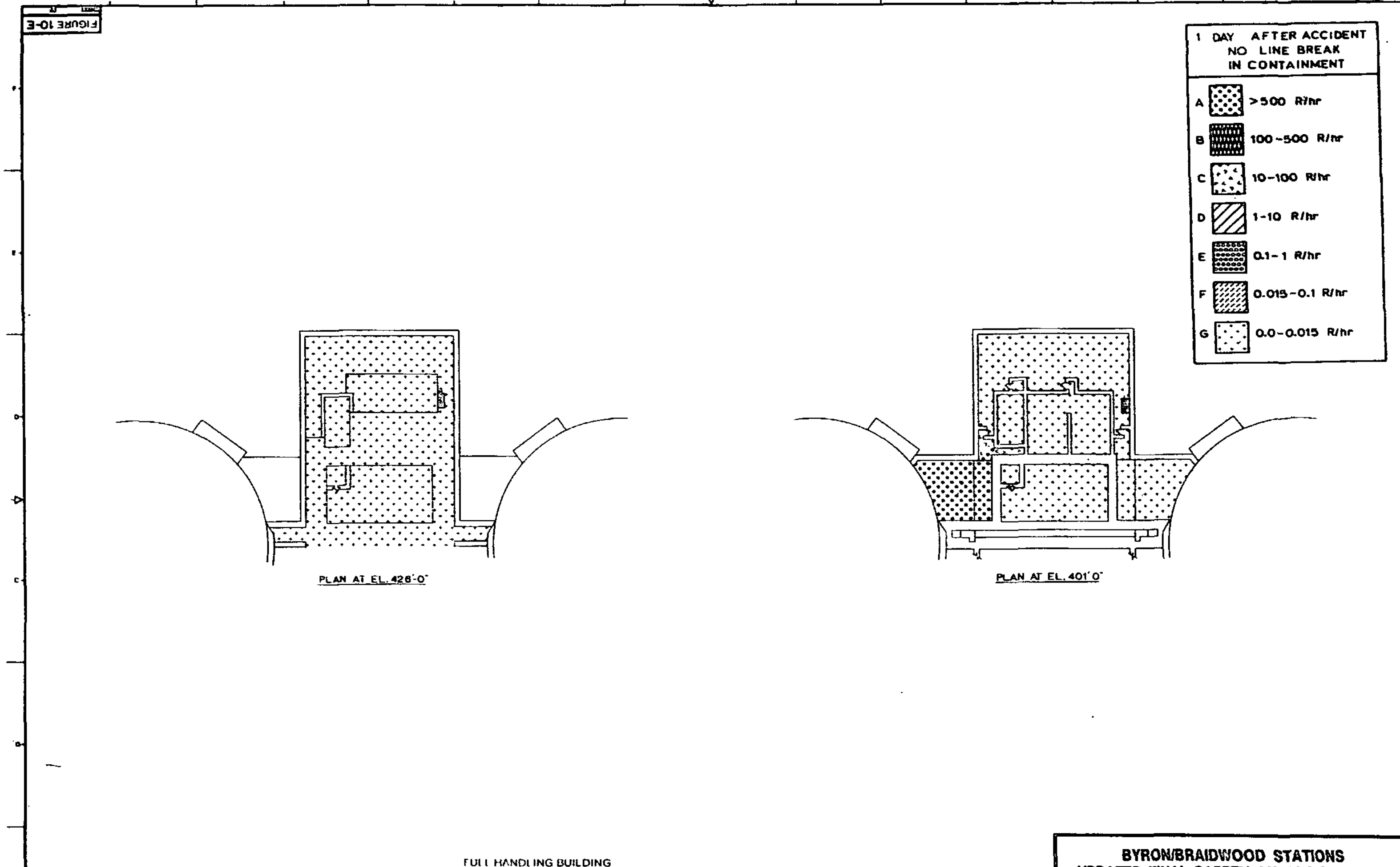
REVISION 3
DECEMBER 1991



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-6

POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—NO LINE BREAK
(SHEET 10 OF 12)

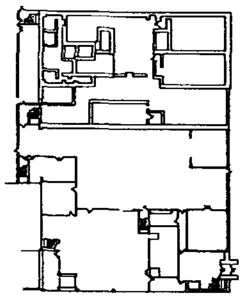


BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

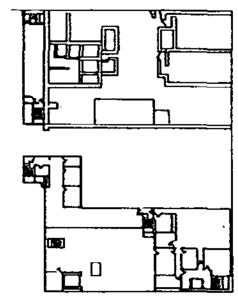
FIGURE E.20-6

POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—NO LINE BREAK
(SHEET 11 OF 12)

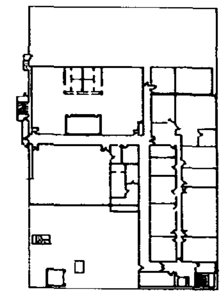
REVISION 3
DECEMBER 1991



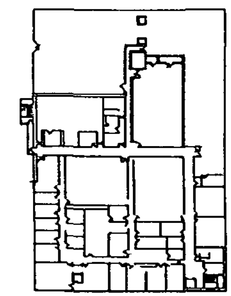
GROUND FLOOR PLAN
EL.397'-0" & 401'-0"



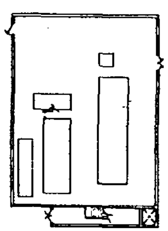
SECOND FLOOR PLAN
EL.410'-0" & 417'-0"



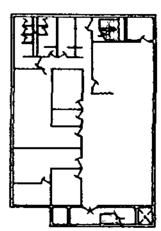
THIRD FLOOR PLAN
EL.433'-0"



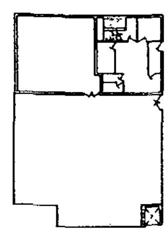
FOURTH FLOOR PLAN
EL.451'-0"



CONDENSATE CLEAN-UP BLDG.
EL.401'-0"



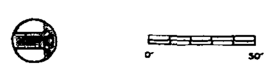
TECHNICAL SUPPORT CENTER
EL.435'-0"



TECHNICAL SUPPORT CENTER
EL 451'-0"

1 DAY AFTER ACCIDENT NO LINE BREAK IN CONTAINMENT	
A	>500 R/hr
B	100-500 R/hr
C	10-100 R/hr
D	1-10 R/hr
E	0.1-1 R/hr
F	0.015-0.1 R/hr
G	0.0-0.015 R/hr

NOTE: THE DOSE RATE IN THE RADWASTE/SERVICE BLDG AND TECHNICAL SUPPORT CENTER IS LESS THAN 15mR/hr. DUE TO AN ACCIDENT IN EITHER UNIT.

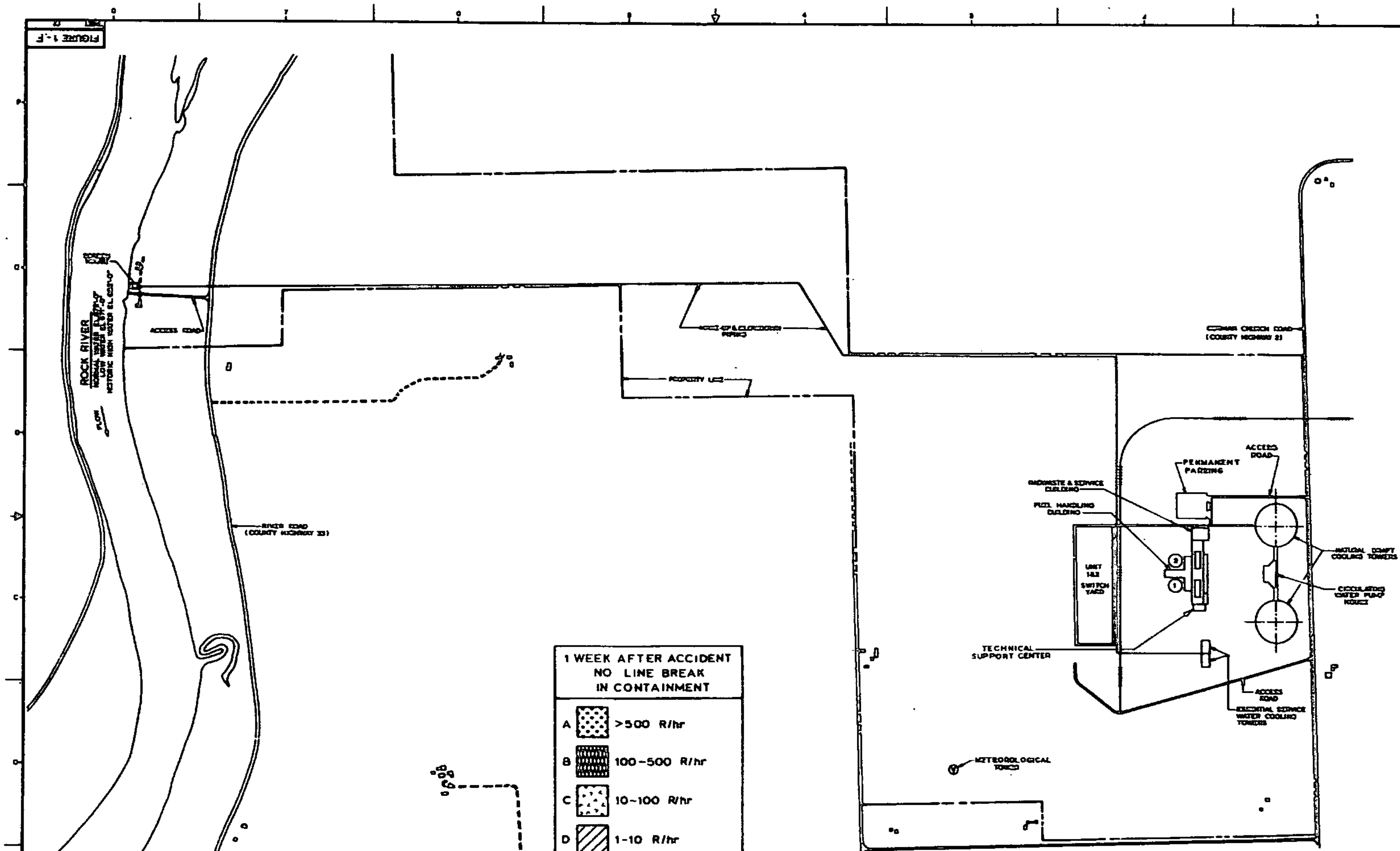


RADWASTE/SERVICE BUILDING/TSC

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-6

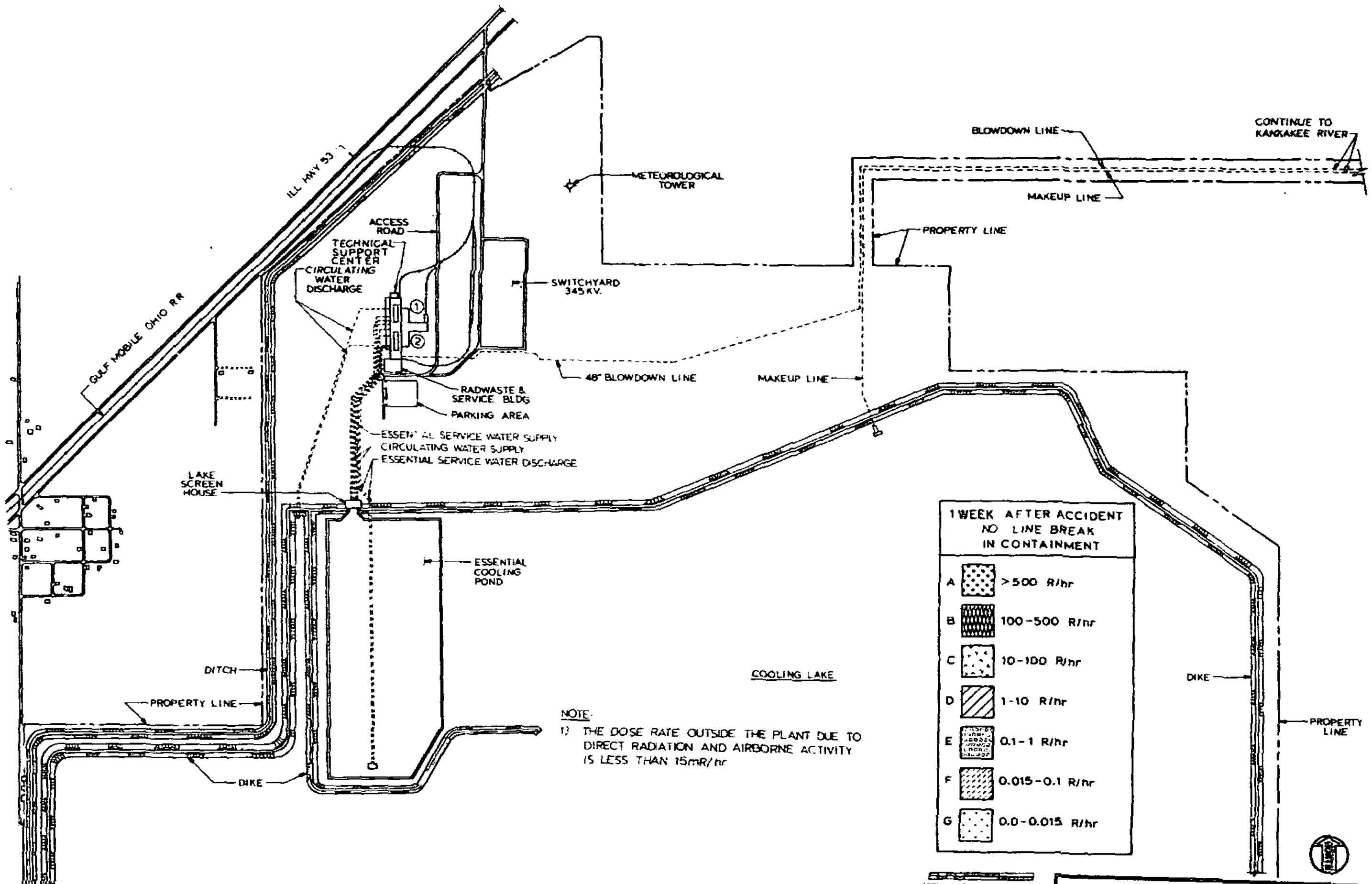
POSTACCIDENT RADIATION ZONE MAP
1 DAY AFTER ACCIDENT—NO LINE BREAK
(SHEET 12 OF 12)



BYRON STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-7

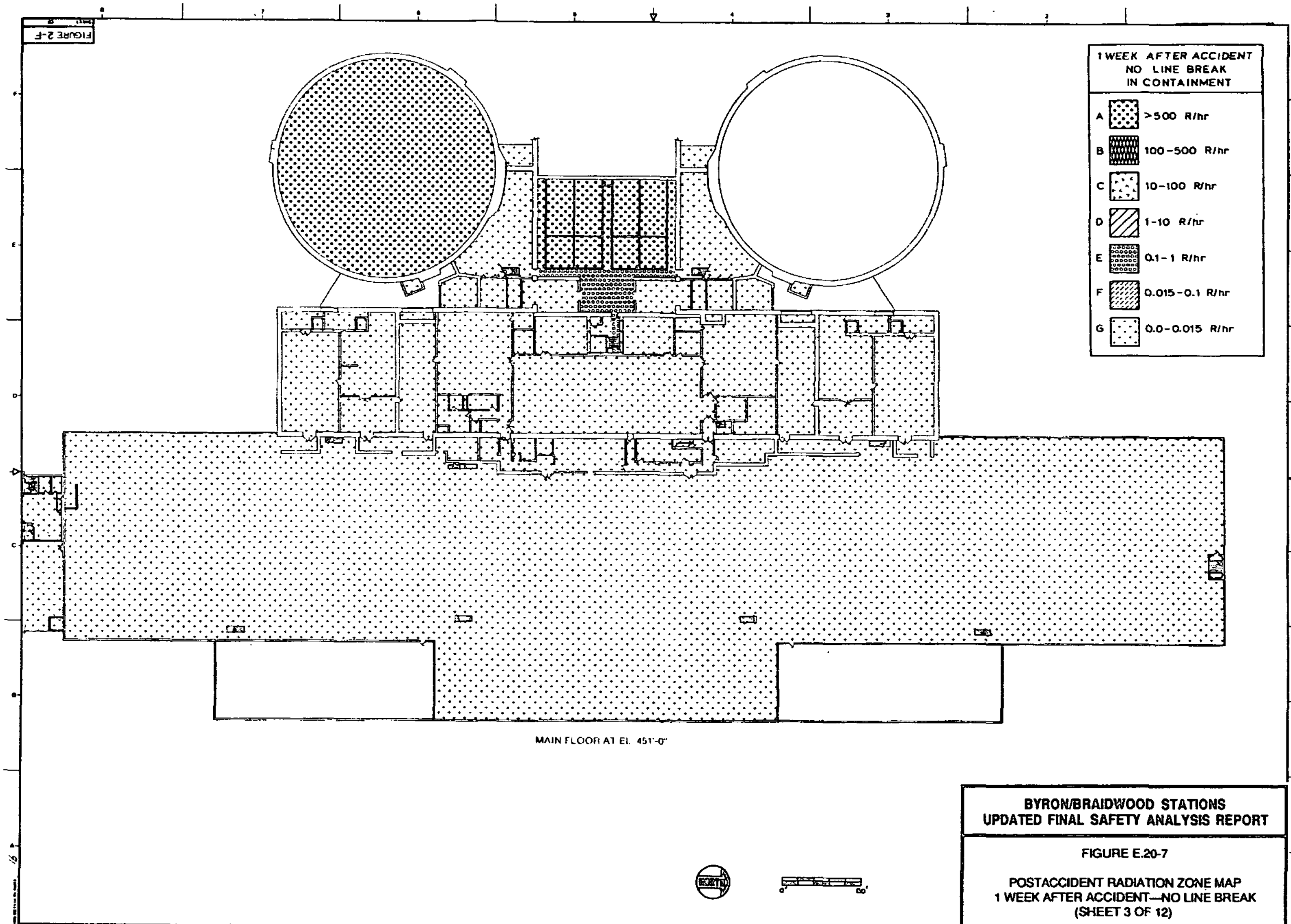
POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—NO LINE BREAK
(SHEET 1 OF 12)



BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-7

POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—NO LINE BREAK
(SHEET 2 OF 12)

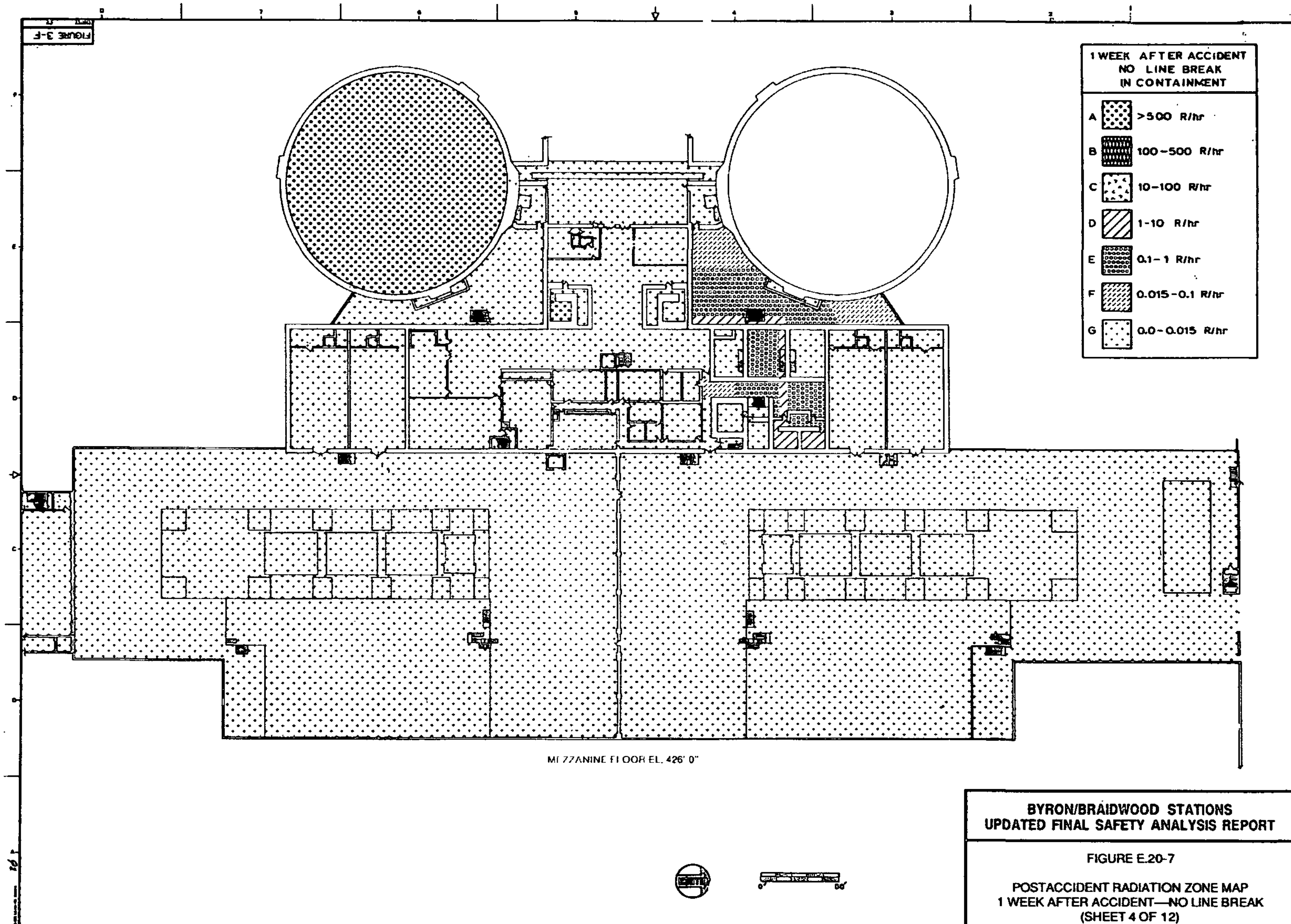


BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-7

POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—NO LINE BREAK
(SHEET 3 OF 12)

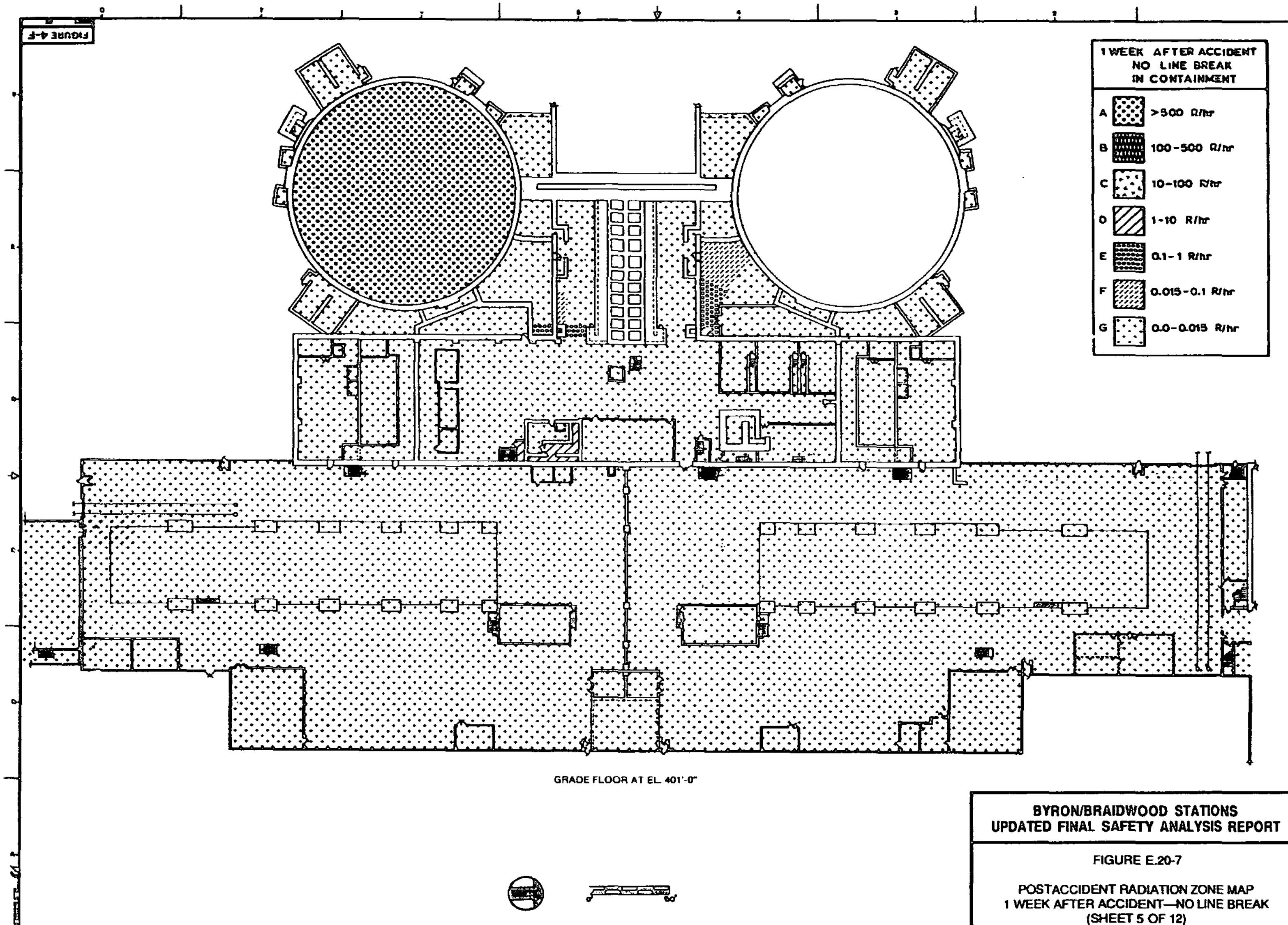
02



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-7

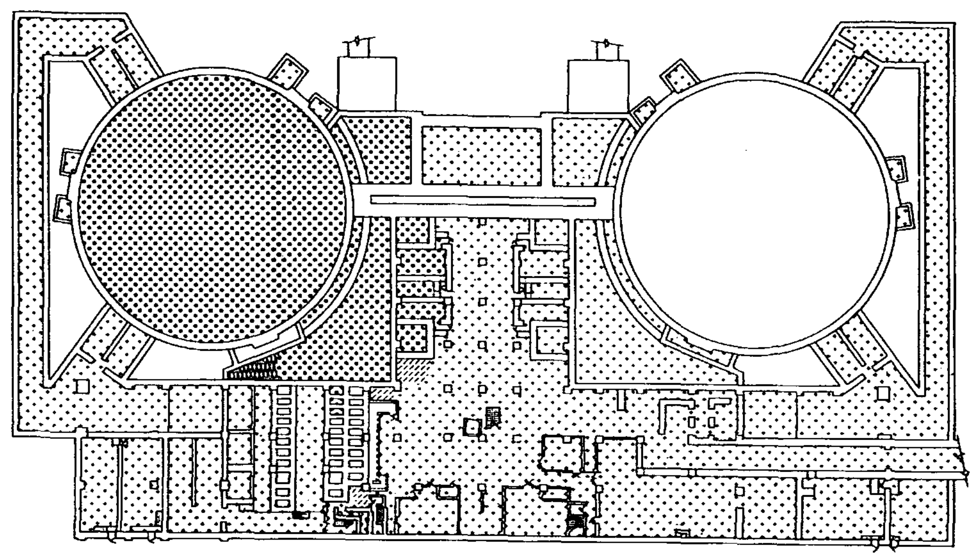
POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—NO LINE BREAK
(SHEET 4 OF 12)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-7

POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—NO LINE BREAK
(SHEET 5 OF 12)



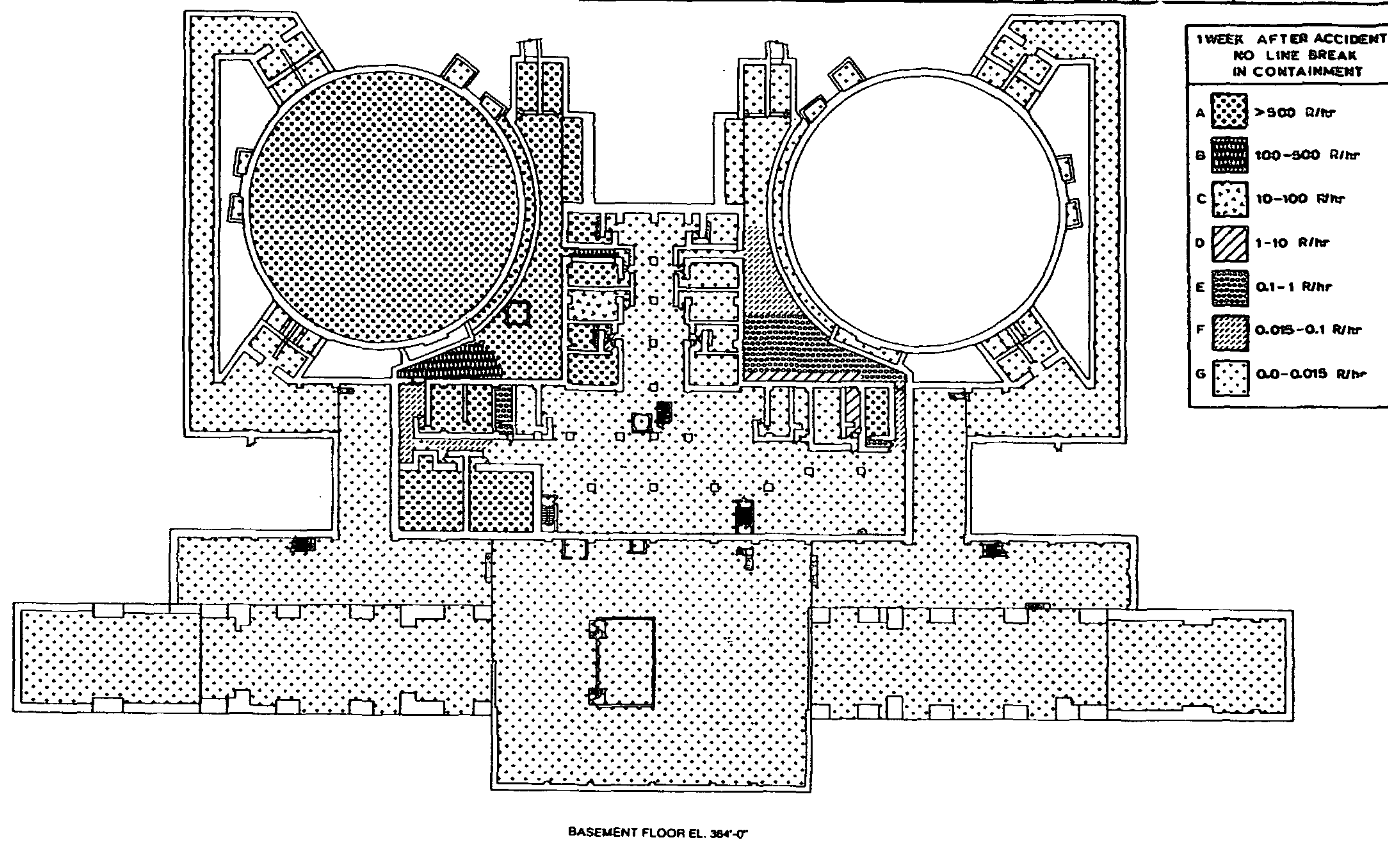
FLOOR PLAN EL. 383'-0"

NOTE: THIS ZONE MAP ASSUMES THE ACCIDENT HAS OCCURRED IN UNIT #1. IF THE ACCIDENT OCCURS IN UNIT #2, THE UNIT #2 PIPING PENETRATION AND HEAT EXCHANGER AREAS WILL BE ZONED IN THE SAME MANNER AS THE UNIT #1 CORRESPONDING AREAS. THE UNIT #1 PENETRATION AREA AND THE REMAINDER OF THE AUXILIARY BUILDING WILL NOT CHANGE SIGNIFICANTLY.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-7

POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—NO LINE BREAK
(SHEET 6 OF 12)

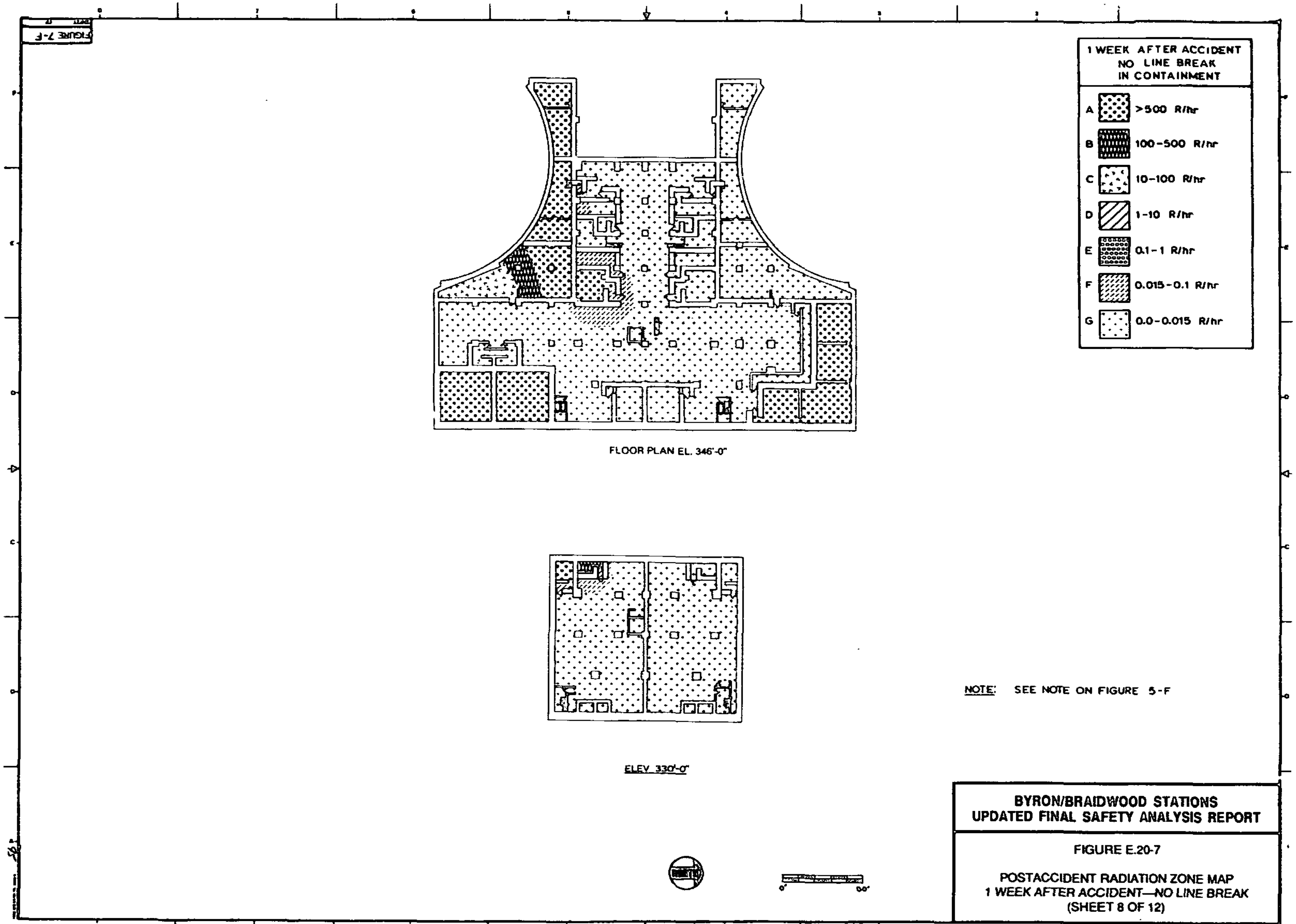


NOTE: SEE NOTE ON FIGURE 5-F

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-7

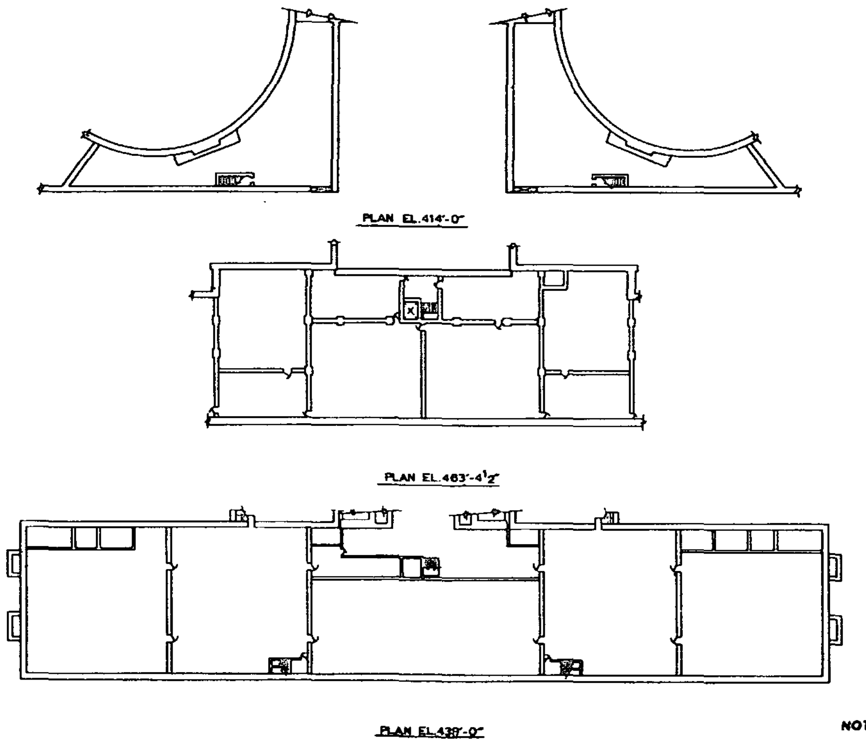
POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—NO LINE BREAK
(SHEET 7 OF 12)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-7

POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—NO LINE BREAK
(SHEET 8 OF 12)



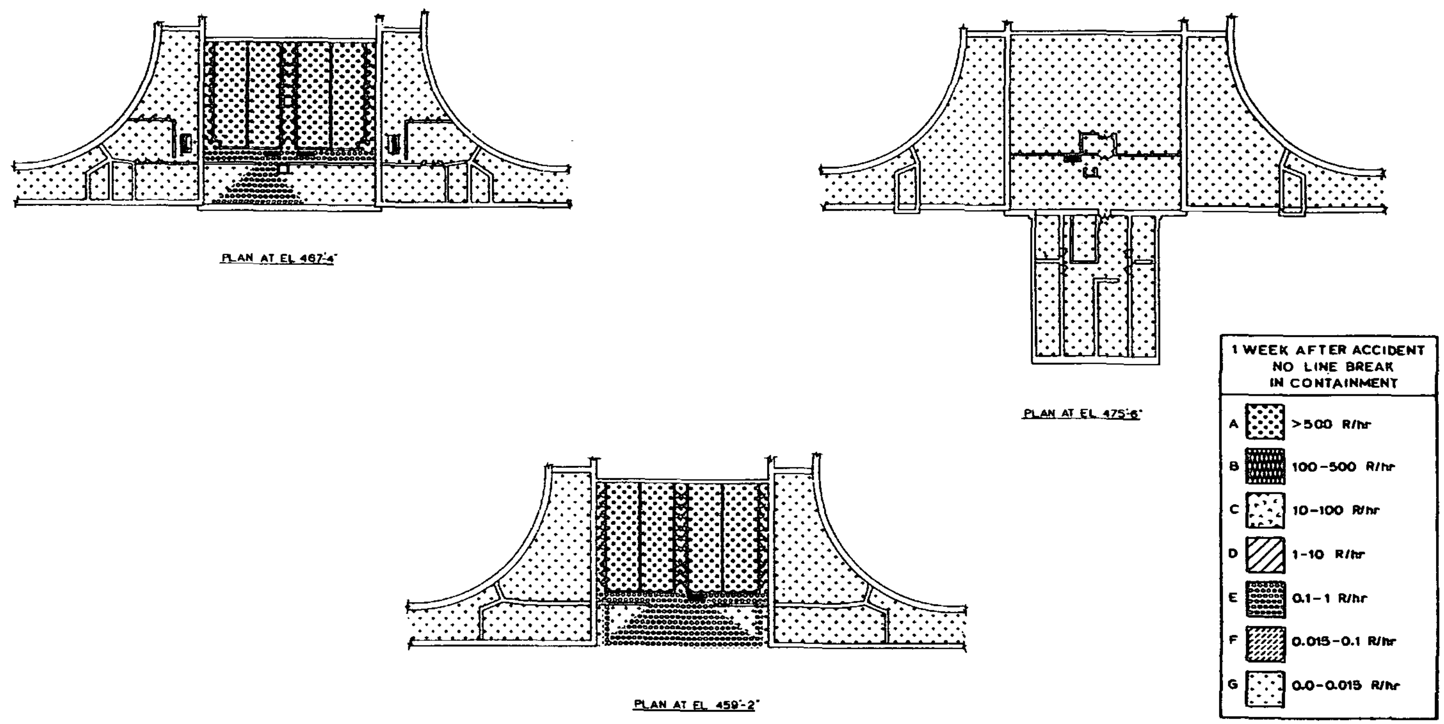
1 WEEK AFTER ACCIDENT NO LINE BREAK IN CONTAINMENT	
A	>500 R/hr
B	100-500 R/hr
C	10-100 R/hr
D	1-10 R/hr
E	0.1-1 R/hr
F	0.015-0.1 R/hr
G	0.0-0.015 R/hr

NOTE: DUE TO THE ACCIDENT, THE RADIATION ZONES ON THIS FIGURE ARE LESS THAN 15mR/hr.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-7

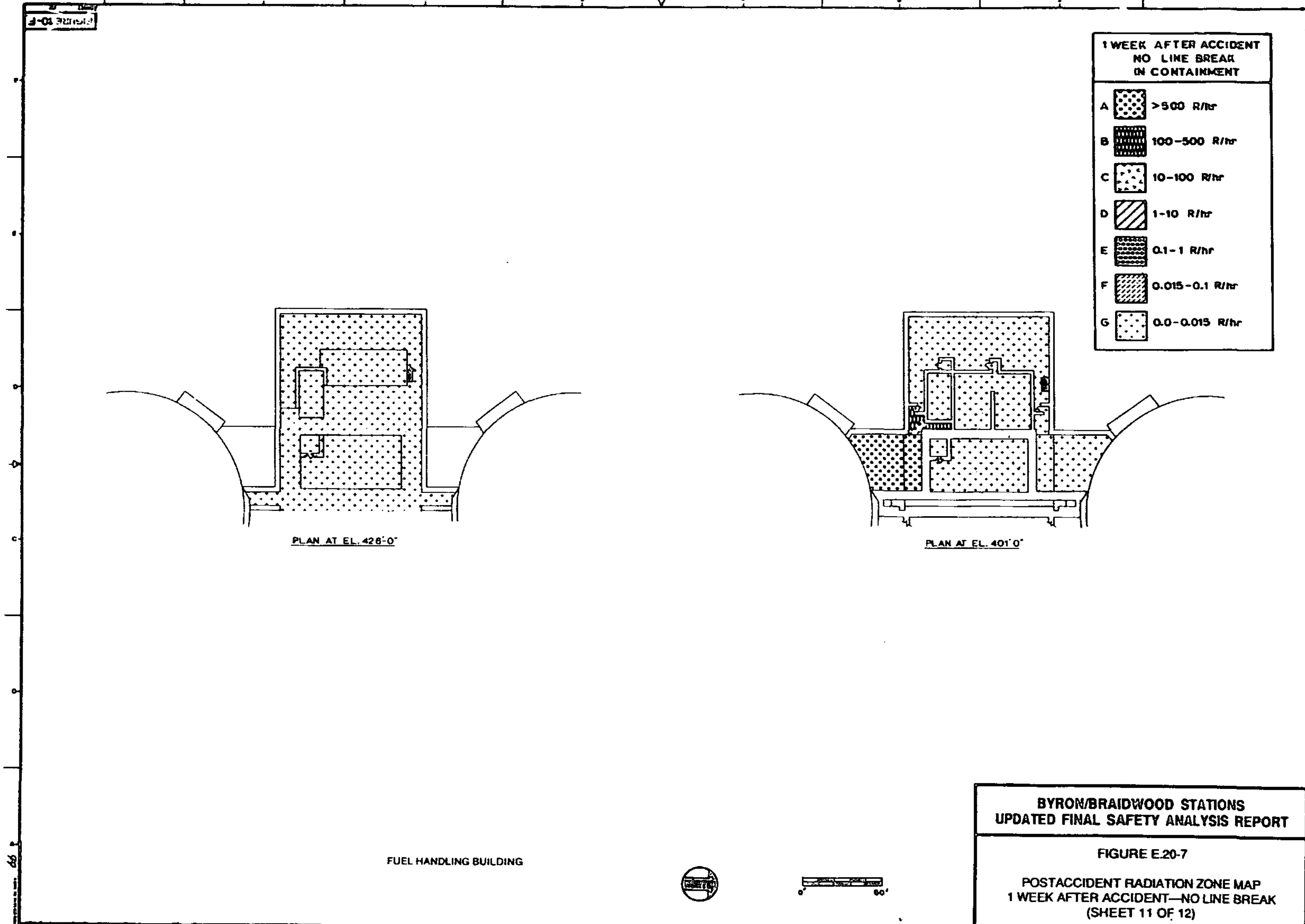
POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—NO LINE BREAK
(SHEET 9 OF 12)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-7

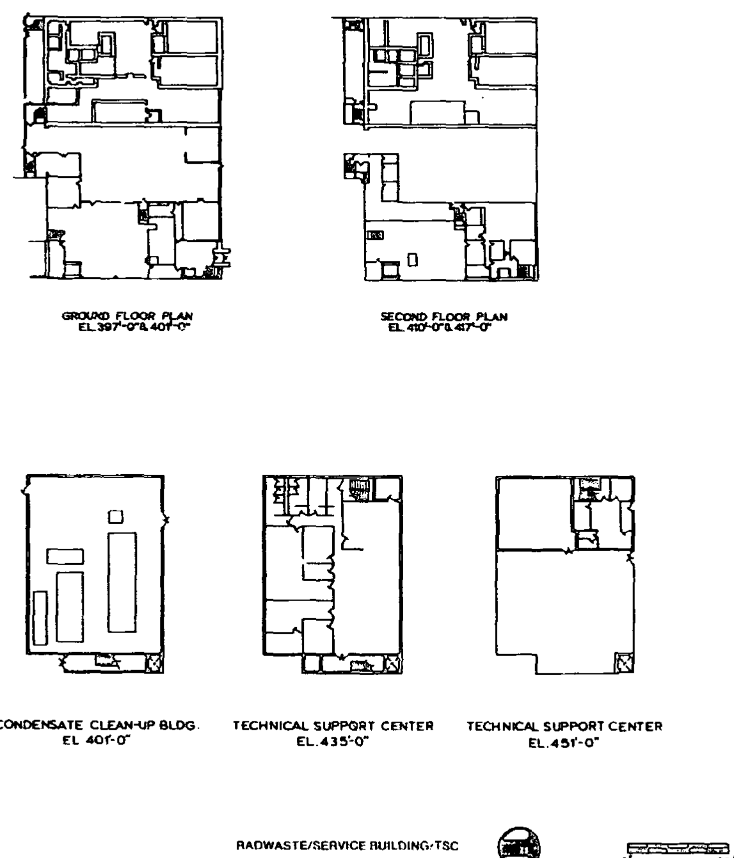
POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—NO LINE BREAK
(SHEET 10 OF 12)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-7

POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—NO LINE BREAK
(SHEET 11 OF 12)



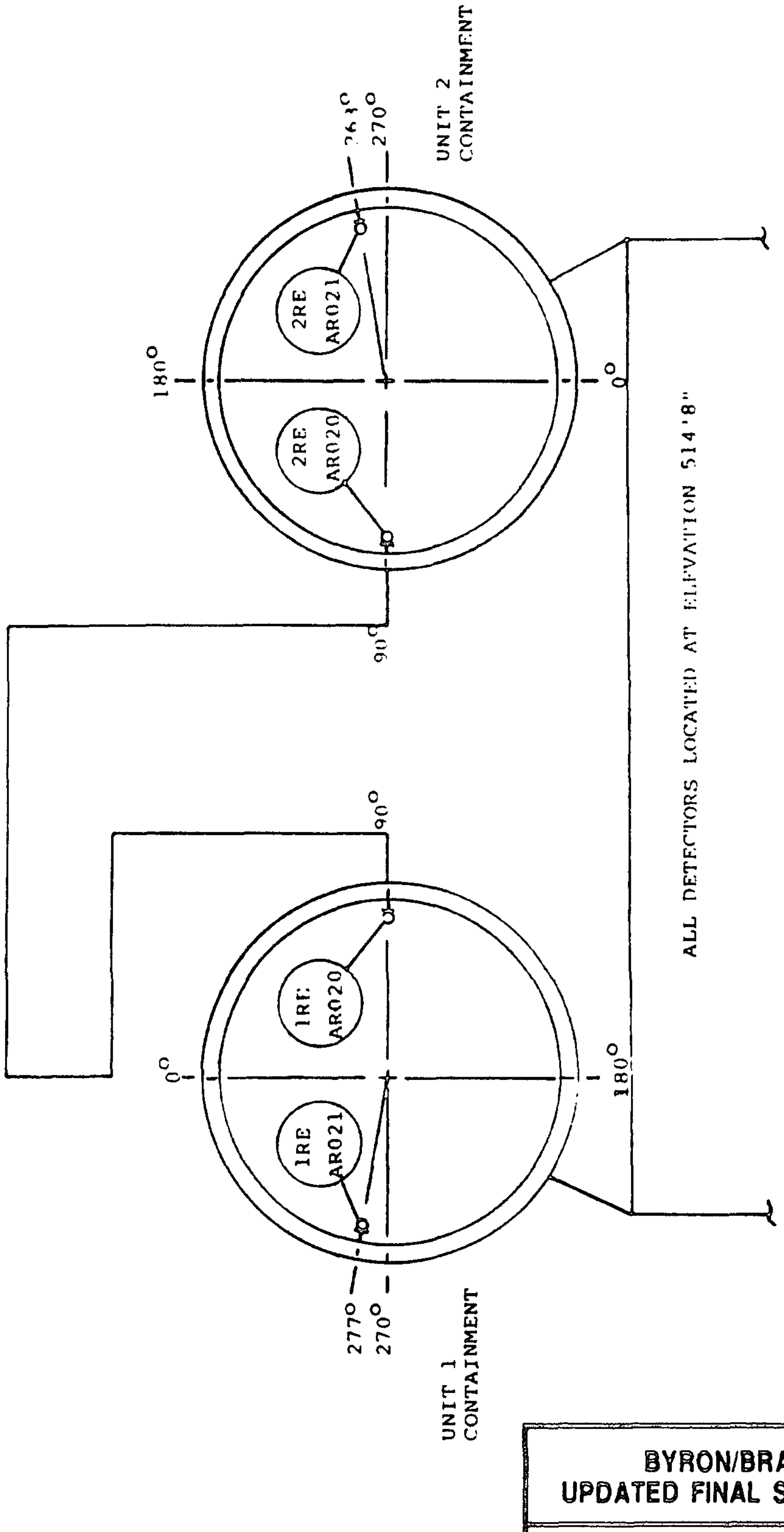
1 WEEK AFTER ACCIDENT NO LINE BREAK IN CONTAINMENT	
A	>500 R/hr
B	100-500 R/hr
C	10-100 R/hr
D	1-10 R/hr
E	0.1-1 R/hr
F	0.015-0.1 R/hr
G	0.0-0.015 R/hr

NOTE: THE DOSE RATE IN THE RADWASTE/SERVICE BUILDING AND TECHNICAL SUPPORT CENTER IS LESS THAN 15mR/hr DUE TO AN ACCIDENT IN EITHER UNIT.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.20-7

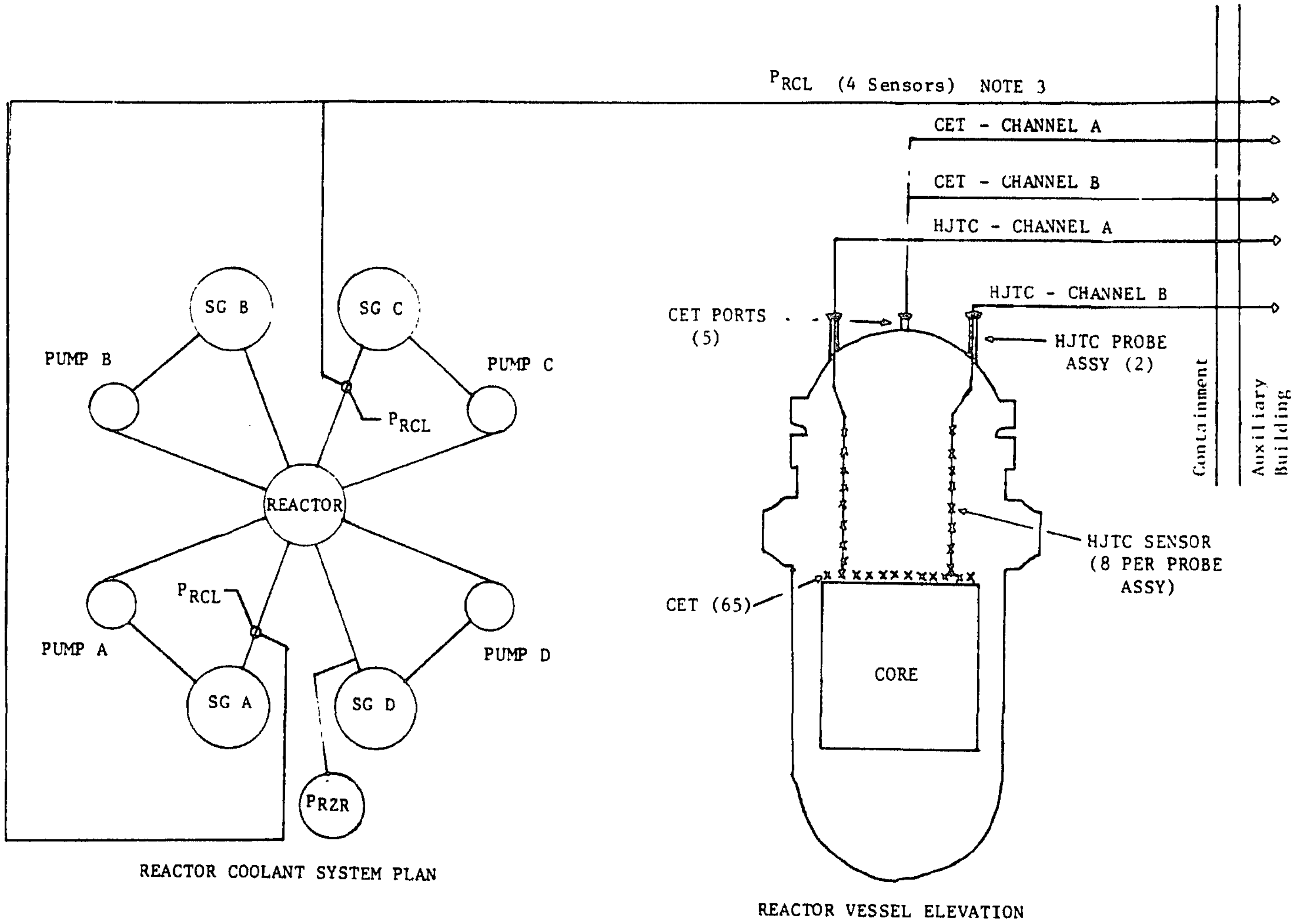
POSTACCIDENT RADIATION ZONE MAP
1 WEEK AFTER ACCIDENT—NO LINE BREAK
(SHEET 12 OF 12)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.30-1

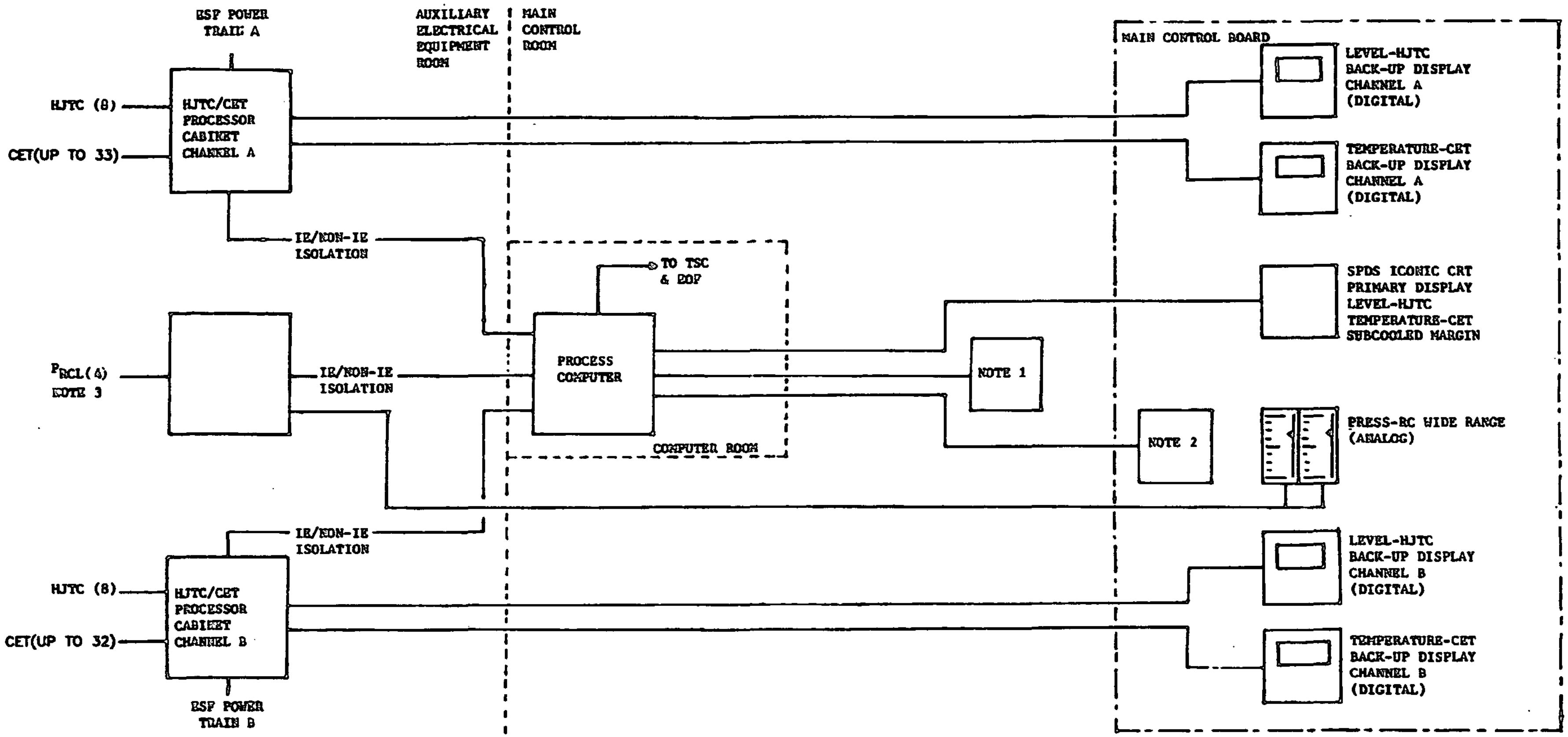
HIGH RANGE CONTAINMENT
RADIATION DETECTORS



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.31-1

ICC DETECTION INSTRUMENTATION SENSORS PROCESSING AND DISPLAY (SHEET 1 OF 2)

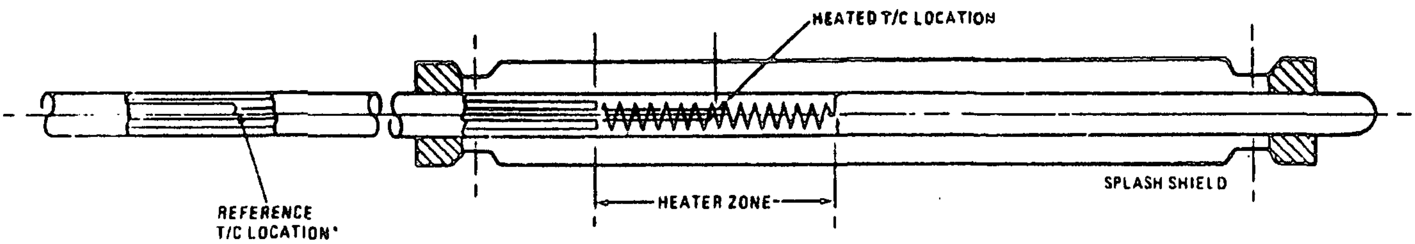


NOTES:
1. THREE MONITORS AND PRINTERS, COMPUTER DRIVEN, NON-DEDICATED FOR OPTIONAL ICC DISPLAY SUPPORT.
2. SIX ANALOG INDICATORS, TWO CONTINUOUS PEN RECORDERS AND ONE MONITOR, COMPUTER DRIVEN, NON-DEDICATED FOR OPTIONAL ICC DISPLAY SUPPORT.
3. FOR BYRON UNIT 1, ONLY 2 SENSORS WILL BE AVAILABLE UNTIL THE FIRST REFUELING OUTAGE.

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.31-1

ICC DETECTION INSTRUMENTATION SENSORS PROCESSING AND DISPLAY
(SHEET 2 OF 2)



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.31-2

HJTC SENSOR—HJTC/SPLASH SHIELD

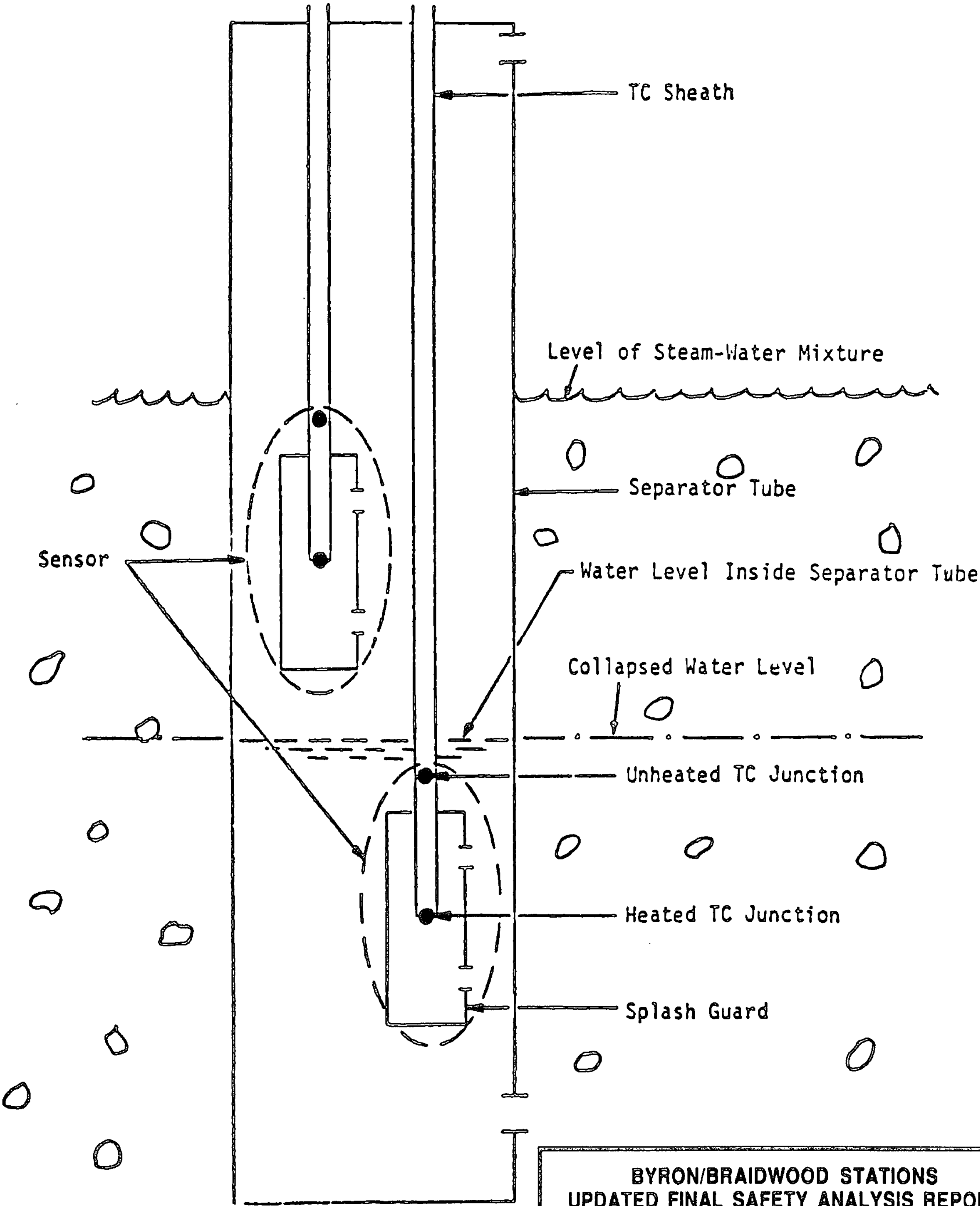
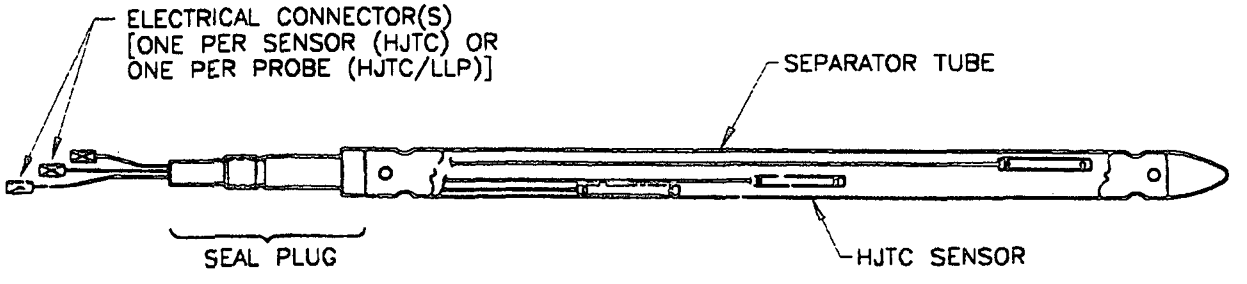


FIGURE E.31-3

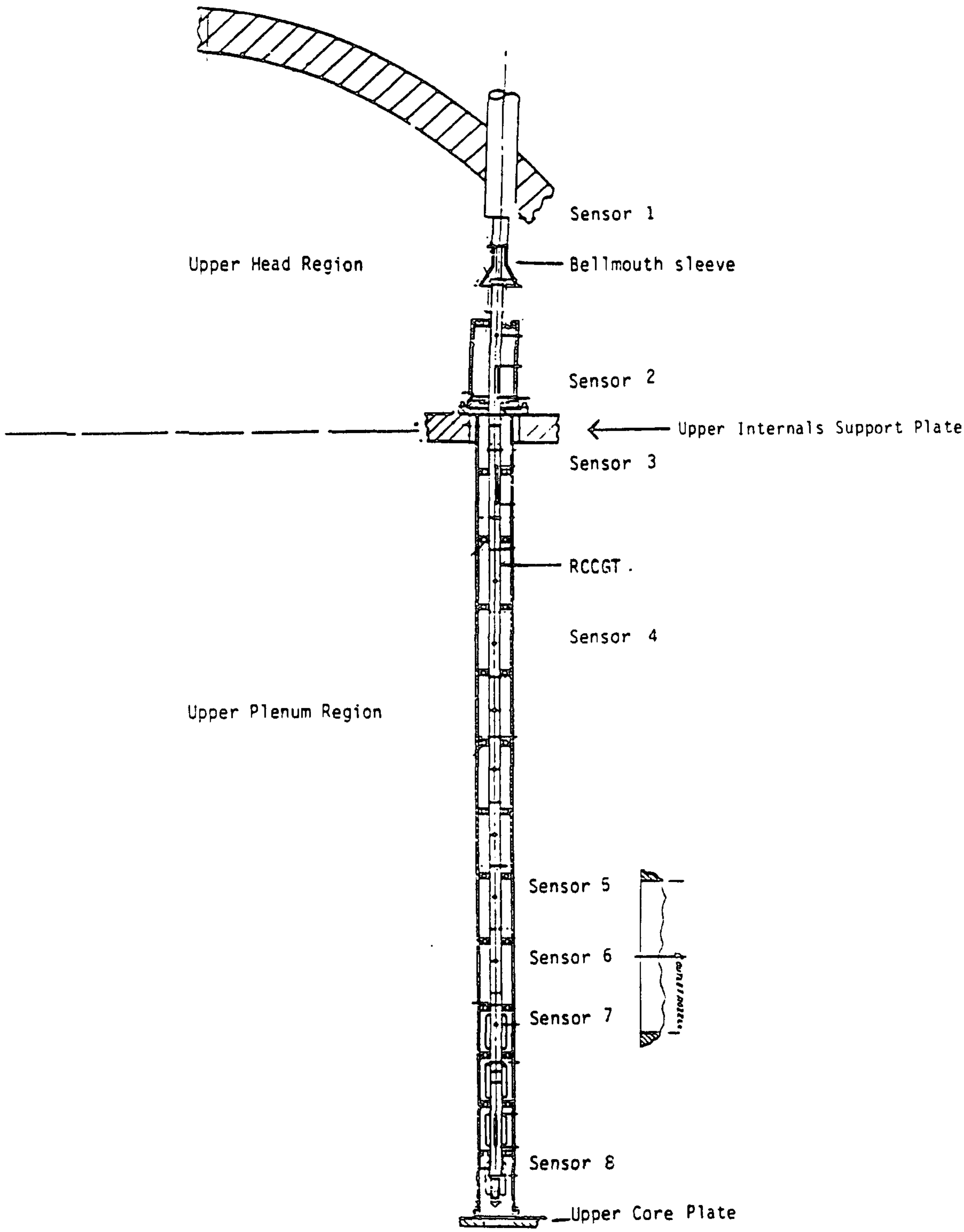
SCHEMATIC OF HJTC PROBE



BYRON/BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.31-4

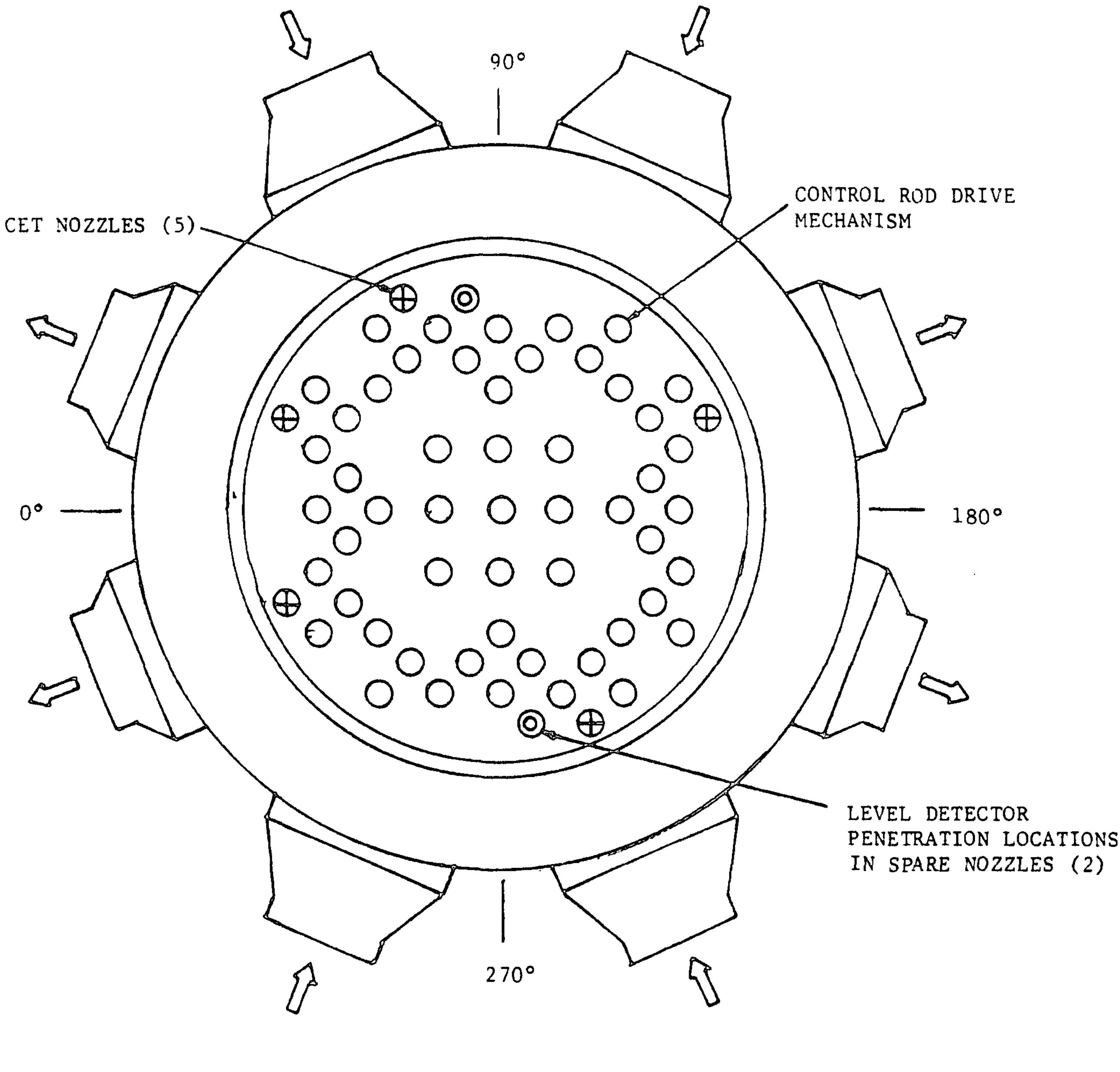
HJTC PROBE ASSEMBLY



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.31-5

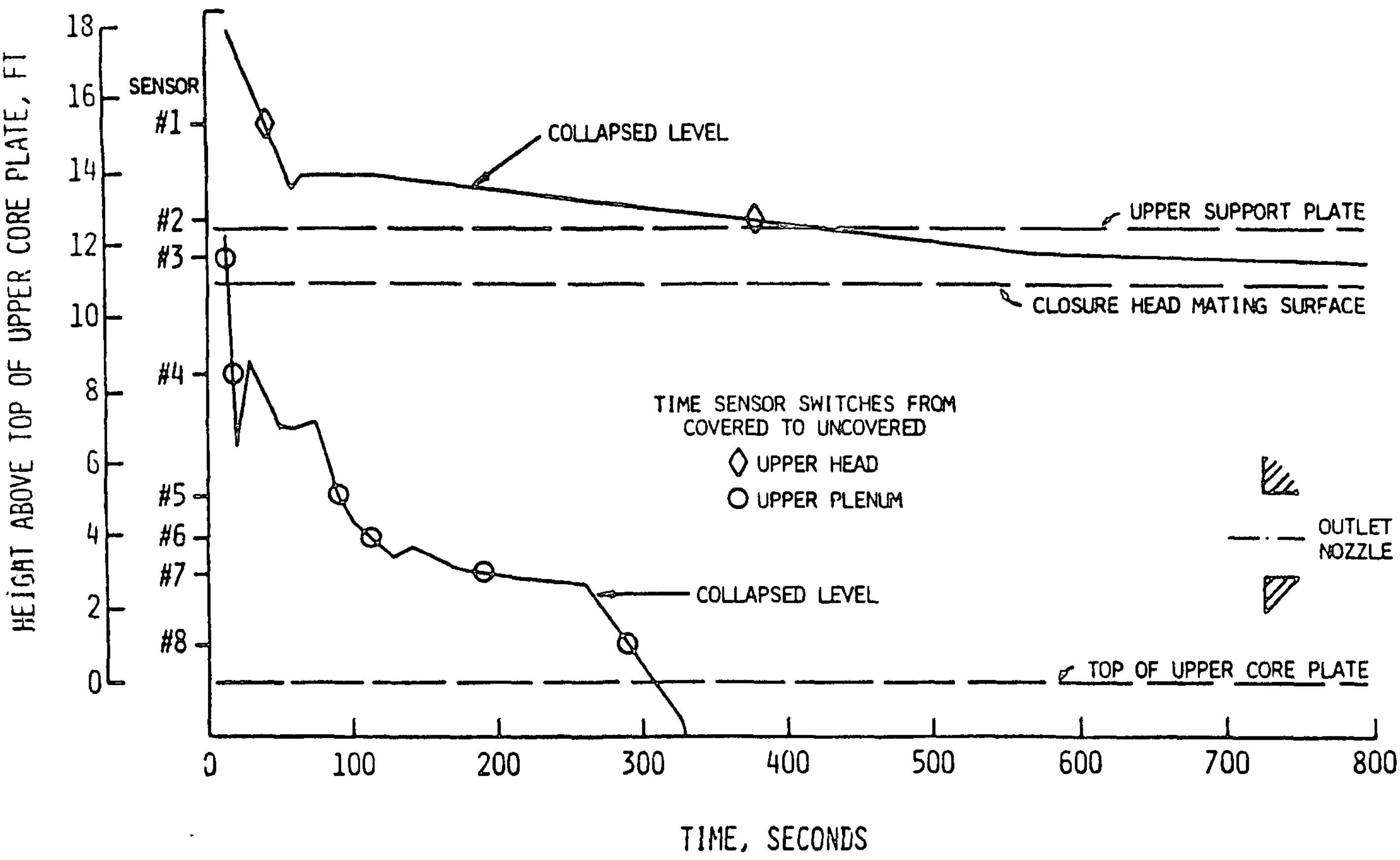
PROBE HOLDER ASSEMBLY AND
SENSOR LOCATIONS



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.31-6

HJTC HOLDER ASSEMBLY LOCATIONS
REACTOR VESSEL PLAN

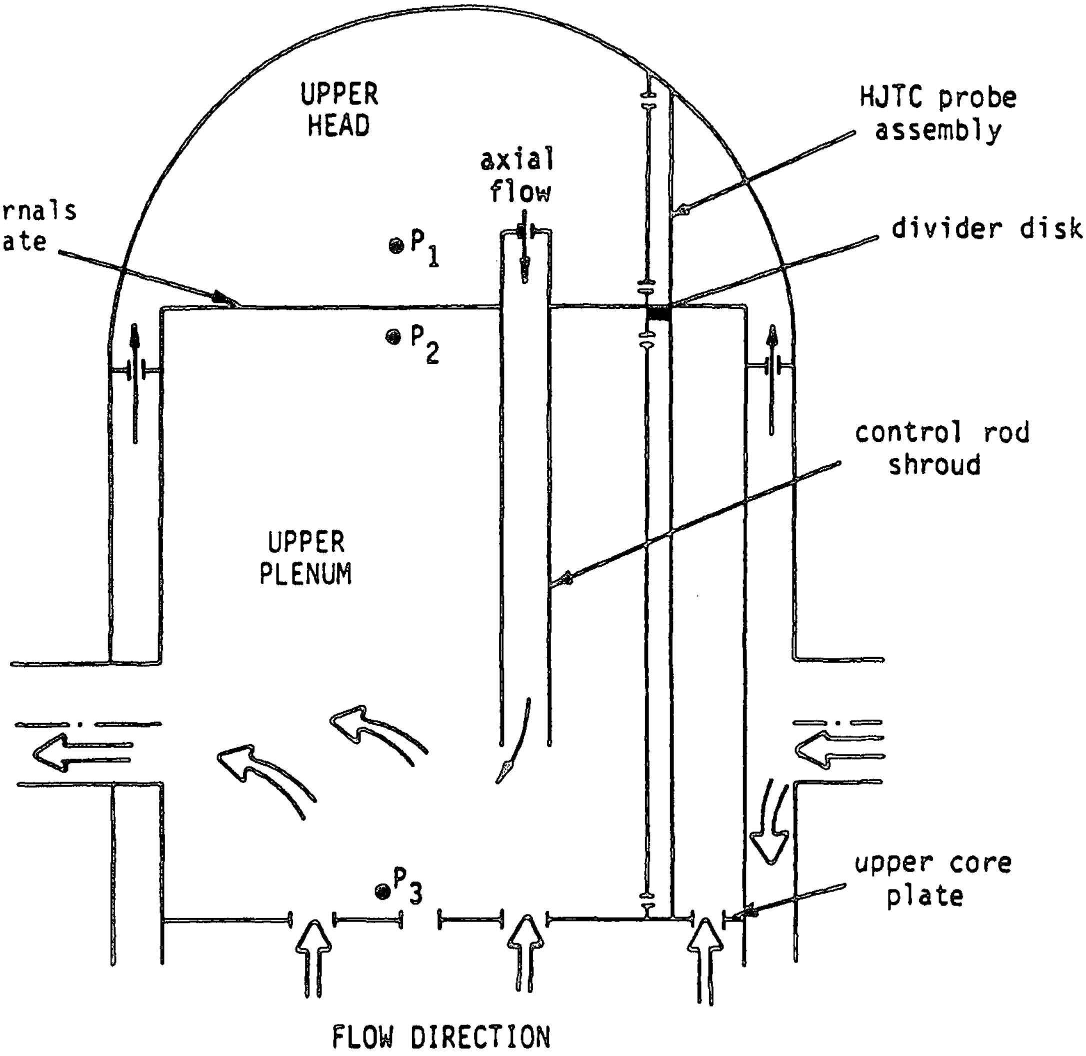


(Based on WCAP 9601, VOL. III, SEC. 4.1)

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.31-7

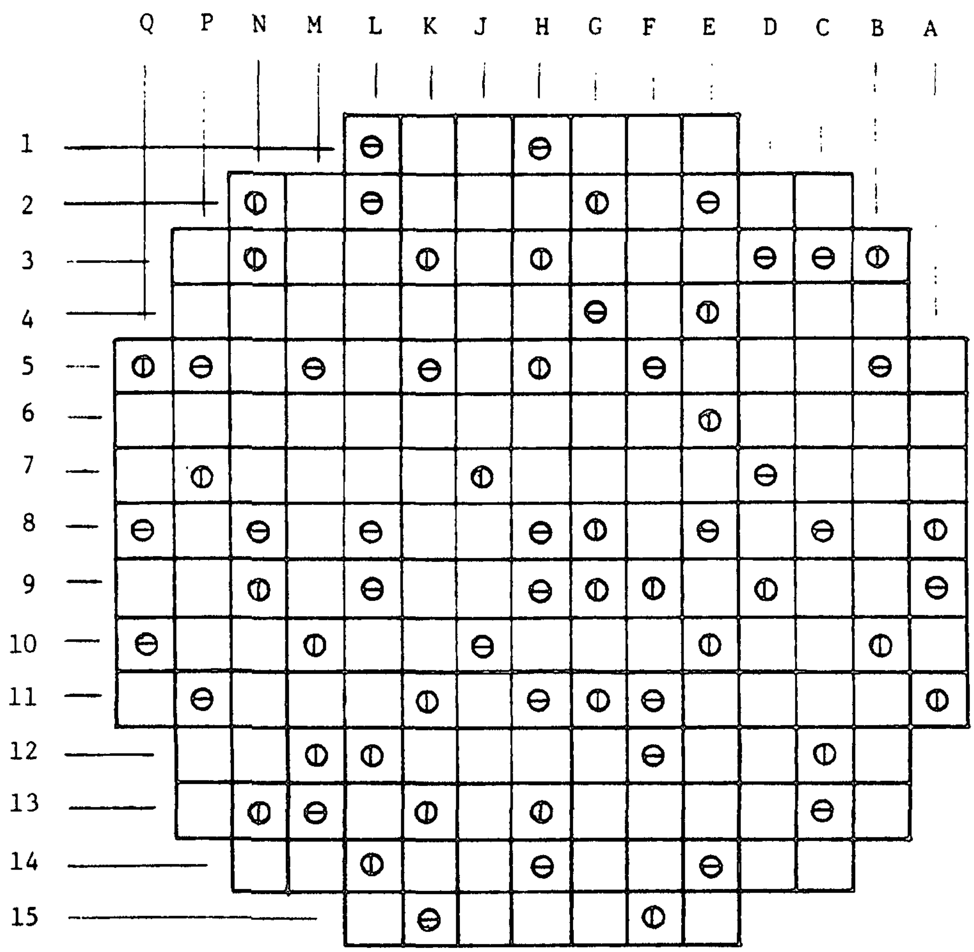
HJTC SENSOR RESPONSE FOR A REPRESENTATIVE 4 IN. (0.09 FT^2) COLD LEG BREAK



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.31-8

NORMAL OPERATING FLOW PATTERN WITH REACTOR COOLANT PUMPS ON



NOTE: FOR BYRON/BRAIDWOOD UNIT 1,
THERMOCOUPLE LOCATION
M10 IS K10.

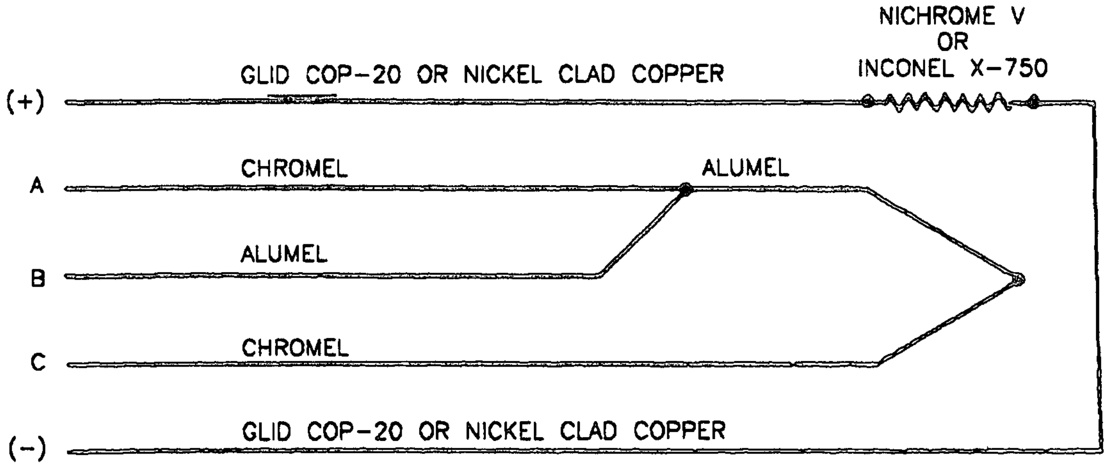
TRAIN A THERMOCOUPLE LOCATION

TRAIN B THERMOCOUPLE LOCATION

BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.31-9

CORE EXIT THERMOCOUPLE
CORE LOCATIONS

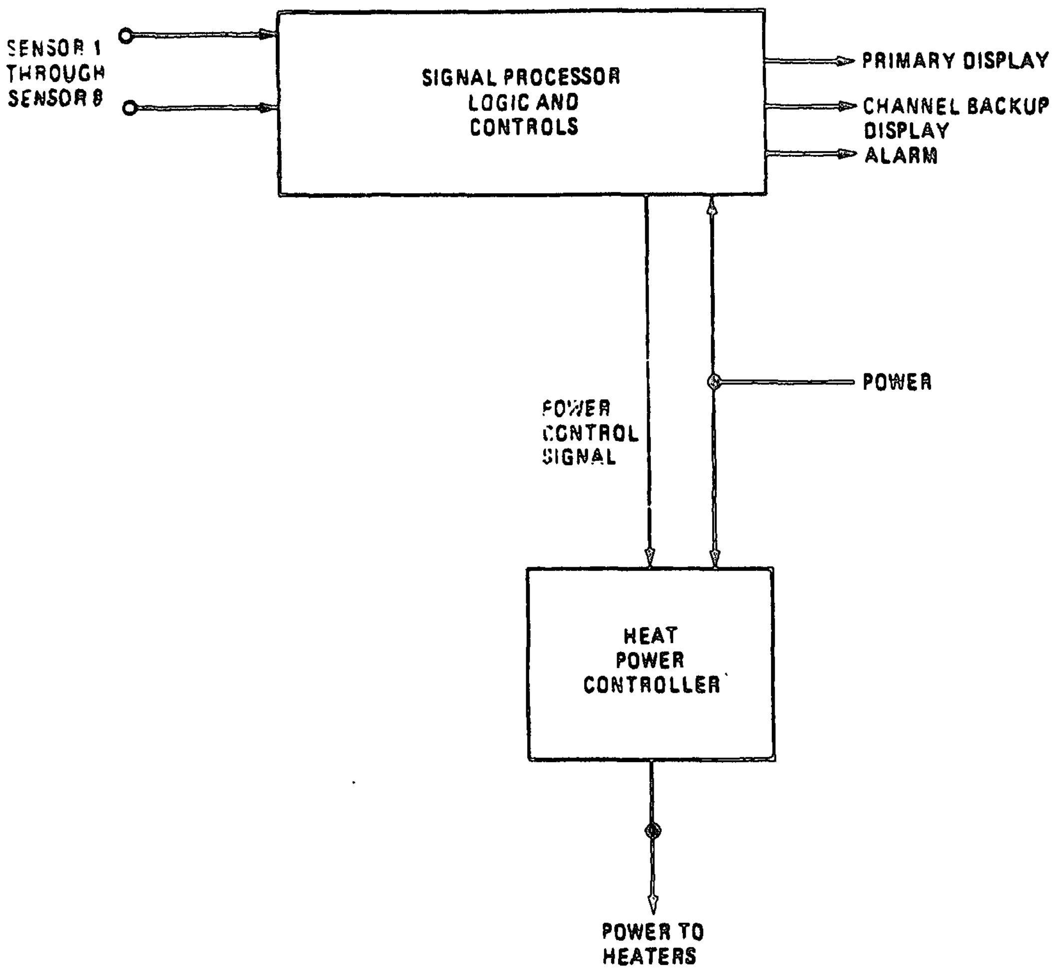


V(A-B) = ACTUAL TEMPERATURE, UNHEATED JUNCTION
V(C-B) = ACTUAL TEMPERATURE, HEATED JUNTION
V(A-C) = DIFFERENTIAL TEMPERATURE

BYRON/BRAIDWOOD STATION
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.31-10

ELECTRICAL DIAGRAM OF HJTC



BYRON/BRAIDWOOD STATIONS
UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE E.31-11

HJTC SYSTEM PROCESSING CONFIGURATION
(ONE CHANNEL SHOWN)

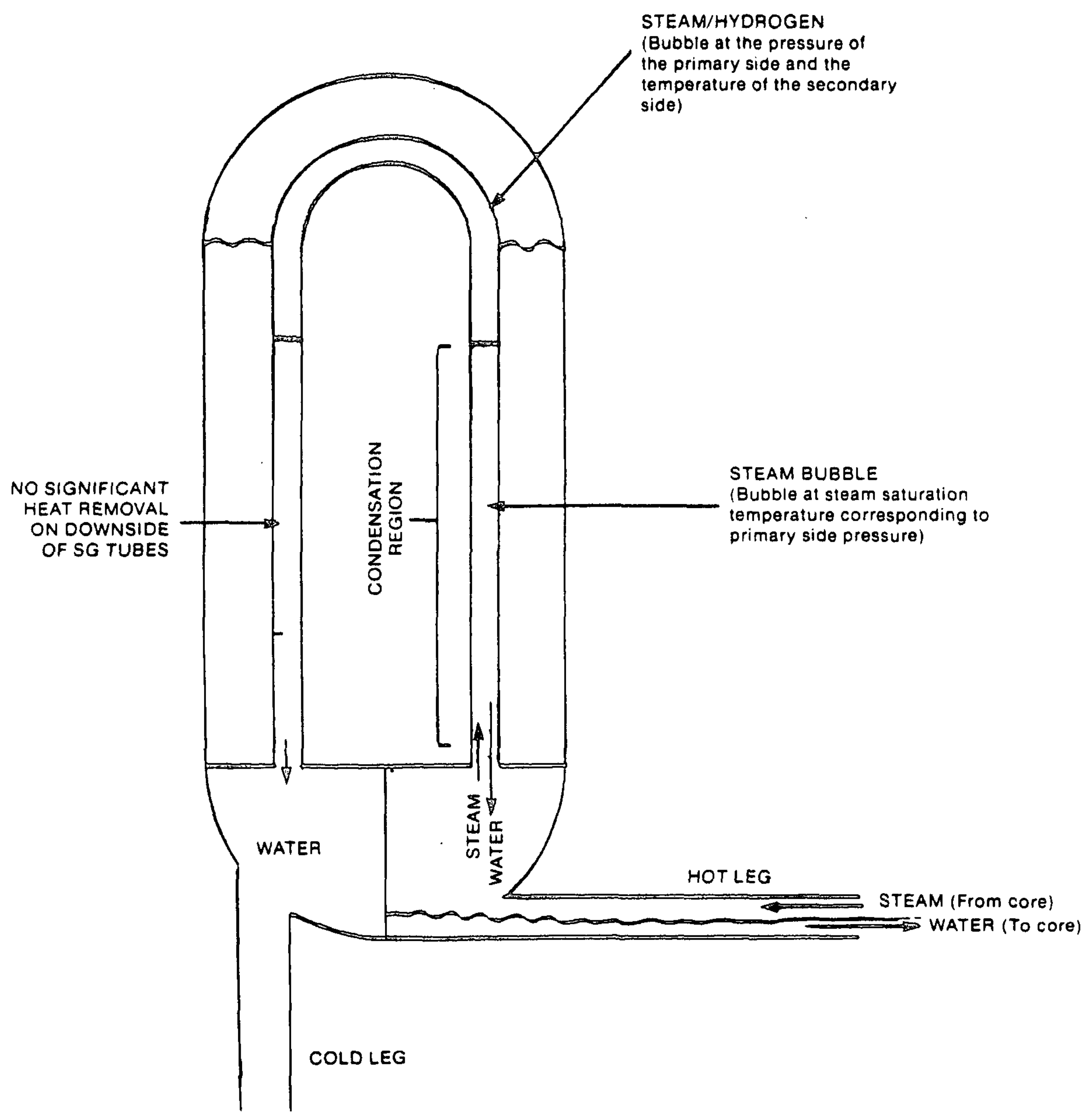


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FIGURE E.47-1

LOOP REFLUX ILLUSTRATION
HYDROGEN BUBBLE

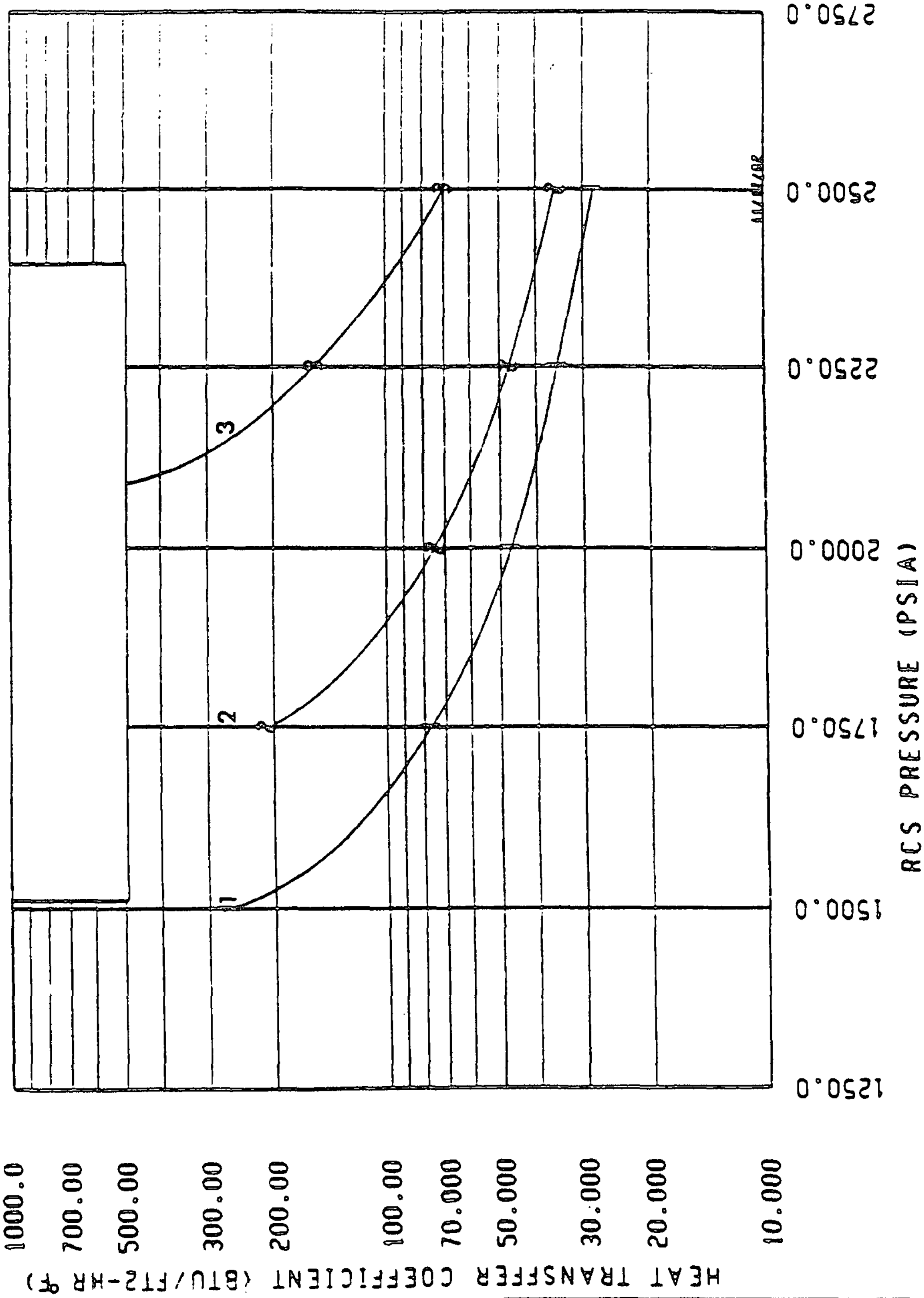

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FIGURE E.47-2

HYDROGEN BUBBLE PRESSURIZED
SECONDARY SIDE
% CLD RX 1 = 5%, 2 = 10%, 3 = 20%

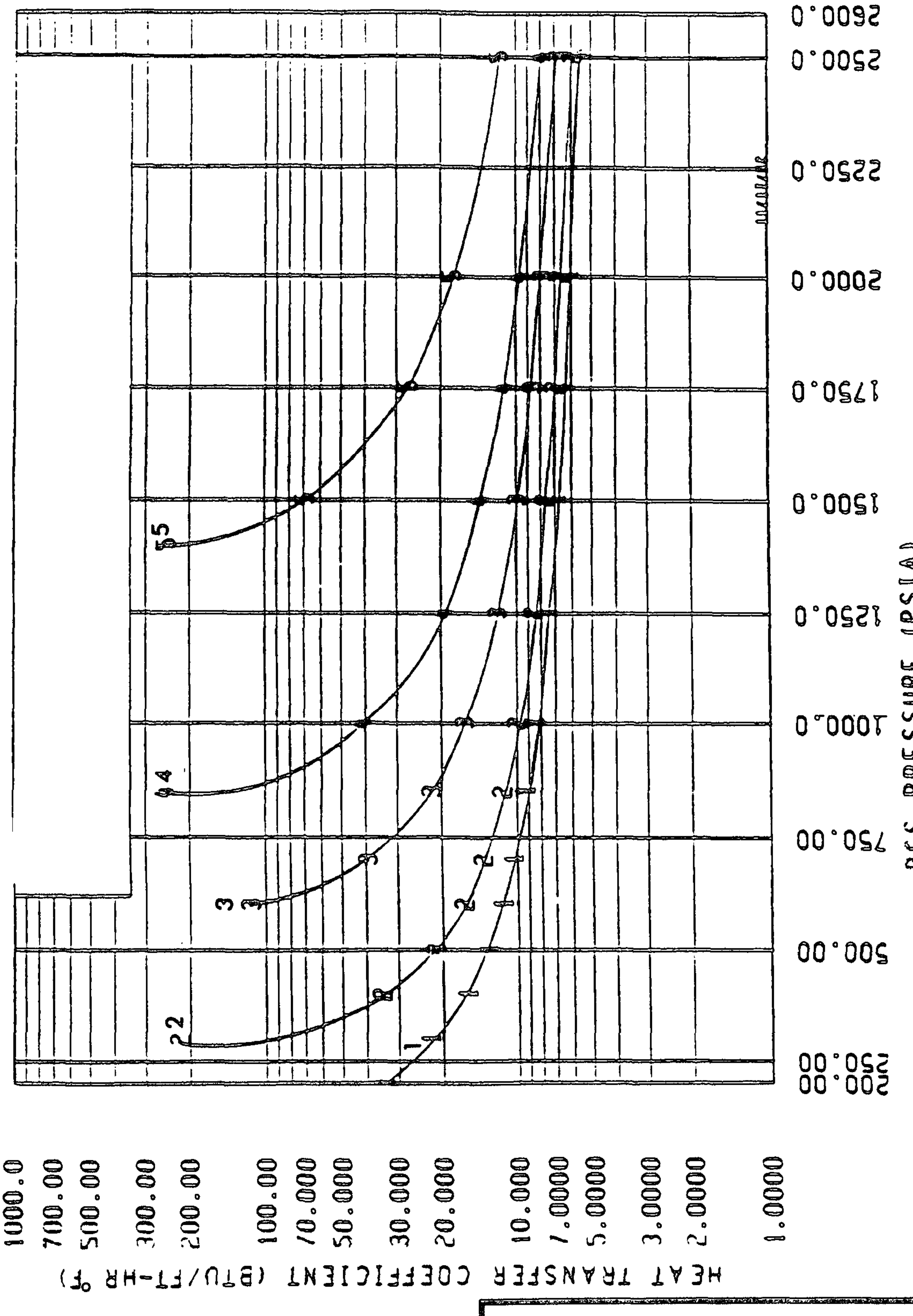

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FIGURE E.47-3

HYDROGEN BUBBLE DEPRESSURIZED
SECONDARY SIDE
% CLD RX 1 = 5%, 2 = 10%, 3 = 20%, 4 = 30%, 5 = 50%

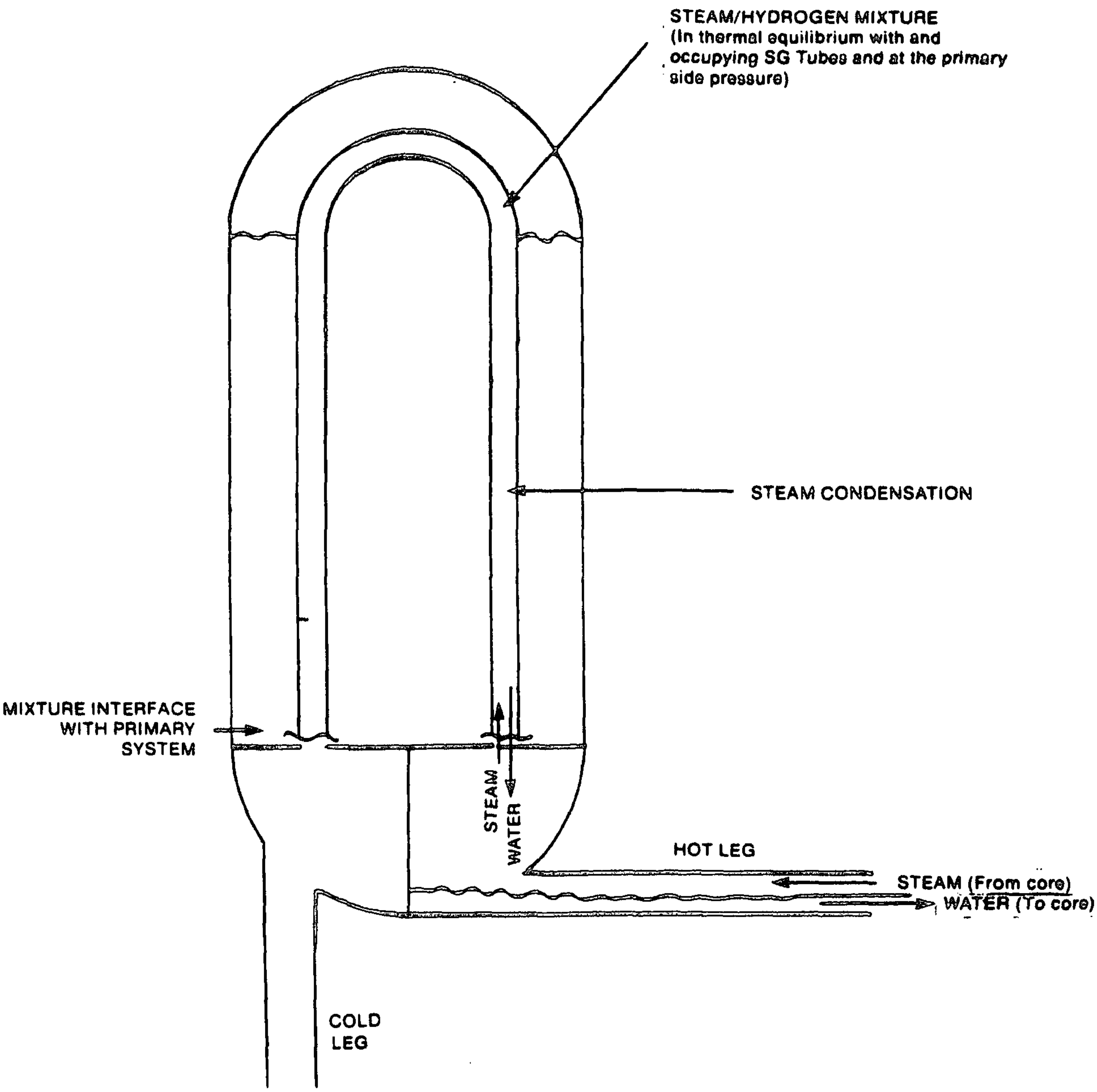

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FIGURE E.47-4

LOOP REFLUX ILLUSTRATION
HOMOGENEOUS MIXTURE

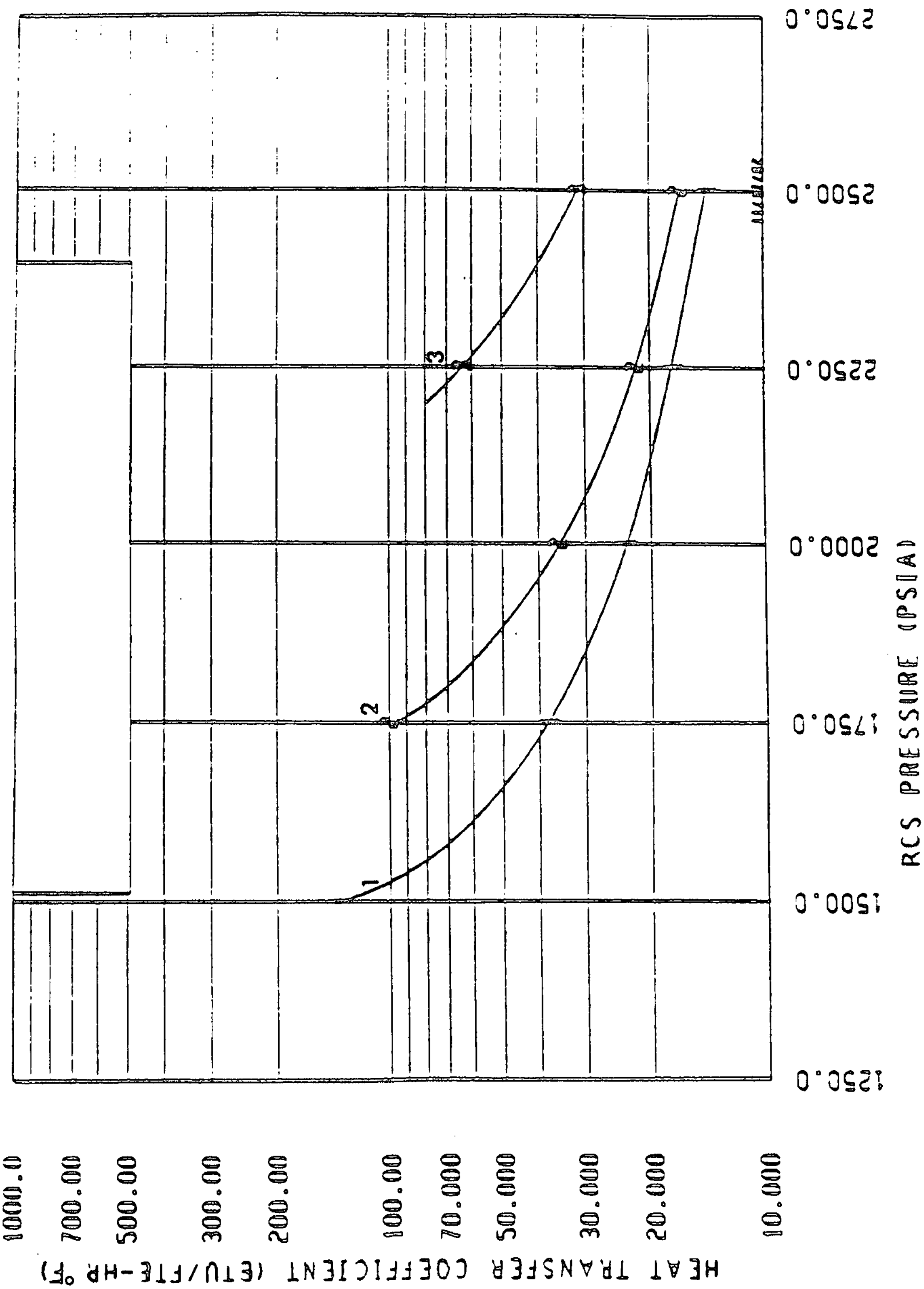

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FIGURE E.47-5

HOMOGENEOUS MIXTURE PRESSURIZED
SECONDARY SIDE
% CLD RX 1 = 5%, 2 = 10%, 3 = 20%

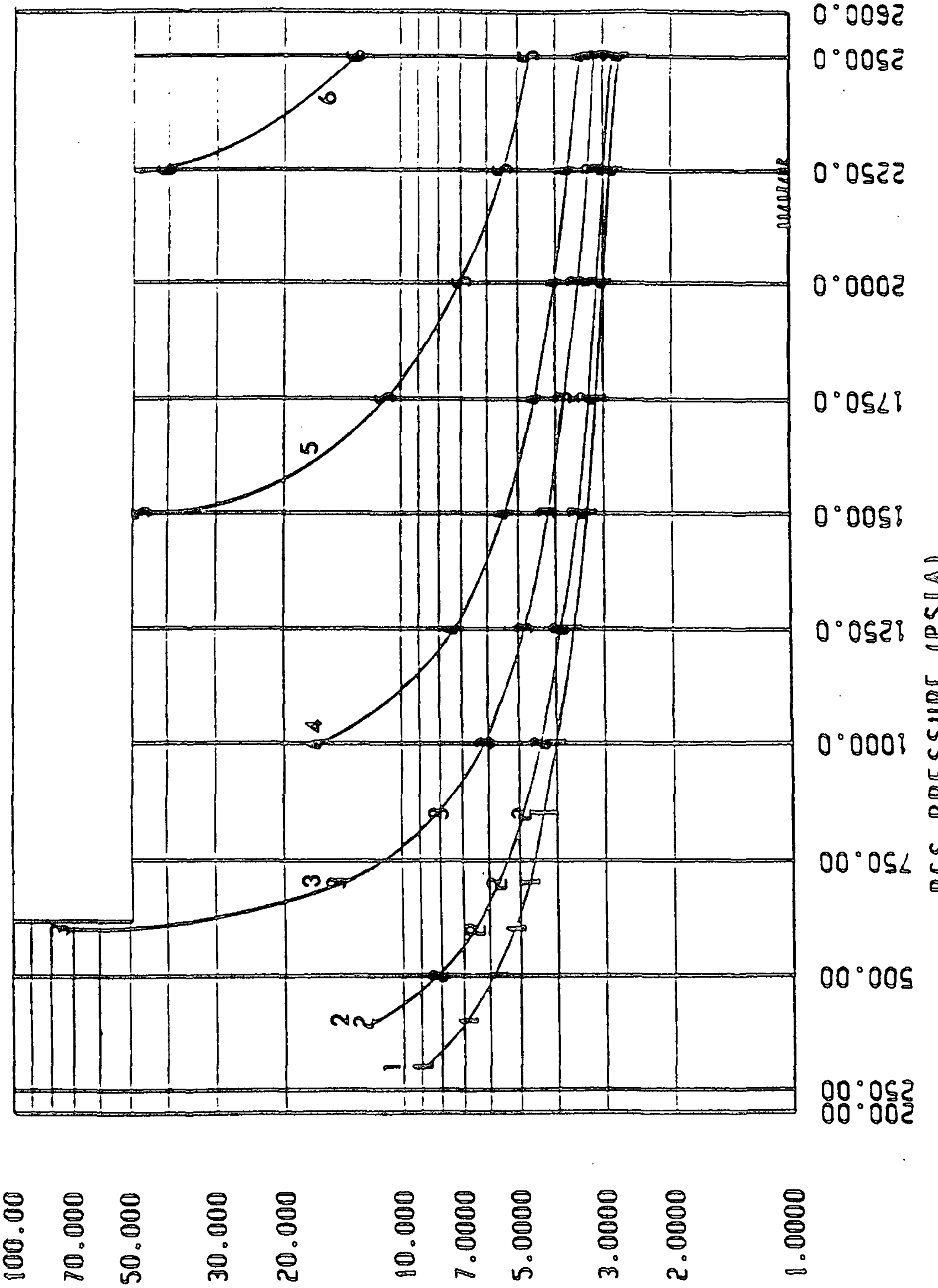

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FIGURE E.47-6

HOMOGENEOUS MIXTURE DEPRESSURIZED SECONDARY SIDE—% CLD RX 1 = 5%, 2 = 10%, 3 = 20%, 4 = 30%, 5 = 50%, 6 = 75%

Figures E.75-1 through E.75-3 have been deleted intentionally.