ERRATA AND ADDENDA SHEET
AMENDMENT 19, NOVEMBER 13, 1981

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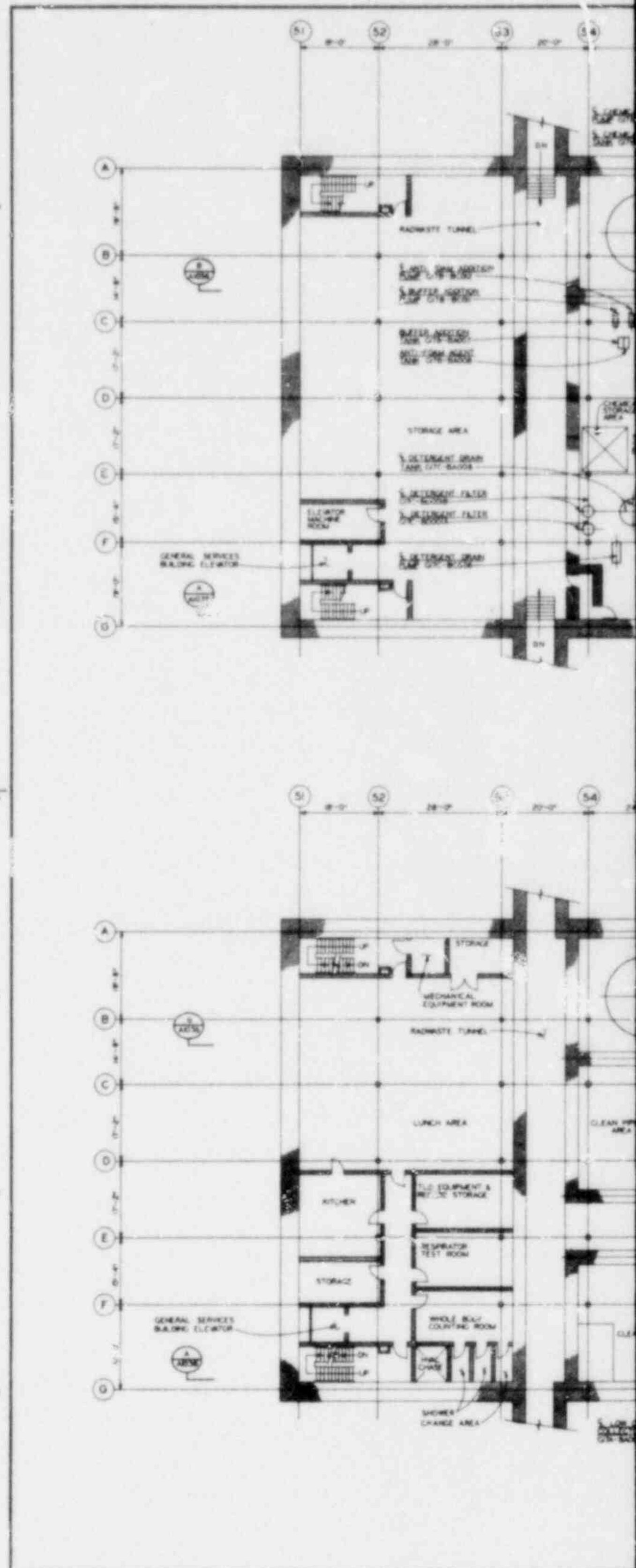

RADWASTE TUNNEL
STORAGE AREA
ELEVATOR MACHINE ROOM
GENERAL SERVICES BUILDING ELEVATOR
DN
UP
STORAGE
MECHANICAL EQUIPMENT ROOM
RADWASTE TUNNEL
LUNCH AREA
KITCHEN
STORAGE
RESPIRATOR TEST ROOM
WHOLE BODY COUNTING ROOM
SHOWER
CHANGE AREA
GENERAL SERVICES BUILDING ELEVATOR

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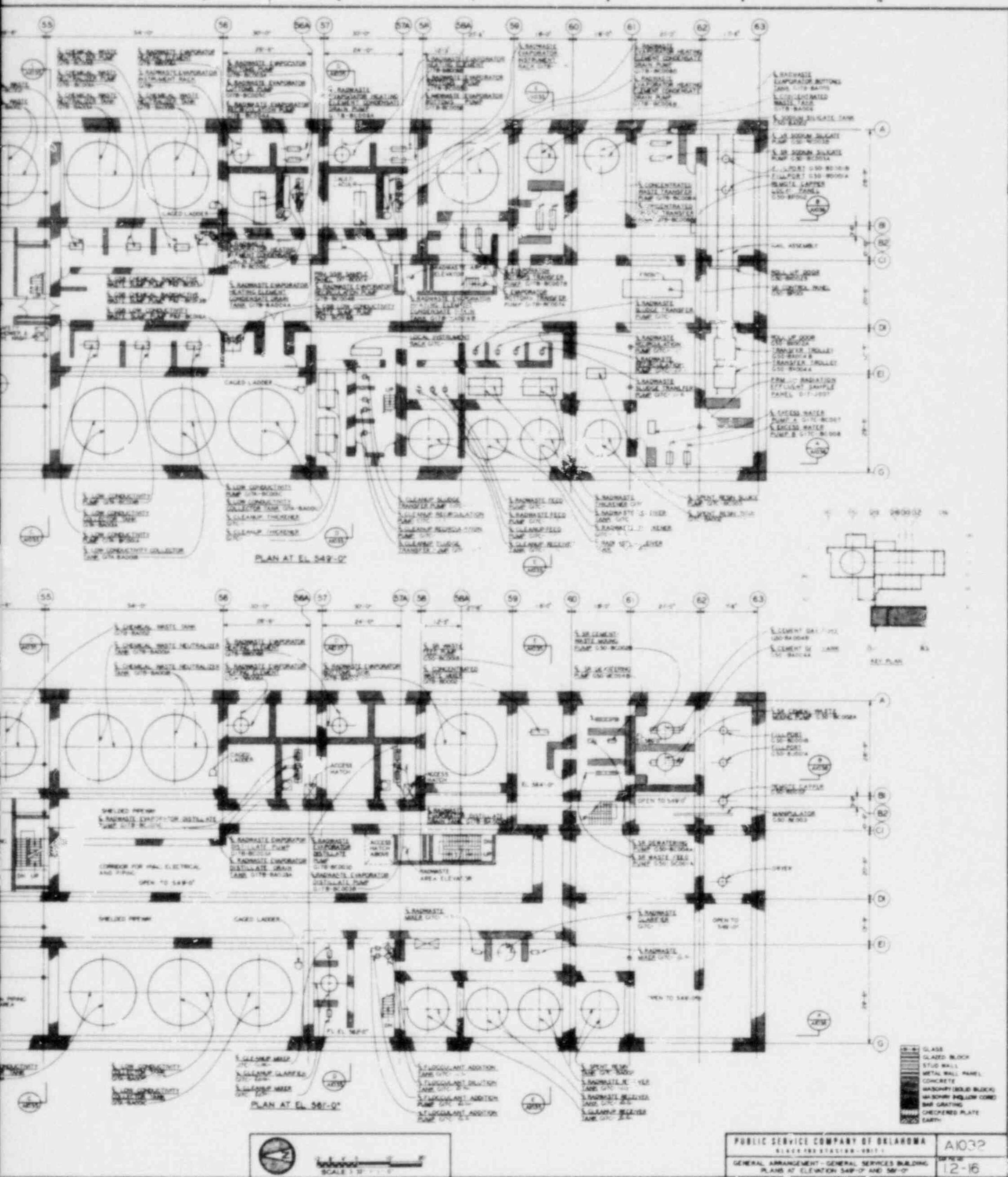

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GENERAL ARRANGEMENT-GENERAL SERVICES BUILDING 19
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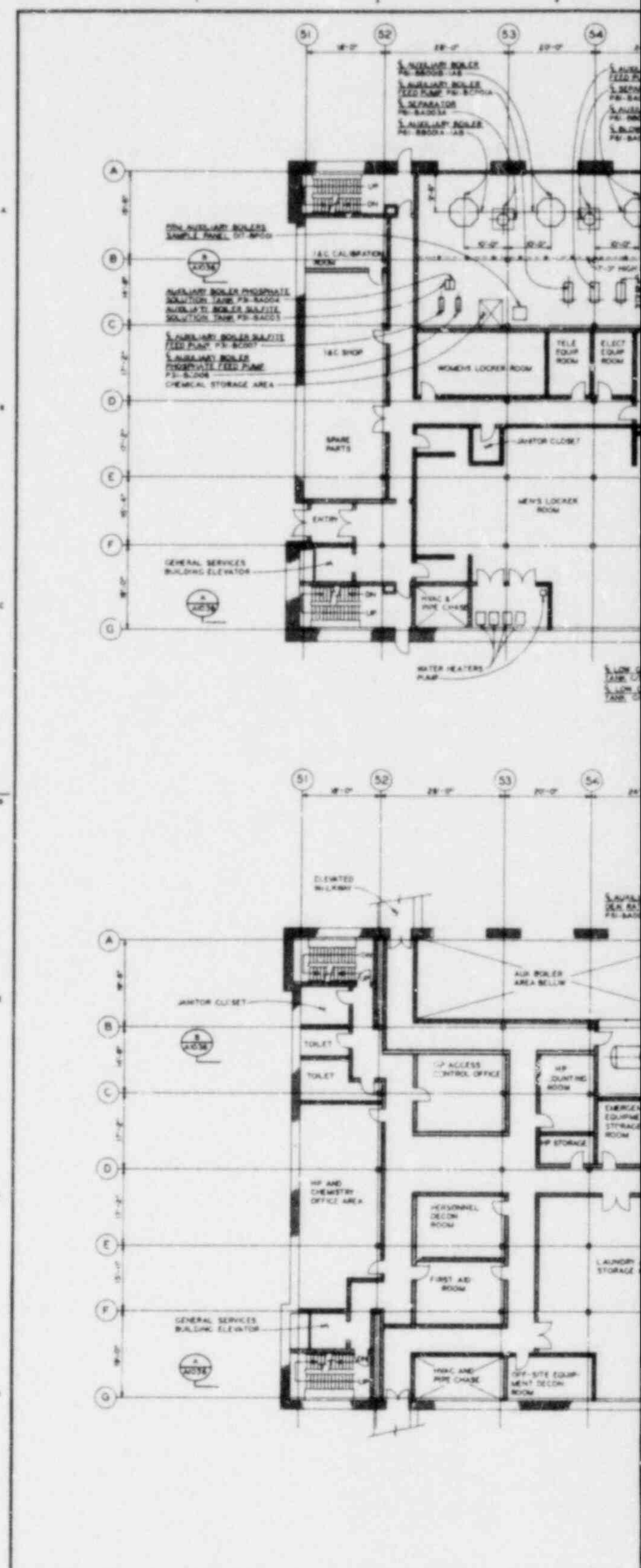

I&C CALIBRATION ROOM
I&C SHOP
CHEMICAL STORAGE AREA
WOMEN'S LOCKER ROOM
TELE EQUIP ROOM
ELECT EQUIP ROOM
JANITOR CLOSET
SPARE PARTS
MEN'S LOCKER ROOM
ENTRY
GENERAL SERVICES BUILDING ELEVATOR
HVAC & PIPE CHASE
WATER HEATERS PUMP
ELEVATED WALKWAY
AUX BOILER AREA BELOW
JANITOR CLOSET
TOILET
TOILET
HP ACCESS CONTROL OFFICE
HP COUNTING ROOM
HP STORAGE
HP AND CHEMISTRY OFFICE AREA
PERSONNEL DECON ROOM
FIRST AID ROOM
GENERAL SERVICES BUILDING ELEVATOR
HVAC AND PIPE CHASE
OFF-SITE EQUIPMENT DECON ROOM

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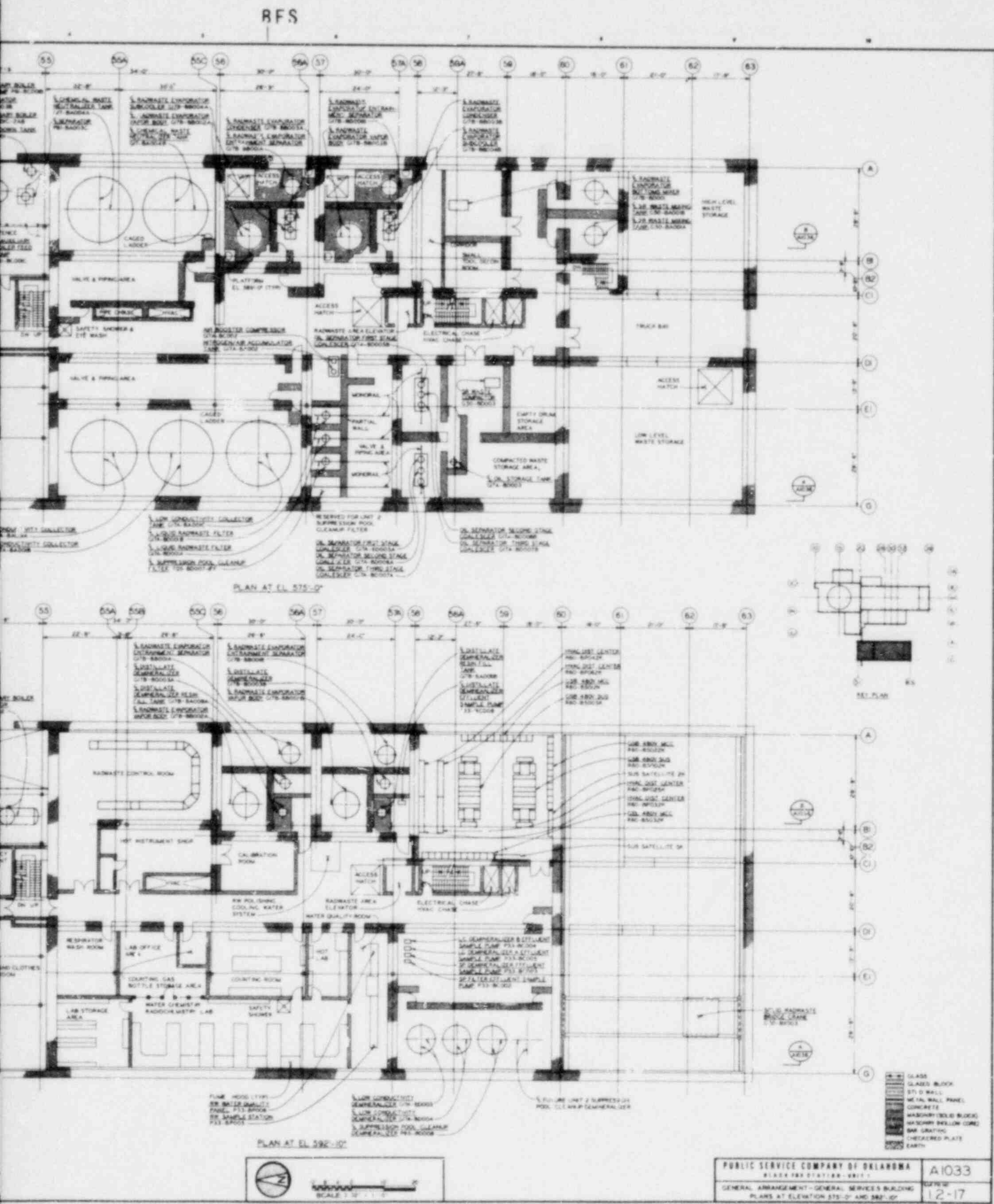

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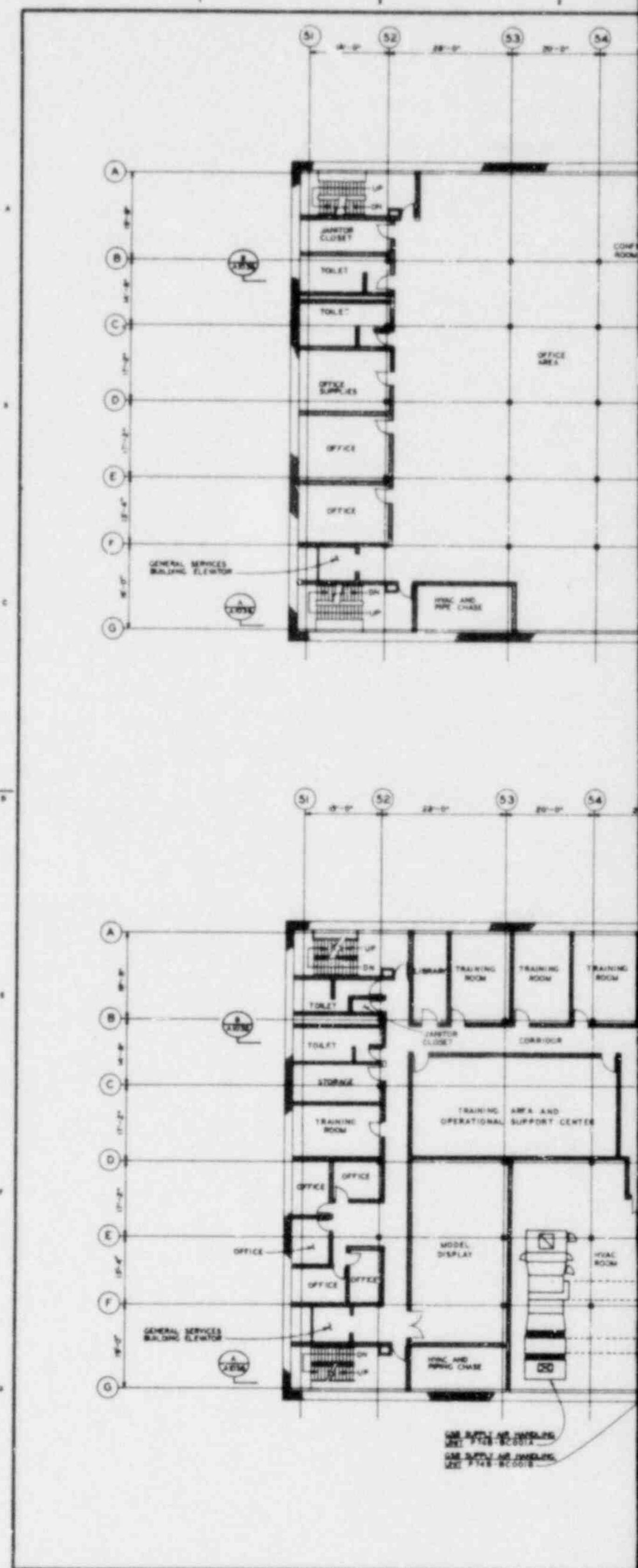
GENERAL ARRANGEMENT-GENERAL SERVICES BUILDING
PLANS AT ELEVATION 575'-0" AND 592'-10" 19

FIGURE 1.2-17



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52
53
54
A
B
C
D
E
F
G
UP
DN
JANITOR CLOSET
TOILET
OFFICE SUPPLIES
OFFICE
OFFICE AREA
GENERAL SERVICES BUILDING ELEVATOR
HVAC AND PIPE CHASE
LIBRARY
TRAINING ROOM
CORRIDOR
STORAGE
TRAINING AREA AND OPERATIONAL SUPPORT CENTER
MODEL DISPLAY
HVAC ROOM
HVAC AND PIPING CHASE

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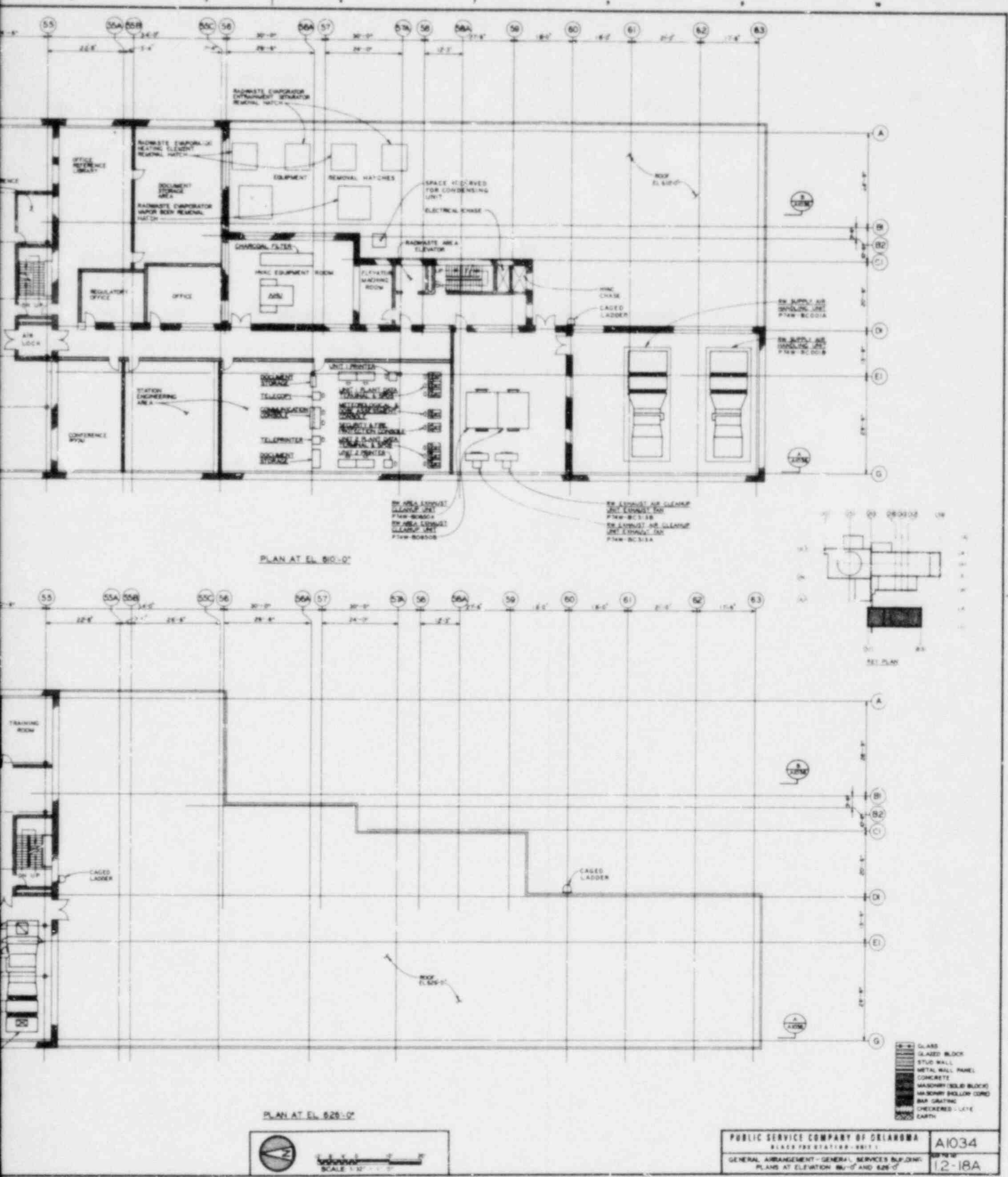

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GENERAL ARRANGEMENT-GENERAL SERVICES BUILDING
PLANS AT ELEVATION 610'-0" AND 626'-0"

FIGURE 1.2-18A

TABLE 1.9-1 (Continued)

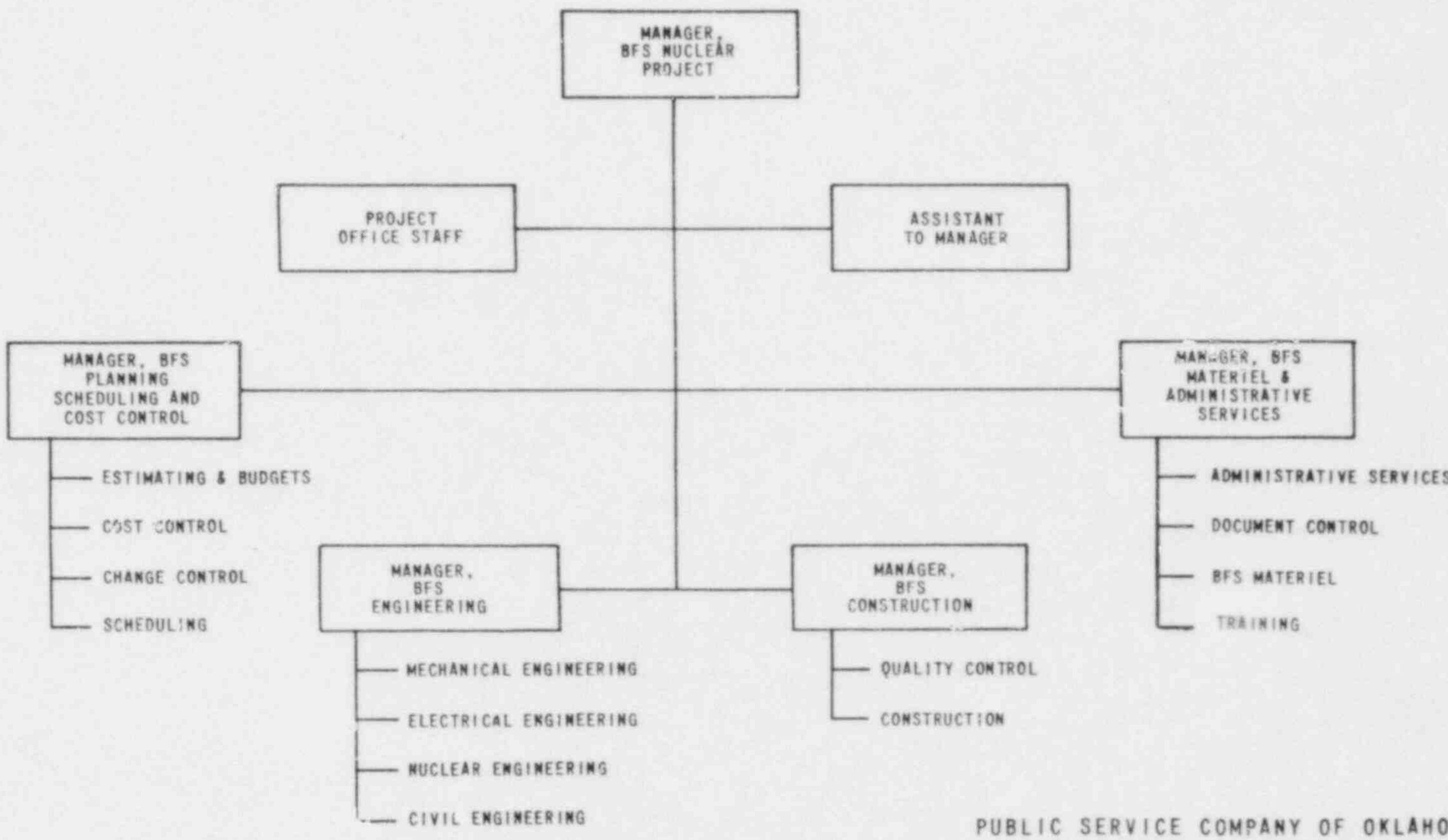
Regulatory Guide	Title and Applicant's Position
1.57 (Continued)	PSO will meet the provisions of this guide with the following clarifications:

(1) The load combinations and stress limits for the steel portions of the drywell, such as the drywell head and hatch covers, will be designed in accordance with Article NE, Section III, of the ASME Code.

(2) Those portions of the guard pipes which form part of the containment boundary are classified as Class MC and designed by the rules of Article NE, Section III of the ASME Code.

(3) The steel containment vessel is classified as a Class MC component and is designed according to the provisions of Subsection NE, Section III, of the ASME Code except with respect to the provisions of position C.1.b. (2). Instead of the normal design limits specified in NE-3131(b) for the accident recovery flooded condition plus OBE, the containment vessel will be designed according to the stress intensity limits established in PSAR Subsection 3.8.2.3.12. 11 11

Regulatory Guide	Title and Applicant's Position
1.58	Qualification Of Nuclear Power Plant Inspection, Examination, And Testing Personnel (Rev. 1, 10/80) 3
	PSO will meet the provisions of this guide, with the exception of regulatory position C.6. PSO will require written proficiency examinations for personnel who do not have a high school diploma or GED equivalent. Q411.9 17 19
1.59	Design Basis Floods For Nuclear Power Plants (Rev. 1, 4/76)
	PSO has complied with the provisions of this guide. See Section 2.4 for details.
1.60	Design Response Spectra For Seismic Design Of Nuclear Power Plants (Rev. 1, 12/73)
	PSO will meet the provisions of this guide. See Subsection 2.5.2 for details.
1.61	Damping Values For Seismic Design Of Nuclear Power Plants (Rev. 0, 10/73)
	PSO will meet the provisions of this guide. See Section 3.7 for details.
1.62	Manual Initiation Of Protective Actions (Rev. 0, 10/73)
	(GESSAR)
1.63	Electric Penetration Assemblies in Containment Structures For Water-Cooled Nuclear Power Plants (Rev. 0, 10/73)
	See Subsection 3.8.6.2 for details.



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PUBLIC SERVICE COMPANY OF OKLAHOMA
BFS NUCLEAR PROJECT ORGANIZATION

FIGURE 13.1-2

17A.1.1.13 Manager, Environmental and Chemistry Control. The Manager, 17
Environmental and Chemistry Control is responsible to the Vice President,
Power Generation for all matters pertaining to in-plant chemistry and cor-
porate environmental affairs.

17A.1.1.14 Manager, Nuclear Licensing. The Manager, Nuclear Licensing is
responsible to the Vice President, Power Generation for the day-to-day 17
maintenance of licensing activities. He also serves as liaison with the NRC
for both safety and environmental licensing. He is directly responsible for
preparation and submittal of all applications to Federal, state, and local
agencies which are required for construction permits and operating licenses, 19
and for advising project personnel of current regulations.

17A.1.1.15 Manager, Quality Assurance. The Manager, Quality Assurance, is
responsible to the Vice President, Power Generation for the preparation and
management of the PSO QA Program and for surveillance and follow-up of pro-
gram implementation. This responsibility extends to all project activities
including design, procurement, construction, preoperational testing, and 10
operations.

The Manager, Quality Assurance, has been delegated the authority and provided the organizational freedom to identify problems and to initiate, recommend, provide solutions, and verify implementation of solutions. He is delegated the authority to oversee the execution and implementation of the QA Program and to perform both internal and external audits as necessary to assure a safe and reliable facility. He has written authority to stop use of unacceptable or unapproved purchase documents, procedures, or instructions and to prevent the continuation of activities performed by PSO or contractors, including construction site activities, which would tend to
degrade the quality of the structures, systems, and components 17
important to safety. He is responsible to assure, through QA audits and surveillance activities, that verification of conformance to established quality requirements is accomplished by individuals or groups who do not have direct responsibility for performing the work being verified. He has delegated to his staff the authority to carry out the duties assigned to

17A.1.2.2 Structures, Systems, and Components Important to Safety. Section 3.2 of this PSAR identifies and classifies the structures, systems, and components in accordance with their importance to safety. 17
10CFR50, Appendix B applies to the PSO QA program for all items identified as "B" in Table 3.2-1 (GER). For all items identified as "S", PSO will apply 19
appropriate portions of the PSO QA program.

17A.1.2.3 Timely Initiation of the Quality Assurance Program

17A.1.2.3.1 Initial Activities. Prior to the submittal of this PSAR, PSO initiated quality assurance activities as indicated below.

(1) Quality Assurance Training
   - (a) Gray Book conference, San Francisco, attended by Manager, QA, and Assistant Vice President-Nuclear, 1973.
   - (b) Orange Book conference, Denver, attended by Manager, QA, 1973.
   - (c) Green Book conference, Denver, attended by Manager, QA, 1974.
   - (d) Five 2-day management meetings conducted by EEI-QA Task Force, attended by Manager, QA, 1973, 1974, 1975.
   - (e) One-day QA management seminar conducted for 54 PSO middle and upper management personnel, 1974.
   - (f) ASQC, 28th Annual Technical Conference, Boston, attended by Manager, QA, 1974.
   - (g) L. Marvin Johnson 5-day audit technique for QA effectiveness seminar, Washington, attended by QA Engineer, 1974.

(2) Pre-award Survey
   - (a) GE Nuclear Steam Supply System, San Jose, California, 1974.
   - (b) GE Nuclear Fuel Fabrication, Wilmington, North Carolina, 1974.

(3) Audits
   - (a) Shannon & Wilson, Core Boring, Field Operations and Records, 1974.
   - (b) Meteorology Research, Inc., Records and Procedures, 1974.

by PSO personnel in the review and management of design, procurement, and construction activities for BFS structures, systems, and components important 17 to safety.

Equipment suppliers and contractors for PSO are required to have written QA programs, procedures, methods, and operating instructions to carry out and verify that specified activities and the quality of materials and workmanship are properly achieved and controlled. PSO or its representatives will include provisions within the overall auditing and surveillance program to assure that those organizations performing specific quality-affecting activities have their respective activities and procedures under control. 10

The testing organization which will perform the preoperational testing at the end of the construction phase is discussed in Chapter 13 (TEST WORKING GROUP) of this PSAR. The technical and quality responsibilities related to the transfer of systems, components, and structures after construction are included in Chapter 13.

17A.1.2.8 Management Review. The Manager, Quality Assurance, will provide the Vice President, Power Generation with a copy of all internal audit reports. Any disputes arising from the implementation of the QA program will be finally resolved by the Executive Vice President. His decision will be 17 documented in a memorandum to the affected responsible parties who will change, modify, or amplify existing procedures, instructions, and manuals to reflect the resolution.

The Manager, Quality Assurance is responsible for performance of audits of the prime contractors' QA programs. He reviews the prime contractors' QA programs to determine that adequate provisions are established for management 19 review and he conducts audits to ensure that the review procedures are in fact being implemented.

A Review and Audit Committee (RAC) will function throughout the design, construction, and operational phases of Black Fox Station (Chapter 13). The 10 Manager, BFS Nuclear Project will serve as chairman of the committee through the Design and Construction phases. Members of the committee will be the 3 Executive Vice President; the Manager, Black Fox Station; the Manager, BFS Q411.6 17 Engineering; the Manager, Quality Assurance; the Manager, BFS Planning, Scheduling and Cost Control; the Vice President, Power Generation; and the Vice President, Materiel and Property Management. The Supervisor, perform surveillance as necessary at the manufacturers' plants and, through the Manager, BFS Construction, will perform receipt inspection and surveillance inspection during installation at the construction site. 10 Inspection will be performed by experienced QC personnel who are 19 cognizant of applicable codes and standards and specification requirements. QA will approve the training requirements of QC personnel and through administration of their certification program will review and verify that the training requirements have been met.

The manufacturers and installers of equipment and materials 17 important to safety will be required to have inspection programs implemented 10 by personnel qualified in accordance with the applicable codes to determine that the specified quality requirements are met.

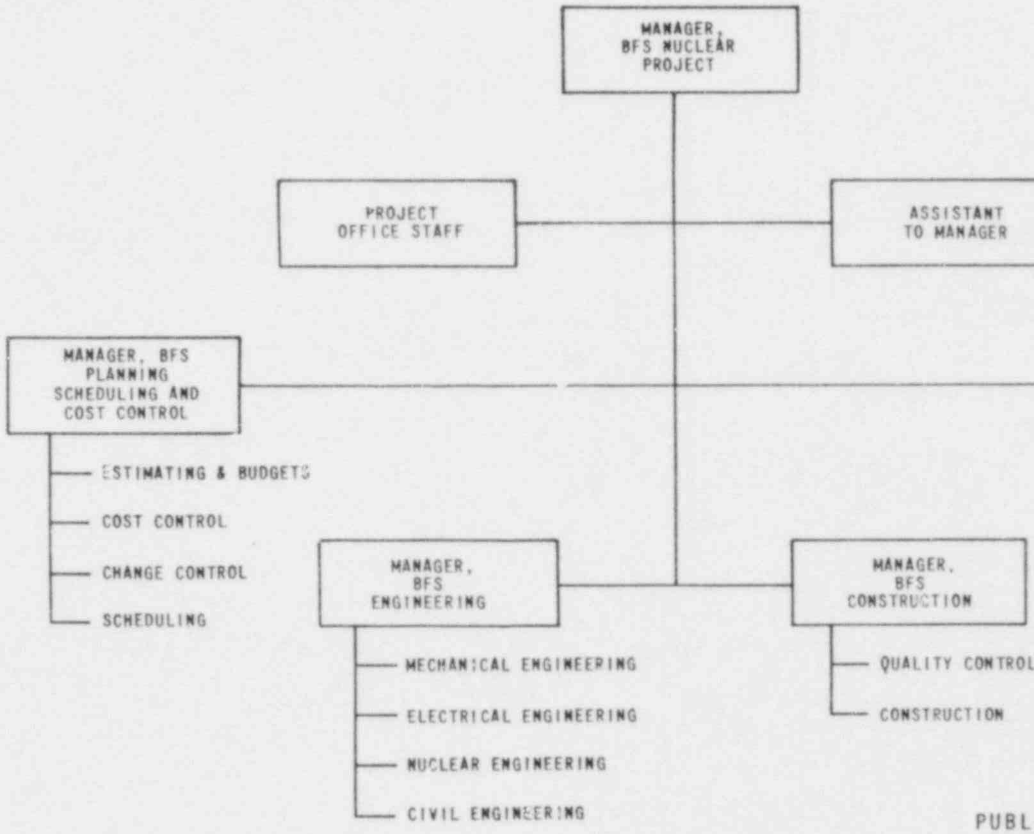
The Manager, BFS Construction (Appendix 17D) will be responsible to verify that materials and equipment delivered to the construction site conform to the purchase requirements and that objective evidence of the quality 17 of the delivered items accompanies the item. QA will review and approve the results of inspection prior to its installation or use. The Manager, BFS Construction will also be responsible to verify that the installation requirements are met by the site contractors. 10

The Manager, Quality Assurance, or his representatives will have access to all manufacturing, fabrication, and construction activities and will perform audits as necessary to assure compliance with contractual requirements. Inspection hold-points will be established as appropriate in procedures and purchase specifications.

17A.1.11 Test Control

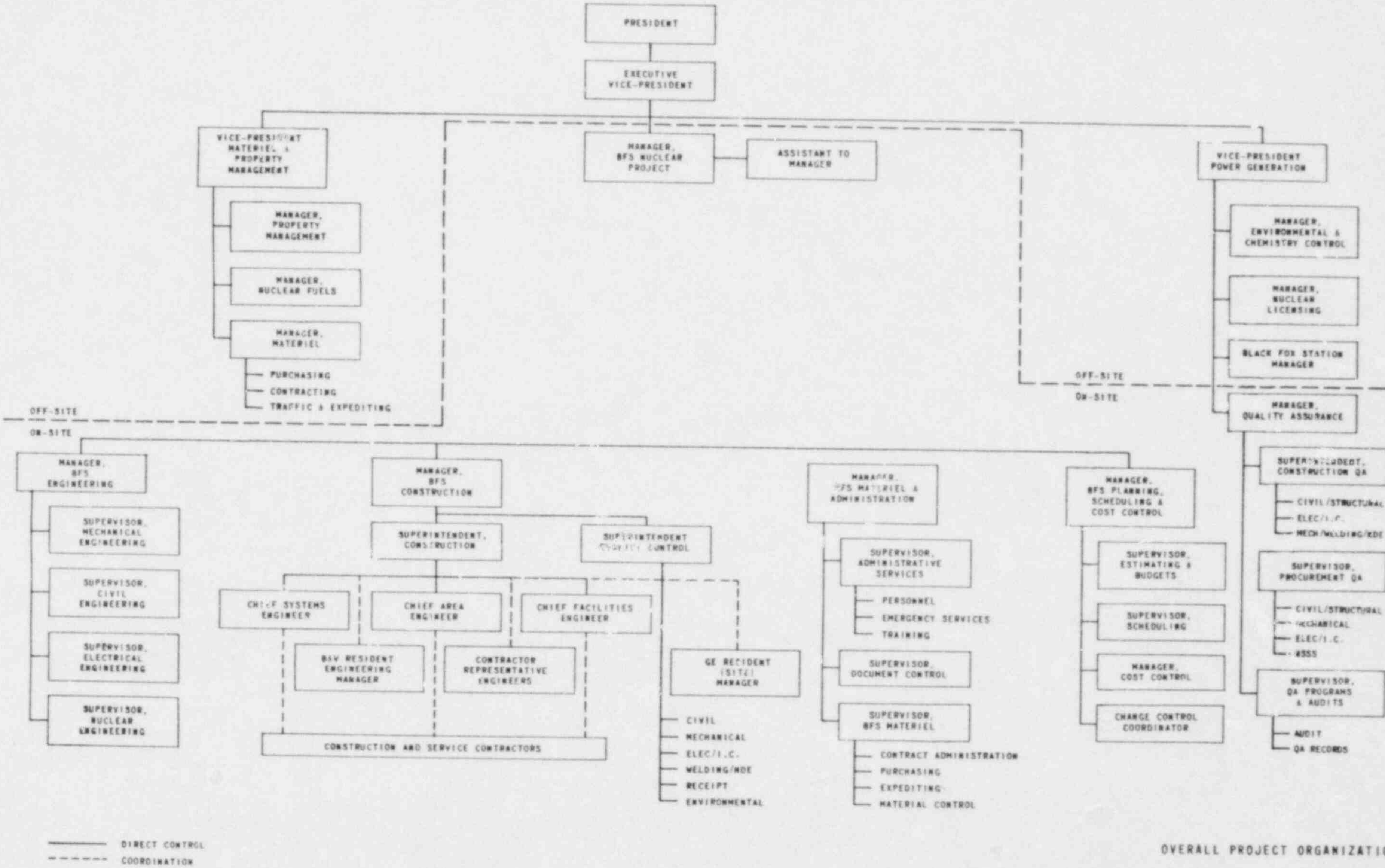
17A.1.11.1 Equipment Testing. The suppliers and contractors furnishing components will be required to have test programs to control the quality 10 of the equipment they supply. These test programs will include mechanical tests, performance tests, and other functional tests to be accomplished on subcomponents or to be performed after manufacture of the component as appropriate. PSO will review test programs to assure that the contractors and suppliers have provisions for the appropriate testing of specified equipment. PSO audit programs will include provisions to assure that the contractors and suppliers accomplish the appropriate and specified tests in accordance with contract requirements.

17A.1.11.2 Construction Testing. Completed systems or subsystems within the plant that require construction testing (i.e., hydrostatic testing,



PUBLIC SERVICE COMPANY OF OKLAHOMA
BFS NUCLEAR PROJECT ORGANIZATION

FIGURE 17A.1-2



OVERALL PROJECT ORGANIZATION

FIGURE 17A.1-3

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(3) Performing reviews of design and procurement documents for quality aspects. 5

(4) Processing Nonconforming Item Reports and Corrective Action Requests that are project related. Q411.33A

Administratively, the Project Quality Control Engineer(s) are responsible to the Division QA Manager and, thus, may obtain resources of the Division QA Group in the execution of their responsibilities.

17B.1.2 Quality Assurance Program

17B.1.2.1 Conformance of Quality Assurance Program to Regulatory Requirements.

The Quality Assurance Program - Nuclear conforms to the applicable provisions of the following.

(1) Regulatory Guide 1.28-June 1972 (ANSI N45.2 - 1971).

(2) Regulatory Guide 1.64-June 1976 (ANSI N45.2.11 - 1974).

(3) Regulatory Guide 1.74-February 1974 (ANSI N45.2.10 - 1973). 11

(4) Regulatory Guide 1.88-December 1975 (ANSI N45.2.9 - 1974).

(5) Regulatory Guide 1.123-July 1977 (ANSI N45.2.13-1976).

(6) ANSI N45.2.12 (Draft 3, Revision 4) - February 1974.

(7) Regulatory Guide 1.146-August 1980 (ANSI N45.2.23 - 1978). 19

The Project Manager is responsible to see that regulatory guides and industry standards are properly translated into appropriate policies, procedures, and other documents necessary to accomplish quality-affecting activities of the Black & Veatch Power Division for a specific project in a controlled manner.

The details of the Quality Assurance Program-Nuclear and program application to be implemented on the BFS project and the means for implementing procedural details are contained in the Quality Assurance Program-Nuclear Standard Procedures. The Quality Assurance Program-Nuclear

(l) Regulatory Guide 1.123, July 1977 (ANSI N45.2.13-1976) 11

(m) ANSI N45.2.12 (Draft 3, Rev. 4) February 1974

(n) Regulatory Guide 1.146, August 1980 (ANSI N45.2.23-1978). 19

(3) Defined accept-reject criteria when not a part of the referenced industry codes and standards.

(4) Requirements for supplier document submittals such as instructions, procedures, drawings, specifications, inspection and test results, and other supplier documentary evidence of quality.

(5) Requirements for submittal or retention, control, and maintenance of quality assurance records.

(6) Statements as to rights of access to the suppliers' facilities and working documents for inspection and audit.

APPENDIX 17C QUALITY ASSURANCE

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17C.1 GENERAL ELECTRIC COMPANY (NEDO-11209-04A, Revision 2) 19

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manpower requirements for area, system and construction facilities to provide a smooth transition from area control to system control at the appropriate time.

It is recognized that early in the construction cycle, area control will be dominant as foundations, slabs, buildings and basic components are installed. As components become part of an integrated system and made ready for functional checking and testing, control of the construction effort will shift from area control to system control. 19

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17D.1.1.3 Chief Area Engineer, Chief Systems Engineer, Chief Facilities Engineer. The Chief Area Engineer and the Chief Systems Engineer will assign responsible field engineers to provide the detailed interfacing necessary for the contractors involved and to coordinate the schedule; engineering changes; equipment changes; material expediting; items found 17
in nonconformance, contract changes as advised by the Supervisor BFS Materiel; recommendations received from contractors; and other information pertinent to the conduct of the job.

As directed by the Superintendent, Construction, the Chief Facilities Engineer will provide the necessary interface with the contractors for required facilities and will provide the assistance that may be required by the contractor.

17D.1.1.4 Supervisor, BFS Materiel. The Supervisor, BFS Materiel reports to the Manager, BFS Materiel and Administration. He supervises the contract administrators and provides interpretation of the various procurement docu- 17
ments to the Superintendents and Supervisors directly involved in the construction efforts with various contractors. The Supervisor, BFS Materiel, through the contract administrators, shall review the progressive activities of each contractor to assure the contractor is performing the required work activity in accordance with contractual requirements. In addition, the Supervisor, BFS Materiel will assure the contractor is furnishing appropriate reports and backup data with accompanying invoices and estimates of completion. The Supervisor, BFS Materiel shall be responsible for additions to and deletions from contracts, as well as provide guidance to all parties involved regarding the applications of pricing for the additions or deletions. If a question arises concerning the quality of

Development and implementation of contractor training programs 19
will be the responsibility of each respective contractor. Each contractor shall furnish the necessary documentation to verify craftsmen skill level for special processes and their latest qualifications.

17D.1.2.4 Review and Evaluation of the Quality Control Program.

The Quality Assurance Organization shall audit FPM and contractor quality 17
control programs to assure that field activities affecting the quality of items important to safety are accomplished under controlled conditions. The results of PSO quality assurance audits will be properly documented, and performed with sufficient frequency and depth to assure adequacy of the scope, implementation, and effectiveness of quality control programs.

17D.1.3 Design Control

Project design is the responsibilitiy of the Engineer, the NSSS vendor and other manufacturers of equipment subject to review and approval by PSO. 3
Modifications will be controlled and documented by appropriate requests, Q411.42
notifications, and disposition approval which will be processed and reviewed by the Engineer. The FPM organization and the Engineer's site organization will evaluate design changes and insure that they are reviewed and approved by the same or equally qualified organization which originally processed the design.

The Manager, BFS Construction through his staff will monitor contract activities to assure that items have been installed in accordance with approved design documents and code requirements. The auditing of design control activities at the site will be the responsibility of Quality Assurance.

The Supervisor, Document Control under the Manager, BFS Materiel and Administration, will control and issue drawings and reproducibles and other information to all parties for use in construction to assure that the latest revisions of documents are available for use. Each contractor shall be required to submit his drawing control procedure to the FPM for approval.
Each contractor shall reproduce, issue and control all drawings for his area 17
of responsibility. The Superintendent, Quality Control will be responsible for the coordination of the evaluation and resolution of

nonconformances at the construction site and verifying corrective action is
taken as required. The Chief Area Engineer will be responsible for imple-
menting changes in the field. 19 17

17D.1.4 Procurement Document Control

Construction specifications and equipment procurement documents will
generally be prepared and controlled by the Engineer as described in Sub-
section 17B.1.4.

The Manager, BFS Materiel and Administration will be responsible to
place into effect and to enforce procedures for preparation, review and
approval of field procurement change orders and any field originated procure-
ments. Field originated procurement documents will be prepared by the Super- 17
visor, BFS Materiel and approved by the Manager, BFS Construction.

Field originated change orders and procurement documents shall
be reviewed by the Quality Assurance Organization to ensure applicable
provisions for inspectability and controllability, adequate acceptance-
rejection criteria, and proper review and approval of these documents.
Originals or copies of all procurement and change order documents for 3
items important to safety will be transmitted to the document Q411.43
control organization to be maintained as quality assurance records.
Revisions of field originated procurement requisition documents will go
through the same review and approval processes as the original documents.

17D.1.5 Instructions, Procedures, and Drawings

All FPM and construction contract activities which affect quality of
items important to safety will be performed in accordance with 17
documented procedures. By delegation from the Manager, BFS Nuclear Project,
the Manager, BFS Construction will approve all FPPM procedures, including
revisions. The procedures will provide 3
that quality affecting activities conform to applicable codes and standards
and will include appropriate quantitative or qualitative criteria for deter-
mining that the work has been accomplished satisfactorily, is acceptable
and has been properly documented. It is the responsibility of the Q411.44 & Q411.45
Quality Assurance Organization to perform a documented review of site
procedures and changes to insure conformance with quality requirements and to 17

(3) Inspection records or certificates of conformance to be supplied by each vendor are reviewed by the Quality Assurance organization prior to installation or use of equipment. 3 Q411.46 17

(4) Items accepted and released for installation or use are identified by proper inspection status tags or records traceable to the item. 19

Record files will be maintained by the contractors describing the processes performed, the procedures followed, qualification of personnel and procedures used in performing the process. Copies of such records will be submitted to the PSO QA organization for review to verify satisfactory 17
accomplishment of the process.

17D.1.10 Inspection

Receipt inspections to verify that items important to 5 17
safety purchased directly by PSO comply with purchase specifications shall be performed by the Superintendent, Quality Control in accordance with approved procedures. Construction contractors will be required to establish a quality control inspection program which will assure that activities within each contractor's scope of work conform to specification requirements and documented instructions, procedures, and drawings. Contractors' inspectors will be required to be appropriately trained, qualified and independent of the craftsmen performing the activities being inspected. Q411.48 & Q411.55

Inspections will be performed in-process as necessary to verify the 19
required quality, and will be performed according to documented instructions, procedures, and checklists which contain the following:

(1) Identification of characteristics to be inspected.

(2) Indentification of the individuals or groups responsible for performing the work as well as for performing the inspection operation.

(3) Acceptance and rejection criteria.

(4) A description of the method of inspection.

(5) Verification of completion and certification of in-process and final inspections.

(6) A record of the results of the inspection operation.

QA will review inspection procedures to verify they contain, as appropriate, the criteria above. 17
Area engineers will assist contractors in obtaining technical information and in interpretation and application of codes, standards, drawings and specifications as a means of assuring proper application by the contractors.

Quality Control. The Superintendent, Quality Control shall review the NR for possible stop work action and for issuance to affected parties for action.

Project personnel will coordinate and expedite resolution of each NR with 3
the affected parties except those dispositioned "scrap, return to vendor" Q411.52 17
(contractor furnished material). BFS Engineering and Quality Assurance shall approve the resolution of all quality-related nonconformances at the site.

All NR's will be maintained as QA records. 19

The Superintendent, Quality Control will be responsible to verify that equipment and component nonconformances which are reworked or repaired are reinspected to the original quality standards and that results are properly evaluated and documented. Copies of completed NR's will be transmitted to the affected vendors and contractors and NR records will be forwarded 17
to management and periodically analyzed by QA for quality trends.

17D.1.16 Corrective Action

The Manager, BFS Construction will be responsible for promptly enforcing necessary corrective actions by vendors and site contractors.

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The Manager, Quality Assurance will be responsible for insuring that corrective action is initiated in a timely manner, for verifying the adequacy of 3
corrective action taken, for insuring that the corrective action taken precludes recurrence of conditions adverse to quality, and for assuring Q411.53
that each corrective action documentation sequence is properly closed out and signed off.

The Manager, BFS Construction will be responsible for reporting construction conditions or practices which may adversely affect quality, and for recommending corrective actions to the Superintendent, Quality Control. The Superintendent, Quality Control will initiate corrective action requests and forward them to the Quality Assurance Organization for issuance to the affected contractor.

17

A nonconforming item, design deficiency or any other condition which is considered a significant deficiency as defined by 10 CFR 50.55 (e) will be reported for evaluation by the Manager, BFS Nuclear Project, Manager, 17
Licensing and the Manager, Quality Assurance. If the deficiency is judged as significant, it will be reported to the Nuclear Regulatory Commission as stated in Subsection 17A.1.16.

17D.1.17 Quality Assurance Records

17D.1.17.1 Extent of Quality Assurance Records. The quality assurance 17
records system developed by QA will be defined, 19

implemented and enforced in accordance with written procedures and instructions. QA will receive, identify and review all documentation of field activities affecting quality. The control of the QA record system, 19
including site generated records and records received at the site which are within the scope of ANSI N45.2.9, is the specific responsibility of the 17
Manager Quality Assurance.

17D.1.17.2 <u>Identification and Retrievability of Records</u>. Each document will be required to have a unique identification number and will be further indexed and coded in accordance with appropriate contract and/or subject category identifications in order to facilitate retrievability. QA records will also be identifiable and traceable through the identification system to the devices or plant systems with which the documents are associated.

17D.1.17.3 <u>Maintenance of Records</u>. To prevent the possibility of loss or destruction, QA records will be maintained in a secure file with access controlled by the Supervisor, Document 17
Control and his assistants.

17D.1.17.4 <u>Content of Inspection and Test Records</u>. The content of inspection and test records prepared at the construction site is described in Subsections 17D.1.10 and 17D.1.11.

17D.1.18 <u>Audits</u>

Construction contractors will be required to perform internal audits to verify conformance to their respective quality programs. The PSO Manager, Quality Assurance or his representatives will periodically audit selected site activities carried out by the FPM and site contractors in accordance with the provisions of Subsection 17A.1.18.

ADDENDUM II

ADDITIONAL TMI-RELATED REQUIREMENTS

ADDENDUM II

Additional TMI-Related Requirements

Addendum II to the Black Fox Station (BFS) Preliminary Safety Analysis Report (PSAR) identifies the Applicant's commitments regarding the design, construction, and operation of the BFS in response to the accident at Three Mile Island, Unit 2.

In accordance with the NRC Staff guidance contained in the July 14, 1981 generic letter (Generic Letter No. 81-26) to all pending construction permit and manufacturing license applicants, Addendum II consists of responses to the requirements embodied in a new paragraph (e) to 10 CFR 50.34, entitled "Additional TMI-Related Requirements." 18

Commitments contained in Addendum II supersede any conflicting statements elsewhere in the PSAR where such conflicting statements were made earlier than the date of the current revision of Addendum II.

ADDENDUM II

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10 CFR 50.34(e)(1)(i) DEGRADED CORE - RELIABILITY ANALYSIS PROGRAM

NRC POSITION:

(1) To satisfy the following requirement, the applicant shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates and a program to ensure that the results of such studies are factored into the final design of the facility. (NUREG-0718, Category 3)

(i) Perform a plant/site-specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. (NUREG-0718, II.B.8(1))

PSO RESPONSE:

Introduction

Reviews of the TMI accident have made the NRC and industry increasingly aware of the need for an improved systems-oriented approach to safety review. In order to provide this review, methods are applied to identify sequences of events that have a high likelihood of occurrence. The methodology, similar to that used in WASH-1400, provides insight into the relative safety significance of reactor systems and design features and allows assessment of the merits of prospective changes to these systems. 19

Black Fox Station (BFS) is a BWR/6 Mark III design which has benefitted from both experience and the application of proven engineering principles to provide a safe and reliable operating station.

To further verify the BFS design, PSO will perform a plant/site specific Reliability Analysis Program with objectives to seek improvements in the reliability of core and containment heat removal systems as are significant and practical and to not impact excessively on the station.

PSO Involvement

A consultant or engineering organization who is highly qualified and experienced in risk assessment methodology will be selected by PSO to conduct the BFS Reliability Analysis Program.

PSO will provide management of the interfaces between PSO, Black & Veatch, the NSSS vendor, and the consultant organization to assure timely and effective development of the program.

PSO is currently developing in-house capability to implement and maintain its program through participation of an engineer in the Oconee

Probabilistic Risk Assessment being sponsored by the Nuclear Safety Analysis Center. This capability will be further developed as PSO personnel participate directly with the consultants in conducting the BFS Reliability Analysis Program.

BFS Reliability Analysis Program Plan

The methodology to be used will be similar to that employed in WASH-1400 updated to be consistent with the IEEE/ANS, and the Interim 19
Reliability Evaluation Program efforts to establish a standard methodology.

The initiating events to be considered will include those indicated in 19
Table (1)(i)-1 together with the accidents and transients identified in the PSAR Chapter 15 and those applicable accidents in WASH-1400. These events will be screened to identify the basic set of initiating events requiring operation of the key safety systems for core protection and release mitigation. The Reliability Analysis Program will focus on core and containment cooling systems in performing event tree/fault tree analysis, and will include environmental effects, system interactions, human error and performance data, interdependence of 19
support systems and system unavailabilities in the event tree/fault tree analysis.

The Reliability Analysis Program will identify common-mode failure mechanisms and sequences and system/component failures which are the dominant contributors to core damage. A component failure data base for use in system fault tree analysis will be developed from recognized reference sources including WASH-1400 and IEEE-500. In addition, prototype specific failure data will be requested from vendors of selected components being supplied to PSO. The data base used in the system fault trees will include methodologies to adjust failure data for varying testing and surveillance strategies. Human error will be considered in the development of the data base.

An uncertainty analysis will be performed to determine propagation of component failure data, including error ranges, through the fault trees.

Sensitivity analyses will be performed by varying the failure rates of key basic events which contribute to dominant event sequences in order to determine the effect on system failure rates and over-all results.

An additional decay heat removal system with its functional requirements and criteria derived from the study will also be considered.

The final report will appear in the format shown in Table (1)(i)-2. 19

PSO RESPONSE: 10 CFR 50.34(e)(1)(i)

Review and Recommendation

Prior to decisions relating to identified design or other improvements, PSO will appoint a third party to conduct a peer review of the reliability study. 19

PSO will evaluate the results of the study, including the peer review, and determine the need for implementing any improvements.

Application of Results to Final Design

Acceptance criteria for the reliability analyses will be established during the initial phase of the program. These acceptance criteria will include both quantitative and qualitative considerations of potential design changes on plant safety, cost, schedule, and availability.

The results of the reliability analyses will be evaluated using the acceptance criteria to determine design or other changes. The results of the study will be used to improve reliability of component selection, specifications, and testing and to improve system interaction. Furthermore, the results of the study will be used to identify improvements to be considered for maintenance, procedures, operator training, operating feedback and to identify those areas where additional quality assurance would improve reliability of core and containment cooling systems.

Schedule

The program will commence after issuance of construction permit. The initial phase of the program is expected to take approximately 15 months and will consist of a reliability analysis of the present BFS design. The final study, including radionuclide release quantification, will be completed within two years of CP issuance.

BFS Engineering Design Status

Over one-half of BFS engineering design has been completed and most of the nuclear steam supply system/emergency core cooling system components have already been fabricated and delivered into storage.

However, the results of the study will be used on a case by case basis to determine whether major redesign, repurchase, or refabrication is warranted, taking into account the significance, practicality, and impact on the Station. 19

Acceptance Criteria

There are currently no established regulatory requirements or acceptance criteria for judging the acceptability of the reliability analysis. Thus, the need for implementing changes in design operating, testing, or maintenance procedures to achieve improvements will be based on judgemental acceptance criteria which are not directly related to licensing requirements.

Radioactive Release

The total radioactive release to the environment will be estimated for the various release categories and will serve as a basis for assessing the effect of improvements to the reliability of the core and containment cooling systems.

TABLE (1)(i)-1
INITIATING EVENTS FOR RELIABILITY ANALYSIS PROGRAM

1. LOCA
2. Transients
3. Steam/Feedwater line breaks
4. Failures during cold shutdown operation
5. Fire
6. Earthquakes*
7. Explosions and missiles, internal and external*
8. Floods*
9. Tornadoes, hurricanes*
10. Station blackout, loss of AC/DC

19

* The best available methodology will be utilized where applicable and will be consistent with IEEE/ANS efforts where appropriate.

TABLE (1)(i)-2 19

OUTLINE OF RELIABILITY ANALYSIS REPORT

I. INTRODUCTION

II. SUMMARY

III. METHODOLOGY OVERVIEW

A. Event Trees
B. Fault Trees
C. Quantification of Accident Sequences
D. Containment Failure Analyses
E. Fission Product Release Analyses
F. Treatment of Uncertainties

IV. SYSTEM DESCRIPTIONS

A. Performance Requirements
B. Actuation
C. Environment Considerations
D. Dependence Diagrams for Support Systems - Power, Cooling, Lubrication 19

V. CORE MELT PROABILITIES

A. Dominant Sequences
B. Dominant Cut-Sets

VI. PLANT MODIFICATIONS THAT ADDRESS DOMINANT SEQUENCES

A. Improvement in Reliability Expected
B. How Factored into Design, Equipment Purchase, Fabrication, Procedures, Operation, etc.
C. Basis for Not Implementing More Reliable Alternatives 19

VII. FISSION PRODUCT RELEASE ANALYSIS

A. Release Groups
B. Containment Failure Probabilities
C. Fission Product Release Fractions
D. Total Radioactive Release from Containment to Environment for the Various Release Groups.

VIII. APPENDICES (DETAILS OF STUDY)

10 CFR 50.34(e)(1)(ii) AUXILIARY FEEDWATER SYSTEM EVALUATION

NRC POSITION:

(1) To satisfy the following requirement, the application shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of such studies are factored into the final design of the facility. All studies shall be completed no later than two years following issuance of the construction permit or manufacturing license. (NUREG-0718, Category 3)

(ii) Perform an evaluation of the proposed auxiliary feedwater system (AFWS), to include (applicable to PWR's only): (NUREG-0718, II.E.1.1)

(A) A simplified AFWS reliability analysis using event-tree and fault-tree logic techniques.

(B) A design review of AFWS.

(C) An evaluation of AFWS flow design bases and criteria.

PSO RESPONSE:

This requirement is applicable to Construction Permit applications for Pressurized Water Reactors only and hence does not apply to Black Fox Station.

10 CFR 50.34(e)(1)(iii) IMPACT OF REACTOR COOLANT PUMP SEAL DAMAGE FOLLOWING SMALL-BREAK LOCA WITH LOSS OF OFFSITE POWER

NRC POSITION:

(1) To satisfy the following requirements, the applicant shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of such studies are factored into the final design. (NUREG-0718, Category 3)

(iii) Perform an evaluation of the potential for an impact of reactor coolant pump seal damage following a small-break LOCA with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break Loss of Coolant Accident with subsequent reactor coolant pump seal damage. (NUREG-0718, II.K.2.16 and II.K.3.25)

PSO RESPONSE:

Introduction

The accident on March 28, 1979 at Three Mile Island, Unit 2 (TMI-2) involved a main feedwater transient coupled with a stuck-open relief valve and a temporary failure of the auxiliary feedwater system. The Bulletins and Orders Task Force (B&OTF) was established following the TMI-2 accident and made responsible for reviewing loss of feedwater transients and Loss of Coolant Accidents (LOCA) for all operating plants. A generic review of the General Electric designed Boiling Water Reactor (BWR) operating plants was conducted by the B&OTF. The B&OTF evaluated an accident scenario similar to the TMI-2 event; i.e., a loss of feedwater transient with a small break LOCA and a loss of high pressure emergency core cooling systems. As part of their review, the NRC Staff evaluated leakage paths for reactor coolant following a small break LOCA. The requirement to evaluate the performance of the reactor recirculation pump seals under accident conditions was one result of the Staff's evaluation.

Study

The BWR Owners' Group performed an evaluation of the effect of loss of alternating current power following a LOCA on pump seals. The study was completed and it has been submitted to the NRC Staff for evaluation. PSO will follow the course of resolution of this generic issue, and will provide a plant specific evaluation of this issue within two years after issuance of the construction permit for BFS and after Staff review adopt the final resolution of the issue in the final design for BFS.

Equipment Description

The BWR recirculation pump design incorporates a dual mechanical shaft seal assembly to control leakage around the rotating shaft of the recirculation pump. Each assembly consists of two seals built into a cartridge that can be replaced without removing the motor from the pump. Each individual seal in the cartridge is designed for full pump design pressure and can adequately limit leakage in the event that the other seal should fail.

During normal operation, the recirculation pump seals require forced cooling due to the temperature of the primary reactor water and due to the friction heat generated in the sealing surfaces. For Black Fox Station, two systems accomplish this forced cooling: the Closed Cooling Water (CCW) system and the seal purge system. Cooling water, provided by the CCW system, flows through a heat exchanger around the seal assembly. This CCW flow cools primary reactor water which flows to the lower seal cavity, thereby maintaining the seals at the correct operating temperature.

The seal purge system injects clean, cool water from the control rod drive system into the lower seal cavity. This seal purge flow also provides efficient cooling for the seals.

Nature of Study

Under normal conditions, with the primary reactor system at or near rated temperature and pressure and the recirculation pumps either operating or secured, both CCW and seal purge are operating. These two systems maintain the seal temperatures at approximately 120° F. Test data indicate that if either one of the seal cooling systems is operating, the seal temperatures remain well below 250° F and no seal deterioration should occur.

Under the abnormal condition of a small break LOCA followed by a loss of offsite power, both cooling systems to the pump seals will be lost. Test data, taken while operating at approximately 530° F/1040 psia, indicate that the seals will heat up, reaching 250° F approximately seven minutes after a total loss of cooling. This will occur whether or not the pump is operating. Test data also indicate seal temperatures exceeding 250° F may deteriorate the seal condition, resulting in primary coolant leakage into the drywell.

An analysis of fluid loss through a degraded seal modeled the fluid leakage path as a series of fluid volumes with interconnecting junctions, each having appropriate initial conditions. The model assumed gross degradation of the mechanical seals. Gross failure of these seals encompasses warpage, fractures and grooving of the seal faces due to excessive thermal gradients and dirt.

PSO RESPONSE: 10 CFR 50.34(e)(1)(iii)

The results of this seal leakage analysis show that even with gross degradation of the seals, the leakage would be less than 70 gallons per minute. This leakage rate is within the compensating capacity of normal or emergency reactor vessel water level control systems. A leakage of 70 gpm is equivalent to a liquid leak of 0.001 ft^2 flow cross-sectional area.

In the unlikely event of a seal failure on both recirculation pumps, the equivalent leak area of 0.002 ft^2 is insignificant compared to the postulated break area, 0.1 ft^2, of a small-break LOCA and does not influence the results of the LOCA analyses. It is emphasized that the seal leakage analysis is extremely conservative and a leakage rate of 70 gpm is not expected upon seal failure.

Conclusion of Study

Two systems provide cooling to the recirculation pump seals. If either one of these systems is operating, recirculation pump operation may continue with no harm to the seals. If both seal cooling systems are inoperable, the pump seals will overheat approximately seven minutes after the total loss of cooling, and seal deterioration may begin. The extent of seal deterioration is dependent on the seal's operating history, the amount of time without cooling, and the peak seal temperature.

Based on fluid loss analysis of extremely degraded seals, the leakage is less than 70 gallons per minute per pump. This amount of leakage is within the capacity of normal or emergency vessel water level control systems and does not influence the results of LOCA analyses.

10 CFR 50.34(e)(1)(iv) REPORT ON OVERALL SAFETY EFFECT OF PORV ISOLATION SYSTEM

NRC POSITION:

(1) To satisfy the following requirement, the application shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of such studies are factored into the final design of the facility. All studies shall be completed no later than two years following issuance of the construction permit or manufacturing license. (NUREG-0718, Category 3)

(iv) Perform an analysis of the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV). If this probability is a significant contributor to the probability of small-break LOCA's from all causes, provide a description and evaluation of the effect on small break LOCA probability of an automatic PORV isolation system that would operate when the reactor coolant system pressure falls after the PORV has opened. (Applicable to PWR's only). (NUREG-0718, II.K.3.2)

PSO RESPONSE:

This requirement is applicable to Construction Permit applications for Pressurized Water Reactors only and hence does not apply to Black Fox Station.

10 CFR 50.34(e)(1)(v) SEPARATION OF HPCI AND RCIC SYSTEM INITIATION LEVELS - ANALYSIS AND IMPLEMENTATION

NRC POSITION:

(1) To satisfy the following requirement, the applicant shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of such studies are factored into the final design of the facility. (NUREG-0718, Category 3)

(v) Perform an evaluation of the safety effectiveness of providing for separation of High Pressure Core Spray (HPCS) and Reactor Core Isolation Cooling (RCIC) system initiation levels so that the RCIC system initiates at a higher water level than the HPCS system, and of providing that both systems restart on low water level. (NUREG-0718, II.K.3.13)

PSO RESPONSE:

Introduction

The NRC Staff has expressed an interest in reducing the number of thermal cycles on the reactor vessel and its internals from the injection of cold water during transients caused by the loss of feedwater. This interest arose out of the Staff's evaluation of the TMI-2 accident in NUREG-0626, "Generic Evaluation of Feedwater Transients and Small Break Loss of Coolant Accidents in GE Designed Operating Plants and Near Term Operating License Applications." Specifically, Item A.1 of NUREG-0626 states that operating licenses and applicants should perform analyses to evaluate changes to the Reactor Core Isolation Cooling System (RCIC) and the High Pressure Coolant Injection System (HPCI) that would reduce the number of challenges to HPCI initiation and result in less stress on the vessel. The first change to be evaluated involves separating the RCIC and HPCI setpoints such that the RCIC system would initiate at a higher water level than HPCI. The second change to be evaluated involves modification of RCIC initiation logic such that the system will restart automatically on recurrence of low water level. Currently, RCIC and HPCS initiate at the same reactor low water level and RCIC logic reset for automatic restart is a manual operation. The NRC Staff has required that these same studies be performed by near term construction permit applicants.

A discussion of automatic restart for the HPCS system is given in the response to Requirement (1)(viii).

Studies

Two separate studies were sponsored by the BWR Owners' Group (BWROG), of which PSO is a member, to evaluate the two NRC recommendations. The first report, transmitted to the NRC on October 1, 1980 under the subject title "NUREG-0660 Requirement II.K.3.13" concludes that neither raising the RCIC setpoint or lowering the HPCI or High Pressure Core

Spray (HPCS) system (for designs using this system instead of HPCI) setpoint would result in substantially reducing thermal fatigue levels on the vessel and internals. This action would, instead, have undesirable consequences such as increasing the number of cold water injections from unnecessary RCIC initiation (raising RCIC initiation level) or decreasing the margin for adequate core cooling (lowering HPCI/HPCS initiation level). The report evaluates the thermal cycles due to RCIC and HPCI/HPCS actuation. The most severe thermal cycle due to RCIC and HPCI/HPCS initiation at the current level 2 water level setpoint is assessed and compared to the thermal cycle analysis for the limiting reactor components. Operating plant data is used to determine the frequency of HPCI/HPCS and RCIC initiation. Finally, the potential for reducing the number of thermal cycles by separating the two initiation setpoints is evaluated.

The second study was submitted to the NRC on December 29, 1980 by the BWROG under the subject title "BWR Owners' Group Evaluation of NUREG-0737 Requirements." This study concludes that changing the RCIC initiation logic for automatic restart would be beneficial to overall plant safety through increasing RCIC system availability.

System Descriptions

The High Pressure Core Spray System (HPCS) and Reactor Core Isolation Cooling Systems (RCIC) are high pressure reactor auxiliary cooling systems which deliver water to the vessel to restore and maintain coolant inventory during abnormal and small break Loss of Coolant Accident conditions. Both pumps take suction from either the condensate storage tanks or the suppression pool and discharge into the vessel at pressures high enough to preclude the need for vessel depressurization. The HPCS system is qualified as part of the Emergency Core Cooling System (ECCS) and is initiated on high drywell pressure and/or low reactor water level (Level 2) signals from the Nuclear System Protection System. The RCIC system functions in addition to the ECCS and is initiated on low reactor water level signal.

BWROG Evaluation of Separating HPCI/HPCS and RCIC Initiation Setpoints

1. Separation of Setpoints

   The BWROG study is generic in nature and applies to BFS. The analysis conducted is for typical BWR-3 and BWR-4 designs which have HPCI systems. The HPCI system consists of a steam turbine driven pump which injected water through the feedwater piping into the reactor vessel. BFS has a HPCS system which consists of an electric motor driven pump which discharges water through its own sparger onto the core. The BFS HPCS design therefore creates a less limiting condition in comparison to the HPCI design with respect to thermal fatigue.

PSO RESPONSE: 10 CFR 50.34(e)(1)(v)

The portions of the reactor vessel and its internals which may be affected by operation of HPCI and RCIC are the reactor vessel shell, core shroud and feedwater nozzles and spargers. Thermal fatigue analyses show that the limiting reactor component is the feedwater nozzle. Upon loss of feedwater, the temperature of the feedwater sparger and nozzle approaches the normal reactor operating temperature. Initiation of HPCI and RCIC at low water level cools the sparger and nozzle. The most severe thermal cycle results in a temperature change from 550°F (reactor operating temperature) to 50°F (HPCI/RCIC injection water temperature). This temperature change is included in the loads assumed in fatigue analyses based on normal operation, which itself includes many cold water injections, as well as expected transients and other postulated events. The duty imposed on the feedwater nozzle from all causes is summed to obtain a fatigue usage of 0.95, which is less than the limit of 1.0. The design basis includes 70 thermal cycles of the type described. The calculated fatigue usage of these cycles is about 17% of the total fatigue usage. It should be noted that there is no significant thermal effect on the reactor vessel shell due to the operation of HPCI and RCIC.

The feedwater nozzle fatigue usage will be even less than 0.95 for BFS because HPCS does not inject through the feedwater nozzle. Therefore, the study results are conservative for BFS.

2. Evaluation of the Potential for Reducing Thermal Cycles by Separation of HPCS and RCIC Setpoints

   There are two classes of transients which can cause RCIC and HPCS initiation:

   a. After feedwater is tripped on high reactor water level, the inventory is lost slowly due to decay heat steam generation.

   b. Following a sudden loss of feedwater, inventory loss is rapid, occurring approximately twenty (20) seconds after event initiation.

   The majority of transients which require HPCS and RCIC initiation can be grouped into category 1. In this case, the level decrease is slow because of the low power condition at the time the feedwater is tripped. A small amount of makeup water is needed and if feedwater cannot be restored, sufficient time is usually available such that RCIC would be started manually as the water level slowly decreases below the normal operating range. Since such manual action has been demonstrated to be successful for avoidance of HPCS actuation, it is considered sufficient and more desirable than an increase of the RCIC setpoint close to the normal operating water level. If neither feedwater or RCIC is manually started, both HPCS and RCIC would automatically be initiated at the low level setpoint.

PSO RESPONSE: 10 CFR 50.34(e)(1)(v)

The second class of transient to be considered is the loss of feedwater event. Sudden loss of feedwater flow is accompanied by a large and rapid drop in water level. Low level scram is initiated in approximately five (5) seconds, with RCIC and HPCS actuation occurring shortly thereafter. With both systems operating, water level is quickly restored. Due to the rapidity of the transient, HPCS initiation cannot be avoided even if the RCIC setpoint is raised to the normal operating level. Therefore, raising the RCIC setpoint for this type of transient can have no beneficial effect on thermal cycles and will interfere with normal plant operation.

For both types of events, automatic RCIC operation could avoid HPCS initiation if the HPCS setpoint was lowered; however, no significant benefit is realized unless the HPCS setpoint is lowered to near the low-low water level (level 1). Since the actuation of RCIC and HPCS has been previously shown to be of minimal impact in fatigue usage analyses, and lowering of the HPCS setpoint lessens the existing margin for assurance of adequate core cooling, such a separation of HPCS and RCIC setpoints by lowering the HPCS setpoint is not warranted.

BWROG Evaluation of Automatic RCIC Reset

The BWR Owners' Group has prepared a second report evaluating automatic RCIC reset. Currently RCIC reset is a manual operation. Depending on the accident or transient, the operator may have to perform other actions or may be distracted to the extent that he may either forget or delay RCIC system reset. To provide assurance that this does not occur, the RCIC system could be modified to incorporate automatic reset logic on high reactor water level. RCIC would restart on low water level and the operator would only have to verify proper system operation. The report states, and PSO agrees, that this design modification would benefit overall plant safety by increasing the availability of the RCIC system.

The proposed change would utilize the steam supply valve rather than the turbine trip valve to shut off steam to the RCIC turbine on high reactor water level. The steam supply valve would be used both to initiate system operation on low reactor water level and terminate operation on high water level.

The cessation of steam will be extended over a longer period of time due to the normal travel time of the steam supply valve. The spring loaded turbine trip valve closes essentially instantaneously. The steam supply valve closes in fifteen (15) seconds or less. Conservatively assuming full rated flow throughout this extended shutoff period with a maximum rated RCIC flow of 700 GPM would cause approximately 175 gallons to be added to the reactor vessel following the high vessel water level trip. This additional volume has an insignificant effect on high vessel level transients.

PSO RESPONSE: 10 CFR 50.34(e)(1)(v)

The duty on the steam supply valve is essentially the same; automatic closure rather than manual closure. The steam supply valve will be subjected to increased wear due to the wire drawing experienced at closure. This effect should be minimal due to the low frequency of closures with steam flow through the valve.

This modification to logic circuitry will increase the complexity of the system a minimal amount. From this standpoint the overall reliability of the system is minimally reduced, but this reduction is more than offset by the increased safety, reliability, and availability created by the fact that the steam supply valve is used to reset the system automatically.

Conclusion

It has been shown that HPCS and RCIC initiation at the current low-water level setpoints are within the design basis thermal fatigue limits for the reactor vessel and internals. Separating the setpoints as a means of reducing the number of thermal cycles caused by HPCS initiation would be of negligible benefit. Current design of the vessel is conservative with respect to thermal cycles and the stresses they cause. Further, it has been shown that the proposed change would be counterproductive; that the disadvantages of unnecessary RCIC actuation or decreased margin for adequate core cooling outweigh the advantage of increased system automation. We therefore conclude that no change should be made in the design for initiation of RCIC and HPCS.

The results of the analyses on the automatic RCIC restart indicate that the proposed logic change would contribute to improved system reliability, be of assistance to the operators, and enhance total plant safety. The change will be incorporated into BFS design after NRC approval of the BWR Owners' Group study.

10 CFR 50.34(e)(1)(vi) REDUCTION OF CHALLENGES AND FAILURES OF RELIEF VALVES - FEASIBILITY STUDY AND SYSTEM MODIFICATION

NRC POSITION:

(1) To satisfy the following requirement, the application shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of such studies are factored into the final design of the facility. All studies shall be completed no later than two years following issuance of the construction permit or manufacturing license. (NUREG-0718, Category 3)

(vi) Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. (NUREG-0718, II.K.3.16)

PSO RESPONSE:

Introduction

The accident at Three Mile Island-Unit 2 (TMI-2) on March 28. 1979 involved a main feedwater transient coupled with a stuck-open pressurizer power-operated relief valve and a temporary failure of the auxiliary feedwater system. The Bulletins and Orders Task Force (B&OTF) was established within the NRC and made responsible for reviewing and directing TMI-2 related staff activities associated with loss of feedwater transients and Loss of Coolant Accidents (LOCA) for all operating plants. A generic review of General Electric designed Boiling Water Reactor (BWR) plants was conducted by the B&OTF, and documented in NUREG-0626, "Generic Evaluation of Feedwater Transients and Small Break Loss of Coolant Accidents in GE Designed Operating Plants and Near Term Operating License Applications." This review identified improvements in systems, procedures, and analysis which, the B&OTF believed, would make GE-designed BWR's less susceptible to core damage during accidents and transients coupled with systems failures or operator errors.

NUREG-0626 showed that relief valves have failed to close and remained stuck open (after opening) at a rate of 0.03 failures per valve challenge in operating BWR plants. A stuck-open relief valve (SORV), which can be equivalent to a small break Loss of Coolant Accident (LOCA), can lead to actuation of the Emergency Core Cooling System (ECCS).

One requirement, proposed by the B&OTF and adopted by the NRC Staff was to study ways to reduce the number of challenges to safety and relief valves, i.e., the number of actuations. The Staff expanded the requirement to also study ways to reduce valve failures. The B&OTF recommended a frequency reduction for SORV's of one order of magnitude (factor of 10).

PSO RESPONSE: 10 CFR 50.34(e)(1)(vi)

Study

The BWR Owners' Group (BWROG), an organization of utilities operating or building boiling water reactors, has completed a study of potential modifications to reduce challenges to SRV's as well as to reduce the frequency of stuck-open relief valves. This study, titled "BWR Owners' Group Evaluation of NUREG-0737, Item II.K.3.16, Reduction of Challenges and Failures of Relief Valves," has been submitted to the NRC. Public Service Company of Oklahoma, as a member of the BWR Owners' Group, participated in the study.

The majority of relief valve failures in operating plants have been with 3-stage Target/Rock valves in BWR-4 reactors. The study, therefore, examined ways to reduce the SORV frequency rate of these valves by the recommended amount.

The conclusions of the study indicated that the Black Fox design incorporates features that reduce the frequency of challenges to SRV's. The frequency of SORV's is further reduced by employment of an improved valve design compared to those in use in currently operating BWR's.

Equipment Description

Each Black Fox Station unit has nineteen dual function safety/relief valves located inside the containment on the four main steam lines which transport steam from the reactor vessel to the turbine. The primary purpose of the valves is to prevent damage to the reactor system resulting from excessive water/steam pressure.

The valves can be opened in two ways. For the "Safety" function, each valve is provided with a spring which keeps the valve closed. The spring is adjusted so that a particular pressure in the steam line must be reached before the valve begins to open. The valve closes when the pressure is reduced below the valve spring "setpoint."

For the "Relief" function, each valve is equipped with an air operator, the actuation of which opens the valve. Actuation is caused by an electrical signal which has three sources: pressure sensors on the reactor vessel which cause the valves to open at pressures slightly less than that of the "safety" function; a switch in the control room with which the operator can open individual valves; and instruments in the Automatic Depressurization System which cause eight valves to open if high pressure cooling systems fail to operate when required. Valve closure can occur either by operator manual action or automatically when reactor pressure is reduced.

PSO RESPONSE: 10 CFR 50.34(e)(1)(vi)

Reduction of Challenges

The study examined ways to reduce SRV challenges. The following were examined to determine their feasibility for effectively reducing SRV challenges:

(1) Additional anticipatory scram on loss of feedwater,

(2) Revised relief valve actuation setpoints,

(3) Increased emergency core cooling (ECC) flow,

(4) Lower operating pressures,

(5) Earlier initiation of ECC systems,

(6) Heat removal through emergency condensers,

(7) Offset valve setpoints to open fewer valves per challenge,

(8) Installation of additional relief valves with a block or isolation valve feature to eliminate opening of the safety/relief valves (SRV's), consistent with the ASME code,

(9) Increasing the high steam line flow setpoint for main steam line isolation valve (MSIV),

(10) Lowering the pressure setpoint for MSIV closure,

(11) Reducing the testing frequency of the MSIV's,

(12) More stringent valve leakage criteria, and

(13) Early removal of leaking valves.

Based on its examination, the BWROG concluded that existing designs for BWR-6 plants like Black Fox Station include several of these features. As a result of a probabilistic assessment and an examination of historical data, the BWROG determined that a reduction of SRV challenges by more than a factor of 2 has been achieved by present BWR-6 designs. These features are:

(1) MSIV closure at reactor water level 1 rather than level 2. The Black Fox Station MSIV's will close at a reactor water level lower than the level at which the MSIV's of previous BWR designs close. Since this lower water level is reached less frequently, the MSIV's will not close as often, and relief valves will not be called upon to open due to the ensuing pressure increase.

(2) Improved relief valve control logic so that only one or two valves will cycle open and closed instead of all nineteen following the initial pressure signal which causes all of the valves to open. This control logic allows actuation of a few selected valves at pressures lower than normal relief operation. Thus only a few valves are required to operate rather than all nineteen.

Reduction of Failures

The study then examined methods of reducing relief valve failures when challenges occur. For BWR-6 plants, including Black Fox Station, the BWROG concludes that reduction has already been accomplished by using an improved valve design. Both the Dikkers safety/relief valve, which Black Fox Station will use, and the Crosby safety/relief valve, which several other BWR-6 plants will use are believed by the BWROG to be less prone to failure than the 3-stage Target/Rock valve. The BWR-6 valves are not pilot-actuated like the Target/Rock valves, and employ fewer moving parts which come in contact with steam. Based on valve qualification test data and limited operating experience, a factor of eight reduction in SRV failures is expected by the BWROG.

Conclusion

The study concluded that the combined effect of reducing challenges and failures of relief valves is to reduce SORV's by a factor of sixteen. PSO will incorporate in the Black Fox Station design the resolution of this item as agreed to by the NRC and the BWROG.

10 CFR 50.34(e)(1)(vii) MODIFICATION OF ADS LOGIC - FEASIBILITY STUDY AND MODIFICATION FOR INCREASED DIVERSITY FOR SOME EVENT SEQUENCES

NRC POSITION

(1) To satisfy the following requirements, the applicant shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of such studies are factored into the final design of the facility. (NUREG-0718, Category 3)

(vii) Perform a feasibility and risk assessment study to determine the optimum Automatic Depressurization System (ADS) design modifications that would eliminate the need for manual activation to ensure adequate cooling. (NUREG-0718, II.K.3.18)

PSO RESPONSE:

Introduction

Following the TMI-2 accident, the NRC created the Bulletins and Orders Task Force (B&OTF) which directed and reviewed NRC activities associated with loss of feedwater transients and small break LOCAs. In the B&OTF generic review of General Electric designed BWR's, documented in NUREG-0626, "Generic Evaluation of Feedwater Transients and Small Break Loss of Coolant Accidents in GE Designed Operating Plants and Near Term Operating License Applications," the B&OTF identified certain improvements to systems and procedures which they believed would make GE designed plants less susceptible to core damage during accidents and transients coupled with system failures and operator errors. Specifically in NUREG-0626, Item A.7, the NRC Staff stated that the ADS initiation logic should be modified to eliminate the need for manual actuation to assure adequate core cooling. A feasibility and risk assessment should be performed to determine the optimum approach. One possible approach suggested by the Staff was to activate ADS on low reactor vessel water level provided no HPCS flow exists and a low pressure emergency core cooling system is running.

In the continuing evolution of this Requirement, the "Clarification of TMI Action Plan Requirements" (NUREG-0737) delineated the required content of the licensee procedures. NUREG-0737 stipulated that Item A.7 was applicable to operators of reactors and applicants for an operating license.

The final Staff revision to this requirement and to which this response is addressed, is embodied in NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License." In NUREG-0718, Staff concerns regarding ADS design are incorporated. As a result, this Requirement is now applicable to near-term construction permit applicants.

PSO RESPONSE: 10 CFR 50.34(e)(1)(vii)

Study

PSO is a member of the BWR Owners' Group which has performed an evaluation of possible design modifications to the ADS that would eliminate the need for manual activation to insure adequate core cooling. The Owners' Group report was forwarded to the NRC under a cover letter dated March 31, 1981 from D. B. Waters to D. G. Eisenhut with the subject title "BWR Owners' Group Evaluations of NUREG-0737 Requirements II.K.3.16 and II.K.3.18."

ADS System Description

The Automatic Depressurization System, through selected safety/relief valves, functions as a backup to the operation of the high pressure coolant systems for protection against excessive fuel cladding heatup upon loss of coolant, over a range of steam or liquid line breaks inside the drywell. The ADS depressurizes the vessel, permitting the operation of the low pressure coolant systems. It is activated automatically upon coincident signals of low water level in the reactor vessel (level 1), high drywell pressure, and low pressure ECCS pumps running. A time delay of approximately two minutes after receipt of the signals allows time for the automatic blowdown to be bypassed if the water level is restored (or to be bypassed manually if the signals are erroneous). The ADS can be manually initiated as well.

Nature of the Study

The point of concern in this issue is with a Loss of Coolant Accident which does not pressurize the drywell. Under such circumstances, as well as for some non-line break events, such as the loss of feedwater, which are further degraded by the unavailability of high pressure injection systems, manual ADS actuation is necessary under current design. The Owners' Group study examined the advantages and disadvantages of the current design, coupled with new symptom oriented emergency procedures and/or four possible ADS modifications. These were: (1) elimination of the high drywell pressure trip, (2) addition of a timer which allows bypass of the drywell pressure trip after a certain period of time, (3) addition of a suppression pool high temperature trip to be used in parallel with the drywell pressure trip, and (4) addition of a high pressure system no-flow trip as suggested by the NRC Staff in NUREG-0626, Item A.7.

Each of the options was evaluated on the basis of whether it assures adequate cooling without operator action for isolations and for stuck open relief valves. For these analyses it was assumed that all high pressure injection systems failed and that the vessel must be depressurized in order for the low pressure systems to inject.

The Owners' Group concluded that the intent of this requirement could be satisfied in two ways. First, the ADS logic could be modified for automatic depressurization for these type of events. Of the four ADS logic modifications, the Owners' Group concluded that a modification which adds the bypass timer or the elimination of the drywell pressure trip would be the most beneficial. The second approach determined by the Owners Group to satisfy the intent of this requirement was to provide the operator with symptom oriented emergency procedures for manual action during any degraded condition.

Discussion of ADS Logic Modifications

a. Elimination of High Drywell Pressure Trip

By eliminating the high drywell pressure trip, the ADS would be activated by low water level and low pressure ECCS pumps operating. When the water level reaches Level 1, the two-minute timer would begin to run. The timer would reset if the low water level signal cleared before the timer runs out.

Under current design, with a pipe break inside the drywell, the high drywell pressure trip would occur before the low water level trip. Eliminating the high drywell pressure trip can be thought of as assuring that this condition will exist for all transients. The water level response for an isolation event is bounded by small break analyses where the majority of inventory loss is through the cycling of relief valves, not through the break. The water level response for a stuck open relief valve is essentially the same as a small recirculation line break. Therefore, the break spectrum analyses provided in Chapter 15 of the PSAR verify that adequate core cooling would be assured for this alternative.

The elimination of the high drywell pressure trip is a simple and effective modification. Maintenance and testing is somewhat easier with fewer trip circuits to be tested and repaired. The primary drawback is that the removal of one of the trip circuits results in a slight increase in the probability of actuation as a result of improper testing or spurious signals. While this does not present a core cooling concern, it tends to decrease plant availability and increase the duty cycles on the vessel and containment due to unneeded depressurizations.

b. Bypass of the High Drywell Pressure Trip

The other viable alternative for ADS modification would involve the addition of a timed bypass of the drywell pressure trip. A timer would be actuated on low water level. When the timer runs out, the high drywell pressure signal would be bypassed and the ADS actuated on water level alone. Once the timer runs out, the option becomes the same as the elimination of drywell pressure trip option. The only difference is that for events which do not produce a high drywell pressure signal, the bypass timer gives the operator additional time to bypass automatic vessel blowdown if the situation is corrected or if ADS is not needed for some other reason.

This alternative presents a feasible modification. Moreover, additional maintenance and testing is minimal.

c. Other Modification Schemes

A system could be developed to measure the rise in suppression pool temperature due to the inventory loss through the relief valves or a break in the drywell. The ADS would then be initiated by either suppression pool temperature or drywell pressure and reactor low water level. Two conditions would be used to provide the pool temperature permissive: (1) when pool temperature reaches a specified value and (2) when pool heat up rate is faster than a specified rate.

This alternative presents many problems. The system would have to be designed to produce a high pool temperature signal before low reactor water level was reached. Additional hardware necessary for this option is complex and expensive. Variations in SRV location and RHR operation raise the possibility that the system might miss a local temperature rise from a relief valve. Maintenance and testing wo[illegible] complex and could increase exposure to maintenance personnel. [illegible] because of the complexity of the system, overall reliability is lower than for the previous options.

The final ADS logic modification considered involved the addition of high pressure system flow measurement and logic in parallel with the high drywell pressure trip. This alternative suffers from the fact that a HPCS pipe break or incorrect valving downstream would provide a false indication of flow without providing makeup to the reactor. Another difficulty is determining an acceptable flow criterion. For example, only 3% feedwater flow is required to maintain vessel inventory for isolation or stuck open relief valve events. It is difficult to accurately and reliably measure small flow with devices that would not interfere with normal operation. In addition, the HPCS is normally cycled on and off to maintain reactor water level. Therefore, the modification is not viable since the flow scheme has a low probability of producing a signal that reflects the availability of the HPCS.

Conclusion

PSO has evaluated the Owners' Group report and it has determined that the report is applicable to Black Fox Station. PSO will incorporate the Owners' Group/NRC Staff resolution of this matter into the design of Black Fox Station.

10 CFR 50.34(e)(1)(viii) RESTART OF CORE SPRAY AND LPCI SYSTEMS ON LOW LEVEL - DESIGN AND MODIFICATION

NRC POSITION:

(1) To satisfy the following requirement, the applicant shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of such studies are factored into the final design of the facility. (NUREG-0718, Category 3)

(viii) Perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present. (NUREG-0718, II.K.3.21)

PSO RESPONSE:

Introduction

Following the TMI-2 accident, the NRC created the Bulletins and Orders Task Force (B&OTF) which directed and reviewed NRC activities associated with loss of feedwater transients and small break LOCA's. In the B&OTF generic review of General Electric designed BWR's, documented in NUREG-0626, "Generic Evaluation of Feedwater Transients and Small Break Loss of Coolant Accidents in GE Designed Operating Plants and Near Term Operating License Applications," the B&OTF identified certain improvements to systems and procedures which they believed would make GE designed plants less susceptible to core damage during accidents and transients coupled with system failures and operator errors. Specifically, Item A.10 of NUREG-0626 states that the core spray and low pressure coolant injection system logic should be modified so that these systems would restart if required to assure adequate core cooling. The Staff further states that because this modification affects several core cooling modes under accident conditions, a preliminary design should be submitted for the Staff's review and approval prior to making any modification.

This requirement was a part of the "Clarification of TMI-2 Action Plan Requirements," (NUREG-0737) which was was applicable to operating reactor licensees and operating license applicants. The final Staff revision of this issue appears in NUREG-0718, "Licensing requirements for Pending Applications for Construction Permits and Manufacturing License." In NUREG-0718, the Staff's requirements regarding core spray and low pressure coolant injection automatic restart became applicable to BFS.

Study

The BWR Owners' Group, of which PSO is a member, has conducted generic reviews of the core spray and low pressure coolant injection systems for a group of plants including BFS and evaluated the impact of the proposed logic changes. The report was forwarded to the NRC Staff under a cover letter dated December 29, 1980, with the subject title, "BWR Owners' Group Evaluation of NUREG-0737 Requirements."

High Pressure Core Spray (HPCS) System Description

The HPCS supplies makeup water to the reactor vessel in order to restore coolant inventory following a LOCA. The system consists of an electric motor driven pump which takes suction from either the condensate storage tanks or the suppression pool and discharges into the vessel through its own sparger onto the core.

Operation of the HPCS is automatically initiated by signals from the Nuclear System Protection System indicating low reactor water level (level 2) or high drywell pressure (2 psig). After startup, the HPCS pump will run continually until secured by operator action. If, during HPCS operation, the water level reaches level 8 (222 inches above top of the active fuel) and the drywell pressure is below 2 psig the valve which allows injection into the vessel will close automatically. The pump will continue to run. A valve which directs flow to the suppression pool will open at this time to prevent the pump from overheating. Indication and annunciation of this bypass is given in the control room. If the water level again drops to level 2, the injection valve will reopen allowing water into the vessel. If the pump is secured by the operator, it will not restart automatically unless both automatic initiation signals, high drywell pressure and low water level, have cleared.

Low Pressure Core Spray (LPCS) System Description

If the HPCS cannot maintain vessel coolant inventory, the vessel will automatically depressurize through operation of the Automatic Depressurization System. Once the vessel is at a lower pressure the LPCS can supply the makeup coolant. For initial core cooling, the LPCS works in conjunction with the HPCS (if available) and the Low Pressure Coolant Injection system. The LPCS by itself can provide long term core cooling. The system consists of an electric motor driven pump which takes suction from the condensate storage tanks or the suppression pool and discharges through its own sparger onto the core.

Automatic operation of the LPCS is initiated by signals indicating water level 1 or high drywell pressure. Upon automatic initiation and when reactor pressure is below about 300 psi, the injection valve opens to allow water into the vessel. A cooling loop is established; water is taken from the suppression pool, through the LPCS pump into the reactor, then out of the reactor through the ADS valves into the suppression pool. The LPCS pump will continue to operate until manually secured by the operator.

PSO RESPONSE: 10 CFR 50.34(e)(1)(viii)

Low Pressure Coolant Injection (LPCI) System Description

The LPCI is one functional mode of the Residual Heat Removal System (RHR). The Containment Spray mode and the Suppression Pool Cooling mode are the other safety related accident modes of the RHR. The RHR System consists of three separate loops, each equipped with a motor driven pump and piping routed to accomplish the many functions the system must perform.

The LPCI mode, in conjunction with the other ECC systems, will restore and maintain vessel coolant inventory. The LPCI is automatically initiated on reactor water level 1 or high drywell pressure. All three loops, A, B and C, inject water into the vessel initially because the primary concern immediately following a LOCA is to restore water level in the reactor. LPCI loop A initiation logic is common to the LPCS. LCPI loops B and C share similar, but separate, initiation logic.

Ten minutes after LPCI initiation, and in the event of high drywell and containment pressure, LPCI loops A and B automatically align themselves to the Containment Spray mode. The Containment Spray mode of the RHR is necessary in order to cool and depressurize the containment so that it can accommodate the bypass of steam through the drywell vent structure. The spray also removes airborne halogen and particulate fission products from the containment atmosphere. Loop C will continue in the LPCI mode.

The Suppression Pool Cooling mode of the RHR cools the pool water by circulating it through the RHR heat exchangers. This mode is manually initiated. The Suppression Pool Cooling mode is important because it helps prevent containment overpressurization and protects the RHR pumps from damage due to inadequate net positive suction head conditions caused by elevated pool temperatures.

Analysis of Logic Changes

HPCS

The BWROG study reviewed the current HPCS design and concluded that additional safety margin may be added by modifying the HPCS control logic to provide automatic restart of the system following manual termination by the operators. Automatic restart would be provided by a system that (1) blocks the high drywell pressure signal to allow logic reset, (2) restarts the pump on reactor water level 2, (3) clears if both reactor low water level and high drywell pressure signals disappear, and (4) still allows system shutdown if necessary. The logic change is under NRC review for approval.

LPCS and LPCI

The BWR Owners' Group study addressed the desirability of adding logic for automatic restart of the LPCS and LPCI as opposed to manual restart by the operator.

Reactor operators presently can stop an ECCS System at any time even if a LOCA signal is present. This manual override option is deliberate and is an important safety feature because it provides the operators flexibility for dealing with credible conditions requiring system shutdown. Examples of such conditions are gross seal leakage, ECCS piping breaks, failed pump motors or load shedding for other post-LOCA operations. Any design change which restricts operator flexibility would not enhance plant safety. Since reactor water level is measured directly in the BWR and is a primary parameter in operator guidelines, operator action is a highly reliable means of reinitiating the low pressure ECCS if needed for core cooling.

Moreover, the Owners' Group study determined that additional complications arose when designing low pressure automatic restart as compared to the HPCS restart. These additional complications include competing priorities for modes of operation of the RHR system, decreased operator flexibility when dealing with unanticipated situations, and the additional complexity of logic required with its resultant decrease in reliability. Because of the complications, no net safety improvement would be realized if the logic changes were made on the low pressure systems. For example, high containment and drywell pressures cause a portion of the LPCI to realign to the Containment Spray mode automatically. Reoccurrence of the LPCI autostart signal would create conflicting simultaneous automatic signals which would have to be resolved by priority logic and its inherent complications. As a further example, automatic realignment of the other modes to the LPCI mode would have to take into account the characteristics of the hardware involved. Thus, the suppression pool return valve requires 90 seconds to close, while the LPCI injection valve requires only 25 seconds to open. This valve travel time mismatch would result in a significant period of time during which the RHR pumps would be supplying water to both flow paths. Since the pumps are not designed for excess duty, pump motor and auxiliary power overloading are problems. Logic to avoid the valve timing mismatch would require additional valve permissives and so adds to the probability of failure.

Conclusion

PSO has evaluated the Owners' Group report and it has determined that the report is applicable to Black Fox Station. PSO will incorporate the Owners' Group/NRC Staff resolution of this matter into the design of Black Fox Station.

10 CFR 50.34(e)(1)(ix) CONFIRM ADEQUACY OF SPACE COOLING FOR HPCI AND RCIC SYSTEMS

NRC POSITION:

(1) To satisfy the following requirement, the applicant shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of such studies are factored into the final design. (NUREG-0718, Category 3)

(ix) Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the Reactor Core Isolation Cooling (RCIC) and High-Pressure Coolant Injection (HPCI)* systems, following a complete loss of offsite power to the plant for at least two (2) hours. (NUREG-0718, II.K.3.24)

*For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, "high pressure core spray" for "high pressure coolant injection" and "HPCS" for "HPCI."

PSO RESPONSE:

Introduction

The question concerning adequate space cooling for the emergency core cooling systems following the loss of offsite power arose indirectly from the NRC Staff's evaluation of the TMI-2 accident in NUREG-0626, "Generic Evaluation of Feedwater Transients and Small-Break Loss of Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications." Specifically, NUREG-0626, Item B.3, requires licensees and operating license applicants to verify the acceptability of a two hour loss of offsite power to BFS and the effect on the Reactor Core Isolation Cooling System (RCIC) and the High Pressure Core Spray System (HPCS) and their support systems. The RCIC system provides makeup water to the reactor vessel to replenish coolant inventory in the event the reactor is isolated from the main condenser accompanied by the loss of normal feedwater supply. The HPCS system supplies makeup coolant to the vessel to restore coolant inventory following a LOCA, thereby preventing core damage caused by excessive temperatures. As a consequence of the Staff's concern about adequate space cooling, they have, in NUREG-0718 (Revision 1), "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License," required BWR Construction Permit applicants to perform a study to determine the need for additional space cooling in order to ensure long-term operation of the RCIC and HPCS systems following a two hour loss of offsite power. Space cooling shall be adequate to maintain pump room temperatures within tolerable limits.

PSO RESPONSE: 10 CFR 50.34(e)(1)(ix)

PSO Study

The Black Fox Station (BFS) RCIC and HPCS systems and their associated space cooling systems have been designed to function continuously for an extended period of time in excess of two hours following the loss of offsite power. The design criteria for the space cooling systems are adequate to maintain pump room temperatures within allowable limits. Only a momentary interruption (approximately 30 seconds) in space cooling will occur at the incidence of loss of offsite power, the time necessary for onsite emergency diesel generator startup and load sequencing. The loss of offsite power is a design basis for the BFS RCIC and HPCS and their associated space cooling systems. A re-evaluation of this design basis confirms that no additional space cooling is needed for BFS to ensure reliable long-term operation of the RCIC and HPCS systems.

Description of Space Cooling for HPCS and RCIC

Environmental conditions in the HPCS and RCIC equipment rooms are maintained within tolerable limits through the operation of the Auxiliary Building Heating, Ventilating and Air Conditioning System (HVAC-AB). The Auxiliary Building is divided into zones that provide control over the spread of extreme environmental conditions from local areas throughout the building. The HPCS and RCIC equipment rooms are cooled by individual safety related recirculating fan coil units located in the equipment rooms. Each fan coil unit contains a fan, filter, and cooling coil. These units are safety Class 3, electrical Class 1E and are manufactured and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section III and IEEE Standard 323-1974. The units are locally controlled by a thermostat which cycles the fan and are provided with an interlock to start whenever the equipment they protect is started. There are also manual start switches located in the main control room.

The RCIC room fan coil unit is provided power from Division 1 of the Standby AC Power Supply System (onsite diesel generators) and cooling water from Division 1 of the Standby Service Water System following loss of offsite power. The HPCS room fan coil unit is powered from Division 3 of the Standby AC Power Supply System and provided cooling water from Division 3 of the Standby Service Water System. In each case, the Space Cooling System is supplied emergency AC power and cooling water from the same division as the equipment being protected.

At the incidence of loss of offsite power, the Auxiliary Building secondary containment is isolated and the RCIC and HPCS fan coil units are deenergized. Thirteen seconds later the diesel generators are synchronized with the ESF buses and are ready to accept loads. Twelve seconds later the HPCS and RCIC fan coil units have been started and have reached rated capacity within thirty seconds of the loss of offsite power if the equipment they serve is operating. Thirty seconds following the loss of offsite power, the air handling units and exhaust fans for the ESF switchgear and battery rooms, Divisons 1, 2, 3 and 4 are automatically started. When the loss of offsite power condition clears, all units are manually returned to normal operation using offsite power.

Conclusion

Loss of offsite power will not affect the performance of the RCIC and HPCS systems or their space cooling systems. These systems have been designed to operate for a period of time in excess of two hours using AC power from the station's onsite emergency diesel generators. Space cooling has been designed to maintain suitable equipment room environments to ensure that the equipment will operate efficiently and reliably.

10 CFR 50.34(e)(1)(x) VERIFY QUALIFICATION OF ACCUMULATORS ON ADS VALVES

NRC POSITION:

(1) To satisfy the following requirement, the application shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of such studies are factored into the final design of the facility. All studies shall be completed no later than two years following issuance of the construction permit or manufacturing license. (NUREG-0718, Category 3)

(x) Perform a study to ensure that the Automatic Depressurization System, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves. (NUREG-0718, II.K.3.28)

PSO RESPONSE:

Introduction

The accident at Three Mile Island, Unit 2 (TMI-2) on March 28, 1979 involved a main feedwater transient coupled with a stuck-open pressurizer power-operated relief valve and a temporary failure of the auxiliary feedwater system. The Bulletin and Orders Task Force (B&OTF) was established by the NRC and made responsible for reviewing and directing TMI-2 related activities associated with loss of feedwater transients and Loss of Coolant Accidents (LOCA) for all operating plants. A generic review of General Electric designed Boiling Water Reactor (BWR) plants was conducted by the B&OTF and documented in NUREG-0626. The B&OTF review identified improvements in systems, procedures, and analysis which the B&OTF believed would make GE-designed BWR's less susceptible to core damage during accidents and transients coupled with systems failures or operator errors. One item identified concerned the design basis for ADS actuation.

The Automatic Depressurization System (ADS) must be able to function after an accident if the reactor vessel remains pressurized and high pressure cooling systems are either inoperative or fail to maintain adequate core cooling. The ADS is then used to depressurize the reactor so that low pressure cooling systems can inject water to maintain adequate core cooling.

One requirement, proposed by the B&OTF and adopted by the NRC Staff, was to verify that the accumulators for the Automatic Depressurization System (ADS) do in fact meet design requirements. The present Automatic Depressurization System (ADS) air accumulators are sized to cycle the ADS valves twice against 70% of containment design pressure (or five times against containment atmospheric pressure) plus accommodate component leakage for several days.

Study

PSO is a member of the Boiling Water Reactor Owners' Group (BWROG) which is performing a study to justify the design basis of the ADS accumulator design. The BWROG study will provide a discussion of the general design bases for both long and short term depressurization needs. PSO will perform and submit to the NRC a study verifying that the BFS ADS will be capable of performing its intended function during and following an accident situation.

System Description

The Automatic Depressurization System utilizes eight of the nineteen dual function safety/relief valves to achieve its function. Two ADS valves are located on each main steam line.

The ADS valves use the "relief" function of the SRV's, opening due to actuation of a pneumatic operator by an electrical signal. The design includes three sources of gas (air or nitrogen) supply for actuation purposes. During normal operation, air is supplied to the valves from the station air system. Two safety grade backup sources are available if the station air supply is unavailable. The short-term backup supply utilizes accumulators located near the ADS valves in the drywell. Also, a supply of nitrogen is provided from storage cylinders located in the Control Building. Additional backup cylinders are available in the Control Building as well as an external connection for additional nitrogen brought from offsite. This arrangement provides for replenishment on a periodic basis, thus ensuring an essentially continuous source of supply.

Requirement Response

PSO will perform and submit to the NRC a study verifying that the ADS accumulators meet the design requirements. The BWROG study, which is generic, will be used as a guide in PSO's plant unique study for BFS.

The study will address the capability of the Black Fox Station ADS to provide both short and long term reactor pressure vessel depressurization during and following an accident situation. This will include justifying the sizing of the accumulators. In addition, PSO will address the guidelines provided in the April 8, 1981 meeting with the NRC Staff, which are:

1. Define the number of times the ADS valves must be capable of cycling using only the accumulator inventory and the length of time these accumulators are required to perform their function following an accident.

PSO RESPONSE: 10 CFR 50.34(e)(1)(x)

2. Provide the criteria for the allowable leakage limits.

3. Commit to periodic leak testing of the ADS accumulator system.

4. Propose technical specifications which will specify leak test frequency, allowable leak rate and the actions to be taken if the leakage limit is exceeded.

5. Commit to seismically and environmentally qualify the ADS accumulator system and associated control circuitry.

PSO will make those design changes deemed necessary as a result of the resolution of this item with the NRC.

10 CFR 50.34(e)(1)(xi) EVALUATE DEPRESSURIZATION WITH OTHER THAN FULL ADS

NRC POSITION:

(1) To satisfy the following requirement, the application shall provide sufficient information to describe the nature of the studies, how they are to be conducted, estimated submittal dates, and a program to ensure that the results of such studies are factored into the final design. (NUREG-0718, Category 3)

(xi) Provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown. (NUREG-0718, II.K.3.45)

PSO RESPONSE:

Introduction

The accident on March 28, 1979 at Three Mile Island, Unit 2 (TMI-2) involved a main feedwater transient coupled with a stuck-open relief valve and a temporary failure of the auxiliary feedwater system. The Bulletin and Orders Task Force (B&OTF) was established following the TMI-2 accident and made responsible for reviewing loss of feedwater transients and Loss of Coolant Accidents (LOCA) for all operating plants. A generic review of the General Electric designed Boiling Water Reactor (BWR) operating plants was conducted by the B&OTF. The B&OTF evaluated an accident scenario similar to the TMI-2 event; i.e., a loss of feedwater transient with a small break LOCA and a loss of high pressure emergency core cooling systems. They determined the transient would be terminated by actuation of the Automatic Depressurization System (ADS) and low pressure emergency core cooling system injection. Since analyses of BWR plants have included only one or two ADS actuations in the calculation of reactor pressure vessel fatigue usage, the NRC Staff determined that an evaluation of depressurization methods, other than by full actuation of the ADS, which would reduce the possibility of exceeding vessel integrity limits during rapid cooldown should be provided.

Study

PSO is a member of the BWR Owners Group which has performed an evaluation of depressurization modes other than full ADS. The study is applicable to BFS and concludes that there is no benefit to be derived from the use of reduced blowdown rates. PSO will incorporate into the BFS design the resolution of this item that is agreed to by the Owners Group and the NRC.

Equipment Description

The ADS consists of dual-function safety/relief valves (SRV) each with an air supply, an accumulator, a discharge line to the suppression pool and associated initiation and leakage detection system. The air supply consists of a connection to the plant air system with the accumulator serving as a backup supply capable of maintaining its associated SRV open until the primary coolant system is depressurized. The SRV's are located on each of the main steam lines between the reactor vessel and the main steam line isolation valves. Each of the SRV's discharge through a separate line to a point below the water level of the suppression pool.

The ADS together with the low pressure emergency core cooling systems serve as a backup to the High Pressure Core Spray (HPCS) system. Without the availability of HPCS, ADS performs the function of vessel depressurization for small and intermediate break LOCA's, thereby permitting the low pressure emergency core cooling systems to operate and provide adequate core cooling. The ADS is actuated on high drywell pressure and reactor vessel low water level if a low pressure emergency core cooling system pump is operating. The ADS valves may also be opened by the control room operator.

Analysis

Depressurization by full ADS actuation constitutes a depressurization from about 1050 psig to 180 psig in approximately 3.3 minutes. Such an event, which is not expected to occur more than once in the lifetime of the plant, is well within the design basis of the reactor pressure vessel. This conclusion is based on the analysis of several transients requiring depressurization via the ADS valves. Results of these analyses indicate that the total vessel fatigue usage is less than 1.0. Therefore, no change in the depressurization rate is necessary. However, to comply with the above requirement two cases of reduced depressurization rates were analyzed and compared with the full ADS actuation. The alternate modes considered cause vessel pressure to traverse the same pressure range as a full ADS actuation. Case 1 depressurization times range from 6-10 minutes depending on plant size and ADS capacity. Case 2 depressurization times range from 15-20 minutes. The Case 1 depressurization gives the results of an intermediate time between the present design and an unacceptable long core uncovery time. The Case 2 depressurization produced an undesirably long core uncovery time thus bounding the possible increase in depressurization times. These modes were achieved by opening a reduced number of relief valves.

Core Cooling Capability

Examination of the reduced depressurization rates under consideration with respect to core cooling concerns shows that:

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1. Case 1 Depressurization

   A. When actuated at the same level as the full ADS case, vessel depressurization for a Case 1 blowdown will result in less vessel inventory at the time of ECCS injection and can result in longer periods of core uncovery.

   B. When actuated considerably earlier than at the ADS initiation setpoint, vessel depressurization for a Case 1 blowdown can result in some improvement in core cooling. However, the operator is required to act more quickly in these cases (i.e., within 1-6 minutes after the accident). This earlier depressurization also reduces the time available to start high pressure system injection and hence to avoid the need for manual depressurization. It also increases the frequency of depressurizations.

2. Case 2 Depressurization

   Vessel depressurization for a Case 2 blowdown causes the core to be uncovered for a lengthy period of time even assuming system initiation at the earliest reasonable time.

Vessel Integrity

Examination of the reduced depressurization rates under consideration with respect to vessel integrity shows that the fatigue usage for the vessel or core support structures is not significantly different for fast and slow blowdown events. Available pressure vessel fatigue analyses show the usage per event to be $<$ 0.1 per full ADS event.

In summary, reactor vessel and core support structure integrity is assured for the blowdown rates considered if an ADS event should occur, and reduced rates of depressurization do not significantly decrease fatigue usage.

Conclusion

The cases considered show that no appreciable improvement can be gained by a slower depressurization based on core cooling considerations. A significantly slower depressurization rate will result in increased core uncovered time. A moderate decrease in the depressurization rate necessitates an earlier actuation time resulting in less time available for operator action to start high pressure ECCS without significant benefit to vessel fatigue usage. This will also result in an increased frequency of ADS actuation.

Full ADS blowdown is well within the design basis of the reactor pressure vessel and ADS is properly designed to minimize the threat to core cooling. Because ADS is not expected to occur more than once per

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plant lifetime as a backup for HPCS, no change in the depressurization rate is necessary.

10 CFR 50.34(e)(2)(i) LONG-TERM TRAINING SIMULATOR UPGRADE

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(i) Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCA's. (Applicable to construction permit applicants only). (NUREG-0718, I.A.4.2)

PSO RESPONSE:

Introduction

Operator training programs should emphasize the basic principles of reactor operation under transient and accident conditions so that during such conditions appropriate corrective measures can be taken expeditiously to prevent onset of core damage. A training program designed to meet this objective requires extensive use of a full-scope simulator which closely models the plant and correctly demonstrates plant reactions to various size loss of coolant accidents (LOCA). PSO is committed to such a training program. The BFS training program for licensed operators, non-licensed operators, engineers, and technicians will be scheduled and implemented to support startup and operation of Black Fox Station. Table (2)(i)-1 provides a projected manpower schedule to support operator training and assignment. The training program will meet the requirements of the following documents: 19

10 CFR Part 55, Operator Licenses

Regulatory Guide 1.149, Nuclear Power Plant Simulators for Use in Operator Training

ANS 3.1, 4/10/81, Standard for Qualification and Training of Personnel for Nuclear Power Plants

Simulator Commitment

PSO has long been committed to ensuring that a simulator referenced to the Black Fox Station will be available to train operators, technicians and engineers for BFS. In 1976 PSO and General Electric Company undertook to develop a full-scope simulator to duplicate the Black Fox Station control room primary operator interface. The Black Fox Simulator has been in operation since October, 1979 at the General Electric BWR/6 Training Center, located near the BFS site.

PSO RESPONSE: 10 CFR 50.34(e)(2)(i)

PSO made the necessary commitment of resources to support the simulator development. At an early stage, balance of plant (BOP) engineering design work was authorized to ensure the most accurate information was included in the simulator model and panel layout. Black & Veatch and PSO engineers used a full scale mockup to integrate the BOP design in with the Nuclear Steam Supply System design. A PSO representative was assigned to General Electric for one year to participate in simulator verification, procedure preparation, training software development, senior reactor operator certification, and operator class instruction. This arrangement provided the training center staff with an interface for obtaining the BOP design information necessary to produce course material, while at the same time allowing PSO to closely monitor the progress of simulator development.

Simulator Model Capability

The Black Fox Simulator was designed to correctly model the BFS Plant and its control room. Modifications will be made to the simulator to increase training effectiveness as required, and update the simulator as required by ANSI/ANS-3.5, 1981 to reflect plant design changes.

The Black Fox Simulator has the capability to simulate various size loss of coolant accidents, including small-break LOCA's. Software flexibility allows for future expansion of transient simulation if deemed necessary.

During the presentation of a simulator training exercise, the evolution in progress may be suspended, terminated, or selected malfunctions activated according to a planned training schedule. Up to fifteen (15) separate malfunctions may be activated simultaneously to assist in skills development associated with recognizing, prioritizing and responding to multiple failures.

The Black Fox simulator design includes 121 generic plant malfunctions, ranging from small isolated equipment failures up to and including the design basis loss of coolant accident. Fifty-three of the 121 malfunctions are provided with multiple initiation inputs which represent common failure modes for redundant components. An annunciator malfunction is provided with the capability to activate any of the assigned annunciators independent of other malfunctions or operating modes. A sufficient number of malfunction spares have been provided for future expansion. All malfunctions required by ANSI/ANS-3.5, 1981 are simulated with two exceptions. One exception is the loss of service water or cooling to individual components. This malfunction will be added to the simulator. The other exception is steam generator tube leaks which are not applicable to the BWR-6.

A simulated initial condition (IC) established for each major training evolution is activated by selecting any one of twenty-six (26) pre-established IC modes. The pre-established Black Fox Simulator initial conditions are representative of plant operating conditions including beginning, middle, and end of core life. Beginning of core life is utilized as the basic IC. All other simulation modes were established from this condition based upon real time operation of the simulated Black Fox Plant.

Eight additional modes are provided for "snapshot" initialization enabling instantaneous establishment of an initial set of plant conditions recorded at the time of the "snapshot." Provisions also exist for two "setback" modes which will re-establish the simulator operating conditions from one to ten minutes prior to the initiation of the "setback" feature.

Simulator Model Verification

To ensure the Black Fox Simulator correctly resembled and modeled the Black Fox Station, several phases of acceptance testing took place. As part of the Black Fox Simulator acceptance procedure, a comparison of simulator response was made with analyzed transients reported in the BWR/6 Transient Analysis and with recent startup test results from a comparable sized BWR/5 plant. The operational transients considered in the report to the NRC were chosen to be representative of the transients reported in the Black Fox Station PSAR and sufficiently diversified to effectively illustrate the operational characteristics of the Black Fox Simulator. Initial test conditions were established on the simulator to closely approximate the initial condition of the reference transient analysis or startup test. The results of the simulator verification process proved that the simulator performance not only agreed with the BWR/6 Transient Analysis and the BWR/5 startup data but also presented the operator with quantitative values of plant parameters within the tolerances specified in ANSI/ANS-3.5, 1981.

In December, 1979 General Electric Company submitted a description of the Black Fox Simulator, its acceptance test results, and the "BWR/6 Integrated Operator Training Program" to the Operator Licensing Branch (OLB) of the NRC. Following review of this material, OLB representatives were sent to observe the simulator in operation. As a result, in April, 1980 OLB notified General Electric by letter approving the use of the Black Fox Simulator in USNRC license related training programs. This letter stated, in addition, that the OLB was "impressed with the quality and degree of simulation."

An on-going program exists at the BWR/6 Training Center to identify and correct deficiencies in both Black Fox Simulator hardware and software. In conjunction with the control room review (NUREG-0718, Rev. 1, Issue (2)(iii)) PSO personnel will further verify the model's accuracy as well as identify modifications that will be needed to duplicate the as-built plant design.

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Simulator Description

The Black Fox Simulator consists of a General Electric NUCLENET-1000* Control Complex and supportive control room panels plus a computer/software system modeling the expected dynamic response of the reference Black Fox Nuclear Power Station.

With the simulator designed to reflect an identical outward appearance and dynamic function, the operator trainees will realistically experience all modes of reactor operation, normal and degraded, as conducted from the control room of the reference plant. Relay racks and backrow benchboards are not included in the physical makeup of the simulator control room although related component and system functions are modeled by the computer software. An exception to this is the inclusion of selected panels housing controls and instrumentation for the Off-Gas System, Containment Atmospheric Monitoring, Standby Gas Treatment, and Nuclear Instrumentation Systems.

A SEL 32/55 model computer serves as the control and simulation subsystem for the Black Fox Simulator. The computer/software system calculates plant parameters corresponding to selected operating conditions, displays these parameters on appropriate instrumentation, and provides proper alarm and/or protective system action when predetermined limits are approached or exceeded. The Reactor, Turbine Generator, Auxiliary Systems and other equipment external to the control room are represented by mathematical models programmed to operate continuously, and in real time. Peripheral systems are simulated to the degree required to provide realistic control room instrument indication.

The simulator duplicates in real time system failures and malfunctions. Changes in system status which would be initiated by the actions of an in-plant equipment attendant are also simulated. The Simulator can process multiple malfunctions as passive failures awaiting actuation, failures which are currently active, and failures to be deactivated. Once initiated, the simulated malfunction(s) results in the same sequence of events that would occur in the reference plant.

Panel Description

The Black Fox Simulator consists of thirteen control panels arranged in a configuration as illustrated by Figure (2)(i)-1.

Duplicating the primary control room interface are the following front row panels:

NUCLENET* Control Console (P680)

Standby Information Panel (P678)

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Supervisory Monitoring Console (P679)

Reactor Core Cooling Benchboard (P601)

Diesel Generator Benchboard (P877)

Balance of Plant Control Benchboard (P870)

Auxiliary Electric Panel (P800)

To provide additional training capability on systems important to control room operations are the following back row panels:

Nuclear Instrumentation Panel (P880)

Containment Atmospheric Monitoring Panel (P639)

Off Gas System Panel (P845)

Division 2 Standby Gas Treatment System Panel (P847)

Division 1 Standby Gas Treatment System Panel (P848)

The Instructor's Console is unique to the training application and is not part of the Black Fox Station control room configuration.

Details concerning panel layout, functional placement, and design philosophy are provided in the PSO response to Requirement (2)(iii).

Summary

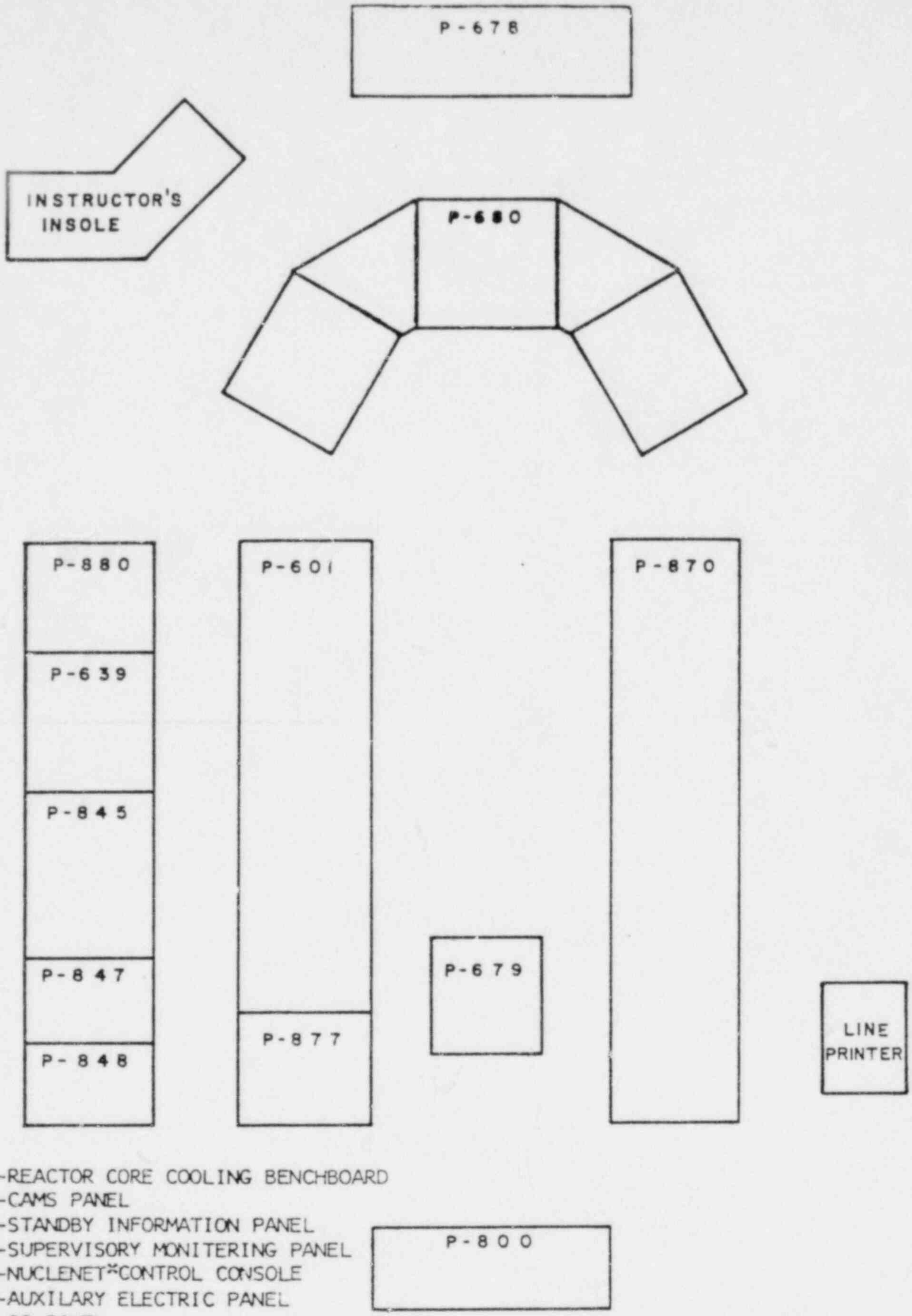
PSO has and will continue to demonstrate its commitment to a superior training program with extensive use of a high-fidelity, full-scope simulator. PSO will ensure that sufficient quality and quantity of simulator instruction will be made available to maintain safe and competent operations and support staffs for Black Fox Station.

TABLE (2)(i)-1
PROJECTED MANPOWER SCHEDULE TO
SUPPORT TRAINING AND ASSIGNMENT

Year	Cumulative Operations Staff	Cumulative SRO and RO Certifications	
1983 (CP)	5	0	
1984	7	0	
1985	28	0	
1986	40	0	
1987	57	8	
1988	115	16	
1989	200	24	19
1990	265	36	
1991 (FL/CO Unit 1)	300	44	
1992	335	52	
1993	382	60	
1994 (FL/CO Unit 2)	450	60	

CP: Receipt of Construction Permit
FL: Fuel Load
CO: Commercial Operation

Assume: CP 1983
Construction Start 1984
Unit 1 Operational 1991
Unit 2 Operational 1994



P601-REACTOR CORE COOLING BENCHBOARD
P639-CAMS PANEL
P678-STANDBY INFORMATION PANEL
P679-SUPERVISORY MONITERING PANEL
P680-NUCLENET*CONTROL CONSOLE
P800-AUXILARY ELECTRIC PANEL
P845-OG PANEL
P847-DIVISION 2 SGTS PANEL
P848-DIVISION 1 SGTS PANEL
P870-BOP CONTROL BENCHBOARD
P877-DIESEL GENERATOR BENCHBOARD
P880-NI PANEL

FIGURE (2) (i)-1
BLACK FOX SIMULATOR
CONTROL ROOM FLOOR PLAN

19-111381

10 CFR 50.34(e)(2)(ii) LONG-TERM PROGRAM PLAN FOR UPGRADING OF PROCEDURES

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(ii) Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts. (NUREG-0718, I.C.9)

PSO RESPONSE:

Introduction

It was concluded by various reviews and evaluations performed by the NRC and others of the TMI-2 accident and industry practice generally that more attention and care should be devoted to writing, reviewing, and monitoring plant procedures. As a result of this concern, the NRC Staff has required construction permit applicants to establish a continuous program to improve plant procedures.

Program Commitment

A detailed program governing the preparation, review, and revision of Black Fox Station procedures will be developed to support the commencement of procedure preparation three years prior to fuel load. The procedure development program will provide mechanisms to incorporate the results of industry and regulatory experiences pertaining to emergency procedures, reliability analysis, human factors engineering, crisis management, operator training, and plant operations into Black Fox Station procedure development. The procedure development program will be an on-going program, beginning during construction and continuing throughout the life of the plant.

Description of Program

The Station Manager will designate a senior BFS staff member to be responsible for development and execution of the program. This individual will be provided with the necessary resources and personnel to effectively accomplish these tasks. Personnel with experience in the appropriate technical discipline will develop procedures according to strict guidelines. Licensed or certified reactor operators will participate in the review and verification of operation-oriented procedures. All other procedures will be reviewed by the plant departments whose activities are governed by those procedures. Plant procedures will be reviewed, as applicable with regard to safety, fire protection, quality control, environmental control, and ALARA. The Plant Operations Review Committee, consisting of the Station Superintendent and the plant department supervisors, will review all procedure drafts and changes which may affect nuclear safety. The PSO Review and Audit Committee will review proposed changes to procedures which involve "unreviewed safety questions" as defined in 10 CFR Section 50.59.

Preliminary Activities

PSO has already established resources for development of high quality plant procedures for BFS. The General Electric procedures used for the Black Fox simulator are available to PSO. These simulator procedures serve as a basis from which control room procedures will evolve. A PSO representative was temporarily assigned to the General Electric BWR-6 Training Center to participate in the initial review and verification of the simulator procedures.

In conjunction with the control room review (Issue (2)(iii)), PSO personnel have begun an in-depth review and revision of all the Black Fox Simulator procedures. This project will not only enhance operator training on the simulator, but will also improve the starting base for preparing the station procedures. Human factors engineering, reliability analysis, operator training, and industry operating experience will be applied throughout the review. Guidelines published by the Electric Power Research Institute (EPRI), Nuclear Safety Analysis Center (NSAC), Institute for Nuclear Power Operations (INPO), and the Nuclear Regulatory Commisssion (NRC) will be referenced, as applicable, during the simulator procedure review. Generic emergency procedure guidelines will be used to draft Black Fox specific symptom-oriented procedures.

Another example of PSO's early interest in procedure development is reflected in the meetings that were held at NASA's Johnson Space Center. A PSO management team reviewing the lessons of the TMI-2 accident as applied to BFS met with project managers of the Space Shuttle Program to examine the NASA approach to managing a complex project organization. Particular emphasis was placed on the system-oriented procedures used by the flight crew and mission control operators. The techniques applied to the NASA procedures will be reviewed for applicability to BFS procedures.

Conclusion

PSO has been and will remain committed to having thorough, technically correct, clear, and concise station procedures. The BFS procedure development program will establish a systematic on-going effort to improve plant procedures prior to and during plant operation. A

PSO RESPONSE: 10 CFR 50.34(e)(2)(ii)

detailed description of the procedure development program will be presented in the Black Fox Station FSAR.

10 CFR 50.34(e)(2)(iii) CONTROL ROOM DESIGN REVIEWS

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(iii) Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts. During a general meeting between the NRC Staff and the near-term construction permit applicants on April 8, 1981, the NRC Staff stated that the words "'a control room design reflects state-of-the-art human factor principles...' means, an advanced design control room utilizing CRTs and computers, having been designed after a full system analysis in accordance with Appendix B of NUREG-0659, and having had all human factors engineering deficiencies, as described in NUREG-0700, corrected." Subsequently NUREG-0659 was superseded by the issuance of NUREG-0700; however, Appendix B was transferred intact from NUREG-0659 to NUREG-0700. (NUREG-0718, I.D.1) 19

PSO RESPONSE:

Introduction

The operators of Three Mile Island, Unit 2 (TMI-2) had difficulty in recognizing and dealing with the accident, in part due to less than adequate control room design. One of the main findings of the various lessons learned documents was that human factors should be considered in the design of nuclear power plant control rooms, to ensure that the operator can effectively monitor the operation of the plant safety systems. This Requirement delineates the actions to be taken by construction permit applicants to allow the NRC to review the proposed design prior to fabrication of control room panels.

The NRC Staff has published guidance for meeting this Requirement. The 19
guidance published to date includes NUREG/CR-1580, "Human Engineering Guide to Control Room Evaluation," which was an interim guide for evaluation of control rooms, and NUREG-0659, "Staff Supplement to the Draft Report on Human Engineering Guide to Control Room Evaluation," which reflects a thorough evaluation of comments made on NUREG-1580, and incorporation of guidance f[illegible]ntrol room design. The current document is NUREG-0700, "Guidelines [illegible]rol Room Design Review," which 19
supersedes NUREG/CR-1580 a[illegible]-0659.

PSO RESPONSE: 10 CFR 50.34(e)(2)(iii)

Summary

PSO has been aware of the importance of human factors engineering application to control room designs since the inception of the Black Fox Station (BFS) Project. This concern was a major factor in the selection of the General Electric Company (GE) NUCLENET* 1000 Control Complex for BFS (* denotes General Electric Company trademark). During the development of NUCLENET*, GE established a design approach for the man/machine interface that is almost identical to that contained in NUREG-0659. Following the selection of NUCLENET*, PSO and Black & Veatch Consulting Engineers (B&V), the architect engineer responsible for the balance of plant design, reviewed and analyzed the design, for proper interfacing of the balance of plant controls and instrumentation in the overall control room complex. Extensive design coordination between PSO, B&V, and GE took place during the development of the overall control room design for BFS. This review and coordination included human factors engineering.

The following parts of the response to this Requirement demonstrate that:

1. The process used by GE in developing NUCLENET* paralleled the guidance presented in NUREG-0700. 19

2. The PSO, GE, and B&V review of NUCLENET* and the balance of plant coordination also included application of human factors engineering principles.

3. PSO commits to a control room evaluation plan which will meet the guidance in NUREG-0700. 19

This response demonstrates that PSO is committed to providing a control room design for BFS that reflects state-of-the-art human factor principles. The final panel insert design, consisting of insert location and arrangement drawings including changes made as a result of the final control room evaluation report, will be provided to the NRC for review prior to panel insert fabrication.

I. SYSTEMS/OPERATING ANALYSIS TECHNIQUES USED IN CONTROL ROOM DESIGN

A. INTRODUCTION

NUCLENET* as an advanced, generic control room design, was developed with the same methodology as that set out in Appendix B to NUREG-0700. This process, similar to the systems/ operations analysis process presented in military specification MIL-H-46855, included an analysis of all functions necessary to operate the plant safely, an allocation of functions between operator and machine, and a qualitative verification of the functional allocation. 19

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G.E. assembled a team to design the NUCLENET* 1000 Control Complex which included expertise in: controls and control systems design, computer technology, industrial design, operator training, power plant test and operations, and behavioral science.

The premise upon which the design is based is that optimum control is achieved when there is an allocation of control functions between the operator and machine which recognize that each performs certain functions better than the other, and that, once the allocation is made, the design permits efficient and effective manipulation of controls by the operator.

NUREG-0700 was prepared by the NRC Staff and incorporates guidelines, provides sample checklists and corresponding human engineering guidelines and acceptability criteria for analyzing operator-control room interfaces, draft systems review guidelines, and a discussion of the Staff's planned procedures for reviewing and evaluating licensee control room design review reports. 19

The objective of Appendix B to NUREG-0700 is to describe the systems/operations analysis which the NRC Staff believes is an acceptable approach to control room design. The technique systematically defines the equipment, personnel, and procedural data requirements to meet all functional objectives of the control room, including safe operation of the plant. The next review step is to compare the planned control room with the design requirements. 19

A systems approach for developing control room design requirements may be characterized as a three-step process. The three steps are functional analysis, functional allocation, and verification of allocation. As stated in NUREG-0700 Appendix B, the purpose of functional analysis is to define all the functions required to operate the plant. The top-level function, safe production of electrical power, is the objective of the total nuclear power plant. 19

One second-level function is nuclear power plant operation, which includes all control room functions. A third-level function is to prevent/mitigate unsafe plant operation. This function consists of many fourth-level functions, such as the shutdown of critical core operation prior to/at the threshold of unsafe operation. Subsequently, fifth-level functions include insertion of sufficient negative reactivity to result in the shutdown of critical core operation, and monitoring to assess the execution and completeness of the shutdown.

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The next step in the development of control room design requirements is the allocation of functions to operator or machine. The basis of the allocation is anticipated performance of the operator or the machine in the implementation of the function. The anticipated performance is based on actions in which humans excel, such as the ability to exercise judgment where events cannot be completely defined, and on actions in which machines excel, such as carrying out many tasks simultaneously. Functional allocation based on anticipated performance of the operator or the machine is preferred. When problems are encountered in verification and validation of functional allocation, it may be necessary to revert to the allocation step to resolve the issue.

The last major step in the development of control room design requirements is verifying functional allocation and validating functional integration. All functions allocated to the control room operating crew should be executable from the control room. Verifying these allocations consists of a series of analyses to ensure that the operator is not overloaded in terms of work and that tasks can be correctly executed. Properly implemented, these analyses will result in optimal utilization of time and space at the human-machine interface, thereby establishing the detailed design requirements for work stations and operator/crew member performance. Upon completing the process of verifying that each system will perform its assigned function, the next effort is to validate the total design to ensure the successful integration of all functions.

The following discussion parallels the format of NUREG-0700, Appendix B and describes how the NUCLENET* design effort corresponds to the recommendations outlined in that document. 19

B. FUNCTIONAL ANALYSIS

1. Definition of Objectives

An essential step in the methodology was to define and evaluate the functions and actions to be performed in meeting system objectives. In determining these functions, it was essential to first identify the many activities necessary to operate the plant safely, under both normal and abnormal conditions.

2. Definition of Functions

Once having identified these activities, the next step was to combine activities under functional groupings, chosen in such a manner that they would be both understandable to the operator and allocated and distributed in such a way to permit effective operator action.

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a. Design Functional Groups

For normal operation the activities were grouped under the following functions:

(1) Provide and maintain normal core coolant

(2) Control reactivity

(3) Monitor performance of the core

(4) Control reactor pressure

(5) Utilize steam for power conversion

(6) Convert mechanical power to electrical power

The above functional grouping is very similar to those listed in Appendix B to NUREG-0659 (p. B-15) which are:

(1) Nuclear reactor reactivity control

(2) Reactor core cooling

(3) Reactor coolant systems integrity

(4) Primary reactor containment integrity

(5) Radioactive effluent control

(6) Power generation

(7) Power transmission

b. Operational Conditions Considered

More detailed identification of plant system control functions has been made by considering operational situations and events that will or may confront operators in the Control Complex. The operational situations and events considered consist of:

(1) All events required to be assessed by Section 15, "Accident Analyses," of the BFS PSAR

(2) Normal operation of the plant

(3) Failures in systems, subsystems, and components, and human errors

(4) Anticipated operational occurrences, including startup and shutdown of the plant

(5) Task Action Plan I.C.1, NUREG-0660 and NUREG-0737 (although these documents did not exist during the initial BFS design process, the documents have been reviewed and a determination made that no design conflicts exist)

3. Decision/Information Requirements

For each significant activity within a functional group, a display was developed which measured each activity against criteria which indicated the type of information essential for making decisions regarding the man-machine interface.

a. Design Objectives

Human factor engineering design objectives were also developed to reflect the goals to be achieved in a new Control Complex design. These objectives are:

(1) Provide a more efficient, coordinated control of the BWR than that attained with a conventional control room.

(2) Integrate planned operation functions for steam supply and power conversion systems into a single operator station.

(3) Improve operator response time and reduce operator errors by determining the optimum quantity of data and number of display devices which the operator must continuously survey, analyze and comprehend.

(4) Improve operator performance requirements by determining how best to centralize and integrate an optimum number of control devices which the operator must manipulate.

(5) Incorporate efficient hardware and software display techniques in order to present timely, useful information which is meaningful to the operator.

(6) Provide for factory testing and evaluation of the entire Control Complex.

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b. Design Criteria

The following design criteria were developed to achieve the design objectives:

(1) Functions in the Control Complex shall be assigned to three types of panels:

(a) Primary Operator Interface panels (sometimes referred to as Front Row panels)

(b) Secondary Operator Interface panels (sometimes referred to as Second Row panels)

(c) Back Row panels

(2) Major power generation systems shall be integrated for planned operations to centralize and minimize the primary operator interface by:

(a) Separating the operator's short response functions from the long response functions

(b) Making frequently used functions and normal reactivity controls readily accessible from or at the operator's normal duty station

(c) Providing a Display Control System for bringing operational data to the operator.

(3) The planned operating functions of the core standby cooling systems shall remain integral with the appropriate cooling system, and their direct support systems in order that the design of the benchboard used for operator interface with these engineered safety features shall not affect licensability of the Control Complex.

(4) Integration of the Nuclear Steam Supply and Balance of Plant functions shall not degrade the capability for power generation.

4. Functional Integration and Interactions

The relationships and interactions between control functions have been defined and evaluated to ensure that all plant operations and safety objectives can be achieved. These relationships and interactions provide a basis for the development of Control Complex design requirements, and, can serve for future design modifications if necessary.

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a. Human Factors Application

(1) The arrangement of panels shall ensure that each panel defined as primary operator interface will have control, display, and annunciator areas visible to an operator from his normal duty station.

(2) The distance from the operator's normal duty station to the most remotely located function on a primary operator interface panel shall not exceed 30 walking-line feet.

(3) Normal operations functions shall be placed within the reach span of a single operator without compromising the integrity of those systems having multifunctional capability.

(4) Align each system's information devices and controls vertically, with information devices above controls.

(5) Align system's operations horizontally, or vertically in the order of the flow path.

(6) Arrange control functions in an array which is meaningful to the operator. Provide mimic of complex control systems representing the system's process flow and component orientation as an aid to the operator's job performance.

(7) Maintain system functional integrity in the human-machine interface to aid operator's comprehension of process behavior.

(8) Use miniature devices for controls without sacrificing safety or reliability.

(9) Provide a Display Control System which presents normal operations information in pre-defined formats, determined by operational analyses, as well as presenting Alarm Initiated Displays (AID). Incorporate: Color and Shape Coding.

(10) Display by exception, where too much information is not meaningful to the operator and could cause sensory overload.

(11) Provide means for power variation and safe shutdown in the event of catastrophic failure of

the Display Control System, yet maximize its availability ($\geq$ 99.5%).

C. ALLOCATION OF FUNCTIONS

1. Selection of Vital Systems

A systems analysis was conducted to determine which systems vital to operation of the plant could be controlled from the single operator station, designed to be the primary operator interface with the control of the plant. It was determined that these systems (defined as System Groups) were:

a. Reactor Water Cleanup System

b. Condensate Pumping System

c. Feedwater Pumping and Reactor Level Control System

d. Reactor Recirculation System

e. Rod Control and Information System

f. Neutron Monitoring System

g. Steam Bypass and Pressure Regulator System

h. Main Turbine Control System

i. Generator Control System

2. Allocation Categories

Human factors engineering principles and criteria were used to evaluate human-machine interfaces in analyzing performance requirements for plant control functions and for the allocation of functions to categorize these nine systems. Allocation categories consisted of:

a. Automatic Operation by Plant Systems Equipment

b. Manual Operation by Control Room Operators

c. Combination of a. and b. above

3. Capability and Limitation Factors

The design evaluation allocation criteria considered the capabilities and limitations of the operator(s) and systems, along with cost-benefit considerations of automating in those instances where the operator and system could perform a given task approximately equally well. Factors comparable to those listed in Exhibit B-3 of Appendix B to NUREG-0700 were used in making the allocation of functions. | 19

a. Operator Processing Capabilities

Plausible human roles of operators and supervisors (e.g., control manipulator, instrument monitor, supervisor, decision maker, communicator, coordinator) have been defined. Qualitative information processing capability in terms of load, accuracy, rate, and time delay have been prepared for each operator/supervisor information processing function.

b. System Processing Capabilities

Plausible system roles of Control Complex equipment (automatic control of reactor power level, reactor trip system, engineered safety feature) have been defined. Information processing capabilities and control function response times of control systems equipment have been defined considering load, accuracy, rate, and time delay for processing and response.

c. Responsibility for Plant Safety

The overall responsibility for the top-level assessment of plant operating and safety status has been allocated to the human operator(s). The rationale for this allocation is based on the cognitive abilities of humans, which cannot be duplicated by a machine. The information requirements to exercise this responsibility determine methods for transfer of plant systems data and information to the operator(s) in the Control Complex.

4. Results of Allocation

A summary description of the allocation of System Group functions follows:

a. Reactor Water Cleanup System

This system is operated manually. This is an instance where the operator or machine can perform approximately equally well, and the system objective is achieved by manual operation. This operation does not overload the operator, and it was not cost-beneficial to automate.

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b. Condensate Pumping System

This system is primarily manual. The operator must reach a decision on when and how much water to pump. The decision is based on the operator's ability to observe a wide variety of stimuli and to reach a judgment based on those observations. This is an activity in which the operator is superior to the machine. Once the operator takes the manual action, other actions in the system are carried out automatically, such as maintaining the hot well water leve... The automatic operation is best suited to the machine.

c. Feedwater Pumping and Reactor Level Control System

During power operation this system operates automatically since its function is to monitor and perform the routine task of maintaining proper reactor water level.

d. Reactor Recirculation System

This system can be operated semi-automatically or manually, and is another example of approximately equal capability between the operator and the machine to perform a task. Manual operation, when used, does not overload the operator.

e. Rod Control and Information System

The operator manually initiates the action for operation of this system, based on a judgment of when it should be operated. This judgment is reached after considering a wide variety of information, a task in which the operator excels. Once control action is initiated by the operator, the system functions automatically to ensure rods do not exceed established limits while being withdrawn. This automatic function is ideal for the machine. The portion of the Rod Control and Information System (RCIS) which controls Control Rod sequences and patterns during startup, shutdown and power operation, namely the Rod Pattern Control System, is not initiated by the operator. This system is a hard wired scheme for which the operator has limited bypassing ability for a limited number of control rods.

f. Neutron Monitoring System

The operator must insert and withdraw Intermediate Range Monitors (IRM) and Source Range Monitors (SRM) during startup and shutdown. Also during these phases of operation the operator must change IRM ranges. Once the limits within which this system must operate are established by the operator, the system performs its monitoring functions automatically. This is a monitoring function in which machines excel.

g. Steam Bypass and Pressure Regulator System

The operator sets the limits appropriate for the phase of operation, and the system operates automatically within those limits.

h. Main Turbine Control System

This system combines manual and automatic operation. The operator manually initiates system operation; the system then operates automatically up to predetermined hold points, to permit the operator to monitor the system's performance and reach a judgment on whether automatic operation should be continued to the next hold point. This system thus combines the most desirable aspects of operator and machine control.

i. Generator Control System

Synchronizing of the unit could be operated approximately equally well either manually or automatically. Manual operation was chosen as preserving the greatest flexibility for integrating balance of plant controls for this system into the Control Complex. Load control is automatically within the limits of the Reactor Recirculation Control.

j. Safety Systems

There are other systems essential to safe operation which are not included in plant control, such as High Pressure Core Spray System, Residual Heat Removal System and most other safety systems. Since NRC requirements dictated that these systems operate automatically, an allocation of functions was not performed for these systems except for those used for surveillance testing and manual intervention.

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k. Balance of Plant Systems

The systems on the BOP Control Benchboard, including Turbine Generator Lube Oil, Steam Supply and Drains, Circulating Water, Condenser Off-Gas and Condensate and Feedwater auxiliary systems, are long-response systems. Most are manually initiated during startup then operate automatically.

D. VERIFICATION OF FUNCTIONAL ALLOCATION

The verification of functional allocation is a detailed assessment and analysis of each allocation to ensure that the correct functional allocation has been made. The verification of functional allocation defines the design requirements and specifications for the systems required by the Control Complex as well as the specifications for quantity of operators, for the interface between operators and a system, for the operational procedures (including emergency procedures), and for maintenance requirements.

1. Verification of Functions Allocated to Machines

For each system function allocated to a machine, the performance requirements of the system, or equipment to execute the function, have been defined. The performance requirement considers such characteristics as response time, accuracy, reliability, and operator interface or display requirements. Points regarding the design of Control Complex systems are:

a. Display Systems

The design requirements for display systems consider established design criteria. Furthermore, the design requirements for display systems contain criteria to display signals that directly and accurately reflect the information to be transferred to the operator. These signals are to the extent practicable a direct measurement of the desired variable. Displayed parameters are selected from which the operator can determine if the systems are performing their design functions or are responding to operator commands.

b. Control Systems

The design of control systems considers the design criteria presented in Appendix A of 10 CFR 50: General Design Criteria 13 and 19 through 29. Utilizing this analysis data and the design criteria previously described, primary and secondary operator interface panels were defined.

(1) The primary operator interface panels include:

(a)	NUCLENET* Control Console	(P680)
(b)	Standby Information Panel	(P678)
(c)	Reactor Core Cooling Benchboard	(P601)
(d)	Diesel Generator Benchboard	(P877)
(e)	BOP Control Benchboard	(P870)
(f)	Auxiliary Electric Panel	(P800)
(g)	Meteorological Information and Dose Assessment Panel	(P900)
(h)	Security/Fire Protection Panel	(P901)
(i)	Supervisory Monitoring Console	(P679)

(2) The secondary operator interface includes all other panels on which are located controls or displays which must be overtly employed by the operator, as opposed to the maintainer.

2. Function Placement

The design criteria were then applied to the operator interface panels.

a. NUCLENET* Control Console (P680)

(1) Normal (after prestart) plant operations functions

(2) Short response functions

(3) Frequently used and/or reactivity controls

(4) Reactor Protection System operator interface

Note 1: Only non-divisional systems related to (1), (2), and (3) above, except Nuclear Steam Supply Shutoff System manual initiation at system level.

Note 2: Exclude functions not related to above.

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b. Standby Information Panel (P678)

(1) Support information of the Display Control System

(2) No process control

c. Reactor Core Cooling Benchboard (P601)

(1) NSS safety systems

(2) NSS long response functions

(3) Standard design with no licensing impact

(4) Maintained divisional integrity

d. Diesel Generator Benchboard (P877)

(1) Safety related diesel generators (Divisions 1 & 2)

(2) Support systems for (1)

(3) Maintained divisional integrity

e. BOP Control Benchboard (P870)

(1) BOP long response functions

(2) Non-frequent use functions

(3) Maintained divisional integrity

f. Auxiliary Electric Panel (P800)

(1) BOP auxiliary electric and power transmission switchyard long response functions

(2) Non-frequent use functions

(3) Maintained divisional integrity

g. Meteorological Information and Dose Assessment Panel (P900)

(1) Plant site meteorological indication

(2) Plant radiation and radioactivity indication

(3) No process control

h. Security/Fire Protection System Panel (P901)

(1) Plant security and fire protection alarms and indication

(2) No process control

i. Supervisory Monitoring Console (P679)

(1) Performance Monitoring System (PMS) interface

(2) No process control

With the Systems Groups assignment to panels determined, the next step was to determine the order of placement of those Systems Groups on the panels. Working on each panel individually, applying the design criteria, a logical order of placement of System Groups upon that panel was deduced.

3. Verification of Functions Allocated to Humans

a. Scope

The most critical portion of the analysis is the verification of functions allocated to humans. Detailed analysis of functions assigned to humans has determined the suitability of the human-machine interface for the performance of the assigned function. Evaluation of the operator's workload has determined if operator overload conditions exist. The product resulting from the analysis of functions allocated to humans should determine requirements for:

(1) Operator training

(2) Operating procedures

(3) Optimal Control Complex human-machine interface and control room configuration

(4) Control Complex staffing.

b. Subfunction and Task Definition

For each function allocated to humans, all subfunctions and tasks including cognitive tasks that must be performed to achieve the function have been defined and arranged in sequence of performance. Manual tasks are specific with regard to actions and information transfers from system to human required to complete the task. The plant procedures used by the control room operator have been reviewed to determine that they provide adequate guidance to perform the plant control functions according to the allocation of functions.

c. Operator Task Analysis

Requirements for operator tasks have been analyzed to ensure that they do not exceed human capabilities. All time-critical functions allocated to the operator have been analyzed to define the time requirements needed to successfully perform each task.

These analyses serve as the basis for specifying the size of the operating crew required, the human performance characteristics required for normal and emergency operations, the operational procedures required for abnormal and emergency operation, and the training requirements for operators.

Based upon the data just derived, the anthropometric data of the intended user population, and the criteria previously stated, a full-scale mockup of the Primary Operator Interface panels was constructed. Sheet styrofoam was used to form the panels. The front surfaces representing the control and display areas were covered with a material whose texture is compatible with the use of "velcro" fasteners.

Systems analysis had determined, in meeting the systems design objectives, which functions were allocated to the operator and which were allocated to the control system. The manner of implementation of those allocations was yet to be tested.

The assumed control and display functional devices, selected for consistency with the design criteria, were photographically reproduced. Small pieces of "velcro" fastener material were adhered to the backs of the devices to permit their placement (and rearrangement) on the mockup.

The system's Operator Interface Devices were placed on the Console and Benchboards in accordance with the design criteria, and in the same order in which they were selected for location on the panel. The devices were rearranged many times, to provide as nearly as possible, the optimum operator orientation.

d. Critical Task Analysis

System operational analyses were performed, by simulating operation of each system, using system operating procedures. The system operating procedures used were those in effect in a plant having nearly identical system(s) design.

Operational analysis was then performed for integrated plant operation, using the plant procedures. As a result of these analyses, device location and arrangement were more nearly optimized.

A task analysis was conducted for those tasks and modes of operation that are likely to have an adverse effect on plant safety if not accomplished in accordance with system requirements. These tasks are identified as critical tasks. An analysis of critical tasks was done to identify:

(1) information required by operator including cues for task initiation

(2) information available to operator

(3) evaluation process

(4) decision reached after evaluation

(5) action taken

(6) body movements required by action taken

(7) workspace envelope required by action taken

(8) workspace available

(9) location and condition of work environment

(10) frequency and tolerances to action

(11) time base and time margins (time margins must be adequate to cover variances in human responses)

(12) feedback informing operator of the adequacy of the actions taken

(13) tools and equipment required

(14) number of personnel required, their specialty, and

experience

(15) job aids or references required

(16) communication required, including types of communication

(17) special hazards involved

(18) Operator interaction where more than one operator is involved

(19) operational limits of personnel (performance)

(20) operational limits of machines and systems

The critical task analysis also included analyzed accident conditions.

During the operational analyses, careful notation was made of the operator's information needs for each phase of system operation. This data would be used to select input variables to the Display Control System (DCS), and to help assign the variables to the various system formats. The immediate use of the data, however, was as a basis for assignment of hardwired, backup information devices to the Standby Information Panel.

e. Work Station Design Analysis

For each work station in the Control Complex, the time sequence of operator activities and the time required for information exchange or transfer to the operator has been defined. The analysis verified that the operator is capable of completing all tasks and that all tasks are capable of being performed using the work station design.

f. Operational Sequence Analysis

An analysis and evaluation of Control Complex sequences of operations, flow of decisions, physical transmissions of data and information, receipts of information, storage of information, monitoring of systems and interactions among operational crew members, work stations, and systems has been conducted. The purpose of the analysis was a validation of the Control Complex capability to successfully complete the intended functions of the design, in both the time and space domain.

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g. Workload Analysis

A workload analysis for all critical functions was conducted to appraise the extent of the Control Complex operator workloads. The analysis was based on the sequential accumulation of task times. Application of this technique permits an evaluation of the capability of the Control Complex operator(s) to perform all assigned tasks in the time required to maintain plant safety.

The detailed workload analysis divided the operator's tasks into categories corresponding to perceptual-motor channels such as vision, left hand, right hand, feet, cognition, auditory, and voice channels. The purpose of this level of detail was to ensure that the operator is not required to perform more than one task at a time if two or more tasks require the simultaneous use of a single perceptual-motor channel nearly 75 percent of the time.

h. Human-Error Analysis

A human-error analysis was conducted for each perceptual-motor channel workload of 75 percent or greater as defined by the results of the workload analysis.

The purpose of the human-error analysis was to investigate the probability of error during high workload conditions and to evaluate the consequences resulting from these errors.

i. Work Station Link Analysis

A work station link analysis has been conducted for each work station used by the operator to perform critical tasks. The analysis defined the frequency and criticality associated with each of the interactions occurring between operator and equipment and/or between one operator and another. The defined frequency and criticality of the interactions are then used to evaluate the design adequacy of the work station layout in terms of time and space utilization. This analysis achieves a near optimal design for the work station, such as the spatial correlation of displays with controls to provide the operator with feedback information as required by General Design Criterion 13, Instrumentation and Control.

j. Procedures Program Requirements

The measurement of human performance in accomplishing the functions and tasks identified by the analyses described above was not completed in the generic design phase. This work will be performed in the BFS Control Room Evaluation described in Part III of this response, and will be used to identify, document and verify the content of the BFS Control Room Normal and Emergency Operating Procedures. Culmination of human factors engineering enhancements will be incorporated into the Black Fox Station procedures as described in Requirement (2)(ii).

4. Validation of System Integrations

A static validation was performed on this initial design effort using the engineering mockup. A dynamic validation of the human factors principles incorporated in the initial design effort will be conducted as part of the BFS Control Room Evaluation described in Part III of this response.

E. INITIAL CONTROL ROOM DESIGN DESCRIPTION

The methodology described above was utilized in the generic design of the NUCLENET* Control Complex. The resulting Primary Operator Interface is described in this section.

1. Display Control System (DCS)

The total design for the DCS required approximately 35 man-years of effort. Some software enhancements continued for almost 7 years after the design initiation. Display format research and development extended over a period of more than 3 years. As a result of studies performed by General Electric, the DCS formats employ the following color coding:

Green - Used only for lines and symbols in process diagrams to represent static system components, i.e., pumps, motors, valves, and piping which are not dynamically presented in the given format. Green was selected for this association because the display elements make up the larger part of the display, and a green hue has been demonstrated to be the least visually fatiguing of the available hues.

Cyan - Used as a supporting hue and applied to alphanumeric identification, scales, and borders.

Yellow - Applied to all dynamic process variable display elements, such as bar graphs and digital data. Selected for this application because of the intensity of its hue. Yellow allows the operator to scan the display and easily identify dynamic information.

Red - Restricted to use as a visual cue for abnormal conditions. Should any variable exceed process limits, the data (bar graph and/or digital) normally displayed in yellow, changes to red. Selected because of the traditional, pre-established psychological associations (populational stereotype) with such conditions, and because intensity allows minimal visual search.

White - Used as a reference mark on scales, adjacent to bar graphs, to indicate process limits, or to present low confidence data.

Magenta - May be used in place of red.

Dark Blue - Shall not be used, due to its visual loss against the normal background color.

Black - Normal background color.

Initial formal definition began from the data gathered during the operational analyses. A set of 63 formats was generated for the process System Groups depicting various levels of each system's operation. Further analysis was performed to determine the relationship of these formats to reactor operation modes.

The DCS design was a continuing process. At this point, however, the operator's controls for retrieval of operational information via the DCS could be defined and located. Each of the ten CRTs on the Control Console would have two multi-position selector switches. One switch would serve for System selection and one for Format selection, thus providing capability of displaying any System Format on any CRT. Two momentary push-buttons would provide a Menu Display and Format Change Enable. It is not necessary that the computer system, which drives the displays, attempt to follow format selection until the operator has placed the Format Select Switch in the position of the Format desired. The operator informs the computer that the System Group and Format selected are those desired for viewing by depressing the Format Change Enable switch. This group of four switches is mounted next to each of the CRTs which they control, including the CRT which is normally assigned to the PMS. Included, for the PMS CRT is a fifth switch (momentary push-button) for assignment of that CRT to the DCS, when necessary.

One of the positions of the Format Select Switch is designated "Master." When any, or all, of the Format Select Switches are in this position, the operator has simultaneous control of those CRTs from a "Master Display Select Matrix" located at his left hand, when seated at the center of the Control Console. The informational needs data, derived from the operational analyses, showed what information the operator needed to either overtly employ, or have available to him, during which phase of plant operation. The Master Display Select Matrix is used, by the operator, to inform the computer which mode of reactor operation he is performing. The computer then displays those System Group Formats determined to be most meaningful to that phase of operation. Thus the operator is only required to perform a single action to have appropriate data retrieved and displayed to him.

2. Front Row (Primary Operator Interface) Panel Layout (Figure (2)(iii)-1)

   a. NUCLENET* Control Console (P680) (Figure (2)(iii)-2)

      The most critical controls and displays should be placed in the center of the operator's work station.

      In a nuclear power plant the most critical controls and displays are those which are used to control and monitor the intended performance of the reactor core. In the BWR, these are the Rod Control & Information System, the Reactor Protection System and the Neutron Monitoring System. These were, therefore, placed nearest the center of the Console.

      There must be water to act as moderator for the fission process and cool the core. In the BWR, steam is generated within the core and, after being scrubbed and dried, carried off to directly drive the Turbine-Generator. With the reactor core at the center of the Console, as the point of reference, if water comes in and steam goes out, there exists the left to right expectancy of: water into the core; water and steam in the core; and, steam out of the core. Therefore, the water system groups were placed on the left side of the Console, and the steam system groups on the right.

PSO RESPONSE: 10 CFR 50.34(e)(2)(iii)

The Reactor Recirculation System controls reactivity, as a function of flow. It was placed on the left side of the Console, nearest the center. The Condensate Pumping and Feedwater Pumping and Level Control Systems indirectly control reactivity. They were placed next to the Reactor Recirculation System. The remaining water system, the Reactor Water Clean-Up System (RWCU), which bears a functional relationship with another system was placed on the far left side of the Console. (This functional relationship will be explained during the discussion of the other system).

Reactor pressure control is performed by the Steam Bypass and Pressure Regulator System. Pressure directly affects reactivity; therefore, this system was placed on the right side of the Console, nearest the center. The Turbine Electro-Hydraulic Control System controls steam utilization by the Turbine. It was placed next to the Steam Bypass and Pressure Regulator System. The Generator is directly coupled to the Turbine, and was therefore placed next to the Turbine.

There are two more systems which were placed on the Console, one of which had been included in the previous analysis. The Performance Monitoring System is an operational aid which provides the capabilities of:

(1) NSS performance calculations, Sequence of Events, Status alarm, and Post-incident data recall

(2) BOP performance calculations and logs

(3) Displays of NSSS performance calculations results

(4) Means of displaying operations information to supervisory personnel

(5) Means of off-line generation of new display formats for both computer systems.

The Performance Monitoring System's Operator Interface was placed on the far right side of the Console.

The other system was the new Display Control System (DCS), so named because it was to be used to provide information displays which bring operations data to the operator.

There were ten color CRTs placed on the Console, one to be associated with each of the System Groups and one to be used primarily by the Performance Monitoring System, with switching capability to the DCS. The operator's controls for the DCS were located on the Console as previously described.

b. Standby Information Panel (P678)(Figure (2)(iii)-3)

Until the calculated DCS reliability (≥.995) could be verified operationally, it was necessary to provide sufficient hardwired information displays (as well as a DCS Configuration/Status Display) to allow continued steady state power operations, reasonable power maneuvers in the Run Mode, or a safe shutdown, without reliance on the DCS. The Standby Information Panel serves no other purpose. There are no process controls or annunciators on the panel. There are no displays which were not determined to be necessary as a result of the operational analyses.

The Standby Information Panel stands behind the Control Console. Initially, it was intended to be in the direct view of a standing operator. It was later determined that the front silhouette of the Control Console could provide a visual path for the seated operator. The standby information displays for each system controlled from the Control Console were located, accordingly, on the panel.

The Standby Information Panel is located four feet behind the Control Console to allow clearance for CRT replacement in the Control Console, but still maintain the information displays within the visual range of a licensed operator.

c. Reactor Core Cooling Benchboard (P601)(Figure (2)(iii)-4)

Order of system placement on P601 was based on the sequence and frequency of operation, as well as the relationship of a particular system to other systems.

Of those systems assigned to P601, there is one system which bears a functional relationship with the RWCU. It is the Control Rod Drive (CRD) System.

During fueling of the reactor, there are times when it is neither desirable nor practicable to operate the Control Rods. Since the Control Rods are hydraulically operated via controlled leakage carbon seals, when the CRD system is operated, water inventory in the reactor vessel is increased, if not compensated for. One of the functions of the RWCU is to compensate for water level increases, during reactor startup, by providing a controlled drain. When the operator starts up the CRD system after an outage he must control reactor water level through the RWCU. This functional relationship establishes the need for the CRD and the RWCU to be in close proximity to each other, even though on two separate panels.

Hence, the CRD system must be located on P601 at the end closest to the Console, and that end of P601 must be located in close proximity to the left side of the Console. Panel arrangement and key plan are both anchored by this relationship.

In a nuclear plant, the integrity of the Nuclear Steam Supply is of vital importance. Leakage from both controlled and uncontrolled sources must be monitored to verify the degree of that integrity. Controlled leakage is collected in Equipment Drain Sump(s) before being pumped to the Low Conductivity Radwaste. Uncontrolled leakage is collected in Floor Drain Sumps before being pumped to the High Conductivity Radwaste. The frequency of monitoring and recording the leakage collected and pumped out to Radwaste dictates that the information would be as close as possible to the operator. This function is therefore located on P601 next to the CRD system.

The next most frequently used functions are those of the Main Steam System: Safety/Relief Valves; Main Steam Line Isolation Valves; and the Steam Line Drains. These functions are located next to the CRD system and Drain Sumps. The Standby Liquid Control System (SLC) has very few Operator Interface devices, and, in point of fact, has never been deliberately operated to inject negative reactivity into the core. The SLC system controls and displays were located next to the Main Steam System.

Core standby cooling is functionally allocated to: the Residual Heat Removal System (RHR); the Low Pressure Core Spray System (LPCS); the Reactor Core Isolation Cooling System (RCIC); and the High Pressure Core Spray (HPCS) System. These systems were assigned to locations on P601 in that order.

d. Diesel Generator Benchboard (P877)(Figure (2)(iii)-4)

The Diesel Generator Panel contains controls for the division 1 and 2 emergency diesels and support systems such as fuel oil transfer. The operation of the emergency diesels is automatic with loss of preferred power or LOCA. This panel contains displays to verify the system is operating according to design such as voltage and frequency. The panel contains controls to synchronize the diesels onto the Standby AC Power Supply (SACPS) System. Alarms and indication are provided for the SACPS System. The panel is located next to the High Pressure Core Spray (HPCS) diesel which is on one end of P601. This allows the operator to address the entire plant standby power system at one station.

e. BOP Control Benchboard (P870)(Figure (2)(iii)-5)

The BOP Control Benchboard contains alarms, indication, and controls for plant systems, BOP containment isolation, and BOP auxiliary systems. Systems and functions on this panel are observed routinely but do not require constant operator attention during normal operation. The most frequently addressed systems are arranged closest to the Control Console. Systems monitored and controlled from the P870 panel include the following (from left to right):

(1) Reactor Feedwater Turbine and Pump Lubricating Oil and Miscellaneous Auxiliary Systems

(2) Main Turbine Lubricating Oil System

(3) Electro-Hydraulic Control System

(4) Main Turbine Turning Gear

(5) Main Steam Drains

(6) Generator Cooling Auxiliaries

(7) Seal Steam System

(8) Reactor Feedwater System

(9) Reactor Feedwater Turbine Steam Drains

(10) Extraction Steam System

(11) Heater Drain System

(12) Condensate System

(13) Main Condenser Evacuation System

(14) Circulating Water System

(15) Service and Instrument Air System

(16) Condensate Storage and Transfer System

(17) Demineralized Water Storage and Transfer System

(18) Station Service Water System

(19) Raw Water System

(20) Closed Cooling Water System

(21) Turbine Building Cooling Water System

(22) Fire Protection System

(23) BOP Containment Isolation Panel Insert - Controls and indication for safety related isolation valves of non-safety systems (i.e., HVAC, Closed Cooling Water System, Condensate Storage and Transfer System, Demineralized Water Storage and Transfer System, Chilled Water System, Radwaste System, and Service and Instrument Air System) are readily accessible to the operator without violating the requirements of separation between safety related components and non-safety related components (Reg. Guide 1.75).

f. Auxiliary Electric Panel (P800)(Figure (2)(iii)-6)

The Auxiliary Electric Panel contains alarms, indication and controls for station electrical loads. Mimic, status lights, ammeters, voltmeters, and breaker controls are provided for the Normal Auxiliary AC Power System from the Main and Reserve Auxiliary Transformers down to 480 VAC Motor Control Centers. The Power Transmission Switchyard Display and Controls section provides complete monitoring and control of the plant substation. Visual Annunciators alert the operator to potential trouble in the following:

(1) Generator Electrical System

(2) Normal Auxiliary AC Power System

(3) Standby AC Power Supply System

(4) DC Power Supply System

(5) Essential AC Power System

(6) Power Transmission System

(7) Plant Auxiliary Transformers and Lock-Out Relays

The P800 panel arrangement provides the operator with a valuable diagnostic tool through a human engineered view of all plant electrical systems and their interrelationships.

g. Meteorological Information and Dose Assessment Panel (P900)(Figure (2)(iii)-6)

The Meteorological Information and Dose Assessment Panel provides the operator with valuable weather condition and plant radioactivity information to assess both potential natural threats to the plant and environmental effects from plant releases. In addition to conventional indication, the operator interface consists of a CRT and keyboard.

h. Security/Fire Protection System Panel (P901)(Figure (2)(iii)-6)

The Security/Fire Protection System Panel allows the operator to monitor potential security and fire emergencies in order to evaluate the danger posed to the reactor as well as the health and safety of plant personnel and general public. This panel contains an annunciator array, CRT, keyboard, and Halon Release Panel.

i. Supervisory Monitoring Console (P679)(Figure (2)(iii)-7)

The Supervisory Monitoring Console allows supervisory personnel access to the same data available to the operator, without creating a disturbance for the operator by looking over his shoulder. The DCS and the PMS have communications links; therefore, all data in the DCS is available to the PMS.

Supervisory personnel wishing to access DCS data may do so on two color CRTs, communicating via a free-standing, multi-function keyboard which is identical to the keyboard supplied the operator for PMS communication.

The Supervisory Monitoring Console is centrally located between, but at the opposite end of, the Benchboards from the Control Console. This provides supervisory personnel with independent visual access to all of the Primary Operator Interface.

3. Second Row (Secondary Operator Interface) Panel Layout

Second Row panels have a support function to the front row panels and contain displays and controls for equipment addressed by the operator on a less frequent basis than front row controls and displays.

4. Back Row Panels

Back Row panels are defined as those panels in the control room which contain no controls and instrumentation for expected operator use. The hardware in these panels is intended for the maintainer (i.e., instrument technicians), primarily for purposes of testing and diagnostics.

5. Front Row/Second Row Interface

Table (2)(iii)-1 lists all the control room panels, including Front Row, Second Row, and Back Row panels. Location of panels within the control room is illustrated in Figure (2)(iii)-8.

The interface between the Front Row and Second Row Panels varies according to mode of plant operation.

a. Normal Operation

During system setup previous to reactor criticality the operator will align service water and service steam systems. Once the reactor has reached criticality the operator has all the controls needed for normal start-up, operation and shutdown on the front row panel group.

b. Abnormal and Emergency Operations

In the event of an accident assessed by "Accident Analyses" Chapter 15 of the BFS PSAR or the sequence of failure events for transients and accidents analyzed to develop upgraded emergency procedures no operator initiated control is required for at least ten (10) minutes. An operator need only monitor that the systems are performing their function. Only if the operator verifies that an automatic function has not initiated will manual initiation be required.

PSO RESPONSE: 10 CFR 50.34(e)(2)(iii)

II. BLACK & VEATCH AND PSO CONTROL ROOM DESIGN EFFORTS

A. INTRODUCTION

PSO has had a long term commitment to optimize the BFS control room design. Early in the design process a group of experienced power plant engineers, designers, and operators from Black & Veatch (B&V) and PSO were assembled. The tasks facing this review group were to evaluate the human engineering qualities of the GE design and to work in conjunction with GE to integrate the balance of plant (BOP) controls and instrumentation into the NUCLENET* Control complex Primary Operator Interface.

B. OBJECTIVES

Since GE had already performed most of the detailed human factors engineering analysis, the PSO/B&V control room group concentrated on specific areas. The following major objectives were stressed:

1. Overall Design Consistancy

   To enhance operator performance capability the hardware used in the BOP portions must be consistent with that supplied by GE.

2. Component Grouping

   To enhance operator orientation, related controls and indications must be grouped together in a meaningful configuration.

3. Component Location

   To enhance efficient plant operation during normal and emergency conditions, system interface must be located with regard to importance to plant safety and frequency of use.

4. Color Schemes

   The use of color schemes must minimize confusion and adverse psychological impact. Component color coding must serve as a guiding tool to plant operation and must be readily distinguishable in each context.

5. Environment

   The lighting, temperature, humidity, air quality or other environmental factors in the control room must not adversely affect operator concentration and alertness.

6. Workspace

The operator must be provided ample, but effective working space. Consideration must be given to location and arrangement of writing surfaces, document storage and viewing, and communication devices. The workspace arrangement must not obstruct access to plant controls and indications.

C. CONTROL ROOM MOCKUP

PSO and B&V constructed a full scale mockup of Primary Operator Interface to test the above listed objectives. Utilizing this tool, experienced operators participated in a walk-through of procedures to identify human factors discrepancies. Design change options were evaluated on the mockup.

D. LIGHTING SYSTEM DESIGN

Specialized lighting systems were designed to accommodate the lighting needs of an operator using paper, normal hardwired instruments, and CRTs to minimize glare and other viewing problems.

E. WARNING SYSTEM DESIGN

Human factors evaluations of the warning system led to several recommendations to improve the perception of plant status while minimizing the operator's search time and irritation with annunciators.

F. WORKSPACE LAYOUT

Within the control room mockup, workspace arrangements were evaluated. A preliminary design was developed which provided the necessary equipment for efficient administrative operations in the control room.

G. CRT FORMAT DESIGN

The advanced CRT display system, which replaces most of the normal power generation indicators and recorders, received special human factors attention. The data set was selected based on established information needs for various plant procedures and modes. This, combined with human engineered formating and consistant placement and color coding, minimizes the operator's search time, visual angle to obtain data, and possible overload by extraneous data. PSO and B&V participated with GE and other utilities in meetings to select CRT format colors and system function displays which would optimize plant status information available to the operator.

H. RESULTS

The work of the PSO/B&V control room review group contributed to the NUCLENET* 1000 Control Complex overall design. Two basic accomplishments resulted. First, B&V's control system design experience and PSO's plant operations experience added new perspectives which led to modifications of the GE design. Second, the control room BOP design was completed sufficiently early to support the construction of the Black Fox Simulator. This simulator has been used to train operators since early 1979 and has served as an extremely valuable tool for evaluating the effectiveness of such an advanced control room.

III. BLACK FOX STATION CONTROL ROOM EVALUATION

A. GENERAL DESCRIPTION

PSO is developing a Control Room Evaluation Plan for Black Fox Station to provide an over-all evaluation of the control room design. The evaluation plan will contain the guidelines and procedures to meet the following objectives:

1. Human Factors Engineering

   Verify state of the art human factors engineering is applied to any changes to the initial panel design.

2. Plant System and Component Operability

   Through the operator interface, verify appropriate size, number, location, and type of components/systems to ensure the safest and most efficient plant operation.

3. Post-TMI Issues

   Analyze post-TMI issues which potentially impact the design and operation of the control room.

4. Plant Procedures

   Analyze and upgrade Black Fox Simulator operating and emergency procedures to enhance operator training and to aid the development of BFS control room procedures (Requirement (2)(ii))

5. Simulator

   Identify and recommend potential changes to improve simulator instructional capabilities, correct modeling deficiencies, and eventually modify the Black Fox Simulator so that it more exactly duplicates the as-built BFS unit one control room (Requirement (2)(i)).

The BFS Control Room Evaluation Plan parallels the guidelines set forth in NUREG-0700. 19

TABLE (2)(iii)-1
BLACK FOX STATION CONTROL ROOM PANELS

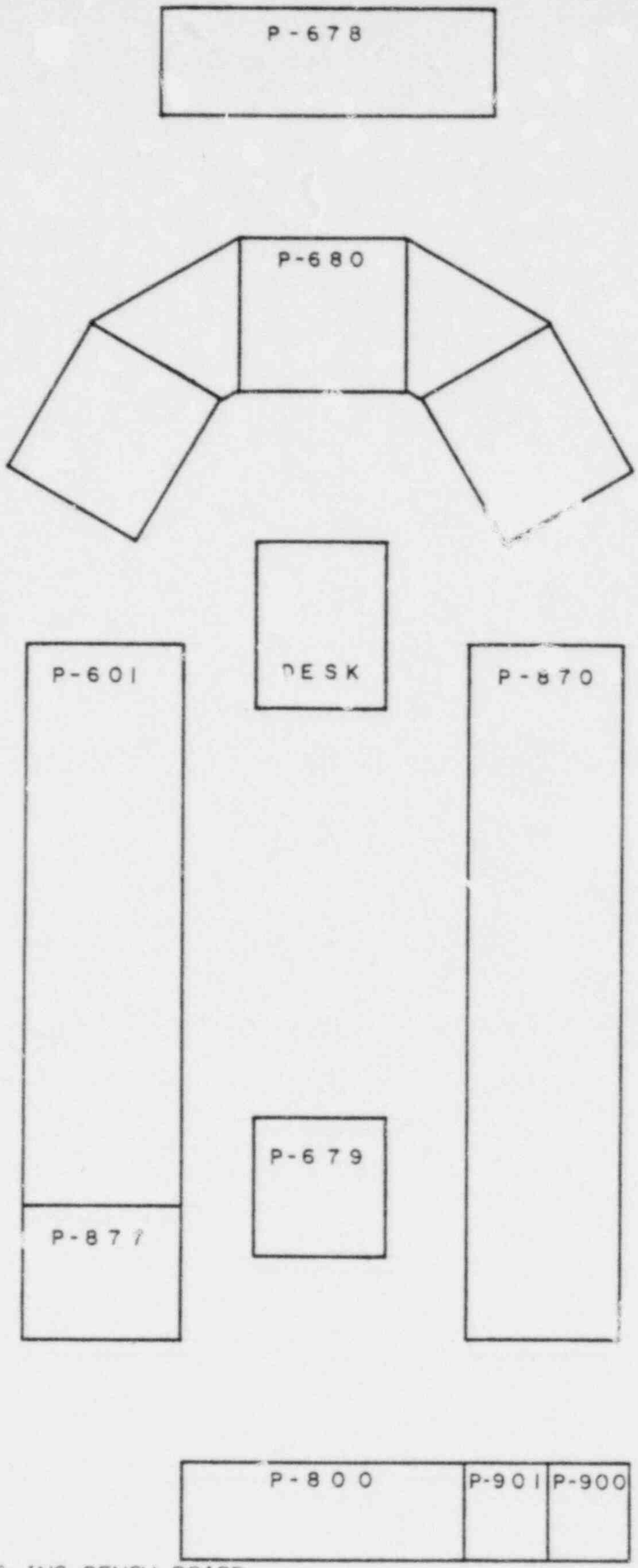
PANEL NUMBER	PANEL NAME
P600	Process Radiation Monitor/Display Control System/Remote Digital and Remote Analog Units Panel
P601	Reactor Core Cooling Benchboard
P604	Process Radiation Monitor Panel
P605	Area Radiation Monitor Panel
P607	Transversing Incore Probe Panel
P610	Control Rod Test Panel
P612	Feedwater Recirculation Panel
P613	Nuclear Steam Supply Panel
P614	Nuclear Steam Supply System Reactor Recirculation Temperature Recorders Panel
P619	Jet Pump Panel
P630	Nuclear Steam Supply Remote Annunciator Electronics Panel
P632	Leak Detection Panel
P634	Recirculation Control Panel
P637	Nuclear Steam Supply System Steam Bypass and Pressure Regulator Panel
P638	Containment Atmosphere Monitor Panel
P639	Containment Atmosphere Monitor Panel
P640	Transient Test Panel
P642	Leak Detection Panel
P651	Rod Control and Information System Panel
P652	Rod Control and Information System Panel
P653	Rod Drive Control Panel
P654	Main Steam Isolation Valve Leakage Control Panel
P655	Main Steam Isolation Valve Leakage Control Panel
P656	Remote Digital Units and Remote Analog Units Panel
P657	Display Control System, Remote Digital Units and Remote Analog Units Panel
P661	Nuclear System Protection System Panel - 1E
P662	Nuclear System Protection System Panel - 2E
P663	Nuclear System Protection System Panel - 3E
P664	Nuclear System Protection System Panel - 4E
P669	Neutron Monitoring and Process Radiation Monitoring Panel - 1E
P670	Neutron Monitoring and Process Radiation Monitoring Panel - 2E
P671	Neutron Monitoring and Process Radiation Monitoring Panel - 3E
P672	Neutron Monitoring and Process Radiation Monitoring Panel - 4E
P678	Standby Information Panel
P679	Supervisory Monitoring Console
P680	Nuclenet* Control Console
P800	Auxiliary Electric Panel
P803	Station Load and Totalizing Equipment Panel
P804	Telemetering Equipment Panel
P821	Turbine Electro Hydraulic Control Panel
P822	Advance Turbine Supervisory Instrumentation Panel
P826	Process Radiation Monitoring System - Balance of Plant Panel 1E
P827	Process Radiation Monitoring System - Balance of Plant Panel 2E
P831	Reactor Feed Pump Turbine Control Equipment Panel

TABLE (2)(iii)-1 (cont'd)
BLACK FOX STATION CONTROL ROOM PANELS

PANEL NUMBER	PANEL NAME
P843	Process Radiation Monitoring System - Balance of Plant Panel 5B
P845	Off-Gas Panel
P847	Standby Gas Treatment System and Containment Combustible Gas Control System Panel - 1E
P848	Standby Gas Treatment System and Containment Combustible Gas Control System Panel - 2E
P850	Balance of Plant Remote Annunciator Electronics and Sequence of Events Recorder Panel
P851	Balance of Plant Engineered Safety Feature Panel 1E
P852	Balance of Plant Engineered Safety Feature Panel 2E/3E
P855	Loose Parts Monitoring Panel
P856	Relay Panel
P857	Relay Panel
P858	Balance of Plant Instrument Power Supply and Signal Condition Panel
P863	Heating Ventilating Air Conditioning Panel 1E
P864	Heating Ventilating Air Conditioning Panel 2E
P865	Flamability Control Panel 1E
P866	Flamability Control Panel 2E
P870	Balance of Plant Control Benchboard
P873	Seismic Vibrations Panel
P874	Vibration Monitoring Panel
P877	Division I & II Diesel Generator Benchboard
P900	(C95-P800) Meteorological Information/Dose Assessment Panel
P901	(P87-P800) Security/Fire Protection Panel
P903	(C94-P800) BOP DCS RAU's & RDU's
COMPUTER PROCESSING UNITS AND PERIPHERIALS	
C-91 Performance Monitoring System (PMS)	
P600	Central Systems Unit
P603	NSS Drum Cabinet
P607	Common Core
P608	Magnetic Tape Cabinet
P609	Disc Memory Cabinet
P612	Display Generator Cabinet
P613	NSS Analog Cabinet
P620	Digital Cabinet
P630	Terminet 1232 MSR
P631	Terminet 1232 MSR
P632	Terminet 1232
P633	Terminet 1232
P636	Line Printer
P638	Card Reader
P639	Card Punch
P642	Results Center Console
P645	Card Reader/Punch Table
R615	Video Copier (Sits on P642)
C-94 Display Control System (DCS)	

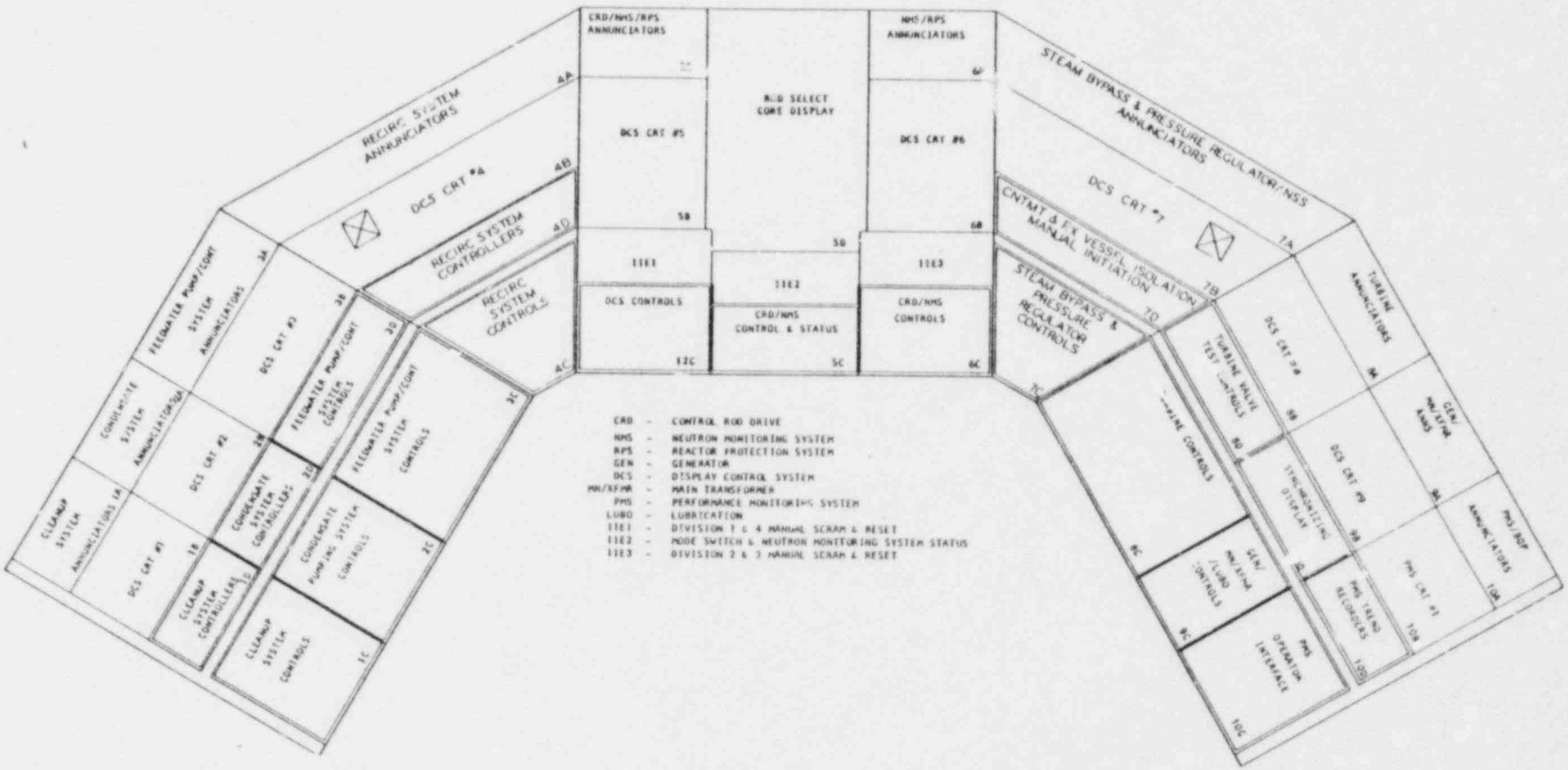
TABLE (2)(iii)-1 (cont'd)
BLACK FOX STATION CONTROL ROOM PANELS

PANEL NUMBER	PANEL NAME
P600	Test and Reconfiguration Unit Cabinet (TRU)
P601	Central Systems Unit
P602	Central Systems Unit
P603	Central Systems Unit
P604	Central Systems Unit
P605	Display Generator Cabinet
P606	DCS/PMS Common Drum



P601-REACTOR CORE COOLING BENCH BOARD
P678-STANDBY INFORMATION PANEL
P679-SUPERVISORY MONITORING PANEL
P680-NUCLENET CONTROL CONSOLE
P800-AUXILARY ELECTRIC PANEL
P870-BALANCE OF PLANT CONTROL PANEL
P877-DIESEL GENERATOR BENCH BOARD
P900-METEROROLOGICAL INFORMATION AND DOSE ASSESSMENT PANEL
P901-SECURITY AND FIRE PROTECTION SYSTEM PANEL

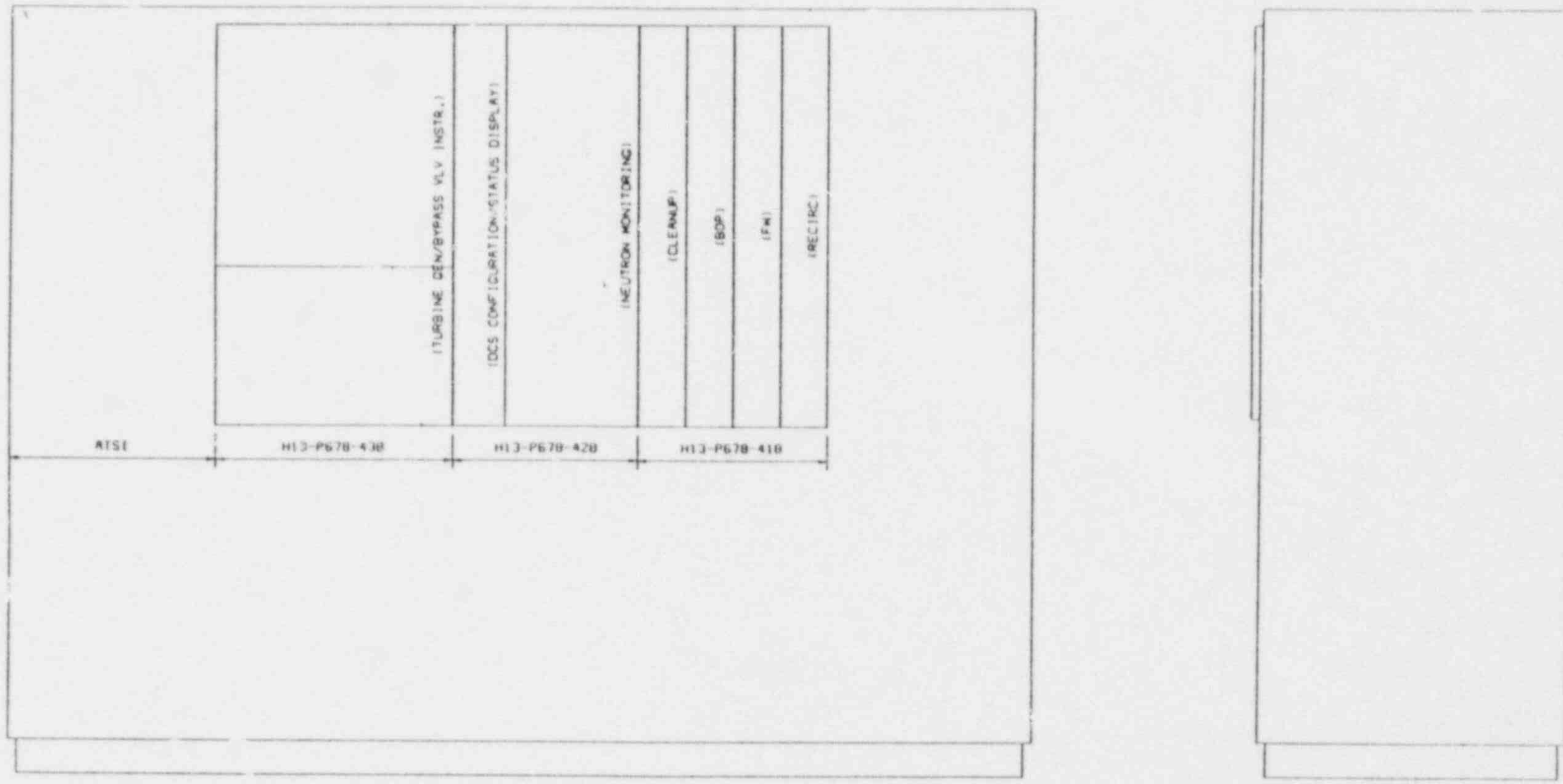
FIGURE (2) (iii)-1
BLACK FOX STATION
CONTROL COMPLEX
PRIMARY OPERATOR INTERFACE



☒ DENOTES HANDSET LOCATIONS

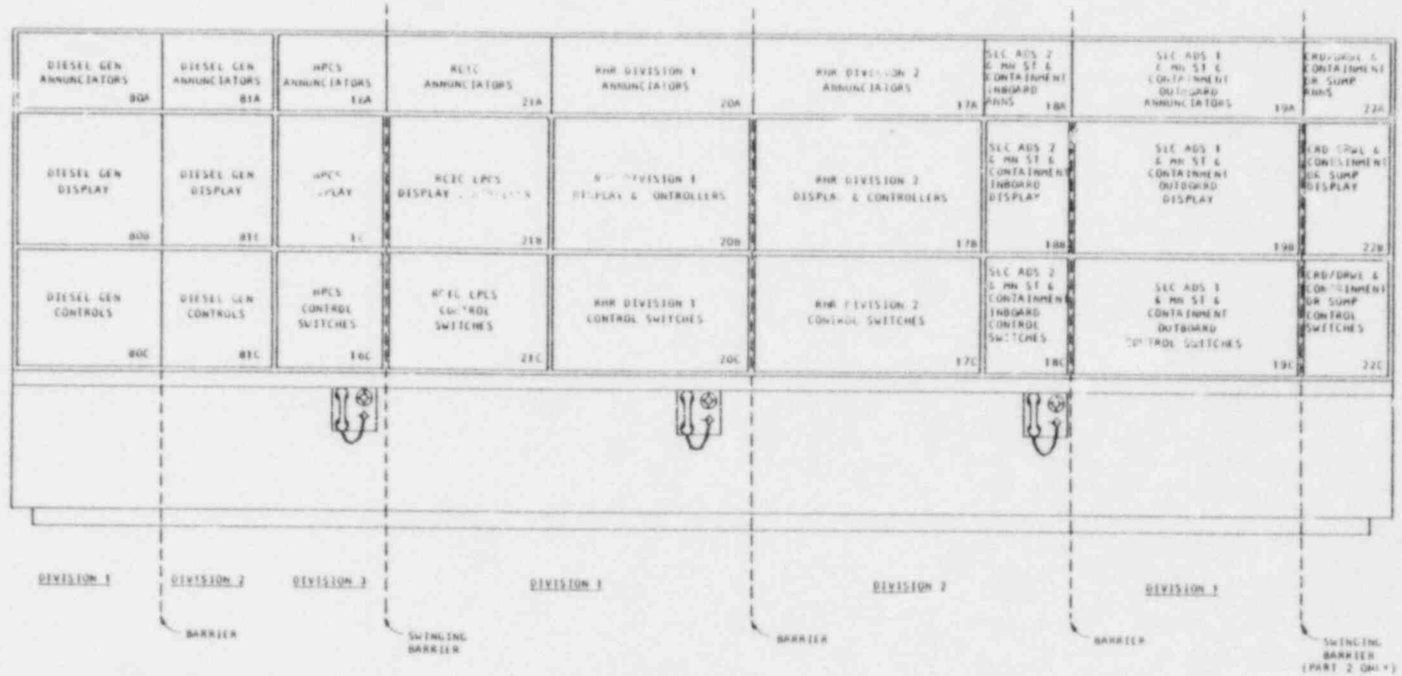
BLACK FOX STATION NUCLENET* CONTROL CONSOLE

FIGURE (2)(iii)-2



BLACK FOX STATION STANDBY INFORMATION PANEL (H13-P678)

FIGURE (2)(iii)-3



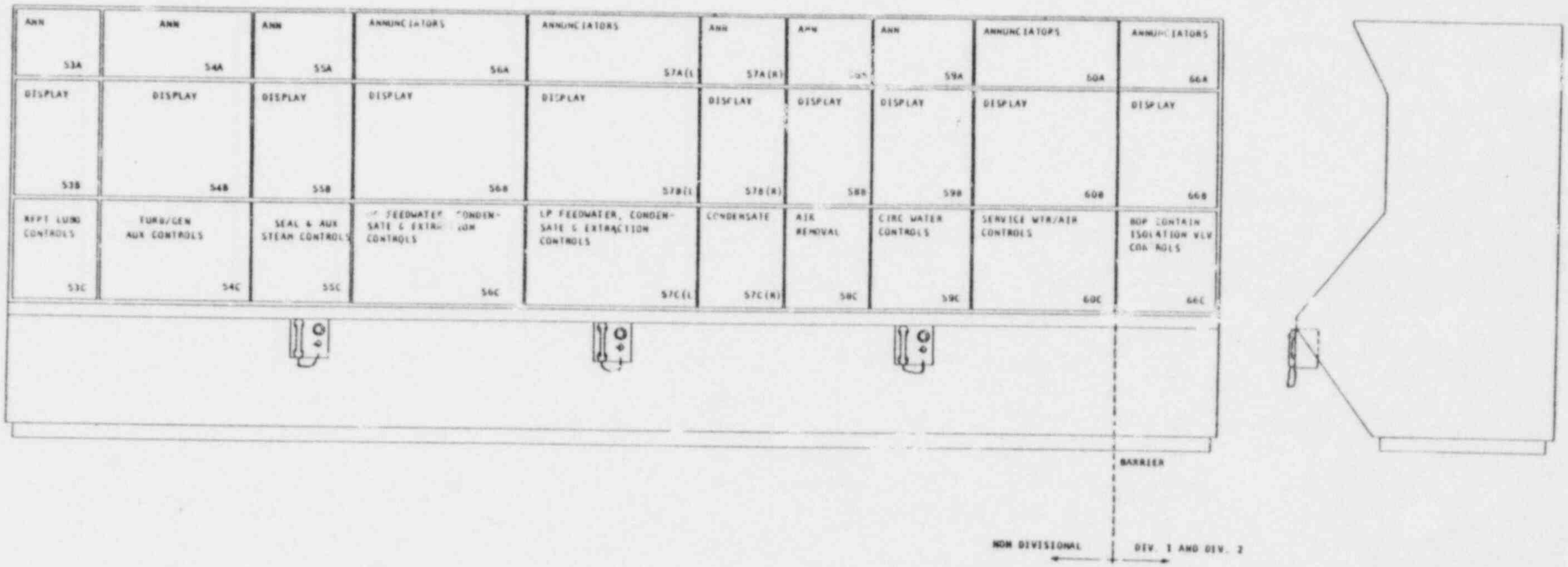
GEN - GENERATOR
HPCS - HIGH PRESSURE CORE SPRAY
RCIC - REACTOR CORE ISOLATION COOLING
RHR - RESIDUAL HEAT REMOVAL
SLC - STANDBY LIQUID CONTROL
ADS - AUTOMATIC DEPRESSURIZATION SYSTEM
MN ST - MAIN STEAM
CRD/DRWL - CONTROL ROD DRIVE AND DRYWELL
DR - DRAIN

BLACK FOX STATION REACTOR CORE COOLING BENCHBOARD (H13-P601) AND DIESEL GENERATOR BENCHBOARD (H13-P877)

FIGURE (2)(iii)-4

68

19-111381



BLACK FOX STATION BOP CONTROL BENCHBOARD (H13-P870)

FIGURE (2)(iii)-5

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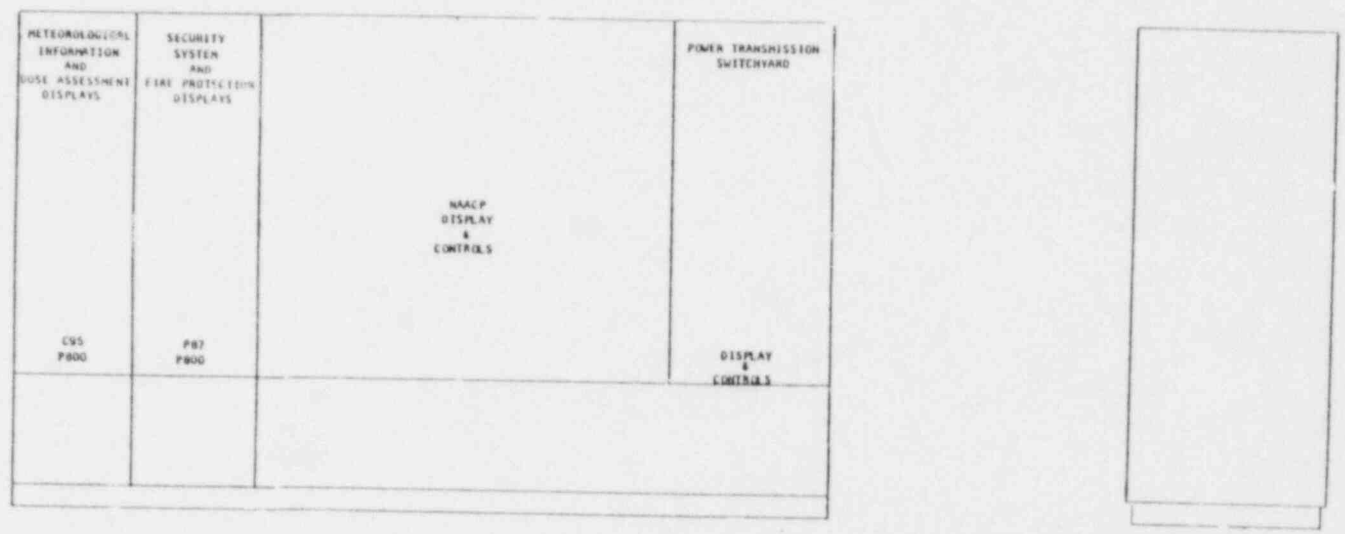
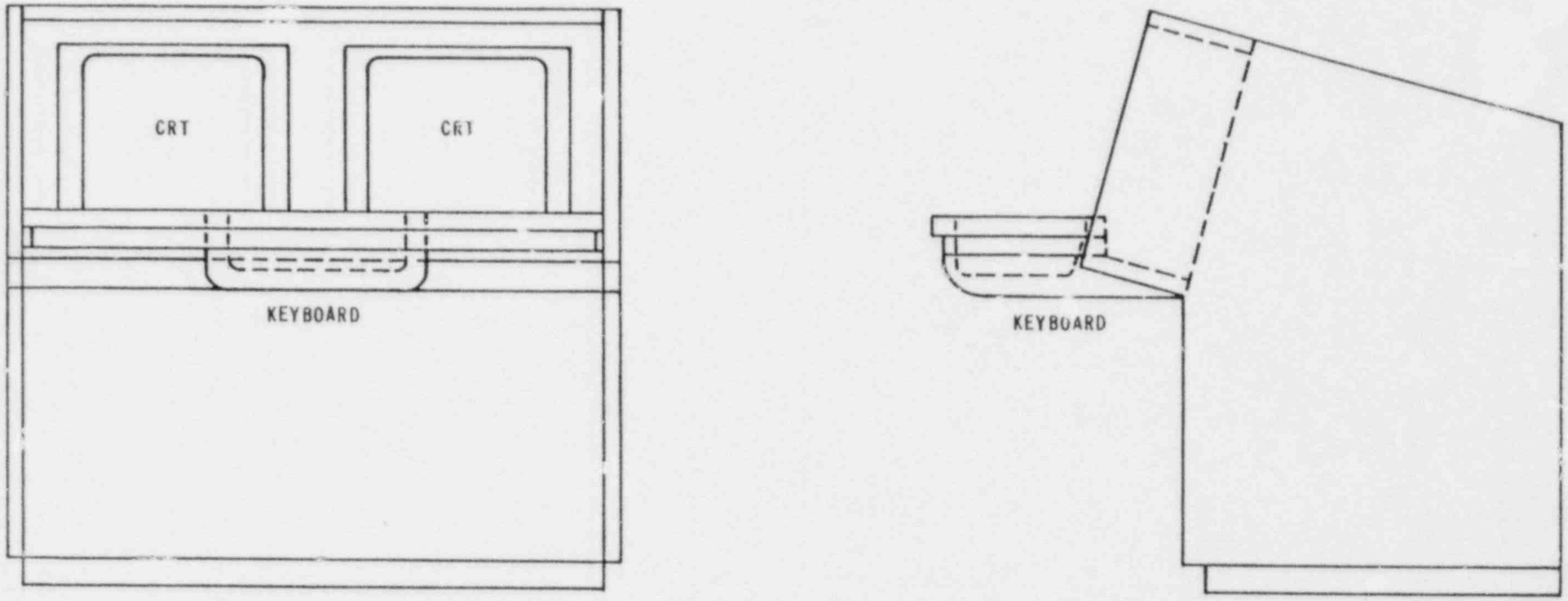


BLACK FOX STATION AUXILIARY ELECTRIC PANEL (H13-P800)
METEOROLOGICAL INFORMATION AND DOSE ASSESSMENT
PANEL (H13-P900) AND SECURITY/FIRE PROTECTION
SYSTEM PANEL (H13-P901)

FIGURE (2)(iii)-6

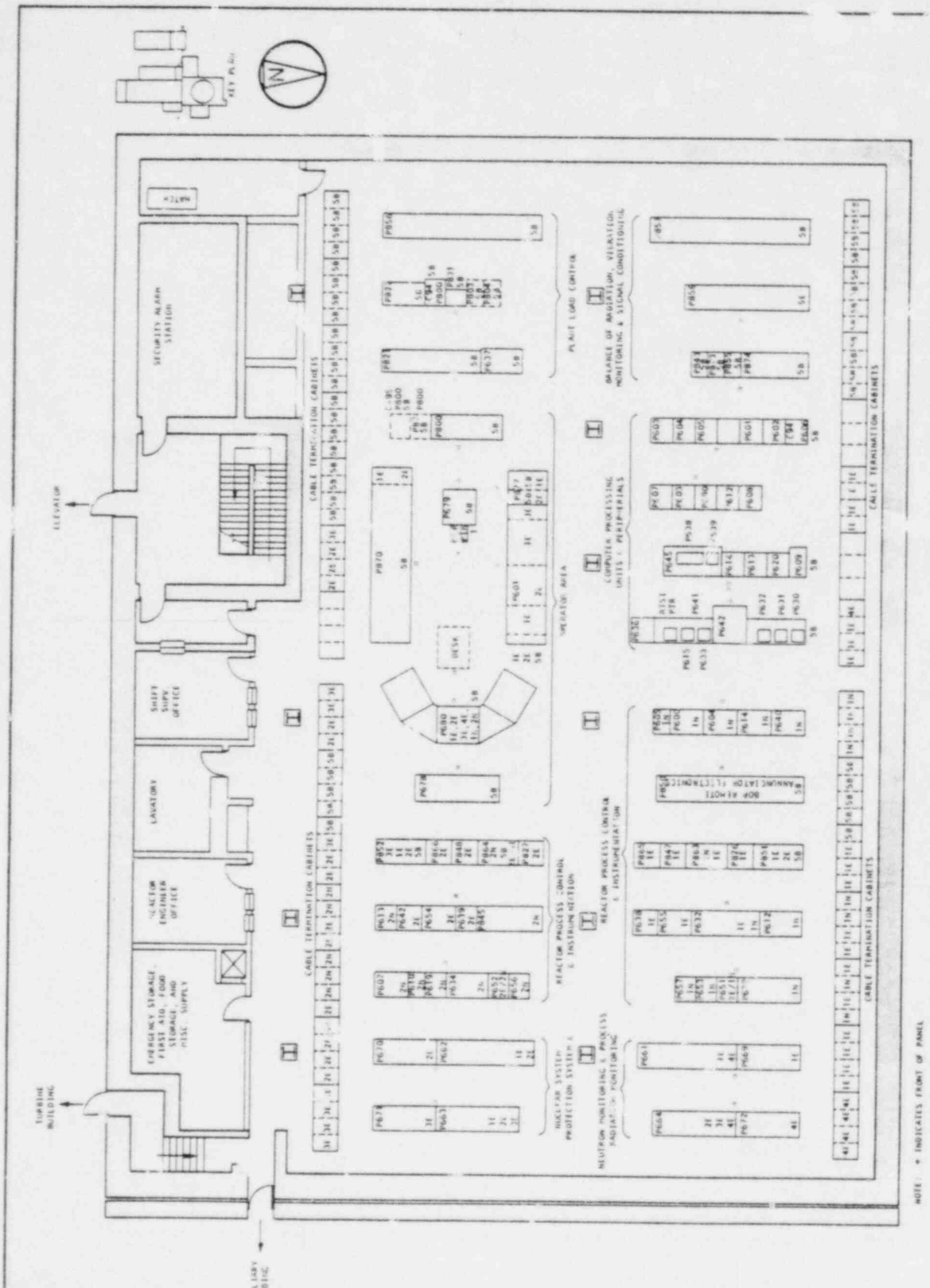


FRONT

PROFILE

BLACK FOX STATION SUPERVISORY
MONITORING CONSOLE (HI3-P679)

FIGURE (2)(iii)-7



BLACK FOX STATION CONTROL ROOM PLAN

FIGURE (2)(iii)-8

10 CFR 50.34(e)(2)(iv) PLANT SAFETY PARAMETER DISPLAY CONSOLE

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(iv) Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded. (NUREG-0718, I.D.2)

PSO RESPONSE:

Introduction

Following the accident at Three Mile Island, Unit 2 (TMI-2), the NRC Staff developed a list of concerns associated with dealing with the lessons learned from the accident. One of the concerns was that the TMI-2 plant did not have a plant safety parameter display to provide sufficient information for the operators to determine the safety status of the plant.

This Requirement is imposed on construction permit applicants to ensure that future plants contain a Safety Parameter Display System (SPDS) to assist them in assessing the safety status of the plant. The Staff published guidance for meeting this Requirement in NUREG-0696, "Functional Criteria for Emergency Response Facilities."

Commitment

PSO recognizes the importance of providing the operators with sufficient information in convenient formats to assist in determining the safety status of the plant. PSO commits to providing a computer based SPDS that will display to the operators a set of parameters for assisting the operators in assessing the safety status of BFS in accordance with NUREG-0696.

System Description

The primary function of the SPDS is to aid the operator in the rapid detection of abnormal operating conditions by providing a continuous indication of plant parameters or derived variables representative of the safety status of the plant. This system is not intended to provide full problem diagnostic capability, but rather serve as an indicator that the plant is either in a safe condition or that an off normal condition exists and further action should be taken to identify and correct it.

PSO RESPONSE: 10 CFR 50.34(e)(2)(iv)

a. Location

The SPDS displays will be available in the control room as well as in the Technical Support Center (TSC) and Emergency Operations Facility (EOF). The displays will be readily accessible and visible to control room operators in the normal operating area but will not interfere with normal movement or with full visual access to other control room systems or displays. The SPDS displays will be readable from the control room senior reactor operator's emergency operating station.

b. Parameters

The set of safety parameters to be displayed will be determined by the ongoing efforts of the BWR Owners' Group when approved by the NRC. The parameters will include, but not be limited to, reactivity control, reactor core cooling, reactor coolant system integrity, containment integrity, and radioactive effluent to the environment. The displayed parameters will be selected to enable the operator to determine if systems are performing their design functions.

The display formats will be designed in accordance with human factors principles. Data trends will be available on demand and the system will be capable of indicating when process limits are being approached or exceeded.

c. Reliability

The SPDS used in the control room will be designed to an operational unavailability goal of .01. The term unavailability is used to express a complete loss of system function.

The primary SPDS information will be provided with a high quality and highly reliable computer system capable of functioning properly in the
control room environment that may be present during transient or 19
accident conditions. Backup display information will be provided in the control room utilizing the normal displays used to comply with Regulatory Guide 1.97.

10 CFR 50.34(e)(2)(v) SAFETY SYSTEM STATUS MONITORING

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(v) Provide for automatic indication of the bypassed and operable status of safety systems. (NUREG-0718, I.D.3)

PSO RESPONSE:

PSO has reassessed safety system status monitoring design criteria for Black Fox Station as described in the PSAR and has concluded that this system adequately addresses the NRC Staff's requirement.

Safety system status monitoring provides the control room operator with a continuous status indication of the operability of reactor safety systems, such as High Pressure Core Spray, Residual Heat Removal, Standby Liquid Control, Reactor Protection and Standby Service Water. This NRC Staff requirement stems from the fact that such a system was not available at TMI-2. However, this requirement has previously been included in the Black Fox Station design prior to the accident at TMI-2 through the application of Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems" which was issued in 1973 and has been applied to both the Nuclear Steam Supply and Balance of Plant Safety Systems.

Table 1.9-1 of the BFS PSAR lists the sections of the PSAR where safety system status monitoring is described. In summary, the design includes automatic indication of the bypassed and operable status of safety systems. To the extent practical, inputs to safety system status monitoring will be direct measurements of the desired variables.

10 CFR 50.34(e)(2)(vi) REACTOR COOLANT SYSTEM VENTS

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(vi) Provide the capability of high point venting of noncondensable gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. (NUREG-0718, II.B.1)

PSO RESPONSE:

Introduction

The NRC Staff is requiring the design of Boiling Water Reactors (BWR's) reactor coolant systems, reactor vessels, and other systems required to maintain core cooling to include the capability of remote venting from the control room. Additionally, the vents shall be safety grade, shall satisfy the single failure criterion, and their operation shall not lead to an unacceptable increase in the probability of a loss-of-coolant accident or an unacceptable challenge to containment integrity. The design shall include an analysis demonstrating that direct venting of noncondensable gases, which may contain high concentrations of hydrogen, does not result in violation of combustible gas concentration limits in containment. Finally, procedural guidelines detailing the operator's use of the vents will be developed prior to operation.

The basis for these requirements stems from the accident at Three Mile Island, Unit 2, (TMI-2) where the collection of noncondensable gases impaired natural circulation cooling capability and also limited reactor coolant pump operational capability because of coolant voids in the system occupied by the gases.

Study

The venting requirement is primarily aimed at the pressurized water reactor (PWR) design which is normally solid (i.e., all water with no vapor present). The generation of noncondensable gases within the PWR results in significant changes in coolant circulation characteristics. The BWR design normally operates with vapor present in the coolant. Historically, the BWR design has included venting capability to handle accumulations of gases in high points of the coolant systems.

Therefore, the main effort to meet this requirement involved a review of the design to again verify that the presence of noncondensable gases within the coolant would not have a deleterious effect on coolant circulation capability and to insure high points within the various coolant loops are adequately vented.

This review was completed generically for all BWR's by the BWR Owners' Group and specifically for Black Fox Station (BFS) by Public Service Company of Oklahoma (PSO).

System Description

The function of the reactor coolant system and the auxiliary and emergency core cooling systems is to remove heat generated by the fission process and the residual decay heat present after the fission process has been terminated. The systems at BFS involved in this cooling process where venting is a concern are the Reactor System, Reactor Recirculation (RR) system, Residual Heat Removal (RHR) system, and the Reactor Core Isolation Cooling (RCIC) system.

The Reactor System (shown in Figure (2)(vi)-1) is comprised of the reactor pressure vessel and internals. The reactor pressure vessel houses the core where the fission process takes place.

Circulation of water in the BFS reactor pressure vessel can be achieved either through natural circulation or by forced circulation of coolant through the core region.

1. Natural Circulation

   The primary natural circulation loop is between the downcomer and the core (see Figure (2)(vi)-1). Due to boiling in the core region, a large difference in densities is available for driving natural circulation flow from the downcomer through the jet pumps and into the shroud region.

   Any noncondensable gases formed in the reactor pressure vessel rise to the top of the vessel by virtue of the same phenomena and by the same route as the steam that is generated in the core. In normal operation, these are swept to the turbine with the main steam. During either normal or emergency reactor shutdown, noncondensable gases are swept with main steam either to the condenser (via the turbine bypass) or to the suppression pool (via the SRV's). The reactor vessel head can also be vented to the drywell remotely from the control room if necessary.

   However, it should be noted that vapor is present in the core during normal operation and natural circulation conditions. Thus, noncondensable gases may change the composition of the vapor but would have an insignificant effect on the circulation itself, since they would rise with the steam to the top of the vessel after leaving the steam separators. Whether or not noncondensable gases are vented from the top of the vessel, the formation of noncondensable gases would not hinder natural circulation during an abnormal event, nor would it result in a blockage condition that could hamper eventual recovery of the core.

The Residual Heat Removal System assists the natural circulation of coolant by acting as a heat sink increasing the driving force of the coolant circulation and by making up coolant inventory. The RHR system (shown in Figure (2)(vi)-2) shutdown cooling mode removes residual heat from the core by circulating hot coolant taken from the suction leg of the reactor recirculation pump (the hot coolant comes from the annulus between the core shroud and the vessel wall) through the RHR pumps and heat exchangers. After the heat is removed via the RHR heat exchangers, the coolant is returned to the vessel through the feedwater lines where natural circulation through the core continues.

The RHR System and the Reactor Core Isolation Cooling (RCIC) System (Figure (2)(vi)-3) work together in the steam condensing mode of RHR to condense steam taken from steam line "A". Steam taken from the main steam line is condensed in the RHR heat exchangers and pumped via the RCIC pump to the Reactor Pressure Vessel head spray nozzle. The condensate makes up the inventory in the annulus for the natural circulation process.

2. Forced Circulation

Forced circulation is achieved through the use of the Reactor Recirculation system. The Reactor Recirculation system (see Figure (2)(vi)-1) consists of two loops external to the reactor pressure vessel, each containing a pump, a flow control valve, and two shutoff valves. The recirculation system utilizes high performance jet pumps within the reactor pressure vessel. The recirculation pumps take suction from the downward flow in the annulus between the core shroud and the vessel wall. Approximately one-third of the core flow is taken from the vessel through the two recirculation nozzles. There, it is pumped at a higher pressure, distributed through a manifold to which a number of riser pipes are connected, and returned to the vessel inlet nozzles. This flow is discharged from the jet pump nozzle into the initial stage of the jet pump throat where, due to a momentum exchange process, it induces surrounding water in the downcomer region to be drawn into the jet pump throat where these two flows mix and then diffuse in the diffuser, to be finally discharged into the lower core plenum.

After exiting the jet pump diffusers, the coolant turns upward, where it flows between the control rod drive guide tubes and enters into the fuel support where the flow is directed to the fuel bundles.

PSO RESPONSE: 10 CFR 50.34(e)(2)(vi)

The coolant water passes along the individual fuel rods inside the fuel channel where it is heated and becomes a two-phase, water-steam mixture. This two-phase flow enters the plenum above the core and then passes through the steam conditioning equipment (steam separator and steam dryer) where the steam is directed to the main steam line nozzles and piped to the turbine and the water is directed back to the annulus between the core shroud and vessel wall.

Revi Design

The BWR Owners' Group, of which PSO is a member, performed a generic review of BWR reactor cooling system designs with respect to the presence of condensable gases. The BWR Owners' Group review concluded that neither natural nor forced circulation were impaired with noncondensable gases present. Further, their study found that the vent locations provided in the BWR design adequately vented the accumulation points for noncondensable gases in the systems.

The PSO review of the specific BFS design confirmed the conclusions of the BWR Owners' Group with respect to BFS. The PSO review verified that ventilation of the reactor vessel and auxiliary and emergency core cooling systems is achievable in the following manners:

1. A normally open 2" reactor vessel head vent continuously vents noncondensable gases in the vessel head to main steam line "A". Normally these gases are carried to the turbine and condenser where they are processed through the Off-Gas System. The noncondensable gases may also be vented to the suppression pool and containment via the main steam Safety Relief Valves (SRV's). The vent, reduced from the 4" tee connection on the vessel head to a 2" line size, is controlled through valve B21-F005. This motor operated, ASME III Safety Class 1, 1500 pound globe valve is powered from a divisional Class 1E 480 volt Motor Control Center. Control for the valve is on the P601 panel in the main control room.

2. Additionally, there is a normally closed 2" reactor vessel head vent which discharges to the drywell equipment drain sump. The vent is isolated by two motor operated, ASME III Safety Class 1, 1500 pound globe valves (B21-F001 and B21-F002). These valves are powered from the same motor control center and control room panel as valve B21-F005.

   All three valves are safety grade and are seismically and environmentally qualified. The operators which are Class 1E are powered from an essential power supply. The valves are operable from the main control room. Due to their size, the vents do not lead to an unacceptable increase in the probability of a loss-of-coolant accident. The vents do not penetrate containment and, therefore, do not challenge containment integrity.

PSO RESPONSE: 10 CFR 50.34(e)(2)(vi)

3. Noncondensable gases below the main steam line nozzles are vented by opening any one of the 19 SRV's on the main steam lines. These power operated relief valves are operable from the control room but additionally provide over-pressure protection for the reactor pressure vessel and as such may open without operator initiation. These valves, their operators, and their instrumentation are Safety Class 1, seismically and environmentally qualified and are operated from the P601 panel in the main control room. In addition, eight of the valves have a backup safety-related air supply, thus providing redundant venting capability. inadvertent opening of a relief valve is a design basis event and a controllable transient and as such does not increase the probability of a loss-of-coolant accident. Since the valves vent within the containment, there is also no challenge to containment integrity.

4. In addition to the path through the SRV's, main steam line "A" can be vented through the Reactor Core Isolation Cooling (RCIC) system. The path is through the RCIC turbine which exhausts to the suppression pool. This method is also operable from the main control room.

5. Noncondensable gases conceivably could come out of solution in the RHR system during operation. These gases are expected to be swept through the system with possible accumulations in the upper regions of the RHR heat exchangers. The RHR heat exchangers have a 1" top vent to the suppression pool for the removal of these noncondensable gases. Each vent has two 1" ASME III Safety Class 2, 1500 pound globe valves with Class 1E electric motor operators which are operable from the main control room. The valves and their operators are seismically and environmentally qualified. The motor operators are powered from an essential power supply. Due to the size of the vents, there is no increase in probability of a LOCA or challenge to containment integrity.

All vent paths lead to the suppression pool and ultimately the containment. Discussion of the hydrogen mixing analysis and control of large volumes of hydrogen mixing analysis and control of large volumes of hydrogen within the containment can be found in the response to Requirement (2)(ix).

Procedures for the use of the various vent paths discussed above will be developed in the future and summarized in the FSAR.

Conclusion

The PSO review of the specific BFS reactor cooling system design confirmed, with respect to BFS, the generic BWR Owners' Group conclusions that:

PSO RESPONSE: 10 CFR 50.34(e)(2)(vi)

1. The presence of noncondensable gases does not impair natural or forced circulation.

2. The locations of reactor coolant system vents in the BWR design provide for adequate venting of accumulation points for noncondensable gases.

The design of the EFS Reactor Coolant System and other systems required to maintain adequate core cooling provides the capability for venting noncondensable gases from high points in the systems. This function is performed by the operator in the control room. The design required to vent these system high points existed prior to the requirements stemming from the accident at TMI-2 and, therefore, do not constitute an increase in the probability of a loss-of-coolant accident or an unacceptable challenge to containment integrity. No additional vent lines are required over those analyzed. There is no new, novel design and there are no concerns regarding technical feasibility, state-of-the-art, or ability to implement the intended venting design. 19

PSO RESPONSE: (2)(vi)

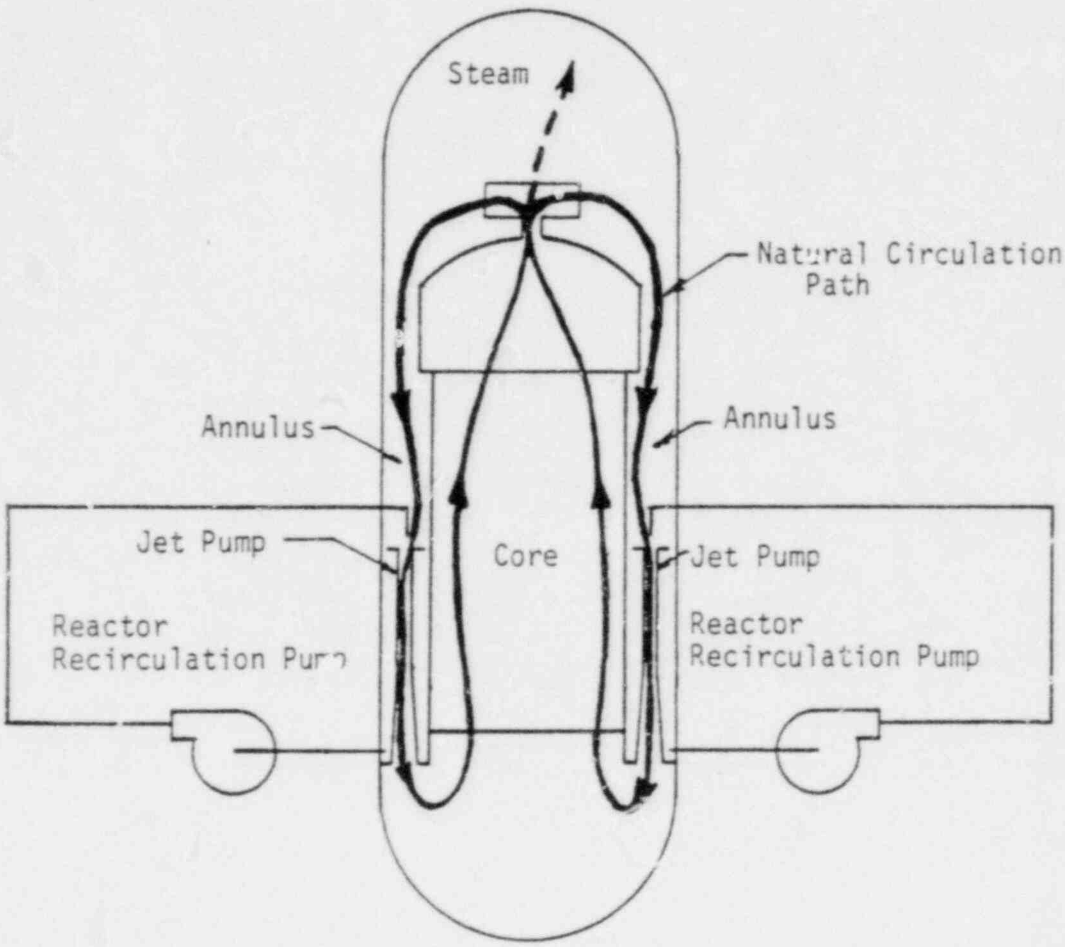


Figure (2) (vi) - 1

Black Fox Station
Reactor System

PSO RESPONSE: (2)(vi)

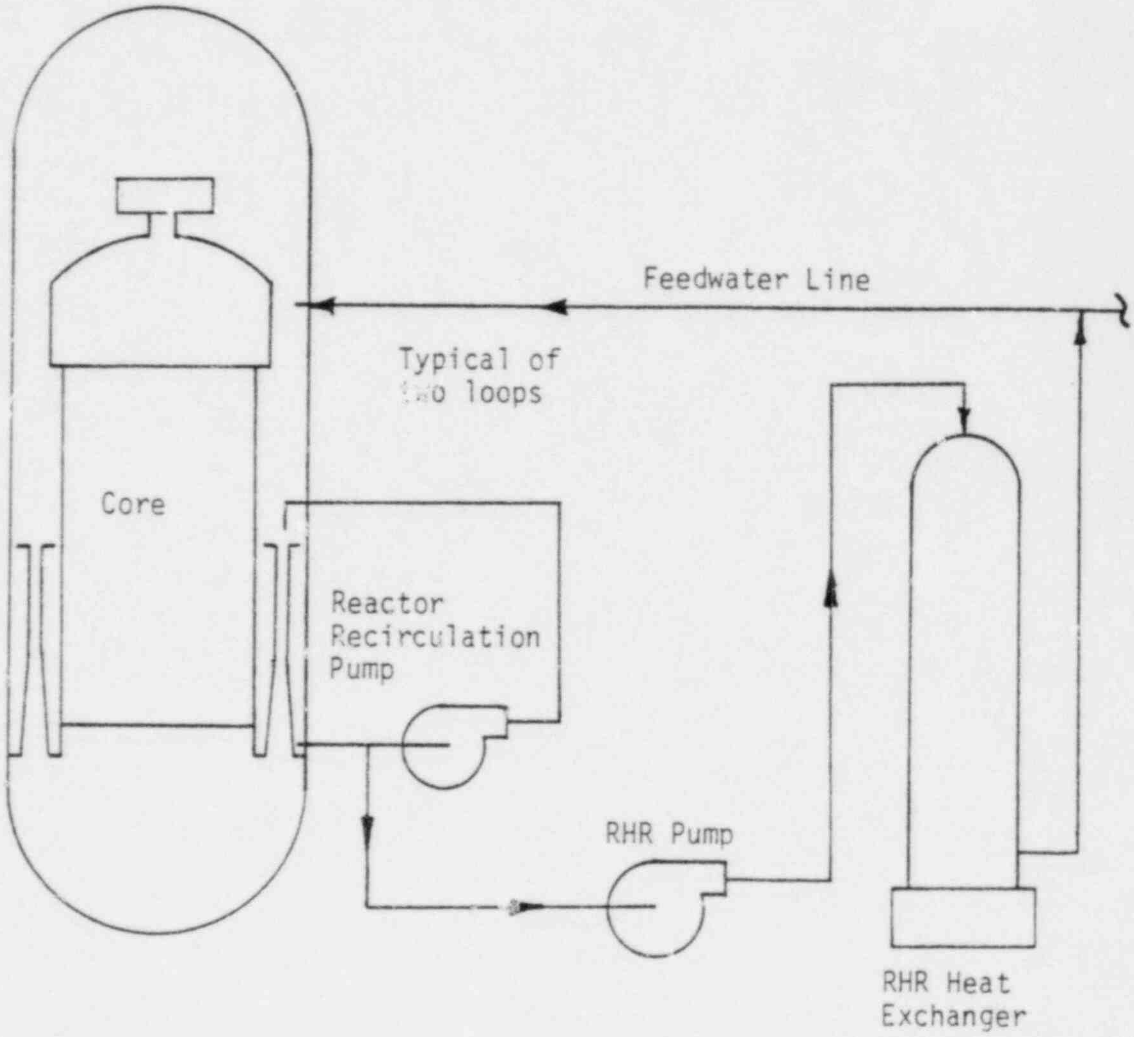


Figure (2) (vi) - 2

Black Fox Station
Residual Heat Removal System
Shutdown Cooling Mode

PSO RESPONSE: (2)(vi)

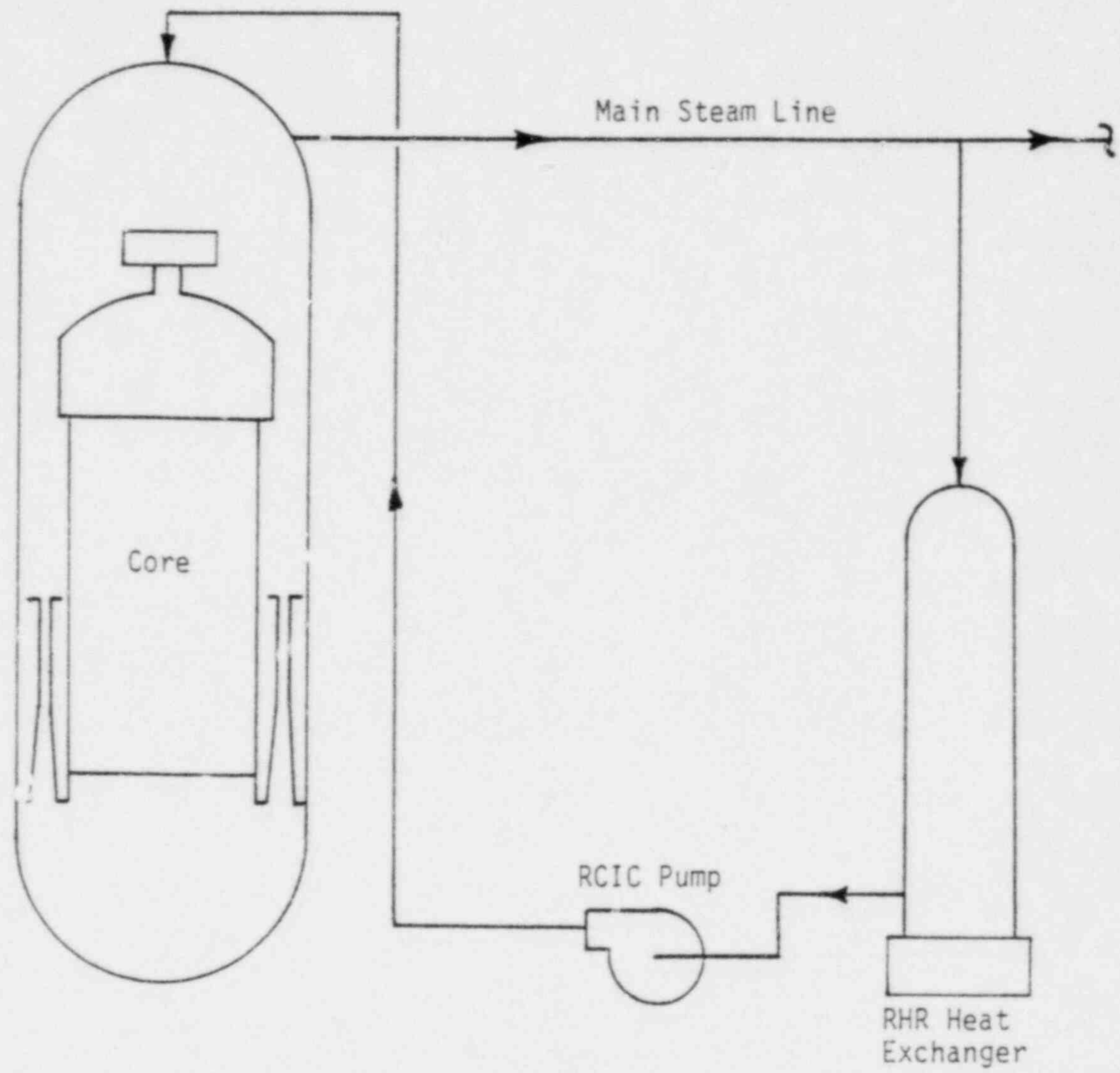


Figure (2) (vi) - 3

Black Fox Station
Residual Heat Removal
System
Steam Condensing Mode

10 CFR 50.34(e)(2)(vii) PLANT SHIELDING TO PROVIDE ACCESS TO VITAL AREAS AND PROTECT SAFETY EQUIPMENT FOR POST-ACCIDENT OPERATION

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(vii) Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain TID 14844 source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. (NUREG-0718, II.B.2)

PSO RESPONSE:

Introduction

One of the consequences of the TMI-2 accident was the release of large amounts of radioactive material to plant systems and rooms which were not specifically designed to contain high levels of radiation. The resulting radiation fields interfered with personnel access required to achieve control of the accident, maintain a safe shutdown condition and perform accident recovery operations. Systems which were not specifically designed to perform post-accident functions were used to mitigate the consequences of the accident. The lack of adequate shielding from accident source terms made maintenance of these systems difficult. Additionally, access was required to important areas such as the radwaste control room, power supplies and instrument racks which were found to be located in high radiation fields. All of these effects contributed to delays to the accident control and recovery operations and in personnel exposures.

The Staff's concern for adequate plant shielding is stated in NUREG-0718, Revision 1, "Licensing Requirements for Pending Applications for construction Permits and Manufacturing License." Construction permit applicants must perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain TID 14844 source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment.

Commitment

PSO commits to perform the required design reviews and to incorporate the results of these reviews into the design of Black Fox Station (BFS). The review process is underway at the present time and is scheduled for completion within six months after the issuance of construction permits for BFS. The purpose of the review is (a) to ensure that vital areas in which personnel will be present during post-accident operations will be accessible; (b) to determine the accessibility of areas where it may be beneficial (although not essential) to have access to support post-accident operations; and (c) to verify the adequacy of protection provided for safety-related equipment.

Should the shielding design review so indicate, design modifications will be implemented as the detailed design progresses to permit adequate post-accident access or to protect safety equipment from the radiation environment. Any required design and/or procedural changes will be made to maintain personnel exposures in vital areas within 10 CFR 50, Appendix A, GDC-19, specified design bases.

Post-Accident Source Terms

TID 14844 source terms will be used in this shielding review.

The source isotopic compositions, as specified in NUREG-0737, Item II.B.2, will be used for pressurized-depressurized reactor coolant and for gas-containing systems. 19

Tabulations of the initial inventory of radioisotopes for these sources are given in Table (2)(vii)-1.

For the calculation of the post-accident radiation source terms, the following assumptions will be employed:

a. No credit will be taken for radioactive decay prior to transport of the source terms to the systems under consideration.

b. A detailed mechanistic approach to develop radiation source terms will not be used.

Systems With Post-Accident Source Terms

The following BFS systems will be assumed to potentially contain high levels of radioactivity in a post-accident situation:

a. Reactor Building (RB)

b. Residual Heat Removal System (RHR) (Suppression Pool Cooling, Containment Spray, Low Pressure Coolant Injection and Shutdown Cooling)

c. High Pressure Core Spray System (HPCS)

d. Low Pressure Core Spray System (LPCS)

PSO RESPONSE: 10 CFR 50.34(e)(2)(vii)

e. Reactor Core Isolation Cooling System (RCIC)

f. Standby Gas Treatment System (SGTS)

g. Reactor Water Cleanup System (RWCS)

h. Main Steam Isolation Valve Leakage Control System (MSIVLC)

i. Containment Atmosphere Monitoring System (CAM)

j. Leak Detection System (LD)

k. Plant Equipment and Floor Drain System (PEFD)

l. Post-Accident Sampling System (PAS)

During the detailed review, all BFS systems will be reexamined to determine whether they might contain accident source terms. If any additional systems are so identified they will be added to the above list.

The following BFS systems will not be assumed to contain accident source terms for the reasons indicated:

a. Hydrogen Recombiner System: BFS utilizes thermal recombiners which are completely internal to the containment.

b. Main Condenser Off Gas and Liquid Radwaste Systems: The radwaste systems outside containment will be isolated from the containment immediately following an accident. 19

Post-Accident Access

All BFS plant areas will be reviewed to determine if they fall into one of the two following categories:

a. Post-Accident Vital Areas: Those areas in which personnel will be present during post-accident operations to perform monitoring and control functions.

b. Potential Post-Accident Support Areas: Those areas other than vital areas in which it is beneficial, although not essential, to have access to support post-accident operations.

The main control room, technical support center, sampling station and sample analysis area are areas where access is considered vital after an accident.

PSO RESPONSE: 10 CFR 50.34(e)(2)(vii)

Personnel Radiation Exposure Guidelines

Personnel radiation exposure doses will be calculated based on calculated dose rates and occupancy assumptions for each area requiring access in a post-accident environment. The calculated exposure doses will be compared to the guidelines contained in 10 CFR 50, Appendix A, General Design Criteria 19 and Standard Review Plan (SRP) Section 6.4.

Occupancy Assumptions

For post-accident vital areas which require continuous occupancy, assumptions will be based on the criteria contained in SRP Section 6.4

For other post-accident vital areas, occupancy assumptions will be determined taking into account the frequency and duration of the activities anticipated for that area.

Furthermore, transit paths and transit time will be included in the review to ensure that radiation doses received in transit are considered in the assessment of personnel doses.

For potential post-accident support areas, occupancy limits will be determined based on the results of the shielding review.

Dose Rates

Average dose rates over the duration of the accident will be used to determine personnel doses. The dose rates will include contributions from containment shine and equipment shine from all significant sources.

Protection of Safety Related Equipment

An analysis for equipment qualification will be performed using the source terms identified in NUREG-0737, Item II.B.2, to establish the 19
integrated dose, including post accident operation, under which safety-related mechanical and electrical equipment located inside and outside containment are required to function. The results of this analysis will be used in the design and specification of this equipment. Design modifications will be implemented where necessary to assure that the safety-related equipment will function when exposed to the radiation fields resulting from systems involved in the mitigation of an accident.

Options for Solving Potential Problems

The shielding analysis will verify the adequacy of the existing BFS design and indicate where changes will need to be made. If changes are required to meet acceptable operator and/or equipment dose levels in certain locations, the following options are available:

PSO RESPONSE: 10 CFR 50.34(e)(2)(vii)

a. Move the offending radiation source to a less sensitive location.

b. Move the target equipment or operator control/work station to a location with an acceptable radiation field.

c. Place additional shielding around the offending radiation source.

d. Place local shielding around the target equipment or operator control/work station.

e. Purchase equipment designed to withstand the newly specified radiation environment.

PSO RESPONSE: (2) (vii)
TABLE (2) (vii)-1. POST-ACCIDENT REACTOR COOLANT SOURCE TERM*

Isotope	Curies	Isotope	Curies	Isotope	Curies	Isotope	Curies	Isotope	Curies
AG108	7.187-03	AG109M	2.78 +05	AG110	5.071+04	AG110M	9.721+02	AG111	5.565+04
AG111M	5.562+04	AG112	2.247+04	AG113M	1.694+04	AG114	1.310+04	AG115	6.890+03
AG115M	2.438+03	AG116	9.545+03	AG117	5.787+03	AS76	6.517+00	AS77	2.439+03
AS78	7.337+03	AS79	1.020+04	AS80	1.843+04	AS81	5.326+04	AS85	7.355+04
BA135M	[illegible].529+01	BA137M	1.196+05	BA139	1.783+06	BA140	1.692+06	BA141	1.736+06
BA142	1.404+06	BA143	[illegible].051+06	BA144	6.406+05	BR80	4.331+01	BR80M	4.295+01
BR82	6.933+04	BR83	5.365+06	BR84	8.382+06	BR84M	7.605+05	BR85	1.162+07
BR87	1.957+07	BR88	2.323+07	BR89	2.031+07	BR90	1.995+07	CD111M	7.498+00
CD113M	9.685-02	CD115	8.736+03	CD115M	6.127+02	CD117	2.935+03	CD117M	2.915+03
CD118	8.769+03	CD119	8.769+03	CD121	9.545+03	CE141	1.765+06	CE142	6.854-10
CE143	1.490+06	CE144	1.258+06	CE145	9.699+05	CE146	7.441+05	CE147	4.835+05
CE148	3.038+05	CS134	6.965+04	CS134M	1.845+04	CS135	2.371-01	CS136	3.209+04
CS137	1.279+05	CS138	1.849+06	CS139	1.781+06	CS140	1.522+06	CS141	1.151+06
CS142	5.855+05	CS143	2.278+05	CS144	6.013+04	DY165	1.248+02	DY165M	1.076+02
DY166	6.693+00	EU154	2.504+03	EU155	6.066+04	EU156	2.045+04	EU157	1.657+04
EU159	3.551+03	EU160	1.168+03	GA72	3.751+00	GA73	6.349+01	GA74	1.641+02
GA75	4.055+02	GA76	9.48[illegible]+02	GD159	4.853+03	GD161	7.244+02	GD162	4.542+02
GE75	4.059+02	GE75M	1.633+01	GE77	1.180+03	GE77M	1.656+03	GE78	7.337+03
HO166	1.913+01	I128	9.975+04	I129	1.833+00	I130	1.547+06	I131	5.258+07

*Values based on 3 years continuous operation at 3,579 MW(t). Release to reactor coolant of 100 per cent of the noble gas inventory, 50 per cent of the halogen inventory, and 1 per cent of the inventory of all others. No credit has been taken for radioactive decay.

PSO RESPONSE: (2) (vii)
TABLE (2) (vii)-1 (Continued). POST-ACCIDENT REACTOR COOLANT SOURCE TERM

Isotope	Curies	Isotope	Curies	Isotope	Curies	Isotope	Curies	Isotope	Curies
I132	7.444+07	I133	8.604+07	I134	1.113+08	I135	8.808+07	I136	3.597+07
I137	4.524+07	I138	3.166+07	I139	3.454+07	IN115	1.688-11	IN115M	8.736+03
IN116	3.715+03	IN116M	2.890+03	IN117	4.295+03	IN117M	2.935+03	IN118	8.769+03
IN119	1.282+03	IN119M	8.110+03	IN120	9.202+03	IN121	4.989+03	IN121M	5.225+03
IN123	2.300+04	KR81	5.494-05	KR83M	1.073+07	KR85	1.073+06	KR85M	2.326+07
KR87	4.134+07	KR88	6.049+07	KR89	7.133+07	KR90	7.308+07	KR91	4.706+07
KR92	2.358+07	KR93	5.841+06	KR94	4.102+06	KR95	8.590+04	KR97	5.136+02
LA140	1.795+06	LA141	1.768+06	LA142	1.553+06	LA143	1.455+06	LA144	1.159+06
MO99	1.807+06	MO101	1.689+06	MO102	1.590+06	MO103	1.317+06	MO104	1.031+06
MO105	6.177+05	NB93M	2.516-01	NB94	2.272-08	NB94M	6.700-04	NB95	1.698+06
NB95M	3.333+04	NB96	2.877+03	NB97	[illegible].745+06	NB97M	1.633+06	NB98	4.875+04
NB98M	1.708+06	NB99	1.751+06	NB100	1.958+06	NB101	1.402+06	ND144	[illegible].388-09
ND147	6.009+05	ND149	3.779+05	ND151	1.865+05	PD107	1.453-01	PD109	2.783+05
PD109M	2.763+02	PD111	5.562+04	PD111M	7.673+02	PD112	2.248+04	PD113	1.673+04
PD114	1.266+04	PD115	8.368+03	PD116	9.373+03	PM147	2.393+05	PM148	[illegible].299+05
PM148M	1.720+04	PM149	4.918+05	PM151	2.025+05	PM152	1.417+05	PM154	5.057+04
PR142	7.559+04	PR143	1.484+06	PR144	1.267+06	PR145	9.824+05	PR146	7.824+05
PR147	5.8[illegible]1+05	PR148	4.832+05	RB86	8.801+02	RB86M	9.628+01	RB87	2.415-05
RB88	6.109+05	RB89	7.817+05	RB90	9.735+05	RB91	9.[illegible]+05	RB92	9.692+05
RB93	8.922+05	RB94	5.605+05	RB95	2.594+04	RB97	9.076+01	RH103M	1.431+06
RH104	5.386+05	RH104M	3.826+04	RH105	1.091+06	RH105M	2.650+05	RH106	7.312+05
RH106M	1.215+05	RH107	7.519+05	RH108	5.683+05	RH109	2.621+05	RU103	1.461+06
RU105	1.262+06	RU106	6.[illegible]+05	RU107	7.301+05	RU108	5.250+05	SN122	1.770+02

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PSO RESPONSE: (2) (vii)
TABLE (2) (vii)-1 (Continued). POST-ACCIDENT REACTOR COOLANT SOURCE TERM

Isotope	Curies	Isotope	Curies	Isotope	Curies	Isotope	Curies	Isotope	Curies
SB122M	2.849+00	SB124	1.059+04	SB124M	9.520+03	SB125	9.044+03	SB126	3.077+03
SB126M	2.249+03	SB127	1.111+05	SB128	4.649+03	SB128M	1.603+05	SB129	2.954+05
SB130	6.904+05	SB131	[illegible].861+05	SB132	9.574+05	SB133	6.818+05	SB134	2.437+05
SB135	1.313+05	SE77M	4.[illegible]70+01	SE79	4.076-01	SE79M	1.021+04	SE81	5.397+04
SE81M	2.141+03	SE83	5.279+04	SE83M	5.404+0[illegible]	SE84	1.640+05	SE85	1.937+05
SE87	2.470+05	SM147	2.556-06	SM148	1.130-09	SM149	6.553-12	SM151	1.363+02
SM153	2.914+05	SM155	5.322+04	SN117M	3.772-02	SN119M	5.526+01	SN121	1.021+04
SN121M	1.301+00	SN123	1.716+03	SN123M	2.330+04	SN125	1.575+04	SN125M	5.784+02
SN126	5.268-01	SN127	[illegible].377+04	SN128	1.550+05	SN130	3.321+05	SN131	2.826+05
SN132	1.370+05	SR89	[illegible].046+05	SR90	8.819+04	SR91	1.026+06	SR92	1.113+06
SR93	1.331+06	SR94	1.281+06	SR95	1.186+06	SR97	1.366+05	TB160	4.427-02
TB161	9.327+02	TB162	5.190+06	TB163	2.163+02	TC99	1.697+01	TC99M	1.590+06
TC100	1.456+05	TC101	1.699+06	TC102	1.642+06	TC103	1.468+06	TC104	1.405+06
TC105	1.180+06	TC107	4.914+05	TC108	2.643+05	TE125M	1.857+03	TE127	1.099+05
TE127M	1.682+04	TE129	2.810+05	TE129M	4.646+04	TE131	8.607+05	TE131M	2.227+05
TE132	1.489+06	TE133	6.356+05	TE133M	1.073+06	TE134	1.694+06	TE135	8.103+05
XE131M	6.299+05	XE133	2.021+08	XE133M	6.993+06	XE135	3.501+07	XE135M	6.077+07
XE137	1.684+08	XE138	1.587+08	XE139	1.414+08	XE140	6.933+07	XE141	2.112+07
XE142	4.728+06	XE143	6.961+05	XE144	7.516+04	Y89M	8.046+01	Y90	9.273+04
Y91M	6.517+05	Y91	1.101+06	Y92	1.167+06	Y93	1.364+06	Y94	1.414+06
Y95	1.626+06	Y96	1.505+06	Y97	1.229+06	ZN72	[illegible].751+00	ZN73	6.349+01
ZR93	3.460+00	ZR95	1.666+06	ZR97	1.652+06	ZR98	1.639+00	ZR99	1.363+06
								TOTAL	2.026+09

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10 CFR 50.34(e)(2)(viii) POST-ACCIDENT SAMPLING

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic issues. (NUREG-0718, Category 4)

(viii) Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID 14844 source term radioactive materials without radiation exposures to any individual exceeding 5 rem to the whole body or 75 rem to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. During a meeting between the NRC Staff and the near-term construction permit applicants on April 8, 1981, the NRC Staff clarified this requirement to indicate that construction permit applicants must commit to meeting the guidelines for post-accident sampling contained in Section II.B.3 of NUREG-0737, "Clarification of TMI Action Plan Requirements." (NUREG-0718, II.B.3)

PSO RESPONSE:

Introduction

Prompt sampling and analysis of reactor coolant and of containment atmosphere can provide information important to the efforts to assess and control the course of an accident. Chemical and radiological analysis of reactor coolant liquid and gas samples can provide substantial information regarding core damage and coolant characteristics. Analysis of containment atmosphere (air) samples can determine if there is any prospect of a hydrogen reaction in containment, as well as provide core damage information. Following an accident, significant amounts of fission products may be present in the reactor coolant and containment air, creating abnormally high radiation levels throughout the facility. These high radiation levels may interfere with timely sampling and analysis activities. In addition, the abnormally high background radiation, high sample radiation, and high levels of airborne contamination may render in-plant radiological spectrum analysis equipment inoperable during and after an accident.

Review of BFS Design

An engineering review of the originally planned BFS sampling and analysis facilities has been performed. The BFS sampling and analysis facilities will be redesigned and will meet the requirements for post-accident sampling and analysis found in Section II.B.3 of NUREG-0737, "Clarification of TMI Action Plan Requirements."

There are no questions regarding the technical feasibility or the state-of-the-art of the required post accident sampling and analysis capability, nor are there any concerns as to the ability to provide appropriate sampling and analysis facilities in the BFS design.

Nature of Review

A summary of the engineering review follows:

1. Sample Station Location

   Since the original sample station was determined to be inaccessible during some accident conditions, a new sample station location was selected using the listed criteria in the following four areas:

   a. Radiation Protection

      Limit radiation exposures to personnel involved in sampling and analysis activities to 5 rem to the whole body or 75 rem to the extremities.

   b. Impact on Present Design

      Location of the sample station will not adversely affect the function of equipment or facilities already located in various BFS buildings.

   c. Accessibility

      Access to the sample station or laboratory facilities will not be prohibited by excessive radiation levels or physical barriers.

   d. Sample Line Length

      Sample line lengths will be minimized to reduce plate out and to reduce the volume of fluids required for purging.

   Eight locations were evaluated using these criteria. A new sample station located in the turbine building on the 621 ft. elevation (see Figure (2)(viii)-1 for the arrangement) was selected as the most appropriate when considering the above criteria.

PSO RESPONSE 10 CFR 50.34(e)(2)(viii)

2. Sample Points

The samples required to provide post accident sampling capability include liquid samples of reactor water, suppression pool water and various containment sumps, and gas samples from the containment atmosphere and Standby Gas Treatment System (SGTS) intakes. The various systems were reviewed to select the most feasible sample point locations for these samples. The sample points selected are shown in Table (2)(viii)-1.

3. Radiation Protection

Preliminary shielding analysis of this sample station location indicates that the existing shielding in this area will maintain the radiation levels from containment shine as well as from equipment/component shine due to systems involved in the mitigation of an accident, assuming TID 14844 source terms, to less than 100 mR/hr. With exposure rates at this level during sampling and with dilution capabilities designed in the sample station to reduce exposure rates during analysis, the exposure limits stated in the above requirement will be met.

4. Sample Analysis

Facilities for the analysis of the post accident samples taken from the sample points listed in Table (2)(viii)-1 will be in the General Services Building. Radiological analyses for certain radionuclides that are indicators of core damage (e.g., noble gases, iodines and cesium and non-volatile isotopes) will be performed in the counting room and chemical analyses (for chloride, hydrogen, dissolved gases, and boron) in the laboratory or with on-line instrumentation. The chemical sample analysis stations are equipped with fume hoods to minimize airborne contamination in the laboratory.

Time for the sample collection and analyses will not exceed the following:

- radiological: three hours
- boron: three hours, if boron injection was initiated
- chlorides: twenty-four hours
- total dissolved gas or hydrogen: three hours
- dissolved oxygen: verification that dissolved oxygen is $<$ 0.1 ppm if chloride concentration exceeds 0.15 ppm

PSO RESPONSE: 10 CFR 50.34(e)(2)(viii)

Accuracy, range and sensitivity of the analyses will be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant system.

Conclusion

The BFS sampling and analysis facilities will be redesigned and will meet the requirements for post-accident sampling and analysis found in Section II.B.3 of NUREG-0737.

TABLE (2)(viii) - 1

SAMPLE POINT LOCATIONS AND ANALYSIS

Sample	Sample Point Description	Radiological	Boron	Chlorides	Total Dissolved Gases or Hydrogen	Dissolved Oxygen	Hydrogen
Reactor Water	Jet Pump 5 lower flow sensing instrument line after drywell penetration.	X	(1)	X	X	(2)	
Reactor Water	Jet Pump 15 lower flow sensing instrument line after drywell penetration.	X	(1)	X	X	(2)	
Reactor or Suppression Pool Water	RHR Heat Exchanger BA001C effluent line	X	(1)	X	X	(2)	
Reactor or Suppression Pool Water	RHR Heat Exchanger BA001D effluent line	X	(1)	X	X	(2)	
Suppression Pool Water	Suppression Pool level instrument line	X	(1)	X			
Containment Sump Water	Containment sump pump discharge header	X					
Auxiliary Bldg. Sump Water	Auxiliary Building floor drain sump pump discharge	X					
Fuel Bldg. Sump Water	Fuel Building floor and equipment drain sump pump discharge	X					
Auxiliary Building Atmosphere	Auxiliary Building exhaust duct	X					
Fuel Building Atmosphere	Fuel Building exhaust duct in SGTS plenum	X					
Annulus Atmosphere	Annulus exhaust duct in SGTS plenum	X					
Annulus Fan Room Duct Atmosphere	Annulus fan room duct in SGTS plenum	X					
Containment Atmosphere	Containment atmosphere samples at two elevations, El 610' and El 634'	X					X

(1) Only if boron injection was initiated.

(2) Verification that dissolved oxygen is < 0.1 ppm if chloride concentration exceeds 0.15 ppm.

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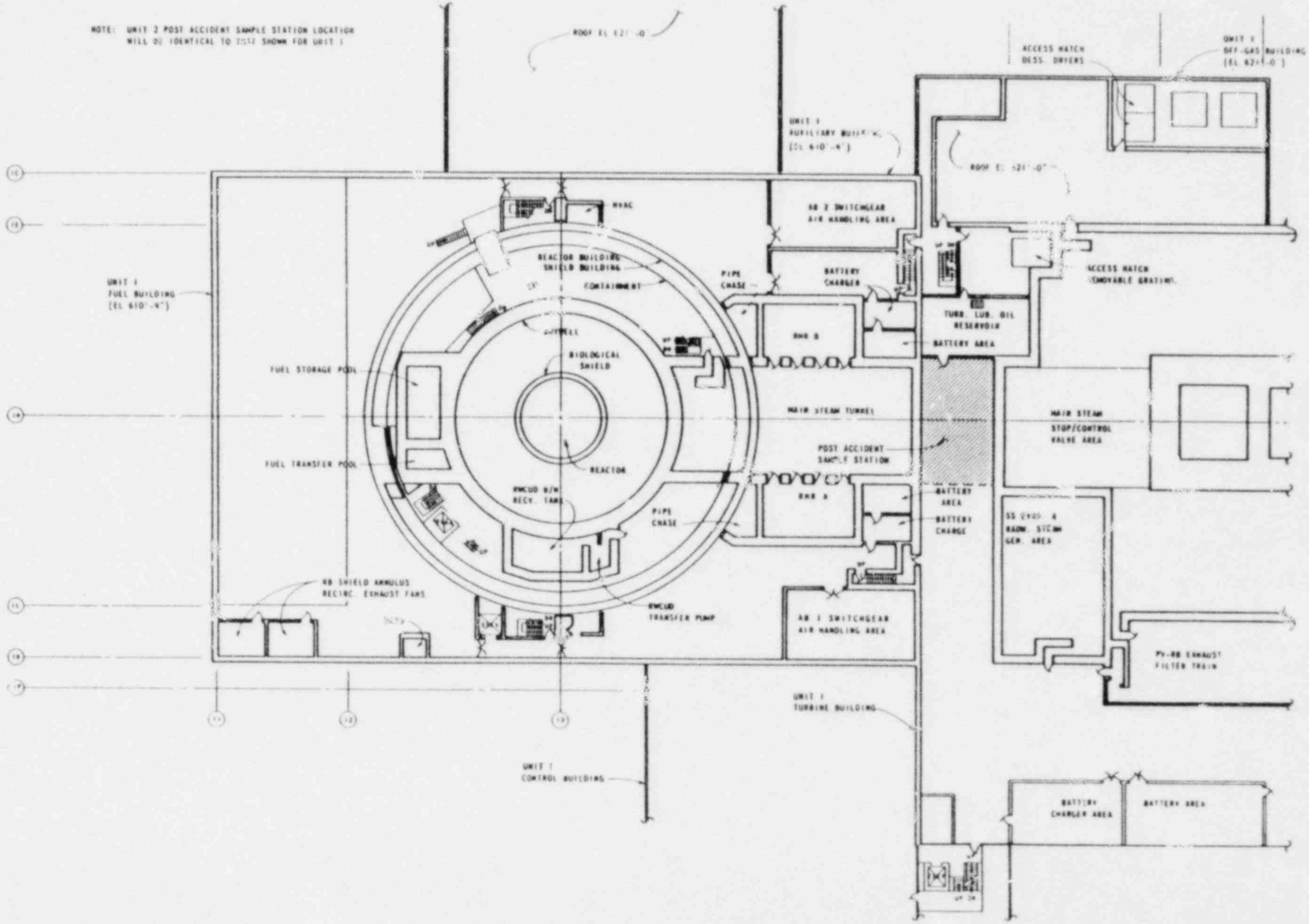


POST ACCIDENT SAMPLE STATION LOCATION
EL. 621'-0"

FIGURE (2)(viii)-1

10 CFR 50.34(e)(2)(x) TESTING REQUIREMENTS

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(x) Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWR's, PORV block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of Anticipated Transients Without Scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed. (NUREG-0718, II.D.1)

PSO RESPONSE:

Introduction

The Three Mile Island-Unit 2 accident sequence included the failure of a power operated relief valve to close. This failure raised a question about the performance qualification of two types of valves in the primary coolant boundary, i.e., safety and relief valves.

The conditions under which relief and safety valves must function and the transients which might affect safety/relief valve operation are different in a boiling water reactor (BWR) like Black Fox as compared to a pressurized water reactor like TMI-2. In a BWR, a stuck-open relief valve, either by itself or in conjunction with a loss of feedwater transient presents no threat to adequate core cooling. It is a design-basis accident and has been analyzed extensively.

Equipment Description

Black Fox Station unit has nineteen dual function Safety/Relief Valves (SRV) located inside the containment on the four main steam lines which transport steam from the reactor vessel to the turbine. The primary purpose of the valves is to prevent damage to the reactor system resulting from excessive water/steam pressure.

The SRV's can be opened in two ways. For the "Safety" function, each SRV is provided with a spring which keeps the valve closed. The spring is adjusted so that a particular pressure in the steam line must be reached before the valve begins to open. The SRV closes when the pressure is reduced below the valve spring "setpoint."

PSO RESPONSE: 10 CFR 50.34(e)(2)(x)

For the "Relief" function, each SRV is equipped with an air operator, the actuation of which opens the valve. Actuation is caused by an electrical signal which has three sources: pressure sensors on the reactor vessel which cause the SRV's to open at pressures slightly lower than those of the "safety" function; a switch in the control room with which the operator can open individual SRV's; and instruments in the Automatic Depressurization System which cause eight SRV's to open if high pressure cooling systems fail to operate when required. Valve closure can occur either by operator manual action or automatically when reactor pressure is reduced.

SRV's for BWR's are designed and qualified for saturated steam flow. However, full water or two-phase (steam/water mixture) flow through the SRV's may occur under some accident conditions, and such flow may increase the dynamic forces on valve internals, piping and supports over those that would be expected from saturated steam flow conditions.

Response to Requirement

The BWR Owners' Group, an organization of utilities operating or building Boiling Water Reactors, developed a program to qualify safety and relief valves in response to the requirements of NUREG-0578. Public Service Company of Oklahoma is a member of this Group.

Nature of Study

The NRC requires that testing be conducted under expected operating conditions for design basis transients and accidents. An evaluation was performed of the events selected from the transients and accidents identified in Regulatory Guide 1.70 Rev. 2, Table 15-1 which have the potential of producing liquid or two-phase flow discharge from the SRV's. The conclusion reached after a detailed review of all identified events is that a test which simulates the alternate shutdown cooling mode should be performed. This is an anticipated operating condition which has been considered in the design and analysis of BWR's. In this event, the SRV is expected to operate with low-pressure (250 psig) water. Two-phase fluid is not expected to flow through the SRV's in this mode of operation.

A test program was developed to qualify the SRV's under the alternate shutdown cooling mode. The test program had two objectives:

1. To demonstrate the capability of each type of SRV used or to be used in BWR's to operate under the bounding cases of low-pressure water with resultant typical pipe loads on the valve.

2. To measure the loads on the valve discharge line during water flow through the SRV's.

PSO RESPONSE: 10 CFR 50.34(e)(2)(x)

The type of SRV purchased for Black Fox Station, the 8 x 10 Dikkers direct-acting valve, was included in the test program. The 8 x 10 Crosby valve used by other BWR-6 plants and which could be used in Black Fox, was also in the test program. Testing was performed, controlled and documented consistent with the requirements of NRC regulations. The tests were performed by Wyle Laboratories in their Huntsville, Alabama facility during the first half of 1981.

The test setup, including the valve, discharge piping and supports was arranged to represent a typical BWR plant and to permit data obtained to be used for analyses of a specific plant. Instrumentation was supplied and located to provide adequate measurement and recording of pipe loads and fluid conditions for proper analyses. An adjustable orifice was installed on the discharge line to vary back pressure at the valve.

The test was designed to simulate, as close as reasonably possible, the conditions for the alternate shutdown cooling mode. After being heated by using 1000 psig saturated steam the valve was commanded to open, reflecting the valve's relief mode of operation. Steam at a pressure of 1000-1200 psig flowed thorugh the valve for approximately five seconds. The valve was then allowed to cool to a temperature of approximately 210° F. Water was then admitted to flow through the valve at pressures up to 250 psig and temperatures slightly lower than saturation. Flow was maintained for approximately five seconds and was regulated by controlling the back pressure through adjustment of the orifice size.

Two tests as described above were made plus another test using water approximately 50° F cooler during the water flow portion of the test. During all portions of the test, data was received from the instruments and recorded. The tested SRV's were given a detailed examination to check for damage. The preliminary conclusion from the results of the test program is that the SRV's qualify for the tested conditions.

A final report will be prepared by the Owners' Group containing all test data and documenting the SRV's successful operation in the alternate shutdown cooling mode. It will also present an analysis of the loads on the valve discharge piping during the test and compare those with the design of the test setup. The Owners' Group report is scheduled to be submitted directly to the NRC Staff in the fourth quarter of 1981.

Conclusion

PSO will review the Owners' Group test report after its acceptance by the NRC Staff. Should the Staff conclude that valves and piping be qualified for operating conditions in addition to that currently defined by the Owners' Group, PSO will participate in any additional testing programs, and document the applicability of the results to Black Fox Station.

10 CFR 50.34(e)(2)(xi) RELIEF AND SAFETY VALVE POSITION INDICATION

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(xi) Provide direct indication of relief and safety valve position (open or closed) in the control room. (NUREG-0718, II.D.3)

PSO RESPONSE:

Introduction

The accident at Three Mile Island-Unit 2 (TMI-2) on March 28, 1979 involved a main feedwater transient coupled with a stuck-open Power-Operated Relief Valve (PORV) and a temporary failure of the auxiliary feedwater system. The PORV position indication in the control room was provided indirectly by the open or close command signals. After opening normally during the initial high pressure transient, the PORV was commanded to close as pressure decreased. The control room indication showed the valve to be closed when it actually remained open. The equivalent system at BFS is the Safety Relief Valve (SRV).

Commitment

PSO recognizes the importance of providing unambiguous indication to the control room operator and commits to provide direct indication of SRV position in the control room with a safety grade valve position detection device and direct indication in the control room.

System Description

The existing SRV position indication operates indicating lights and an alarm signal in the control room. The lights and alarm indicate an SRV open condition. Input to the indicators is provided by the automatic or manual open-close command signals initiated by the operator or by the logic. This system indicates only desired valve position and not actual position.

In addition, valve leakage monitoring is provided to detect a situation in which the actual valve position does not correspond to the command signal. SRV leakage monitoring consists of a thermocouple in the discharge pipe of each SRV. The thermocouple signal is input to a control room back panel recorder with annunciation at the normal operator location. If a temperature sensor exceeds the alarm setpoint, the recorder chart drive will automatically start to record and the annunciator will alert the operator.

PSO RESPONSE: 10 CFR 50.34(e)(2)(xi)

The main function of the monitoring system is to detect a leaking SRV, but it can also detect an inadvertently opened SRV. The high temperature indication, however, is ambiguous, because the temperature ranges for leaking and open valves overlap. The alarm condition does not necessarily indicate whether the valve is open or leaking. This ambiguity misled the operators at TMI-2, who otherwise would have been able to terminate the accident on receipt of an unambiguous correct position indication.

Design Approach for Sensor

PSO proposes to use hermetically sealed limit switches mounted on the valve as the position detection device. This indicator will be safety grade and will be seismically and environmentally qualified. The switch will be powered from a 1E power source. This approach is known to be technically feasible and within the present state-of-the-art.

Control Room Indication

Direct open and closed position indication will be provided to the operator for each SRV on the Emergency Core Cooling benchboard. The direct position indication signal will provide an audible and visual alarm signal to the operator in the control room. Black Fox Station 19
also incorporates the GE Nuclenet 1000 Control Complex which will include an advance design, computer based CRT display system. SRV position signals will be provided as input to this system.

10 CFR 50.34(e)(2)(xii) AUXILIARY FEEDWATER SYSTEM AUTOMATIC INITIATION AND FLOW INDICATION

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(xii) Provide automatic and manual auxiliary feedwater (AFW) system initiation, and provide auxiliary feedwater system flow indication in the control room. (Applicable to PWR's only). (NUREG-0718, II.E.1.2)

PSO RESPONSE:

This requirement is applicable to Construction Permit applications for Pressurized Water Reactors only and hence does not apply to Black Fox Station.

10 CFR 50.34(e)(2)(xiii) RELIABILITY OF POWER SUPPLIES FOR NATURAL CIRCULATION

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(xiii) Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available. (Applicable to PWR's only). (NUREG-0718, II.E.3.1)

PSO RESPONSE:

This requirement is applicable to Construction Permit applications for Pressurized Water Reactors only and hence does not apply to Black Fox Station.

10 CFR 50.34(e)(2)(xiv) ISOLATION DEPENDABILITY

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(xiv) Provide containment isolation systems that:

(A) ensure all non-essential systems are isolated automatically by the containment isolation system,
(B) for each non-essential penetration (except instrument lines) have two isolation barriers in series,
(C) do not result in reopening of the containment isolation valves on resetting of the isolation signal,
(D) utilize a containment setpoint pressure for initiating containment isolation as low as is compatible with normal operation,
(E) include automatic closing on a high radiation signal for all systems that provide a path to the environs. (NUREG-0718, II.E.4.2)

PSO RESPONSE:

Introduction

The NRC Staff evaluation of the containment isolation experience at TMI-2 showed that design features at some other plants may be inadequate in three respects. First, the lack of diverse actuation signals was a contributing factor at TMI-2 in not isolating the containment until after a significant quantity of water had been pumped from the containment sump into the auxiliary building. Second, the sequence of events at TMI-2 illustrated the need for careful reconsideration of the isolation provisions of non-essential systems inside containment. Third, the experience gained at TMI-2 indicates that the resetting of the containment isolation signal in some designs may result in automatic reopening of some containment isolation valves. The NRC Staff's continued evaluation of this experience resulted in the above requirement.

Containment Isolation Description

The primary objective of the BFS containment isolation design basis is to provide protection against releases of radioactive materials to the environment as a result of accidents. This objective is accomplished by automatic isolation of appropriate lines that penetrate the containment vessel. Containment isolation is automatically initiated by diverse signals as illustrated in PSAR Table 6.2.9.

PSO RESPONSE: 10 CFR 50.34(e)(2)(xiv)

The containment isolation signals result in closure of those fluid penetrations that support systems not required for emergency operation. Those fluid penetrations for essential systems have manually initiated isolation valves which may be operated from the control room.

The isolation criteria for BFS isolation valves conforms to the General Design Criteria 54, 55, 56, 57 and NRC Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment." Redundancy and physical separation is included in the electrical and mechanical design to ensure that no single failure prevents containment isolation.

On signals of high drywell pressure, low water level in the reactor vessel or high radiation all isolation valves that are part of systems not required for emergency shutdown of the plant are closed. The same signals will initiate the operation of systems associated with the Emergency Core Cooling Systems (ECCS). The isolation valves which are part of the ECCS may be closed by manual initiation from the control room.

Compliance With Standard Review Plan 6.2.4

The BFS containment isolation design meets the recommendations of Standard Review Plan Section 6.2.4, Rev. 1. The present containment isolation design has been reviewed and accepted by the NRC (Reference: Black Fox Station SER, NUREG-0190, Section 6.2.4).

Identification of Essential and Nonessential Systems

All BFS systems penetrating the containment have been designated to be essential or nonessential systems according to the following definition:

1. Essential

   Essential systems are those critical to the immediate mitigation of any event that results in automatic containment isolation. Essential systems are not automatically isolated by accident signals.

2. Nonessential

   Nonessential systems are those not critical to the mitigation of any event that results in containment isolation. Nonessential systems are automatically isolated by accident signals. After automatic isolation, the operator may choose selectively to reopen the valves as they are needed, while the accident signal is still present. This permits the operator to use all available systems to cope with an accident, while still maintaining the effectiveness of the containment.

PSO RESPONSE: 10 CFR 50.34(e)(2)(xiv)

PSAR Table 6.2.9 lists the systems penetrating the containment, provides the source of the actuation signals and gives a justification for the essential or nonessential designation of each system.

Isolation of Nonessential Systems

All nonessential penetrations meet the requirements of GDC 54, 55, 56, or 57 as clarified by Standard Review Plan 6.2.4, Rev. 1 and Regulatory Guide 1.11. These penetrations are listed in PSAR Table 6.2.9 and the valve arrangement is shown in PSAR Figure 6.2-15. The isolation of nonessential systems is performed automatically by independent signals derived from diverse parameters. Separate switches are provided, whereby the operator may reset the containment inboard or outboard isolation signal with an accident signal present. The operator may then selectively open the individual valves needed to operate available systems to cope with the accident.

Resetting of Isolation Signal

The design of the controls for automatic containment isolation is such that the containment isolation valves will not reopen on reset of the isolation signal. Each valve must be individually opened by deliberate operator action. Therefore, ganged reopening of containment isolation valves will not be utilized.

Containment Pressure Isolation Setpoint

The containment isolation setpoint pressure for BFS is approximately 2 psig (drywell pressure). Under normal operating conditions, fluctuations in the atmospheric pressure as well as heat inputs from such sources as pumps can result in containment pressure increases on the order of 1 psi. Consequently, the isolation setpoint of 2 psig provides a 1 psi margin above the maximum expected operating pressure to allow for instrument error. It reduces the possibility of spurious containment isolation and provides a very sensitive and positive means of detecting and protecting against the consequences of breaks and leaks in the reactor coolant system.

High Radiation Isolation of Open Path Lines

All systems that provide a path from the containment atmosphere to the environs (e.g., the containment purge and vent systems) will close on a safety-grade high radiation signal. The Containment Purge supply and exhaust isolation valves automatically isolate upon detection of high radiation in the purge exhaust duct.

Containment Purge Isolation Valves

The containment purge and vent isolation valves will satisfy the operability criteria of CSB 6-4. See the response to Requirement (2)(xv) for details.

PSO RESPONSE: 10 CFR 50.34(e)(2)(xiv)

Conclusion

The present design of BFS for containment isolation meets the NRC Staff requirements, and hence no modification of design is necessary.

10 CFR 50.34(e)(2)(xv) PURGING

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(xv) Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions. (NUREG-0718, II.E.4.4)

PSO RESPONSE:

Introduction

The accident at Three Mile Island Unit 2 resulted in unanticipated leakage paths for radioactive gases and liquids from the containment to the auxiliary buildings. During the review of this matter the NRC staff became concerned with respect to the adequacy of the purge system and the isolation of that system. The containment purge system for some plant designs may provide an open path to the environs for accident releases prior to containment isolation. This concern is minimal for the Mark III containment due to the "defense in-depth" design of the drywell, suppression pool, and containment. The reactor coolant system piping is enclosed in the drywell which communicates with the containment only through the suppression pool. Releases from the reactor system are subjected to the quenching and scrubbing action of the suppression pool before entering the containment, so the containment purge for Black Fox does not provide an open path for reactor system releases in the same manner as other containment designs may. Even so, the containment purge system has been reviewed for ALARA considerations and isolation reliability.

Minimizing Purging Time

The requirement for minimizing containment purging/venting is primarily applicable to plants with containments that are not designed for continuous occupancy during normal operation. In these plants purging is only necessary to reduce airborne radioactivity levels consistent with maintaining occupational radiation exposures ALARA prior to containment entry. The Black Fox Station containment is designed for continuous occupancy during operation to facilitate plant operations and maintenance. This led to a PSO decision to continuously purge the containment during operation in order to maintain occupational exposures ALARA.

PSC RESPONSE: 10 CFR 50.34(e)(2)(xv)

The containment purge for BFS is accomplished by the containment Heating, Ventilating and Air Conditioning (HVAC) system. The containment HVAC system consists of two supply Air Handling Units (AHU), six recirculating AHU's, two exhaust fans, and two dome recirculating fans (refer to Figure (2)(xv)-1). The objective of the containment HVAC system is to provide an environment with a level of air quality (temperature, humidity, radioactivity, etc.) that will ensure personnel comfort, health, and safety and efficient equipment operation, while maintaining the exposure of personnel in the containment ALARA.

During normal reactor operation, the containment is continuously purged by the operation of a containment supply AHU and a containment exhaust fan consistent with ALARA considerations. The AHU supplies outside air that has been filtered and tempered to the containment. The exhaust fan exhausts the purged air to the Plant Exhaust Vent System charcoal filter (see Figure (2)(xv)-2). This facilitates maintaining off-site doses ALARA.

Isolation Valve Performance

Isolation provisions for containment HVAC system penetrations consist of three 18-inch diameter air operated butterfly valves in series on each penetration. The isolation valves will be the quick closure type capable of full closure in 5 seconds at the pressure, temperature, and flow rate existing at the time of the accident. The valves will be capable of closing at the containment design pressure, but the actual pressure at the time of isolation will be much less. The isolation valves are Safety Class 2 and Quality Group B and will be designed in accordance with Seismic Category I criteria.

The HVAC containment isolation valves will be independently actuated from two separate divisions of standby power. The isolation valves are designed to fail closed on loss of power. No single failure of the safety-related actuating systems will preclude at least two of the three series isolation valves from closing.

The isolation valves will automatically close within 5 seconds in response to any one of the following conditions:

(1) High drywell pressure

(2) Low reactor water level

(3) High radiation level in the containment exhaust air flow

The signals for conditions (1) and (2) are provided by the Nuclear Systems Protection System. The signal for condition 3 is provided by four redundant, Class 1E monitors of the Process Radiation Monitoring System located in the exhaust duct upstream of the inboard isolation valve.

PSO RESPONSE: 10 CFR 50.34(e)(2)(xv)

Isolation Valve Operability

The isolation valves for the containment HVAC system will meet the interim NRC guidelines on valve operability (see Table (2)(xv)-1) under accident conditions as certified in PSAR Section 9.4.5.4 Inspection and Testing Requirements, summarized below.

The ability of the HVAC duct containment isolation valves to meet the valve isolation requirements will be demonstrated prior to delivery to the station by any one of the following methods:

(1) Each HVAC duct containment isolation valve will be tested in the manufacturer's shop with test conditions imposed during the demonstration of the isolation valve closing equivalent to the combined conditions which the valve is expected to withstand when the isolation function is required.

(2) If the tests described in Item (1) above are not feasible or practical, each HVAC duct containment isolation valve will be tested in the manufacturer's shop under conservative conditions which separately simulate each of the loadings which the valve is expected to withstand in combination during the isolation valve closing function. This testing program will be supplemented by analyses which demonstrate that the individual test loadings are sufficiently higher than the anticipated conditions in combination to ensure there are adequate margins for assurance of operability under the combined loading conditions.

(3) If the tests described in Item (2) above are not feasible or practical because of the manufacturer's test facility limitations, the tests described in Item (2) will be performed at various locations such as the manufacturer's plant, an independent testing facility, or at the site following installation. This method of testing will be verified by analyses in the same manner as described in Item (2) above.

(4) If the testing procedures described in Item (3) above are not feasible or practical, containment isolation valves that can be demonstrated to be equivalent to a prototype isolation valve, which have successfully met the test requirements of a valve operability assurance program, will not be tested if the loading conditions for these valves are equivalent or less severe to those imposed during testing of the prototype valve. The test results of the prototype valve will be documented. The prototype valve may be selected from a group of similar valves which will be used in the unit. A prototype valve used in one nuclear power plant will be deemed to qualify as a prototype valve for the other plant

provided the system operating conditions of both plants and the valve loading conditions at the time when the isolating function is required are equivalent or the operating and loading conditions in the proposed plant are less severe than the conditions in the plant for which the prototype valve was initially used. Valve suppliers will be required to demonstrate by analysis, testing, or combination of testing and analysis that valves which may be open to the containment following a pipe break accident will function under the dynamic loadings associated with the accident. Analytical and test results will be provided by the valve supplier in the stress reports and test reports for the respective valve.

The HVAC containment isolation valves will be analyzed using the most stringent combinations of loads resulting from the OBE or SSE. This is considered to be the normal condition for these valves. Therefore, deformation and damage will not occur and the valves will perform their intended function.

In addition to these tests and analyses, the HVAC duct containment isolation valves will be tested for verification of operability during a simulated seismic event (SSE) by demonstrating operational capabilities within the specified limits.

The seismic test proposed is described as follows. The valve will be mounted in a manner which will be conservatively representative of the plant installation. They will be installed including the actuator and all appurtenances normally attached to the valve in service. The operability of the valve during the SSE shall be demonstrated by satisfying the following criteria.

(1) The valves will be designed to have a fundamental frequency which is greater than 33 Hz. This will be shown by test or analysis.

(2) The actuator and yoke of the valve system will be statically loaded to a load greater than that determined by analysis as representing SSE acceleration applied at the center of gravity of the actuator alone in the direction of the weakest axis of the yoke. The simulated operational differential pressure will be simulataneously applied to the valve during the static deflection tests.

(3) The valve will then be operated while in the deflected position to perform its safety-related function within the specified operating time limits.

(4) Motor operators and other electrical appurtenances necessary for operation will be qualified as operable during the SSE by the appropriate seismic qualification standards described in IEEE-344-1975.

PSO RESPONSE: 10 CFR 50.34(e)(2)(xv)

If the fundamental frequency of the valve, by test or analysis, is less than 33 Hz, a dynamic analysis of the valve will be performed to determine the equivalent acceleration which will be applied during the static test. The analysis will provide the amplification of the input acceleration considering the natural frequency of the valve and the applicable floor response spectra to determine a conservatively adjusting acceleration. The adjust acceleration will then be used in the static analysis and the valve operability will be assumed by the methods outlined in steps (2) through (4) above. As an alternate, the valve including the actuator and other accessories may be qualified by a shake table test.

Using the methods described above, the HVAC containment isolation valves will be qualified for operability during a seismic event to ensure that they will perform their isolation functions where necessary.

Prior to installation, in addition to the seismic tests, the following tests will be performed on the HVAC containment isolation valves.

(1) Shell hydrostatic test to ASME III requirements

(2) Seat leakage tests

(3) Disc hydrostatic test

(4) Functional tests to verify that the isolation valves will close within the specified time limit when subject to the design differential pressure and other loading and environmental conditions. Alternately, this testing requirement may be performed as part of the plant pre-operational test program. Valve actuators will be qualified in accordance with IEEE-382-1972, as modified by Regulatory Guide 1.73 (valves inside containment only).

Conclusion

Since the BFS containment is designed for continuous occupancy, the containment purging system for BFS functions continuously during normal operation in a manner consistent with maintaining occupational exposures ALARA. Hence, the requirement for minimizing purging time is not a concern. In addition, as the foregoing discussion shows, the containment purge for BFS is designed and will be tested to demonstrate reliable isolation under accident conditions.

TABLE (2)(xv)-1

Guidelines for Demonstration
of Operability of Purge and
Vent Valves

Operability

In order to establish operability, it must be shown that the valve actuator's torque capability has sufficient margin to overcome or resist the torques and/or forces (i.e., fluid dynamic, bearing, seating, friction) that resist closure when stroking from the initial open position to full seated (bubble tight) in the time limit specified. This should be predicted on the pressure(s) established in the containment following a design basis LOCA. Considerations which should be addressed in assuring valve design adequacy include:

1. Valve closure rate versus time - i.e., constant rate or other.
2. Flow direction through valve; across valve.
3. Single valve closure (inside containment or outside containment valve) or simultaneous closure. Establish worst case.
4. Containment back pressure effect on closing torque margins of air operated valve which vent pilot air inside containment.
5. Adequacy of accumulator (when used) sizing and initial charge for valve closure requirements.
6. For valve operators using torque limiting devices - are the settings of the devices compatible with the torques required to operate the valve during the design basis condition.
7. The effect of the piping system (turns, branches) upstream and downstream of all valve installations.
8. The effect of butterfly valve disc and shaft orientation to the fluid mixture egressing from the containment.

Demonstration

Demonstration of the various aspects of operability of purge and vent valves may be by analysis, bench testing, in-situ testing or a combination of these means.

Purge and vent valve structural elements (valve/actuator assembly) must be evaluated to have sufficient stress margins to withstand loads imposed while valve closes during a design basis accident. Torsional shear, shear, bending, tension and compression loads/stresses should be considered. Seismic loading should be addressed.

Once valve closure and structural integrity are assured by analysis, testing or a suitable combination, a determination of the sealing integrity after closure and long term exposure to the containment environment should be evaluated. Emphasis should be directed at the effect of radiation and of the containment spray chemical solutions on seal material. Other aspects such as the effect on sealing from outside ambient temperatures and debris should be considered.

The following considerations apply when testing is chosen as a means for demonstrating valve operability:

Bench Testing

A. Bench testing can be used to demonstrate suitability of the in-service valve by reason of its tracability in design to a test valve. The following factors should be considered when qualifying valves through bench testing.

   1. Whether a valve was qualified by testing of an identical valve assembly or by extrapolation of data from a similarly designed valve.

   2. Whether measures were taken to assure that piping upstream and downstream and valve orientation are simulated.

   3. Whether the following load and environmental factors were considered:

      a. Simulation of LOCA
      b. Seismic loading
      c. Temperature soak
      d. Radiation exposure
      e. Chemical exposure
      f. Debris

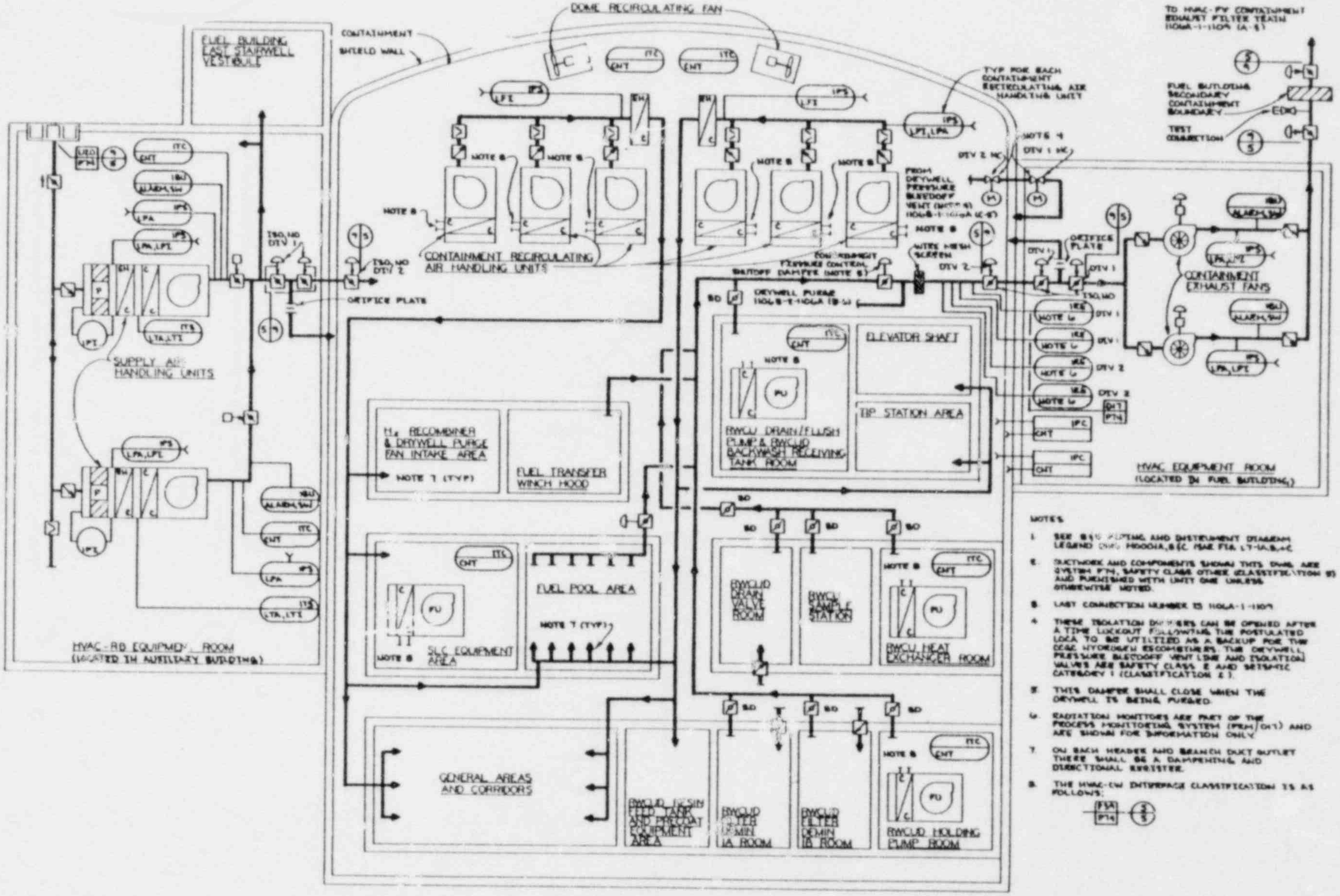
B. Bench testing of installed valves to demonstrate the suitability of the specific valve to perform its required function during the postulated design basis accident is acceptable.

   1. The factors listed in items A.2 and A.3 should be considered when taking this approach.

In-Situ Testing

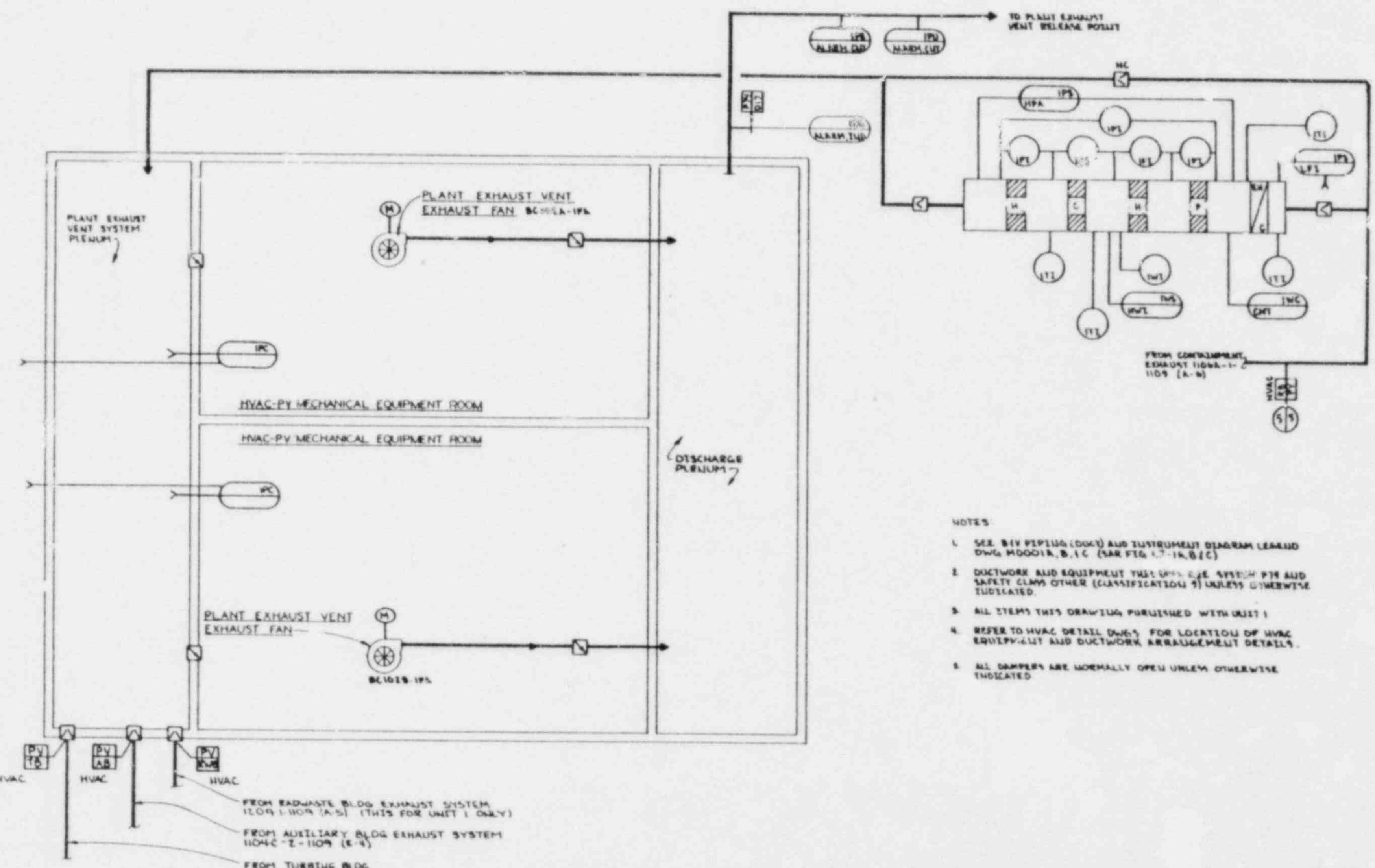
In-situ testing of purge and vent valves may be performed to confirm the suitability of the valve under actual conditions. When performing such tests, the conditions (loading, environment) to which the valve(s) will be subjected during the test should simulate the design basis accident.

NOTE: Post test valve examination should be performed to establish structural integrity of the key valve/actuator components.



CONTAINMENT HVAC

FIGURE (2)(XV)-1



PLANT EXHAUST VENT SYSTEM

FIGURE (2)(XV)-2

10 CFR 50.34(e)(2)(xvi) DESIGN EVALUATION

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(xvi) Establish a design criterion for the allowable number of actuation cycles of the emergency core cooling system and reactor protection system consistent with the expected occurrence rates of severe overcooling events (considering both anticipated transients and accidents). (Applicable to B&W designs only). (NUREG-0718, II.E.5.1)

PSO RESPONSE:

This requirement is applicable to Construction Permit applications for Babcock & Wilcox designed Pressurized Water Reactors only and hence does not apply to Black Fox Station.

10 CFR 50.34(e)(2)(xvii) ADDITIONAL ACCIDENT MONITORING INSTRUMENTATION

NRC POSITION:

(2) To satisfy the following requirement, the applicant shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(xvii) Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level) and (E) noble gas effluents at all potential accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples. (NUREG-0718, II.F.1)

PSO RESPONSE:

Introduction

The NRC Staff, as a part of its continuing program to provide regulatory guidance to licensees and license applicants, has identified the need for instrumentation to aid in accident diagnosis and control. Information regarding a minimum group of plant variables is required by the operators to (1) take preplanned manual action to accomplish safe reactor shutdown, (2) determine whether the reactor trip, engineered safety-feature systems, and manually initiated safety systems and other systems important to safety are performing their intended functions (i.e., reactivity control, core cooling, maintaining reactor coolant system integrity, and maintaining containment integrity), (3) determine the potential for causing a gross breach of the barriers to radioactivity release (i.e., fuel cladding, reactor coolant pressure boundary, and containment) or determine if such a breach has occurred. Instruments designed for monitoring normal operations may be insufficient for monitoring accident extremes. Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant & Environs Conditions During and Following an Accident," has been revised by the NRC to provide guidance in the light of the TMI-2 accident. Among the variables to be monitored for a BWR as described in Regulatory Guide 1.97 are (1) containment pressure, (2) suppression pool water level, (3) containment hydrogen concentration, (4) containment radiation intensity (high level) and (5) noble gas effluents from all potential accident release points. NUREG-0718, Revision 1, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License," states the requirements for these five monitors for plants awaiting construction permits shall be the same as the requirements for plants now operating and plants awaiting operating licenses as set forth in NUREG-0737, "Clarification of TMI Action Plan Requirements." Public Service Company of Oklahoma (PSO) intends to provide these five monitors in compliance with Regulatory Guide 1.97, Revision 2 and in so doing will also be in compliance with NUREG-0737.

Instrumentation Description

All the instrumentation will be capable of functioning to the extent required during and following an accident and all will have ranges sufficiently large so as to be able to measure the accident extremes. All instruments described below are within the state-of-the-art and will be redundant, safety grade, seismically and environmentally qualified for accident conditions including the span of its own measured parameter range and powered from the onsite electrical system.

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 will be performed taking into consideration (a) the use of this information by an operator during normal and abnormal plant conditions, (b) the integration of this instrumentation into emergency procedures, (c) the integration of this instrumentation into operator training, and (d) other alarms occurring during an emergency and the need for establishing the priority among alarms.

1. Containment Pressure Monitors

   Two redundant channels measuring and recording containment pressure are provided. Containment pressure is taken with respect to the shield annulus so that the real impact of pressure transients on the containment structure can be ascertained. The range of the transmitters and recorders shall be at least -5 to 60 psid. The scale maximum represents four times the design pressure of the containment building. The transmitters may be located outside of the shield building in order to make these instruments accessible during post-accident operation and to facilitate access for maintenance. The recorders are located in the main control room as shown in Figure (2)(xvii)-1.

   Containment high pressure alarms and annunciator windows are located in the control room as shown in Figure (2)(xvii)-1. These alarms, located in the normal operating area of the control room are sufficient to alert the operators to abnormal pressure conditions within the containment. Inputs from these instruments will be provided for the operator CRT display.

2. Suppression Pool Level Monitoring

Suppression pool level is monitored as part of the Suppression Pool System. Two pairs of redundant level transmitters measure suppression pool level for control room indication, recording and alarming. Ranges of these instruments will be specified from 1.5 feet above the bottom of the pool (which is below the ECCS suction line) to 30 feet, which is about 4 feet above the top of the weir wall. Normal pool level is 20.5 feet.

Suppression pool transmitters are located outside of the shield building for ease of maintenance and accessibility during post-accident situations. The recorders are located as shown on Figure (2)(xvii)-1.

3. Containment Hydrogen Monitors

Two redundant channels are provided for hydrogen monitoring. By utilizing solenoid valves, each monitor is capable of obtaining a sample from either the drywell or containment atmospheres. The monitors are mounted in the containment, approximately 180° apart, close to their respective drywell penetrations. Hydrogen concentration measurements are made locally, and an electrical signal proportional to concentration is transmitted to strip chart recorders in the main control room. The sample is then returned to its place of origin.

The hydrogen monitoring equipment is automatically activated upon LOCA signal. Those components located in the drywell and containment have been designed to withstand the post LOCA environment for 100 days. Ranges for the monitors will be from 0 to 30% by volume.

4. Containment Radiation Intensity (High Level)

The BFS drywell and containment high level ($1-10^7$ R/hr gamma)
radiation monitors will be located in a manner to provide a
reasonable assessment of area radiation conditions in the drywell
and containment. Four high level monitors will be provided, two in
the drywell and two outside the drywell in the containment. These 19
monitors will be widely separated to provide independent monitoring
of a large portion of the containment and drywell volume.

These monitors will be designed and calibrated to meet the energy
response, redundancy, qualification and recommendations of 19
NUREG-0737, Item II.F.1, Attachment 3.

5. Noble Gas Effluent Monitors

The plant exhaust vent and the standby gas treatment system vent are gaseous release points for the BFS units. These vents will be monitored by multiple channels in order to detect noble gas concentrations in the range from 10^{-6} μCi/cc to 10^{4} μCi/cc.

The requirement for sampling of plant effluents is not monitoring instrumentation per se, but is rather a sample collection and analysis capability. This will be provided in the manner specified in NUREG-0737, as described below:

Sample collection: The release points with high range noble gas effluent monitors will also have particulate and iodine sampling capability. Iodine samples will be taken with a charcoal or silver-zeolite cartridge and particulate samples with a filter. The post-accident iodine and particulate samples are extracted from the release point via the same sample line as the monitoring line.

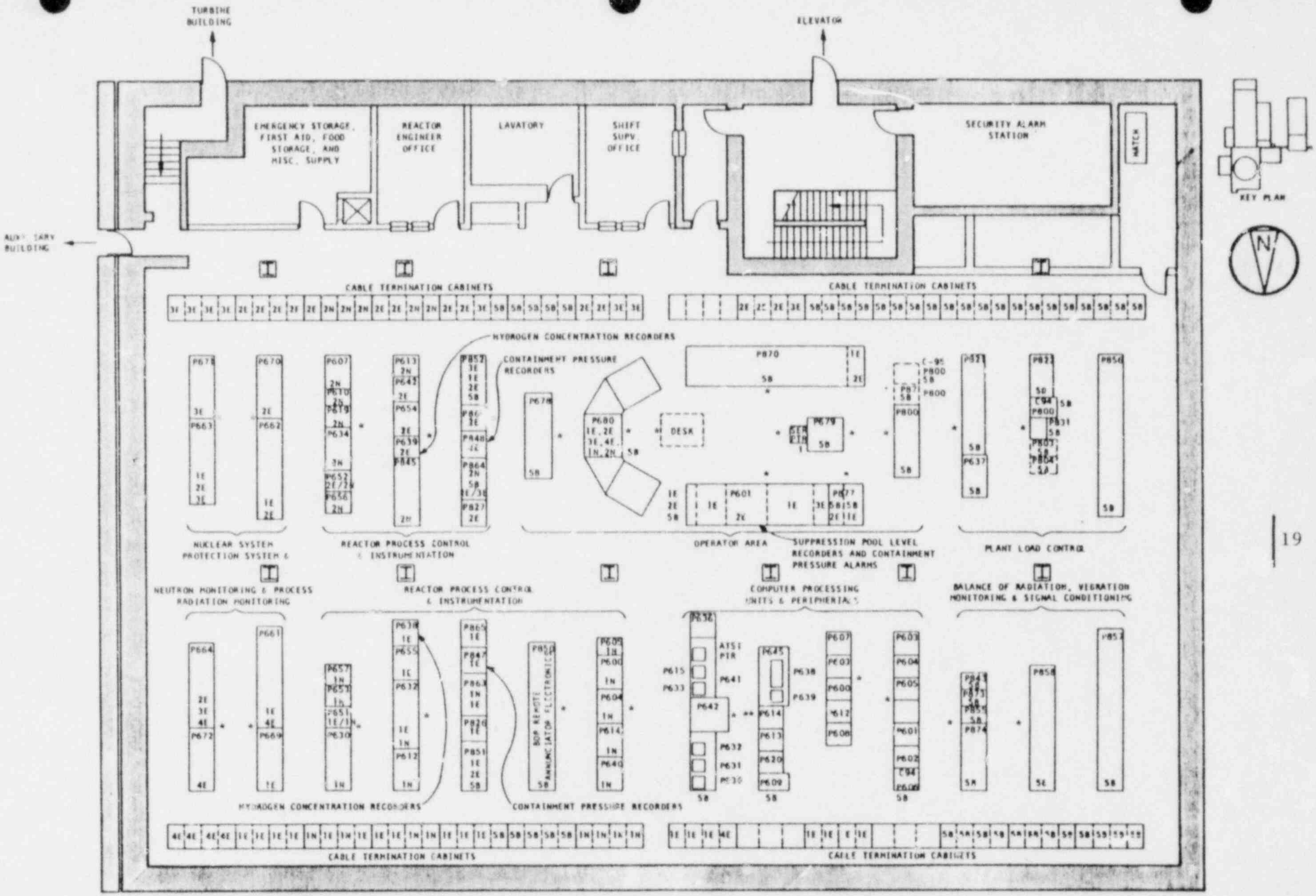
Sample transport: The sample cartridges will be placed in a portable shielded cask and taken to the counting room.

Sample analysis: Capability for the analysis of sample cartridges will be provided. Design of the counting facility will consider the design basis sample.

The precise location of the sample collection station will be selected upon completion of the post-accident shielding study (Requirement (2)(vii)), and the location will assure that a worker involved in the sample collection and transport operation will not receive an exposure greater than 5 rem to the whole body and 75 rem to the extremities.

Conclusion

The BFS design will provide for instrumentation to monitor (1) containment pressure, (2) suppression pool water level, (3) containment hydrogen concentration, (4) containment radiation intensity (high level), and (5) noble gas effluents from all potential accident release points. PSO will provide these five monitors in compliance with Regulatory Guide 1.97, Revision 2 and in so doing will also meet the guidance provided in NUREG-0737.



NOTE: * INDICATES FRONT OF PANEL

19

BLACK FOX STATION CONTROL ROOM RECORDER LOCATIONS

FIGURE (2) (xvii)-1

10 CFR 50.34(e)(2)(xviii) IDENTIFICATION OF AND RECOVERY FROM CONDITIONS LEADING TO INADEQUATE CORE COOLING

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(xviii) Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWR's, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWR's and BWR's. (NUREG-0718, II.F.2)

PSO RESPONSE:

Introduction

During the Three Mile Island accident, a condition of low water level in the reactor vessel and inadequate core cooling apparently existed and was not recognized for a period of time. The NRC Staff has concluded that the problem was the result of a combination of factors including an insufficient range of existing instrumentation, inadequate emergency procedures, human factor problems with control room indicator locations and the absence of water level measuring instrumentation on the reactor vessel of pressurized water reactors (PWR). This problem does not exist in BWR's for the following reasons.

Water Level Measurement

An observable water level during operation is an inherent feature of the BWR concept and monitoring this water level provides a direct indication of the status of core cooling. Principles of BWR operation rely on a high quality water level instrumentation system that display the reliable information to the reactor operator. Such a system is included in the Black Fox Station (BFS) design as described below.

Vessel water level is measured by differential pressure transmitters which measure the difference in static head between columns of water. One column is the constant "reference leg" outside the reactor vessel while the other is the "variable leg" of reactor water level inside the reactor vessel. The BFS station design uses 30 differential pressure transmitters to provide level signals to automatic safety systems, 6 front panel indicators, 5 front panel recorders and the 10 front panel CRT's. Figure (2)(xviii)-1 illustrates the overlapping ranges that measure water level from below the bottom of the active fuel to the top of the vessel.

PSO RESPONSE: 10 CFR 50.34(e)(2)(xviii)

Multiple and redundant channels of hardwired water level indicators and trend recorders are provided in the control room for the operator in addition to the information on 10 CRT's. The BFS design also incorporates visual and audible alarm systems to alert the operator and provide advance warning of water level perturbations that might lead to inadequate core cooling. Figure (2)(xviii)-2 shows the location of the CRT's and readout instrumentation on the control room panels.

The reactor water level measurement techniques provided on the BWR-6 with the Nuclenet 1000* control room will perform satisfactorily for all modes of normal operations, anticipated transient conditions, and credible accident conditions.

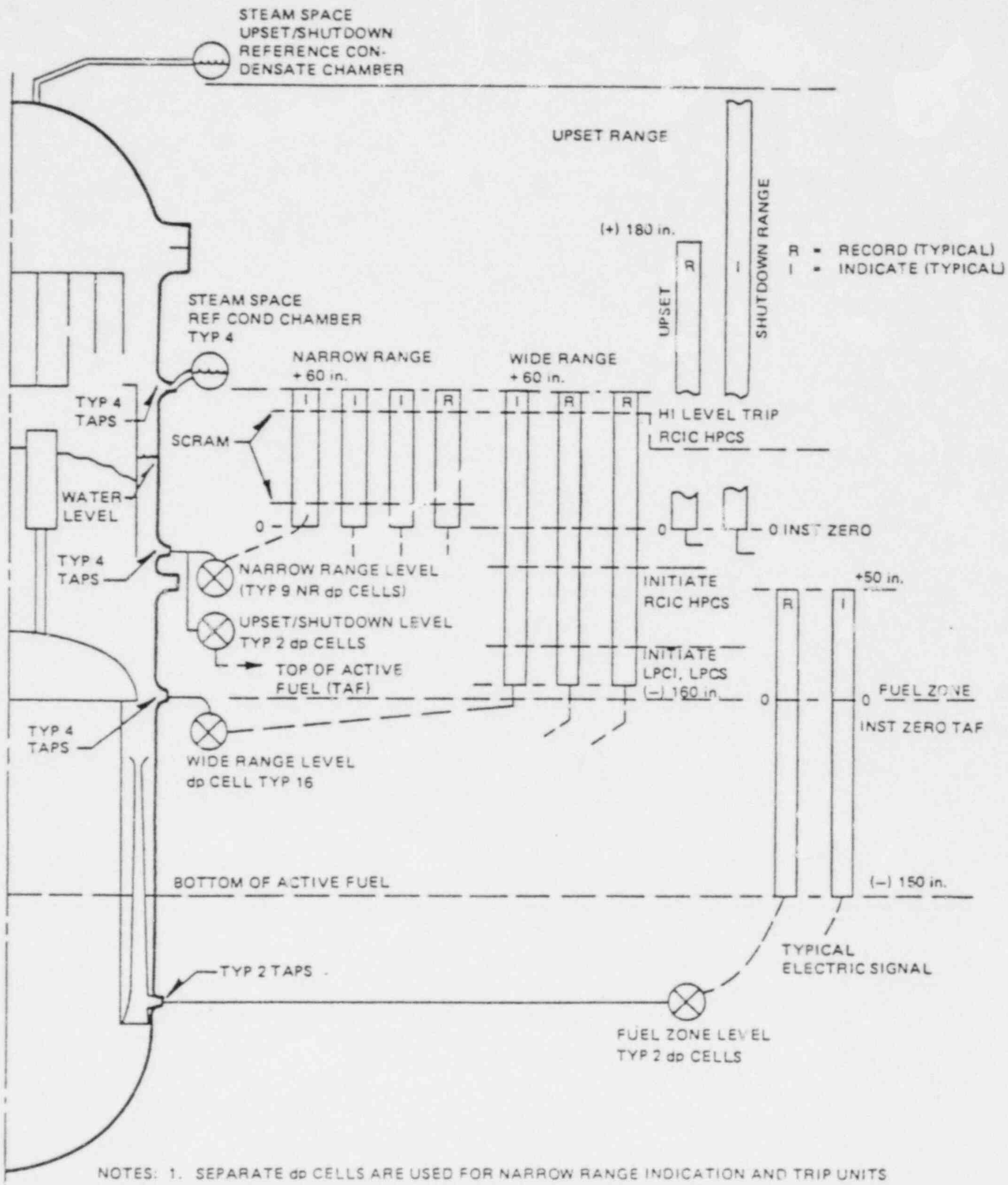
PSO believes that no additional instrumentation is needed for Black Fox Station to monitor inadequate core cooling. Nevertheless due to the insistence of the NRC Staff, PSO will comply with the requirement in Regulatory Guide 1.97, Revision 2 for in-core thermocouples with the recognition and understanding that the requirement is being reconsidered on the docket for the La Salle Nuclear Station which is presently being constructed by Commonwealth Edison Company. Thus if the requirement for in-core thermocouples is changed from that set forth in Revision 2 of Regulatory Guide 1.97, the requirement, as revised, will apply to Black Fox Station. This commitment is not intended to limit the flexibility provided by Regulatory Guide 1.97 to permit the adoption of NRC-approved alternatives to the requirement for core thermocouples as it is presently described in Revision 2 or as the requirement may be amended on the La Salle docket.

Emergency Procedure Guidelines

As discussed in PSO response to Requirement (2)(ii), "Long-Term Program Plan for Upgrading Procedures," an early interest was shown by PSO in symptom oriented emergency procedures. PSO commits to incorporate emergency procedure guidelines for recognizing the approach to inadequate core cooling into the Black Fox Station procedures.

In summary, Black Fox is a BWR-6. Reactor vessel water level is the primary indication of adequate core cooling in a BWR and water level information is presented to the operator in an advanced human engineered Nuclenet 1000* control room. The operator will be well trained and have available up-to-date procedures that incorporate symptom based emergency procedure guidelines.

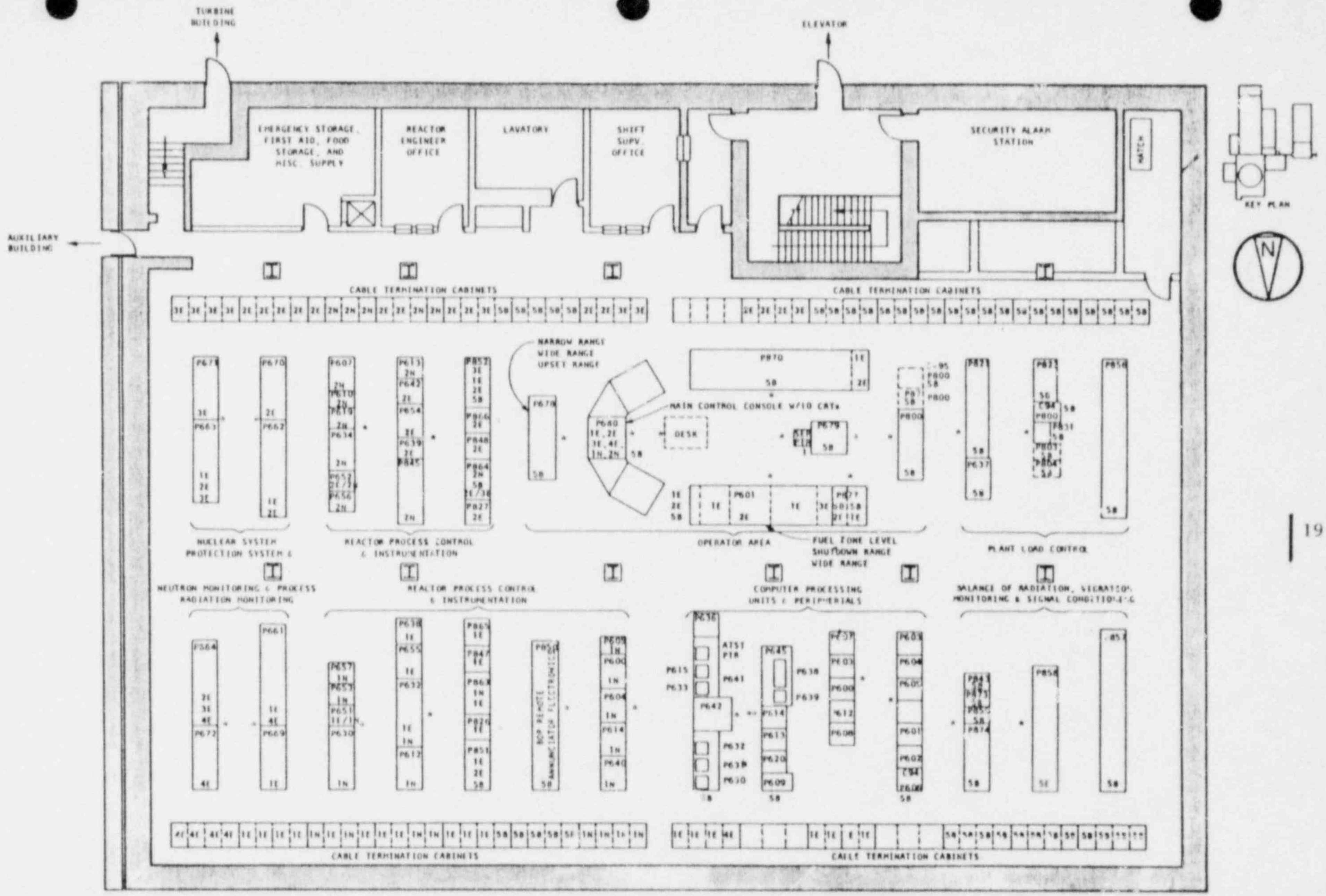
*General Electric Company trademark



NOTES:
1. SEPARATE dp CELLS ARE USED FOR NARROW RANGE INDICATION AND TRIP UNITS
2. INDICATION/RECORD AND TRIP UNITS FOR WIDE RANGE USE COMMON dp CELLS
3. WATER LEVEL FOR INITIATION OF RCIC AND HPCS IS ALSO PERMISSIVE FOR ADS INITIATION

Figure (2)(xviii) - 1

Typical BWR-6 Level Indicators on Reactor Control Panels



NOTE: * INDICATES FRONT OF PANEL

BLACK FOX STATION CONTROL ROOM
REACTOR LEVEL INDICATORS

FIGURE (2) (xviii)-2

19

10 CFR 50.34(e)(2)(xix) INSTRUMENTATION FOR MONITORING ACCIDENT CONDITIONS (REGULATORY GUIDE 1.97)

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(xix) Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage. (NUREG-0718, II.F.3)

PSO RESPONSE:

Introduction

Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," is a pre-TMI Regulatory Guide that was written to describe acceptable methods of providing instrumentation to monitor plant variables and systems during and following an accident. The revised version of Regulatory Guide 1.97, Revision 2 includes post-TMI guidance.

The NRC Staff, as a part of its continuing program to provide regulatory guidance to licensees and license applicants, has identified the need for instrumentation to aid in accident diagnosis and control. Information regarding a minimum group of plant variables is required by the operators to (1) take preplanned manual action to accomplish safe reactor shutdown, (2) determine whether the reactor trip, engineered safety-feature systems, and manually initiated safety systems and other systems important to safety are performing their intended functions (i.e., reactivity control, core cooling, maintaining reactor coolant system integrity, and maintaining containment integrity), (3) determine the potential for causing a gross breach of the barriers to radioactivity release (i.e., fuel cladding, reactor coolant pressure boundary, and containment) or determine if such a breach has occurred.

Following the TMI-2 accident, the NRC Staff developed Revision 2 to Regulatory Guide 1.97 entitled, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Access Plant and Environs Conditions During and Following an Accident." This Regulatory Guide sets forth guidance for meeting this Requirement.

Commitment

PSO will meet the requirements of Regulatory 1.97, Revision 2 as clarified below and in PSO's Response to Requirement (2)(xviii) concerning in-core thermocouples.

PSO RESPONSE: 10 CFR 50.34(e)(2)(xix)

PSO will develop a plan for the selection and location of radiation monitors in containment penetration areas and in areas where access to service safety equipment is required. This plan will be developed in conformance with the provisions of Revision 2 to Regulatory Guide 1.97. Any exceptions will be identified and justification provided. This plan will be submitted to the NRC Staff prior to the procurement of these monitors. 19

Clarifications to Table 1 of Regulatory Guide 1.97, Revision 2

1. The references to "Containment and Drywell Oxygen Concentration," "Drywell Spray Flow" and "Isolation Condenser System Shell Side Water Level and Valve Position" are not applicable to BFS.

2. The references to "Suppression Chamber Spray Flow," "HPCI Flow" and "Core Spray System Flow" mean for purposes of BFS "Containment Spray Flow," "HPCS Flow" and "LPCS Flow" respectively.

Type A Variables

The term "Type A Variables" is defined in Regulatory Guide 1.97, Revision 2 and PSO has developed the variables and necessary manual actions listed below. A final list will be submitted to the NRC during the FSAR review.

The final list of Type A variables and the instrumentation for these variables will satisfy the provisions of Revision 2 to Regulatory Guide 1.97. Any exceptions will be identified and justification provided. 19

Type A Variable	Required Manual Action
1. Suppression Pool Temperature	Initiate Suppression Pool Cooling
2. Containment Hydrogen Concentration	Initiate Containment Combustible Gas Control System
3. Ultimate Heat Sink Basin Level	Ensure Adequate Water Level Is Maintained

10 CFR 50.34(3)(2)(xx) POWER SUPPLIES FOR PRESSURIZER RELIEF VALVES, BLOCK VALVES, AND LEVEL INDICATION

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(xx) Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) level indicators are powered from vital buses; (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety; and (C) electric power is provided from emergency power sources. (Applicable to PWR's only). (NUREG-0718, II.G.1)

PSO RESPONSE:

This requirement is applicable to Construction Permit applications for Pressurized Water Reactors only and hence does not apply to Black Fox Station.

10 CFR 50.34(e)(2)(xxi) DESCRIBE AUTOMATIC AND MANUAL ACTIONS FOR PROPER FUNCTIONING OF AUXILIARY HEAT REMOVAL SYSTEMS WHEN FEEDWATER SYSTEM IS NOT OPERABLE

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic issues. (NUREG-0718, Category 4)

(xxi) Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable. (NUREG-0718, II.K.1.22)

PSO RESPONSE:

Introduction

The accident at Three Mile Island-Unit 2 (TMI-2) on March 28, 1979 involved a main feedwater transient coupled with a stuck-open pressurizer power-operated relief valve and a temporary failure of the auxiliary feedwater system. The temporary failure of the auxiliary feedwater system at TMI created a concern with respect to the effectiveness of automatic and manual operation of auxiliary heat removal systems to mitigate the consequences of a loss of feedwater transient. Consequently, the NRC Staff established this Requirement to assure that BWR plants are adequately designed to provide proper functioning of auxiliary heat removal systems.

Summary

The BWR-6 design has been reviewed by the General Electric Company as documented in NEDO 25224, "GESSAR Assessment Report, Review of BWR-6 Protection in Depth for Transient and Accident Events." PSO has reviewed this document and concluded that it is applicable to the BFS. Based on a review of the auxiliary heat removal systems described below, PSO has concluded that the BFS auxiliary heat removal systems are adequately designed such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable.

System Descriptions

1. Feedwater System

The Feedwater System is a reliable, normally operating system which replenishes reactor coolant inventory to make up for extraction of steam out of the reactor vessel. The Feedwater System consists of two steam turbine driven pumps, each capable of providing 80% of total rated feed flow.

Loss of feedwater can result from various problems including pump failures, feedwater controller failures, operator errors, or reactor system variables such as a high vessel water level (Level 8) trip signal. Table (2)(xxi)-1 lists the reactor vessel level setpoints, their relation to the top of active fuel, and the automatic actions that occur. The loss of one feedwater pump at full reactor power is an easily controlled perturbation and does not result in a plant shutdown. With the loss of one feedwater pump and a subsequent low water level (Level 4) signal, the Reactor Recirculation Flow Control System automatically responds to reduce reactor recirculation flow. This allows the remaining feedwater pump to recover and maintain normal operating water level at a stable lower reactor power level. In the case of a complete LOF, the systems discussed in the following paragraphs are available to inject water into the reactor vessel. A summary of all the plant systems which can supply water to the reactor vessel is provided in Table (2)(xxi)-2.

2. High Pressure Core Spray and Reactor Core Isolation Cooling Systems

In the event of a LOF transient, reactor vessel water level is automatically controlled by backup systems which are diverse and redundant. The High Pressure Core Spray (HPCS) System and Reactor Core Isolation Cooling (RCIC) System are high pressure systems with sufficient capacity such that either system can maintain reactor vessel water level with a LOF. HPCS can maintain level with a LOF and a stuck open relief valve (SORV).

3. Low Pressure Emergency Core Cooling Systems

In the unlikely event of a LOF with a failure of both HPCS and RCIC, the next backup is the set of high flow, low pressure emergency core cooling systems. These systems are the Low Pressure Core Spray (LPCS) System and the Residual Heat Removal (RHR) System operating in the Low Pressure Coolant Injection (LPCI) mode. In a situation where a decreasing reactor vessel water level is accompanied by a relatively slow decrease in reactor pressure, the Automatic Depressurization System (ADS) will open eight of the nineteen safety relief valves. This rapidly drops reactor pressure so that the low pressure emergency core cooling systems can begin supplying large volumes of water to the reactor vessel. These systems initiate automatically.

4. Condensate System

In the long term recovery from a LOF after reactor pressure is reduced, the normal makeup water supply to the reactor vessel is from the Condensate System (CS). The CS can be used provided offsite power is available and a flow path exists through the feedwater piping to the reactor vessel. When reactor pressure falls below the discharge pressure of the condensate booster pumps (approximately 600 psig), these pumps can begin delivering water to the reactor vessel at a high flowrate. With the condensate booster pumps unavailable the condensate pumps can supply water at a high flow rate when reactor pressure decreases below approximately 100 psig. Reactor vessel water level control with the CS can be either automatic or manual. The CS, however, must be manually initiated.

5. Control Rod Drive Hydraulic System

The Control Rod Drive (CRD) Hydraulic System is a normal operating system which adds a relatively small flow to the reactor vessel water inventory. Water is supplied to the reactor vessel from the CRD system via control rod drive mechanism cooling flow and flushing of the reactor recirculation (RR) pump shaft seals. While the CRD system is not a safety related water source for reactor vessel makeup, it will supplement the other systems.

6. Standby Service Water System

For long term cooling, assuming all other systems are unable to provide makeup water to the reactor vessel, Standby Service Water System (SSWS) may be used. Provisions are made in the RHR system whereby a SSWS pump can be manually valved through the RHR system to flood the reactor vessel with water from the ultimate heat sink storage basin.

Transient Descriptions

1. Loss of Feedwater With All Backup Systems Operable

The Loss of Feedwater (LOF) event is an operational transient which occurs with a frequency of approximately 1-2 times per plant-year. The LOF event is a mild transient with respect to maintaining acceptable pressure and fuel thermal margins. Black Fox Station is designed so that the high pressure makeup and inventory maintenance systems (RCIC, HPCS) are independently capable of maintaining the water level above the top of the active fuel given a loss of feedwater. Redundancy of systems and components is provided in the BWR design to provide margin against core uncovery and a high probability that core uncovery will be avoided. The following paragraphs describe the scenario of a LOF with backup systems fully operational and no operator intervention.

Loss of feedwater flow results in a reduction of reactor vessel inventory. Corrective action begins as soon as low feedwater flow is sensed and low level alarm (Level 4) is reached. A reduction of the core recirculation flow is initiated to reduce power and thereby reduce the rate of level decrease. Feedwater pump coastdown causes flow to terminate at approximately five seconds.

Subcooling decreases resulting in a reduction of reactor power level and pressure.

Water level continues to drop until the vessel level (Level 3) scram occurs. As power level decreases, the turbine steam flow starts to drop off because the pressure regulator is attempting to maintain pressure by closing the turbine control valves.

Vessel water level continues to drop reaching the Level 2 trip at about twenty seconds. At this time, the recirculation system is completely tripped and HPCS and RCIC operation is initiated. HPCS and RCIC inject into the vessel causing the vessel water level inside the shroud to reach its minimum value about 6.5 feet above the top of active fuel. In addition, operation of both HPCS and RCIC will cause the vessel pressure to decrease to the point at which a low pressure main steamline isolation occurs.

After flow to the reactor vessel from HPCS and RCIC has been terminated at high vessel water level (Level 8), the vessel will repressurize to the set point of the lowest set safety relief valve (SRV) which will open to limit the pressure rise caused by decay heat of the fuel. One or more SRVs will cycle open and closed to maintain pressure control. Vessel inventory will be lost through the open SRV's. When water level decreases to Level 2, HPCS will begin injecting water again into the reactor vessel. PSO's response to Requirement (1)(v) describes how RCIC control logic will be modified to automatically restart RCIC flow when Level 2 is reached.

Reactor vessel water level will be maintained between Level 2 and Level 8 by RCIC and HPCS. Reactor overpressure will be prevented by SRV cycling. A gradual decrease in pressure will result from the operation of RCIC and HPCS, and from decreasing decay heat generation.

2. Loss of Feedwater With A Stuck Open Safety Relief Valve

BFS is adequately equipped to mitigate the consequences of the LOF event as it relates to core cooling without operator assistance under all conditions within the design basis, with or without a stuck-open relief valve (SORV). This is achieved through the automatic functioning of various mitigating systems.

A stuck-open relief valve, even with a complete loss of feedwater, is a controllable event. The consequences of a SORV can be mitigated by the operation of RCIC and HPCS. The scenario of automatic actions will be essentially the same as described in the previous section down to the point where one of the SRVs fails to close. The cooldown rate and depressurization of the reactor will be accelerated under these conditions. If the SRV does not reseat, reactor pressure will decrease to 300 psig in about one hour.

3. Loss of Feedwater With Concurrent Failure of HPCS and RCIC

Under normal conditions, the high pressure makeup water systems will provide sufficient water to restore the level to the normal range. Plant shutdown or restart can then be accomplished. If the HPCS and RCIC should fail to start, the vessel water level continues to drop and the level outside the core shroud reaches the low level (Level 1) trip at about 425 seconds. At this time the main steam line isolation valves will close. LPCS and RHR (in LPCI mode) pumps start. Division 1 and 2 Diesel Generators start and come up to speed. ADS could be manually activated to depressurize the reactor so that LPCS and LPCI can begin injecting into the reactor vessel. PSO's response to Requirement (1)(vii) describes how ADS control logic will be modified to avoid the potential need for operator intervention to assure adequate core cooling. Due to the large capacity of the low pressure systems, they will rapidly reflood the reactor. Once the vessel is reflooded, the operator can then proceed to place the reactor in cold shutdown.

4. Operator Response to Loss of Feedwater

The primary function of the operator is to monitor the operation of the automatic systems and to assume control in restoring the system to a normal operational condition. As has been described, if HPCS and/or RCIC initiate automatically, and if SRV's operate properly, no operator intervention is required. However, the operator may assume manual control of HPCS and RCIC in order to effect a smoother recovery from the RPV level transient. In this case, the unit may be returned to service when the problem which initiated the trip of feedwater is corrected. During this period, the principal duty of the operator will be to monitor system operation and ensure that reactor water inventory is maintained.

BFS administrative, operating, and emergency procedures will specify detailed actions for the operator so that all systems capable of providing makeup water to the reactor vessel will be most effectively employed to ensure adequate reactor core cooling. These emergency procedures will be summarized in the FSAR. The BFS operator training program will emphasize the analysis of and corrective action for loss of feedwater transients from various causes, including situations of concurrent backup equipment failures. 19

Conclusion

The BFS Nuclear Steam Supply System is designed to be self-contained and self-actuated to assure reactor core cooling. An isolation event can be totally accommodated initially by automatic operation of engineered safety feature systems and the Reactor Core Isolation Cooling (RCIC) System which are redundant and diverse. These systems restore and maintain system parameters. During the long term, however, there is adequate time for the operator to take appropriate action. The operator need monitor and control only reactor vessel pressure and level. Furthermore, the operator has multiple parameters available to provide additional information on system conditions.

In summary, the Black Fox Station auxiliary heat removal systems are designed such that automatic and manual actions are available to ensure proper functioning when the feedwater system is not operable.

TABLE (2) (xxi)-1
SUMMARY OF REACTOR VESSEL WATER LEVEL TRIPS

Reactor Vessel Setpoints	Inches Above Top of Active Fuel	Automatic Actions
8	222	Reactor Scram
		Trip Main Turbine
		Trip Feed Pump Turbines & Condensate Booster Pumps
		Shutdown RCIC Turbine
		Close HPCS Injection Valve
7	207	High Level Alarm
5	203	Normal Water Level
4	199	Low Level Alarm
		Reactor Recirculation (RR) Flow Control Valve Runback (with concurrent loss of one feed pump)
3	177	Reactor Scram
		ADS Confirmation Signal
		RR Pump Shift to Slow
		RHR Isolation (Shutdown Cooling Mode)
2	131	Initiate RCIC
		Initiate HPCS (and start Division 3 Diesel Generator)
		Trip RR Pumps
		Isolate Reactor Water Cleanup (RWCU) System
		Partially Isolate Containment and Selected Reactor Plant Systems via Nuclear Steam Supply Shutoff System (NSSSS)
		Initiate Standby Gas Treatment System (SGTS)
1	19	Initiate RHR (LPCI Mode)
		Initiate LPCS
		Start Division 1 and 2 Diesel Generators
		Shut Main Steamline Isolation Valves (MSIV)
		ADS Actuation Logic Signal

NOTE: Level 6 is not used.

TABLE (2)(xxi)-2

FUNCTIONS AVAILABLE AFTER LOSS OF FEEDWATER

Function	BWR/6 System	No. Pumps	Approximate Flow per Pump (gpm)	Power Source: Off-Site Electrical or Steam	Power Source: On-Site Diesel Generator
Supply Water to Maintain the Core Covered					
a. High Pressure	Nat Circ[a]	N/A			
	HPCS	1	1550	Off-site	Division III
	RCIC	1	700	Steam	Divisio I - Controls only
	Restore FW	2	18000	Steam/Off-site	None
	CRD[b]	2	50	Off-site	None
b. Low Pressure	Nat Circ[a]	N/A			
	HPCS	1	6100	Off-site	Division III
	RCIC	1	700	Steam	Division I - Controls only
	LPCS	1	6200	Off-site	Division II
	LPCI	3	7200	Off-site	Division I; Division II
	CS	3(Cond) 3(Cond. Boost)	6000	Off-site	None
	CRD[b]	1 (+1 Backup)	50	Off-site	None
	RHR/SSWS	1	5000	Off-site	Division II

[a] The BWR has inherent strong natural circulation. It is only necessary to maintain the core covered with water to assure adequate core cooling.

[b] In a post-scram configuration, with scram inlet valves open, flow to the reactor vessel may increase up to the runout flow of the CRD pump (approximately 100 gpm at high reactor pressure and approximately 170 gpm at low reactor pressure).

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10 CFR 50.34(e)(2)(xxii) ANALYSIS AND UPGRADING OF INTEGRATED CONTROL SYSTEM

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(xxii) Perform a failure modes and effects analysis of the integrated control system (ICS) to include consideration of failures and effects of input and output signals to the ICS. (Applicable to B&W-designed plants only). (NUREG-0718, II.K.2.9)

PSO RESPONSE:

This requirement is applicable to Construction Permit applications for Babcock & Wilcox designed Pressurized Water Reactors only and hence does not apply to Black Fox Station.

10 CFR 50.34(e)(2)(xxiii) HARD-WIRED SAFETY-GRADE ANTICIPATORY REACTOR TRIPS

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(xxiii) Provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip. (Applicable to B&W-designed plants only). (NUREG-0718, II.K.2.10)

PSO RESPONSE:

This requirement is applicable to Construction Permit applications for Babcock & Wilcox designed Pressurized Water Reactors only and hence does not apply to Black Fox Station.

10 CFR 50.34(e)(2)(xxiv) CENTRAL WATER LEVEL RECORDING

NRC POSITION:

(2) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(xxiv) Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements. (NUREG-0718, II.K.3.23)

PSO RESPONSE:

Introduction

After the accident at Three Mile Island-Unit 2 (TMI-2), the Bulletins and Orders Task Force (B&OTF) was established within the NRC and made responsible for reviewing and directing TMI-2 related staff activities associated with loss of feedwater transients and Loss of Coolant Accidents (LOCA) for all operating plants. A generic review of General Electric designed Boiling Water Reactor (BWR) plants was conducted by the B&OTF, and documented in NUREG-0626, "Generic Evaluation of Feedwater Transients and Small Break Loss of Coolant Accidents in GE Designed Operating Plants and Near Term Operating License Applications."

This review identified improvements in systems, procedures, and analyses which, the B&OTF believed, would make GE-designed BWR's less susceptible to core damage during accidents and transients coupled with systems failures or operator errors. The recommendation of NUREG-0626 stated that "In order to simplify the reading of the water level in the vessel and to provide the opertors with a record of water level during transients, all BWR's should have the capability to record vessel water level over the range from the top of the vessel dome to the lowest pressure tap. This range of water level should be available in one location on recorders which meet normal post-accident recording requirements. The recorders should be started on a reactor trip signal."

A revised version of this B&OTF recommendation was subsequently incorporated into Revision 2 of Regulatory Guide 1.97 and identified as a requirement for construction permit applicants in NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License." Regulatory Guide 1.97, Revision 2 requires centrally located post-accident monitoring instrumentation with the capability to continuously record reactor vessel water level over the range from the bottom of the core support plate to the center line of the main steam lines.

PSO RESPONSE: 10 CFR 50.34(e)(2)(xxiv)

Commitment

The Black Fox Station design will provide post-accident monitoring instrumentation with the capability to continuously record reactor vessel water level over the range from the bottom of the core support plate to the center line of the main steam lines. This instrumentation will meet the requirements of Regulatory Guide 1.97, Revision 2.

10 CFR 50.34(e)(2)(xxv) UPGRADE LICENSEE EMERGENCY SUPPORT FACILITIES

NRC POSITION:

(2) To satisfy the following requirements, the application will provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(xxv) Provide an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a nearsite Emergency Operations Facility. (NUREG-0718, III.A.1.2)

PSO RESPONSE:

Introduction

The accident at Three Mile Island-Unit 2 identified the need for improvements in the response to and control of accidents at nuclear power stations. Some of the identified improvements include:

- Establishing formal licensee, local, state, and federal organizations to better manage and effectively coordinate emergency response support;
- Developing integrated emergency response facilities and data systems to aid in this management;
- Providing for better information needed to assess conditions at a station and its environs prior to, during, and following an accident;
- Providing an improved capability by the licensee and federal organizations to provide recommendations to state and local authorities on actions protecting the public; and
- Providing transmission of more accurate information to federal, state, and local emergency response organizations, and to the general public.

With respect to near-term construction permit applicant activities and responsibilities, the NRC has determined that the emergency response facilities that will provide the necessary improvements are the onsite Technical Support Center (TSC), onsite Operational Support Center (OSC), and the Emergency Operations Facility (EOF). These facilities will operate as an integrated system to support the control room in the mitigation of the consequences of accidents and to enhance the capability to respond to abnormal station conditions. These facilities will help in providing a graduated response capability dependent on the severity of an emergency. Figure (2)(xxv)-1 shows the location of each of PSO ...gency Response Facilities (ERF's) with respect to Black Fox Sta... FS). Figure (2)(xxv)-2 shows an expanded view of the major sta... uildings. The preliminary location of key BFS emergency response personnel is shown in Table (2)(xxv)-1. 19

Emergency Operations Facility

1. General Description

   The EOF is a nearsite support facility for the management of PSO's over-all emergency response (including coordination with federal, state and local officials, coordination of radiological and environmental assessments, and determination of recommended public protective actions. The EOF will have appropriate technical data displays and station records to assist in the diagnosis of station conditions to evaluate the potential or actual release of radioactive materials to the environment. A senior PSO official in the EOF will organize and manage PSO offsite resources to support the TSC and the control room operators.

2. EOF Function

   The BFS Emergency Operations Facility will be controlled and operated by PSO and will serve as the location for performing the following functions:

   o Management of over-all PSO emergency response,

   o Coordination of radiological and environmental assessment,

   o Determination of recommended public protective actions, and

   o Coordination of emergency response activities with federal, state and local agencies.

   The EOF space may be used for other purposes during normal operations. Provisions will be set forth to assure the emergency functions of the EOF are not degraded by those activities and will ensure all necessary systems meet required availability. These provisions will include adequate security protection of the facility during normal and emergency conditions.

3. Activation and Use

   The EOF will be activated for Site Area Emergency and General Emergency classes.

PSO RESPONSE: 10 CFR 50.34(e)(2)(xxv)

4. EOF Location

The EOF will be located approximately 0.9 miles east of the station. In locating the EOF, several factors were considered:

- Whether the location provides optimal functional and availability characteristics for carrying out the licensing functions specified for the EOF (i.e., over-all strategic direction of PSO onsite and support operations, determination of public protective actions to be recommended by PSO to offsite officials, and coordination of PSO with federal, state, and local organizations including the NRC).

- Whether the EOF functions would be interrupted during radiation releases for which it was necessary to recommend protective actions for the public to offsite officials.

A conceptual floor layout for the EOF is shown on Figure (2)(xxv)-3.

5. Location, Structure, and Habitability

The EOF is located within ten (10) miles of the TSC; therefore the following habitability criteria will be met:

- The EOF will be well engineered for the design life of BFS in accordance with the Uniform Building Code. The EOF will be able to withstand the expected adverse conditions of high winds (other than tornadoes) and floods.

- A radiation reduction factor greater than or equal to five will be provided to those areas of the EOF in which dose assessments, communications, and decision making take place.

- Ventilation protection will be accomplished with HEPA filters (no charcoal) and will function in a manner comparable to the control room and TSC ventilation systems.

Because the nearsite EOF is within ten (10) miles of the TSC, a preliminary EOF backup location will be provided at the PSO corporate offices in Tulsa, approximately twenty-three (23) miles west of the TSC. The additional three (3) miles beyond the twenty (20) mile siting criterion will not impair movement between the nearsite and backup EOF's nor will it impede communications with emergency response personnel. The backup EOF location is shown on Figure (2)(xxv)-3a. 19

6. EOF Staffing and Training

The EOF will be staffed to provide the over-all management of PSO resources and the continuous evaluation and coordination of PSO activities during and after an accident. Upon EOF activation, designated personnel will report directly to the EOF to achieve full functional operation within one hour. A PSO senior management official will be in charge of all PSO activities in the EOF. The EOF staff will include personnel to manage PSO onsite and offsite radiological monitoring, to perform radiological evaluations, and to interface with offsite officials. The specific number and type of personnel assigned to the EOF may vary according to the emergency class. The staffing for each emergency class will be fully detailed in the BFS Final Emergency Response Plan. The EOF staff will participate in EOF activities drills, conducted periodically in accordance with the BFS Final Emergency Response Plan. These drills will include operation of all facilities that will be used to perform the EOF functions.

7. EOF Size

The EOF building complex will be large enough to provide the following:

- Working space for the personnel assigned to the EOF as specified in the BFS Final Emergency Response Plan, including federal, state and local agency personnel. A working space of approximately 75 square feet per person will be used as a basis for size and layout of the EOF. The conceptual EOF layout provided in Figure (2)(xxv)-3 assumes approximately 25 persons from PSO, 10 persons from state and local agencies, 9 persons from NRC and 1 person from FEMA.

- Space for EOF data system equipment needed to transmit data to other locations.

- Sufficient space to perform repair, maintenance, and service of equipment, displays and instrumentation.

- Space for access to communications equipment by all EOF personnel who need communications capabilities to perform their functions.

- Space for access to functional displays of EOF data.

- Space for storage of station records and historical data or space for means to readily acquire and display those records.

- Separate office space to accommodate at least five NRC personnel during periods that the EOF is activated for emergencies.

- A space to brief select groups of approximately 50 persons.
- A secured entrance.
- Sufficient space outside the EOF for parking PSO, federal, state, and local vehicles.

8. EOF Communications

The EOF will have reliable voice communications facilities to the TSC, the control room, NRC, and state and local emergency operations centers. The normal communication path between the EOF and the control room will be through the TSC. The primary functions of the EOF voice communications facility will be:

- EOF management communications with the designated senior PSO official in charge of the TSC,
- Communications to manage PSO emergency response resources,
- Communications to coordinate radiological monitoring,
- Communications to coordinate offsite emergency response activities, and
- Communications to disseminate information and recommended protective actions to responsible government agencies.

The EOF voice communications facilities will include reliable primary and backup means of communication. PSO will provide a means for EOF telephone access to commercial telephone common-carrier services that bypass any local telephone switching facilities that may be susceptible to loss of power during emergencies. PSO will insure that spare commercial telephone lines to the station are available for use by the EOF during emergencies. The EOF voice communications equipment will include:

- Hot line telephone (located in the NRC office space) on the Emergency Notification System (ENS) to the NRC operations center;
- Dedicated telephone (located in the NRC office space) on the NRC Health Physics Network (HPN);
- Telephones for management communications with direct access to the TSC and the control room;
- Telephones reserved for EOF use to provide access to onsite and offsite locations;

- Radio communications to PSO mobile monitoring teams;
- Communications to state and local operations centers; and
- Communications to facilities outside the EOF used to provide supplemental support for EOF evaluations.

The EOF communications system will also include designated telephones (in addition to the ENS and HPN telephones) for use by NRC personnel. PSO will provide at least three telephone lines for NRC use while the EOF is activated. PSO will also furnish the access facilities and cables to the NRC for the ENS and HPN telephones. Facsimile transmission capability between the EOF, the TSC, and the NRC operations center will be provided.

9. EOF Instrumentation, Data Systems Equipment, and Power Supplies

The EOF will contain equipment for the acquisition, display, and evaluation of radiological, meteorological and station system data necessary to determine protective measures recommended to offsite authorities. This equipment will also be used to evaluate the magnitude and effect of potential or actual radioactive releases and to project offsite doses. Data will be transmitted to the EOF from station process computer systems. The data will be presented in the EOF using equipment such as CRTs, standard keyboards and a printer/plotter. The details of the data acquisition system will be provided in the FSAR.

The data system will display the Safety Parameter Display System (SPDS) formats and data needed in the EOF to analyze and exchange information needed on station conditions with the designated senior PSO station official in charge of the TSC. The system will perform these functions independently from actions in the control room without degrading or interfering with control room and station functions. Trend information display capability will be available in the EOF.

The total EOF data system will be designed to achieve an operational unavailability goal of 0.01 during all station operating conditions above cold shutdown. The term unavailability is used to express a complete loss of system function. Mathematically, it is expressed as a ratio of time duration when a function is lost, to the total duration when the function is required to be available.

The EOF electrical equipment load will not degrade the capability or reliability of any safety related power source. Circuit transients or power supply failures or fluctuations will not cause a loss of any stored data vital to the EOF functions.

PSO RESPONSE: 10 CFR 50.34(e)(2)(xxv)

10. EOF Technical Data and Data Systems

The EOF data set will include radiological, meterological and other environmental data as needed to:

- Assess environmental conditions,
- Coordinate radiological monitoring activities, and
- Recommend implementation of offsite emergency plans.

A sufficient number of data display devices will be provided in the EOF to allow all EOF personnel to perform their assigned tasks. They include:

- Station systems variables,
- Instation radiological variables,
- Meteorological information, and
- Offsite radiological information.

As a minimum EOF data set, selected variables specified in Regulatory Guide 1.97, Rev. 2, Table 1, and selected meteorological variables specified in proposed Rev. 1 to Regulatory Guide 1.23, will be available for display in the EOF. Station system data that is available for display in the control room will be available in the EOF. The sample frequency will be chosen to be consistent with the use of the data.

11. Records Availability and Management

The EOF will have access to up-to-date station records, procedures, and emergency plans needed to exercise over-all management of PSO emergency response resources. The EOF records will include:

- Station technical specifications,
- Station operating procedures,
- Emergency operating procedures,
- Final Safety Analysis Report
- Up-to-date records related to PSO, state and local emergency response plans,
- Offsite population distribution data,

- Evacuation plans,
- Environs radiological monitoring records,
- Licensee employee radiation exposure histories,

And up-to-date drawings, schematics and drawings showing:

- Conditions of station structures and systems down to the component level, and
- Instation locations of these systems.

These records will either be stored and maintained in the EOF (such as a hard copy or microfiche) or will be available via transmittal to the EOF from other records storage locations.

Technical Support Center

1. General Description

The Technical Support Center (TSC) is an onsite facility located close to the control room that will provide station management and technical support to the reactor operating personnel located in the control room during emergency conditions. It will have technical data displays and station records available to assist in the detailed analysis and diagnosis of abnormal station conditions and any significant release of radioactivity to the environment. The TSC will be the primary onsite communications center for the station during an emergency. A senior station official, designated in the BFS Final Emergency Response Plan, will use the resources of the TSC to assist the control room operators by handling the administrative items, technical evaluations, and contact with offsite activities, relieving them of these functions.

A Secondary Technical Support Center (STSC) will be provided in close proximity to the Unit 2 control room to provide a Unit 2 emergency condition management area for key technical and management personnel. 19

2. TSC Activation and Use

The onsite TSC will be activated for the Alert, Site Area, and General Emergency classes.

When the TSC is functional, emergency response functions, except direct supervision of reactor operations and manipulation of reactor system controls, will shift to the TSC. Station administration, technical support functions, and contact with offsite activities to assist the control room operator will be performed in the TSC throughout the course of an accident.

3. TSC Data Systems Reliability

The data systems of the TSC will be designed and constructed to provide a very high degree of reliability. The operational unavailability goal of 0.01 is applicable to the TSC data systems when the reactor is above cold shutdown status. The term unavailability is used to express a complete loss of system function. Mathematically, it is expressed as a ratio of time duration when a function is lost, to the total duration when the function is required to be available.

4. TSC Function

The onsite TSC will provide the following functions:

- Provide station management and technical support to station operations personnel during emergency conditions.
- Relieve the reactor operators of peripheral duties and communications not directly related to reactor system manipulations.
- Prevent congestion in the control room.
- Perform EOF functions for the Alert, Site Area, and General Emergency classes until the EOF is functional.

The TSC will be the emergency operations work area for designated technical, engineering, and senior station management personnel; and other PSO designated personnel required to provide the needed technical support; and a small staff of NRC personnel.

The TSC will have facilities to support the station management and technical personnel and will be the primary onsite communications center for the station during the emergency. TSC personnel will use the TSC data system to analyze the station steady-state and dynamic behavior prior to and throughout the course of an accident.

The TSC facilities may be used by designated operating personnel for normal daily operations as well as for training and emergency drills.

5. TSC Location

PSO has reviewed the existing BFS design to determine the optimum location for the TSC. The objective of the review was to select a location that would provide the maximum facility resources for both Units 1 and 2. A centralized TSC near the normal work place of the technical staff using the TSC is particularly useful because the facilities will more readily become an integral part of normal station operation. Thus, such a location will enhance (1) personnel familiarity with TSC equipment, (2) TSC readiness due to usage for normal daily station analyses, and (3) maintenance reliability through daily use of equipment. Moreover, the document control center for the station can be readily located in a centralized TSC, thereby assuring the ready availability of important station documentation.

Based on the foregoing criteria, a preliminary location on the 610 foot elevation level of the General Services Building has been selected. Travel time between the TSC and the control room for Unit 1 is approximately two (2) minutes. The travel time to the control room for Unit 2 is approximately four (4) minutes. The movement of personnel between the TSC and the two control rooms can be accomplished without difficulty under accident conditions. The guidance in NUREG-0696, "Functional Criteria for Emergency Response Facilities," suggests a two (2) minute travel time objective. In PSO's judgment, the disadvantage of the modest departure from the two (2) minute travel time objective to the Unit 2 control room is more than offset by the substantial benefit of a centralized location near the normal work place of the engineers using the facility.

However, in view of the travel time between the primary TSC and the Unit 2 control room, a secondary TSC for key technical and management personnel will be provided contiguous to the Unit 2 control room. 19

The conceptual layout of the TSC may be found on Figure (2)(xxv)-4. The location of the TSC with regard to the location of the OSC is illustrated on Figure (2)(xxv)-4a. 19

6. TSC Staffing and Training

Upon activation of the TSC, designated personnel will report directly to the TSC and achieve full functional operation within 30 minutes. The TSC staff will consist of sufficient technical, engineering, and senior PSO personnel to provide the needed support to the control room during emergency conditions. A PSO senior station official will coordinate activities in the TSC and interface with the control room, the OSC, and the EOF.

7. TSC Size

The TSC will be large enough to provide:

- Approximately 75 square feet per person;
- Space for the TSC data system equipment needed to acquire, process, and display data used in the TSC;
- Sufficient space to perform repair, maintenance, and service of equipment, displays and instrumentation;
- Space for data transmission equipment needed to transmit data originating in the TSC to other locations;
- Space for personnel access to functional displays of TSC data;
- Space for access to communications equipment by all TSC personnel who need communications capabilities to perform their functions;
- Space for storage of and/or access to station records and historical data; and
- A separate room adequate for at least three persons to be used for private NRC consultations.

The TSC working space will be sized for a minimum of 25 persons, including 20 persons designated by PSO and 5 NRC personnel.

8. TSC Structure

The TSC structure will be built in accordance with sound engineering practices to withstand the expected adverse station conditions during the design life of BFS including adequate capabilities for (1) earthquakes, (2) high winds (other than tornadoes), and (3) floods.

9. TSC Habitability

The TSC will be radiologically habitable to the same degree as the control room under accident conditions, but the ventilation system will not be safety-related. The TSC ventilation system will function in a manner comparable to the control room ventilation system and will include HEPA and charcoal filters. The TSC ventilation system will not be seismic Category I qualified, redundant, or instrumented in the control room. To ensure adequate radiological protection of TSC personnel, radiation monitoring equipment will be provided. Protective equipment will also be provided for the staff who must travel between the TSC and the control room under adverse radiological conditions. Should the TSC become uninhabitable, the TSC station management function will be transferred to the control room.

PSO RESPONSE: 10 CFR 50.34(e)(2)(xxv)

10. TSC Communications

The TSC will have reliable voice communications to the control room, the OSC, the EOF, and the NRC. The TSC voice communications facilities will include means for reliable primary and backup communication.

The TSC voice communications equipment will include:

- Hotline telephone (located in the NRC consultation room) on the NRC Emergency Notification System (ENS) to the NRC operations center;

- Telephone (located in the NRC consultation room) on the NRC Health Physics Network (HPN);

- Telephones for management communications with direct access to the control room, the OSC and the EOF;

- Telephones that provide access to onsite and offsite locations;

- Communications to PSO mobile monitoring teams and to state and local operations centers prior to EOF activation.

The TSC communications system will also include designated telephones (in addition to the ENS and HPN telephones) for use by NRC personnel. PSO will provide two telephone lines for NRC use when the TSC is activated. In addition, PSO will furnish the onsite access facilities and cables to NRC for the ENS and HPN telephones.

11. TSC Instrumentation, Data System Equipment and Power Supplies

Station data will be available for display in the TSC. Hard copies of displays can be made by the video copiers or line printer located in the work area.

The TSC electrical equipment load will not degrade the capability or reliability of any safety-related power source. Sufficient alternate or backup power sources will be provided to maintain continuity of TSC functions and to resume display of TSC data if loss of the primary TSC power sources occurs.

12. TSC Data Systems

The TSC technical data system will receive and display information acquired from the station as needed to perform the TSC function. The data available for display in the TSC will enable the station management, engineering, and technical personnel to aid the control room operators in handling emergency conditions.

Data that is available for display in the control room will be available in the TSC without interference to the control room during emergency operations. The data selected system variables specified in Regulatory Guide 1.97, Rev. 2, Table 1 will be available for display and printout in the TSC.

The TSC displays will include:

- Station systems variables,
- Instation radiological variables,
- Meteorological information, and
- Offsite radiological information.

Data trending capability and SPDS formats will be available in the TSC.

13. TSC Records Availability and Management

The TSC will have access to station records to aid in technical analysis and evaluation of emergency conditions. The station records, operational specifications, and procedures include:

- Station technical specifications,
- Station operating procedures,
- Emergency operating procedures,
- Final Safety Analysis Report,
- Station operating records,
- Station operations reactor safety committee records and reports.

And up-to-date, as-built drawings, schematics, and diagrams showing:

- Conditions of station structures and systems down to the component level,
- Instation locations of these systems.

The Technical Support Center will be fully discussed in the FSAR.

14. Secondary Technical Support Center (Unit 2)

A Secondary Technical Support Center (STSC) will be provided contiguous to the Unit 2 control room. The STSC will provide facilities for Unit 2 emergency management for 5 to 10 key technical and management personnel. Figure (2)(xxv)-4b shows a preliminary layout of the STSC.

The same ventilation system will be used for both the STSC and control room; therefore, habitability provisions will be the same in the STSC and the Unit 2 control room. 19

Plant data and emergency condition status will be provided by the plant computers. The operator interface and the CRT data formats will be the same as that provided in the primary TSC. Reliable voice communication with the support personnel in the primary TSC will also be available.

Operational Support Center

1. General Description

The Operational Support Center (OSC) is an onsite assembly area separate from the control room and the TSC where PSO operations support personnel will report in an emergency. There will be direct communications between the OSC and the control and between the OSC and the TSC so that the personnel reporting to the OSC can be assigned to duties in support of emergency operations.

2. Activation and Use

The OSC will be activated for the Alert, Site Area, and General Emergency classes.

3. OSC Function

The OSC is an onsite area separate from the control room and the TSC where PSO operations support personnel will assemble in an emergency. The OSC will:

- Provide a location where station logistic support can be coordinated during an emergency,
- Restrict control room access to those support personnel specifically requested by the Shift Supervisor. When the OSC is activated, it will be supervised by PSO operations management personnel designated in the Black Fox Station Final Emergency Response Plan to perform these functions.

PSO RESPONSE: 10 CFR 50.34(e)(2)(xxv)

4. OSC Habitability

The OSC's habitability is not comparable to that of the control room; therefore, the Black Fox Station Final Emergency Response Plan will include procedures for evacuation of OSC personnel in the event of a large radioactive release. The BFS Final Emergency Response Plan will include provisions for the performance of the OSC functions by essential support personnel from other onsite locations.

5. OSC Communications

The OSC will have direct communication with the control room and with the TSC so that the personnel reporting to the OSC can be assigned duties in support of emergency operations. The OSC communications system will consist of one telephone extension to the control room, one telephone extension to the TSC, and one telephone capable of reaching onsite and offsite locations.

6. OSC Location

The preliminary location selected for the OSC is the training area large classroom at the north end of the General Services Building on the fourth floor. Figure (2)(xxv)-5 illustrates the preliminary OSC location. This area provides sufficient space in a central location. Its location does not interfere with access to the control room, the TSC, or the unaffected unit.

7. OSC Details

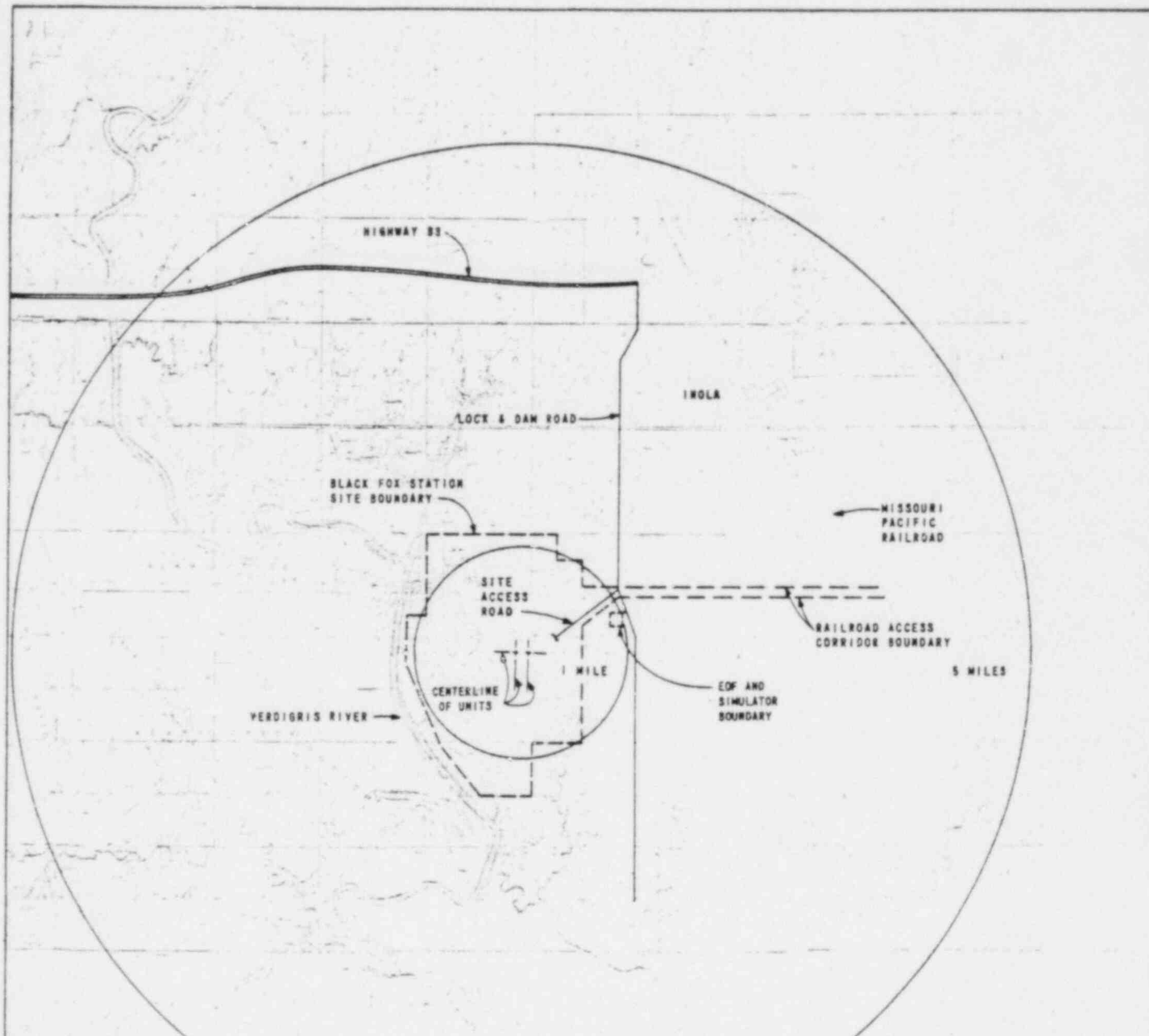
Details concerning the OSC final location, backup assembly area, station access control, staffing requirements, conduct of operations, training, and equipment storage locations will be provided in the BFS Final Emergency Response Plan.

TABLE (2)(xxv)-1

Preliminary location of key BFS emergency response personnel during alert or greater emergencies.

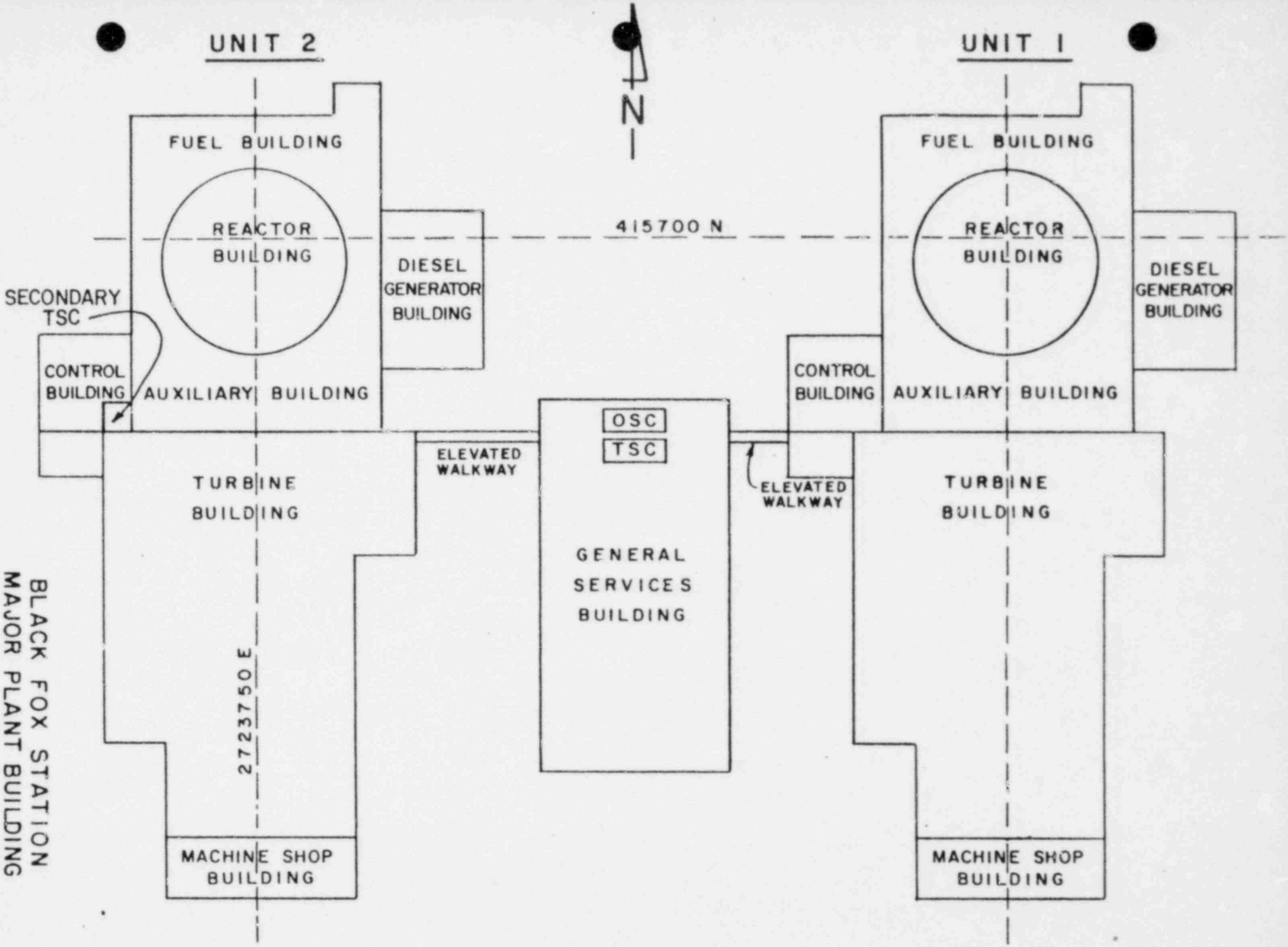
	Emergency Class			
Personnel	Alert	Site Area	General Emergency	
Recovery Manager	TSC*	EOF	EOF	
Public Relations Director	EOF	EOF	EOF	
Emergency Coordinator	EOF	EOF	EOF	
Station Manager	TSC*	TSC*	TSC*	
Health Physics Supervisor	TSC	TSC	TSC	19
Shift Technical Advisor	TSC*	TSC*	TSC*	
Operations Supervisor	CR	CR	CR	
Shift Supervisor	CR	CR	CR	
Fire Brigade, Damage Control	OSC	OSC	OSC	
Engineering Support	TSC	TSC	TSC	

* Also Secondary Technical Support Center (STSC)



BLACK FOX STATION
SITE ARRANGEMENT

Figure (2) (xxv)-1



BLACK FOX STATION
MAJOR PLANT BUILDING
ARRANGEMENT FIGURE (2)(XXV)-2

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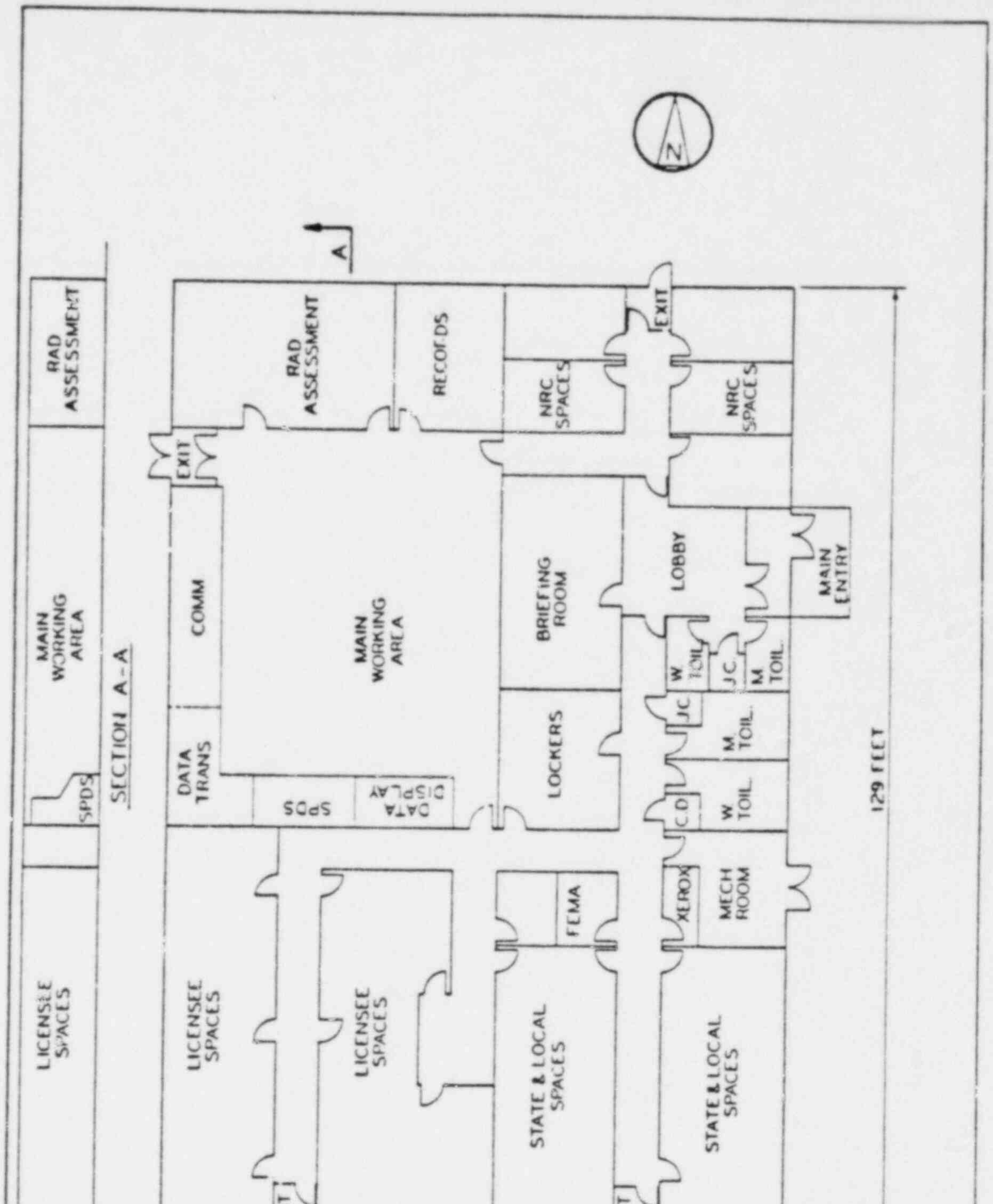
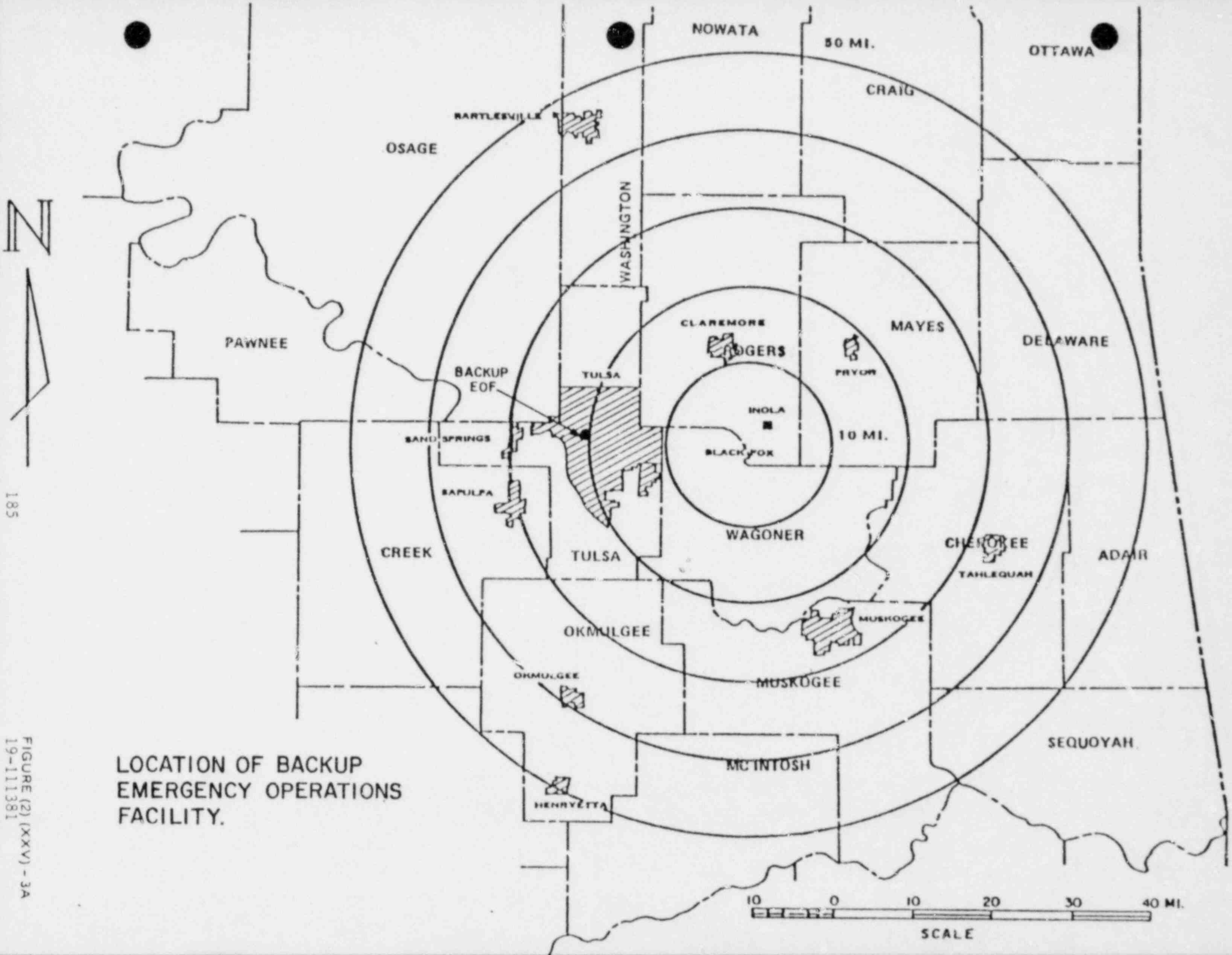


BLACK FOX STATION EMERGENCY OPERATIONS FACILITY
CONCEPTUAL FLOOR PLAN

Figure (2)(xxv)-3



LOCATION OF BACKUP EMERGENCY OPERATIONS FACILITY.

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FIGURE (2) (XXV) - 3A
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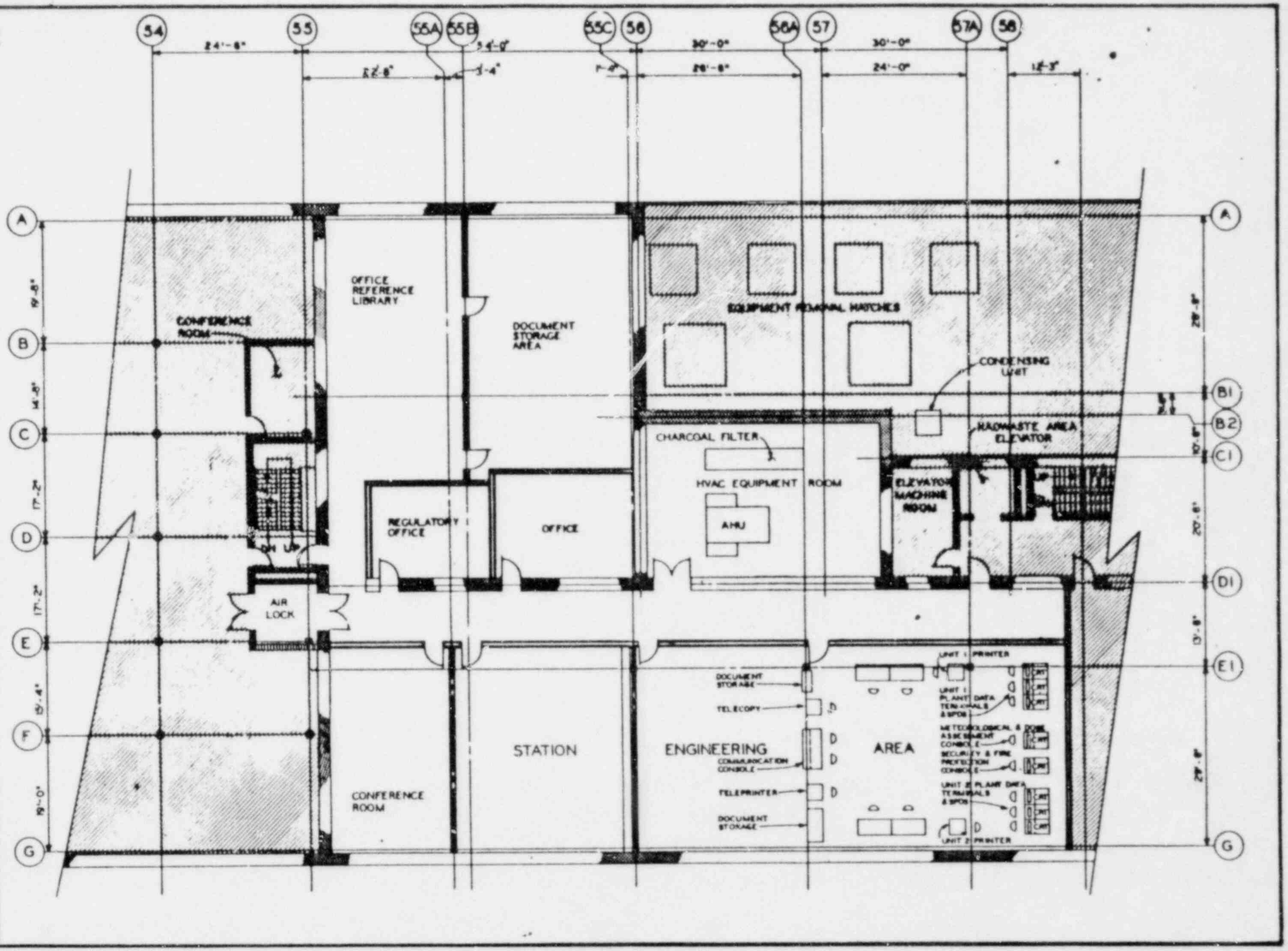

BLACK FOX STATION TECHNICAL SUPPORT CENTER
GENERAL SERVICES BUILDING
EL. 610'-0" FLOOR PLAN FIGURE (2)(XXV)-4

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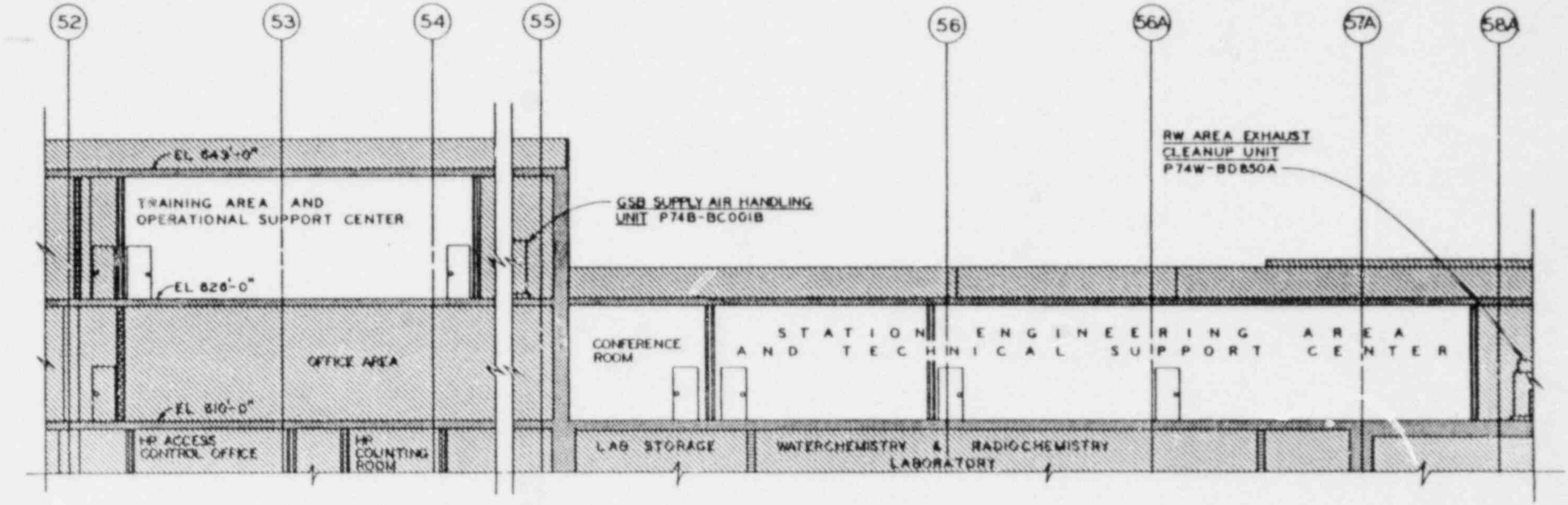

PARTIAL SECTION

BLACK FOX STATION
PARTIAL SECTION GENERAL SERVICE BUILDING
TECHNICAL & OPERATIONAL SUPPORT CENTER

FIGURE (2) (xxv)-4A

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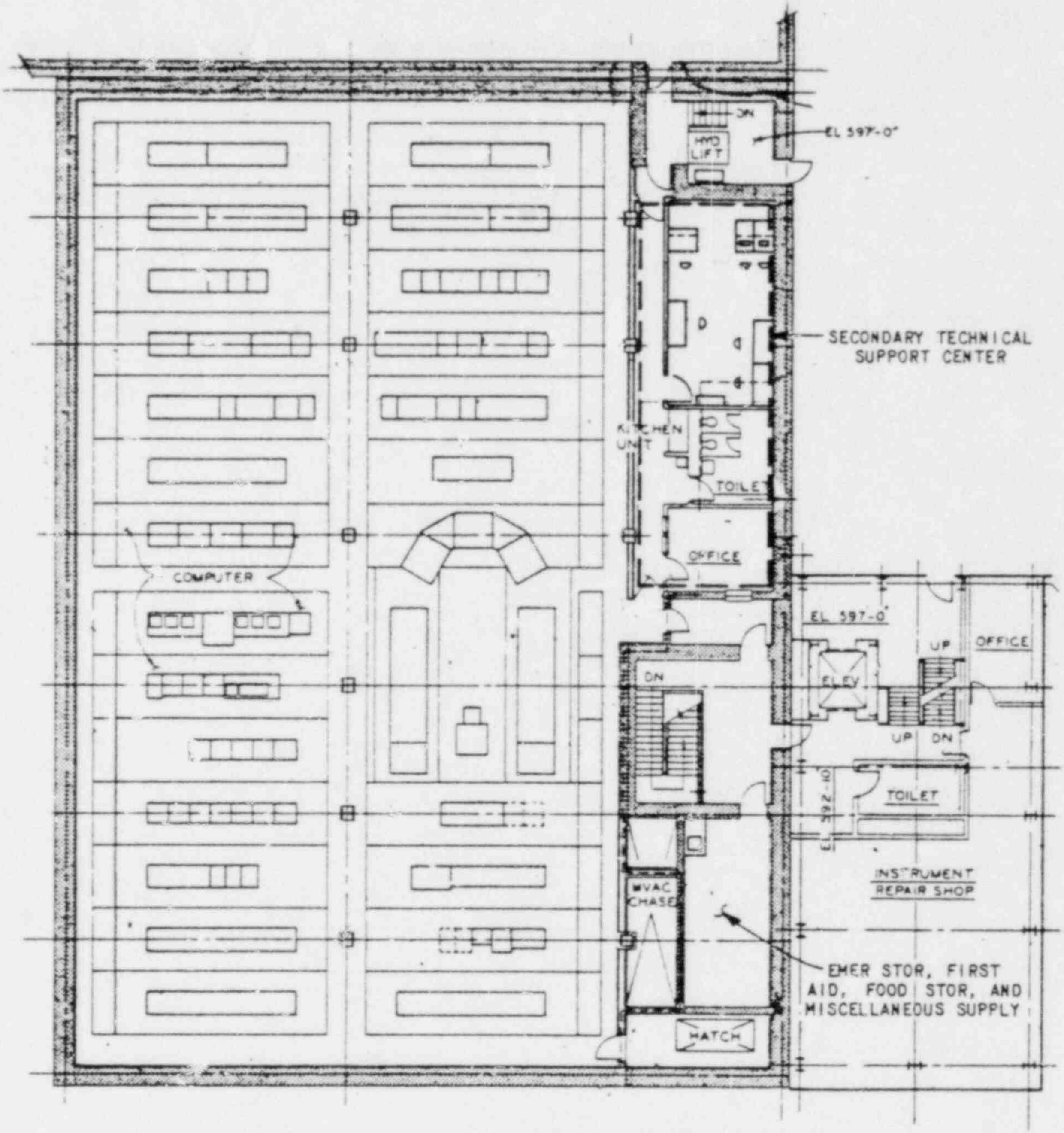
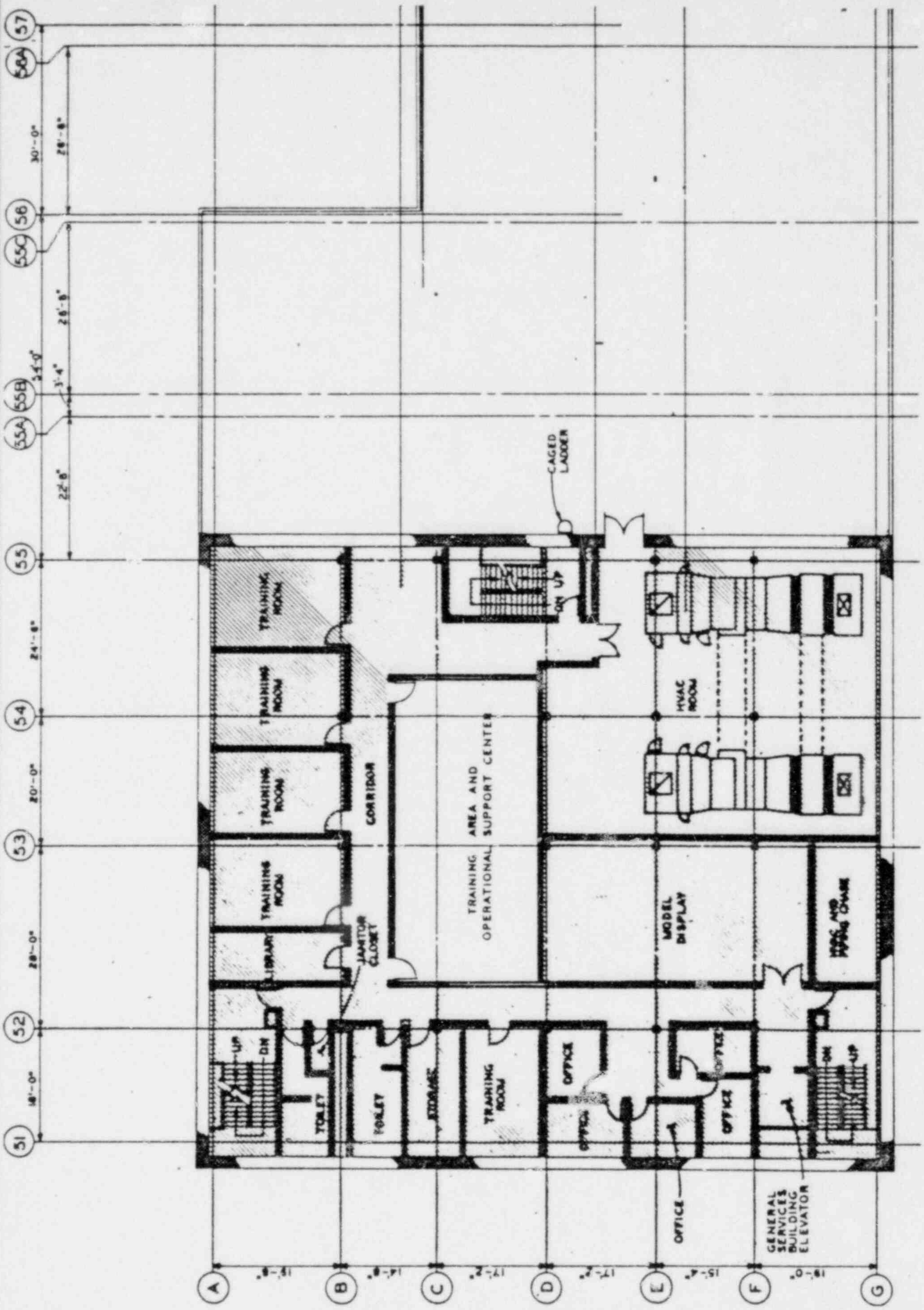


UNIT 2 - CONTROL BUILDING
PLAN AT ELEVATION 592'-10"

BLACK FOX STATION -
SECONDARY TECHNICAL
SUPPORT CENTER
FIGURE (2) (XXV)-4B

19-111381



BLACK FOX STATION OPERATIONAL SUPPORT CENTER
GENERAL SERVICES BUILDING
EL. 626'-0" FLOOR PLAN FIGURE (2)(XXV)-5

10 CFR 50.34(e)(2)(xxvi) PRIMARY COOLANT SOURCES OUTSIDE THE CONTAINMENT STRUCTURE

NRC POSITION:

(2) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(xxvi) Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID 14844 source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. (NUREG-0718, III.D.1.1)

PSO RESPONSE:

Introduction

During the accident at Three Mile Island, systems located outside the containment were used with resulting releases of radioactive material to the ventilation systems. These releases resulted from leaking valves, waste gas compressor seals and open rupture discs. The residual heat removal system was not used as designed for several reasons, one of which was the uncertainty of the leakage characteristics of the system.

As a consequence of the Staff's concern about adequate leak detection, they have, in NUREG-0718 (Revision 1), "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License," required BWR Construction Permit applicants to submit a leakage control program, including an initial test program, a schedule for retesting these systems, and actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not present the use of systems needed in an emergency.

Purpose of Leakage Control

In both PWR's and BWR's, primary coolant may be circulated outside primary containment in a post-accident environment. Consequently, emphasis is placed on minimization of leakage paths during design, leak detection, leak collection, and leak treatment during operation. The

PSO RESPONSE: 10 CFR 50.34(e)(2)(xxvi)

BFS design includes a secondary containment because of the potential for leakage during normal operation. The secondary containment houses most of the auxiliary systems which may contain primary coolant in the event of an accident. The standby gas treatment processes the air inside the secondary containment for radioactivity control before discharge to the atmosphere.

Leakage Control Program

1. System Description

   The BFS design incorporates several systems, portions of which are located outside primary containment, that are provided for accident prevention and mitigation. Consequently, the design requires that some of these continue to function in an accident environment.

   Systems located outside of the containment structure have been reviewed to determine which may contain TID 14844 source term radioactive materials following an accident. These systems are listed below:

   a. Residual Heat Removal
   b. Low Pressure Core Spray
   c. High Pressure Core Spray
   d. Reactor Core Isolation Cooling
   e. Reactor Water Cleanup
   f. Plant Equipment and Floor Drains
   g. Main Steam Isolation Valve Leakage Control
   h. Standby Gas Treatment
   i. Post Accident Sampling
   j. Containment Atmosphere Monitoring

   The above systems were also reviewed to determine which systems would not isolate as the result of an accident. The non-isolating systems, utilized to mitigate the consequences of a serious transient or accident, are listed below:

   a. Residual Heat Removal
   b. Low Pressure Core Spray
   c. High Pressure Core Spray
   d. Reactor Core Isolation Cooling
   e. Main Steam Isolation Valve Leakage Control
   f. Standby Gas Treatment

2. Design Basis to Minimize Leakage

   Systems located outside of containment that may contain radioactive material are designed, to the maximum extent practicable, tominimize radiation exposures to workers and to the general public. Some of the generic guidelines used during system design are given below:

PSO RESPONSE: 10 CFR 50.34(e)(2)(xxvi)

a. Welded construction will be used to the maximum extent practicable.

b. Valve and pump selection criteria will include packing and seal considerations to minimize leakage.

c. Pressure relief valves will be piped to enclosed vessels or to the Plant Equipment and Floor Drain System.

d. Test connections will be provided to aid in identification of leakage during initial and periodic leak testing.

e. Pathways for containment bypass leakage have been analyzed and means for controlling and minimizing leakage to the environment are included in the BFS design.

f. A quality assurance program will be implemented to assure that system components that may contain highly radioactive materials are designed and fabricated, and materials selected in accordance with requirements commensurate with their importance to safety or in mitigating radioactivity release.

3. Special Systems

The design of BFS incorporates special systems to detect and/or reduce the consequences of system leakage as described below:

a. The Leak Detection System provides monitoring and detection of excessive leakage from the following systems:

- Low Pressure Coolant Injection (Emergency Core Cooling System mode of Residual Heat Removal)
- Low Pressure Core Spray (Emergency Core Cooling System)
- High Pressure Core Spray (Emergency Core Cooling System)
- Reactor Core Isolation Cooling
- Reactor Water Cleanup
- Residual Heat Removal (Emergency Core Cooling System)

The leak detection system monitors equipment room temperatures, process system flow, sump flow rate, and radiation level. The systems above are isolated automatically on detection of excessive leakage by the Leak Detection System except for the Emergency Core Cooling System function. Excessive leakage is annunciated for all systems.

b. The Process Radiation Monitoring System will monitor the radioactivity of in fluid streams and automatically isolate the following:

- Fuel Building and General Services Building Ventilation Exhaust
- Habitability Area of Control Building
- Main Condenser Vacuum Pump
- Main Steam Isolation Valves

The Process Radiation Monitoring System will also monitor the radioactivity in fluid streams which do not contain radioactivity during normal operation but which have the potential for becoming contaminated through inter-system leakage. Such leakage is annunciated when the preset limits are exceeded.

c. Waste treatment systems have been provided for BFS to process and reduce the concentration of radioactive fluids before further usage or release to the environment. These systems are discussed below:

- A Standby Gas Treatment System is provided to control exfiltration of contaminated air from the secondary containment following an accident which results in abnormally high airborne radioactivity levels in the Shield Building, Auxiliary Building and Fuel Building. The Standby Gas Treatment System will operate to maintain a subatmospheric pressure in these areas. Gaseous radioactive discharges from Engineered Safety Features Systems, which are not isolated from the containment following an accident, will be collected and filtered by the Standby Gas Treatment System before release to the environment.

- The Main Steam Isolation Valve Leakage Control System is designed to minimize the release of fission products which could bypass the Standby Gas Treatment System after a LOCA. This is accomplished by directing the leakage from the closed main steam lines through a bleed line into an area served by the Standby Gas Treatment System, eliminating direct leakage to the environment.

Leak Testing Program

PSO will determine actual base line data during construction and preoperational testing. Leakage rate tests will be performed in accordance with the criteria of Appendix J of 10 CFR Part 50 and ASME Section XI. Systems containing gases will be tested, using tracer gases, by pressure decay testing or by metered makeup tests. Acceptance criteria will be established and guidance for corrective actions will be included to assure continued low leakage rates.

Periodic tests will be conducted by surveying the affected systems under normal or test pressures. PSO will also implement preventative and periodic maintenance programs to minimize system leakage. The

PSO RESPONSE: 10 CFR 50.34(e)(2)(xxvi)

testing schedule and the details of the testing and surveillance program will be described in the BFS PSAR.

10 CFR 50.34(e)(2)(xxvii) IN-PLANT RADIATION MONITORING

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(xxvii) Provide for monitoring of in-plant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions. (NUREG-0718, III.D.3.3)

PSO RESPONSE:

Introduction

Following the accident at TMI-2, it was determined that the monitoring of in-plant radiation and airborne radioactivity was not adequate for emergency conditions. The number and location of area radiation monitors was inadequate to provide information required to minimize post accident radiation exposures. Also, the equipment used to determine airborne radioactivity concentrations was inadequate and the resulting conservative assumptions used to estimate these concentrations caused the airborne concentrations to be overstated. The overstated airborne radioactivity concentrations caused plant operations personnel to use cumbersome respiratory equipment thereby unnecessarily impeding their efforts to deal effectively with the accident. The NRC Staff has developed the requirement for augmented sampling and monitoring equipment to assure adequate monitoring of in-plant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions.

Description of BFS Equipment and Facilities

In-plant radiation and airborne radioactivity will be monitored at BFS by the Area Radiation monitoring System (ARM) and the Process Radiation Monitoring System (PRM). The PRM is described in the BFS PSAR Section 11.4 and the ARM is described in Section 12.3.4.

In addition to the ARM and the PRM, portable airborne iodine samplers will be provided in sufficient quantities to sample all vital areas. This equipment will be used for routine and emergency conditions as required to supplement permanently installed process radiation monitoring equipment. Plant personnel will be trained in the use of this equipment under both routine and emergency conditions.

A counting room will be located on the 592'-10" elevation of the General Services Building. This low-background counting room will house a Multi-Channel Analyzer capable of accurately measuring iodine, noble gases, and other radioactive material. 19

PSO RESPONSE: 10 CFR 50.34(e)(2)(xxvii)

The above BFS equipment and facilities will be provided in accordance with the guidelines of NUREG-0737, "Clarification of TMI Action Plan Requirements," Item III.D.3.3.

10 CFR 50.34(e)(2)(xxviii) CONTROL ROOM HABITABILITY

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(xxviii) Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in a TID 14844 source term release, and make necessary design provisions to preclude such problems. (NUREG-0718, III.D.3.4)

PSO RESPONSE:

Introduction

Following the TMI-2 accident, it was determined that the following factors contributed to the control room contamination: (a) lack of adequate control room access control, (b) access by contaminated personnel, (c) doors that were left open, and (d) the inability to monitor accurately the control room atmosphere in the recirculation mode. As a result of the NRC Staff review of the TMI-2 accident, the NRC Staff has developed the requirement for construction permit applicants to evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions, and make necessary design provisions to preclude such problems.

Resolution of Specific TMI-2 Problems

1. Lack of Adequate Control Room Access Control

   Control room access control will be limited by administrative procedures, and will be enhanced due to the dedicated Technical Support Center (TSC) and the on-site Operations Support Center (OSC) provided at BFS. The TSC and OSC will provide work areas for additional operating personnel during accident conditions, thereby reducing the congestion in the control room.

2. Access By Contaminated Personnel

   A combination of radiation monitoring devices and administrative procedures will be provided to minimize the potential for contamination of the control room by contaminated personnel.

3. Control Room Doors Left Open

The double entry doors into the control room will be designed to prevent an open path for contaminated air to enter the control room.

4. Inability to Monitor Accurately the Control Room Atmosphere

The BFS control room atmosphere will be monitored continuously by a three-stage particulate, iodine and noble gas radiation monitor during accident conditions.

Description of the BFS Control Room Habitability System

1. Normal Operation

During normal operations, one of two full-capacity air handling units will maintain the control room habitability area environment within the design basis envelope by supplying conditioned air to the habitability area. Outside air, in a quantity sufficient to meet ventilation requirements, shall be supplied through the operating air handling units to the habitability area.

2. Emergency Operation

The control room is supplied with two redundant emergency air cleanup units. The emergency air cleanup units will not operate during normal conditions with the exception of short periods for surveillance and inspection purposes. The selected emergency air cleanup units and associated isolation and control dampers will be automatically initiated if any of the following conditions occur:

- Incidence of a LOCA (indicated by high drywell pressure or low reactor water level).
- Indication of high radiation levels at the normal outside air intake.
- Indication of chlorine gas at the normal outside air intake.
- Indication of smoke at the normal outside air intake.

If the selected emergency air cleanup unit fan fails to maintain air flow following the automatic start of the unit, then the standby air cleanup unit and associated equipment will be automatically energized.

PSO RESPONSE: 10 CFR 50.34(e)(2)(xxviii)

Summary

The BFS control room habitability design concept has not changed from that presented in the PSAR. The design was previously reviewed against Regulatory Guides 1.78 and 1.95 and Standard Review Plans 2.2 and 6.4 by the NRC and found acceptable (Reference Black Fox Station SER, NUREG-0190, June, 1977 and Supplement No. 2, March, 1979.)

The BFS control room habitability design has been re-reviewed by PSO and found to be in conformance with the following:

1. Standard Review Plans 2.2.1-2.2.2: "Identification of Potential Hazards in Site Vicinity."

2. Standard Review Plan 2.2.3: "Evaluation of Potential Accidents."

3. Standard Review Plan 6.4: "Habitability Systems."

4. Regulatory Guide 1.78: "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."

5. Regulatory Guide 1.95: "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release."

6. K. G. Murphy and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criteria 19," 13th AEB Air Cleaning Conference, August, 1974.

A review of the control room shielding will be performed as described in PSO's response to Requirement (2)(vii).

10 CFR 50.34(e)(3)(i) PROCEDURES FOR FEEDBACK OF OPERATING, DESIGN AND CONSTRUCTION EXPERIENCE

NRC POSITION:

(3) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy 10 CFR 50.34(a)(1) or to address the applicant's technical qualifications and management structure and competence. (NUREG-0718, Category 5).

(i) Provide administrative procedures for evaluating operating, design and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant. (NUREG-0718, I.C.5).

PSO RESPONSE:

Introduction

Prior to and following the accident at Three Mile Island-Unit 2, the Office of Nuclear Reactor Regulation (NRR) of the Nuclear Regulatory Commission (NRC) saw the need for substantially improving the feedback of industry operating experience to individual licensee staffs for information pertinent to the operation of their plant. The primary document arising from post-TMI requirements was NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident." In short, this document stated that licensee procedures should effectively be reviewed and revised as necessary to assure that important operating experience both within and outside the organization was continually provided to operators and other personnel. In addition, the procedures should assure that high priority items received prompt attention while keeping operating personnel from being deluged with paper or instructions on less important matters which could be detrimental to their over-all job proficiency. This feedback of operating experience was applicable to operators of reactors and applicants for an operating license.

In the continuing evolution of this requirement, NUREG-0737, "Clarification of TMI Action Plan Requirements" delineated the required content of the licensee procedures. NUREG-0737, Item I.C.5, stipulated that this requirement was applicable to operators of reactors and applicants for an operating license.

The final Staff revision to this requirement and to which this response is addressed, is embodied in NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License." In NUREG-0718, Staff concerns regarding procedures for feedback of design and construction experience in addition to operating experience are incorporated. As a result, this requirement is now applicable to construction permit applicants.

PSO RESPONSE: 10 CFR 50.34(e)(3)(i)

The Black Fox Station (BFS) project procedures will have provisions for the evaluation and feedback of industry operating, design and construction experience to the personnel involved in the design, construction and operation of BFS.

Organizational Responsibilities

Specific documents containing design, construction and operating experience, as defined by project procedures, are directed to the BFS Nuclear Licensing organization. The Manager, Nuclear Licensing is responsible for implementation of the PSO program that feeds back 19
industry experience into the design, construction and operation of BFS. The Manager, Nuclear Licensing or his designee will initially screen the information to discard information clearly not applicable to BFS and direct the remainder to the appropriate BFS manager for further disposition. If the information is applicable, the manager will notify the Manager, Nuclear Licensing of his recommended action for resolution. If the recommended action reflects a change from information submitted in previous licensing documents, the Manager, Nuclear Licensing, or his designee will ensure the change is reflected in future submittals to the NRC.

Administration and Review of Information

1. General

   As part of its responsibilities, GE has, within its Nuclear Services Department, established and maintained a formal service advisory communication system that is designed to provide the BWR Owner-Operator with a broad coverage of BWR operating and maintenance information and recommendations. In addition, GE routinely reviews other available industry experience for applicability to the equipment and services it supplies to BWR Owner-Operators. Similarly, Black and Veatch and PSO review available industry experience for applicability to the design and other services provided to PSO for the BFS.

2. Public Service Company of Oklahoma

   a. Program Description

      PSO functions within the review process for operating design 19
      and construction experience to:

      1) Review information obtained from industry feedback and determine recommendations for applicability to BFS,

      2) furnish to principal contractors information uniquely available to PSO,

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3) provide direction to project staff and principal contractors for incorporating and implementing operating, design and construction experience into the BFS design and construction,

4) provide direction to the PSO project staff for incorporating and implementing operating experience into training and procedures for BFS, and

5) audit and monitor the principal contractors' implementation of their programs.

Operating, design, and construction experience information enters the BFS nuclear project review from two general categories; regulatory agencies, and industry sources. Examples of both categories include:

1) Regulatory Agency Information:

- USNRC Regulatory Guides
- USNRC Inspection and Enforcement Bulletins, Circulars and Information Notices
- Standard Review Plans (including Branch Technical Positions)

2) Industry Information:

- Reports from GE (Service Information Letters (SILs), and Application Information Documents (AIDs))
- Reports from the Nuclear Safety Analysis Center (NSAC) and the Institute of Nuclear Power Operations (INPO)
- NSAC/INPO Significant Events Evaluation Information Network (SEE-IN)
- Other industry experience from participation in various industry groups, e.g., EEI, AIF, owners groups, etc.

As external information from the above general categories is received, it is directed to the Nuclear Licensing organization for review. Their review is three-fold in purpose:

1) To reduce the quantity of information received to manageable amounts by discarding information clearly non-relevant to BFS, and

2) To separate the information into licensing, operations, design or construction disciplines of concern. Operational information is transmitted to the Manager, Black Fox Station; design information to the Manager, BFS Engineering; and construction information to the Manager, BFS Construction. If the received information may impact more than one discipline, the information will be transmitted to all of the affected parties.

3) Eliminate conflicting or contradictory information.

The review of operating, design or construction information has been or will be addressed in the following project procedures:

1) Review of IE Bulletins, Circulars and Information Notices--defines responsibilities and establishes systematic review of IE documents.

2) Review of and Commitment to Regulatory Guides--defines responsibilities and establishes guidelines for systematic review and evaluation of Regulatory Guides and Draft Regulatory Guides.

3) Review of Industry Experience--provides mechanism for review of operating, design and construction experience from various industry sources that might be applicable to BFS.

PSO will continually monitor the timeliness and sufficiency of the review process for the BFS by utilizing an in-house tracking system. In accordance with formal procedures, PSO will continually monitor the timeliness and sufficiency of the review process for BFS. Commitments resulting from this review process are incorporated into PSO's Licensing Commitment 19
Tracking System (LCTS). The LCTS provides a computerized data base to aid in assuring compliance with these commitments.

b. Operating Experience

Operating experience information is routed by the BFS Nuclear Licensing Organization to the Manager, BFS, or his designee for review. In addition, the Manager, BFS will be privy to a continuous flow of information resulting from PSO's participation in owners groups and other formal and informal contacts.

The Manager, BFS will determine if the information is applicable to BFS and if the disposition needs to be pursued with the principal contractors or the applicable vendors. A primary source of information is the INPO/NSAC "SEE-IN" program of which PSO is a subscriber.

A short time after the accident at Three Mile Island the Nuclear Safety Analysis Center (NSAC), with the support of its utility advisory group, began developing a program to improve the means by which the benefits of shared nuclear plant experience are attained. In early 1980, shortly after its formation, the Institute of Nuclear Power Operations (INPO) joined NSAC in the development and implementation of the program. The program has been named the "Signficant Event Evaluation and Information Network" (SEE-IN). It is a network in the sense that it involves NSAC, INPO, the nuclear utilities, the nuclear steam supply system vendors, and appropriate contractor support.

The objective of SEE-IN is to provide a high degree of assurance that the cumulative learning process from operating experience works well, and that the lessons learned are reported in a timely manner to improve both plant safety and availability. This objective is met by systematically screening all available nuclear plant event information, identifying and evaluating the important or significant events and communicating the results to the utilities and applicable contractors.

The principal organizations involved in the initial screening of plant event data are the utilities, NSAC, and INPO. It is essential for these organizations (i.e., the utilities, NSAC, and INPO) to interface and supplement each other in the screening process for maximum efficiency to be realized.

The primary data used as input to the screening process are Licensee Event Reports (LERs) and outage reports. Both of these report types are submitted in accordance with Nuclear Regulatory Commission (NRC) requirements. The objective of the SEE-IN screening process is to identify those plant events which are most likely to justify further action on the part of the utilities to upgrade nuclear safety or reduce financial risk. Events which become candidates for action analysis (i.e., products of the screening) will be termed "significant". INPO and NSAC have designed a Licensee Event Report Tracking System (LERTS) to track the status of screening and any followup action on all LERs. Once a significant event has been identified from the screening process, it will undergo an action analysis. The purpose of the action analysis is to investigate the event in some detail and develop and evaluate practical remedies. It may be discovered that no further action is required or that it is only necessary to make certain organizations aware of the event.

For those events requiring further action, the results of the action analysis will be communicated to the utility. In some instances, recommendations will be made for mitigating the underlying problems. The recommendations will be made for consideration purposes only.

The Manager, Nuclear Licensing or his designee will receive the SEE-IN recommendations and evaluate the information for applicability to BFS. The eventual result of the reviews will be implemented as appropriate by PSO.

c. Design Experience

For design experience, BFS Engineering has primary responsibility for resolving concerns once the information is received from the Nuclear Licensing organization. The Manager, BFS Engineering will review the information and direct it to the responsible engineer for a determination of the necessary action. The responsible engineer may consult with either Black and Veatch, GE, or the supplying vendor to evaluate the concern. The Manager, Nuclear Licensing, will be apprised of any design changes to determine if they require revision of licensing documents.

d. Construction Experience

The Manager, BFS Construction, has primary responsibility for resolving construction concerns once the information is received from the Nuclear Licensing organization. The Manager, BFS Construction, or his designee may request assistance from the Manager, BFS Engineering, or Black and Veatch as appropriate to resolve construction concerns. The Manager, Nuclear Licensing, will be apprised of any construction changes to determine if they require revision of licensing documents.

3. General Electric

a. Advisory Service

The GE-Nuclear Services Department maintains a service advisory communication system that is designed to provide the BWR Owner-Operator with a broad coverage of BWR operating and maintenance information and recommendations. This system, implemented by the Service Information Letter (SIL), is designed to collect, process, and disseminate information pertinent to:

1) unique operating conditions and experiences
2) improved methods, techniques and procedures for operating and maintaining BWR plant equipment

3) plant performance improvement and equipment upgrading
4) safety, licensing and other regulatory matters.

The major sources of information, including data, drawings, equipment, catalog/part numbers, problem definition, technical work recommendations, and other technical material required to prepare SILs include:

1) Application Information Documents (AIDs)
2) Field Engineering Memos (FEMs)
3) Product Experience Reports (PERs)
4) Safety and Licensing Reports
5) Reports and Instructions prepared by GE Engineering organizations
6) GE and Vendor Equipment Instruction Manuals
7) Equipment Failure and Reliability Reports
8) BWR Plant Owner-Operator(s) and utility management suggestions
9) Startup and Preoperational Test Reports

Occasionally, a need may arise to transmit to the utility owners with operating BWRs an urgent announcement of a potential operational situation which could adversely impact plant operations. In general, such announcements will consist of a complete explanation of the situation with advice or precautionary measures to be observed.

Prior to release from GE-Nuclear Services Department, SILs will undergo formal review by responsible design engineers, other cognizant engineers and GE management representing various disciplines including engineering, startup tests, licensing, and services.

PSO has found the GE feedback program acceptable. | 19

b. NRC Information

Information received from the Nuclear Regulatory Commission falls into the following categories:

1) I&E Bulletins, Circulars and Information
2) NUREGs, Regulatory Guides and SRPs

I&E Documents are received by one individual within the GE licensing department, who reviews and routes it to the proper unit within the department. In turn, that particular unit will review and communicate with each project to which that information may be applicable.

NUREGs, Regulatory Guides and SRPs are received directly from the NRC distribution list by the following organizations within the GE licensing department:

1) Standardization
2) Operating Reactor Service
3) BWR Project Licensing
4) BWR System Licensing
5) Washington Liaison Office

Each organization reviews the documents received and disseminates the data to the proper individual within the licensing organization. At that time all Project Managers are made aware of the information if their project is affected.

c. Field Information

Within the General Electric Nuclear Division, all systems are assigned a Lead System Engineer with the prime responsibility for that particular system. If at any time a problem is encountered in the field by PSO, Black and Veatch or GE field representatives, GE personnel will write a field deviation disposition report (FDDR) describing in detail the problems encountered. At the same time, that report may suggest a solution which is transmitted back to the GE Lead System Engineer in San Jose. That particular Lead Engineer will review the FDDR for its application. If it is a generic problem, an Engineering Change Authorization (ECA) will be written for review and approval. If the ECA is approved, then an Engineering Change Notice (ECN) will be issued to all projects to correct the problem. If the Lead System Engineer finds that the problem is only applicable to a certain project, the same procedure described above will take place but only the specific project management will be notified.

4. Black and Veatch

a. Operations and Design Experience

Black and Veatch BFS project personnel have a responsibility to identify and resolve design and operational feedback concerns for the BFS Project. Sources utilized for feedback include:

1) NRC Inspection and Enforcement Bulletins, Circulars and Notices
2) INPO/NSAC Significant Operating Experience Reports
3) Various Internal Black & Veatch Sources
4) Various External Sources

PSO RESPONSE: 10 CFR 50.34(e)(3)(i)

NRC Inspection and Enforcement Documents are received and reviewed in accordance with established project instructions to determine applicability to and impact upon BFS design, construction and/or operation by the Licensing and Project Design Engineers (PDE). PSO has reviewed and approved the Black & Veatch program for feedback of operating experience 19
into the design of BFS.

Examples of other documents providing operating and design experience feedback that may be routed directly to the PDE for review and identification of concerns, if applicable, include:

1) Licensee and Vendor Inspection Status Reports issued by the NRC.
2) Power Reactor Events, issued by the NRC.
3) Nuclear Power Experience, issued by Nuclear Power Experience, Inc.
4) Atomic Energy Clearing House information
5) EPRI Reports
6) General Electric experience feedback from other BWR's
7) Information obtained from various committees such as the AIF Reactor Licensing & Safety Committee
8) NUREG Documents
9) NRC Generic Letters

Liaison with other nuclear projects, which Black and Veatch is involved in, is maintained by project management. Contacts with utilities, vendors and other engineering firms are also maintained and provide valuable experience feedback. Feedback applicable to BFS is routed to the appropriate PDE or PSO for resolution.

Significant experience feedback, if applicable, is incorporated into design criteria documents such as System Analyses Reports, the Project Design Manual and System Design Specifications. These engineering documents are utilized in developing detailed design documents. Likewise, experience feedback, if applicable, is incorporated into System Descriptions which are utilized in preparing the preoperational, startup and operating procedures for the Black Fox Station.

Most items applicable to the project will be resolved in the design process and not require a change to one of the documents described above. All items requiring a change are submitted to PSO for review and concurrence using either a design change request, design change document or letter correspondence in accordance with documented procedures in the BFS Project Instructions.

PSO RESPONSE: 10 CFR 50.34(e)(3)(i)

b. Construction Experience

Black and Veatch has a dedicated project constructability team which interfaces directly with the BFS site through the Black and Veatch Site Liaison organization and PSO construction management. The constructability team obtains construction experience data through reports from the field, review of I&E Bulletins, Circulars and Information Notices, and review of construction practices at similar sites. The significant experience data obtained from these sources is communicated to the Black and Veatch PDE's, Black and Veatch Site Liaison and PSO Construction Management in the form of letter reports to alert both design and construction personnel to potential problems that may be encountered during the construction phase.

Avoidance of Extraneous and Unimportant Information and Conflicting or Contradictory Information

PSO's Nuclear Licensing organization will, through its screening process, remove extraneous or unimportant information to reduce the unnecessary distraction of Project personnel. The Nuclear Licensing organization will also assure that potentially conflicting or contradictory information is identified and resolved if possible before transmitting to the appropriate organization for review.

Practical Interim Audits

PSO will monitor compliance with these requirements by conducting periodic audits of PSO, Black and Veatch, and General Electric activities in accordance with project procedures.

10 CFR 50.34(e)(3)(ii) EXPAND QA LIST

NRC POSITION:

(3) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy 10 CFR 50.34(a)(1) or to address the applicant's technical qualifications and management structure and competence. (NUREG-0718, Category 5)

(ii) Ensure that the quality assurance (QA) list required by Criterion II, Appendix B, 10 CFR 50 includes all structures, systems, and components important to safety. (NUREG-0718, I.F.1)

PSO RESPONSE:

Introduction

During the investigations of the Three Mile Island Unit 2 accident, the NRC determined that the Quality Assurance List for the power plant did not include many of the items that had contributed to the significance of the accident. As a part of the lessons learned from Three Mile Island, the NRC has established the requirement for applicants to ensure that methods and procedures are in place to assure that all structures, systems, and components of a nuclear power plant are appropriately classified in accordance with their importance to safety, and that appropriate quality assurance actions are taken to assure implementation of the associated criteria. The expanded Quality Assurance List to meet this requirement provides a listing of structures, systems, and components that are important to safety, including items specifically safety related or items that contribute to or whose failure could affect the proper functioning of safety related items.

Description of Program

The Q List for BFS has been designated as an Essential Items List (E-List). This E-List actually consists of several separate, but complementary lists. The BFS Project Design Manual includes a listing of BFS systems and structures important to safety, identified by name and acronym. The Project Design Manual also contains major design criteria unique to the Black Fox site and the NSSS. This Project Design Manual is prepared, maintained, and approved by Black & Veatch. In addition, PSO reviews and concurs with all revisions of the Project Design Manual.

The portion of the E-List contained in the Project Design Manual consists of a listing of all major components within the systems and structures of BFS that are important to safety and consequently subject to the provisions of the Quality Assurance Program. For items that fall within the GE scope of work, inputs to the E-List are implemented based on GE recommendations. This portion of the E-List was reviewed and approved by the B&V Manager, Design and reviewed and concurred with by the PSO Manager, BFS Engineering in accordance with the procedures governing the preparation, review, and approval of the Project Design Manual. Changes to the Project Design Manual are reviewed and approved in a similar manner.

In addition to the above listing, several other lists are prepared, reviewed, and approved by Black & Veatch to complete the entire definition of the E-List. These additional listings are as follows:

- A detailed listing of all major equipment important to safety within the B&V scope of work is maintained as a computer data base and contains the unique identification number for each piece of equipment as well as the defined safety class, quality group, seismic category and electrical classification for the equipment, as appropriate.

- Several detailed listings of all components important to safety within the B&V scope of work, sorted by type of component (such as valves, pipelines, electrical devices, etc.) are also developed, reviewed and approved by B&V. These listings contain the unique identification number and defined safety class, quality group, seismic category, and electrical classification as appropriate for the type of equipment.

E-List Development

The Project Design Manual listing is the initial development stage of the E-List. The Project Design Manual contains the listing of the structures and systems important to safety. The major components of those systems are identified. The preparation of the E-List as a portion of the Project Design Manual is controlled by procedures contained within the Black & Veatch Quality Assurance Program which cover the development, review, approval, and issue control. The primary criteria utilized by personnel at Black & Veatch to determine the importance to safety for the structures, systems, and major components of BFS come from the following documents:

- 10 CFR 50 Appendix A, "General Design Criteria for Nuclear Plants."

- Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants."

- Regulatory Guide 1.29, "Seismic Design Classification."

- ANSI N212, "Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants."

PSO RESPONSE: 10 CFR 50.34(e)(3)(ii)

- GESSAR, "General Electric Standard Safety Analysis Report."

ANSI N212 defines a graded approach for assigning numerical safety classifications, as appropriate to the level of importance to safety, for all types of equipment. Regulatory Guide 1.26 defines a further alphabetic classification scheme for fluid containing components to relate to application of industry standards, as appropriate to the level of importance to safety. Regulatory Guide 1.29 provides two classification categories, seismic and non-seismic, as appropriate for structures and components, as related to their importance to safety.

The various classification schemes contained within the primary criteria documents listed above were grouped into 15 possible combinations. The resulting system provides a method for defining the level of importance to safety of the individual components.

In addition, the inclusion of specific designations on the E-List for quality assurance applicability, safety class, quality group, seismic category, electrical classification and other appropriate information provides definition of the design, procurement, fabrication, and construction requirements to the personnel involved in these activities.

Following the development of the major systems and structures E-List (in the Project Design Manual), the development of the detailed parts of the Essential Items List was initiated based on the development of a System Design Specification (SDS) for each system identified in the PDM. The SDS was developed to define and document each system function, the exact system boundaries, the interfaces with other systems, the NSSS/BOP design interfaces, the regulatory criteria applicable to the design of the system, the industry codes and standards applicable to the system, and the special functions of each component within the system with respect to its importance to safety (both as a primary function and/or its relationship to other systems and components important to safety). The SDS is prepared and approved by Black & Veatch in accordance with B&V Quality Assurance Program procedures. Each SDS is reviewed and approved by the B&V Manager, Design. PSO reviews and concurs with the SDS for applicability to Black Fox Station in accordance with procedures in the PSO Quality Assurance Program. The SDS serves to define all portions of the system that are important to safety and provides the criteria to be used for each portion of the system for assigning classifications (safety class, quality group, seismic category, and electrical classification, as appropriate) to the individual components within the system. The detailed parts of the E-List are prepared and approved as final design is accomplished. The SDS is used as a basis to ensure that all components are entered on the respective E-List prior to their procurement. The development, identification of authorized personnel to approve the E-List and changes thereto, and the distribution of the approved parts of the E-List are controlled in accordance with procedures contained within the respective PSO and B&V Quality Assurance Programs.

Continuing Development

During the design and licensing process for BFS, various additional regulatory positions and requirements have been defined which affect the quality requirements for structures, systems, and components. These have included such items as quality assurance requirements for fire protection system, branch technical position APCSB 9.5-1, quality assurance requirements for radiological waste systems defined in branch technical position ETSB 11-1, and additionally, specialized requirements defined for feedwater and main steam lines covered by RSB 3-2. Such additional requirements have been incorporated in the E-List for BFS.

The BFS E-List will be periodically updated, reviewed, and approved to maintain it current with the design for BFS. Particular attention will be directed to any new structures, systems or components to be included in the design of BFS as a result of lessons learned from TMI. As such items are defined in the licensing and detailed design process, appropriate classifications and entries on the Essential Items List will be completed.

PSO Review

For purposes of this Requirement, PSO will review the Q List used on BFS by comparing the structures, systems and components of the Q List with a systems analysis study. The discussion of this systems analysis study follows.

The systems analysis is performed to provide a systematic classification of components by examining plant events by frequency of occurrence, radiological impacts, and allowable limits of the safety criteria.

The systems analysis is constructed by first defining categories of plant operation and potential events in each plant operating category. The events are ordered by frequency of occurrence. Unacceptable safety criteria are established according to the expected frequency of occurrence.

For planned (normal) operation, the unacceptable results criteria are:

- Release of radioactive material to the environs exceeding the limits of either 10 CFR 20 or 10 CFR 50.

- Fuel failure to such an extent that if the freed fission products were released to the environs via the normal discharge paths for radioactive material, the limits of 10 CFR 20 would be exceeded.

PSO RESPONSE: 10 CFR 50.34(e)(3)(ii)

- Nuclear system stresses exceeeding those allowed for planned operation by applicable industry codes.

- Existence of a plant condition not considered by plant safety analysis.

For anticipated (expected) operational transients with calculated probabilities of occurrence of once per day to once in 20 years, the unacceptable results criteria are:

- Release of radioactive material to the environs exceeding the limits of 10 CFR 20.

- Any fuel failure calculated as a direct result of the transient analyses.

- Nuclear system stresses exceeding those allowed for transients by applicable industry codes.

- Containment stresses exceeding those allowed for transients by applicable industry codes when containment is required.

For abnormal (unexpected) operational transients with calculated probabilities of occurrence of less than one event in 20 years to one in 100 years, the unacceptable results criteria are:

- Radioactive material release exceeding the guideline values of a small fraction of 10 CFR 100.

- Failure of the fuel barrier as a result of exceeding mechanical or thermal limits (failure means gross core-wide fuel cladding perforations).

- Nuclear system stresses exceeding those allowed for transients by applicable industry codes.

- Containment stresses exceeding that allowed for accidents by applicable industry codes when containment is required.

For design basis (postulated) accidents, i.e., events with low probability of occurrence (once in 100 years to once in 10,000 years), the unacceptable results criteria are:

- Radioactive material release exceeding the guidelines values of 10 CFR [illegible].

- Failure of the fuel barrier as a result of exceeding mechanical or thermal limits. Failure includes fuel cladding fragmentation (Loss of Coolant Accident) and excessive fuel enthalphy (control rod drop accident).

- Nuclear system stresses exceeding those allowed for accidents by applicable industry codes.
- Containment stresses exceeding those allowed for accidents by applicable industry codes when containment is required.
- Plant main control room personnel overexposure to radiation.

Nuclear safety operational requirements are diagrammed for each event to obtain minimum acceptable results and identify those systems required to function. The systems required to function become, by definition, systems important to safety. By inspection of Protection Sequence Diagrams (described below) those systems required to function will be determined and the requirements for satisfaction of single failure criteria observed.

Four operating states are identified in order to establish initial conditions of each protection system sequence analysis. The four states are: (1) reactor shutdown, vessel head off (2) reactor not shutdown, vessel head off (3) reactor shutdown, vessel head on, and (4) reactor not shutdown and vessel head is on. For each state, required safety actions are defined to assure adequate control. For example, in state (4) the required safety actions are as follows:

- Radioactive material release control
- Core coolant flow rate control
- Core power level control
- Core neutron flux distribution control
- Reactor vessel water level control
- Reactor vessel pressure control
- Nuclear system temperature control
- Nuclear system water quality control
- Nuclear system leakage control
- Core reactivity control
- Control rod worth control
- Containment and Reactor/Auxiliary Building pressure and temperature control
- Stored fuel shielding, cooling, and reactivity control

Planned operations for each operating state are identified and safety action sequences are diagrammed to demonstrate system requirements. The six planned operations are: refueling, achieving criticality, reactor heat up, power operation, achieving reactor shutdown, and reactor cooldown. In addition to planned operations, anticipated operational transients, design basis accidents, and special events are defined for each operating state and planned operating condition.

For each event, protection sequences are diagrammed to show acceptable success paths including consideration of single active component failure and single operator error conditions. From these diagrams, safety-related system requirements are determined.

PSO RESPONSE: 10 CFR 50.34(e)(3)(ii)

Those additional systems or components identified as safety-related from the results of the systems analysis diagrams, will be added to the Q List as described above. Then, for any such additions, the QA Program will be applied to all subsequent system design, procurement, construction and operation activities.

Conclusion

It is concluded, based on the foregoing, that procedures have been and will be established as necessary to ensure that the Q List mandated by Criterion II of Appendix B to 10 CFR Part 50 and this Requirement includes all structures, systems and components important to safety.

10 CFR 50.34(e)(3)(iii) DEVELOP MORE DETAILED QA CRITERIA

NRC POSITION:

(3) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy 10 CFR 50.34(a)(1) or to address the applicant's technical qualifications and management structure and competence. (NUREG-0718, Category 5)

(iii) Establish a quality assurance (QA) program based on consideration of:

(A) Ensuring independence of the organization performing checking functions from the organization responsible for performing the functions.

(B) Performing the entire quality assurance/quality control function at the construction site.

(C) Including QA personnel in (the review and concurrence) of quality related procedures associated with design, construction and installation.

(D) Establishing criteria for determining QA requirements for specific classes of equipment.

(E) Establishing minimum qualification requirements for QA and QC personnel;

(F) Sizing the QA staff commensurate with its duties, responsibilities, and importance to safety.

(G) Establishing procedures for maintenance of "as-built" documentation;

(H) Providing a QA role in design and analysis activities. (NUREG-0718, I.F.2)

Guidance related to meeting the criteria of the Requirement was received from the NRC on April 21, 1981, in a document entitled, "Proposed Quality Assurance Guidance to Satisfy NUREG 0718 and Proposed Rule," which is included as an attachment to this response and hereafter called the "Staff's QA Guidance Document."

PSO RESPONSE:

Background

As a result of the accident at TMI-2, the NRC Staff has determined that near-term construction permit applicants should review and evaluate the accident from the standpoint of Quality Assurance (QA) matters to determine whether their respective QA programs should be revised to accommodate the lessons learned from the reviews. In addition, the NRC staff has provided the QA Guidance Document referred to above for the purpose of establishing a framework for satisfying this Requirement.

PSO has conducted an independent review of the TMI-2 accident from the standpoint of QA matters and it has reviewed its QA Program with respect to the matters set forth in the Requirement.

Introduction

PSO's QA program for the Black Fox Station has been reviewed against the requirements of Appendix B to 10 CFR Part 50, and the NRC Staff has determined that the principal elements of the QA Program as described in Chapter 17 of the PSAR meets the requirements of Appendix B. Nevertheless, it is appropriate to restate in broad terms the philosophical underpinnings of PSO's QA program and to describe the actions being taken to constantly improve that program.

PSO is committed to a highly effective and comprehensive QA program for Black Fox Station. PSO, in furtherance of this commitment, has established an independent QA organization that is responsible for developing the QA program and verifying its effectiveness. Towards these ends PSO is qualifying its QA program under Section III of the American Society of Mechanical Engineering (ASME) Boiler and Pressure Vessel Code with the objective of obtaining the ASME "N" Stamp Certification. Moreover, PSO recently completed a Management Review of the Black Fox Station project.

Two aspects of the QA program are discussed in the Management Review report with recommendations for the consideration of PSO management. One recommendation suggests including the Quality Control (QC) in the organizational structure of the QA organization for the Black Fox project. Under the present structure, the QC group reports to the Manager, BFS Construction to assure timely integration of QC activities in the construction effort. The QA organization presently maintains control over the QC group by approving the qualifications of QC personnel and QC procedures, and by auditing their activities. The organization of the QC group was thoroughly considered during NRC Staff's review of Chapter 17 of the PSAR and it was concluded (and recognized by the Staff in its Safety Evaluation Report) that, for the Black Fox project, the present structure is acceptable. Nevertheless, PSO, as a part of its remobilization after the issuance of the construction permit, will reconsider this matter at the highest management level in accordance with the recommendation of the Management Review report.

The second recommendation of the report suggests that communication within the Black Fox QA organization requires improvement. This recommendation was immediately followed-up with the QA organization to assure that channels of communication were clearly established and opened. An executive review of this effort was conducted with the Manager, QA.

The results of PSO's independent review of the QA aspects of the TMI-2 accident and the eight QA criteria of the Requirement as defined for acceptance in the Staff's QA Guidance Document are discussed below. The detailed implementation of these criteria are provided for in Table 17A.1-5 of Chapter 17 of the PSAR.

New commitments or changes to the existing PSO QA program resulting from this review are highlighted by underlining in the response. The response is organized in the following by repeating the NRC guidance statement, followed by a description of the PSO QA program that relates to the guidance.

A. Independence of Organization Performing Checking Functions

NRC Acceptance Criterion 2A1

Verification of conformance to established requirements is accomplished by individuals or groups within the QA organization who do not have direct responsibility for performing the work being verified. Rationale and justification must be provided if performed by other than the QA organization.

PSO Response to Criterion 2A1

Verification of conformance to established QA program requirements is done by the PSO QA organization which is independent from the organizations or groups responsible for performing the work. The Manager, Quality Assurance reports directly to the Vice-President of Power Generation who reports to the Executive Vice-President. PSO QA reviews and approves the suppliers' and contractors' QA programs and verifies implementation through audits. PSO QC performs surveillance of contractor installation to assure that the contractor's QC program is performing its function. PSO QC has the responsibility to report quality concerns to the Manager, BFS Nuclear Project. The Manager, BFS Nuclear Project reports directly to the Executive Vice-President.

NRC Acceptance Criterion 2A2

The QA organizational responsibilities for inspection are described. Individuals performing inspections report to the QA organization.

PSO RESPONSE: 10 CFR 50.34(e)(3)(iii)

PSO Response to Criterion 2A2

Off-site suppliers and site contractors are responsible for inspections in accordance with a QA program approved by PSO QA. For suppliers, the PSO QA organization performs audits and surveillances to assure compliance with their programs. For site contractors, PSO QA performs audits and PSO QC performs surveillance to assure compliance with their programs.

NRC Acceptance Criterion 2A3

Verification of suppliers' activities during fabrication, inspection, testing, and shipment of materials, equipment, and components is planned and performed with QA organization participation in accordance with written procedures to assure conformance to the purchase order requirements. These procedures, as applicable to the method of procurement, provide for:

a. Specifying the characteristics or processes to be witnessed, inspected, or verified, and accepted; the method of surveillance and the extent of documentation required; and those responsible for implementing these procedures.

b. Audits, surveillances, or inspections which assure that the supplier complies with the quality requirements.

PSO Response to Criterion 2A3

The PSO QA organization performs surveillance at the source of supply in accordance with procedures to verify that suppliers are meeting procurement document requirements.

a. The PSO QA organization reviews procurement documents to verify inclusion of appropriate witness and hold points such as start of fabrication, initial welding, non-destructive testing, hydrostatic testing and preparation for shipment. PSO QA also reviews to verify that appropriate documents and records are specified to be submitted or retained by suppliers. In addition to witness and hold points included in the specification, the PSO QA organization develops a surveillance plan on each supplier. The plan specifies the characteristics to be witnessed or inspected, the method of surveillance and the documentation required. The PSO QA organization implements the source surveillance procedures and plans.

b. As stated in "a" above, the PSO QA organization performs surveillance in accordance with procedures to verify the supplier complies with procurement document requirements. QA also approves the suppliers' QA programs and conducts audits of suppliers to assure that they comply with their Quality Assurance Programs.

PSO RESPONSE: 10 CFR 50.34(e)(3)(iii)

NRC Acceptance Criterion 2A4

Receiving inspection is performed by the QA organization to assure:

a. The material, component, or equipment is properly identified and corresponds to the identification on the purchase document and the receiving documentation.

b. Material, components, equipment, and acceptance records satisfy the inspection instructions prior to installation or use.

c. Specified instruction, test and other records, (such as certificates of conformance attesting that the material, components, and equipment conform to specified requirements) are available at the nuclear power plant prior to installation or use.

PSO Response to Criterion 2A4

Receipt inspection is performed by the PSO Quality Control organization. The receipt inspections plans, procedures and results are approved by PSO QA. The plans and procedures assure:

a. Material, component or equipment is identified in accordance with procurement document requirements and is traceable to the receiving documentation.

b. Material, components, equipment, and acceptance records satisfy the inspection instructions prior to installation or use. The PSO QA organization verifies that the QA records are acceptable prior to installation or use of the material or equipment.

c. Specified inspection, test, and other records, (such as certificates of conformance attesting that the material, components, and equipment conform to specified requirements) are available at the nuclear power plant prior to installation or use.

NRC Acceptance Criterion 2A5

Correct identification of material, parts, and components is verified and documented by the QA organization prior to release for fabrication, assembling, shipping, and installation.

PSO Response to Criterion 2A5

PSO QA requires the supplier's QA program to provide for correct identification of material, parts, and components prior to release for fabrication, assembling and shipping. Verification of implementation of the requirements is done by PSO QA through audits and surveillance. PSO QC verifies at receipt inspection correct identification of material, parts, and components. PSO QA reviews PSO QC receipt inspection checklists prior to release for installation.

NRC Acceptance Criterion 2A6

Procedures are established for recording evidence of acceptable accomplishment of special processes using qualified procedures, equipment, and personnel. The QA organization verifies the recorded evidence and documents the results.

PSO Response to Criterion 2A6

Procurement documents instruct the supplier or contractor to submit special process procedures for approval. For suppliers this submittal is to General Electric, the NSSS supplier, and Black & Veatch, the Architect Engineer. For site contractors this submittal is to PSO QA. Through surveillance and audits, PSO QA verifies that supplier special process procedures have been approved, that equipment and personnel are qualified for the process, and that the process is done in accordance with the approved procedures. For site contractors, audits are performed by PSO QA and surveillance by PSO QC. The results of these functions are documented.

NRC Acceptance Criterion 2A7

Inspection and test results are documented, evaluated, and their acceptability determined by a responsible individual or group. The QA organization, as a minimum, evaluates, verifies, and documents completeness of this activity.

PSO Response to Criterion 2A7

The suppliers' and site contractors' QA programs, which are reviewed and approved by PSO QA, require a responsible individual or group to document, evaluate, and determine acceptability of inspection and test results. The PSO QA organization, through source surveillance, evaluates and verifies that suppliers are accomplishing this activity. The PSO QC organization, through surveillance, evaluates and verifies that site contractors are accomplishing this activity. These surveillances are documented.

NRC Acceptance Criterion 2A8

Follow-up action is taken by the QA organization to verify proper implementation of corrective action and to close out the corrective action in a timely manner.

PSO RESPONSE: 10 CFR 50.34(e)(3)(iii)

PSO Response to Criterion 2A8

The PSO QA organization documents QA program deviations on a Correction Action Report (CAR) which states the proposed corrective action and specifies a date for completion. PSO QA personnel maintain logs to track the completion of corrective action of each CAR. Proper implementation and timeliness of the corrective action and its effectiveness is evaluated. Black & Veatch (B&V), General Electric (GE), suppliers, and contractors are required to have a corrective action system in their respective QA programs. PSO QA verifies proper implementation of these corrective action systems through audits.

B. Performing QA/QC Functions at Construction Site

NRC Acceptance Criterion 2B1

The person at the construction site responsible for directing and managing the site QA program is identified by position. He reports to the offsite QA organization and has appropriate organizational position, responsibilities and authority to exercise proper control over the QA program. This individual is free from non-QA duties and can thus give full attention to assuring that the QA program at the plant site is being effectively implemented.

PSO Response to Criterion 2B1

The Manager, Quality Assurance is responsible for directing and managing the BFS QA Program. He is located at the construction site and reports off site to the Vice-President of Power Generation. He is free of any non-QA duties and is responsible for developing and verifying implementation of the BFS Quality Assurance Program. He has the authority and organizational freedom to identify problems, recommend or provide solutions, and verify implementation of solutions. He has written authority to prevent the continuation of activities which are detrimental to the quality of the plant. The Superintendent, QC is located at the construction site and is responsible for verifying that contractors' QC organizations are performing their functions and that installation activities are done in accordance with specifications. The Superintendent, QC also has written authority to prevent the continuation of activities detrimental to the quality of the plant. He has access to the Manager, BFS Nuclear Project to resolve conflicts affecting quality which cannot be resolved at lower management levels.

NRC Acceptance Criterion 2B2

Designated QA individuals are involved in day-to-day plant activities important to safety (i.e., the QA organization routinely attends and participates in daily plant work schedule and status meetings to assure they are kept abreast of day-to-day work assignments throughout the plant and that there is adequate QA coverage relative to procedural and inspection controls, acceptance criteria, and QA staffing and qualification of personnel to carry out QA assignments).

PSO Response to Criterion 2B2

PSO QA and QC personnel are involved in day-to-day plant activities important to safety and are kept abreast of work schedule and construction activities by routinely attending construction status meetings. With information obtained at these meetings PSO QA and QC personnel ensure adequate coverage relative to procedural and inspection controls, acceptance criteria, and QA and QC staffing and qualification of personnel to carry out QA and QC assignments.

C. QA Review and Concurrence of Quality Related Procedures

NRC Acceptance Criterion 2C1

Provisions are established to assure that quality-affecting procedures required to implement the QA program are consistent with QA program commitments and corporate policies and are properly documented, controlled, and made mandatory through a policy statement or equivalent document signed by the responsible official.

PSO Response to Criterion 2C1

Procedures are written to implement the policies in the QA Policy Manual and the commitments in Chapter 17 of the PSAR. Procedures are reviewed by QA personnel to assure that they adequately address PSO policies and QA program commitments and to assure consistency. PSO QA reviews and approves all quality-affecting procedures. Each procedure manual contains a policy statement signed by the President of PSO which makes implementation of the procedures mandatory.

NRC Acceptance Criterion 2C2

The QA organization reviews and documents concurrence with these quality-related procedures.

PSO Response to Criterion 2C2

The PSO QA organization reviews and documents approval of each quality-related procedure.

PSO RESPONSE: 10 CFR 50.34(e)(3)(iii)

NRC Acceptance Criterion 2C3

Procedures are established for the review of procurement documents to determine that quality requirements are correctly stated, inspectable, and controllable; there are adequate acceptance and rejection criteria; and procurement documents have been prepared, reviewed, and approved in accordance with QA program requirements. To the extent necessary, procurement documents should require contractors and subcontractors to provide an acceptable quality assurance program. The review and documented concurrence of the adequacy of quality requirements stated in procurement documents is performed by QA personnel.

PSO Response to Criterion 2C3

Procedures are established for the review of procurement documents by PSO QA to determine that quality requirements are correctly stated, inspectable, and controllable and that appropriate acceptance and rejection criteria are included. PSO QA verifies that procurement documents have been prepared, reviewed, and approved in accordance with QA program requirements. Procurement documents, to the extent necessary, require contractors to provide an acceptable QA program and to pass this requirement on to their subcontractors. The PSO QA organization reviews and documents that bid specifications, bid proposals and procurement documents contain adequate quality requirements.

NRC Acceptance Criterion 2C4

Procedures for the review, approval, and issuance of documents and changes thereto are established and described to assure technical adequacy and inclusion of appropriate quality requirements prior to implementation. The QA organization reviews and documents concurrences with these documents with regards to QA-related aspects.

PSO Response to Criterion 2C4

Procedures have been established to control the review, approval, and issuance of documents and changes thereto. PSO Engineering reviews the documents as a surveillance and monitoring function for assuring technical adequacy. PSO QA personnel review the documents to verify inclusion of appropriate quality requirements and to assure that the documents have been reviewed and approved in accordance with established procedures prior to implementation. 19

NRC Acceptance Criterion 2C5

Inspection procedures, instructions, or checklists provide for the following as reviewed and concurred with by the QA organization for QA aspects and other technical organizations, as appropriate:

PSO RESPONSE: 10 CFR 50.34(e)(3)(iii)

a. Identification of characteristics and activities to be inspected.

b. A description of the method of inspection.

c. Identification of the individuals or groups responsible for performing the inspection operation in accordance with the provisions of Item 10B1.

d. Acceptance and rejection criteria.

e. Identification of required procedures, drawings, and specifications and revisions.

f. Recording inspector or data recorder and the results of the inspection operation.

g. Specifying necessary measuring and test equipment including accuracy.

PSO Response to Criterion 2C5

Procurement documents require suppliers and site contractors to have inspection procedures, instructions, or checklists and submit these for review and concurrence. For suppliers this submittal is to B&V. For site contractors this submittal is to PSO for a technical and a QA review. PSO QA conducts audits and surveillance to verify the procedures, instructions, or checklists have been reviewed and concurred with by the appropriate organization. The procurement documents contain requirements, as appropriate, for:

a. Identification of characteristics and activities to be inspected.

b. A description of the method of inspection.

c. Identification of the individuals or groups responsible for performing the inspection operation in accordance with the provisions of Item 10B1.

d. Acceptance and rejection criteria.

e. Identification of required procedures, drawings, and specifications and revisions.

f. Recording inspector or data recorder and the results of the inspection operation.

g. Specifying necessary measuring and test equipment including accuracy.

PSO RESPONSE: 10 CFR 50.34(e)(3)(iii)

NRC Acceptance Criterion 2C6

Test procedures or instructions provide for the following as reviewed and concurred with by the QA organization for QA aspects and by other technical organizations for technical aspects.

a. The requirements and acceptance limits contained in applicable design and procurement documents.

b. Instruction for performing the test.

c. Test prerequisites such as calibrated instrumentation, adequate test equipment and instrumentation including their accuracy requirements, completeness of item to be tested, suitable and controlled environmental conditions, and provisions for data collection and storage.

d. Mandatory inspection hold points for witness by owner, contractor, or inspector (as required).

e. Acceptance and rejection criteria.

f. Methods of documenting or recording test data and results.

g. Provisions for assuring test prerequisites have been met.

PSO Response to Criterion 2C6

Procurement documents require suppliers and site contractors to have test procedures or instructions and submit these for review and concurrence. For suppliers this submittal is to B&V. For site contractors this submittal is to PSO for a technical and a QA review. This review will verify that the procedures or instructions address the following:

a. The requirements and acceptance limits contained in applicable design and procurement documents.

b. Instruction for performing the test.

c. Test prerequisites such as calibrated instrumentation, adequate test equipment and instrumentation including their accuracy requirements, completeness of item to be tested, suitable and controlled environmental conditions, and provisions for data collection and storage.

d. Mandatory inspection hold points for witness by owner, contractor, or inspector (as required).

e. Acceptance and rejection criteria.

f. Methods of documenting or recording test data and results.

g. Provisions for assuring test prerequisites have been met.

NRC Acceptance Criterion 2C7

Procedures are established and described for calibration (technique and frequency), maintenance, and control of measuring and test equipment (instruments, tools, gauges, fixtures, reference and transfer standards, and non-destructive test equipment) that is used in the measurement, inspection, and monitoring of structures, systems, and components. The review and documented concurrence of these procedures is described and the organization responsible for these functions is identified.

PSO Response to Criterion 2C7

Suppliers and site contractors are required by procurement documents to have procedures for calibration, maintenance, and control of measuring and test equipment; including fixtures, reference standards, and non-destructive testing equipment. PSO QA will audit suppliers, site contractors, and calibration service suppliers to verify implementation of calibration procedures.

NRC Acceptance Criterion 2C8

Procedures are established and described to control the cleaning, handling, storage, packaging, and shipping of materials, components, and systems in accordance with design and procurement requirements to preclude damage, loss, or deterioration by environmental conditions such as temperature or humidity. The QA organization reviews and documents concurrence of these procedures.

PSO Response to Criterion 2C8

Suppliers will be required to have procedures for cleaning, handling, storage, packaging, and shipping of material and equipment in accordance with criteria specified in procurement documents. These procedures are required to be submitted to B&V, GE, or PSO for concurrence. Cleaning, handling, and storage procedures for use on the site will be developed by BFS Engineering based on procurement specifications and the suppliers' specifications and instructions, and PSO QA will review and document concurrence of these procedures.

NRC Acceptance Criterion 2C9

Procedures are established to indicate the inspection, test, and operating status of structures, systems, and components throughout fabrication, installation, and test. The QA organization reviews and documents concurrence with these procedures.

PSO RESPONSE: 10 CFR 50.34(e)(3)(iii)

PSO Response to Criterion 2C9

PSO QA reviews and approves suppliers' QA programs, which require procedures to indicate inspection, test and operating status of structures, systems, and components throughout fabrication. PSO QA performs audits and surveillance of suppliers to verify compliance.

PSO QA reviews and approves site contractors' QA programs, which require procedures to indicate inspection, test and operating status of structures, systems and components throughout installation and test. These procedures are reviewed and approved by PSO QA.

NRC Acceptance Criterion 2C10

Procedures are established and described to control the application and removal of inspection and welding stamps and status indicators such as tags, markings, labels, and stamps. The QA organization reviews and documents concurrence with these procedures.

PSO Response to Criterion 2C10

The PSO QA program has procedures which describe methods to control the application and removal of status indicators, such as tags, markings, labels and stamps. Site contractors will be required to use the PSO tagging system. Tags applied by the PSO QC organization can be removed only by PSO QC. The use of welder identification stamps will be controlled by each contractor in accordance with procedures or plans approved by PSO QA.

NRC Acceptance Criterion 2C11

Procedures are established and described to control altering the sequence of required tests, inspections, and other operations important to safety. Such actions should be subject to the same controls as the original review and approval. The QA organization reviews and documents concurrence with these procedures.

PSO Response to Criterion 2C11

The PSO QA organization reviews and approves suppliers' and contractors' QA manuals. Part of the review is to verify that changes to inspection and test procedures or specifications are reviewed and approved by the same organization that approved the original document. PSO QA reviews inspection plans, and results during surveillance, and audits to verify proper implementation.

PSO RESPONSE: 10 CFR 50.34(e)(3)(iii)

NRC Acceptance Criterion 2C12

Procedures are established and described for identification, documentation, segregation, review, disposition, and notification to affected organizations of nonconforming materials, parts, components and as applicable to services (including computer codes) if disposition is other than to scrap. The procedures provide identification of authorized individuals for independent review of nonconformances, including disposition and closeout.

NRC Acceptance Criterion 2C13

QA and other organizational responsibilities are described for the definition and implementation of activities related to nonconformance control. This includes identifying those individuals or groups with authority for the disposition of nonconforming items and involvement of the QA organization in documenting concurrence to the disposition, satisfactory completion of the disposition, and corrective action.

PSO Response to Criteria 2C12 and 2C13

PSO procurement documents require suppliers and site contractors to have established procedures to control the identification, documentation, segregation, review, disposition and notification of nonconforming items. Suppliers and site contractors will be required to document nonconformances which provide identification of individuals who initiate, provide disposition, approve, and close-out the nonconformance.

For site contractors PSO QA will approve the disposition of all nonconformances affecting safety-related equipment and other equipment deemed important to safety. Nonconformances on items which are dispositioned use-as-is or repair also require approval of the responsible designer of the item. Verification of close-out of nonconformances is done by PSO QC personnel and the final closed out report is sent to PSO QA for verification that it meets QA record requirements.

Off-site suppliers are required to have a system for the control of nonconformances as part of their QA program which is approved by PSO QA. Nonconformances on items which are dispositioned use-as-is or repair require approval by B&V and by PSO QA. Implementation of nonconformance control procedures will be verified through surveillance and audits by PSO QA.

NRC Acceptance Criterion 2C14

Procedures are established and described indicating an effective corrective action program has been established. The QA organization reviews and documents concurrence with the procedures.

PSO RESPONSE: 10 CFR 50.34(e)(3)(iii)

PSO Response to Criterion 2C14

PSO QA is responsible for the development and implementation of a corrective action program. Procedures have been established by PSO QA which provide for identification of QA program deviations, recommended corrective action, proposed disposition and verification of close-out. These procedures provide for corrective action to prevent recurrence. PSO requires suppliers and contractors to have a corrective action program and verifies compliance by audit.

D. Criterion for Requirements for Specific Classes of Equipment

NRC Acceptance Criterion 2D1

The QA organization and the necessary technical organizations participate early in the QA program definition stage to determine and identify the extent QA controls are to be applied to specific structures, systems, and components. This effort involves applying a defined graded approach to certain structures, systems, and components in accordance with their importance to safety and affects such disciplines as design, procurement, document control, inspection tests, special processes, records, audits and other described in 10 CFR 50 Appendix B.

PSO Response to Criterion 2D1

The PSO QA program includes provisions to assure that the PSO QA organization and the necessary BFS technical organizations participate early in the QA program definition stage to determine and identify the extent QA controls are to be applied to specific structures, systems, and components. This effort involves applying a defined graded approach to certain structures, systems, and components in accordance with their importance to safety and affects such disciplines as design, procurement, document control, inspection tests, special processes, records, audits and others described in 10 CFR 50 Appendix B.

NRC Acceptance Criterion 2D2

For commercial "off-the-shelf" items where specific quality assurance controls appropriate for nuclear applications cannot be imposed in a practicable manner, special quality verification requirements shall be established and described to provide the necessary assurance of an acceptable item by the purchaser.

PSO Response to Criterion 2D2

For commercial "off-the-shelf" items where specific quality assurance controls appropriate for nuclear applications cannot be imposed in a practicable manner, special quality verification requirements shall be established and described to provide assurance of acceptability of the item by PSO QA. Factors which will be considered in approving a supplier are an acceptable QA program, approval by the Coordinating Agency for Supplier Evaluation (CASE), a pre-award survey, or prior history. The "off-the-shelf" items will be receipt inspected as described in 2A4.

NRC Acceptance Criterion 2D3

The scope of the inspection program is described that indicates an effective inspection program has been established. Program procedures provide criteria for determining the accuracy requirements of inspection equipment and criteria for determining when inspections are required or define how and when inspections are performed. The QA organization participates in the above functions.

PSO Response to Criterion 2D3

PSO QA reviews and approves suppliers' and site contractors' QA programs. These programs are required to have procedures which provide criteria for determining the accuracy requirements of inspection equipment and criteria for determining when inspections are required or to define how and when inspections are performed. PSO QA performs audits and surveillance to verify compliance by suppliers. PSO QA audits and PSO QC performs surveillance of site contractors to assure compliance.

NRC Acceptance Criterion 2D4

Procedures are established and described with the involvement of the QA organization to identify, in pertinent documents, mandatory inspection hold points beyond which work may not proceed until inspected by a designated inspector.

PSO Response to Criterion 2D4

PSO QA reviews and approves the suppliers' and site contractors' QA program which will establish and describe the involvement of their QA organization to identify mandatory inspection hold points beyond which work may not proceed until inspected by a designated inspector. PSO QA shall audit the suppliers and site contractors for compliance.

NRC Acceptance Criterion 2D5

The description of the scope of the test control program indicates an effective test program has been established for tests including proof tests prior to installation and preoperational tests. Program procedures provide criteria for determining the accuracy requirements of test equipment and criteria for determining when a test is required or how and when testing activities are performed.

PSO Response to Criterion 2D5

Test procedures are prepared by the responsible organization and reviewed by QA as specified in paragraph 2C6. Proof tests prior to installation are delineated in procurement documents and will be performed by contractors. Proof tests on all safety-related systems and on systems important to safety are designated as mandatory hold points and will be witnessed by QC personnel. QA will review the results of the proof tests prior to acceptance of the item for preoperational testing. Preparation, review and approval of preoperational tests are coordinated by the Test Working Group which determines how, when and by whom testing activities are performed. The requirement for testing and the accuracy requirements for test equipment will be provided in test procedures approved by the Test Working Group.

NRC Acceptance Criterion 2D6

Audit data are analyzed by the QA organization and the resulting reports indicating any quality problems and the effectiveness of the QA program, including the need for reaudit of deficient areas, are reported to management for review and assessment.

PSO Response to Criterion 2D6

Audits are conducted and the results analyzed by the PSO QA organization. Audit reports indicate any quality problems and the effectiveness of the QA program. Reaudits of deficient areas are conducted as necessary. Audit reports are provided to management for review and assessment.

E. Qualification Requirements for QA and QC Personnel

NRC Acceptance Criterion 2E1

Indoctrination, training, and qualification programs are established such that:

a. Personnel responsible for performing quality-affecting activities are instructed as to the purpose, scope, and implementation of the quality-related manuals, instructions, and procedures.

b. Personnel verifying activities affecting quality are trained and qualified in the principles, techniques, and requirements of the activity being performed.

PSO RESPONSE: 10 CFR 50.34(e)(3)(iii)

c. For formal training and qualification programs, documentation includes the objective, content of the program, attendees, and date of attendance.

d. Proficiency tests are given to those personnel performing and verifying activities affecting quality, and acceptance criteria are developed to determine if individuals are properly trained and qualified.

e. Certificates of qualifications clearly delineates (a) the specific functions personnel are qualified to perform and (b) the criteria used to qualify personnel in each function.

f. Proficiency of personnel performing and verifying activities affecting quality is maintained by retraining, reexamining, and/or recertifying as determined by management or program commitment.

g. The description of the training program provisions listed above satisfies the regulatory position in Regulatory Guide 1.58, Rev. 1.

PSO Response to Criterion 2E1

The PSO QA program includes provisions for the establishment of an indoctrination, training, and qualification program to assure the following:

a. Personnel responsible for performing activities affecting quality are indoctrinated in the purpose, scope, and implementation of instructions and procedures they use to accomplish these activities.

b. Personnel who perform audits, source surveillance, site contractor surveillance, receipt inspection and other verification activities are trained and qualified in the principles, techniques, and requirements of the activity being performed.

c. PSO has established a nuclear training organization who assures that formal training and qualification documentation includes the objective, content, attendees, and date of attendance.

d. Proficiency tests to evaluate initial capability will be given to lead auditors in accordance with Reg. Guide 1.146 and to QC inspection personnel in accordance with Reg. Guide 1.58, Rev. 1. The initial capability of personnel who witness inspections or tests (source surveillance) will be determined by an evaluation of their education, experience, and training, and by test results or capability demonstration as provided in ANSI N45.2.6 Certification procedures include acceptance criteria for qualification or reference applicable codes and standards which state the criteria.

e. Certificates of qualification will state what activities the individual is qualified to perform and the basis used for the qualification. PSO QA will issue the certifications for PSO Quality Assurance and Quality Control personnel.

f. Proficiency of personnel performing and verifying activities affecting quality is maintained by retraining, reexamining, and/or recertifying. The method used is delineated in training procedures.

g. The training program will comply with the requirements in ANSI N45.2.6 as amended or clarified by Reg. Guide 1.58, Rev. 1.

Those contractors which are responsible for performing quality-affecting activities will be required to have a training program. PSO QA will verify this through audits or surveillance.

NRC Acceptance Criterion 2E2

A qualification program for inspectors (including NDT personnel) is established under direction of the QA organization and documented, and the qualifications and certifications of inspectors are kept current.

PSO Response to Criterion 2E2

A qualification program for PSO QC inspection personnel has been established and approved by PSO QA. Each individual certification is reviewed and approved by PSO QA to verify compliance with ANSI N.45.2.6 and applicable regulatory guides. PSO QA performs periodic audits to assure that training and qualification files are kept current. The PSO QA organization tests and certifies NDT personnel in accordance with the American Society for Nondestructive Testing, Recommend Practice No. SNT-TC-1A.

F. Sizing of QA Staff

NRC Acceptance Criterion 2F1

Organization charts identify the "onsite" and "offsite" organizational elements which function under the cognizance of the QA program (such as design engineering, porcurement, manufacturing, construction, inspection, test, instrumentation and control, nuclear enigneering, etc.), the lines of responsibility, and a description of the criteria for determining the size of the QA organization including the inspection staff.

PSO RESPONSE: 10 CFR 50.34(e)(3)(iii)

PSO Response to Criterion 2F1

Organization charts (PSAR, Chapter 17) identify the offsite and onsite organizations which function under the cognizance of the QA program and illustrate the lines of functional and administrative reporting. The staffing levels of the PSO QA and QC organizations were determined by listing activities to be performed; such as audits, surveillance, procedure reviews, and inspections, and assigning an estimated duration time and manpower needed for each activity. The manpower requirement per activity was based on experience and is periodically updated based on actual time it took to accomplish specified activities.

NRC Acceptance Criterion 2F2

The QA organization is involved in establishing long range project work schedules and staffing of QA and QC personnel and evaluates these periodically (i.e., monthly) to assure they are valid or if necessary, modify staffing level.

PSO Response to Criterion 2F2

In conjunction with manpower estimates, the PSO QA organization maintains long range schedules based on BFS Project schedules to determine when personnel are needed and what type of QA expertise is needed. These schedules are re-evaluated periodically depending upon Project requirements. The PSO QC organization's work activities and long range schedules will be reviewed by PSO QA during surveillance and audits to assess adequacy of staffing level.

G. Procedures for Maintenance of "As-Built" Documentation

NRC Acceptance Criterion 2G1

The scope of the document control program is described, and the types of controlled documents are identified. As a minimum, controlled documents include as-built documents.

NRC Acceptance Criterion 2G2

Procedures are established and described to provide for the preparation of as-built drawings and related documentation in a timely manner to accurately reflect the actual plant design.

PSO Response to Criteria 2G1 and 2G2

The PSO QA program and procedures describe the scope of the document control program and includes "as-built" drawings. Procedures have been established to assure provision for the preparation of as-built drawings and related doucmentation in a timely manner to accurately reflect the actual plant design.

H. QA Role in Design and Analysis Activities

NRC Acceptance Criterion 2H1

Procedures are established and described requiring a documented check to verify the dimensional accuracy and completeness of design drawings and specifications.

PSO Response to Criterion 2H1

Design organizations for BFS (GE and B&V) have procedures in accordance with the PSO QA program which requires a documented check of drawings to verify that dimensions are accurate and that the drawing is complete and meets drafting standards. These requirements are established for other design organizations through procurement documents.

NRC Acceptance Criterion 2H2

Procedures are established and described requiring that design drawings and specifications be reviewed by the QA organization or other individuals knowledgeable and qualified in QA/QC techniques to assure that the documents are prepared, reviewed, and approved in accordance with company procedures and that the documents contain the necessaryquality assurance requirements such as inspection and test requirements, acceptance requirements, and the extent of documenting inspection and test results.

PSO Response to Criterion 2H2

Procedures in B&V's QA Program, which have been approved by PSO QA, require that specifications and associated drawings be reviewed by B&V QA to assure that they are prepared, reviewed and approved in accordance with B&V procedures. B&V QA personnel also do a documented review to assure that inspection and test requirements, accept/reject criteria, and documentation requirements are included. PSO QA reviews procurement documents submitted by B&V to verify that access requirements, accept/reject criteria, appropriate witness and hold points, QA records, and other QA requirements are specified. GE has the responsibility of designing and procuring the Nuclear Steam Supply System for BFS in accordance with their QA program which has been approved by PSO QA.

Attachment

(3) (iii) DEVELOP MORE DETAILED QA CRITERIA

Proposed Quality Assurance Guidance to

Satisfy NUREG-0718 and Proposed Rule

As a result of NUREG-0718, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses," and the proposed rule associated with this NUREG, additional QA requirements were identified for pending construction permits and manufacturing license applications to address in their docketed Quality Assurance (QA) program description. In this regard, the QAB has developed the following guidance to determine the acceptability of the improved QA program.

1. Proposed Rule (3)ii — Ensure that the quality assurance (QA) list required by Criterion II, App. B, 10 CFR Part 50 includes all structures, systems, and components important to safety. (I.F.1)

   Acceptance Guidance: The scope of the QA program includes:

   1a (2A1)* — A commitment that activities affecting structures, systems, and components important to safety will be subject to the applicable controls of the QA program and meet Regulatory Guide 1.29 and 10 CFR 50 Appendix A.

   1b (-) — A description of the management plan for determining and identifying those structures, systems, and components that meet Regulatory Guide 1.29 and 10 CFR 50 Appendix A and fall under the control of the docketed QA program description.

   1c (-) — The identification of structures, systems, and components and related consumables covered by the QA program and controlled measures identifying authorized personnel to approve changes to this list and describing methods controlling its distribution.

2. Proposed Rule (3)iii — Establish a quality assurance (QA) program based on consideration of:

   (A) ensuring independence of the organization performing checking functions from the organization responsible for performing the functions;

   Acceptance Guidance: The QA program includes:

   2A1 (1B2) — Verification of conformance to established requirements is accomplished by individuals or groups within the QA organization who do not have direct responsibility for performing the work being verified. Rationale and justification must be provided if performed by other than the QA organization.

   2A2 (10B1) — The QA organizational responsibilities for inspection are described. Individuals performing inspections report to the QA organization.

* These numbers in parentheses correlate with the numbers in the proposed Rev. 2 to SRP Section 17.1.

2A3 (7A2) Verification of suppliers' activities during fabrication, inspection, testing, and shipment of materials, equipment, and components is planned and performed with QA organization participation in accordance with written procedures to assure conformance to the purchase order requirements. These procedures, as applicable to the method of procurement, provide for:

a. Specifying the characteristics or processes to be witnessed, inspected, or verified, and accepted; the method of surveillance and the extent of documentation required; and those responsible for implementing these procedures.

b. Audits, surveillance, or inspections which assure that the supplier complies with the quality requirements.

2A4 (7B1) Receiving inspection is performed by the QA organization to assure:

a. The material, component, or equipment is properly identified and corresponds to the identification on the purchase document and the receiving documentation.

b. Material, components, equipment, and acceptance records satisfy the inspection instructions prior to installation or use.

c. Specified inspection, test and other records, (such as certificates of conformance attesting that the material, components, and equipment conform to specified requirements) are available at the nuclear power plant prior to installation or use.

2A5 (8B3) Correct identification of material, parts, and components is verified and documented by the QA organization prior to release for fabrication, assembling, shipping, and installation.

2A6 (9B2) Procedures are established for recording evidence of acceptable accomplishment of special processes using qualified procedures, equipment, and personnel. The QA organization verifies the recorded evidence and documents the result.

2A7 (10C3) (11C1) Inspection and test results are documented, evaluated, and their acceptability determined by a responsible individual or group. The QA organization as a minimum evaluates, verifies, and documents completeness of this activity.

2A8 (16.3) Followup action is taken by the QA organization to verify proper implementation of corrective action and to close out the corrective action in a timely manner.

Proposed Rule (3)iii	(B)	performing the entire quality assurance/quality control function at construction sites;
Acceptance Guidance:		The QA program provides provisions to assure that:
2B1 (1C3)		The person at the construction site responsible for directing and managing the site QA program is identified by position. He reports to the offsite QA organization and has appropriate organizational position, responsibilities, and authority to exercise proper control over the QA program. This individual is free from non-QA duties and can thus give full attention to assuring that the QA program at the plant site is being effectively implemented.
2B2 (1B6)		Designated QA individuals are involved in day-to-day plant activities important to safety (i.e., the QA organization routinely attends and participates in daily plant work schedule and status meetings to assure they are kept abreast of day-to-day work assignments throughout the plant and that there is adequate QA coverage relative to procedural and inspection controls, acceptance criteria, and QA staffing and qualification of personnel to carry out QA assignments).
Proposed Rule (3)iii	(C)	including QA personnel in (the review and concurrence) of quality-related procedures (and documents) associated with design, construction, and installation;
Acceptance Guidance:		The QA program includes:
2C1 (2B1a)		Provisions are established to assure that quality-affecting procedures required to implement the QA program are consistent with QA program commitments and corporate policies and are properly documented, controlled, and made mandatory through a policy statement or equivalent document signed by the responsible official.
2C2 (2B1b)		The QA organization reviews and documents concurrence with these quality-related procedures.
2C3 (4A1)		Procedures are established for the review of procurement documents to determine that quality requirements are correctly stated, inspectable, and controllable; there are adequate acceptance and rejection criteria; and procurement documents have been prepared, reviewed, and approved in accordance with QA program requirements. To the extent necessary, procurement documents should require contractors and subcontractors to provide an acceptable quality assurance program. The review and documented concurrence of the adequacy of quality requirements stated in procurement documents is performed by QA personnel.

2C4 (6A2) Procedures for the review, approval, and issuance of documents and changes thereto are established and described to assure technical adequacy and inclusion of appropriate quality requirements prior to implementation. The QA organization reviews and documents concurrences with these documents with regards to QA-related aspects.

2C5 (10C1) Inspection procedures, instructions, or checklists provide for the following as reviewed and concurred with by the QA organization for QA aspects and other technical organizations, as appropriate:

a. Identification of characteristics and activities to be inspected.

b. A description of the method of inspection.

c. Identification of the individuals or groups responsible for performing the inspection operation in accordance with the provisions of item 10B1.

d. Acceptance and rejection criteria.

e. Identification of required procedures, drawings, and specifications and revisions.

f. Recording inspector or data recorder and the results of the inspection operation.

g. Specifying necessary measuring and test equipment including accuracy requirements.

2C6 (11B1) Test procedures or instructions provide for the following as reviewed and concurred with by the QA organization for QA aspects and by other technical organizations for technical aspects:

a. The requirements and acceptance limits contained in applicable design and procurement documents.

b. Instructions for performing the test.

c. Test prerequisites such as calibrated instrumentation, adequate test equipment and instrumentation including their accuracy requirements, completeness of item to be tested, suitable and controlled environmental conditions, and provisions for data collection and storage.

d. Mandatory inspection hold points for witness by owner, contractor, or inspector (as required).

e. Acceptance and rejection criteria.

f. Methods of documenting or recording test data and results.

g. Provisions for assuring test prerequisites have been met.

2C7 (12.3) Procedures are established and described for calibration (technique and frequency), maintenance, and control of the measuring and test equipment (instruments, tools, gages, fixtures, reference and transfer standards, and nondestructive test equipment) that is used in the measurement, inspection, and monitoring of structures, systems, and components. The review and documented concurrence of these procedures is described and the organization responsible for these functions is identified.

2C8 (13.2) Procedures are established and described to control the cleaning, handling, storage, packaging, and shipping of materials, components, and systems in accordance with design and procurement requirements to preclude damage, loss, or deterioration by environmental conditions such as temperature or humidity. The QA organization reviews and documents concurrence of these procedures.

2C9 (14.1) (14.4) Procedures are established to indicate the inspection, test, and operating status of structures, systems, and components throughout fabrication, installation, and test. The QA organization reviews and documents concurrence with these procedures.

2C10 (14.2) (14.4) Procedures are established and described to control the application and removal of inspection and welding stamps and status indicators such as tags, markings, labels, and stamps. The QA organization reviews and documents concurrence with these procedures.

2C11 (14.3) (14.4) Procedures are established and described to control altering the sequence of required tests, inspections, and other operations important to safety. Such actions should be subject to the same controls as the original review and approval. The QA organization reviews and documents concurrence with these procedures.

2C12 (15.1) Procedures are established and described for identification, documentation, segregation, review, disposition, and notification to affected organizations of nonconforming materials, parts, components and as applicable to services (including computer codes) if disposition is other than to scrap. The procedures provide identification of authorized individuals for independent review of nonconformances, including disposition and closeout.

2C13 (15.2) QA and other organizational responsibilities are described for the definition and implementation of activities related to nonconformance control. This includes identifying those individuals or groups with authority for the disposition of nonconforming items and involvement of the QA organization in documenting concurrence to the disposition, satisfactory completion of the disposition, and corrective action.

2C14 (16.1) Procedures are established and described indicating an effective corrective action program has been established. The QA organization reviews and documents concurrence with the procedures.

Proposed Rule (3)iii (D) establishing criteria for determining QA requirements for specific classes of equipment;

Acceptance Guidance: The QA program provides provisions to assure that:

2D1 (2B3) The QA organization and the necessary technical organizations participate early in the QA program definition stage to determine and identify the extent QA controls are to be applied to specific structures, systems, and components. This effort involves applying a defined graded approach to certain structures, systems, and components in accordance with their importance to safety and affects such disciplines as design, procurement, document control, inspection, tests, special processes, records, audits and others described in 10 CFR 50 Appendix B.

2D2 (7B4) For commercial "off-the-shelf" items where specific quality assurance controls appropriate for nuclear applications cannot be imposed in a practicable manner, special quality verification requirements shall be established and described to provide the necessary assurance of an acceptable item by the purchaser.

2D3 (10A) The scope of the inspection program is described that indicates an effective inspection program has been established. Program procedures provide criteria for determining the accuracy requirements of inspection equipment and criteria for determining when inspections are required or define how and when inspections are performed. The QA organization participates in the above functions.

2D4 (10C2) Procedures are established and described with the involvement of the QA organization to identify, in pertinent documents, mandatory inspection hold points beyond which work may not proceed until inspected by a designated inspector.

2D5 (11A1) The description of the scope of the test control program indicates an effective test program has been established for tests including proof tests prior to installation and preoperational tests. Program procedures provide criteria for determining the accuracy requirements of test equipment and criteria for determining when a test is required or how and when testing activities are performed.

2D6 (18B1) — Audit data are analyzed by the QA organization and the resulting reports indicating any quality problems and the effectiveness of the QA program, including the need for reaudit of deficient areas, are reported to management for review and assessment.

Proposed Rule (3)iii — (E) establishing minimum qualification requirements for QA and QC personnel;

Acceptance Guidance: — The QA program provides provisions to assure that:

2E1 (2D) — Indoctrination, training, and qualification programs are established such that:

a. Personnel responsible for performing quality-affecting activities are instructed as to the purpose, scope, and implementation of the quality-related manuals, instructions, and procedures.

b. Personnel verifying activities affecting quality are trained and qualified in the principles, techniques, and requirements of the activity being performed.

c. For formal training and qualification programs, documentation includes the objective, content of the program, attendees, and date of attendance.

d. Proficiency tests are given to those personnel performing and verifying activities affecting quality, and acceptance criteria are developed to determine if individuals are properly trained and qualified.

e. Certificate of qualifications clearly delineates (a) the specific functions personnel are qualified to perform and (b) the criteria used to qualify personnel in each function.

f. Proficiency of personnel performing and verifying activities affecting quality is maintained by retraining, reexamining, and/or recertifying as determined by management or program commitment.

g. The description of the training program provisions listed above satisfies the regulatory position in Regulatory Guide 1.58, Rev. 1.

2E2 (10B2) — A qualification program for inspectors (including NDT personnel) is established under direction of the QA organization and documented, and the qualifications and certifications of inspectors are kept current.

Proposed Rule (3)iii — (F) sizing the QA staff commensurate with its duties, responsibilities, and importance to safety.

Acceptance Guidance: — The QA program provides provisions to assure that:

2F1 (1A5) — Organization charts identify the "onsite" and "offsite" organizational elements which function under the cognizance of the QA program (such as design engineering, procurement, manufacturing, construction, inspection, test, instrumentation and control, nuclear engineering, etc.), the lines of responsibility, and a description of the criteria for determining the size of the QA organization including the inspection staff.

2F2 (-) — The QA organization is involved in establishing long range projected work schedules and staffing of QA and QC personnel and evaluates these periodically (i.e., monthly) to assure they are valid or if necessary modify staffing level.

Proposed Rule (3)iii — (G) establishing procedures for maintenance of "as-built" documentation;

Acceptance Guidance: — The QA program provides provisions to assure that:

2G1 (6A1) — The scope of the document control program is described, and the types of controlled documents are identified. As a minimum, controlled documents include:

As-built documents.

2G2 (6C1) — Procedures are established and described to provide for the preparation of as-built drawings and related documentation in a timely manner to accurately reflect the actual plant design.

Proposed Rule (3)iii — (H) providing a QA role in design and analysis activities.

Acceptance Guidance: — The QA program provides provisions to assure that:

2H1 (3E1) — Procedures are established and described requiring a documented check to verify the dimensional accuracy and completeness of design drawings and specifications.

2H2 (3E2) Procedures are established and described requiring that design drawings and specifications be reviewed by the QA organization or other individuals knowledgeable and qualified in QA/QC techniques to assure that the documents are prepared, reviewed, and approved in accordance with company procedures and that the documents contain the necessary quality assurance requirements such as inspection and test requirements, acceptance requirements, and the extent of documenting inspection and test results.

3. Proposed Rule (3)vii Provide a description of the management plan for design and construction activities, to include: (a) the organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (b) technical resources directed by the applicant; (c) details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (d) proposed procedures for handling the transition to operation; (e) the degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort. (II.J.3.1)

Acceptance Guidance: Fred Allenspach is primary reviewer for this item; however, QAB should assure that sufficient information has been provided either in Fred's section or the QA program to satisfy the following:

3-1a (-) The role and attitude of top management towards QA should be described including: (a) the extent of their involvement in emphasizing and aggressively supporting the QA program as a highly important fundamental tool in assuring the plant is designed and constructed correctly; (b) techniques in conveying the importance of implementing the QA program to all managers, supervisors, technicians, formen, craft personnel, and others performing quality affecting activities and in emphasizing that the working staff and the QA and QC organization is a team cooperative effort; (c) the extent top management keeps informed of major problems and assures timely investigations and resolution of these problems; and (d) the extent top management regularly meets with the QA organization to determine the status and adequacy of the QA program and design and construction activities.

3-1b (-) Utility management should establish a strong discipline QA management organization staffed with well qualified individuals knowledgeable in QA/QC principles with sufficient authority and responsibilities to carry out the QA/QC functions.

3-2 (1A1) The responsibility for the overall program is retained and exercised by the applicant.

3-3 (1A2) The applicant has identified and described major delegation of work involved in establishing and implementing the QA program or any part thereof to other organizations.

3-4 (1A3) When major portions of the applicant's program are delegated:

a. Applicant describes how responsibility is exercised for the overall program. The extent of management oversight should be addressed including the location, qualifications, and criteria for determining the number of personnel performing these functions.

b. Applicant evaluates the performance (frequency and method stated - once per year although longer cycle acceptable with other evaluations of individual elements) of work by the delegated organization.

c. Qualified individual(s) or organizational element(s) are identified within the applicant's organization as responsible for the quality of the delegated work prior to initiation of activities.

3-5 (1A4) Clear management controls and effective lines of communication exist for QA activities among the applicant and the principal contractors to assure direction of the QA program.

3-6 (1A6) The applicant (and principal contractors) describes the QA responsibilities of each of the organizational elements noted on the organization charts.

3-7 (1B1) The applicant (and principal contractors) identifies a position that retains overall authority and responsibility for the QA program (normally, this position is QA Manager) and this position has the following characteristics:

a. Is at the same or higher organization level as the highest line manager directly responsible for performing activities affecting quality (such as engineering, procurement, construction, and operation) and is sufficiently independent from cost and schedule.

b. Has effective communication channels with other senior management positions.

c. Has responsibility for approval of QA Manual(s).

d. Has no other duties or responsibilities unrelated to QA that could prevent his full attention to QA matters.

3-8 (1B3) Persons and organizations performing QA functions have direct access to management levels which will assure the ability to:

a. Identify quality problems.

b. Initiate, recommend, or provide solutions through designated channels.

c. Verify implementation of solutions.

3-9 (1B4) a. Designated QA personnel, sufficiently free from direct pressures for cost/schedule, have the responsibility delineated in writing to stop unsatisfactory work and control further processing, delivery, or installation of nonconforming material.

b. The organizational positions with stop work authority are identified.

3-10 (1B5) Provisions are established for the resolution of disputes involving quality, arising from a difference of opinion between QA personnel and other department (engineering, procurement, manufacturing, etc.) personnel.

3-11 (1C1) Policies regarding the implementation of the QA program are documented and made mandatory. These policies are established at the Corporate President or Vice President level.

3-12 (1C2) Position description assures that the individual directly responsible for the definition, direction, and effectiveness of the overall QA program has sufficient authority to effectively implement responsibilities. This position is to be sufficiently free from cost and schedule responsibilities. Qualification requirements for this individual are established in a position description which includes the following prerequisites:

a. Management experience through assignments to responsible positions.

b. Knowledge of QA regulations, policies, practices, and standards.

c. Experience working in QA or related activity in reactor design, construction, or operation or in a similar high technological industry.

The qualification of the QA Manager should be at least equivalent to those described in Section 4.4.5 of ANSI/ANS-3.1-1978, "Selection and Training of Nuclear Power Plant Personnel," as endorsed by the regulatory positions in Regulatory Guide 1.8.

3-13 (2A2) A brief summary of the company's corporate QA policies is given.

10 CFR 50.34(e)(3)(iv) DEGRADED CORE - DEDICATED CONTAINMENT PENETRATION

NRC POSITION:

(3) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy 10CFR50.34(a)(1) or to address the applicant's technical qualifications and management structure and competence. (NUREG-0718, Category 5)

(iv) Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system. (NUREG-0718, II.B.8(2))

PSO RESPONSE:

The NRC has indicated, by notice in the Federal Register (45 FR 193, page 65474, October 2, 1980), its intent to conduct a rulemaking concerning measures to deal with degraded core conditions. One measure, the installation of a filtered vented containment system, will be examined to determine whether significant risk reduction would accrue from its inclusion in reactor design. The requirement for one or more dedicated containment penetrations (equivalent in size to a single 3-foot diameter opening) is being imposed on construction permit applications to avoid possible foreclosure of this system by construction should it ultimately be determined in the NRC rulemaking to require the installation of a filtered vented containment system.

Provisions will be made for including an additional single 3-foot diameter penetration in the BFS design. The provisions will consist of a capped 42 inch diameter sleeve in the containment vessel and a sealed 48 inch diameter sleeve in the Shield Building. These sleeves will be oriented radially at approximately elevation 629'-0" and have an azimuth angle of approximately 125 degrees. If required, space is available in the containment for an inboard isolation valve and in the Fuel Building for an outboard isolation valve. These sleeves and the associated space will be dedicated for compliance with this requirement.

To assure that the containment penetration will satisfy the establishment of an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the design conditions important to safety are not exceeded for as long as postulated accident conditions require, the design requirements for penetrations as specified in Section 3.8 of the BFS PSAR will be followed. The containment penetration will be designed and constructed in accordance with the ASME Boiler and Pressure Vessel Code, 1974 Edition, with Addenda through Summer 1976, Section III, Subsection NE, Class MC Components, including the quality assurance requirements of Article NA-4000, and inspection requirements of Article NA-5000.

10 CFR 50.34(e)(3)(vi) DEDICATED CONTAINMENT PENETRATION

NRC POSITION:

(3) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy 10 CFR 50.34(a)(1) or to address the applicant's technical qualifications and management structure and competence. (NUREG-0718, Category 5)

(vi) For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere. (NUREG-0718, II.E.4.1)

PSO RESPONSE:

This requirement stems from the concern that, for plants without hydrogen recombiners, should the need arise to connect external hydrogen recombiners to the containment following an accident, the process of connecting those hydrogen recombiners not be complicated by the potential involvement of other external systems. This requirement does not apply to Black Fox Station since the BFS design includes fully redundant hydrogen recombiners permanently located within the containment. Hence there is no need to establish a dedicated containment penetration for operation.

10 CFR 50.34(e)(3)(vii) ORGANIZATION AND STAFFING TO OVERSEE DESIGN AND CONSTRUCTION

NRC POSITION:

(3) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy 10 CFR 50.34(a)(1) or to address the applicant's technical qualifications and management structure and competence. (NUREG-0718, Category 5)

(vii) Provide a description of the management plan for design and construction activities, to include: (A) the organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources directed by the applicant; (C) details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) proposed procedures for handling the transition to operation; (E) the degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort. (NUREG-0718, II.J.3.1)

PSO RESPONSE:

INTRODUCTION

In the aftermath of TMI-2, a number of studies and investigations, including those of the President's Commission on Three Mile Island, the NRC Special Inquiry Group, the NRC Staff Lessons Learned Task Force, and the Atomic Industrial Forum, concluded that improvements were necessary in the organization and management of activities relating both to the operation and to the design and construction of nuclear power plants. NRC Staff reviews have resulted in various documents setting forth new requirements, including NUREG's-0578, 0585, 0626, 0660, and 0737. All of these studies and documents called for upgrading in certain areas of management oversight and technical competence in nuclear activities. The application of these requirements for better management and increased oversight to construction permit applicants has been incorporated in NUREG-0718 and in the Rule for the Near-Term Construction Permit and Manufacturing License.

General plans for PSO's first nuclear facility were initiated in the early 1970's, culminating with the Black Fox Station (BFS) Project announcement in January, 1973. Site preparation was begun under the authority of a Limited Work Authorization (LWA) issued July 26, 1978, by the Nuclear Regulatory Commission (NRC). PSO expected to have a full construction permit issued by the NRC in July of 1979, after the closing of the public health and safety hearing record before the Atomic Safety and Licensing Board in February, 1979. However, because of the Three Mile Island-Unit 2 (TMI-2) accident on March 28, 1979, the NRC suspended all licensing activity while it conducted investigations. As a result of the licensing moratorium, PSO put the Project into a holding status by suspending hiring, suspending or cancelling selected contracts, and reducing existing staff. The Project will be fully reactivated upon receipt of the construction permits for Black Fox Station.

The following discussion sets forth PSO's response to this Requirement.

A. ORGANIZATION AND MANAGEMENT STRUCTURE

1. THE COMPANY AND THE PROJECT OWNERSHIP

Ownership in Black Fox Station is shared by three participants. Public Service Company of Oklahoma (PSO), a wholly owned subsidiary of Central and South West Corporation, has an ownership interest of approximately 61%. Western Farmers Electric Cooperative (Anadarko, Oklahoma) has an ownership interest of approximately 17% and Associated Electric Cooperative, Inc. (Springfield, Missouri) owns approximately 22%. Pursuant to the ownership agreement among PSO, Western Farmers Electric Cooperative, and Associated Electric Cooperative, Inc., PSO has been designated the Project Manager and as such exercises sole control and management of the design, construction and operation of the plant. Western Farmers Electric Cooperative and Associated Electric Cooperative, Inc. conduct budget reviews and expenditure audits for the Project from time to time. Similarly, Central and South West's overview of PSO's activities is limited to budget and cash flow review and construction management audits.

2. PSO UPPER LEVEL MANAGEMENT ORGANIZATION

PSO is headed by the Board of Directors and the President, who is the chief executive officer of the company. Reporting to the President are the heads of each functional group of the company who are:

- Executive Vice President
- Senior Vice President, Finance
- Vice President, Division Operations

Each functional group is further subdivided into divisions and departments. The corporate functional organization as it relates to the Black Fox Project is depicted in Figure (3)(vii)-1.

PSO RESPONSE: 10 CFR 50.34(e)(3)(vii)

The President has delegated broad authority for the conduct of the Project to the Executive Vice President. Under this general authority, the Executive Vice President establishes the policies and administrative controls necessary to assure proper design, procurement, construction, and safe operation of all Company power plants, including Black Fox Station. Because of the size and importance of this Project to the Company, the Executive Vice President has retained a more direct personal involvement in the detailed execution of the functions relating to the design, procurement, construction, and operation of the Black Fox Station than for a fossil fueled electric generating station. This involvement is evidenced by such activities as the monthly one-on-one meetings between the Executive Vice President and the Manager, Quality Assurance; the participation of the Executive Vice President as a member of BFS procurement review boards; involvement in the periodic management review 19
meetings with General Electric and Black & Veatch; membership on the BFS Review and Audit Commmittee; and frequent personal contact between the Executive Vice President and BFS Project managers. The Executive Vice President established the Project management framework also shown in Figure (3)(vii)-1 in order to carry out the management of the Project. Reporting to him are the following subordinate managers:

The Manager, BFS Nuclear Project, who has no responsibilities outside the Project, is responsible to the Executive Vice President for all design and construction phase activities These responsibilities include the development of capabilities within the company to control the design and construction phases of the Black Fox Station Project and the coordination, scheduling and construction of BFS from inception to completion of the facilities. The Manager, BFS Nuclear Project has the financial and managerial authority of a company executive staff member on the vice-presidential level. He is a regular 19
participant in the chief executive officer's staff meetings.

His responsibilities further include the requirements to coordinate the efforts of all internal PSO and external Project organizations and to enforce compliance with the Black Fox Quality Assurance Program. He administers the contract with Black & Veatch Consulting Engineers. The Manager, Planning, Scheduling, and Cost Control; Manager, BFS Engineering; Manager, BFS Construction; and Manager, BFS Materiel and Administration report to him. The Manager, BFS Nuclear Project 19
has consolidated his organization at the construction site.

The Vice President, Power Systems Engineering is responsible to the Executive Vice President for substation, transmission, and distribution engineering for the entire PSO system. He is responsible for the design of the BFS Substation and the connecting transmission network.

The Vice President, Power Generation is responsible to the Executive Vice President for the safe, efficient, and economical operation of each generation facility in the PSO system. He is responsible for the establishment of programs to safeguard the facilities, station personnel, and the public while maintaining the operational capabilities of the facilities. With respect to the Black Fox Station Project, the Executive Vice President has delegated to him responsibility for ensuring corporate implementation of the comprehensive BFS Quality Assurance Program, and for representing PSO before regulatory agencies with respect to quality assurance and environmental and nuclear licensing matters. The BFS Station Manager; the Manager, Environmental and Chemistry Control; the Manager, Quality Assurance; and the Manager, Nuclear Licensing report to him.

The Manager, Corporate Information Resources is responsible to the Executive Vice President for corporate records management functions and for corporate computer support. He is also responsible for maintaining the BFS Project document security file.

The Vice President, Materiel and Property Management is responsible to the Executive Vice President for preparation of procurement documents, including purchase orders and contracts, for the acquisition of goods, services, and property for the Company and for the BFS Project. The Manager, Nuclear Fuel, and the Manager, Materiel report to him.

The Manager, Personnel Resources is responsible to the Executive Vice President for management of all corporate personnel functions. He provides personnel services support to the BFS Project.

3. THE MANAGEMENT APPROACH

PSO's approach to the Black Fox Station Project differed from that of many utilities approaching the construction of a nuclear power station. As a result of extensive study of nuclear construction management, PSO decided to adopt the approach of performing its own construction management for the Black Fox Station Project. On the basis of past experience, PSO believed that direct utility management of construction offered the Company an opportunity to alleviate many of the difficulties with controlling cost, maintaining schedule, and assuring quality. Under the system adopted, the Company acts as its own construction manager using multiple specialty contractors instead of dealing with a combination architect/engineer-constructor firm or a separate large constructor firm. In this system, PSO also controls all purchasing, from the bidding process through the life of the contract, and supervises scheduling and execution of all contractor work.

PSO has had excellent experience with this approach in the past. PSO has over 60 years' experience in the construction and operation of fossil fueled electric generating stations, having retained independent engineers to design its generating stations under the close supervision of Company engineers. This supervision consisted of design surveillance to assure that the Engineer implemented those design features proven in operation to provide safe, reliable, and economic operation and maintenance. This design surveillance involvement enabled PSO to thoroughly understand new and developing technologies and to avoid the designing-in of operational and maintenance problems.

PSO has also performed its own construction management for the construction of fossil units. Construction management consisted of direct on-site PSO supervision of individually awarded construction contracts combined with PSO purchasing of materials through letting of procurement packages from specifications prepared by the Engineer (and reviewed by PSO). This system enabled PSO to construct fossil units on schedule and at costs among the lowest in the industry. The overall Company involvement in station design and construction for these units resulted in constructing generating units that performed with a remarkable minimum of unplanned outages and in establishing a greater level of skill in operating personnel as a result of experience with construction of the plant.

Between 1973 and 1979 the management structure to execute the BFS Project evolved to carry out the PSO philosophy of direct owner management of Project activities.

4. FUNCTIONAL CONTROLS FOR DESIGN AND CONSTRUCTION

The Manager, BFS Nuclear Project is responsible directly to the Executive Vice President for the functional coordination and control of the design and construction activities of all internal PSO and external BFS Project participants.

a. External BFS Project Participants

Equipment Suppliers

1) Nuclear Steam Supply System Vendor

The General Electric Company (GE) designs and manufactures the nuclear steam supply system (NSSS), nuclear fuel, and the turbine generators for the

Project. As the NSSS and nuclear fuel supplier, GE is assigned the responsibility to provide PSO with the engineering, design, procurement, fabrication, vendor surveillance and QA services for the NSSS and Nuclear Fuel. GE is required to provide a QA program acceptable to PSO for the activities that have been delegated to GE. The GE management structure for accomplishing these activities is set forth in PSAR Chapter 17, Appendix C.

2) Other Suppliers

A large number of other suppliers provide other components of the plant. These suppliers are required to provide QA programs acceptable to PSO for the activities that have been delegated to them.

Service Contractors

1) Engineering and Design Services

Black & Veatch Consulting Engineers (B&V) provides PSO with consulting, design, and engineering services. Black & Veatch is one of the ten largest consulting firms in the world and is a leader in the design of power generation, distribution, and related facilities. The firm was founded in 1915 and now has more than 2,900 personnel in eleven offices in the United States and overseas.

The firm's Power Division provides complete engineering design and consulting services for the electric utility industry for both nuclear and fossil generating stations. The Power Division is staffed by more than 1,500 professional and support personnel.

Black & Veatch is responsible to PSO for the engineering and design of the structures and balance of plant systems up to the interface with the NSSS and their integration with the equipment and systems provided by the NSSS supplier, together with licensing support service activities for Black Fox Station. B&V is also responsible for providing on-site engineering services to assist in the resolution of construction problems and design problems arising during construction. B&V serves as a general consultant to PSO with respect to the entire conduct of the Project. B&V is required to provide a QA program acceptable to PSO for the activities that have been delegated to B&V. The B&V management structure for the accomplishment of these tasks is set forth in PSAR Chapter 17, Appendix B.

2) Other Service Contractors

Many other service contractors provide numerous services such as construction inspection and testing, construction equipment operation, and others. PSO exercises control over these contractors by careful contract administration and monitoring of performance. These service contractors are required to provide QA programs acceptable to PSO for the activities that have been delegated to them.

b. PSO Management of Design and Construction

1) Background

At first a functional line organization was established to carry out management of the Project. Over the years since Project inception, the management organization was changed to a "matrix" system in which many Project participants were responsible both to the Project and to their normal Company organizational subdivision. The organization has subsequently evolved to a more functionally-oriented structure.

2) Current Management Structure

The interaction of the present PSO management structure with BFS Project suppliers and contractors in the existing system is depicted graphically in Figures (3)(vii)-2, -3, and -4, below. The four control functions inherent in the system are control of design, control of contracts, control of construction, and control of quality

a) Design Control. Existing methods for design control of Project work are generally effective. The Company has lead responsibility for, and coordinates design activities among, all Project suppliers and contractors. Actual design is performed and directly controlled by Black and Veatch for Balance of Plant and by General Electric for NSSS. The interface points between the two principal contractors are identified specifically by GE and coordination of those interfaces is carefully monitored by PSO in its design surveillance program.

Black and Veatch is responsible to PSO for design integration. Black and Veatch exercises design control over equipment suppliers other than the NSSS supplier (GE) and its subcontractors (design control by GE). Black and Veatch designs and specifications must be reviewed by, and receive concurrence from, PSO prior to attaining release status, necessary for inclusion in contract and purchase order bidding documents.

Design relationships are shown in Figure (3)(vii)-2.

The Manager, BFS Engineering is responsible to the Manager, BFS Nuclear Project, to provide efficient and economical execution of design, administrative control and technical direction to mechanical, electrical, civil, and nuclear engineering functions. He is responsible for surveillance of the BFS design, PSO's surveillance of design coordination among all participants, PSO's compliance with the technical requirements of the ASME "N" Stamp Program, oversight of the Q List, technical assistance for QA audits and vendor surveillance, technical surveillance of procurement activities, and evaluation and surveillance of design changes.

b) Contract Control. PSO control of Project activities continues after the initial design. At that point contract control takes over the lead from design control and carries control forward in the process of obtaining goods and services.

The initial design merged with appropriate commercial terms and conditions by PSO provide a bidding document for each proposed contract or purchase order. Bidder negotiations conducted by PSO (participated in by both PSO and Black & Veatch engineering organizations) result in revised specifications integrated by Black and Veatch under PSO design surveillance to produce final purchase order or contract documents. Once awarded to the successful bidder by PSO, the work required by the purchase order or contract is administered by PSO. The PSO positions described below and in Figure (3)(vii)-3 outline the responsibilities and lines of control for administration of the various contracts and purchase orders.

PSO RESPONSE: 10 CFR 50.34(e)(3)(vii)

Based on a study of general procurement practices completed by Booz, Allen & Hamilton in January, 1980, PSO has established a Project team to implement improvements in its procurement management for the entire Company. BFS Project procurement procedures will be reviewed and updated to reflect more recent Company practice.

The Manager, Materiel is responsible to the Vice President, Materiel and Property Management for coordinating the preparation of BFS Project bid documents, evaluation of bids, and the negotiation and award of procurements, whether by purchase order or contract; he provides for the administration of the General Electric contracts.

The Manager, BFS Materiel and Administration is responsible to the Manager, BFS Nuclear Project for post-award administration of purchase orders and contracts, for expediting and timely delivery of all material, for site warehousing and inventory control, and for site procurement actions. He provides for Project document control through development and implementation of the records management system and for management of the Project safety, security and training programs. The Manager, BFS Nuclear Project retains responsibility for administration of the Black and Veatch design contract.

c) Construction Control. Direct control of day-to-day contractor construction activities is provided through coordination, scheduling, and quality control surveillance by PSO. Any change in the work required by the contract design and specifications is controlled from proposal through final PSO approval by the formal change control system which involves both the design control and contract control functions. Figure (3)(vii)-2 also depicts the direct control over construction site activities exercised by the Manager, BFS Construction.

The Manager, BFS Construction will directly manage the Project field organization and provide technical and administrative direction for his superintendents. He is directly responsible to the Manager, BFS Nuclear Project for coordination of construction activities, for ensuring an effective quality control function, and for enforcing compliance within contractors' and his own organization with the Quality Assurance Program. He provides surveillance of all site construction activities and acts as the coordinator for field activities. He controls mobilization, scheduling, and support of contractor effort. His charter to control construction is complete with stop work and final acceptance authorities.

d) Control of Quality. PSO is responsible for monitoring and controlling all Project QA activities involved in design, procurement, and construction of BFS. This control is provided primarily by PSO staff, supplemented as necessary by contracted capability.

Direct responsibility for those QA/Quality Control (QC) functions inherent in the production of goods and services obtained by purchase order or contract is delegated to the cognizant vendor. Thus, the NSSS supplier, other equipment and material suppliers, Black and Veatch, and the construction or erection contractors each have QA responsibilities specified in their respective contract or purchase order.

PSO controls the vendor QA programs by means of pre-award evaluation, approval of proposed QA programs, source surveillance inspection, and audits of supplier and contractor program execution.

Internally, PSO audits its QA Program execution regularly and makes use of initial QA indoctrination training for all Project staff members and requires periodic refresher training.

PSO established a Review and Audit Committee (RAC) at the inception of the Project. The RAC reviews and evaluates quality-related Project activities as proposed by its members or assigned by the Chairman. It recommends to the responsible manager and to the President changes or improvements in the means of executing the QA Program deemed necessary to achieve and maintain a safe and reliable facility. The RAC is chaired by the Manager, BFS Nuclear Project and includes the Executive Vice President; Manager, QA; Vice President, Power Generation; Vice President, Materiel and Property

Management; Manager, BFS Engineering; Manager, BFS Planning, Scheduling, and Cost Control; and the Black Fox Station Manager.

A special subordinate Technical Advisory Committee (TAC) was established shortly after the TMI-2 accident to study the accident and the lessons learned from the several resulting investigations. The TAC's charge is to determine where BFS design and construction plans might be improved.

By instruction, the Manager, QA, reports on BFS Project status directly to the Executive Vice President; he has the responsibility and authority to coordinate QA matters directly with all PSO and external organizations. The QA department is responsible for the entire management and oversight of the PSO QA Program. It exercises direct QA supervisory authority over all suppliers and contractors. PSO QA staff has direct access to their counterparts in contractor and supplier organizations.

Figure (3)(vii)-4 shows the organization for control of quality.

The Manager, Quality Assurance is responsible to the Vice President, Power Generation for the preparation and management of the PSO QA Program and for surveillance and followup of program implementation. This responsibility extends to all Project activities including design, procurement, construction, construction and preoperational testing, startup testing and operations.

The Manager, Quality Assurance has been delegated the authority and provided the organizational freedom to identify problems and to initiate, recommend, provide solutions, and verify implementation of solutions. He is delegated the authority to oversee the execution and implementation of the QA Program and to perform both internal and external audits as necessary to assure a safe and reliable facility. He has written authority to stop use of unacceptable or unapproved purchase documents, procedures, or instructions and to interrupt the continuation of activities performed by PSO, contractors, or suppliers, including construction site and offsite activities, which would tend to degrade the quality of the structures, systems, and components important to safety. He is responsible to assure, through QA audits and surveillance activities, that verification of conformance to established quality requirements is accomplished by individuals or groups who do not have direct responsibility for performing the work being verified. He has delegated to his staff the authority to carry out the duties assigned to them to meet all responsibilities assigned to the Manager, QA.

The Superintendent, Construction QA is responsible to the Manager, QA for implementing an effective construction QA program, including review of contractors' QA programs, certification of PSO site inspectors, approval of contracted inspector certification, and approval of nonconformance resolutions.

The Supervisor, Procurement QA is responsible to the Manager, QA for implementation of the source surveillance program, including pre-award surveys of suppliers and contractors; source surveillance of contractors, subcontractors, and major suppliers; and review of procurement documents for QA aspects.

The Supervisor, Quality Programs and Audits is responsible to the Manager, QA for conducting QA audits, including supplier QA program approval, contractor QA program approval, and PSO and supplier/contractor program compliance audits, developing and maintaining QA manuals, and reviewing and approving Project procedures from a QA Program standpoint.

The Superintendent, Quality Control is responsible to the Manager, BFS Construction for implementation of the site quality control and construction contractor surveillance programs, including acceptance inspection for site receiving and of construction work for items important to safety. He is empowered to stop work that adversely affects the quality of areas, equipment, and systems. He is responsible for initiating and coordinating nonconformance reports in accordance with established procedures.

PSO RESPONSE: 10 CFR 50.34(e)(3)(vii)

B. TECHNICAL RESOURCES DIRECTED BY THE APPLICANT

1. BACKGROUND

BFS Project staffing began in 1973 with selection of ten engineers from the company. They were trained in reactor engineering fundamentals at a special course presented by Oklahoma State University. Eight of the ten were sent for periods of six to twelve months to nuclear stations under construction or in operation to acquire firsthand experience in design, construction, startup, and operation of a nuclear power station. Their duties were responsible job assignments functioning as utility engineers and obtaining pertinent experience. Since then, PSO has loaned additional engineers, quality control, quality assurance, startup, training, and construction personnel in similar fashion.

The Project staff built gradually. By the time of ACRS hearings in 1977 Project personnel had accumulated over 229 man-years of nuclear experience with a total of 56 full-time employees. Forty-one of them had technical backgrounds, including 27 engineering or science graduates. Of this total 229 man-years, the group possessed some 34 man-years of commercial nuclear power plant experience and 78 man-years of nuclear Navy experience.

By February of 1978, 82 full-time personnel were assigned to BFS Project. The Project staff included its own recruiting specialist, and personnel were being added as required to meet the various Project needs. A year later, strength had reached 160, and by the time of the TMI-2 accident stood at 220. Project staffing reached its maximum to date in October, 1979 with 248 PSO personnel.

With the decision by PSO in late 1979 to reduce its activity pending development of the NRC post-TMI action plans and resumption of the licensing process, personnel strength reduced gradually as Project activity slowed until reaching a current level of about 50. Some staff members are on loan to other active nuclear power plant construction projects, gaining practical experience that will be applicable to BFS Project work after their return.

Complete rebuilding of the Project staff is not planned until construction permits for BFS have been issued. Engineering, procurement, and construction will proceed only at a rate consistent with adequate staffing levels.

During the period from inception of the Project to the present, Black & Veatch built its Project staff to an equivalent peak strength of over 400 full-time Project personnel. This manpower level has been gradually reduced over the past two years to approximately 40, again because of the NRC licensing moratorium. Many of the staff members have been diverted to other nuclear related work that will maintain and broaden their technical skills pertinent to BFS work. They will return to BFS work upon resumption of Project activity following receipt of construction permits for BFS.

2. STAFFING LEVELS

During BFS construction, PSO will maintain a Project staff to oversee the design, procurement, fabrication, and construction management activities and to verify conformance with applicable regulations, codes, and design criteria. In specific cases where BFS Project staff is not sufficient to meet requirements, temporary technical help is available from PSO's in-house organizations or outside consultants contracted to work under the direction of PSO personnel. To support construction of BFS, PSO envisions staffing levels as shown in Table (3)(vii)-1. The figures in Table (3)(vii)-1 reflect the fact that no personnel buildup is planned prior to receipt of CP. Cognizant PSO managers annually develop manpower plans based on projected work requirements for ten years. Adjustments to the manpower plans are made periodically as required by actual workload.

3. LEVEL OF EDUCATION

Table (3)(vii)-2 lists present PSO staffing characteristics regarding education and experience. In addition to these technical resources there is a wide range of technical expertise within the PSO and B&V corporate organizations covering major engineering disciplines plus some of the more highly-specialized fields. Included among these assets are expertise in substation, transmission, and distribution design; results engineering; station electrical, instrumentation and control, environmental, chemistry, and mechanical disciplines. If a technical issue arises that is outside the scope of PSO and B&V's technical staffs' engineering capabilities, services of outside experts may be utilized to assist in resolving the issue.

4. TRAINING AND EXPERIENCE FEEDBACK

PSO builds on the experience of the technical personnel it hires by means of technical training programs incorporating academic work, seminars, workshops, and specific experience-building temporary assignments with other utilities. A conscious management effort is made to develop the Project technical staff's abilities as described above.

Important inputs to the technical staff include the operating experience represented in such documents as I&E Bulletins and LER's. These are routed to staff members to give them the benefit of others' experience. PSO participation in BWR Owners' Group activities broadens the experience background of technical staff members. Similarly, participation in general and specialized committee work of other technical and trade organizations such as ANS, ASME, IEEE, EEI, and INPO serve to provide the benefit of concentrated, evaluated experience to BFS staff members.

Project restaffing studies will investigate ways to bring more personnel with recent applicable experience into the Project. That effort will consider both in-house and external resources, including B&V and specialized outside consultants.

C. INTERACTION OF DESIGN AND CONSTRUCTION

PSO retains overall design and construction responsibility for the Project. The Company exercises authority over the design process through review and concurrence with design documents at various stages of the process.

The Project progresses from design through construction in several stages. In the first stage, that of Criteria Development, all major participants play a part. Black & Veatch develops design criteria for the Balance of Plant (BOP) segments. GE develops design criteria for the NSSS and the turbine generator system. GE also establishes interface requirements for the interfaces between NSSS and BOP systems. B&V prepares, reviews, and approves system design specifications for all plant systems and structures. These design specifications include all pertinent design criteria, interface requirements, and PSO unique requirements. PSO's review and concurrence results in an approved system design specification.

The next stage of the process is the translation of the system design specification's requirements into an initial design. Black & Veatch is responsible to PSO for overall engineering design and design coordination. However, PSO Project Management exercises control of design through review of and concurrence with designs. During this phase of the Project, PSO construction personnel conduct periodic constructibility reviews providing feedback to the design process to assure that construction problems are eliminated from the design. PSO personnel also conduct periodic model reviews during this phase. The design effort includes construction of a scale model of the NSSS containment, the turbine, fuel, control, and auxiliary buildings, and major systems therein. Project personnel review the model on a regular basis to identify potential design, construction, and operational problems. In addition, PSO has taken steps to ensure that industry experience in design, construction, and operation of similar plants is factored into the design. (See PSO response to Requirement (3)(i)). Upon completion, the design process results in a complete set of drawings and engineering specifications upon which procurement and construction is based.

During fabrication and construction, PSO Project Management administers a change control system which controls the interaction of the construction, manufacturing, and design processes. Black & Veatch, GE, construction contractors, or others involved in construction may initiate requests for design changes, specification changes, or contract changes. The Project Management organization reviews and approves or disapproves the request and, if approved, the formal change is issued. In order to facilitate the effective interaction of the construction and design processes, PSO has requested that Black & Veatch provide an on-site engineering organization to participate in the change authorization process during construction. All contracts and purchase orders are based on approved initial designs and specifications. The system for initiation, approval, and implementation of design changes is set forth in the Project procedures, which govern all activities relating to design and construction.

The bases for assuring close coordination of B&V and GE are the work scopes contained in their contracts with PSO. GE is charged with providing appropriate design criteria and submitting equipment specifications and other design documents for NSSS items that interface with BOP or which otherwise could influence design of BOP. Similarly, B&V is required to coordinate design activities by GE and other suppliers with design responsibilities. PSO provides surveillance of design activities and their coordination by B&V.

Necessary design information passes through these design interfaces, including changes to the design information as work progresses. Interface control documents identify the positions and titles of key persons in the communication channels and their responsibilities for decision-making, for resolution of problems, and for providing and reviewing information. Project procedures require all design change proposals to be reviewed by the same design organization involved with the original design.

Coordination of B&V and GE design activities is also assured by interface design reviews in accordance with the formal Project procedures including, as a minimum, personnel from the organizations responsible for each aspect of the design interface plus representatives of the PSO Manager, BFS Engineering and B&V.

Coordination of post-award changes is assured by PSO's Change Control System. Here, too, the system provides for the review of all types of changes by the organizations responsible for review of the original documents. Periodically, a Change Control Status Report is published to aid in management overview.

Design changes are brought about by means of a Description of Change Document (DCD) used in design activities.

PSO is ultimately responsible for the overall design, construction, and operation of BFS in accordance with NRC regulatory requirements, including the Quality Assurance requirements of 10CFR50, Appendix B. PSO's Project Management Organization is responsible for providing management oversight of principal contractor activities, obtaining Federal licenses and permits, approving basic design criteria, releasing selected design documents, and authorizing expenditures of funds. PSO also retains stop work authority over contractor design and construction activities.

The PSO Manager, BFS Construction and his staff are responsible for construction overview of contractor performance. The contractors and sub-contractors under PSO construction management are responsible for construction in a manner that conforms to design quality requirements. The Manager, BFS Construction and his staff: monitor construction activities; approve schedules, field procurements, selected invoices, and other financial controls; monitor compliance with permit and license requirements; monitor procedure compliance; and coordinate contractor turnover of plant systems to the plant operating organization.

In addition, QA provides construction overview through monitoring the QA aspects of site construction, including: review of contractor site procedures; audits and surveillance of construction; identification of quality problems and monitoring their resolution; and acceptance reviews of components, constructed structures, and completed systems. PSO has approved procedures for construction activity. These procedures will be revised and updated as needed to reflect the organization and will conform to applicable regulatory requirements, contractual arrangements, and the Black Fox Station Quality Assurance Program. Procedures will exist for each organizational element involved in construction overview activities.

D. TRANSITION TO OPERATION

1. TECHNICAL CONTINUITY

The PSO Executive Vice President is responsible both for nuclear plant engineering, procurement, construction, fuel, QA, and for operation. This centralized authority will greatly facilitate the transition from construction of Black Fox Station to operation.

Once Black Fox Station becomes operational, PSO will have in place the required technical support necessary to assure safe and reliable plant operation. The BFS Engineering organization responsible for review and surveillance of plant design will be available as the technically cognizant expert resource when BFS operates, performing the same functions of engineering support as they do now for the BFS Project. Technical specialty support from outside sources will be employed when necessary.

Since the BFS Engineering Organization will be physically located at the site during construction and start-up, the members of the organization will have excellent familiarity with the equipment. These individuals will be a basic resource for actual transfer to the operations or engineering support groups. Keeping this group on site will improve its performance by giving the technical support staff maximum access to systems that they will be working on and by developing a close relationship with the operating staff. This relationship should serve to improve communications. Although there will be formal procedures by which the plant staff can request design changes, this close relationship should improve the mutual understanding and performance of both groups.

PSO's goal for technical skill level is to have on hand individuals who are technically capable of performing design verification for all technical areas, especially those that are uniquely nuclear. For very specialized and complex areas, such as seismic analysis, PSO intends to continue to employ outside expert consulting assistance.

2. OPERATIONAL CONTINUITY

Both BFS Operations and PSO's fossil plant operating organization have had personnel involved in BFS design reviews during the design process to ensure that operational aspects are factored into the plant. PSO intends to acquire the operating staff with ample lead time for them to learn the plant design and operation. Furthermore, it is PSO personnel policy to open new technical staff positions within the Company first to existing PSO staff members, and to encourage transfers within the organization. Thus, engineering and management personnel involved in BFS design and construction phases who also have operating experience will be encouraged to transfer to the operational positions as they are available. This will facilitate the transfer of expertise to operation. The BFS Operations group will be deeply involved in construction turnover, construction testing, preoperational testing, and start-up. At the BFS Project, the Station Superintendent, who reports to the Station Manager, is also Chairman of the Test Working Group, which is responsible for the conduct of the formal pre-operational and startup testing programs.

3. CONTRACTOR CONTINUITY

GE, the NSSS vendor, operates the BWR-6 Training Center adjacent to the BFS site. This Center models the BFS Unit 1 control room and provides the most important tool to be used by PSO in training its operators and technical personnel for station operation. GE also supplies technical personnel to support installation and startup of GE-supplied equipment as well as technical personnel to support PSO's general startup work.

GE will provide instruction manuals for various NSSS equipment. These manuals will include operation and maintenance instructions which will be used as references during formation of the BFS Startup, Maintenance, and Operation procedures. PSO may request additional procedure guidance from GE during all phases of plant construction or operation. This will help ensure that plant operations reflect the engineering expertise in plant design.

The services of B&V, the architect engineer, will be required to support the post operating license modification program. Because of their experience during the design and construction phase, this support will provide the continuity for the BFS unique design for which they were responsible.

In summary, PSO's internal organization and policies are such that a smooth transition to operation will be facilitated.

E. MANAGEMENT OVERSIGHT

1. BACKGROUND

PSO, under the joint ownership agreement with Associated Electric and Western Farmers Electric Cooperatives, has sole responsibility and is fully authorized to act for the owner utilities with respect to construction, fueling, and operation of BFS.

PSO exercises top level management oversight by assigning the responsibility for design, procurement, construction, and operation of BFS to the Executive Vice President. The Executive Vice President reports significant developments in the Project to the President and directly to the other members of the Board of Directors on a regular basis. He directly controls the Project by approving funds to implement Project decisions, by approving staffing complements, and by executing contracts. He regularly reviews the Project status, progress and current activities, and sets policy for future activities. He maintains close contact with Project activities through personal contact and review of Project status daily, weekly, or monthly, as circumstances require, with the Manager, BFS Nuclear Project and other project managers.

The Executive Vice President is designated "Engineer-in-charge" of the BFS Nuclear Project in accordance with ANSI N18.1-1971, "Selection and Training of Personnel for Nuclear Power Plants." As a member of the QA Review and Audit Committee, he reviews the QA Program at least annually to determine the need for corrective action and to identify those areas in need of increased emphasis. He has reporting responsibility under 10 CFR 55, 10 CFR 21, and other regulations.

The Manager, BFS Nuclear Project provides routine periodic reports, generally bi-weekly, to the Executive Vice President. 19
These reports identify progress, current difficulties and planned activities. They ensure that top-level management is aware of BFS activities. The Manager, BFS Nuclear Project holds meetings with Black & Veatch and with General Electric Company executives, enabling their management to be informed of Project status, management and technical issues, as well as plans for the future.

2. THE BFS MANAGEMENT REVIEW OF 1980

The construction activity under the LWA served to a large extent as a "shakedown" period for the project management system and the construction management methodology. During this time the Company tested its construction procedures by using them in connection with the non-safety related work. The basic management procedures developed and used during the LWA period remained in place during the moratorium but are in the process of being reviewed to accommodate newly-developing circumstances.

The NRC licensing moratorium offered a significant opportunity for the Company's management to assess the effectiveness of the project management organization.

In 1980 the PSO President appointed a four-man internal Management Review Team to review the project. They submitted a five-phase report for senior executive review in February 1981.

Major recommendations of the Management Review Team were:

- The Project should adopt a functional line organization headed by a full-time executive, on site, with previous nuclear project experience, in lieu of the matrix approach to management of the Project.

- Renegotiate and redefine relationships between PSO Project Management and its principal contractors, Black & Veatch Consulting Engineers and the General Electric Company.

- Augment the BFS construction management methodology to include an outside organization with proven construction management experience if that alternative is confirmed by a recommended review to determine the selection of the appropriate management methodology.

- Increase the number of personnel with previous nuclear power plant project experience.

The first recommendation coincided with previously evolving project plans to shift from a matrix system to a project management system. With respect to the second recommendation, discussions have been held with Black & Veatch to redefine its responsibilities with respect to the Project, and negotiations are ongoing with General Electric to clarify future relationships. For the other recommendations, PSO executive management, after its review, decided to delay decision on implementation until the schedule for resumption of construction became clearer.

As Project restart begins, PSO management will review the recommendations of the Management Review Team and other inputs, and implement any changes deemed necessary in the light of current circumstances to assure the most effective management control and support of the Project. PSO will provide timely notification to the NRC of any significant changes in organizational structure.

3. OTHER MANAGEMENT REVIEWS

PSO Executive Management maintains a dedication to careful oversight of the conduct and management of the BFS Project. It has repeatedly called on outside management consultants for analysis of Project performance. Studies of the BFS Project have been performed by Tera Corporation and Management Analysis Company. Booz, Allen & Hamilton performed a study of PSO's over-all procurement system and recommended a number of changes, many of which have been or are being implemented. Arthur Young & Company performed a study of PSO's personnel resources and the resulting recommendations have been implemented. As previously noted, a Technical Advisory

PSO RESPONSE: 10 CFR 50.34(e)(3)(vii)

Committee was established to review the implications of the TMI-2 accident with respect to Black Fox Station.

TABLE ([illegible]vii)-1

PSO MANPOWER ESTIMATE

End of Year Staffing Levels

Functional Areas	1983 CP	1984	1985	1986	1987	1988	1989	1990	1991 FL CO#1	1992	1993	1994 FL CO#2
Licensing	3	3	3	3	3	3	3	3	3	3	3	3
Engineering	27	31	35	40	42	51	57	62	64	59	40	30
Administration	61	135	189	214	220	226	216	184	137	115	90	65
Construction	21	60	100	120	120	120	110	90	90	85	75	35
Operations	5	7	28	40	57	115	200	265	300	335	382	450
Quality Assurance	12	21	33	48	59	62	42	28	28	28	28	28

CP: Receipt of Construction Permits

FL: Fuel Load

CO: Commerical Operation

Assume: CP 1983

Construction Start 1984

Unit 1 Operational 1991

Unit 2 Operational 1994

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TABLE (3)(vii)-2

BFS PROJECT TECHNICAL QUALIFICATIONS

ORGANIZATION	NUMBER OF PERSONS	CREDENTIALS												EXPERIENCE		
		ACADEMIC DEGREES											REGISTRATION STATUS			
		B.S.					M.S.					Ph.D.				
		CE	EE	ME	NE	OTHER	CE	EE	ME	NE	OTHER			BFS	NUCLEAR	TOTAL PROFESNL.
Executive																
Management	2		1	1									2-PE	16	16	61
Project Staff																
Management	5			2		2				1		1	2-PE	22	66	136
Technical Staff	13	1	2	3	1	4					2		3-PE 1-EIT	55	92	175
Project Support																
Management	5		2	2		1				1		1	3-PE	36	46	136
Technical Staff	11		5	1	2	3		1					6-PE 2-EIT	35	35	135
Quality Assurance																
Management	1					1							1-PE	8	8	18
Technical Staff	1													3	7	7

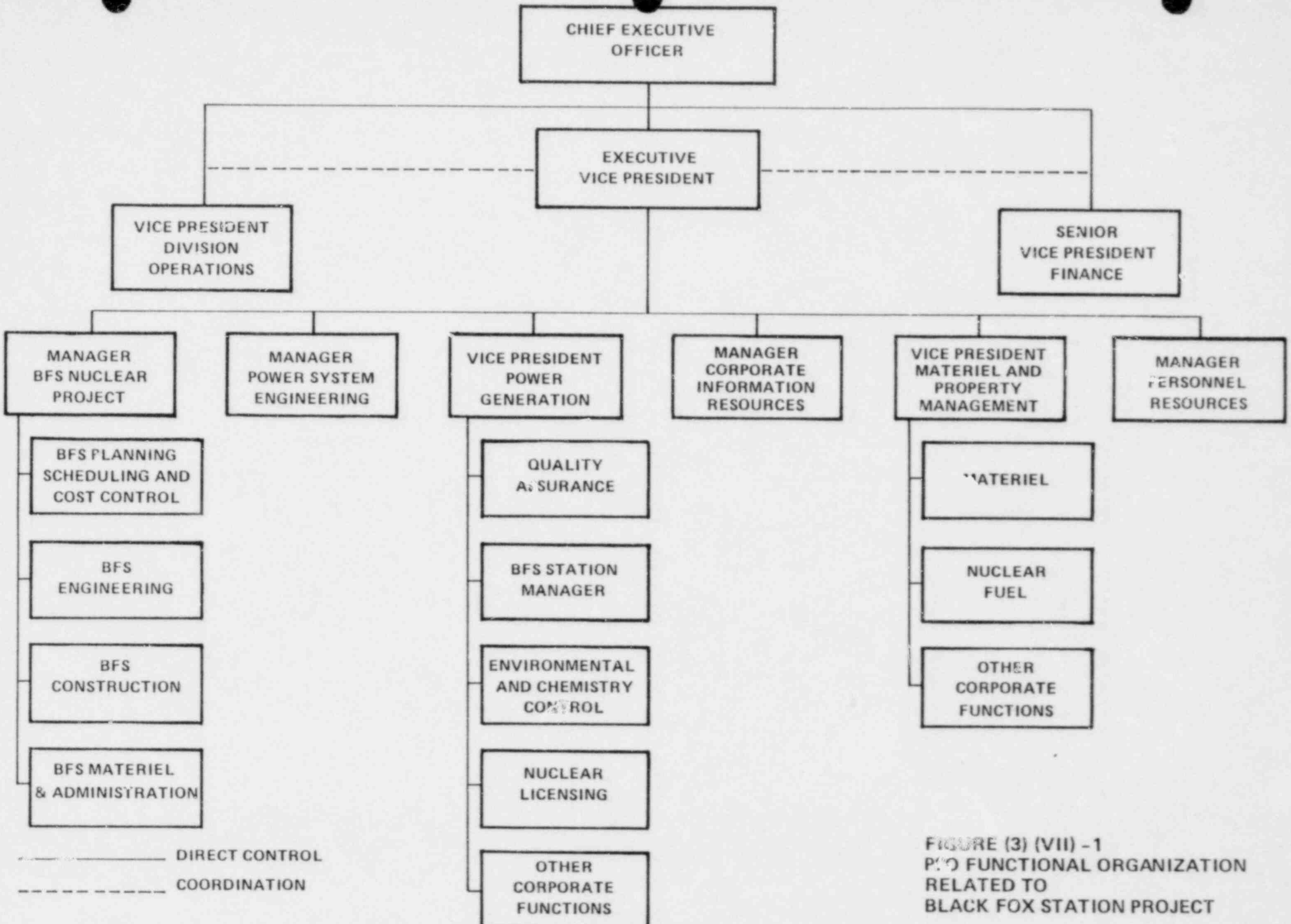

FIGURE (3) (VII) - 1
PSO FUNCTIONAL ORGANIZATION
RELATED TO
BLACK FOX STATION PROJECT

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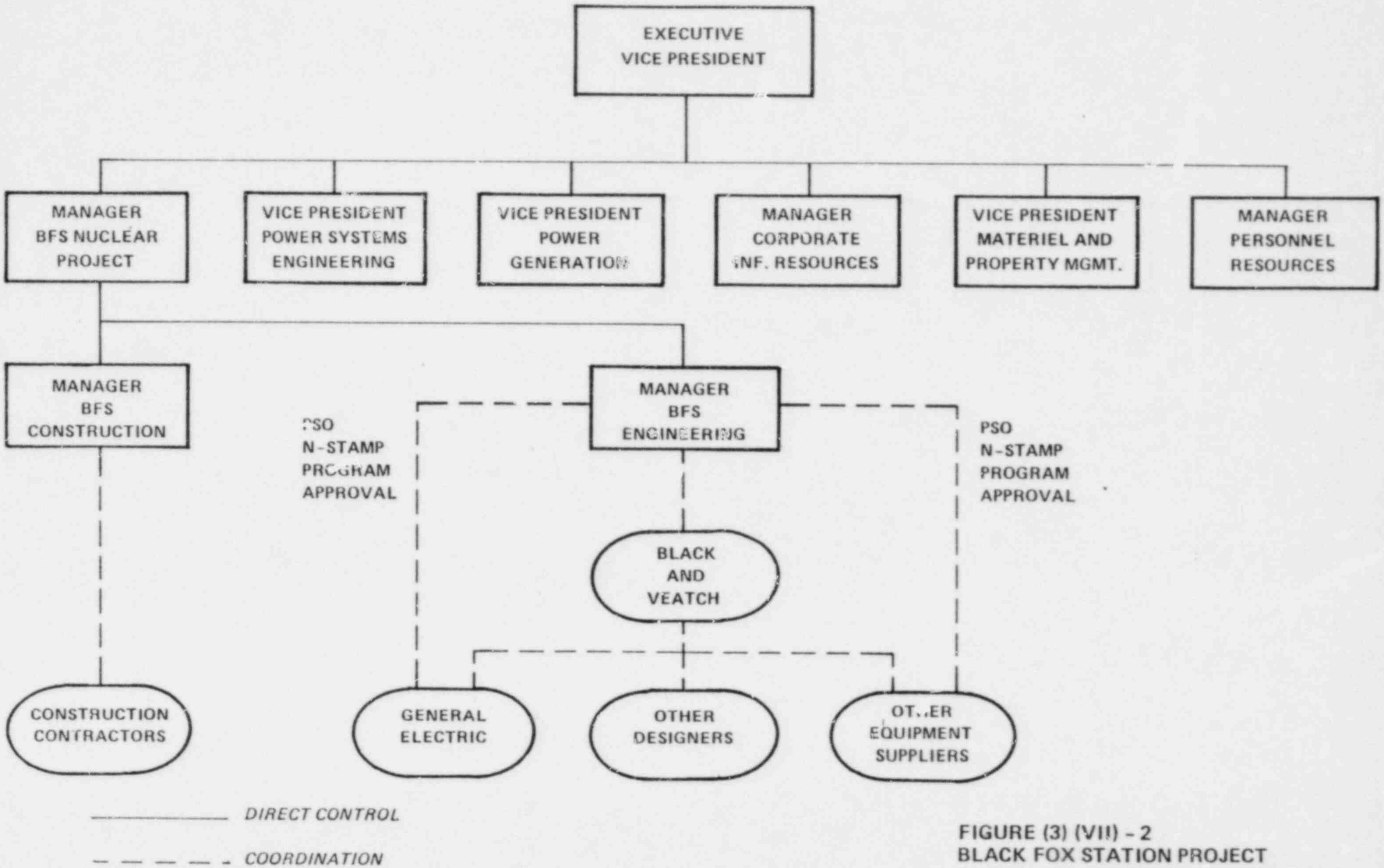

FIGURE (3) (VII) - 2
BLACK FOX STATION PROJECT
DESIGN AND CONSTRUCTION CONTROL

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19-111381

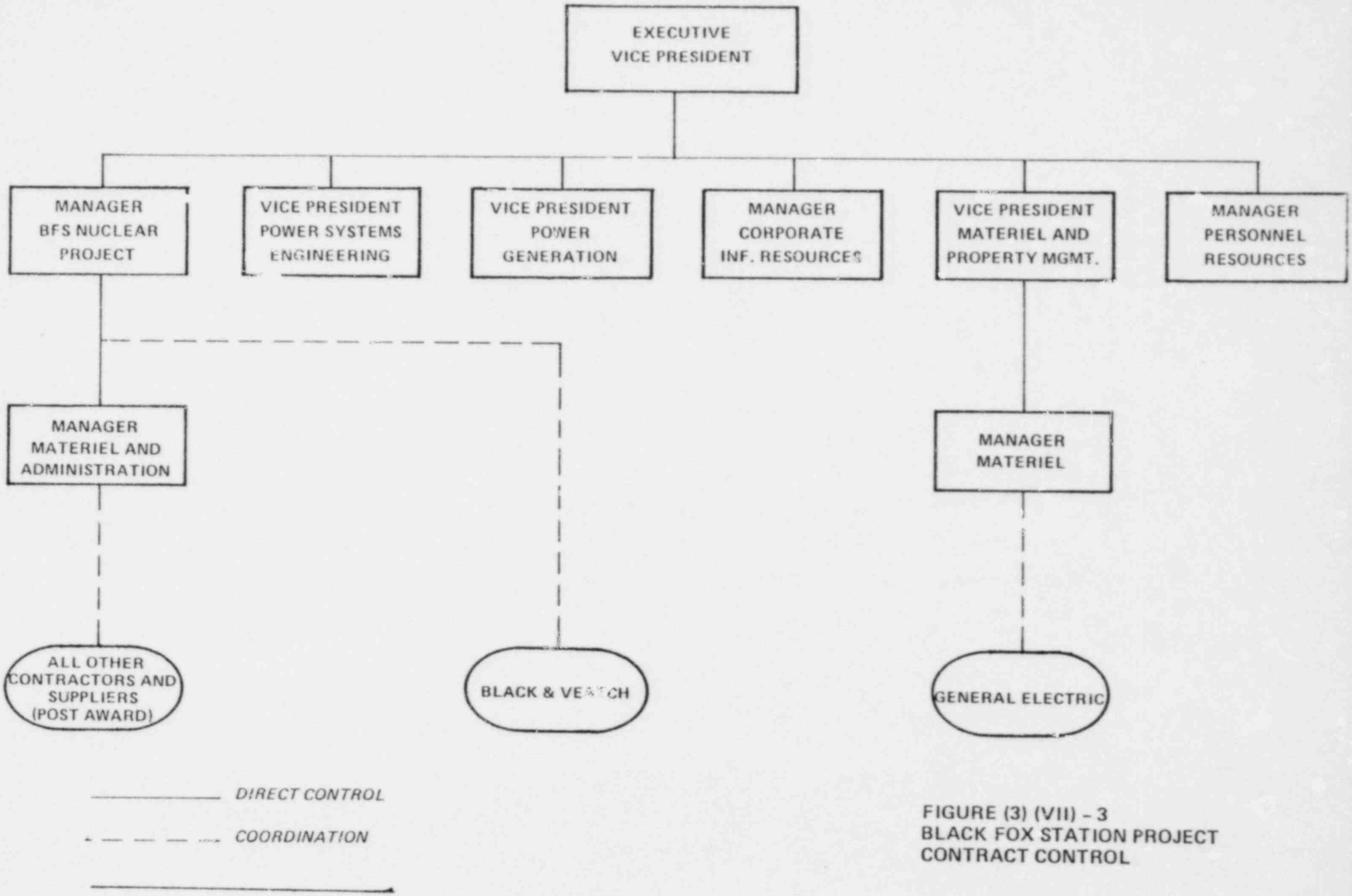

FIGURE (3) (VII) - 3
BLACK FOX STATION PROJECT
CONTRACT CONTROL

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19-111381

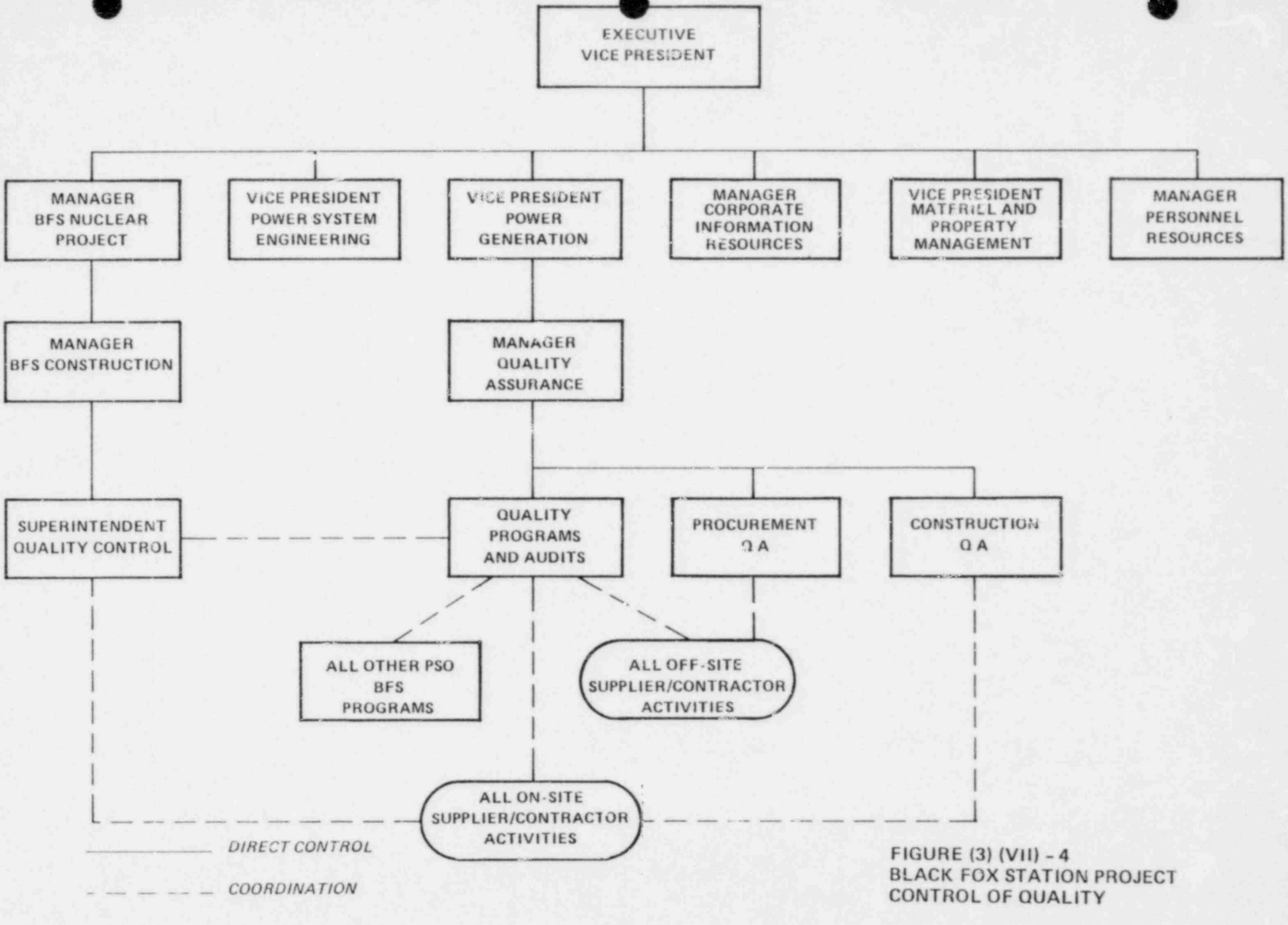

FIGURE (3) (VII) - 4
BLACK FOX STATION PROJECT
CONTROL OF QUALITY

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19-111381

10 CFR 50.34(e)(2)(ix)/(3)(v) RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS

NRC POSITION:

(2) To satisfy the following requirement, the application shall provide sufficient information to demonstrate that the required actions will be satisfactorily completed by the operating license stage. This information is of the type customarily required to satisfy 10 CFR 50.35(a)(2) or to address unresolved generic safety issues. (NUREG-0718, Category 4)

(ix) Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. (NUREG-0718, II.B.8)

(3) To satisfy the following requirements, the application shall provide sufficient information to demonstrate that the requirement has been met. This information is of the type customarily required to satisfy 10 CFR 50.34(a)(1) or to address the applicant's technical qualifications and management structure and competence. (NUREG-0718, Category 5)

(v) Provide preliminary design information at a level of detail consistent with that normally required at the construction permit stage of review sufficient to demonstrate that: (II.B.8)

(A) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-3220, Service Level C limits, except that evaluation of instability is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler Pressure Vessel Code, Section III, Division 2, Subarticle CC-3720, factored load category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100 percent fuel-clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent, depending upon which option is chosen for control of hydrogen. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

(B) The containment and associated systems will provide reasonable assurance that uniformly distributed hydrogen concentrations do not exceed 10 percent during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel-clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

(C) The facility design will provide reasonable assurance that, based on a 100 percent fuel-clad metal-water reaction, combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

(D) If the option chosen for hydrogen control is post-accident inerting: (1) Containment structure loadings produced by an inadvertent full inerting (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subarticle CC-3720, Service Load Category), (2) A pressure test, which is required, of the containments, at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting can be safely conducted, (3) Inadvertent full inerting of the containment can be safely accommodated during plant operation.

(E) If the option chosen for hydrogen control is a distributed ignition system, equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity shall be designed to perform its function during and after being exposed to the environmental conditions created by activation of the distributed ignition system.

PSO RESPONSE: 10 CFR 50.34(e)(2)(ix)/(3)(v)

PSO RESPONSE:

INTRODUCTION

The basis for the requirement for a hydrogen control system which is capable of dealing with rapid generation of large quantities of hydrogen is the TMI-2 accident, which resulted in the generation of hydrogen beyond the limits specified in 10 CFR 50.44. As a consequence the NRC has identified hydrogen control arising from a degraded core as deserving special attention. The Commission has imposed new hydrogen control requirements on plants about to receive operating licenses, and more recently has issued hydrogen control requirements as part of the newly issued Near-Term Construction Permit/Manufacturing License Regulations. These construction permit hydrogen control requirements are hereafter referred to in this response as the "Hydrogen Control Rule."

COMMITMENT

PSO commits to provide a Hydrogen Control System (HCS) which will safely accommodate, in accordance with the Hydrogen Control Rule, the hydrogen generated by the equivalent of a metal-water reaction which consumes 100 percent of the zirconium metal in the active fuel cladding.

DESCRIPTION OF THE RESPONSE

The following four sections of this response provide the detailed information necessary to support PSO's commitment to comply with the requirements of the Hydrogen Control Rule. The first of these sections presents a description of PSO's long-range hydrogen control program, the bases for the preliminary system selection and a conceptual system description. The second of these sections presents a description of the preliminary design parameters which were used to assess the adequacy of the HCS. The third of these sections presents the preliminary system description and the assessment of the system's performance. The last of these sections presents a detailed discussion of the analytical methodology used in completing the performance assessment.

A. HYDROGEN CONTROL PROGRAM

In response to the NRC's hydrogen control requirements, PSO has undertaken a long-range hydrogen control program.

1. PROGRAM PLAN

The hydrogen control program is proceeding in several phases, as described below.

- Phase 1--Preliminary selection of a hydrogen control system. The results of this preliminary assessment are presented in this response.

  PSO will continue its evaluation of the various alternative systems for hydrogen control, including a consideration of the various industry activities in this area. The final evaluation and selection among the various hydrogen control systems will be completed and submitted to the NRC Staff two years after the issuance of the construction permits. Table (2)(ix)-1 provides a list of topics which will be addressed 19
  in this post-construction permit submittal. An evaluation program, similar to that described in Phase 2 below, will be carried out for the final hydrogen control system selected.

- Phase 2--Preliminary evaluation of the selected system against the requirements of the Hydrogen Control Rule and other specific design and performance criteria. This effort has been completed and the results are presented in this response.

- Phase 3--Detailed system evaluation which culminates in a final design. The results of this effort will be submitted with the FSAR.

PSO recognizes the existence of the many ongoing and planned research and development programs in the area of hydrogen control. Examples of these programs are identified in Table
(2)(ix)-1a. As part of its long-range hydrogen control 19
program, PSO is committed to active participation in the BWR Hydrogen Control Owner's Group and to maintaining cognizance of industry efforts in this area.

2. PRELIMINARY SYSTEM SELECTION

   a. Selection Criteria

      A number of approaches to hydrogen control have been proposed. These approaches, as integrated into a total BFS HCS, were evaluated against the following criteria:

      1) The HCS and its supporting systems must be able to safely control the hydrogen generated by the equivalent metal-water reaction which consumes 100 percent of the zirconium metal in the active fuel cladding, such that containment integrity and safe shutdown capability will be achieved and maintained.

      2) The system must be able to maintain the hydrogen concentration below the detonable limits or create an atmosphere incapable of supporting combustion.

3) The operation, including inadvertent operation, of the HCS should not endanger the health and safety of the public.

4) The inadvertent operation of the system should not result in unacceptable damage to station safety systems or pose an undue risk to station personnel.

5) The HCS must be able to assure that no stagnant areas exist where unintended combustion or detonation could result in a loss of containment integrity or loss of any required mitigating features.

6) The HCS must be able to function adequately over a wide variety of postulated events.

7) The components of the HCS, insofar as possible, should be a standard design and not require extensive development. If major components or subsystems require developmental work, the potential for substantial improvement over current performance levels should exist.

8) The preliminary assessment of each alternative should be based on an appropriately conservative analysis.

b. Evaluation and System Selection

There is a considerable amount of research under way to evaluate various aspects of hydrogen control. These activities are expected to provide valuable information in a time frame that will support the detailed design and procurement of a final HCS for BFS. The following four potential hydrogen control systems were selected for preliminary evaluation against the above listed criteria:

- Water fogging
- CO_2 post-accident inerting
- Halon post-accident inerting
- Distributed igniter system

The conclusion of the preliminary evaluation was that a Distributed Igniter System satisfies all of the above specified evaluation criteria. Based on a qualitative evaluation of these systems, PSO has tentatively selected a HCS consisting of a Distributed Igniter System (DIS) operated in conjunction with a spray system equivalent to a single loop of the containment spray operating mode of the Residual Heat Removal (RHR) system.

c. Conceptual System Description

The DIS would react large quantities of hydrogen with oxygen by controlled combustion of the hydrogen as it is released to the drywell and containment volumes. The rate of hydrogen combustion within a given volume is influenced by the number and distribution of the igniting elements. These elements would be provided in sufficient quantities and at the proper locations to ensure that local hydrogen concentrations remain below the detonable range as long as the local atmosphere is capable of supporting combustion.

The heats of combustion, metal-water reaction, and radioactive decay would be absorbed by the suppression pool, the containment sprays, and the large thermal mass of the containment concrete and steel structures. This heat absorption will limit the temperature effects of hydrogen combustion to localize transients which will be analyzed in detail during the equipment qualification review. The 19
pressure suppression effect of the containment spray will act to limit the peak pressures resulting from the controlled deflagrations to values below the minimum required pressure used in the evaluation of containment integrity, as defined in Subsection C.2.d.

The DIS will be designed with suitable reliability such that proper functioning of the system is assured. The DIS will be powered from two independent sources such that each source will supply power to one-half of the igniter assemblies.

B. PRELIMINARY DESIGN PARAMETER DEVELOPMENT

1. INTRODUCTION

The three major design parameters which are used in the assessment of performance adequacy are as follows:

- Hydrogen release rates
- Hydrogen release points
- Hydrogen combustion characteristics

These design parameters were utilized in the hydrogen migration and combustion analyses which are described in Section D, as part of the assessment of the adequacy of the proposed system relative to the requirements of the Hydrogen Control Rule. The selection of parameter values and analytical techniques was based on engineering judgment and experience in performing similar work in other areas. Where appropriate, parametric analyses were performed prior to selecting the base case values.

2. DESIGN CRITERIA

a. Rates of Release

Conditions were postulated such that the resultant mass and energy release to the containment system is accompanied by significant hydrogen generation during a time frame which would reasonably permit recovery of core cooling prior to significant core degradation. Three system parameters are of major significance in defining the hydrogen release. These are:

1) Rate of Reactor Coolant Loss

The time to o- of hydrogen generation is determined by the rate o lowdown, as significant hydrogen generation does not begin until the water level in the RPV has dropped below the active core region. The rate of coolant loss also influences the rate of hydrogen generation by limiting the flow of steam and hydrogen from the core during the reaction period. A steam line break area of 0.163 ft^2 was assumed. This is equivalent to a stuck-open safety-relief valve.

2) Rate of Makeup

Depending on the rate of coolant loss and the power history of the core, makeup may be required to avoid core slumping prior to achieving a significant amount of metal-water reaction. For this analysis, the reactor coolant venting was assumed to occur concurrently with a makeup of 30,000 lbm/h injected into the lower plenum until the peak fuel centerline temperature reached 4130° F. This makeup flow is equivalent to that provided by the Control Rod Drive (CRD) system.

3) Previous Core History and Core Characteristics

These will determine the decay heat values and the fuel temperature distribution. The reactor was assumed to be scrammed at 100 percent power with an equilibrium power history.

In order to develop detailed hydrogen release data, PSO utilized the MARCH[1] computer code along with certain assumptions chosen to approximate the conditions leading to a maximum calculated cumulative release. The following assumptions were made in addition to those given above.

a) When the peak fuel centerline temperature reached 4130° F at 34 minutes, only the hydrogen mass flow rate was taken from the MARCH calculation and the mass and energy release rates for steam were based on the assumed availability of sufficient makeup to remove the total energy generated by decay heat and the metal-water reaction.

b) No fuel was allowed to slump into the lower plenum until all fuel reached the melting point. Prior to this point, the reaction became steam-limited and the results of the MARCH calculation were modified as discussed below.

c) At 70 minutes, the reaction rate was severely limited by the amount of flashing steam from the lower plenum. At this point, approximately 65 percent of the active fuel cladding had reacted. To satisfy the Hydrogen Control Rule requirement to safely control the hydrogen generated by the equivalent of a metal water reaction which consumes 100 percent of the zirconium metal in the active fuel cladding, the reaction rate was assumed to be constant at 48.5 lbm/min hydrogen until 84.2 minutes, at which time the reaction was complete.

The resultant mass, energy, and hydrogen release rates are shown in Tables (2)(ix)-2 and -3. These release rates were used for performing the preliminary evaluation of the performance adequacy of the DIS. A more detailed description of the methodology used, and the parametric analyses performed to establish this hydrogen generation rate is presented in Section D of this response.

The BWR Hydrogen Control Owner's Group has undertaken the evaluation of a BWR hydrogen source term, and of the factors which are expected to influence the design basis for evaluating the adequacy of the Hydrogen Control System performance. These factors include:

- Time to start of generation
- Rate of generation
- Mass and energy release to containment
- Location of release
- Total hydrogen release

The results of the BWR Hydrogen Control Owner's Group evaluation will be compared with the values selected for this preliminary assessment and any modifications which are indicated as a result of this comparison will be reflected in the Phase 3 detailed design effort.

b. Release Points

The BFS containment shown in Figure (2)(ix)-1 is relatively insensitive to the precise release modes. The two cases for evaluation are: a single asymmetric release through a safety-relief valve (SRV) downcomer in the suppression pool; and, a symmetric discharge through the drywell vents.

Releases inside the drywell volume would channel the mass and energy of the release to the containment volume through the horizontal vents at the base of the drywell into the suppression pool. Because of the symmetry of the vents, such a release would produce an axially symmetric distribution of the steam and noncondensible gases to the containment volume.

Potential release points outside the drywell but inside the containment can be subcategorized into three general groups--high energy fluid system lines, low energy and small diameter piping, and large diameter inactive piping such as the safety-relief valve discharge lines (SRVDL) and ECCS test return lines. Of these groups, only the SRVDL creates the potential for continuous release of significant mass and energy to the containment. The SRVDL terminate outside the drywell and in the suppression pool as indicated in Figure (2)(ix)-2. The failure of a SRV to close when required would result in a significant rate of mass and energy release to the suppression pool.

c. Hydrogen Combustion Parameters

The basic analytical tool used to assess the performance adequacy of the DIS is the B&V computer code HYBRID. A functional description of HYBRID is presented in Section D.4. of this response. HYBRID utilizes the hydrogen release rates developed in Section B.2.a. and, by appropriate modeling, injects this hydrogen into the containment system at the release points identified in Section B.2.b. To provide the pressure and temperature time histories which would result from controlled combustion of this hydrogen, it is necessary to specify the relevant combustion parameters.

NUREG/CR-1561[2] provides a concise summary of the current literature relative to hydrogen combustion. Based on NUREG/CR-1561 and other available information, PSO has selected base case values for the parameters listed in Table (2)(ix)-4.

C. SYSTEM DESCRIPTION AND PERFORMANCE ASSESSMENT

1. SYSTEM DESCRIPTION

a. Preliminary Layout

The results of the hydrogen migration analysis, which are described in Section C.2.a. of this response, form the basis for the preliminary layout of the DIS. The relative locations of the igniters are shown in Figures (2)(ix)-3 through (2)(ix)-8. To ensure prompt ignition of hydrogen exiting from either a single point or distributed release, a ring of 18 igniters (9 per division) will be placed in the vicinity of the platform at Elevation 576' 7". A second ring of 12 igniters (6 per division) will be placed near the HCU floor at Elevation 592' 10". Thus, a total of 30 igniters (15 per division) will be available to provide positive ignition of hydrogen in the wet well region of the containment volume.

Above the HCU floor, the flow of hydrogen is directed by the floors and walls of the subcompartments which span the area from the drywell to the containment vessel. Hydrogen which exits the wet well region will be channeled by the steam tunnel and the suspended concrete slabs beneath the HCU modules at Elevation 592' 10" into one of the four relatively open quadrants between the sides of the concrete slabs. A total of 8 igniters (4 from each division) will be placed in these areas in the vicinity of the platform at Elevation 641' 5". A total of 8 igniters (4 from each division) also will be placed in these areas in the vicinity of the platforms at Elevation 618' 11".

To provide for reliable combustion of any hydrogen which reaches the containment dome, a total of 12 igniters (6 from each division) will be placed in the volume above the polar crane.

The containment air recirculation system supplies chilled air to various general areas of the containment. This system isolates on a LOCA signal but can be manually restarted by the operator. With the sole exception of the Main Steam Tunnel, subcompartments within the containment are cooled by internal fan coil units. There is very little air movement between these subcompartments and the general containment atmosphere even under conditions of full forced recirculation. To ensure controlled ignition of any hydrogen which might migrate into open subcompartments, 2 igniters (1 from each division) will be placed in the Main Steam Tunnel at about Elevation 525' 0" and in the Reactor Water Cleanup Demineralizer (RWCUD) pump and tank room on Elevation 641' 5". The total number of igniters to be used in the containment is 62.

To provide for controlled ignition of any hydrogen released to the drywell, two rings of 8 igniters each (4 from each division) will be provided. The first ring will be located at about Elevation 590' 0". The second ring will be in the vicinity of the platform at Elevation 616' 11½". As discussed below, combustion in the drywell is expected to occur under hydrogen-rich rather than oxygen-rich conditions. Air is expected to reenter the drywell primarily through the drywell vacuum-relief line. To control the rate of oxygen buildup, 6 igniters (3 from each division) will be located in the upper part of the drywell. These igniters will also provide protection against potential pocketing in the drywell head region. The total number of igniters to be used in the drywell is 22.

b. Igniter Assembly Description

The igniter assembly proposed for the preliminary BFS DIS is similar to that employed at Sequoyah Nuclear Station and proposed for Grand Gulf Nuclear Station. The igniter, as presently envisioned, is a General Motors AC Division Model 7G glow plug which will be mounted in a welded steel box. The glow plug will be provided with a spray shield to protect the igniter element from containment spray. The igniters located in the wet well region will either be provided with deflectors for pool swell and froth protection, or will be shown to have an acceptable surface temperature recovery time following immersion. A heat shield will be provided, if necessary, to protect the igniters from high temperatures.

c. Igniter Supports

The igniter assemblies will be adequately supported to withstand, without loss of function, the loads associated with seismic events (SSE), and hydrodynamic (pool swell, jet impingement, and pressure spikes associated with pipe rupture and hydrogen ignition) and thermal (pipe rupture and hydrogen ignition) transients.

d. Power Supplies

The igniter assemblies will be provided power from two 480 V ESF power buses, one for each division. These are Class 1E power supplies which, in the event of the failure of the normal power supplies, will be fed from the station's emergency diesel generators. A preliminary one-line diagram is shown on Figure (2)(ix)-9. All electrical components except the local junction boxes will be located outside the containment and are therefore accessible for inspection and repair, even during operation of the system.

e. Proposed Method of Operation

1) System Operation

This DIS will be designed to prevent the accumulation of detonable concentrations of hydrogen. The DIS will not be required for events which result in the generation of hydrogen less than or equal to the amounts and release rates considered in the design of the present Containment Combustible Gas Control System (CCGC) as described in PSAR Subsection 6.2.5. It is intended that the DIS be manually actuated for all event sequences which possess the potential to generate excessive amounts of hydrogen. The design of the DIS will be such that planned or inadvertent actuation of the system will not adversely affect the operational safety of the plant nor increase the severity of a particular event.

2) Initiation Criteria

As shown in section B.2.a. an event which will require operation of the DIS proceeds at such a rate as to allow actuation by a control room operator in accordnce with emergency operating procedures. Reactor water level is considered to be the best indication of the potential for rapid hydrogen generation in a BWR. System initiation details will be included in the FSAR.

f. Tests and Inspection

1) Preoperational Testing

The DIS will be preoperationally tested to ensure correct functioning of all controls, instrumentation and wiring, transformers and igniters. The test will consist of energizing one of the two ESF power distribution panels from the control room and verifying that all igniters powered from the associated panel are functional. The identical procedure will be followed for the remaining igniters powered off the remaining ESF panel.

2) Surveillance Tests

During plant operation, the igniter assemblies, power distribution panels, instrumentation, and associated wiring can be visually inspected (outside the drywell) and operationally tested at any time. All igniter assemblies will be tested periodically to verify operability. The test procedure will be similar to the preoperational test procedure discussed above.

g. Instrumentation and Controls

The DIS will be manually initiated from the control room. Instrumentation for the DIS consists of two control room handswitches, one for each of the two Class 1E power divisions. Each handswitch energizes the igniters in its respective division.

2. DISTRIBUTED IGNITION SYSTEM PERFORMANCE ASSESSMENT

During the course of the preliminary performance assessment described below, other advantages of the DIS were identified. These other advantages include:

- o The hydrogen deflagration process produces pressure time--histories which have frequencies which are significantly lower than the major structural response frequencies.

- o The system configuration is flexible. That is, should design parameters change, the system can be expanded or altered (e.g., adding or relocating igniters) with minimal impact on the remainder of the plant. Moreover, the igniters can be located in such a manner that the loss of one or more igniters will not limit the ability of the DIS to perform its intended function.

a. Unintended Local Conbustion

The potential for localized high concentrations of hydrogen (pocketing), which might lead to unintended combustion or detonation was evaluated by performing a hydrogen migration analysis using the SOLA-DF computer code. A description of this analysis is presented in Section D.3. The following discussion presents the results of the analyses performed.

1) Gratings

The annular region of the BFS containment volume is subdivided by several levels of grating which serve as both personnel access platforms and as supports for equipment. The locations of the gratings (single-link shading) are shown in Figures (2)(ix)-3 through -8. This grating has an open area of 50 percent, not counting the area occupied by equipment base pads. To determine the potential effects of gratings on hydrogen flow, two separate SOLA-DF evaluations were performed.

The first evaluation was a free rise simulation in which the hydrogen was released from the suppression pool and encountered no obstacles except the drywell and containment walls. Typical results are shown in Figures (2)(ix)-10 and -11. Each small square symbol represents 0.5% within a grid. Figure (2)(ix)-10 displays an area 18 feet wide by 30 feet high by 18 feet deep divided into a 15 by 15 by 15 grid. The bottom of the figure represents the surface of the suppression pool. The Figure (2)(ix)-10 cross-section is taken approximately 8.5 feet out from the drywell wall. The plume remains relatively compact (the half-angle of expansion is about 10 degrees). Figure (2)(ix)-11 shows the plume horizontal cross section approximately 10 feet above the surface of the suppression pool. The plume is reasonably symmetric and nearly circular.

The second evaluation was a grating analysis performed by closing off alternate rows of cells in two layers, resulting in two layers each with 50 percent open area. The long axis of the top row of cells was oriented at right angles to the long axis of the lower layer. Except for the simulated grating, all conditions are identical for the two cases. Typical results are shown in Figures (2)(ix)-12 and -13. The differences between the two analyses are readily apparent, as the gratings cause the hydrogen to disperse horizontally to a much greater degree, when compared to the free rise simulation.

It has been concluded from these analyses that gratings or other large obstacles (e.g., pipe support structures, large equipment) can have a significant dispersive effect on hydrogen flow and should be included in detailed migration analyses.

2) Reactor Building 360 Degree Evaluations

The purpose of this series of analyses was to gain an understanding of how hydrogen which is released either at a single point or in a uniform distribution can be expected to flow in a large, internally segmented structure like the BFS containment. In particular, it was considered necessary to determine the size of volume for which the assumption of instantaneous, homogeneous mixing would be reasonable. An SRV release under the RWCU equipment area was selected for evaluation. A relatively coarse nodularization scheme was used. For this relatively coarse model, the effects of gratings and small subcompartments were not included. This is considered to be conservative with respect to dispersal effects for evaluating uniform distribution. The effect of grating and small subcompartments is considered in the 90-degree evaluation as discussed in section C.2.a.3.

The results of the RWCU release are shown in Figures (2)(ix)-14 and -15. Figure (2)(ix)-14 shows a vertical cross-section taken near the drywell wall. Figure (2)(ix)-14A shows the hydrogen distribution after 2 minutes of release and Figure (2)(ix)-14B after 4 minutes. As indicated, the hydrogen plume can be expected to rise fairly slow with minimal initial dispersal until the plume reaches the vicinity of the refueling floor. After 4 minutes of release, the do hydrogen concentrations are starting to approach the lower flammable limit (LFL) of 4 percent, while this limit was exceeded in the wet well region very shortly after the release started. The potential for structures and solid floors to create temporary asymmetric flow patterns is evident.

Figure (2)(ix)-15 shows two separate cross-sections taken after 12 minutes of release. Section 1-1 indicates the start of hydrogen migration horizontally into the wet well region. Section 2-2, taken nearer the RPV center line, shows no evidence of horizontal flow into this area.

On the basis of this preliminary analysis and the grating analysis described above, it has been concluded that for a single point release, the homogeneous mixing assumption is reasonable for a 90 degree arc centered about the release point.

The distributed release case was analyzed by simulating 18 equally spaced release points around the drywell wall, as shown in Figure (2)(ix)-16. The total hydrogen release rate was identical to that used in the single point release evaluation. The holdup and dispersive effects of the Main Steam Tunnel (shown at the right center of Figure (2)(ix)-18) is indicated on Figure (2)(ix)-17. The grating analysis described above indicates the strong dispersive effect of gratings and examination of the BFS arrangement drawings indicates the presence of large areas of grating in all areas except the Equipment Removal Hatch area.

Based on the grating and distributed release analyses, it has been concluded that a distributed discharge will result in relatively uniform concentrations in the wet well region.

3) Reactor Building 90 Degree Evaluations

a) Equipment Removal Hatch. An SRV release under the Equipment Removal Hatch area was simulated in detail. The results are shown on Figure (2)(ix)-19. This area was selected because it is the only region which offers a substantially unrestricted migration path from the suppression pool surface to the dome region, as discussed above in the distributed release analysis. The flow behavior observed is very similar to that seen in the free rise and grating analyses, that is, limited horizontal dispersal until grating is encountered. Based on this preliminary analysis, it has been concluded that igniters should be located as near as possible to the suppression pool surface and that other igniters should be placed above them to ensure positive ignition of any hydrogen released or drawn into this area of the containment.

b) Reactor Water Cleanup Area. A release under the RWCU area was simulated because this region of containment is a mixture of grating platforms, solid floors which extend almost the entire width of the annular space, and small subcompartments which might pocket hydrogen. The results are shown in Figures (2)(ix)-20 through -23.

The degree of pocketing and dispersal in this area is consistent with the results obtained in the grating and Main Steam Tunnel analyses described above. All of the RWCU subcompartments are isolated from the Reactor Building atmosphere by solid, normally closed doors, except the holding pump area at Elevation 641' 5" (depicted on Figure (2)(ix)-23). This area is, in effect, an enlarged walkway. If the hydrogen release is symmetric (as assumed in the analysis shown in Figures (2)(ix)-20 through -23), then hydrogen will tend to stream past this area with little or no lateral movement. A more detailed study using asymmetric release indicates that cross-drafts could be set up which might draw hydrogen into the holding pump area. This potential lateral movement is slightly evident on Figure (2)(ix)-23, which shows the start of hydrogen migration into the walkway area between the RWCU wall and the containment shell. Therefore, igniters have been placed in these areas to ensure controlled combustion and to preclude pocketing.

c) Main Steam Tunnel Area. A release under the Main Steam Tunnel was simulated beause the steam tunnel presents a large, flat expanse which is a nearly complete obstruction to upward flow. The potential for temporary pocketing offered by such an obstruction was expected to be high in this area and is confirmed by the analysis, as shown in Figures (2)(ix)-24 through -28. The actual Main Steam Tunnel construction calls for an air gap between the concrete floor and walls and the steel containment. This was simulated by leaving two cells open, as shown on Figures (2)(ix)-25 and -26. This gap may allow hydrogen to migrate into the tunnel, as shown in Figure (2)(ix)-28, to be consumed by the igniters placed in the upper areas of the tunnel.

The analysis also indicates hydrogen will readily migrate into the walkway area between the RWCU heat exchanger compartment and the steel containment. Igniters will be provided near this area to ensure controlled ignition.

4) Conclusion

Based on the preliminary evaluations described above, PSO believes that a sufficient number of igniter locations (as shown on Figures (2)(ix)-3 through -8) can be provided for reasonable assurance that controlled combustion will occur in the containment and drywell well before the localized concentrations of hydrogen could approach the detonable range.

b. Hydrogen Concentration

The uniformly distributed hydrogen concentration should not excee: 10 percent during and following an accident or that the post-accident atmosphere should not support combustion.

A pos' lated accident of the type required by the Hydrogen Contro Rule has three major periods:

- Initi' L RPV blowdown
- Hydrogen generation and release with controlled combustion
- Post-accident completion of hydrogen generation and release into an oxygen-depleted atmosphere.

The following sections describe the performance assessment of the proposed DIS during the hydrogen generation (Period 2 above) and post-accident completion (Period 3 above) periods for the two release cases considered.

1) SRV Discharge

Figures (2)(ix)-29 through -34 depict the transient hydrogen and oxygen concentrations in each of the subvolumes used in the HYBRID combustion analysis. Refer to Figure (2)(ix)-68 for a description of the subvolumes. Figure (2)(ix)-35 shows the transient uniform mixed concentrations in the containment. Inspection of these figures shows that at no time during Period 2 (hydrogen combustion) does either the localized or uniform hydrogen concentration exceed 10 percent while the atmosphere is capable of supporting combustion.

At the end of the hydrogen burning period, the post-accident containment atmosphere is a turbulent mixture of oxygen, nitrogen, water and water vapor, and hydrogen and other noncondensible gases. Continued operation of the spray system rapidly brings the containment atmosphere into temperature equilibrium with spray water. The hydrogen generation process continues to inject hydrogen into the containment, raising the uniformly mixed hydrogen concentration to approximately 28.7 percent. However, the uniformly mixed oxygen concentration has decreased to 4.4 percent, which is well below the generally recognized limit for combustion.

2) Drywell Discharge

The initial period of a drywell discharge is different than the initial period of an SRV release. The blowdown period causes the drywell to pressurize and to eventually clear the horizontal vents. This allows steam and air to enter the suppression pool, where the steam condenses and the air migrates to the containment atmosphere. At the end of the Period 1 blowdown, it is assumed that all drywell air has been transferred to the containment and the drywell atmosphere is 100 percent steam.

During Period 2, hydrogen is released to the drywell and eventually passes through the suppression pool to the containment atmosphere where it is consumed by controlled combustion. Figures (2)(ix)-36 through -41 depict the transient hydrogen and oxygen concentrations in each of the subvolumes used in the HYBRID combustion analysis. Figure (2)(ix)-42 shows the uniformly mixed concentrations in the containment. Inspection of the figures shows that at no time during Period 2 does either the localized or the uniform hydrogen concentration exceed 10 percent while the local atmosphere is capable of supporting combustion.

Continued release of hydrogen during Period 3 raises the uniformly mixed hydrogen concentration in the containment to approximately 17 percent. The drywell hydrogen concentration is approximately 6 percent. The containment hydrogen concentration is above the 10 percent limit of the Hydrogen Control Rule. However, the uniformly mixed oxygen concentration is 2.3 percent, which is well below the generally recognized limit for combustion.

c. Equipment Qualification

The burning of hydrogen in the Black Fox Station containment is expected to result in temperature spikes with high peaks but relatively short total durations. The temperature time histories for various containment subvolumes, as calculated by HYBRID, are shown in Figures (2)(ix)-43 through -47 for the single point release case and Figures (2)(ix)-48 through -52 for the drywell release case. These containment subvolumes are defined in Table (2)(ix)-6 and Figure (2)(ix)-68. No drywell temperature time histories are provided because no burns occurred in the drywell for the cases considered.

While the peak calculated temperatures are significantly above the bulk or average values for the containment which have been used to establish the existing environmental qualification limits for BFS, the effects of these repeated, short temperature pulses and the other

environmental conditions created by the burning of hydrogen
need not disqualify the existing equipment for service in
the containment or drywell. This determination can only be
made on the basis of detailed evaluations. The
qualification program will be developed and submitted for
NRC approval within two years after issuance of 19
construction permits for BFS. The results of the required
qualification program will be described in the FSAR. The
qualification program will consist of seven steps: 19

- Establish the criteria for equipment selection and identify the vital equipment list for BFS. The BWR Hydrogen Control Owner's Group has directed the General Electric Company to undertake this effort on a generic basis. The results will be used as a foundation for identifying BFS specific vital equipment. In anticipation of the Owner's Group report, PSO has performed a preliminary review of BFS and established a preliminary list of safety-related systems and components which are located inside containment and are necessary for achieving and maintaining the safe shutdown of the plant and/or maintaining containment integrity. All systems which are located, totally or partially, inside the containment vessel were considered. Those preliminarily identified in Table (2)(ix)-5 were selected on the following basis:

  Function A--System or component must function to recover the reactor core.

  Function B--System or component must function to maintain containment pressure boundary.

  Function C--System or component must function to mitigate the consequences of the post-accident events.

  Function D--Systems components whose failure could negatively affect systems or components identified as necessary in accordance with Function (A), (B) or (C).

  Function E--Systems or components whose function might be desirable, e.g., to monitor the course of the event.

  These systems and components, which will be reviewed for potential exposure to post-accident environmental conditions, are listed by function in Table (2)(ix)-5:

- Calculate the environmental parameters. This step will establish the transient temperature and pressure profiles.

- Determine the equipment parameters. This step will establish the external equipment parameters (geometry, composition, emissivity, etc.), the internal parameters (geometry and composition), the present qualification limits, the critical components and the expected failure mechanisms, and the equipment environment (location, existing thermal shielding, etc.).

- Evaluate the response of vital equipment to repeated hydrogen burns and document the qualification status of the equipment. 19

- Compare the results of the analytical models with the performance of equipment exposed to hydrogen flames. PSO notes that tests recently completed at Fenwal Laboratories on behalf of TVA indicate that typical samples of equipment and materials normally used inside the containment show resistance to the effects of repeated hydrogen burns.

- Take corrective action as necessary. Those vital equipment items for which qualification cannot be demonstrated will be upgraded. Examples of possible actions are: 19

  - Improve the equipment's resistance to surface heat transfer and pressure.

  - Provide separate thermal shielding.

  - Requalify critical components. 19

  - Relocate equipment.

  - Replacement of equipment with units of demonstrated qualification. 19

- Qualification of the DIS igniters.

d. Containment Integrity

A preliminary evaluation of the current BFS containment vessel, including vessel anchorage in the Reactor Building foundation mat, indicates, based on the Hydrogen Control Rule, that containment integrity will be maintained during a condition of an internal pressure of 45.0 psig. This pressure envelopes the peak calculated pressure generated by an accident that releases hydrogen generated by the equivalent of a metal-water reaction which consumes 100 percent of the zirconium metal in the active fuel cladding accompanied by controlled hydrogen combustion. The basis for that conclusion is set forth in the following text.

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1) Description of the Containment

The containment vessel is a free-standing fixed and vertical cylindrical steel pressure vessel with an ellipsoidal head and a flat bottom steel liner plate. The cylindrical shell is anchored into the concrete foundation mat. The cylindrical steel shell is backed by reinforced concrete in the suppression region to mitigate structural response due to the hydrodynamic effects of the suppression pool. The physical dimensions of the containment vessel are as follows:

- Inside diameter of 120'-0".
- Shell height to tangent of 153'-7".
- Ellipsoidal head with a ratio of 2:1 with an inside height of 30'-0".

The containment vessel, including all penetration sleeves, welded attachments, and the reinforced concrete backing in the suppression pool area are designed to act as an independent structural component within the Shield Building.

Anchorage of the containment vessel is accomplished by extending the vessel shell into the concrete foundation mat for an approximate distance of 6 feet.

Within the suppression pool area the bottom liner plate is a leaktight membrane which is designed to resist the hydrodynamic effects of the suppression pool. For all other areas, the bottom liner plate serves as a leaktight membrane. The liner plates are continuously supported by the foundation mat. The bottom liner plate, except in the suppression pool area, is covered by concrete which forms the internal structures to the Reactor Building and which protects the liner plate from the Reactor Building environment. A torodial knuckle plate forms the transition piece from the containment cylinder to the flat bottom plate in the suppression pool.

Major attachments and appurtenances to the containment vessel cylinder and head include two personnel air locks, an equipment hatch, polar crane girder, fluid and electric system penetration sleeves, supports for internal framing and platforms, equipment and component supports, and inspection platforms and ladders.

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The base material for the vessel shell, stiffeners, and the bottom liner plates conforms to SA 516 Grade 70. For this evaluation, the vessel shell was assumed to have a uniform shell thickness of 1-3/4 inch, which is the maximum shell thickness permitted by the ASME Code without post-weld heat treatment, and no external shell stiffeners. The actual thickness of the vessel shell and extent of the use of stiffening of the vessel will be determined during the final design process. PSO anticipates that the final vessel configuration which accommodates the design conditions outlined in PSAR Subsection 3.8.2 and these supplemental requirements will utilize thinner shell thicknesses and vessel stiffening to optimize the vessel design.

2) Applicable Codes, Standards, and Specifications

a) Codes, Standards, and Specifications. In order to conform with the Hydrogen Control Rule, the following codes, standards, and specifications are used in this evaluation:

- ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1980 Edition with Addenda through summer 1980, Subarticle NE-3220, Service Level C limits, except that evaluation of instability is not required, considering pressure and dead load alone.

- ASME Boiler and Pressure Vessel Code, Section III, Division 2, 1980 Edition with Addenda through summer 1980, Subarticle CC-3720 Liner (Factored Load Category only).

3) Structural Acceptance Criteria

For this evaluation, the structural acceptance criteria for the steel containment vessel are based on the limits for primary stresses defined in Subarticle NE-3220, Service Level C. The following allowable limits are considered:

Primary Stress	
General Membrane	Larger of 1.2 S_m or 1.0 S_y
Local Membrane	Larger of 1.8 S_m or 1.5 S_y
Bending plus Local Membrane	Larger of 1.8 S_m or 1.5 S_y

Where:

S_m is the allowable stress intensity for the steel material

S_y is the minimum yield strength for the steel material

The structural acceptance criteria for the bottom liner plate are based on the allowables as defined in Subarticle CC-3720, Liners, considering the allowables for factored load conditions.

For concrete and reinforcing steel in the Reactor Building foundation mat and the reinforced concrete backing in the suppression pool, the acceptance criteria are based on the allowables as defined in Subarticle CC-3420, Allowable Stresses for Factored Loads. In particular, the allowable stresses for compression, shear, and bearing in the concrete are as specified in paragraph CC-3421. The allowable stresses for tension and compression in reinforcing steel are as specified in paragraph CC-3422.

4) Loads and Load Combinations

For the evaluation performed in response to the Hydrogen Control Rule, the following loads and load combinations are considered:

a) Containment Internal Pressure. The containment internal pressure P_a' is the pressure which results from either of the two following conditions, whichever produces the larger load effect for the component being considered.

- Required Pressure. The minimum required containment, static pressure is 45 psig.

- Calculated Internal Pressure. The calculated internal pressure for the Black Fox Station containment is the pressure which occurs during an accident that releases hydrogen generated by the equivalent of a metal-water reaction which consumes 100 per cent of the zirconium metal in the active fuel cladding accompanied by hydrogen combustion. The pressure is a time-dependent function. The pressure time-history is computed using the methods discussed in Section 3 and 5 of this response. The containment model consists of six volumes: two in the suppression pool and wet well region, two in the subcompartment area directly above the respective suppression pool region, one for the volume above the refueling floor, and one for the drywell. Pressure time-histories, based on the combustion analyses, were computed for each volume for both the single release point and the distributed (axially symmetric) release point cases.

  Figures (2)(ix)-53 through (2)(ix)-58 show the preliminary pressure time histories for the six volumes inside the containment for the SRV release case. The pressure wave forms are characteristically overpressure impulses of approximately 5 to 10 seconds in duration. The maximum observed peak pressure is approximately 16.8 psig (31.5 psia) and occurs as the result of a burn in the suppression pool area. The pressure time histories for each compartment during the period when this peak pressure occurs have been superimposed and are presented on Figure (2)(ix)-59. The total period of the impulse is approximately 10 seconds. The fundamental period of the steel containment vessel is approximately 0.06 seconds. Therefore, the relationship of the forcing function to the dynamic characteristics of the vessel indicate that the effect of the pressure impulse is quasi-static and can be compared directly to the design pressure stipulated in "Minimum Required Pressure" subsection described above.

In addition, inspection of Figure (2)(ix)-59 indicates that significant pressure differentials, i.e., 1.0 psid, do not exist between the various containment volumes. Therefore, asymmetric pressure distributions due to burning of the hydrogen are negligible. The calculated peak pressure of 16.8 psig (31.5 psia) is below the required pressure of 45 psig specified by the Hydrogen Control Rule.

Figures (2)(ix)-60 through (2)(ix)-65 show the preliminary pressure time histories for the six volumes inside the containment vessel for the drywell release case. The pressure wave forms are characteristically over-pressure impulses of approximately 5 to 10 seconds duration. The maximum observed peak pressure is 27.8 psig (42.5 psia) in the containment and 29.3 psig (44 psia) in the drywell. Due to the dynamics of vent clearing, these drywell and containment pressure transients are separated slightly in time, producing a maximum drywell-to-containment pressure differential of 5.5 psid which is significantly below the 30.0 psid design pressure for the drywell. These peaks occur as the result of a burn in the containment region. The pressure time histories for each compartment during the period when this peak occurs have been superimposed and are presented in Figure (2)(ix)-66 for the containment burn. The shape of this curve is very similar to that resulting from the SRV release, i.e., the pressure time-histories are quasistatic and pressure differentials are negligible. The calculated peak pressure of 27.8 psig is below the minimum required pressure of 45 psig specified by the Hydrogen Control Rule.

b) Dead Loads. The deadloads (D) consist of the following:

- Weight of the steel of the containment vessel and its appurtenances.
- Crane weight.
- Empty weights of attached piping.
- Weight of electrical connections, mechanisms, ladders, and platforms contributory to the containment vessel shell.

In addition, an equivalent hydrostatic pressure of 25 feet 10 inches in the suppression pool area, corresponding to the suppression pool inventory following the upper pool dump which occurs with the Loss of Coolant Accident, is considered.

c) Load Combinations. The following supplemental load combination applies to this evaluation.

(1) $D + P_a'$

This load combination is considered in addition to the requirements of the ASME Code, Section III, Subarticles CC-3000 and NE-3000, and Regulatory Guide 1.57.

5) Design and Analysis Procedures

a) Steel Containment Vessel. The analysis of the containment vessel was carried out by using the containment vessel model developed by Chicago Bridge and Iron Company (CBI). This model is based on a proprietary finite element computer code, CBI Program Z1374, for the solution of problems involving shells of revolution. This program calculates the deflections, forces, moments, and stresses for each output point in the model.

b) Reinforced Concrete in the Suppression Pool Area. The evaluation of the reinforced concrete in the suppression pool area was performed using finite-element computer code, Black & Veatch Program 373. In this evaluation, shell elements are used to represent the steel containment vessel and axisymmetric quadrilateral elements are used for reinforced concrete backing in the suppression pool region. The program calculates the time histories and the maximum values for displacements, forces, moments, and stress for each output point.

6) Results

A preliminary evaluation of the Black Fox containment vessel indicates that the containment integrity can be maintained within the acceptance criteria outlined in Subsection C.2.d.3) (page 302) when the containment vessel is subjected to the required pressure of 45 psig. As indicated above, the required pressure envelops the effects of the peak pressure resulting from an event that releases hydrogen generated by the | 19 equivalent of a metal-water reaction which consumes 100 percent of the zirconium metal in the active fuel cladding accompanied by controlled combustion. Therefore, this preliminary evaluation satisfies the requirements of the Hydrogen Control Rule Subpart (A) to Hydrogen Control Rule (3)(v).

D. DETAILED ANALYTICAL METHODOLOGY

1. INTRODUCTION

The analytical approach used to assess the performance adequacy of the proposed DIS consists of three major parts. These are discussed in detail in the following sections.

In summary, the MARCH computer code, supplemented by hand calculations, was used to derive the mass, energy, and hydrogen release rates to be used. Using the hydrogen release rates as input, the modified SOLA-DF computer code was used to evaluate the potential for pocketing under idealized conditions and without considering combustion-induced turbulence. The HYBRID computer code was used to determine the pressure and temperature response of the containment environment to controlled combustion.

2. HYDROGEN GENERATION RATES

PSO has performed a preliminary parametric analysis and has used the results of this analysis as a basis for selecting the release rate shown in Tables (2)(ix)-2 and (2)(ix)-3.

a. Computer Code

The only publicly available computer code to analyze the combined phenomena reactor heat-up, boil-off, and hydrogen production under degraded core conditions is MARCH[1]. MARCH was developed by Battelle-Columbus for the Probabilistic Analysis Branch of the NRC Staff. The development of the MARCH code is an extension of the meltdown analysis work performed by Battelle-Columbus for the Reactor Safety Study[3] in which the original BOIL[4] code, a subroutine of MARCH, was developed. Most of the models used in the BOIL subroutine of MARCH are the same as those reported in the Reactor Safety Study.

1) Description of MARCH Model

The BOIL subroutine calculates core heat-up in an accident where the fission-product decay heat boils the water out of the pressure vessel and uncovers the core. The reactor water volume is divided into two regions: steam and liquid. The core is divided into small volumes or nodes. Calculations are performed to determine the heat produced in each node by performing heat balances between the fuel and coolant. A steam boiloff rate and the water-steam mixture level in the reactor core is also calculated.

The reactor core in the Black Fox calculations was modeled in the BOIL subroutine using 10 radial and 24 axial power zones. The appropriate axial and radial power distributions were used to simulate the axial and radial region power peaking factors. Core nodes in the mixture region are assumed to be well cooled. Nodes in the steam space are convection cooled by the steam boiling out of the mixture regions.

The BOIL subroutine models radiation heat transfer from the top fuel nodes in the core to structures above the core and from core nodes just above the mixture to the water region. Four heat structures were modeled above the core for the Black Fox calculation. These heat structures represented the nonactive top of core; the core shroud dome and steam separators; the steam dryer; the reactor pressure vessel steel; and miscellaneous piping. Three heat structures were modeled directly below the core. These heat structures represented the nonactive bottom of core; the guide tubes and shroud support legs; and reactor pressure vessel steel.

2) Discussion of the MARCH Model

The MARCH computer code uses conservative assumptions and approximations to model the behavior of a reactor core and containment system undergoing a degraded core accident. The result is that calculations using MARCH are expected to be conservative with regard to the rate and amount of hydrogen generated prior to significant core degradation.

The BOIL subroutine uses the Dittus-Boelter correlation to model forced convection steam cooling heat transfer. The Electric Power Research Institute (EPRI) and the NRC have undertaken an extensive experimental program[5] to determine the actual heat transfer mechanisms and to establish both best-estimate and licensing bases for modeling core heat removel. The results obtained to date[5] indicate that for the conditions expected to exist in a core undergoing significant hydrogen generation, the Dittus-Boetler correlation is conservative. That is, the steam is expected to be more effective at cooling the rods than predicted by the MARCH code and, therefore, the rate of hydrogen generation should be lower than predicted.

The zirconium metal-water reaction rate is modeled in the BOIL subroutine using the Baker-Just gaseous diffusion formulation.[6] In each of the fuel nodes, the metal-water reaction is generally a two-step process which is initially controlled by the gaseous diffusion of water vapor toward the hot fuel rod and by the gaseous diffusion of the hydrogen away from the fuel rod. At a later time, as determined by the diameter of the fuel rod, the thickness of the oxidized layer, and the temperature of the steam and fuel rod, the reaction rate becomes controlled by the solid-state diffusion of oxygen into the cladding. The rate at which the thickness of the oxidized layer increases when solid-state diffusion controls is calculated by the Baker-Just solid-state diffusion formulation. The use of the Baker-Just diffusion correlations has been found to predict twice the rate of hydrogen production as obtained in experimental results.[7]

MARCH calculations generally predict[8] that 30 to 60 percent of the active fuel cladding is oxidized during fuel heat-up, prior to the time the core collapses. The range of cladding oxidation results from uncertainties in modeling assumptions and the type of fuel heat-up being analyzed. Additional cladding oxidation may occur when the core collapses into the lower plenum water of the reactor vessel. This additional oxidation would occur very quickly and generally does not produce a large amount of additional hydrogen since the water in the lower plenum would quickly cool the cladding, thereby quenching the reaction.

b. Parametric Analysis

To evaluate the effects of break size on the hydrogen generation rate, three different breaks were postulated.

- A 0.163 ft^2 steam line break, equivalent to a full-open SRV.
- A 2-inch steam line break.
- A 1-inch line break equivalent in size and location to an RPV instrument line, so that the blowdown would be

subcooled liquid until the water level had dropped below the level of the break.

Cumulative hydrogen generation curves as a function of time for each break are shown in Figure (2)(ix)-67, with the times normalized to the start of hydrogen generation. The curve for the 0.163 ft^2 break envelopes the other two curves over most of the range.

3. HYDROGEN MIGRATION ANALYSIS

The possibility of localized high concentrations of hydrogen (pocketing) in subcompartmented containment structures has been identified as an item of concern in the Hydrogen Control Rule. To address this concern, PSO has performed a preliminary hydrogen migration analysis for the BFS containment. The results of the analysis are presented in this response.

a. Objectives of the Analysis

- To determine the rate of hydrogen buildup in various containment subcompartments.
- To evaluate the potential for hydrogen maldistribution and pocketing.
- To provide a rationale for selecting the number and location of igniters.
- To assist in developing an igniter control philosophy.

b. Computer Code

The evaluation of multi-component gas flows featuring both asymmetric (SRV) and axially symetric (drywell vent) discharges into a subcompartmented closed structure was performed using a modified version of SOLA-DF[9], a public domain solution algorithm for nonequilibrium two-phase flow. In short, SOLA-DF is a finite difference code which uses the implicit continuous fluid Eulerian method to solve the mass, momentum, and energy equations which describe the system under evaluation. To provide a more complete and flexible analysis of the hydrogen migration problem, the original SOLA-DF code has been modified to include three-dimensional capability, rectangular as well as cylindrical coordinates, and the necessary constitutive relationships to describe the behavior of hydrogen-air mixtures.

For each of the cases described below, the containment was the area of interest, since the drywell is effectively sealed to ordinary gaseous inflows. Hydrogen gas (all steam was assumed to be condensed) was released at a temperature of 100° F into an initially quiescent atmosphere also at a temperature of 100° F. No heat transfer between the air and the various structures or the suppression pool was permitted. The release rate for hydrogen was the same as that used for the combustion analysis. The area of release for the single point release was twice the circumscribed area of the SRV quenchers, or about 160 160 ft .

The dispersive effects of buoyancy, momentum, convection, diffusion, temperature, pressure, and gravitation were included in the analysis. Turbulence induced by the sprays and the controlled combustion of hydrogen were not included in this analysis. Neglecting these effects in the migration analysis is a conservative assumption as these effects are expected to increase hydrogen dispersal and thereby further reduce the potential for local pocketing.

PSO has performed a high temperature release analysis, in which a "hot" hydrogen release was simulated by discharging the hydrogen at 1642° F into an atmosphere at 100° F. The upward velocity of the hydrogen increased by a factor of about 3 over the low temperature case and horizontal dispersal was reduced.

c. Description of Completed Cases

To meet the objectives of the preliminary analysis, the following cases have been evaluated:

- Three-dimensional evaluation of pocketing due to grating in the annular volume between the suppression pool and the refueling floor resulting from single point discharge of hydrogen gas in thermal equilibrium with the suppression pool.

- Three-dimensional 360 degree evaluations of the containment without fans operating. A single point release under the RWCU equipment area was simulated. For the drywell release case, uniform discharge through the vents was simulated.

- Three-dimensional, 90 degree evaluations of the annular region for three single point releases. These were the RWCU equipment area, the Main Steam Tunnel area, and the Equipment Removal Hatch area. These finely nodalized studies provide a detailed evaluation of the pocketing potential in both open and congested areas.

4. HYDROGEN COMBUSTION ANALYSIS

The BFS containment pressure/temperature analysis was done using the Computer Code HYBRID. The HYBRID code models the suppression pool and vent flow between the drywell and suppression pool. Other features of the code include a variable heat transfer spray model, entrained water fallout model, heat transfer to specified heat sinks, and the simulation of various engineering safeguard equipment such as fans and heat exchangers.

HYBRID can simulate the multicomponent (H_2, O_2, N_2, and CO_2) gas and two-phase fluid transfer between compartments due to the burning of hydrogen and/or due to the mass and energy release from a pipe break. The computer code can simulate multicompartment (up to 100 volumes) transient pressure and temperature responses and track the distribution of the noncondensible gases.

The model used to determine the pressure/temperature responses due to controlled burning by the DIS was a multi-node model which divides the BFS Reactor Building into discrete volumes based on flow area and natural divisions to flow. The nodal diagram is presented on Figure (2)(ix)-68 and the associated volume descriptions are given in Table (2)(ix)-6. The HYBRID model contains six compartments, a suppression pool at the bottom of Volumes 5 and 6, vents connecting the drywell to Volumes 5 and 6, containment spray with spray carry-over and a vacuum breaker simulation. | 19

The flow paths connecting the compartments are represented as shown on Figure (2)(ix)-68 by arrows pointing in the direction of allowed flow. The junction (flow path) parameters are presented on Table (2)(ix)-7. The junction flow areas represent the minimum flow area of the connecting compartments. The junction loss coefficient calculations were done by evaluating the obstruction losses using the handbook by Idel'chik[10]. Included in the loss coefficient calculations are the losses through grating and other losses due to miscellaneous obstructions. Effects of flow inertia were also included in the HYBRID calculations.

The vents and suppression pool and related parameters are shown in Table (2)(ix)-8. Included are the total volume of water at the normal water level, pool surface area in the wet well and drywell, normal pool height above basemat floor, and the drywell weir height above the normal water level. Other related parameters include specifications for the vents, drywell holdup volume and upper pool dump parameters.

The drywell vacuum breaker is represented as a one-way flow path in the diagram presented on Figure (2)(ix)-68. The drywell vacuum-relief line opening is a function of the vacuum breaker valve, butterfly valve, and associated controls. At 2.0 psid (containment to drywell), or at 0.2 psid (containment to drywell) if the drywell pressure is above 2.0 psig, a signal is generated to open the butterfly valve. After a 3 second delay, the butterfly valve opens within 5 seconds and remains open until closed by operator action. Once the butterfly valve is open, the vacuum breaker valves are activated. At 0.2 psid (containment to drywell) the magnetic latch on the vacuum breaker valve releases and the disc immediately swings wide open. The disc remains fully open (area is 0.5475 ft^2) until the differential pressure falls to about 0.1 psid. Below 0.1 psid, the disc is partially open until it reseats at approximately 0.02 to 0.03 psid. The equivalent loss coefficient for the vacuum-relief lines is 5.51.

The containment spray system parameters used in the BFS HYBRID calculations are presented in Table (2)(ix)-9. The spray is released from two spray rings located in the containment dome. The inner ring is located approximately 59 feet above the operating floor and the outer ring is located approximately 48 feet above the operating floor. The spray falls from the spray rings through the containment dome until the spray pattern is disturbed by various obstructions. These obstructions consist of storage pools, the reactor well, drywell head storage area, reactor head storage area, RWCU heat exchanger removal hatches, and gratings. Part of the spray will collect in the upper pool and is assumed to drain directly into the suppression pool. A large part of the spray will strike the obstructions and agglomerate forming large masses or sheets of water which either flow directly down into the lower compartments or run down the walls of the compartments. It is assumed for these calculations that a small fraction of the spray remains as the initially specified spray droplets and the effective carry-over spray flow fraction was conservatively estimated at 10 percent.

The passive heat sinks used for these calculations were the containment steel shell adjacent to the containment and associated compartments and the concrete and steel in the drywell. Only steam condensing heat transfer was taken into account, using the Uchida correlation. Radiant heat transfer was not considered in these analyses. Neglecting radiant heat transfer produces higher compartment temperatures to minimize the heat transfer to the structures, as was done in these analyses.

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Two analyses were performed to determine the effectiveness of the DIS to reduce the hydrogen concentration, namely the stuck-open SRV which releases the hydrogen directly into the suppression pool, and the same release directed to the drywell (see Figure (2)(ix)-68 for release points with respect to the HYBRID burn model). The mass and energy release rates for water and hydrogen used for both analyses are presented in Tables (2)(ix)-2 and (2)(ix)-3, respectively.

The results of the HYBRID burn analyses are presented in Section C. of this response.

PSO RESPONSE: 10 CFR 50.34(e)(2)(ix)/(3)(v)

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TABLE (2)(ix)-1
LIST OF TOPICS TO BE ADDRESSED
IN EVALUATION TO BE SUBMITTED TO NRC
TWO YEARS AFTER ISSUANCE OF CONSTRUCTION PERMITS

1. Hydrogen Generation Rates
2. Igniter Performance
3. Spray Effectiveness
4. Hydrogen Mixing
5. Accident Sequences
6. Combustion Characteristics 19
7. Single Failure Assumptions
8. Sensitivity Studies on Hydrogen Burn Analysis
9. Potential and Consequences of Local Detonations
10. Analysis to Demonstrate That Containment Pressure Will Not Exceed Service Level C Limits
11. Equipment Qualification

TABLE (2)(ix)-1a
REVIEW OF COMPLETED, ONGOING, AND PLANNED TESTS

Conductor	Sponsor	Status	Goal
TVA	TVA	Completed and Ongoing	Examine performance of GM glow plug, and conduct life-time tests on various igniters.
Fenwal	AEP/DUKE/TVA	Completed	H_2 combustion tests designed to look at conditions simulating in containment environment (mostly quiescent chamber tests)
LLNL	NRC	Completed	H_2 combustion tests to look at LFL under high steam loadings in a well-mixed quiescent chamber.
AECL	EPRI/AEP/ DUKE/TVA	Ongoing	Effect of turbulence on H_2 combustion; igniter effectiveness studies (mostly well-mixed chamber tests)
HEDL	EPRI/AEP/ DUKE/TVA	Ongoing	Mixing, stratification, and distribution of H_2 following LOCA accident (very large chamber but not combustion)
ACUREX	EPRI/AEP/ ACUREX/DUKE/ TVA	Ongoing	Igniter location effect during dynamic injection of H_2; suppression characteristics of microfogs during dynamic injection of H_2; equipment survivability.
SANDIA	NRC	Ongoing	Basic and applied research into H_2 combustion; effects of microfogs in well-mixed quiescent chambers.
FMRC	EPRI/AEP DUKE/TVA	Ongoing	LFL of H_2 in the presence of microfogs.
VARIOUS	TEC (IDCOR)	Planned	Develop adequate technological basis for decision making for Degraded Core Rulemaking.

TABLE (2)(ix)-2
BLACK FOX REACTOR COOLANT MASS AND ENERGY RELEASE RATES

Time (Min)	Mass Release Rate (lbm/min)	Energy Release Rate** (Btu/Min)
0.0	1.36×10^4	1.62×10^7
.106	2.48×10^4	2.95×10^7
5.94	2.48×10^4	2.95×10^7
6.0	1.51×10^4	1.83×10^7
10.0	1.29×10^4	1.54×10^7
14.0	9.31×10^3	1.16×10^7
18.0	5.8×10^3	7.80×10^6
22.0	3.77×10^3	5.40×10^6
26.0	2.61×10^3	3.92×10^6
30.0	1.88×10^3	2.96×10^6
34.0*	3.58×10^3	4.27×10^6
38.0	4.82×10^3	5.76×10^6
42.0	6.15×10^3	7.34×10^6
46.0	7.19×10^3	8.59×10^6
50.0	7.98×10^3	9.53×10^6
54.0	7.21×10^3	8.61×10^6
58.0	7.54×10^3	9.00×10^6
62.0	6.62×10^3	7.90×10^6
70.0	4.96×10^3	5.92×10^6
84.2	4.91×10^3	5.86×10^6
84.21***	2.09×10^3	2.50×10^6
150.0	1.65×10^3	1.97×10^6

*March output modified at 34 minutes to include enough ECC flow to remove energy produced by decay heat and metal-water reaction.

**The fission product decay heat was not identified separately since the total decay heat was included in the energy release rates.

***Hydrogen generation ends.

TABLE (2)(ix)-3
BLACK FOX HYDROGEN RELEASE RATES AND TEMPERATURES

Time (min)	Hydrogen Release Rate (lbm/min)	Temperature (F)
0.0	0.0	534
18.0	1.47×10^{-3}	772
26.0	.449	1,001
30.0	1.2	1,240
34.0	14.8	1,642.0*
38.0	38.4	1,642.0
42.0	63.3	1,642.0
46.0	81.7	1,642.0
50.0	97.9	1,642.0
54.0	85.3	1,642.0
58.0	92.0	1,642.0
62.0	76.4	1,642.0
70.0**	48.5	1,642.0
84.2	48.5	1,642.0
84.21	0.0	1,642.0
150.0	0.0	1,642.0

*Hydrogen temperature was assumed constant when core begins to melt.

**At 70.0 minutes the metal-water reaction becomes severely limited by the amount of flashing steam. From this point onward, the reaction rate was assumed to be constant until all of the active zirconium clad had reacted.

TABLE (2)(ix)-4
BLACK FOX STATION BASE CASE COMBUSTION PARAMETERS

Hydrogen Lean	
Volume percent H_2 for initiation	8.0 percent
Volume percent O_2 for initiation	5.0 percent
Volume percent H_2 for propagation	greater than 0.0 percent (upward), greater than 9.0 percent (horizontal and downward
Flame speed	6 feet per second
Burnup (of available hydrogen)*	85 percent
Burnup (of available oxygen)*	100 percent
Steam effects	Dilutent only**
Oxygen Lean (Dry Well Only)	
Volume percent H_2 for initiation	Less than 90 percent
Volume percent O_2 for initiation	5.0 percent
Volume percent steam for initiation	Less than 60 percent
Flame speed	6 feet per second
Burnup (of available oxygen)*	100 percent

*For individual ignitions, stoichiometry is maintained. For example, if there is sufficient oxygen to initiate a burn but insufficient to consume all available hydrogen, the total burnup is limited by the available oxygen.

**Steam and water vapor are treated as dilutents for combustion purposes. The effects of water as a heat-absorbing material are included in calculating the pressure and temperature response.

TABLE (2)(ix)-5

SYSTEMS, COMPONENTS, OR STRUCTURES REQUIRED FOR
SAFE SHUTDOWN AND MAINTAINING CONTAINMENT INTEGRITY

System	Function*				
	(A)	(B)	(C)	(D)	(E)
Automatic Depressurization System	X		X		
RHR System					
Containment Spray Mode		X			
Suppression Pool Cooling Mode		X			
Shutdown Cooling Mode			X		
Standby Service Water System				X	
Containment and Reactor Isolation Systems	X	X			
Containment Vacuum Relief		X			
Drywell Vacuum Relief			X		
Suppression Pool Makeup		X			
MSIV Leakage Control		X			
Distributed Ignition System		X			
Hydrogen Recombiners			X		
Post Accident Monitoring System**					X
Containment Atmospheric Monitoring System					X
High Pressure Core Spray	X				
Low Pressure Core Spray	X				
Electric Power Distribution (Cable)				X	
Standby Gas Treatment System			X		

*The functions A, B, C, D, and E are defined in the text of Section C.2.c.

**Examples of desirable post-accident monitoring instrumentation located inside containment include: containment pressure, reactor water level, suppression pool water level, and hydrogen monitoring instrumentation.

TABLE (2)(ix)-6
BLACK FOX COMPARTMENT DESCRIPTIONS

Volume Number*	Volume (ft^3)	Temperature (F)	Relative Humidity	Initial Pressure (psia)	Description
1	551,628	90	.70	14.7	Containment Dome Above El 666'-5"
2	274,310	135	.70	14.7	Drywell
3	142,911	90	.70	14.7	Containment between El 610'-4" and El 666'-5" from Az. 46 degrees to Az. 314 degrees
4	49,587	90	.70	14.7	Containment between El 666'-5" and El 610'-4" from Az. 314 degrees to Az. 46 degrees
5	126,115	90	.70	14.7	Containment between El 610'-4" and top of water from Az. 46 degrees to Az. 314 degrees
6	43,759	90	.70	14.7	Containment between El 610'-4" and top of water from Az. 314 degrees to Az. 46 degrees

*See Figure (2)(ix)-68 for nodal diagram.

TABLE (2)(ix)-7
DESCRIPTION OF BLACK FOX JUNCTION PARAMETERS

Junction Number*	$Volume_i$	to $Volume_j$	Minimum Area (ft^2)	K_{ij}**	K_{ji}**	L/A, ft^{-1}
1	1	3	1,251	3.05	3.25	.0288
2	1	4	402	3.05	3.25	.0879
3	3	5	1,053	1.84	1.72	.008
4	4	6	366	1.84	1.72	.023
5	3	4	251	.434	.370	.233
6	5	6	388	.042	.040	.161

* See Figure (2)(ix)-68.

** Loss coefficients are based on minimum area.

TABLE (2)(ix)-8
BLACK FOX SUPPRESSION POOL AND RELATED PARAMETERS

Pool Water			
Density (lbm/ft^3)		62.2	
Volume (ft^3)		133,672	
Temperature (F)		100	
Heat capacity (Btu/lb-F)		1.0	
Pool surface area in wet well, ft^2		5,899	
Pool surface area in drywell, ft^2		482	
Normal pool height above basemat floor, ft		20.2	
Weir height above water level, ft		5.92	
Vents	Row 1	Row 2	Row 3
Number of vents	40	40	40
Flow area per vent, ft^2	4.12	4.12	4.12
Vent length, ft	5.0	5.0	5.0
Water depth at vent bottom, ft	8.4	12.9	16.3
Equivalent vent length added for inertia effects, ft	2.86	2.86	2.86
Turning loss coefficient	1.2	1.2	1.2
Dry Well			
Holdup volume*, ft^3		44,855	
Holdup surface area, ft^2		2,836	
Upper Pool**			
Volume dumped to suppression pool, ft^3		34,150	
Dump time, minutes		6	
Water temperature, F		100	

*Net free volume below and inside top of weir wall in drywell.

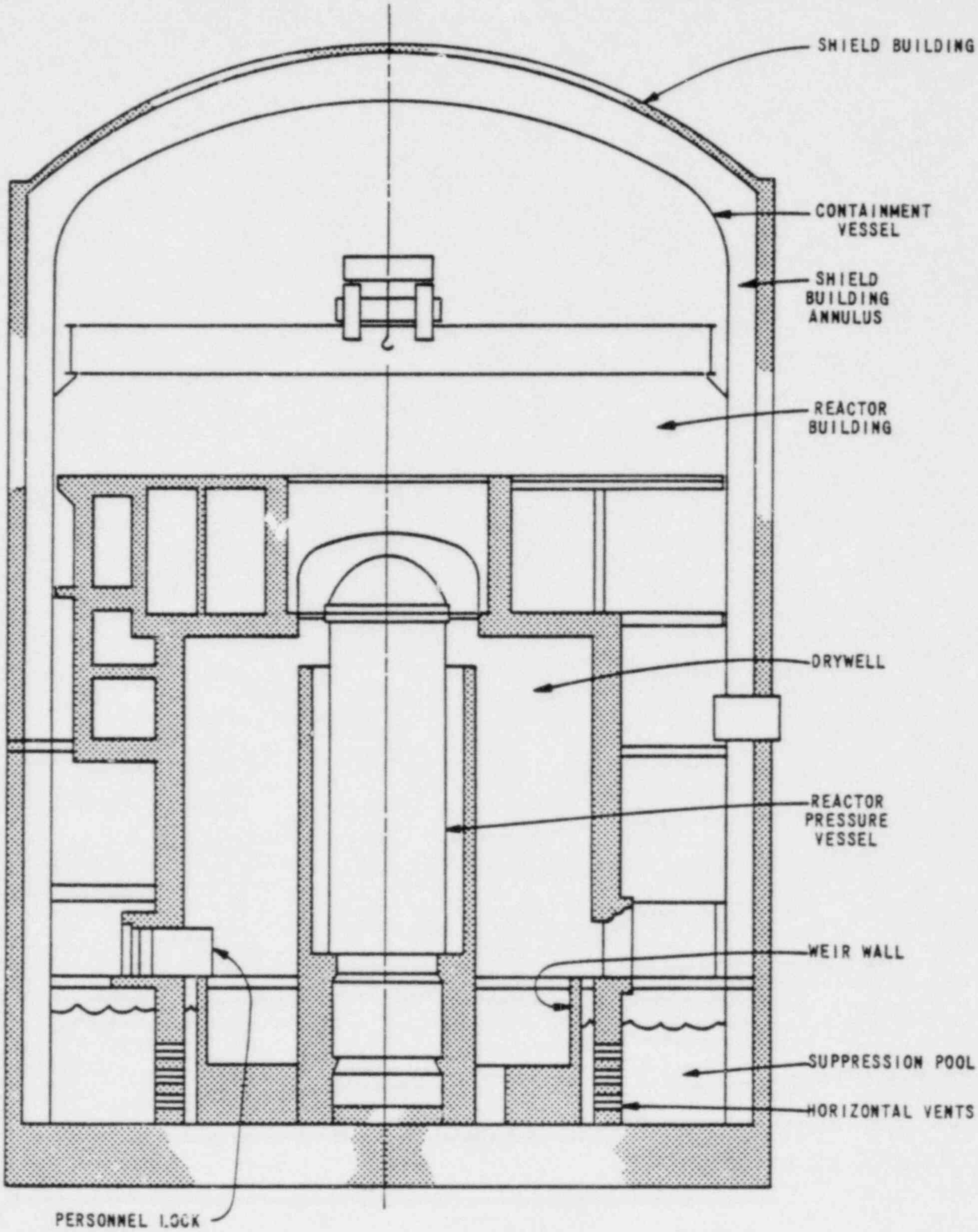
**The upper pool will be dumped automatically 30 minutes after a LOCA or after a LOCA when the suppression pool level drops to low-low water level.

TABLE (2)(ix)-9
BLACK FOX SPRAY SYSTEM PARAMETERS

Flow rate, gpm per spray loop	5,250 (See Note 1)
Temperature, F	132
Drop diameter, microns	350 (See Note 2)
Fall time, seconds	See Note 3
Heat transfer coefficient, Btu/h-ft^2-F	See Note 4
Containment dome to lower compartments carry-over fraction	.1
Initiation	See Note 5
Time to attain full flow, minutes	3
Termination	Operator Action

Notes:

1. Only one loop was assumed operational for these calculations.
2. The arithmetic mean drop diameter for a Spraco 1713A nozzle.
3. Fall time is dependent on local conditions in the compartment and height of compartment.
4. This value is based on the local conditions of the compartment.
5. Spray initiation is when 10 minutes have elapsed after the drywell reaches 2 psig, or the spray will be automatically initiated if the containment pressure is greater than or equal to 9 psig, or manually initiated by the operator.



MARK III CONTAINMENT
Figure (2)(ix)-1

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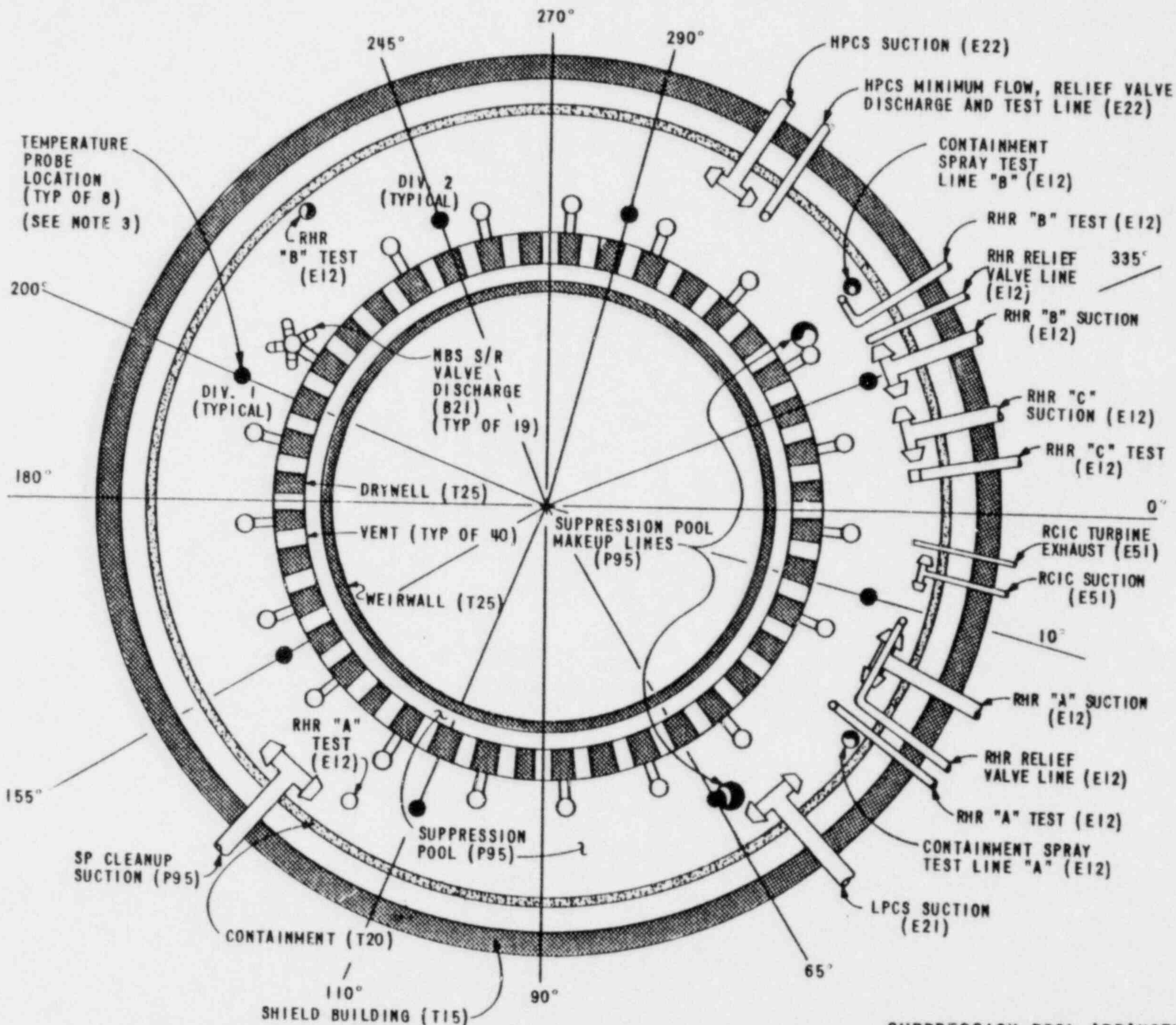
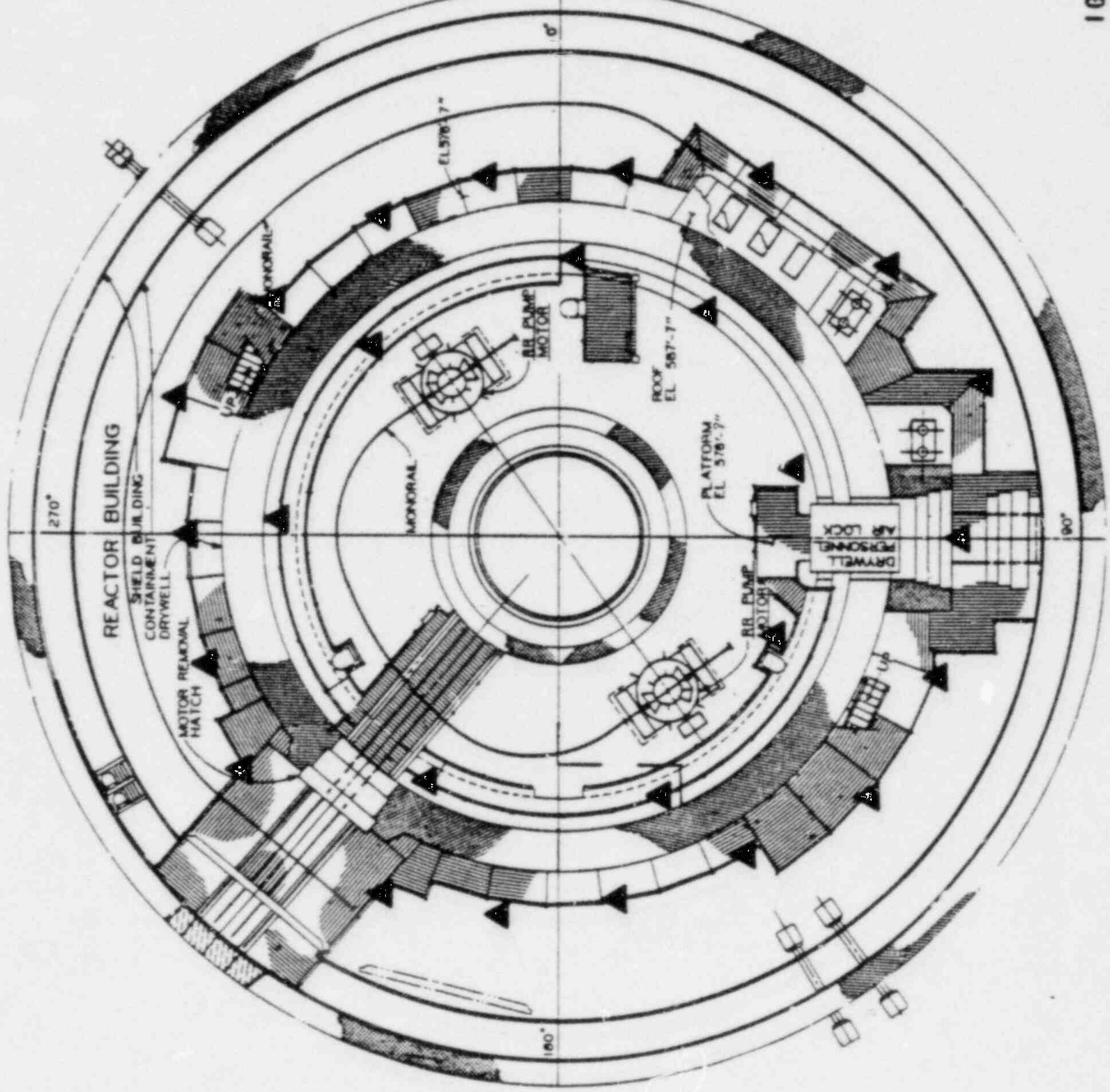


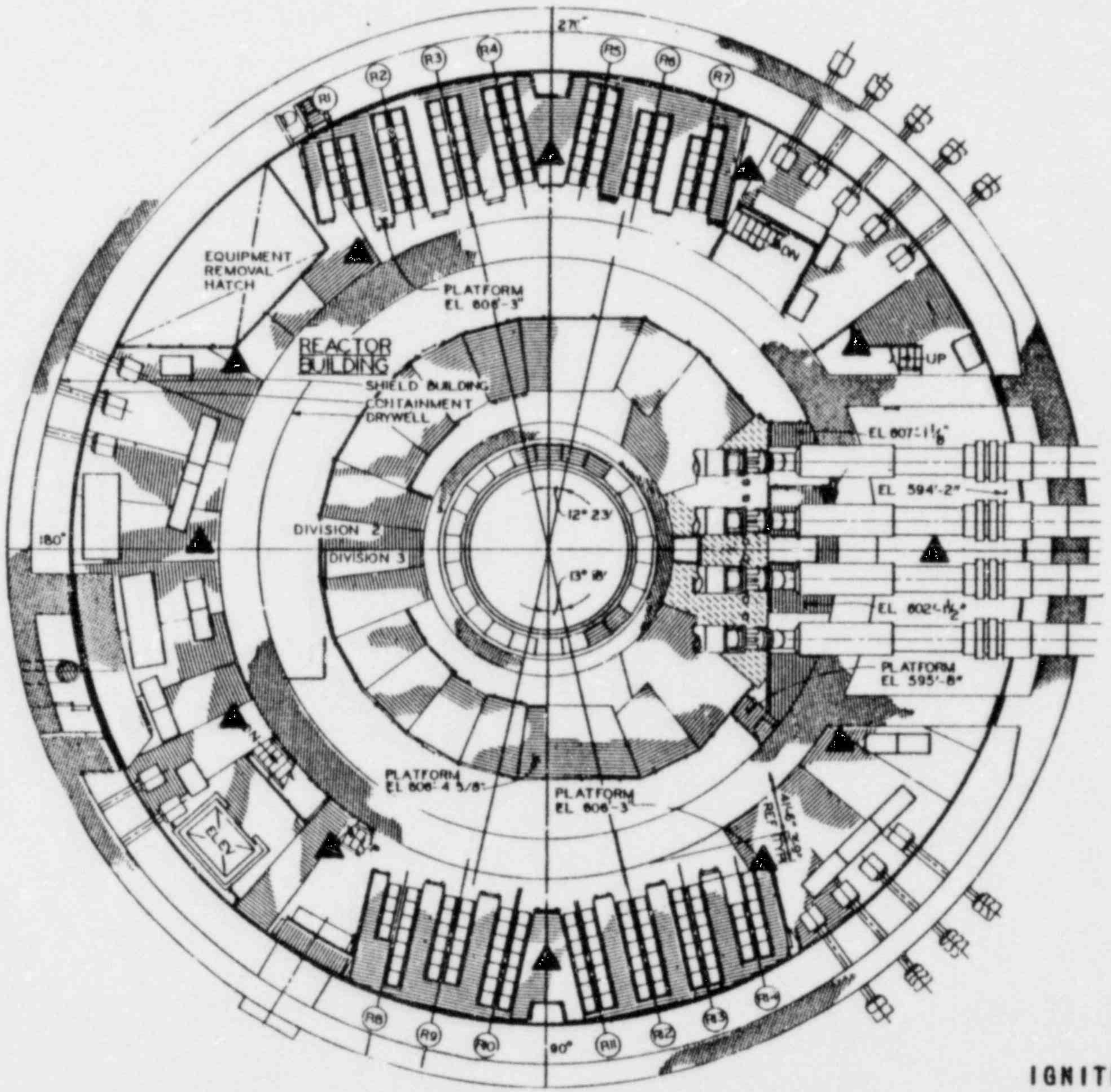
SUPPRESSION POOL ARRANGEMENT

Figure (2)(1x)-2



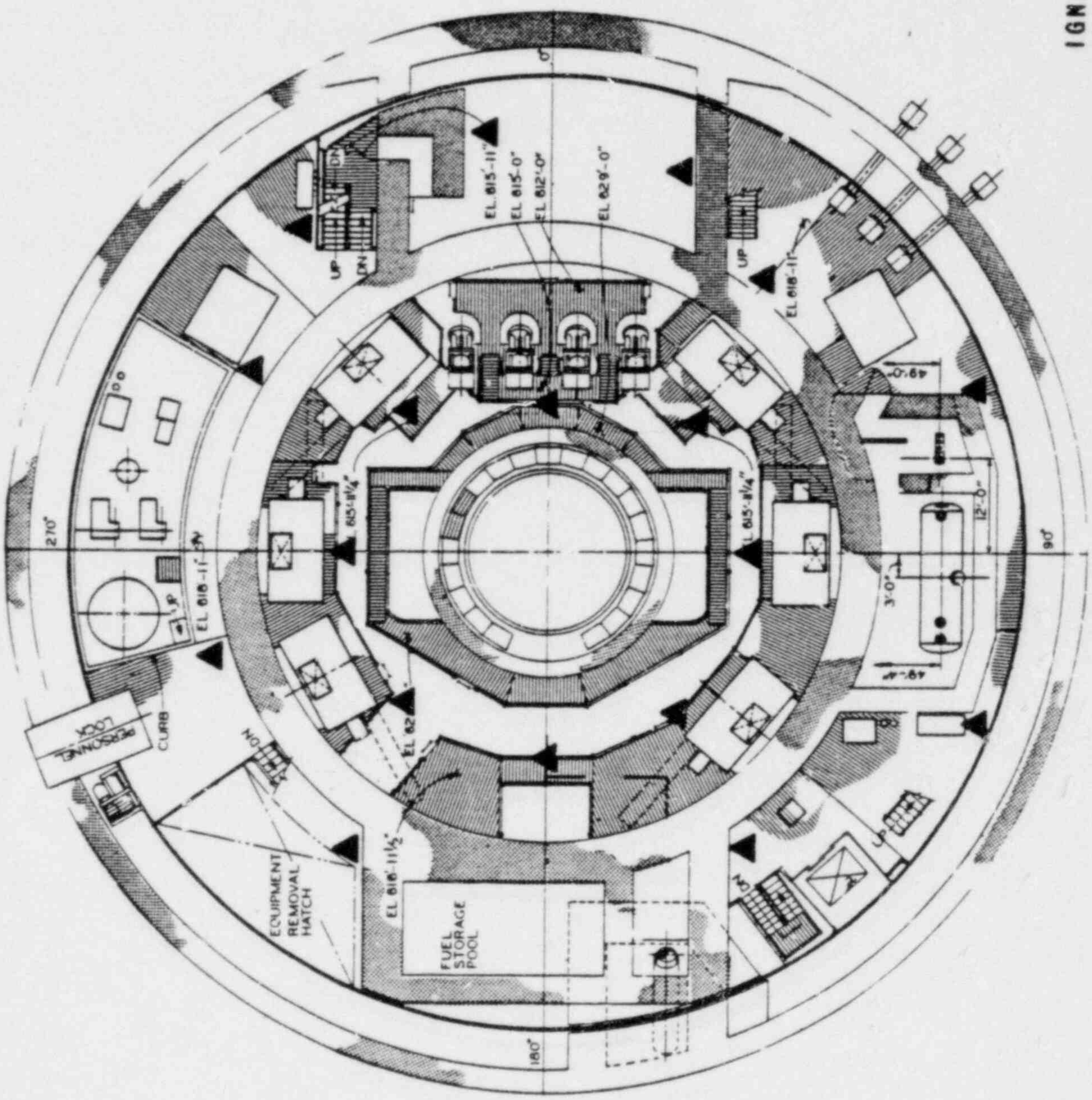
IGNITER LOCATIONS AT ELEVATION 575'-0"

Figure (2)(1x)-3



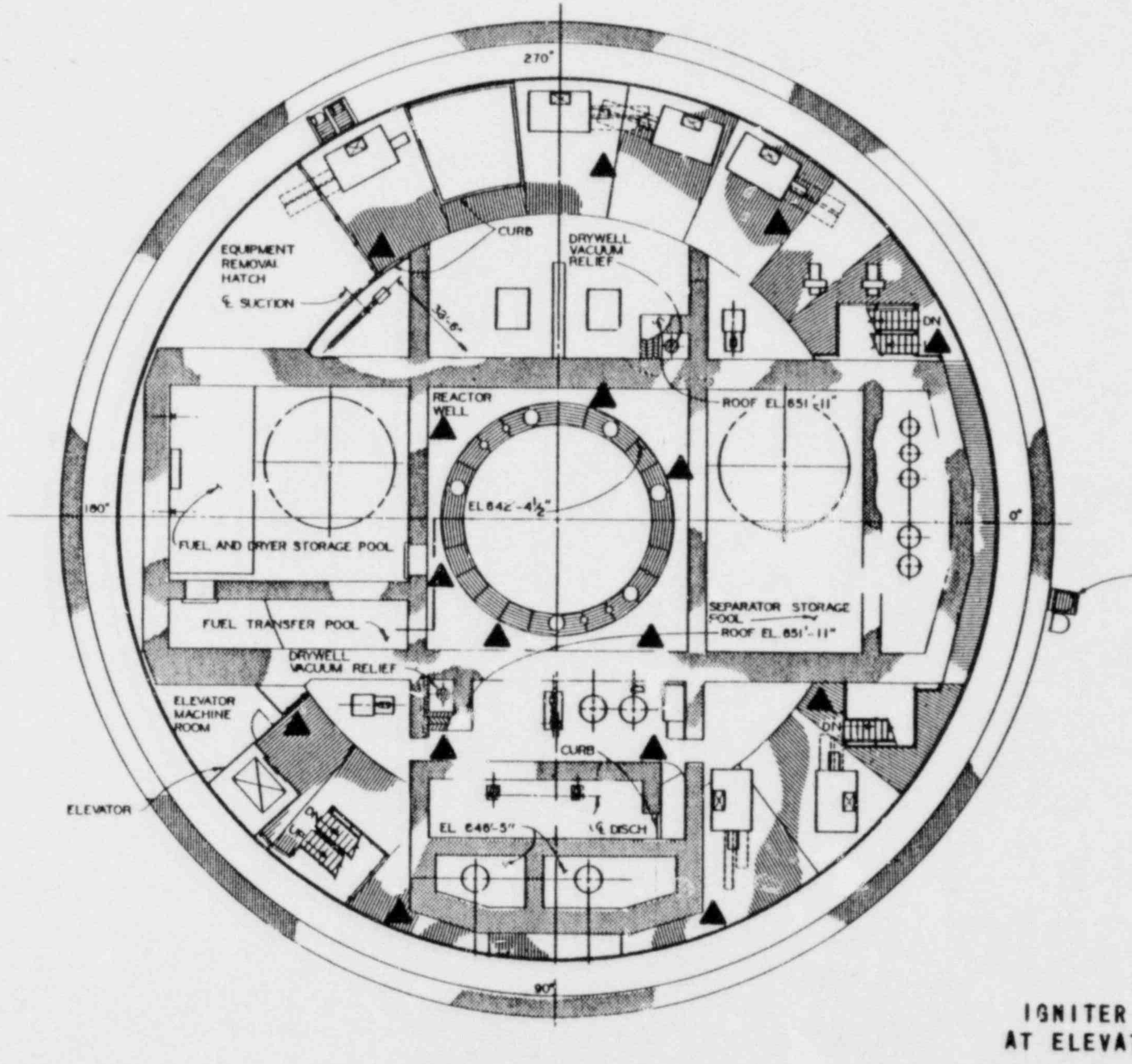
IGNITER LOCATIONS
AT ELEVATION 592'-10"

Figure (2)(1x)-4



IGNITER LOCATIONS AT ELEVATION 610'-4"

Figure (2)(1x)-5



IGNITER LOCATIONS AT ELEVATION 641'-5"

Figure (2)(1x)-6

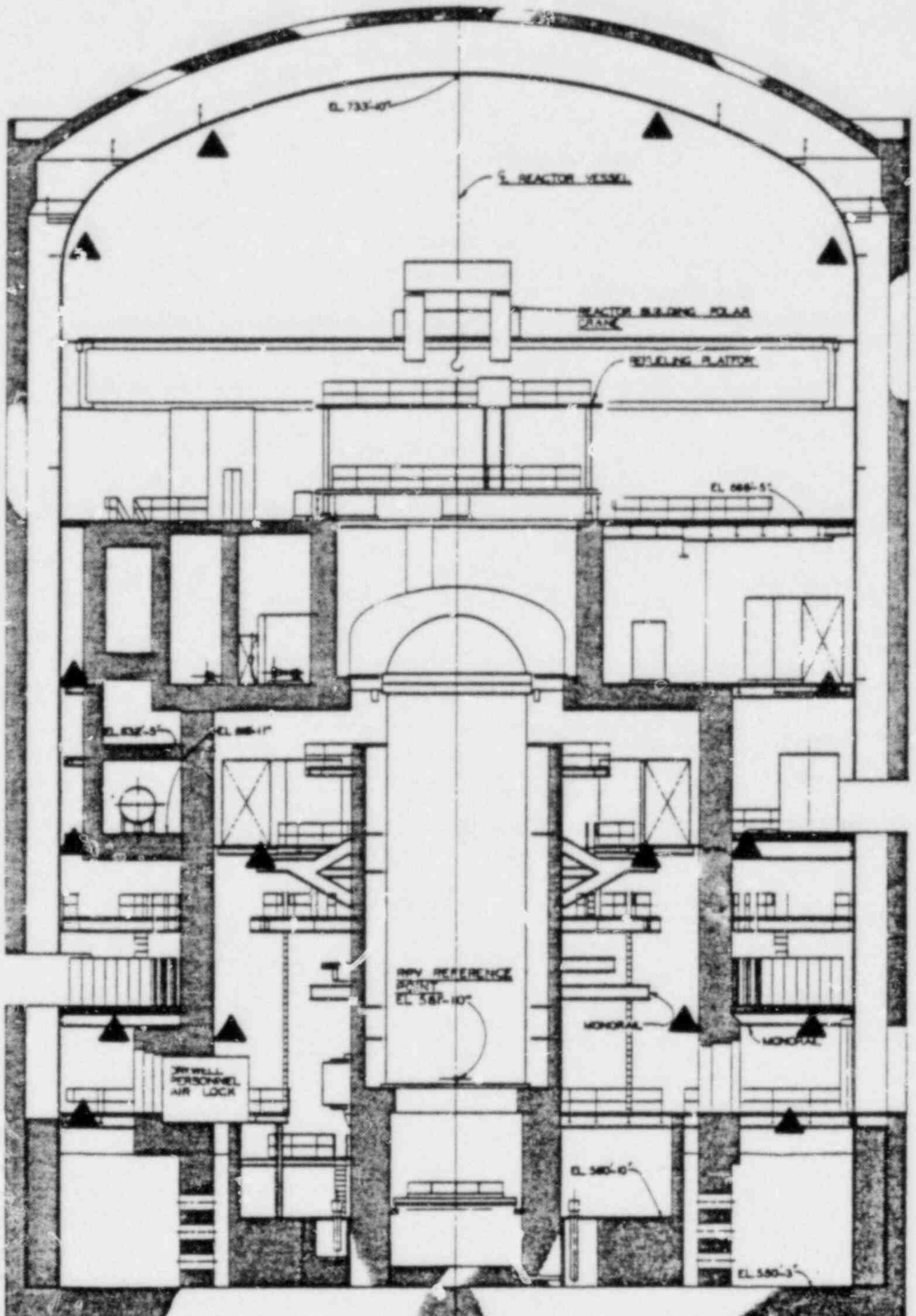


Figure (2)(ix)-7

IGNITER LOCATIONS
SECTJON C

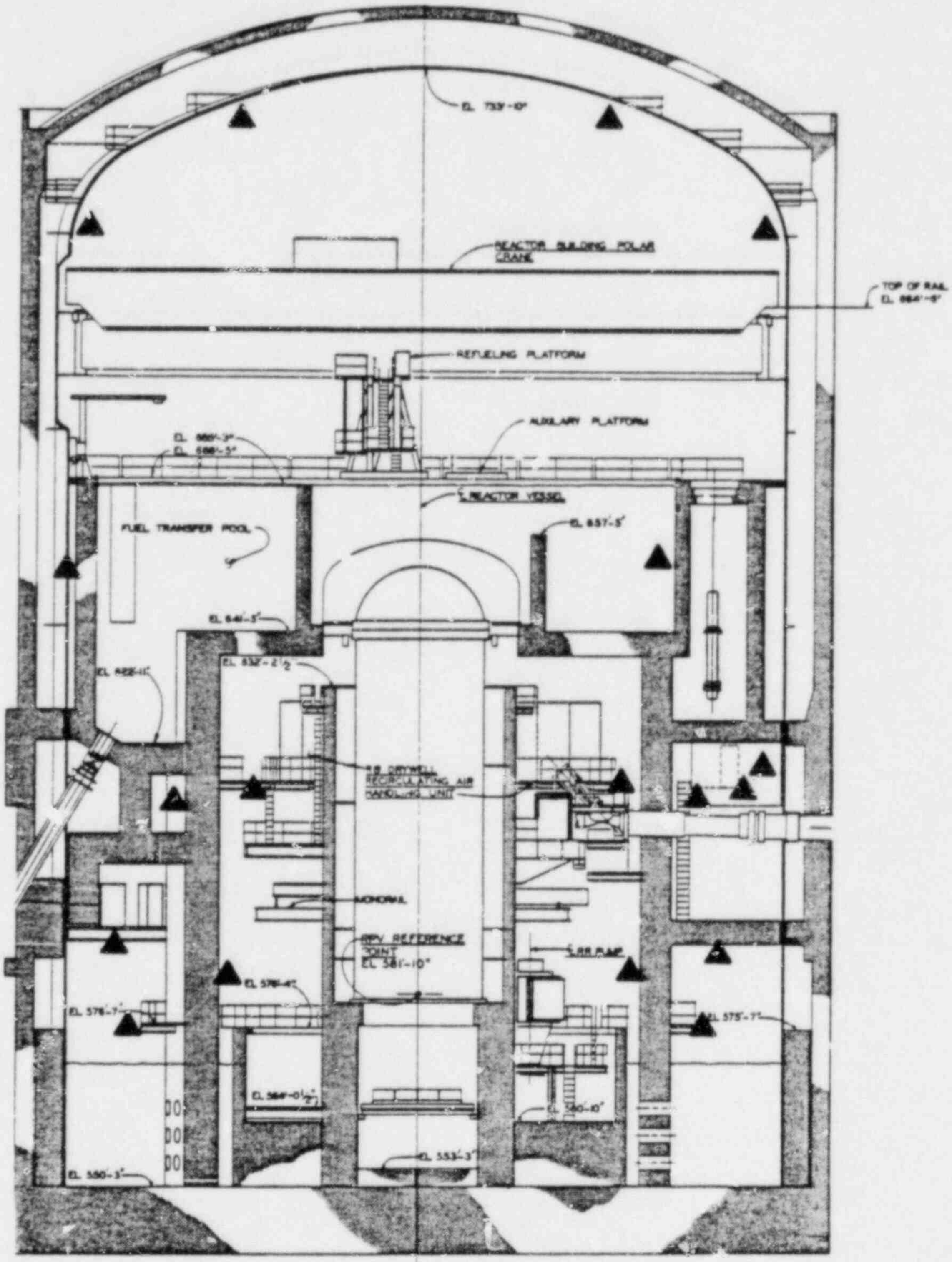
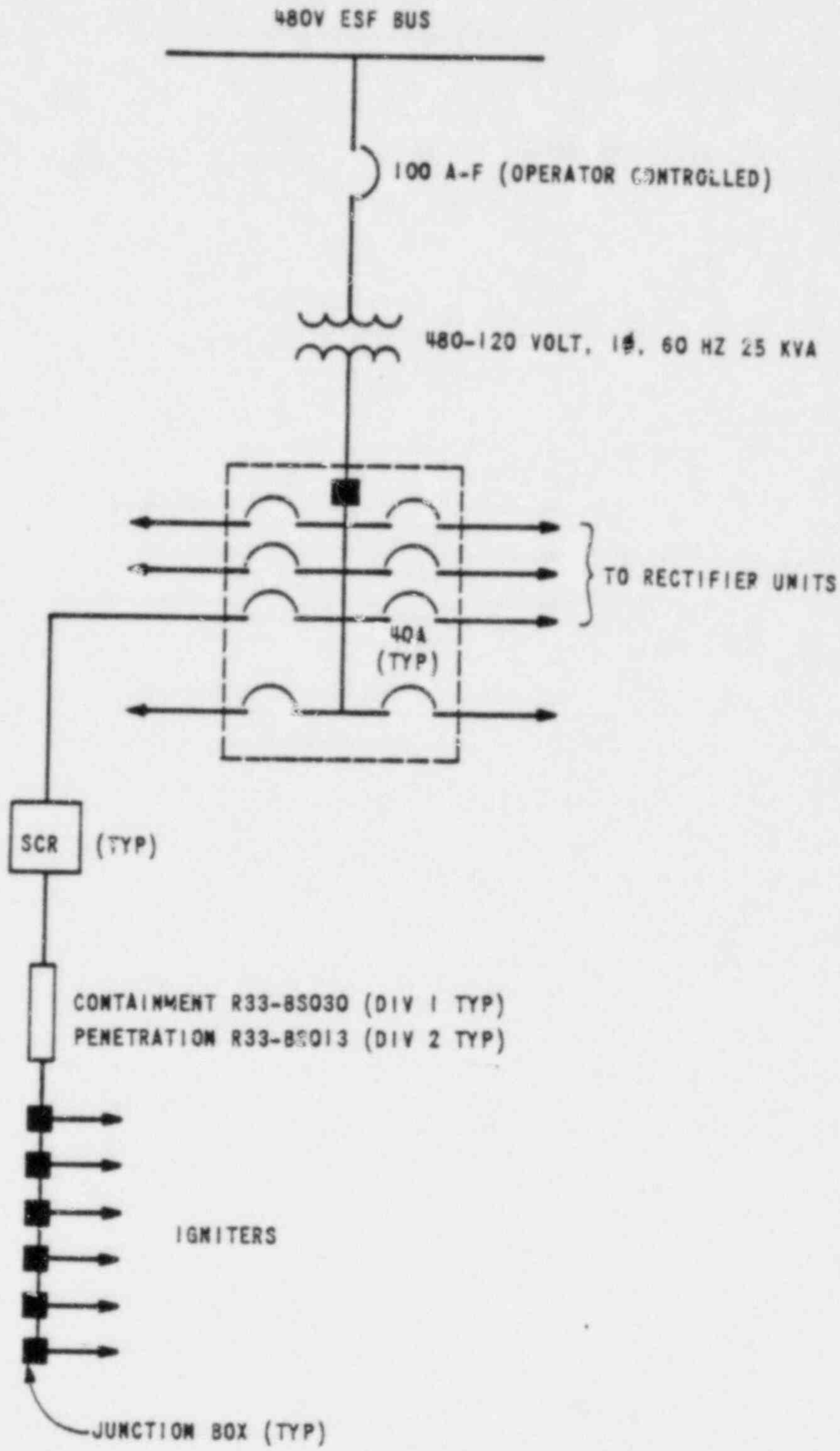


Figure (2)(ix)-8

IGNITER LOCATIONS
SECTION A



PROPOSED ONE-LINE
(SHOWING ONE DIVISION ONLY)

Figure (2)(ix)-9

FIGURE (2) (Ix)-11

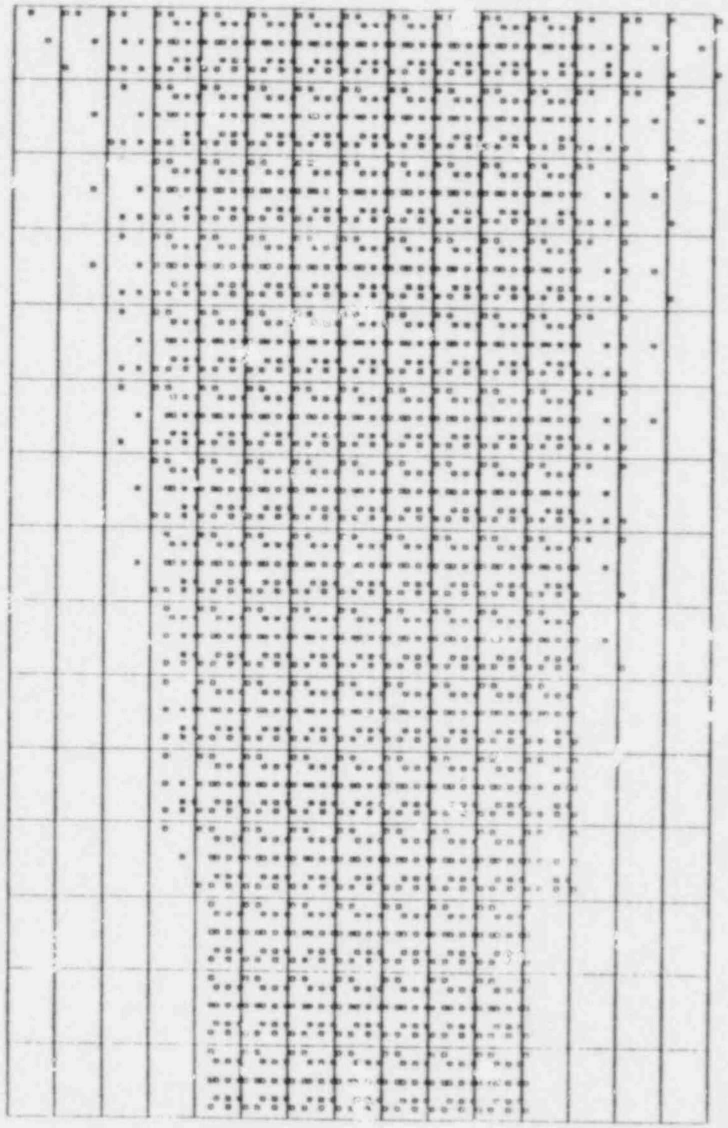


FIGURE (2) (Ix)-11

SECTION
TIME=60. SECONDS

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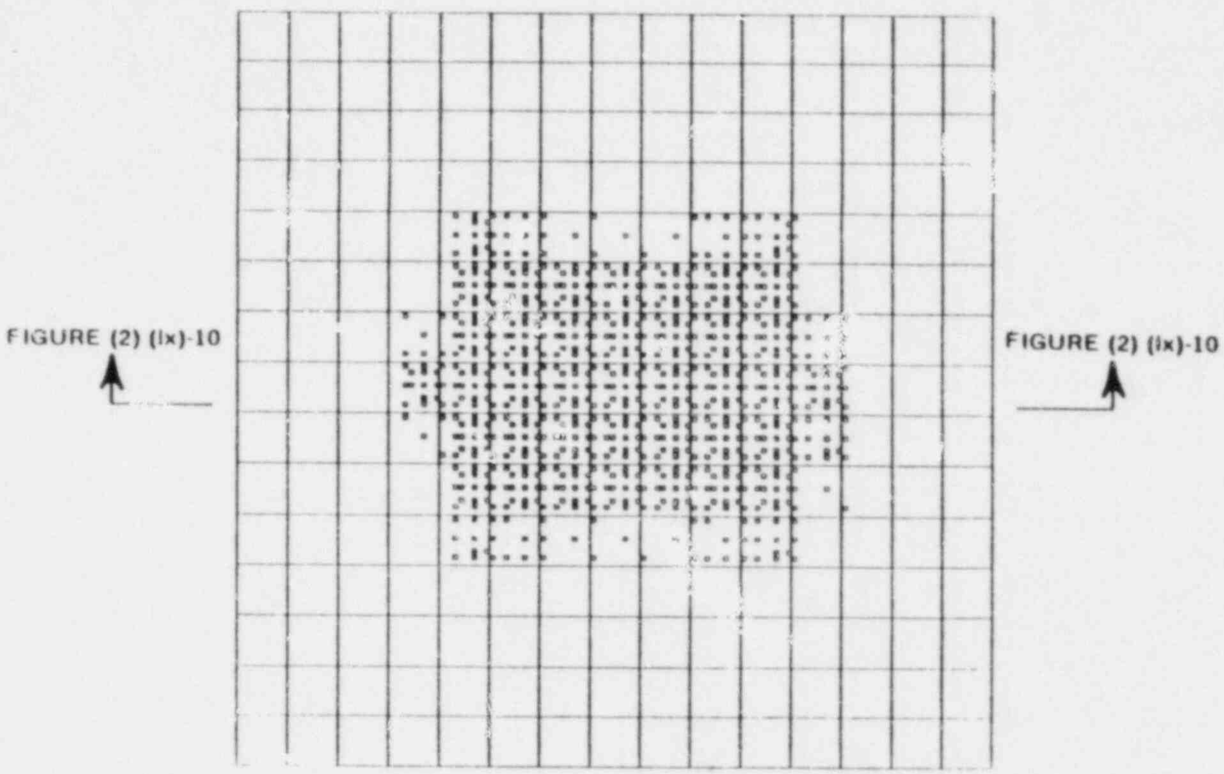
BLACK FOX STATION
HYDROGEN MIGRATION ANALYSIS
FREE RISE
FIGURE (2) (Ix)-10

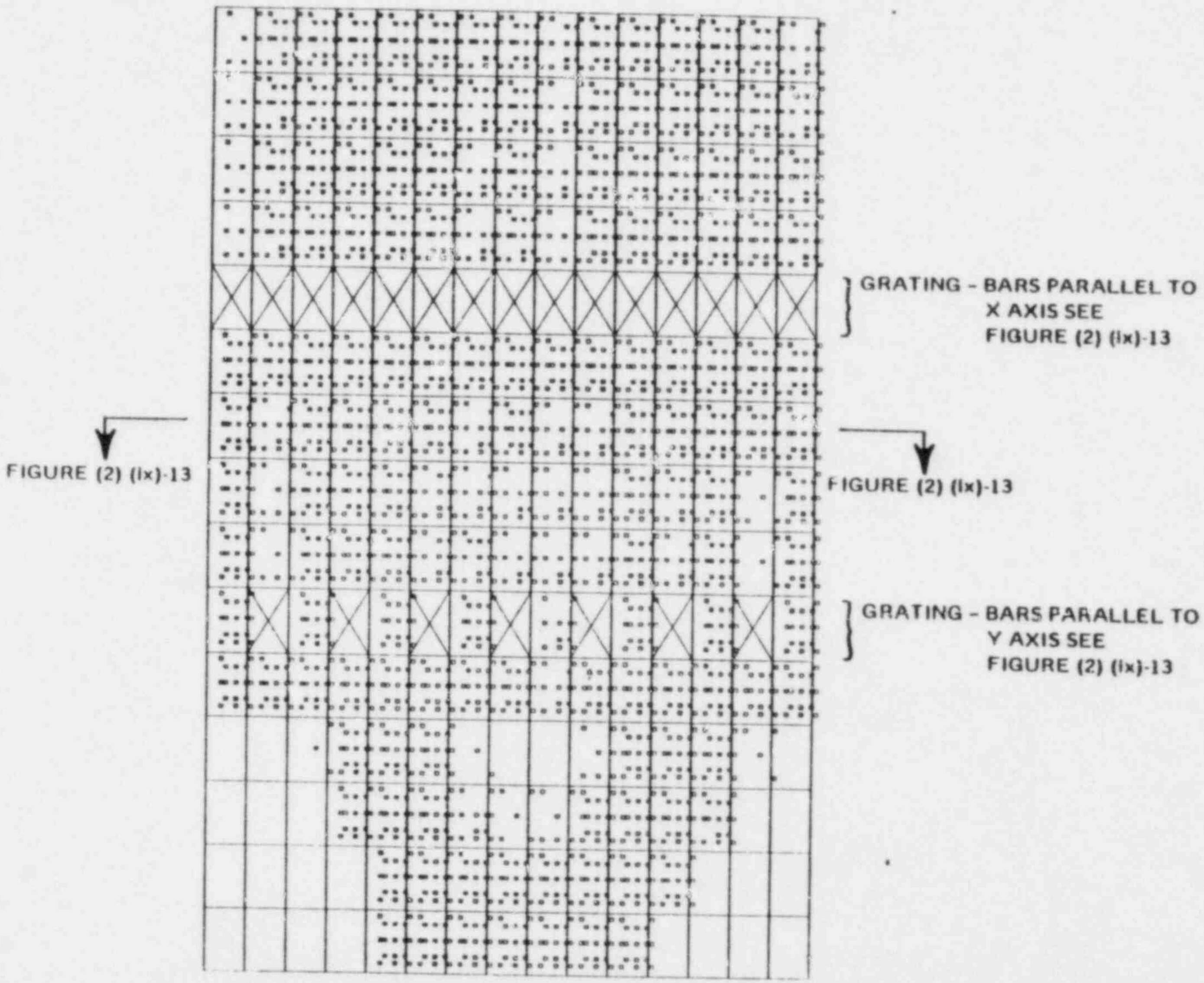


PLAN
TIME=60. SECONDS

335

19-111381

BLACK FOX STATION
HYDROGEN MIGRATION ANALYSIS
FREE RISE
FIGURE (2)(Ix)-11

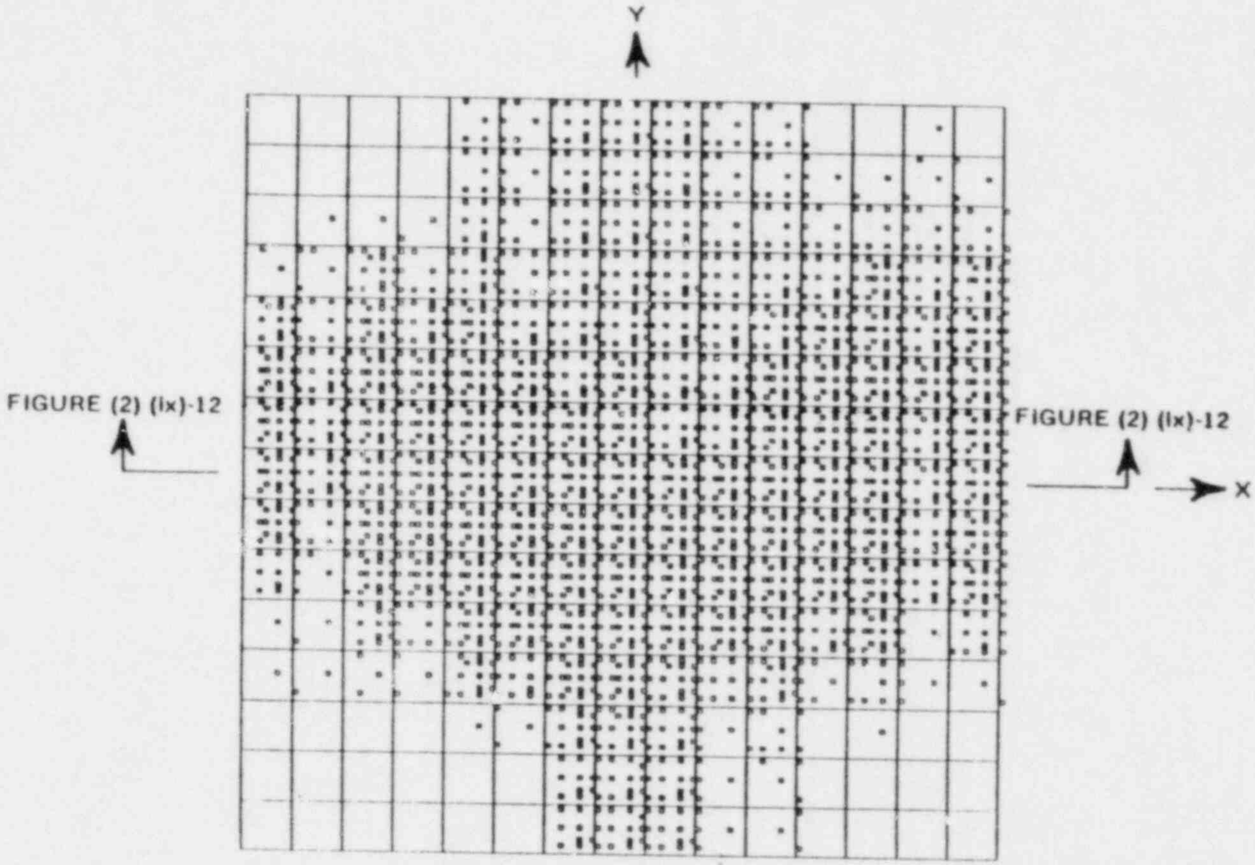


SECTION .
TIME=60. SECONDS

BLACK FOX STATION
HYDROGEN MIGRATION ANALYSIS
GRATING EFFECTS
FIGURE (2)(ix)-12

336

19-111381



PLAN
TIME=60. SECONDS

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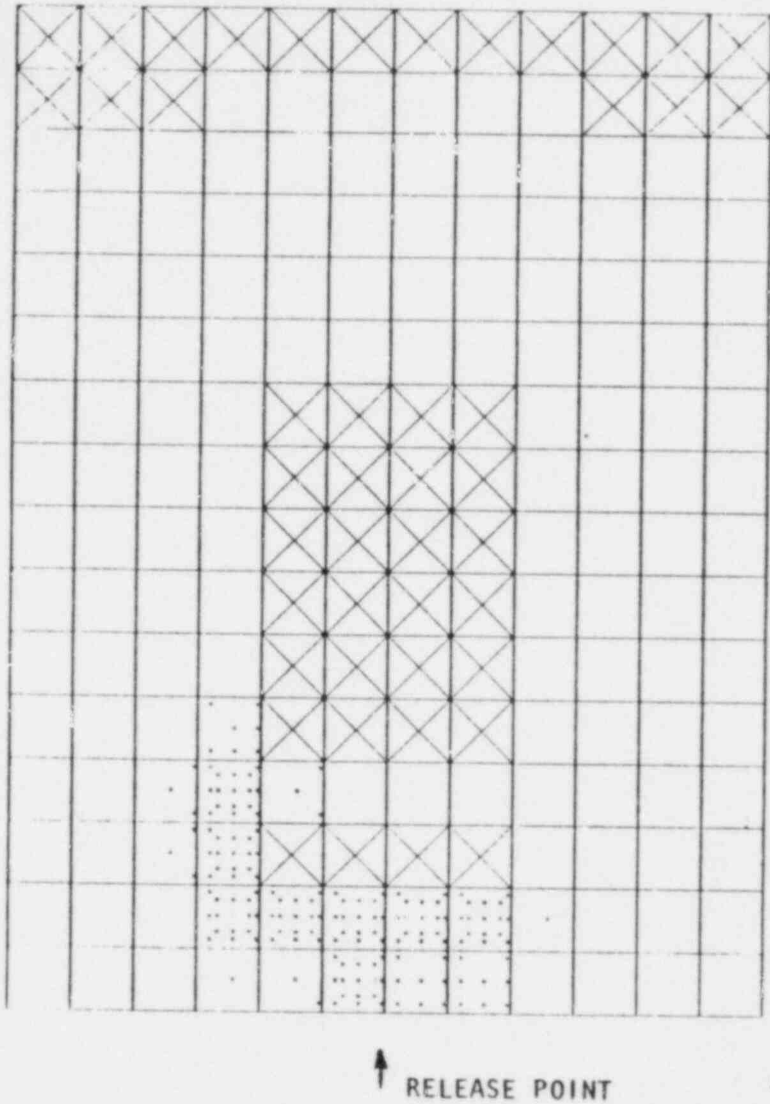
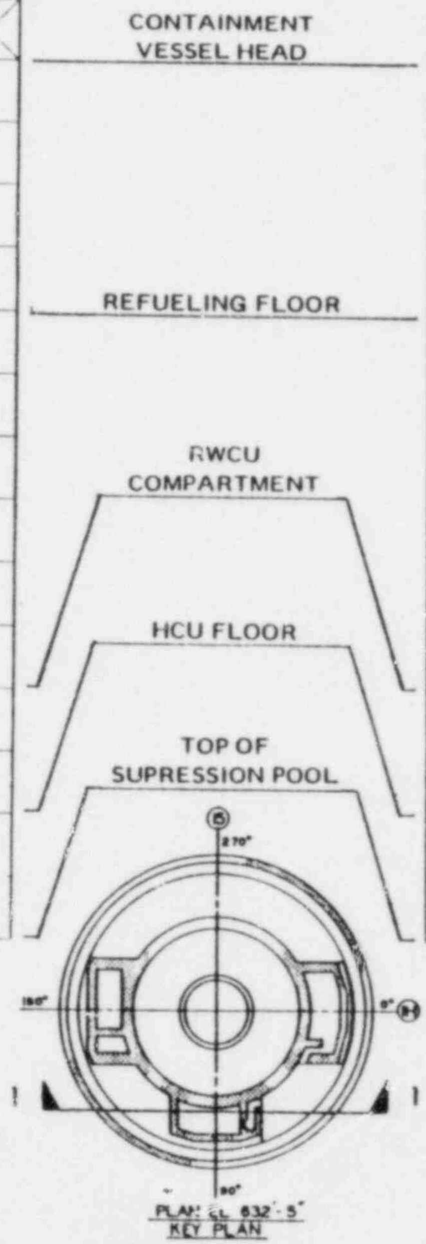
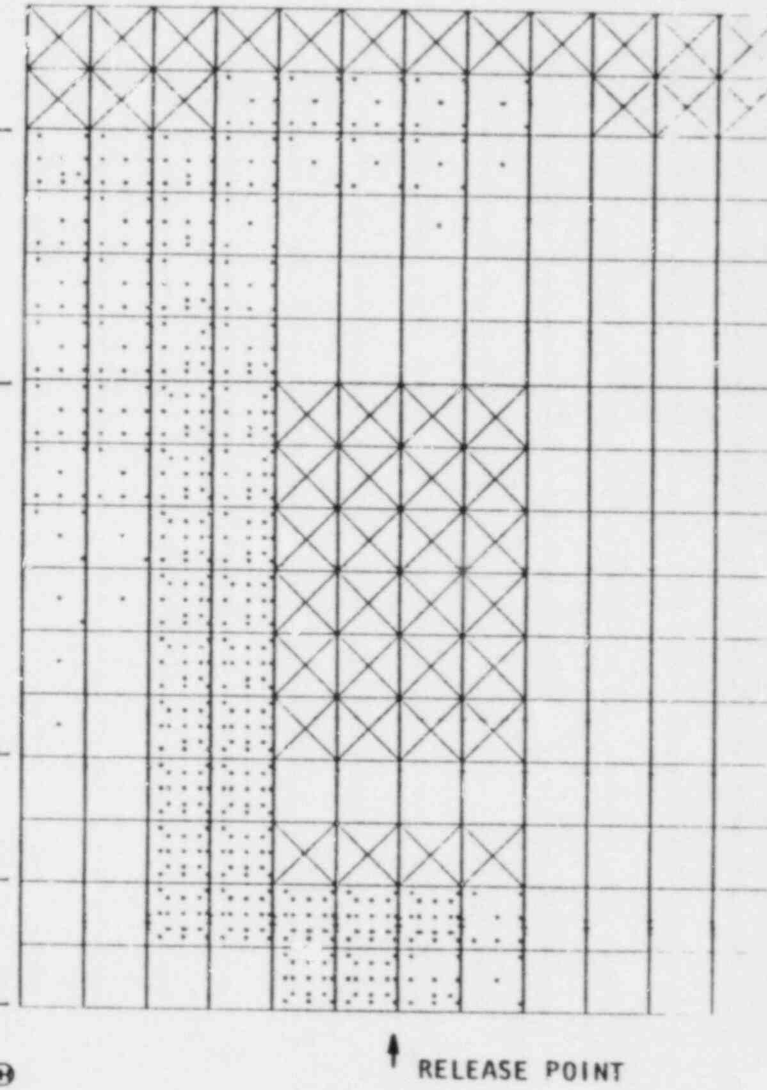
19-111381

BLACK FOX STATION
HYDROGEN MIGRATION ANALYSIS
GRATING EFFECTS
FIGURE (2)(ix)-13



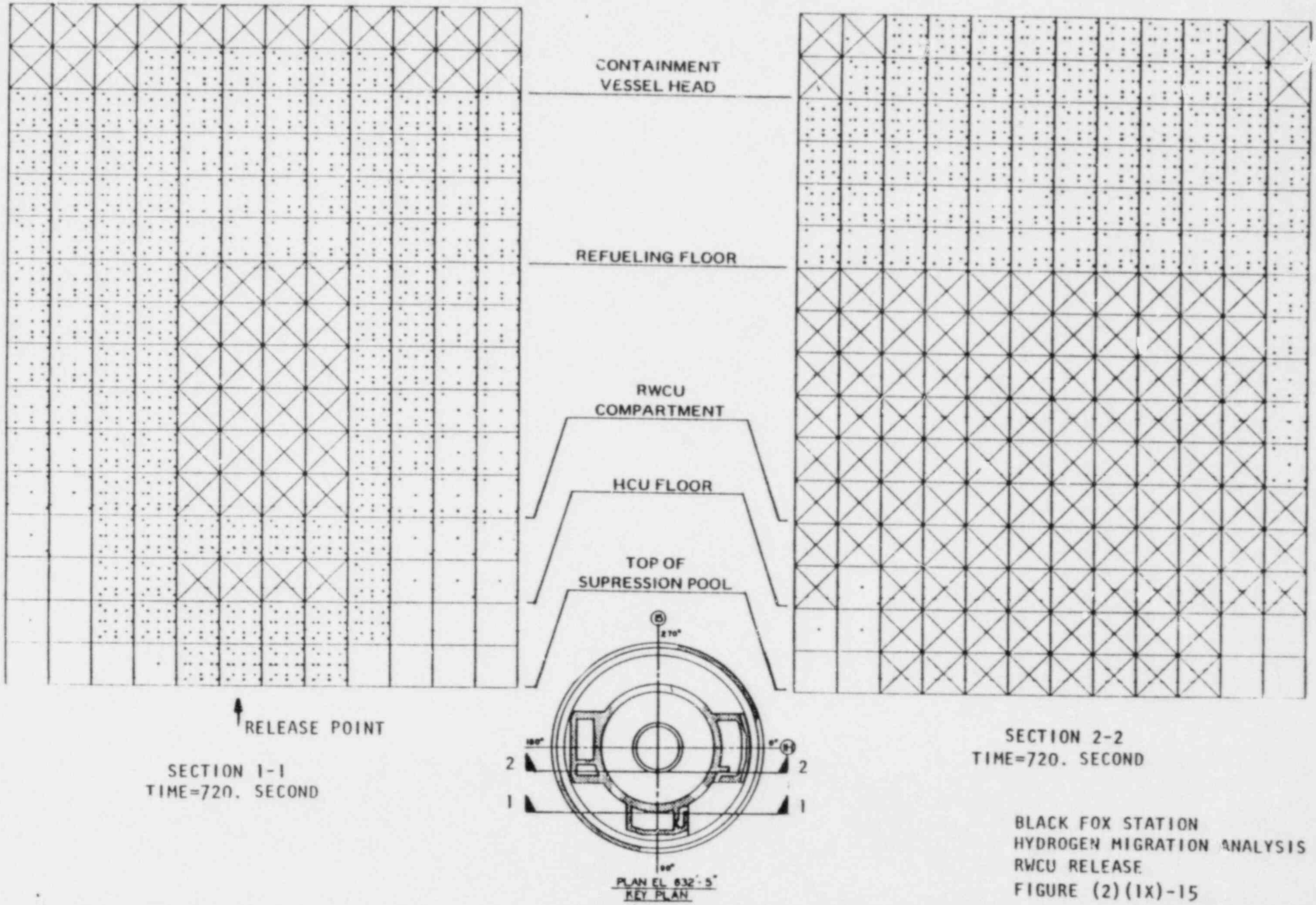
SECTION 1-1
TIME=121.SECONDS
FIGURE (2)(ix)-14A



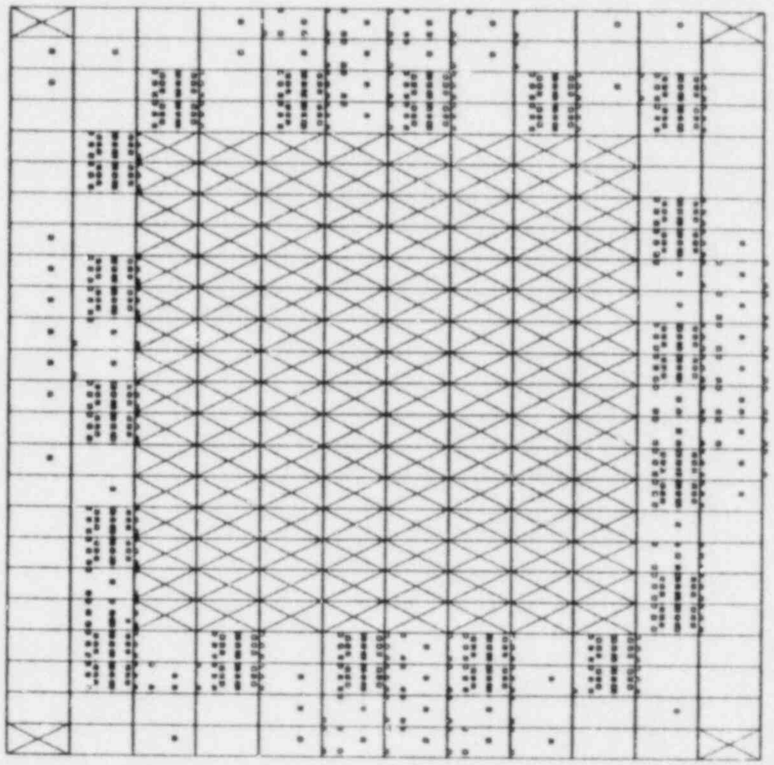


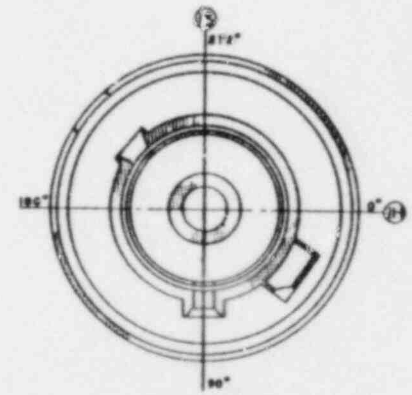
SECTION 1-1
TIME=240. SECONDS
FIGURE (2)(ix)-14B

BLACK FOX STATION
HYDROGEN MIGRATION ANALYSIS
RWCU RELEASE
FIGURE (2)(ix)-14



BLACK FOX STATION
HYDROGEN MIGRATION ANALYSIS
RWCU RELEASE
FIGURE (2)(1X)-15





PLAN EL. 570'-7"
KEY PLAN

PLAN
TIME=240. SECONDS

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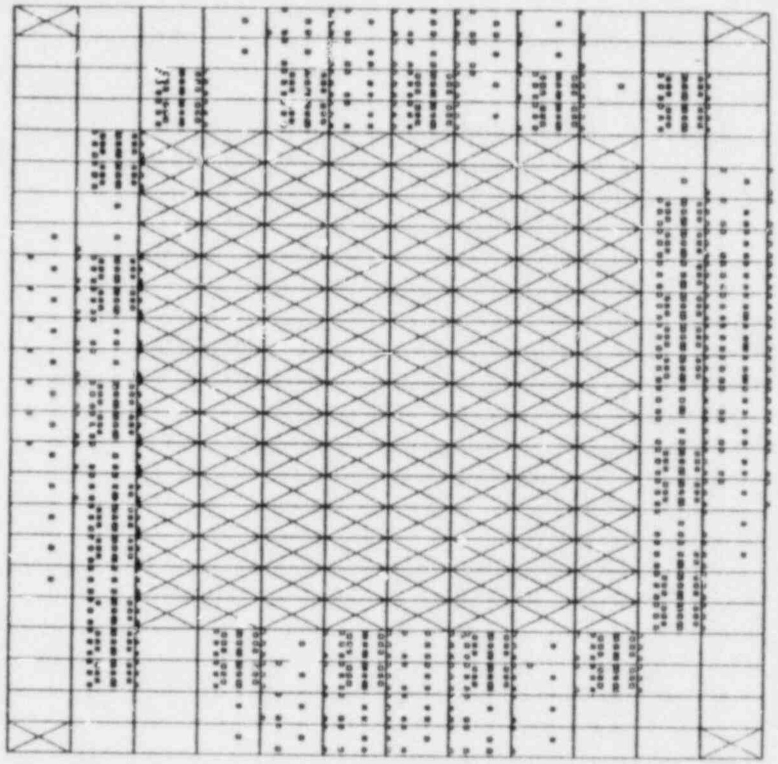
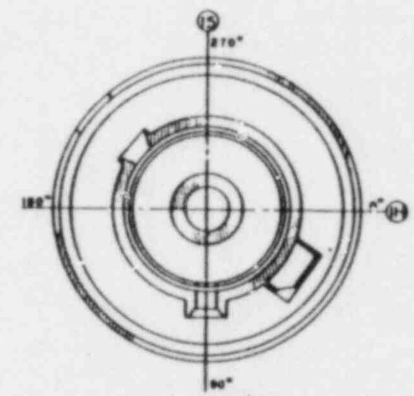
19-111381

BLACK FOX STATION
HYDROGEN MIGRATION ANALYSIS
DISTRIBUTED DISCHARGE
FIGURE (2)(1x)-16



PLAN
TIME=480. SECONDS



PLAN EL 570'-7"
KEY PLAN

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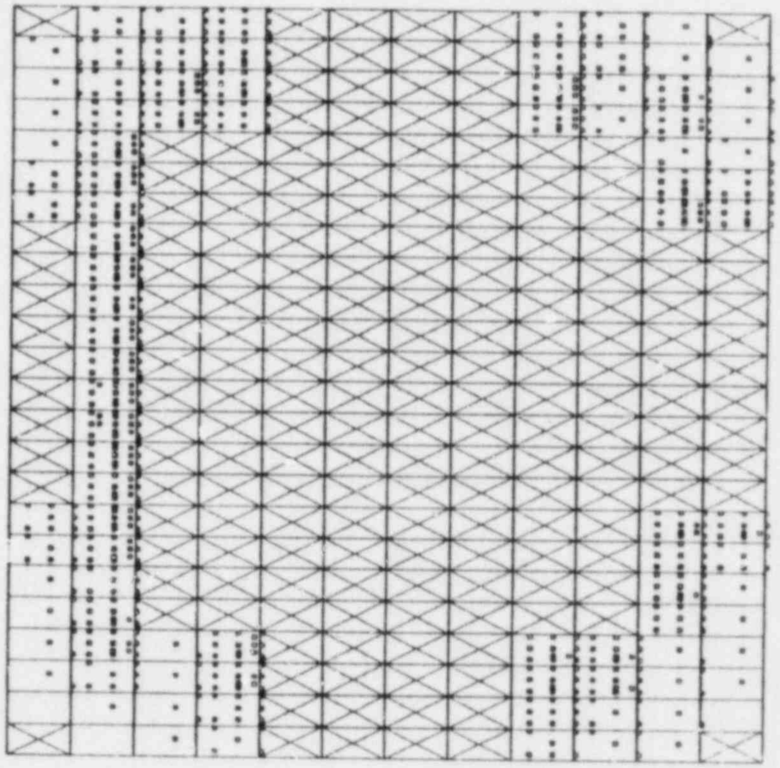
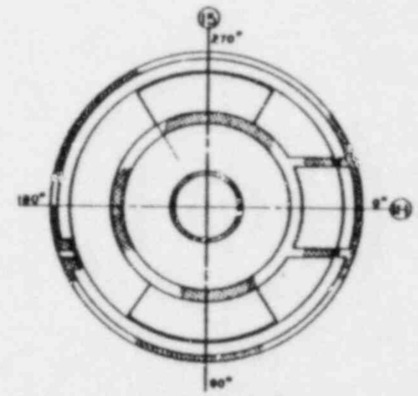
19-111381

BLACK FOX STATION
HYDROGEN MIGRATION ANALYSIS
DISTRIBUTED DISCHARGE
FIGURE (2)(ix)-17



PLAN
TIME=240. SECONDS



PLAN EL 592'-10"
KEY PLAN

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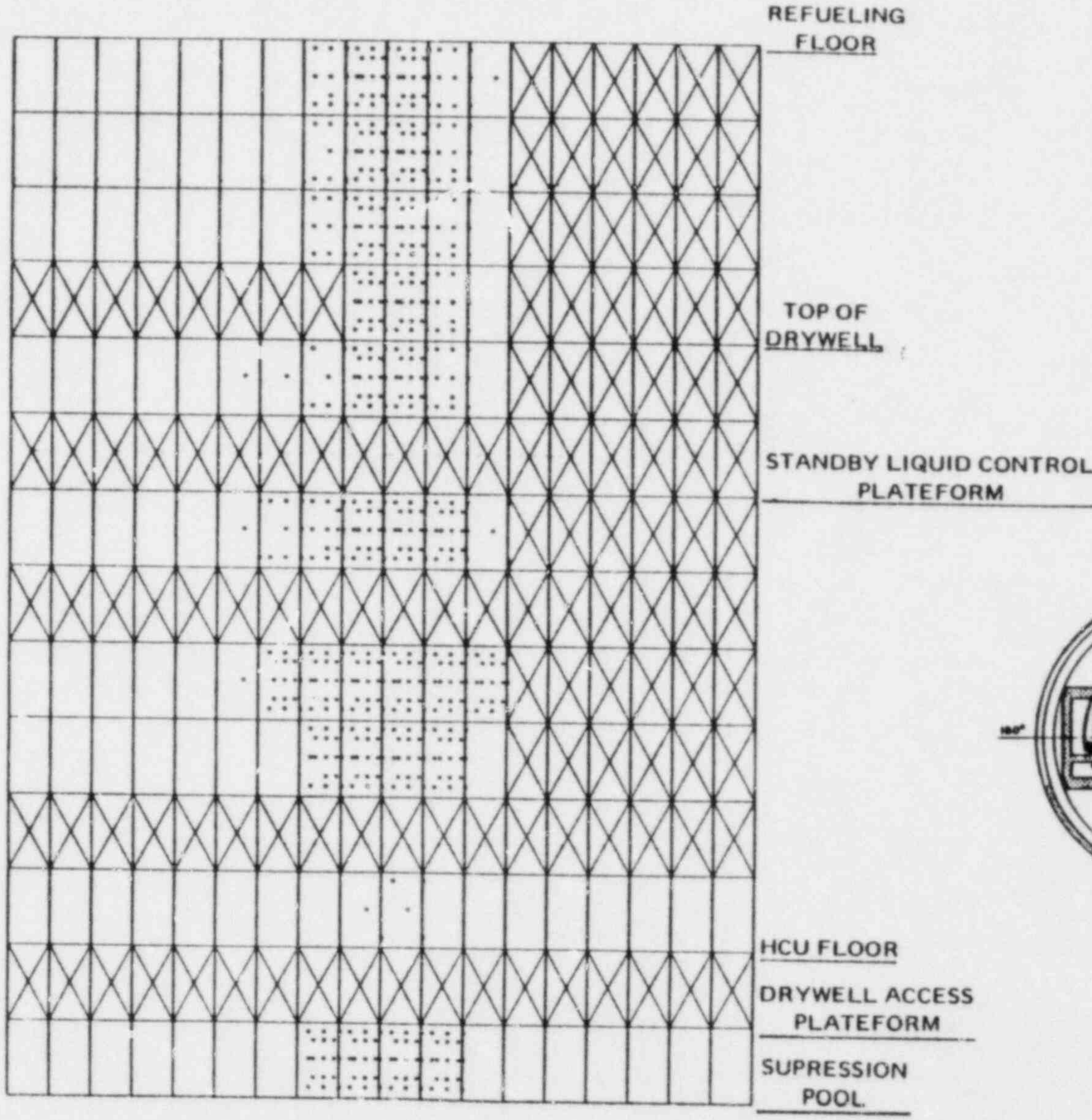
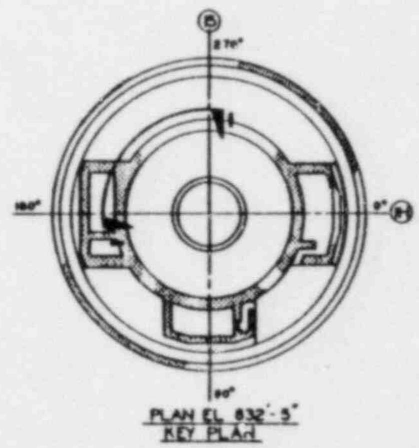
19-111381

BLACK FOX STATION
HYDROGEN MIGRATION ANALYSIS
DISTRIBUTED DISCHARGE
FIGURE (2)(ix)-18



SECTION I-I
TIME=120. SECONDS



BLACK FOX STATION
HYDROGEN MIGRATION ANALYSIS
EQUIPMENT HATCH
FIGURE (2)(ix)-19

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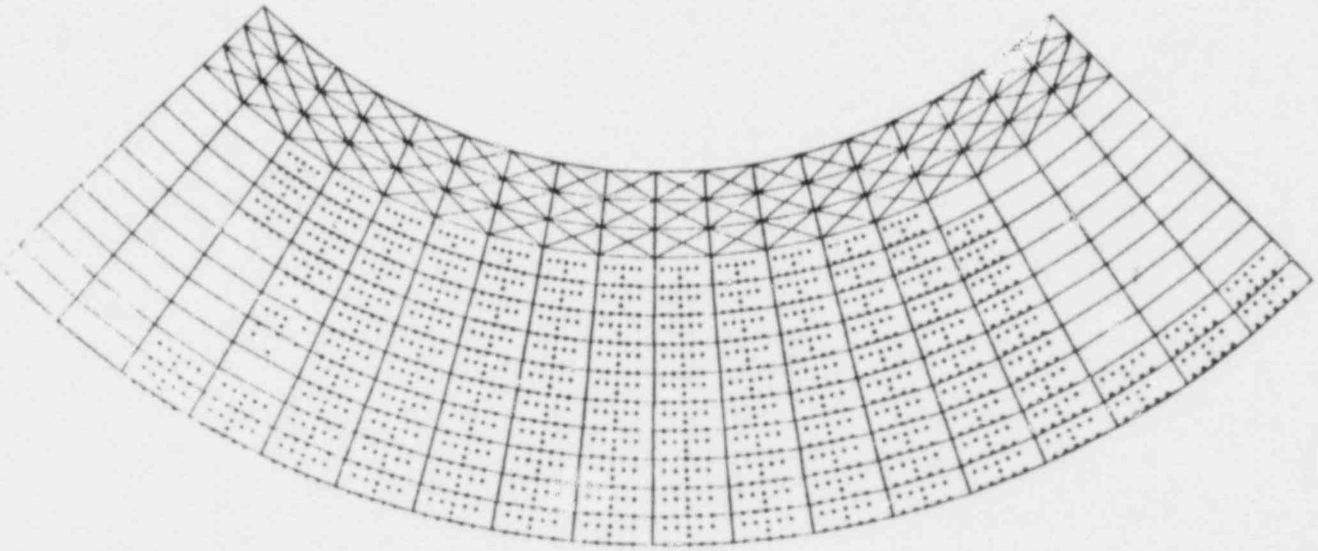
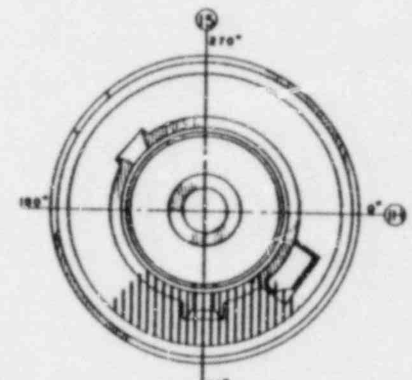
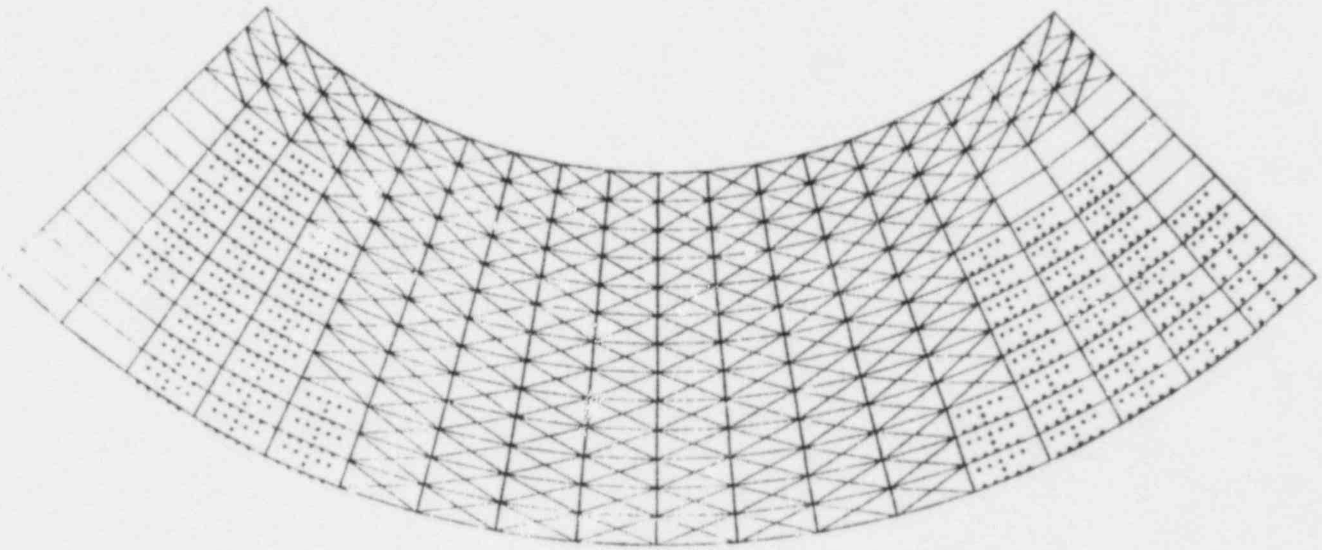


PLAN
TIME=360. SECONDS



BLACK FOX STATION
HYDROGEN MIGRATION ANALYSIS
RWCU SECTION
FIGURE (2)(Ix)-20



PLAN
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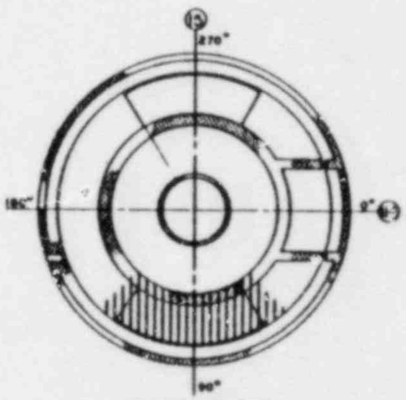
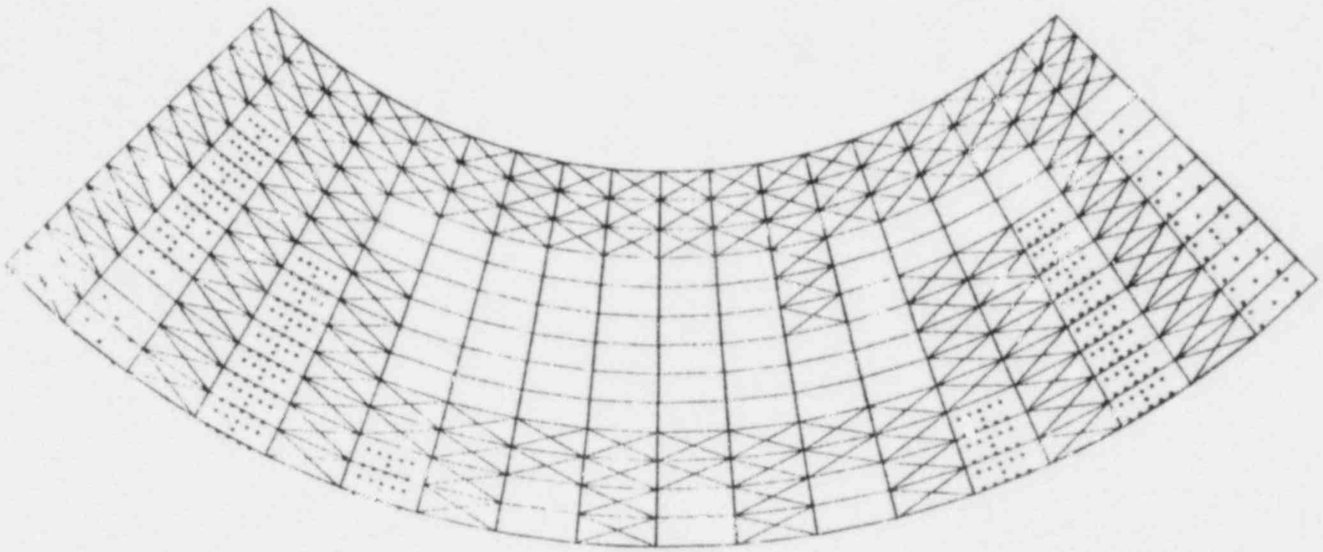
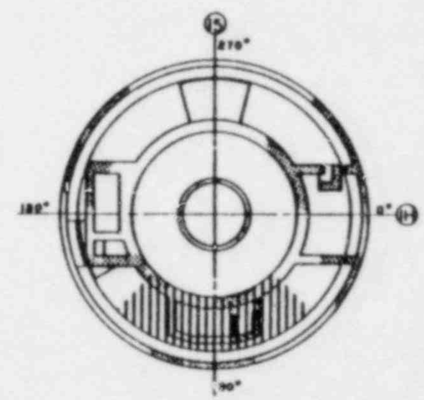


PLAN EL 59'-10"
KEY PLAN

BLACK FOX STATION
HYDROGEN MIGRATION ANALYSIS
RWCU SECTION
FIGURE (2)(1x)-21

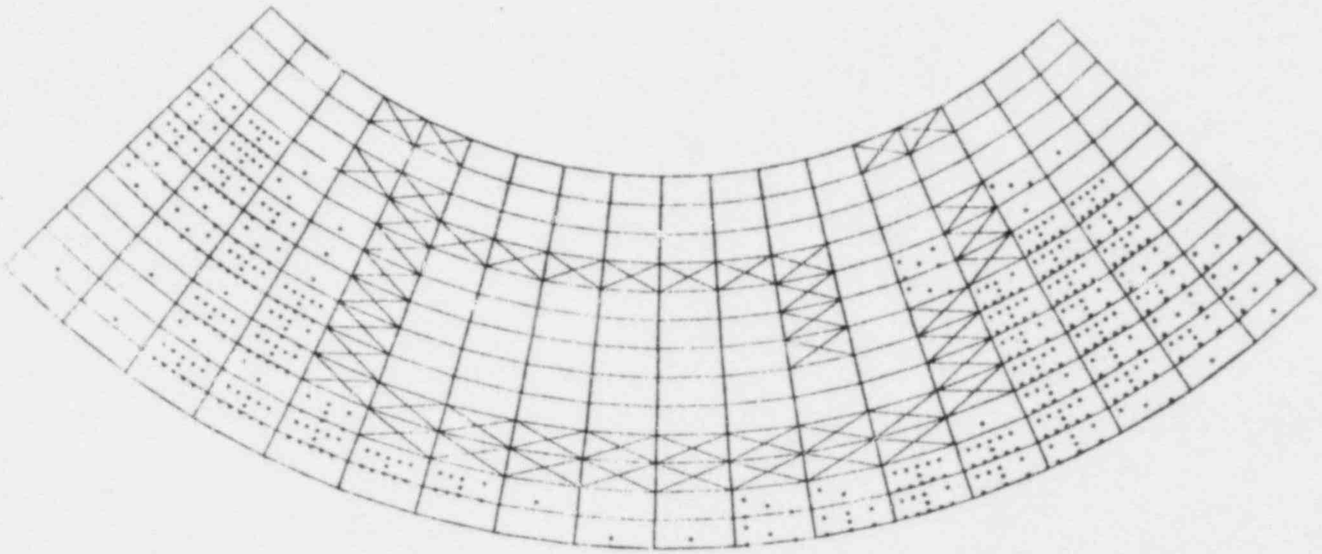


PLAN
TIME=360. SECONDS



PLAN EL 816'-11"
KEY PLAN

BLACK FOX STATION
HYDROGEN MIGRATION ANAYLSIS
RWCU SECTION
FIGURE (2)(ix)-22



PLAN
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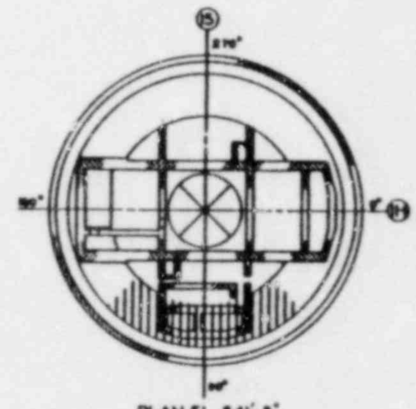
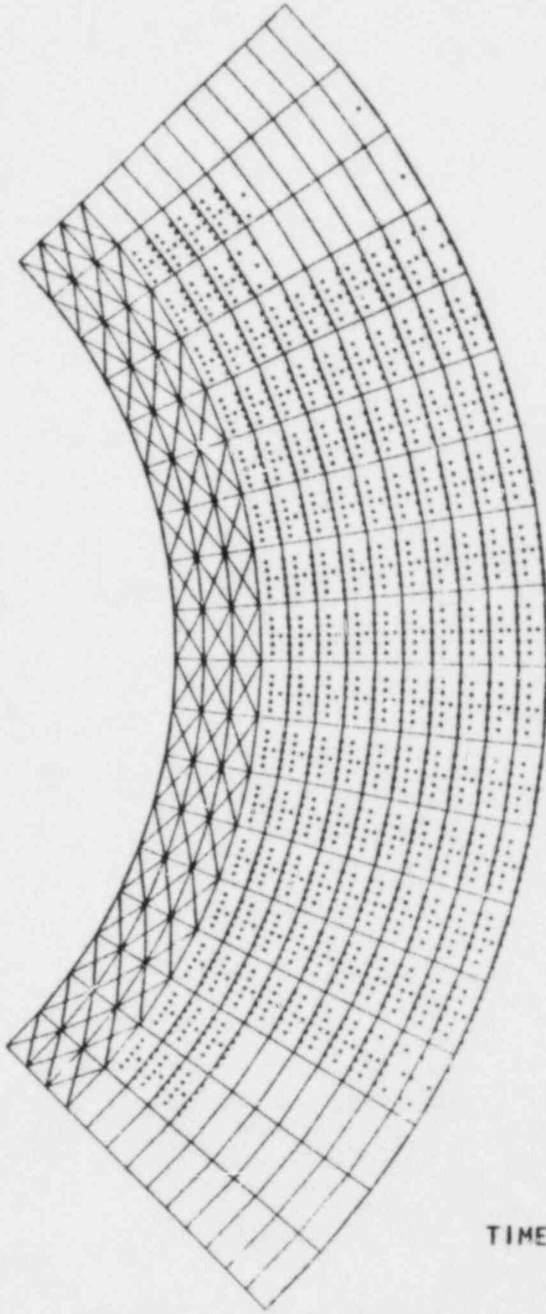
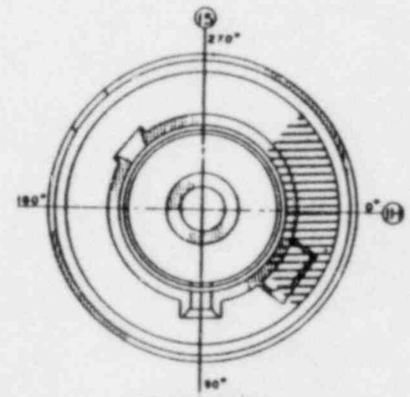


BLACK FOX STATION
HYDROGEN MIGRATION ANALYSIS
RWCU SECTION
FIGURE (2)(1x)-27





PLAN EL 576'-7"
KEY PLAN

PLAN
TIME=361. SECONDS

BLACK FOX STATION
HYDROGEN MIGRATION ANALYSIS
STEAM TUNNEL
FIGURE (2)(ix)-24

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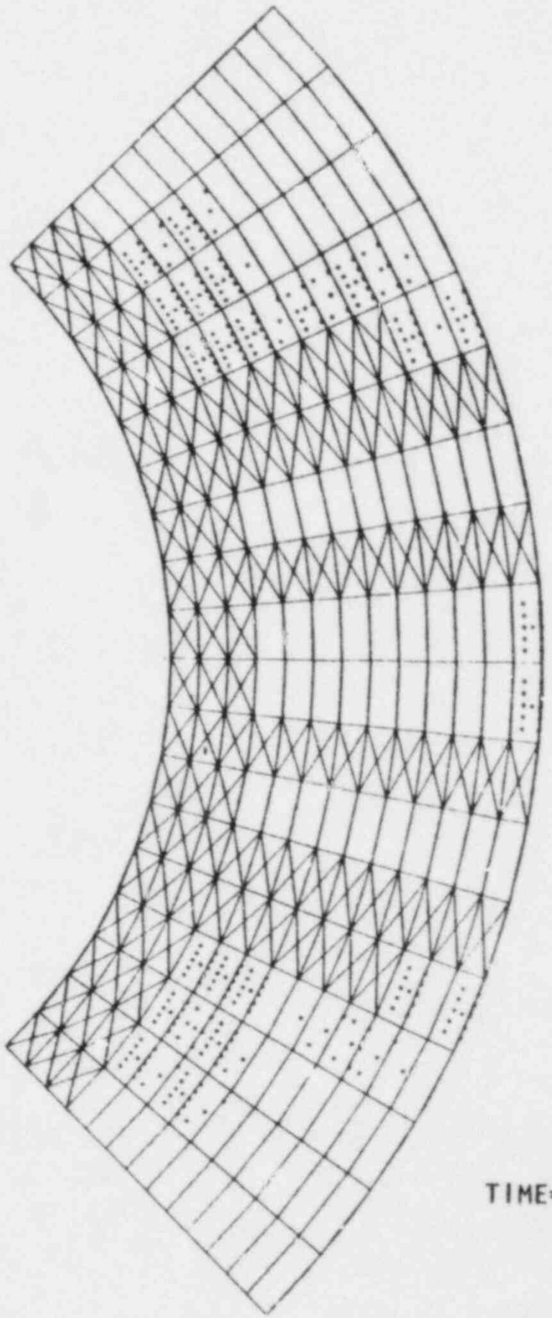
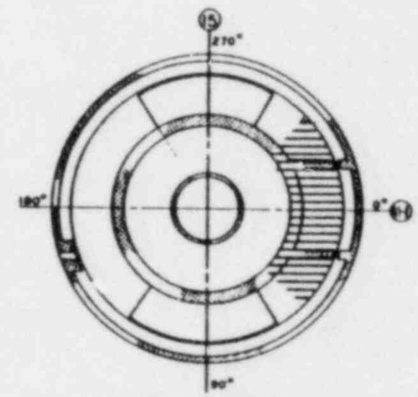


PLAN
TIME=361. SECONDS



PLAN EL 592'-10"
KEY PLAN

BLACK FOX STATION
HYDROGEN MIGRATION ANALYSIS
STEAM TUNNEL
FIGURE (2) (ix)-25

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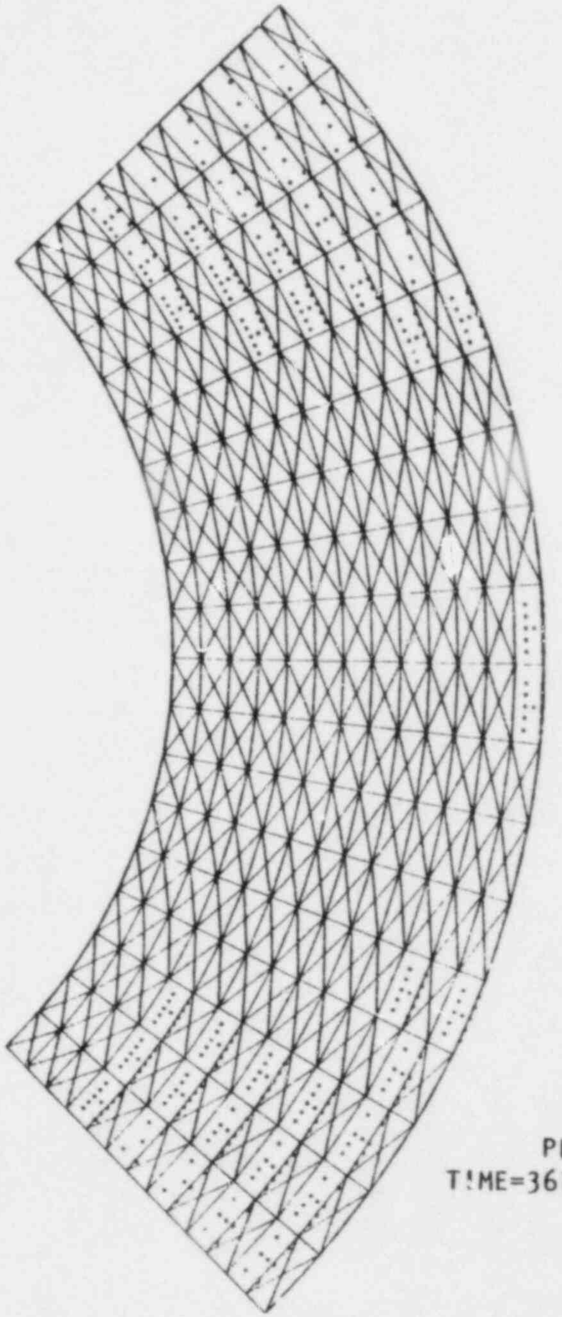
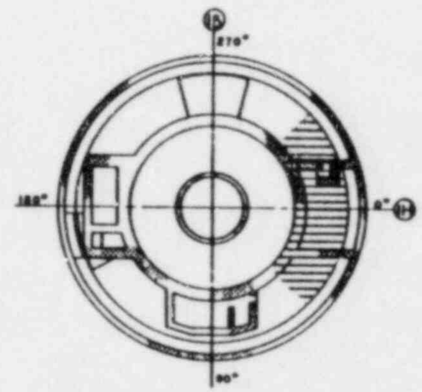


PLAN
TIME=361.0 SECONDS



PLAN EL 610'-11"
KEY PLAN

BLACK FOX STATION
HYDROGEN MIGRATION ANALYSIS
STEAM TUNNEL
FIGURE (2)(ix)-26

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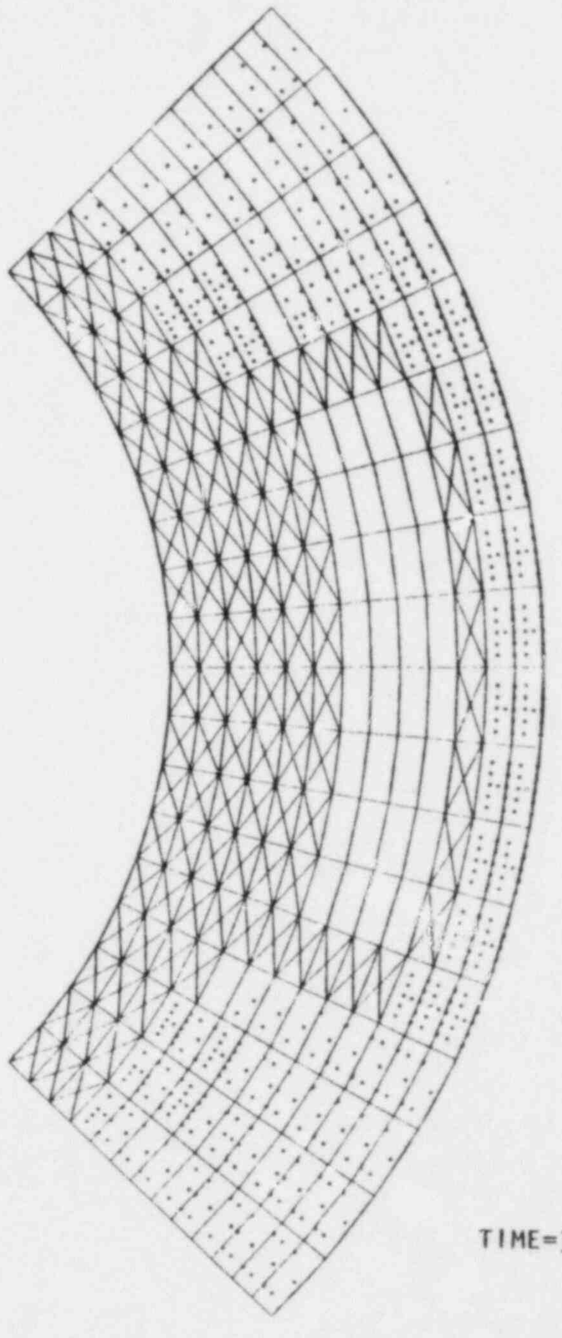
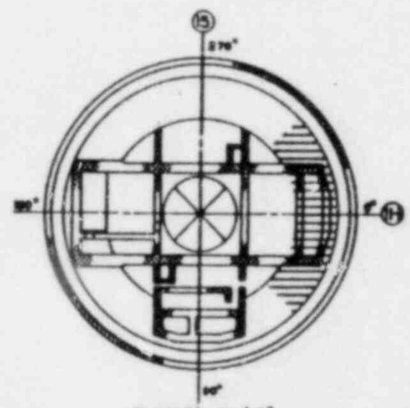
19-111381

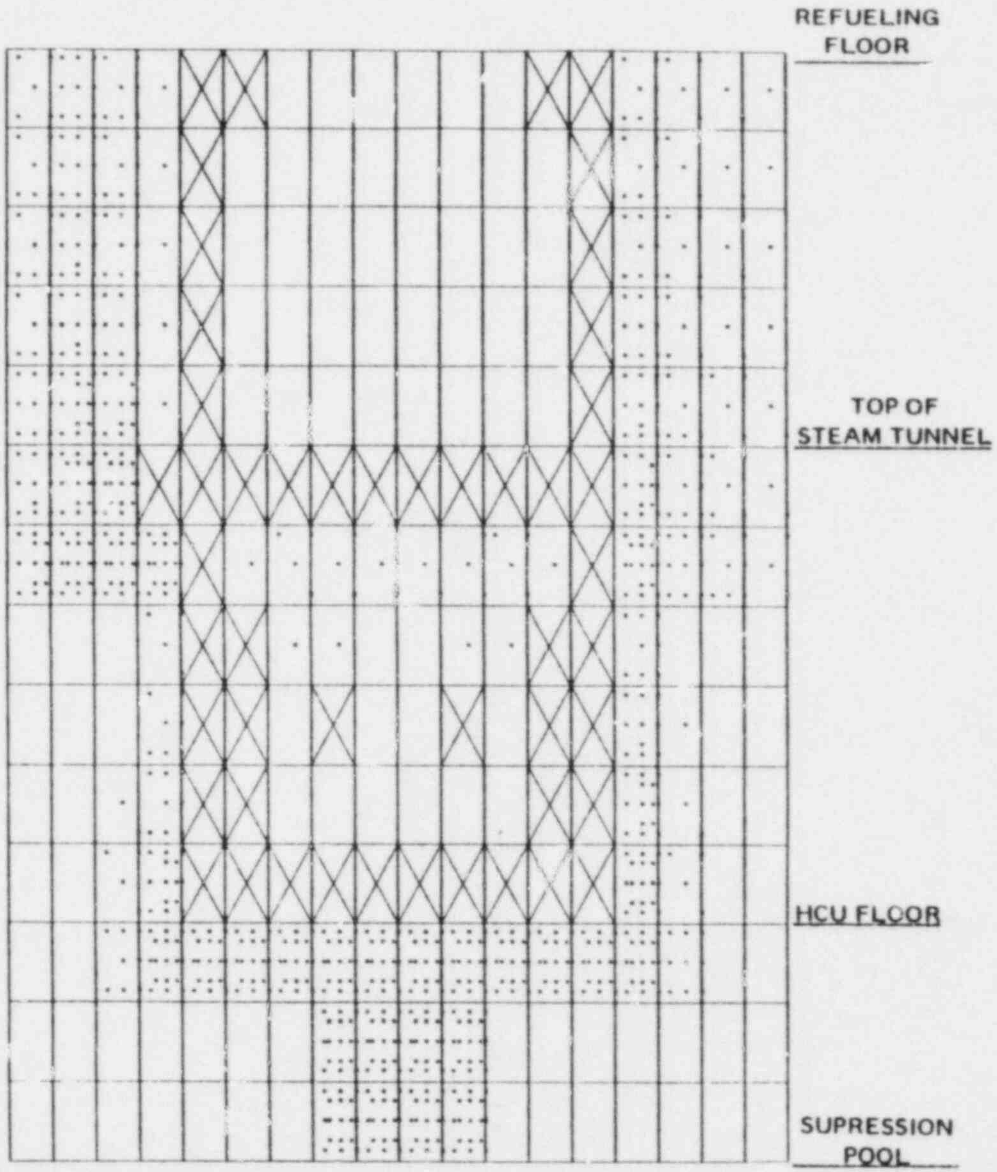


PLAN
TIME=361. SECONDS

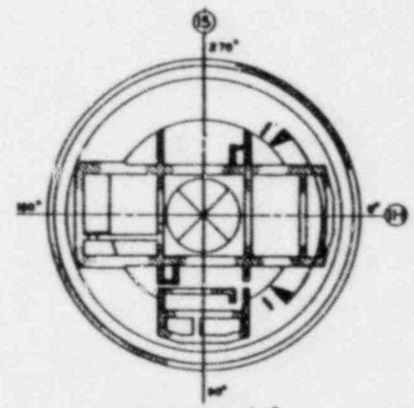


PLAN EL. 641'-5"
KEY PLAN

BLACK FOX STATION
HYDROGEN MIGRATION ANALYSIS
STEAM TUNNEL
FIGURE (2)(1x)-27



SECTION 1-1
TIME=120. SECONDS



PLAN EL 641'-5"
KEY PLAN

BLACK FOX STATION
HYDROGEN MIGRATION ANALYSIS
STEAM TUNNEL
FIGURE (2)(ix)-28

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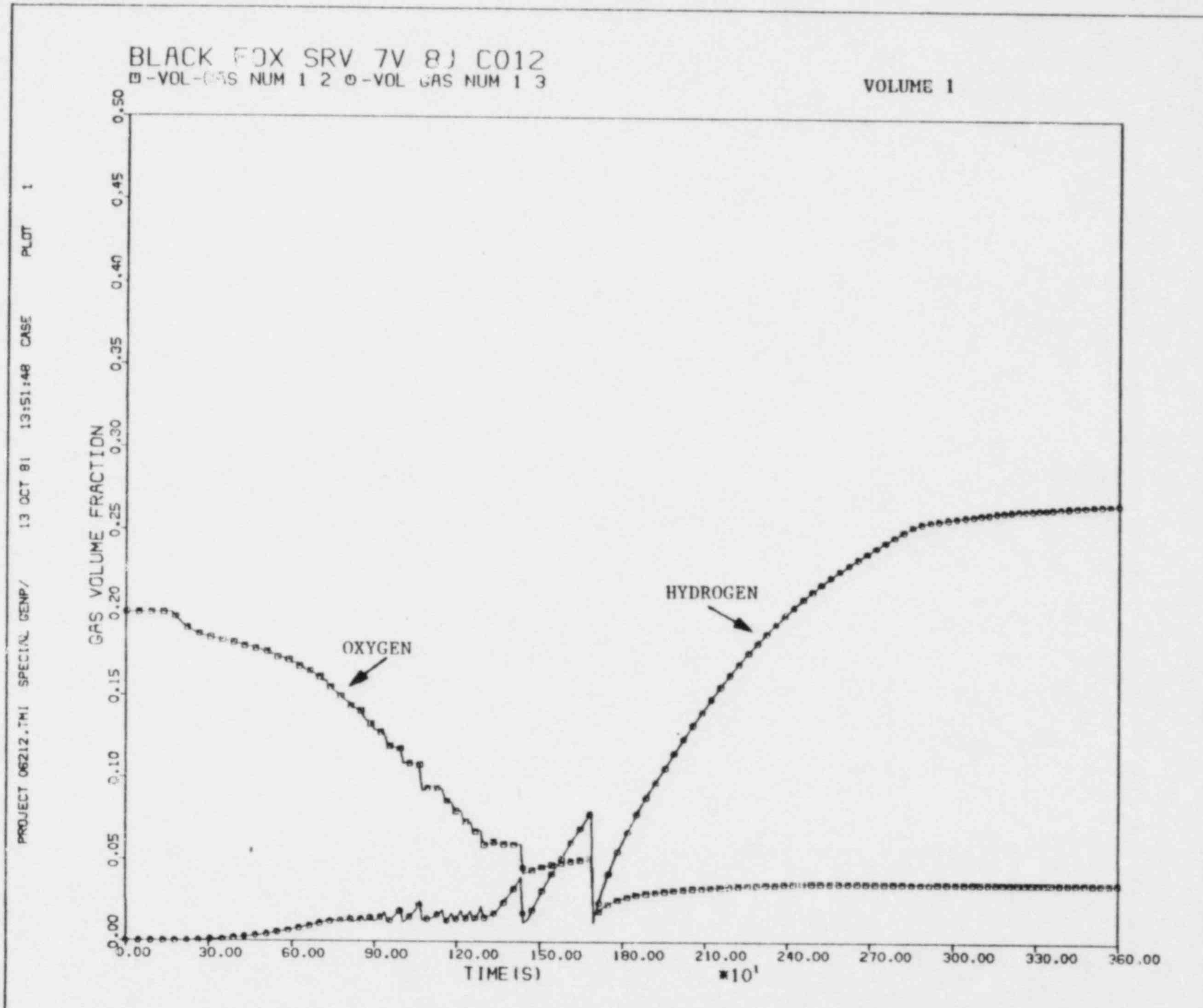
19-111381



USER: 03636SCHADE PLOT FILE: 03636SCHADE*PL*135148A.

FIGURE (2)(1x)-29

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19-111381

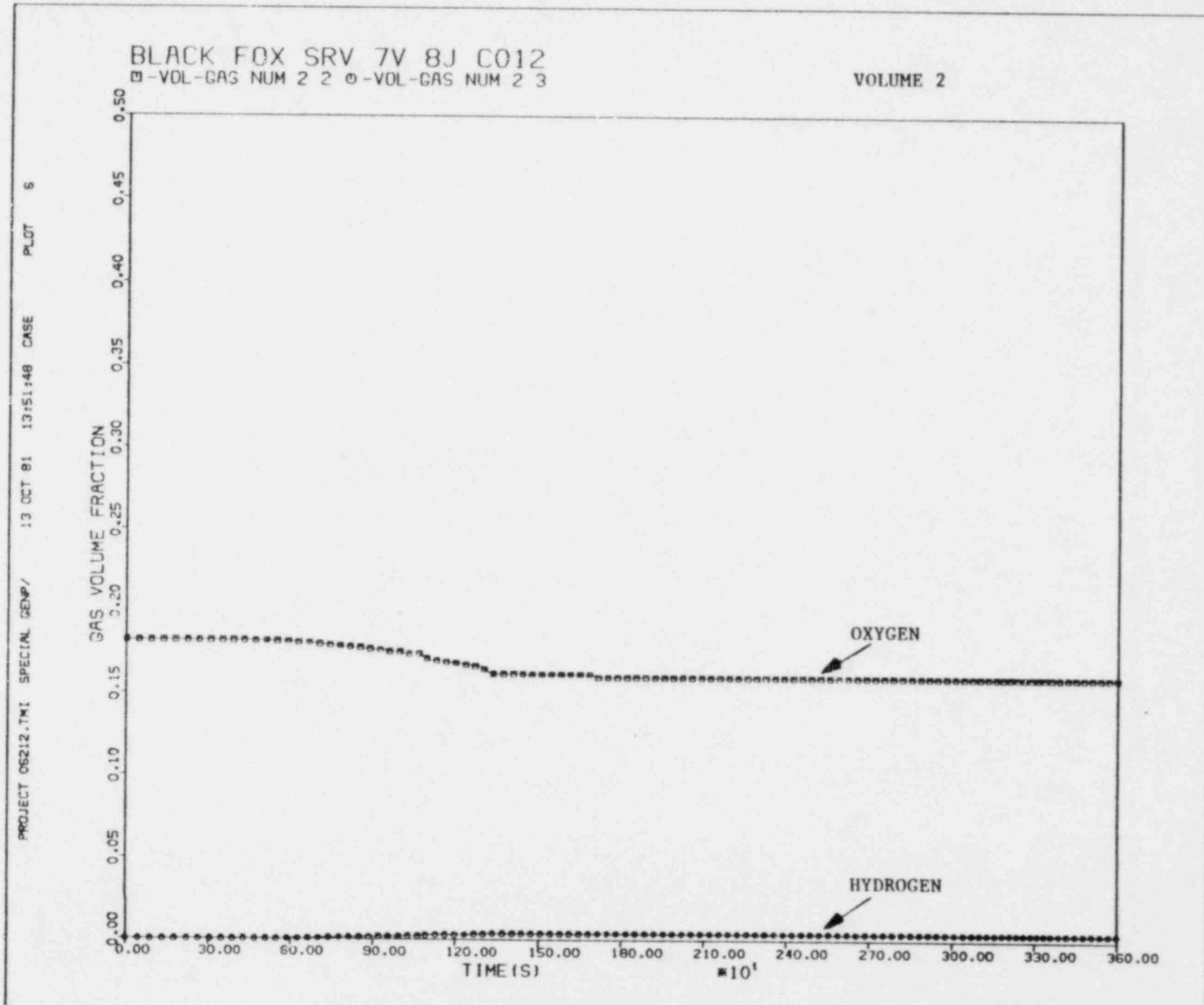


FIGURE (2)(ix)-30

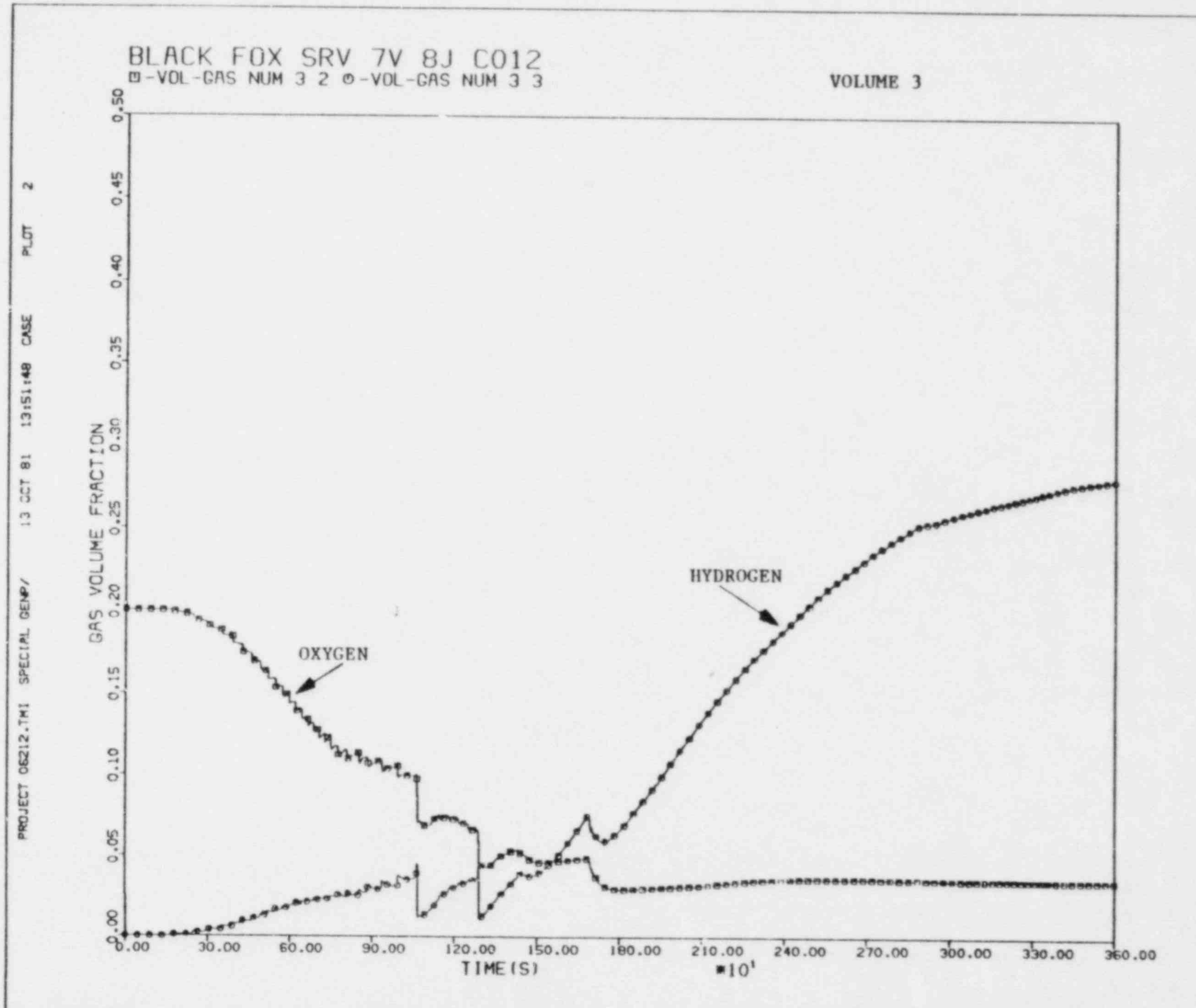


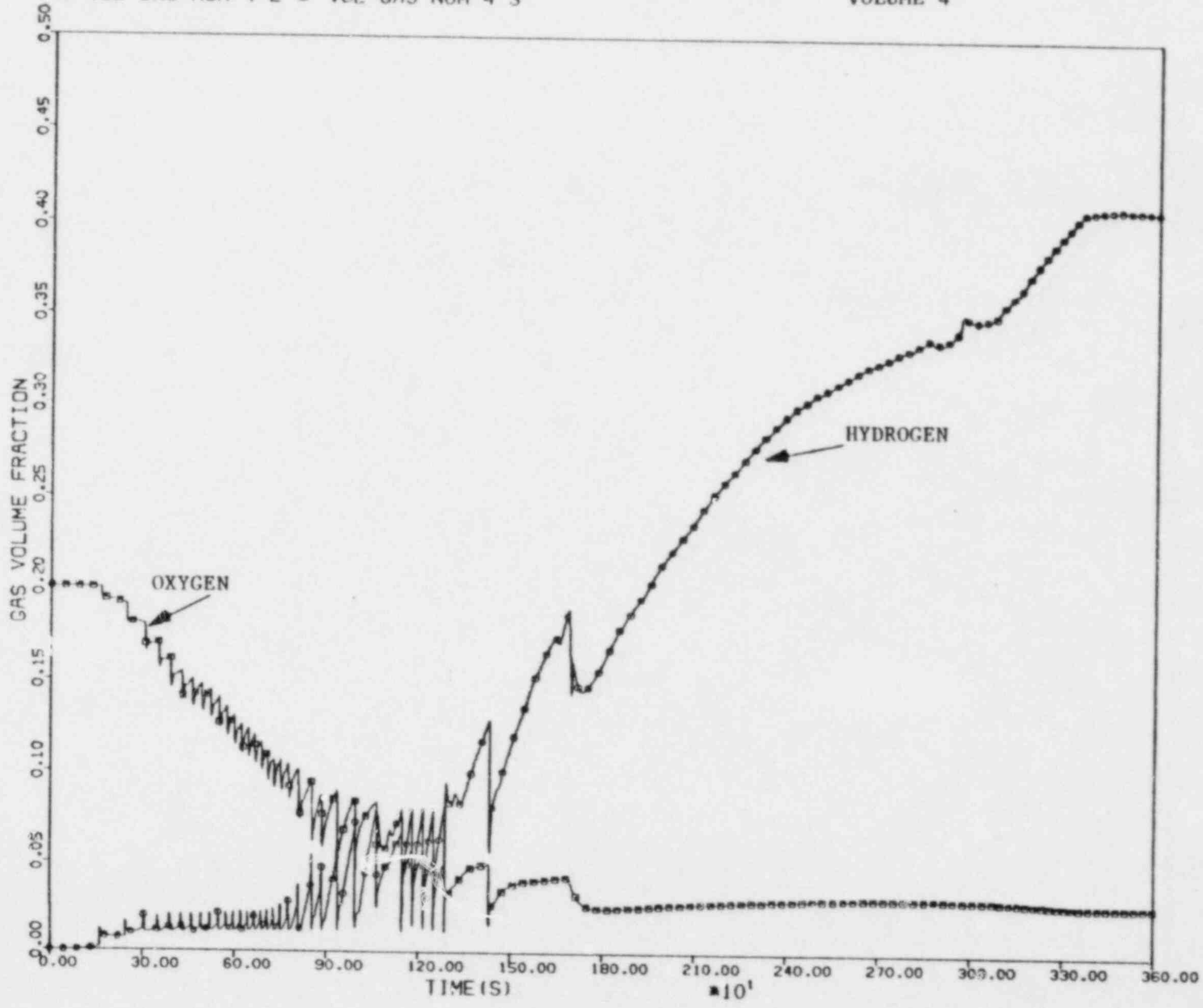
FIGURE (2)(ix)-31

USER: 03636SCHADE PLOT FILE: 03636SCHADE*PL*135148A.

BLACK FOX SRV 7V 8J C012

⊡-VOL-GAS NUM 4 2 ⊙-VOL-GAS NUM 4 3

VOLUME 4



USER: 03636SCHADE PLOT FILE: 03636SCHADE#PL#135148A.

PROJECT 06212.TM1 SPECIAL GENP/ 13 OCT 81 13:51:43 CASE PLOT 3

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19-111381

FIGURE (2)(ix)-32

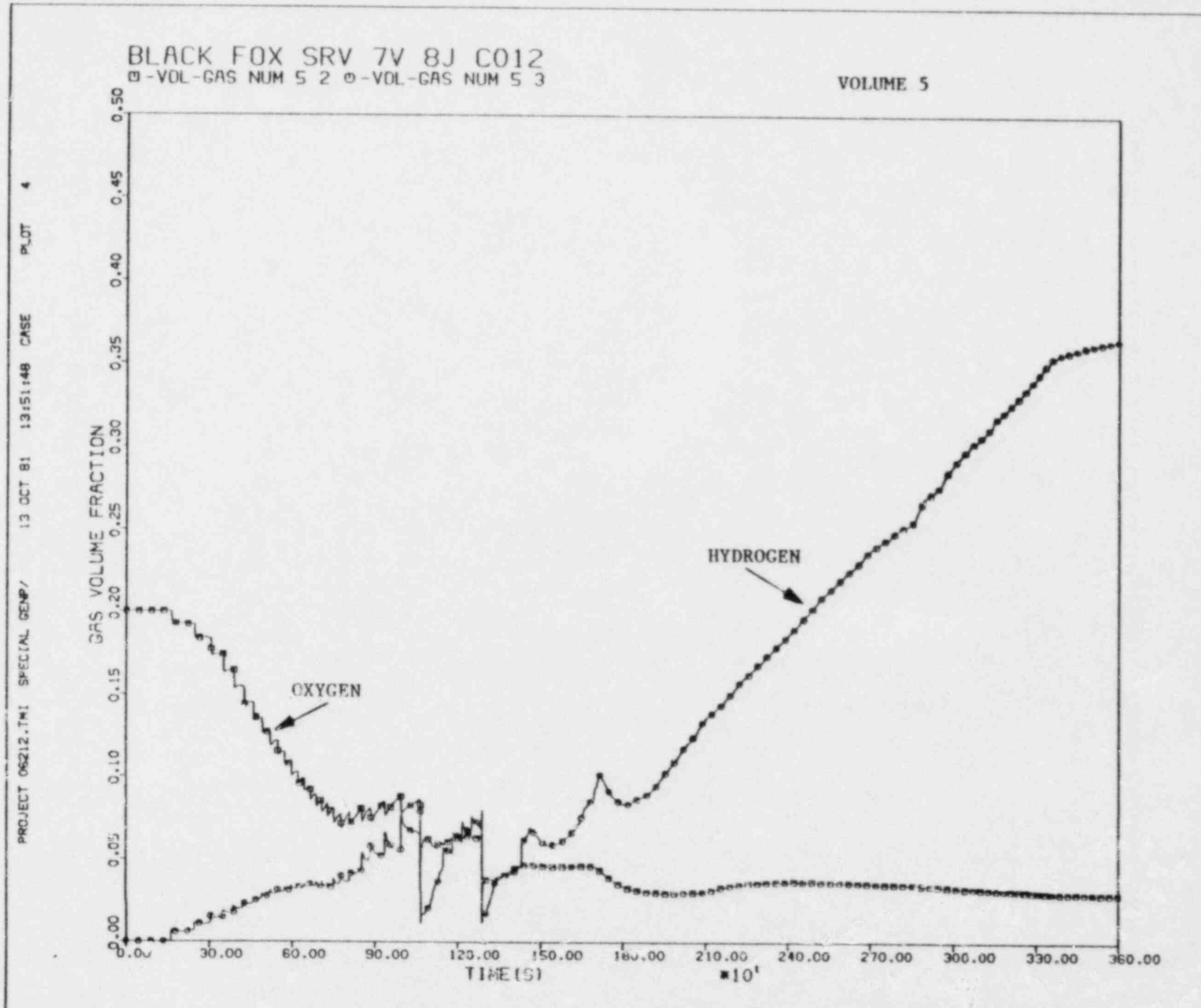
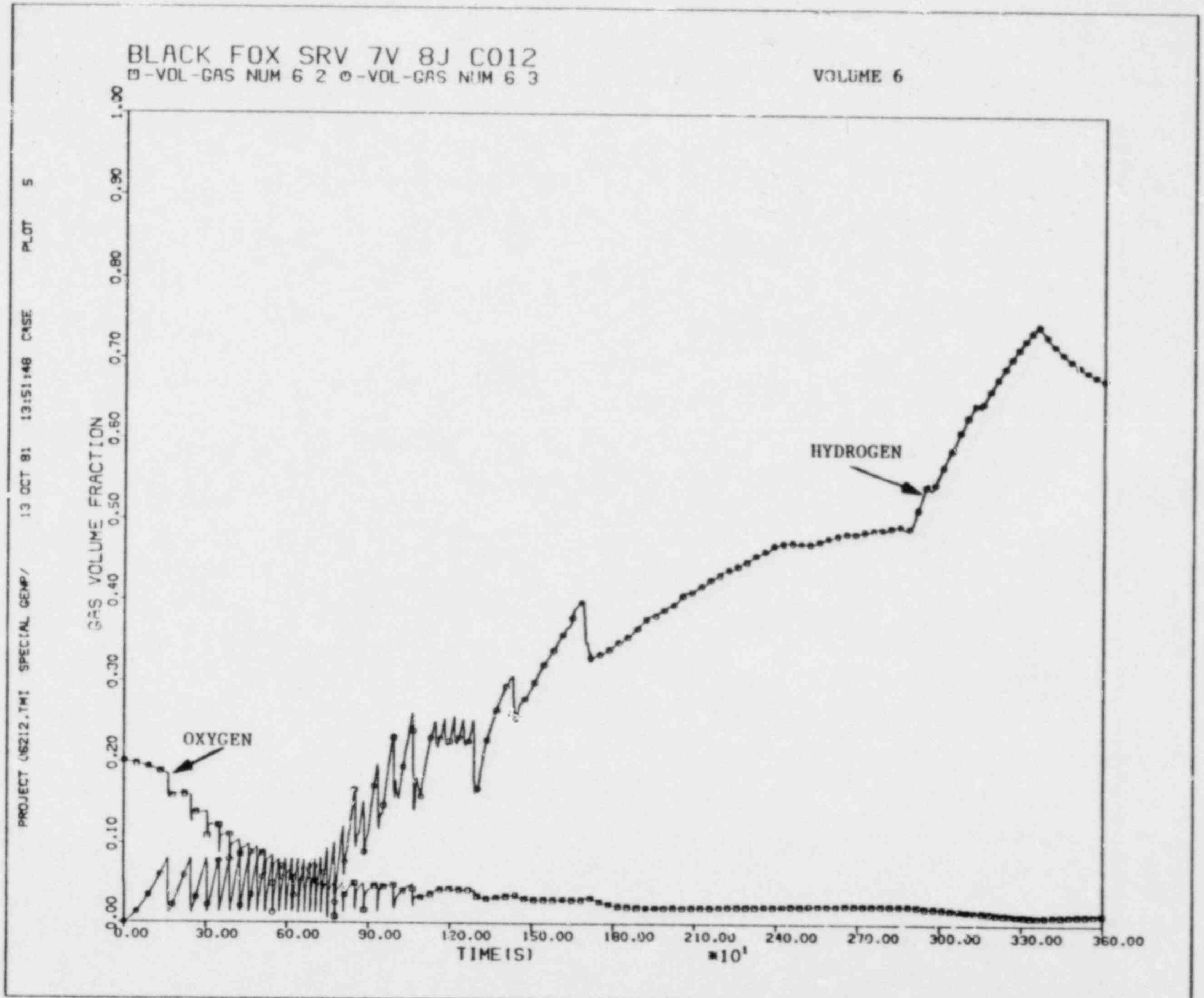


FIGURE (2)(1x)-33

357

19-111381

USER: 03636SCHADE PLOT FILE: 03636SCHADEWPL#135148A.



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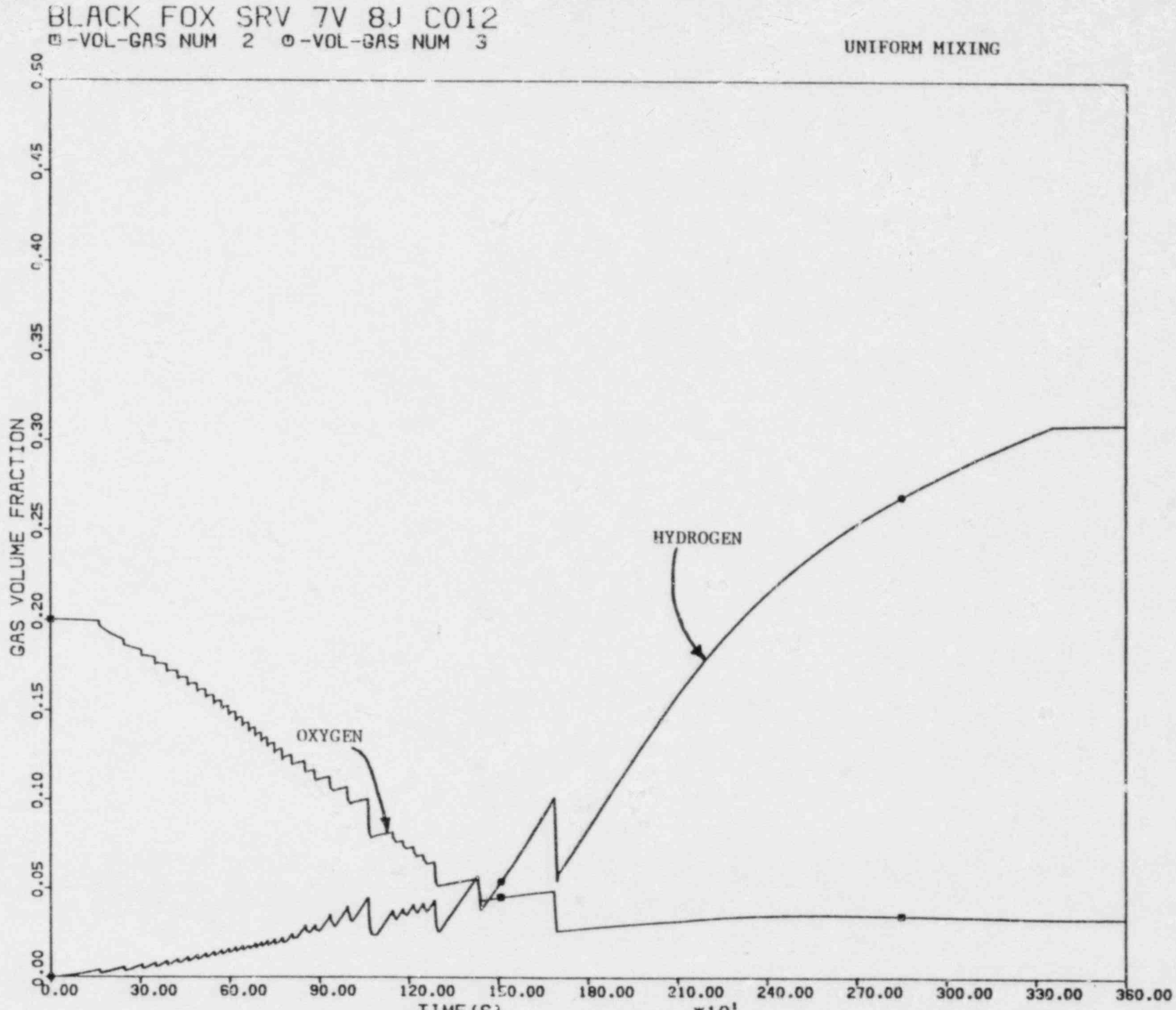
19-111381

FIGURE (2)(1x)-34



PROJECT 06212.TMI SPECIAL 0000/ 14 OCT 81 23:35:50 CASE PLOT 1

(2)(1x)-35

19-111381

USER: 03636SCHADE PLOT FILE: 03636SCHADE*PL*23354OA.

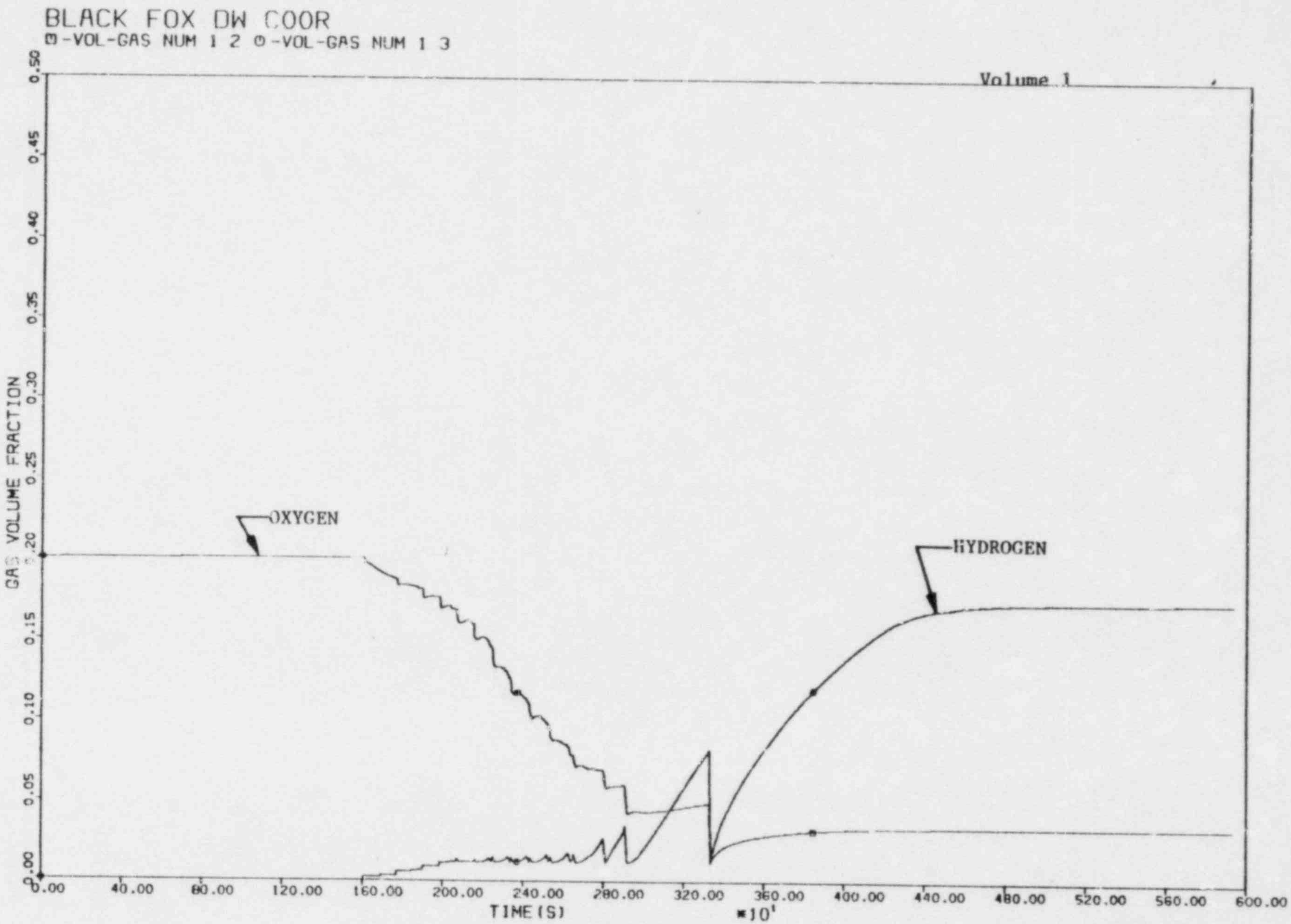


Figure (2)(1x)-36

PROJECT 06212.TMI SPECIAL GENP/ 16 OCT 81 16:18:32 CASE PLOT 1

19-111381

USER: 03638SCHADE PLOT FILE: 03638SCHADE*PL*181832A.

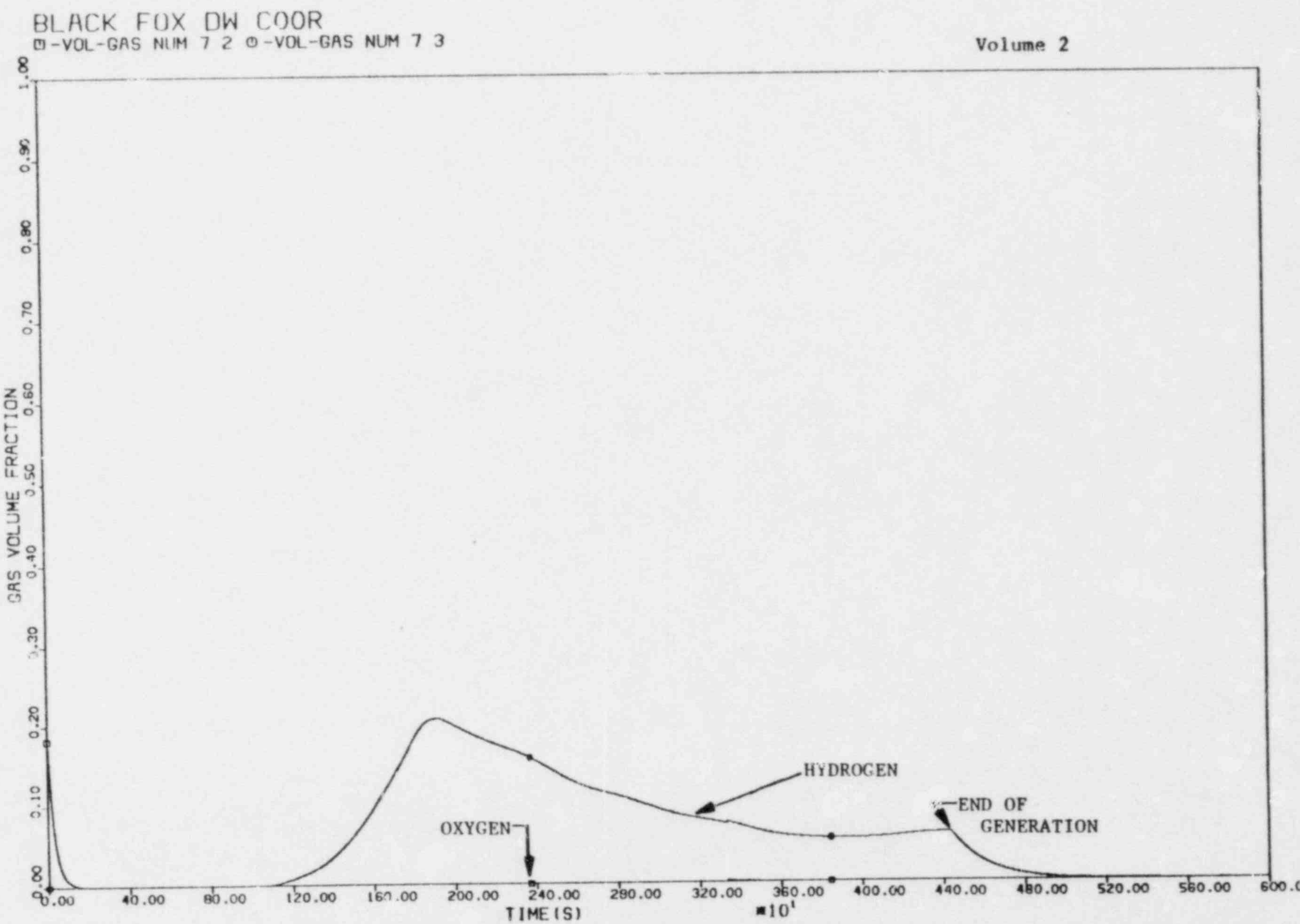


Figure (2)(ix)-37

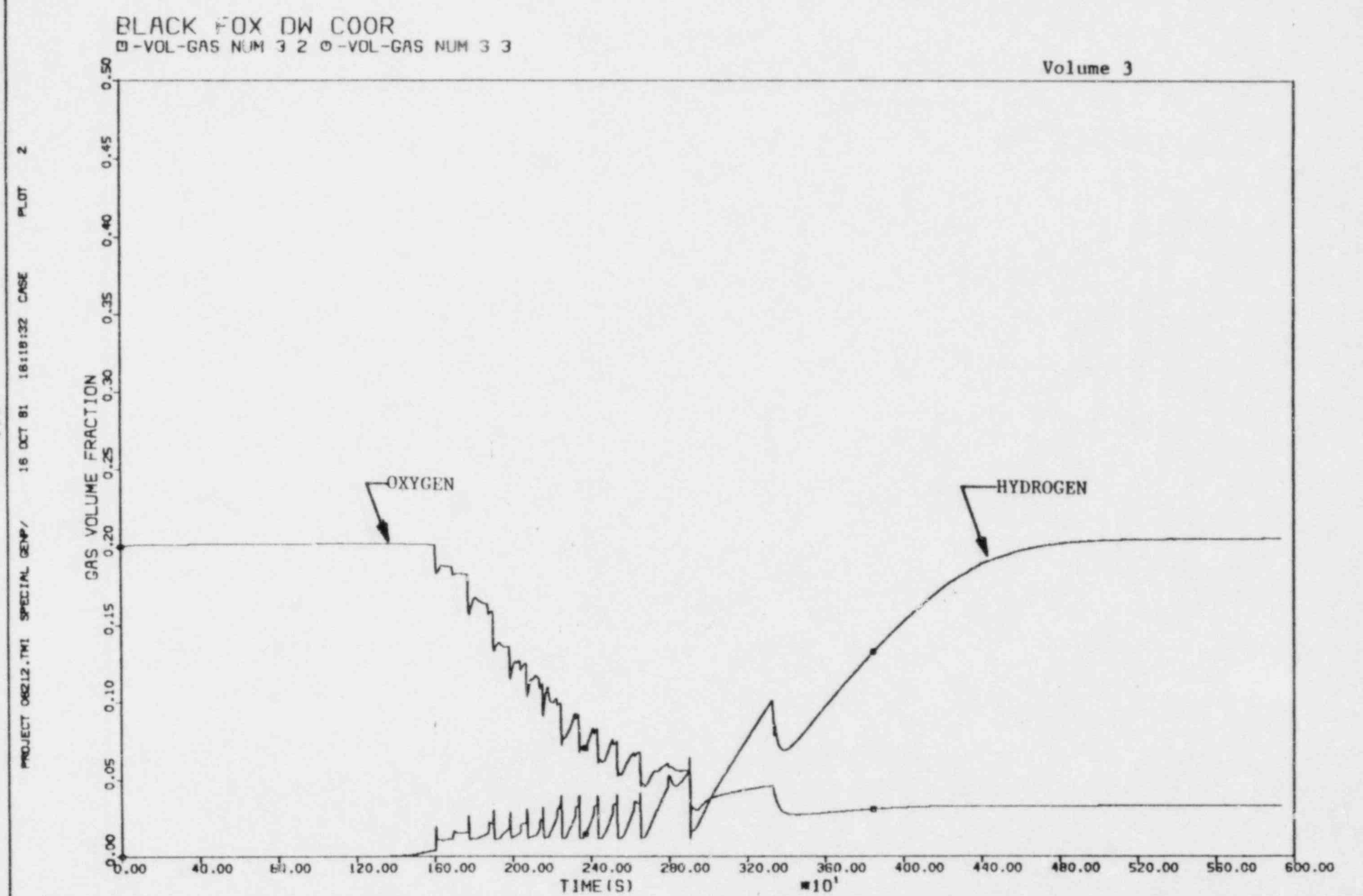


Figure (2)(ix)-38

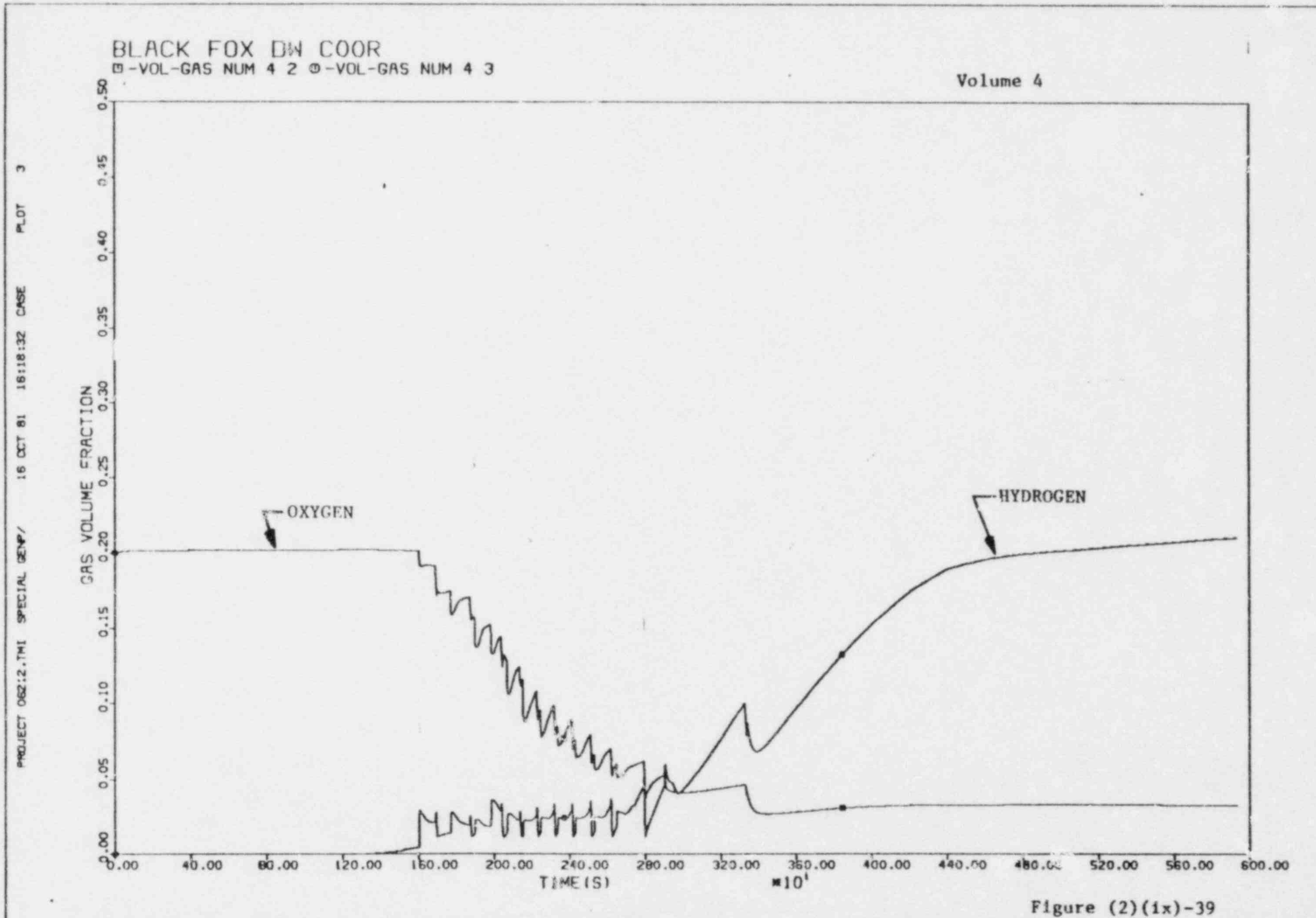


Figure (2)(1x)-39

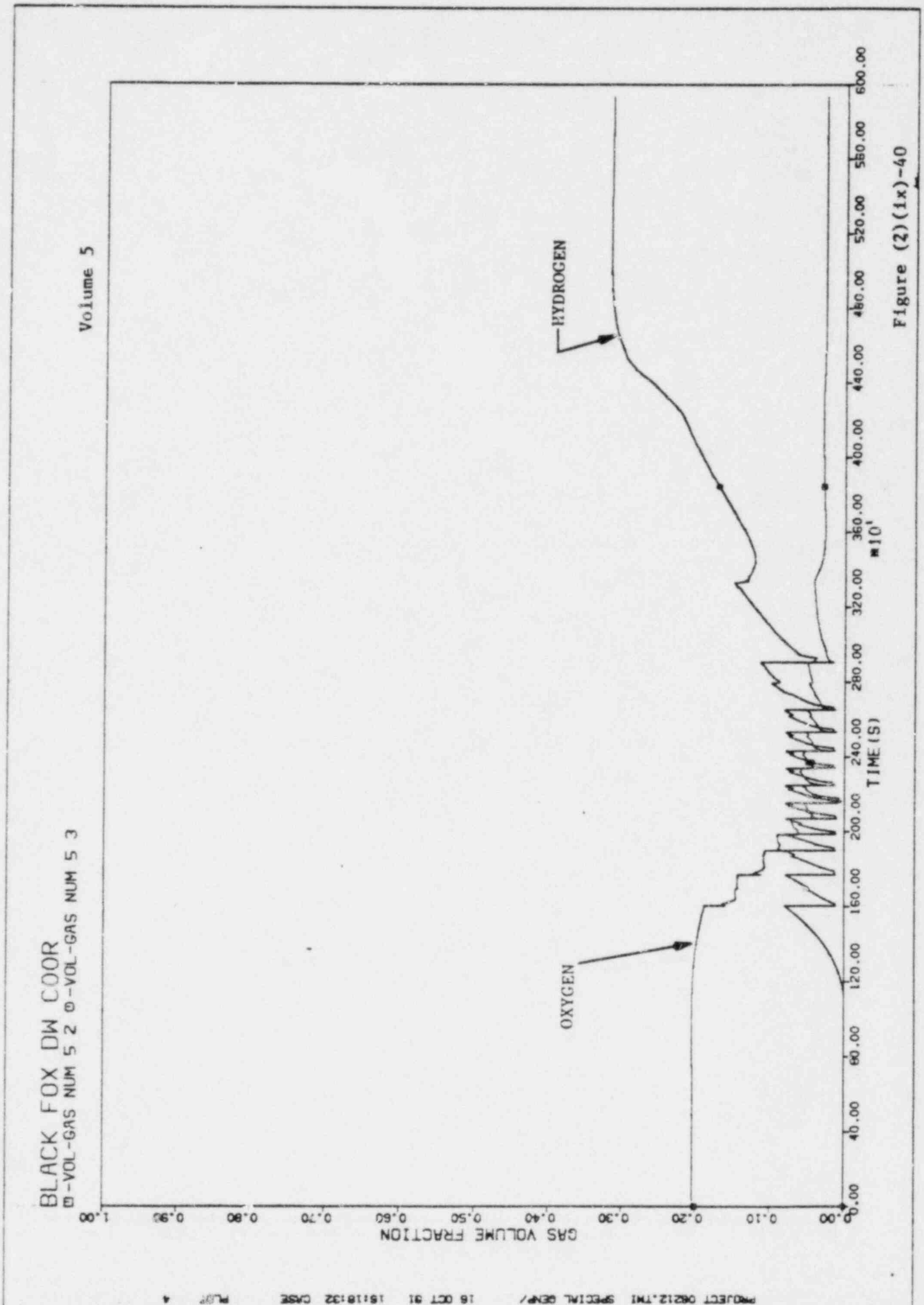


Figure (2)(1x)-40

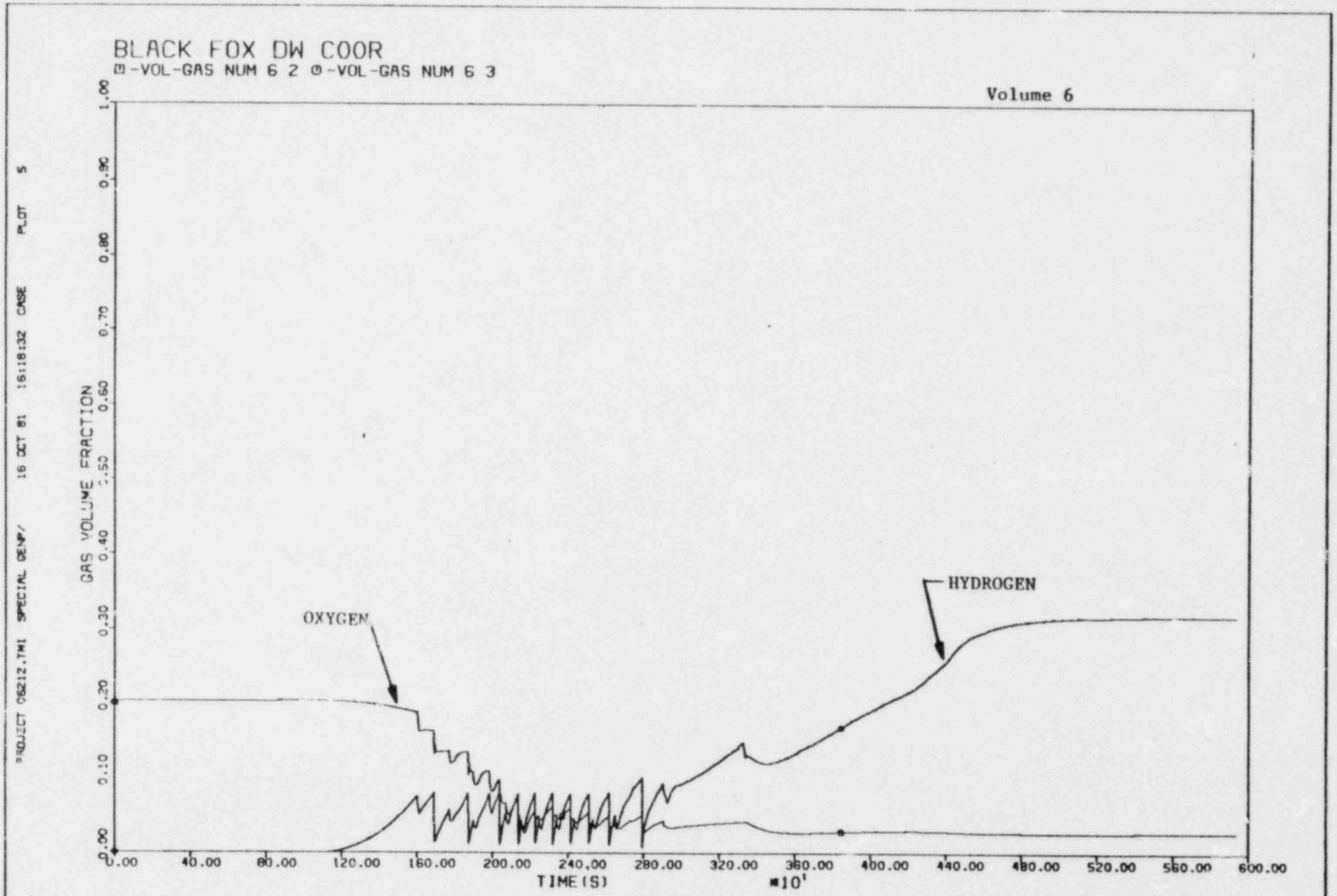


Figure (2)(ix)-41

USER: 03636SCHADE PLOT FILE: 03636SCHADE=PL=161932A.

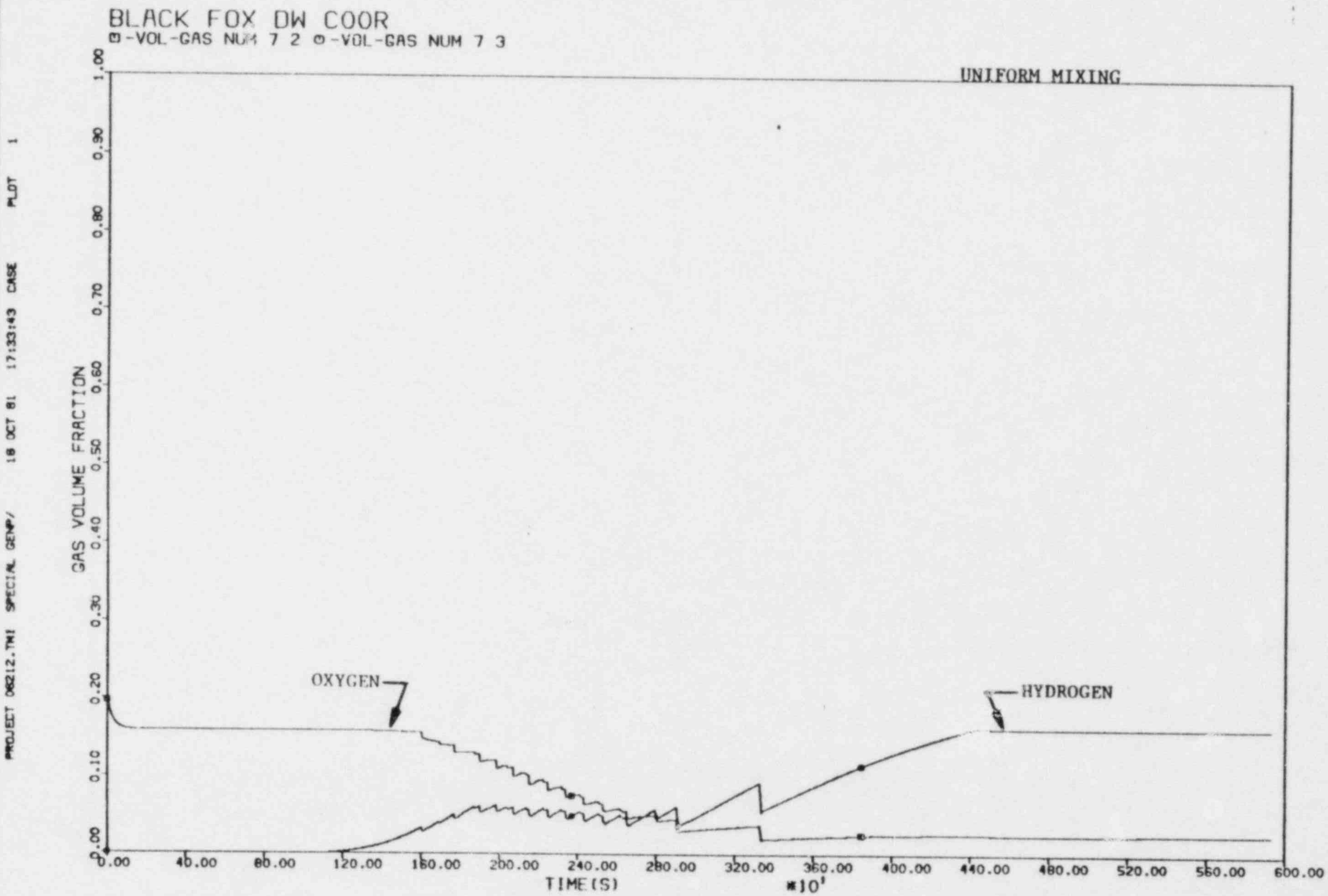
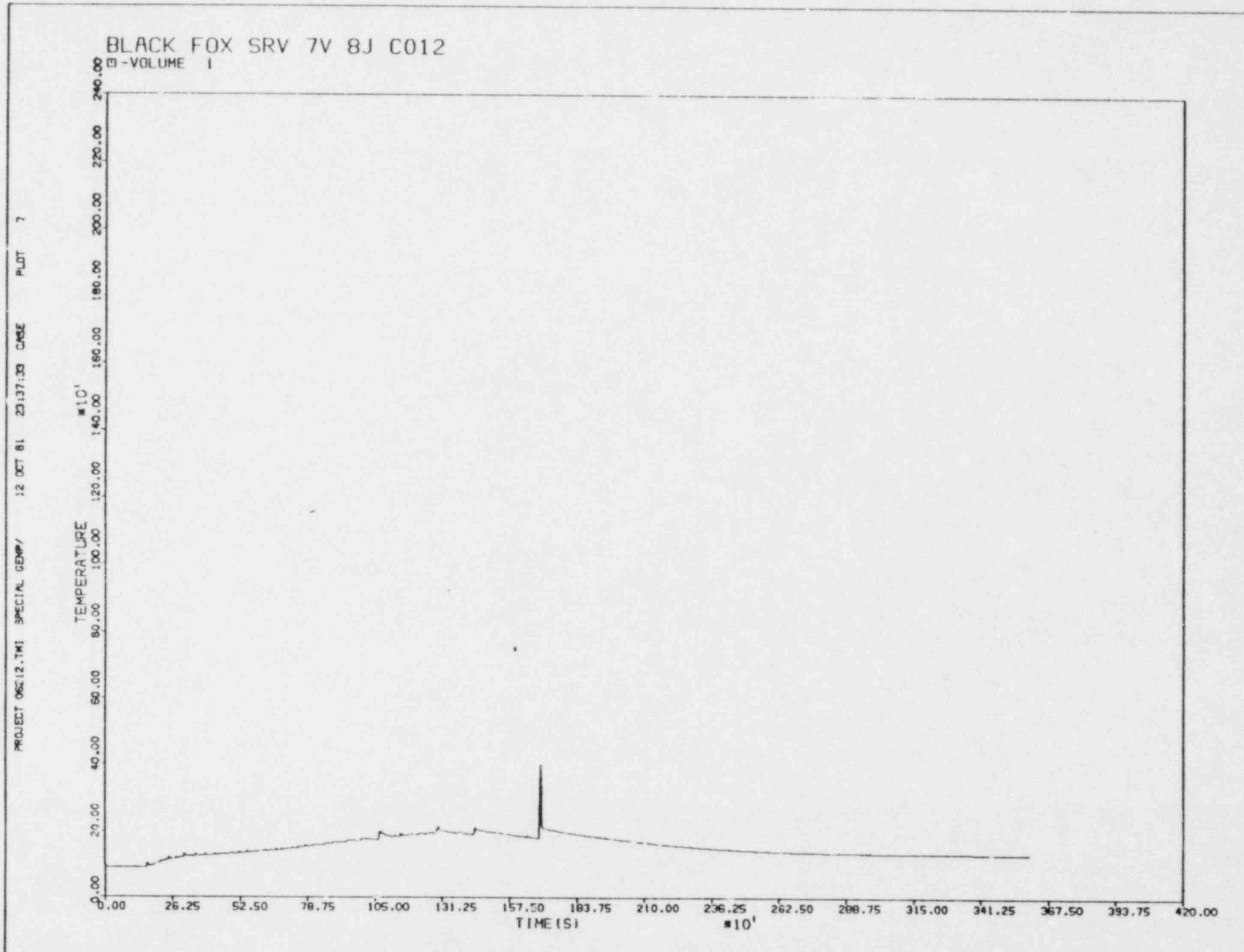


Figure (2)(ix)-42



USER: 03638SCHADE PLOT FILE: 03638SCHADE*PL*2337733A.

FIGURE (2)(ix)-43

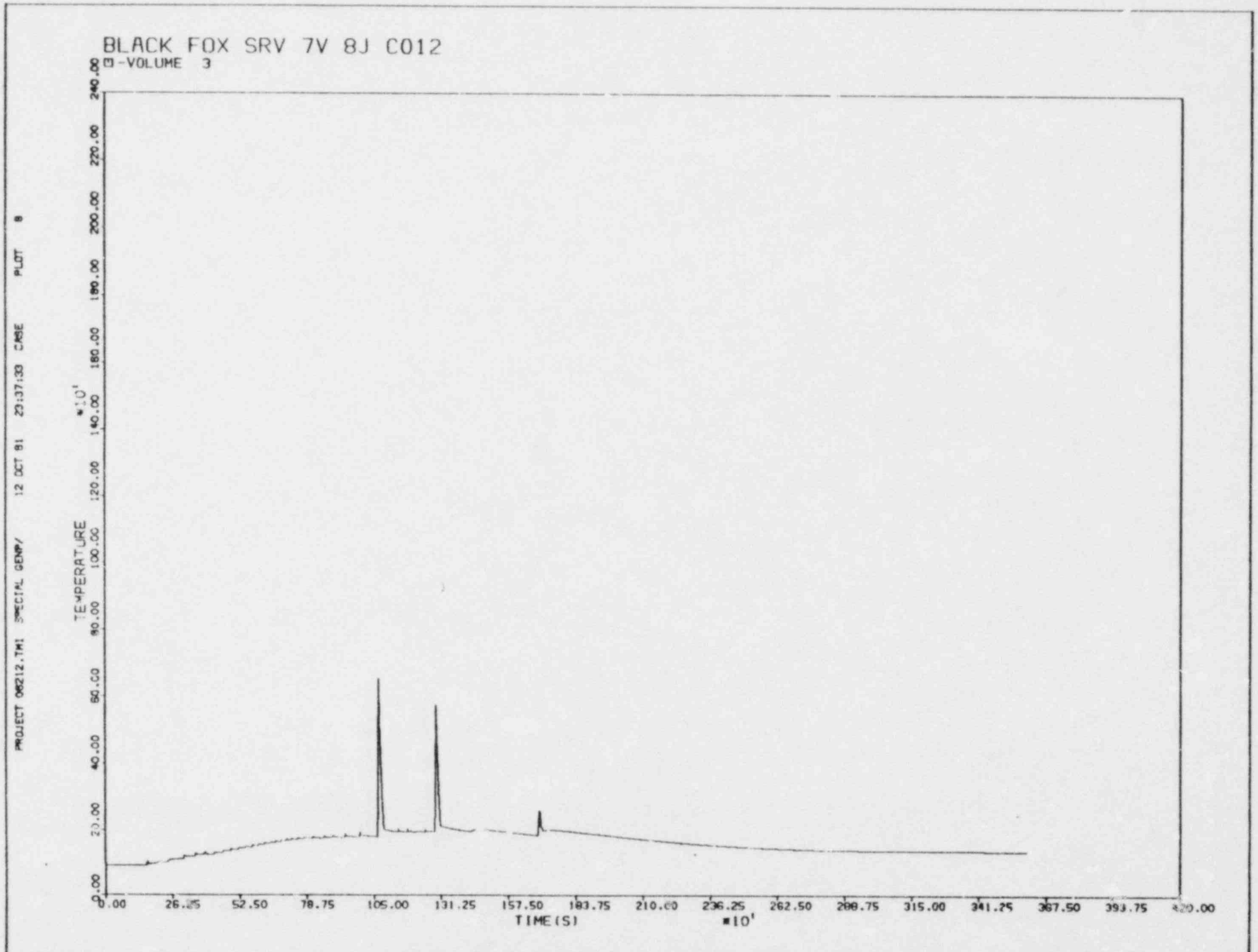


FIGURE (2)(1x)-44

USER: 036365SCHADE PLOT FILE: 036365SCHADE*PL#233733A.

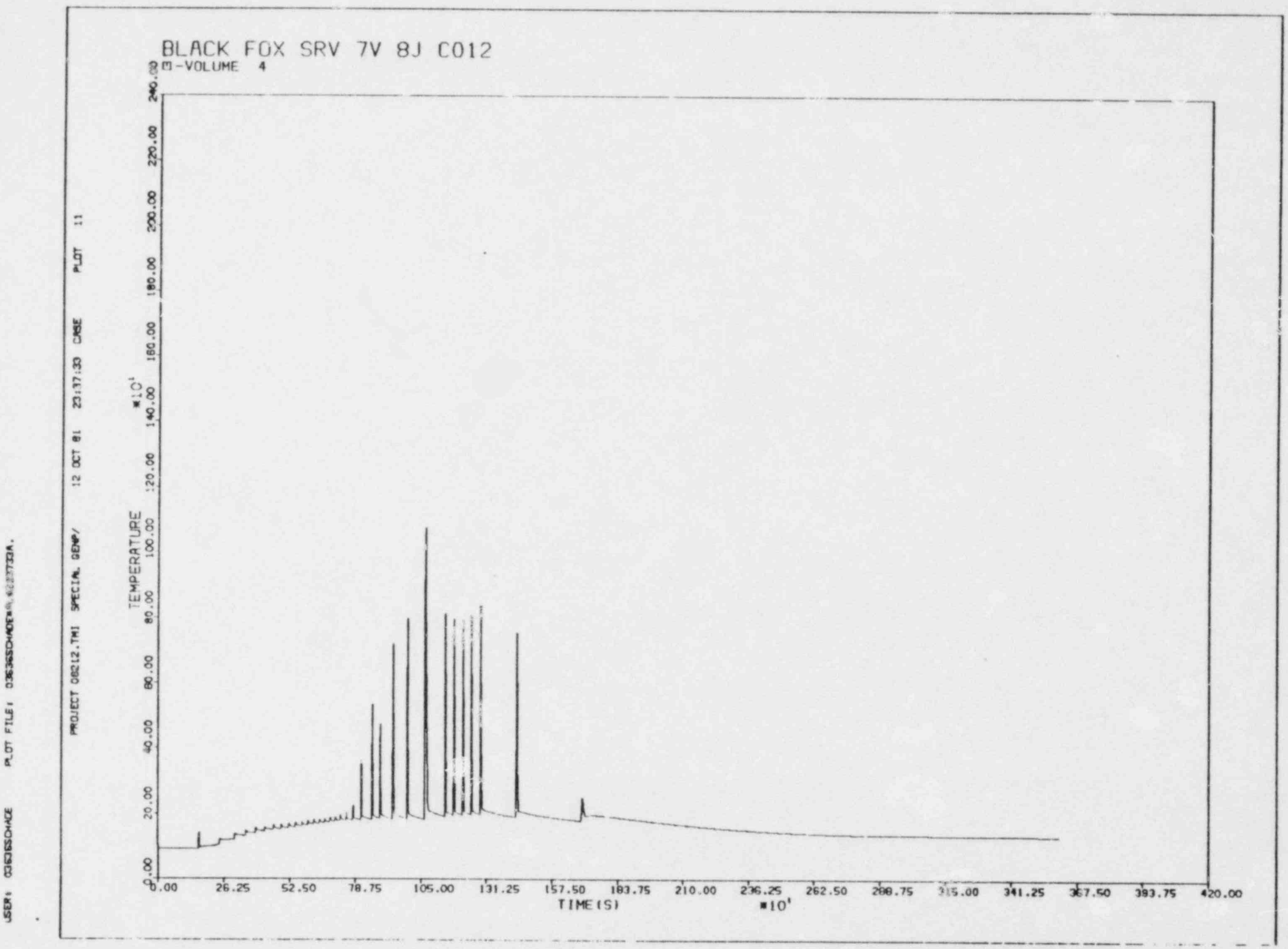


FIGURE (2)(1x)-45

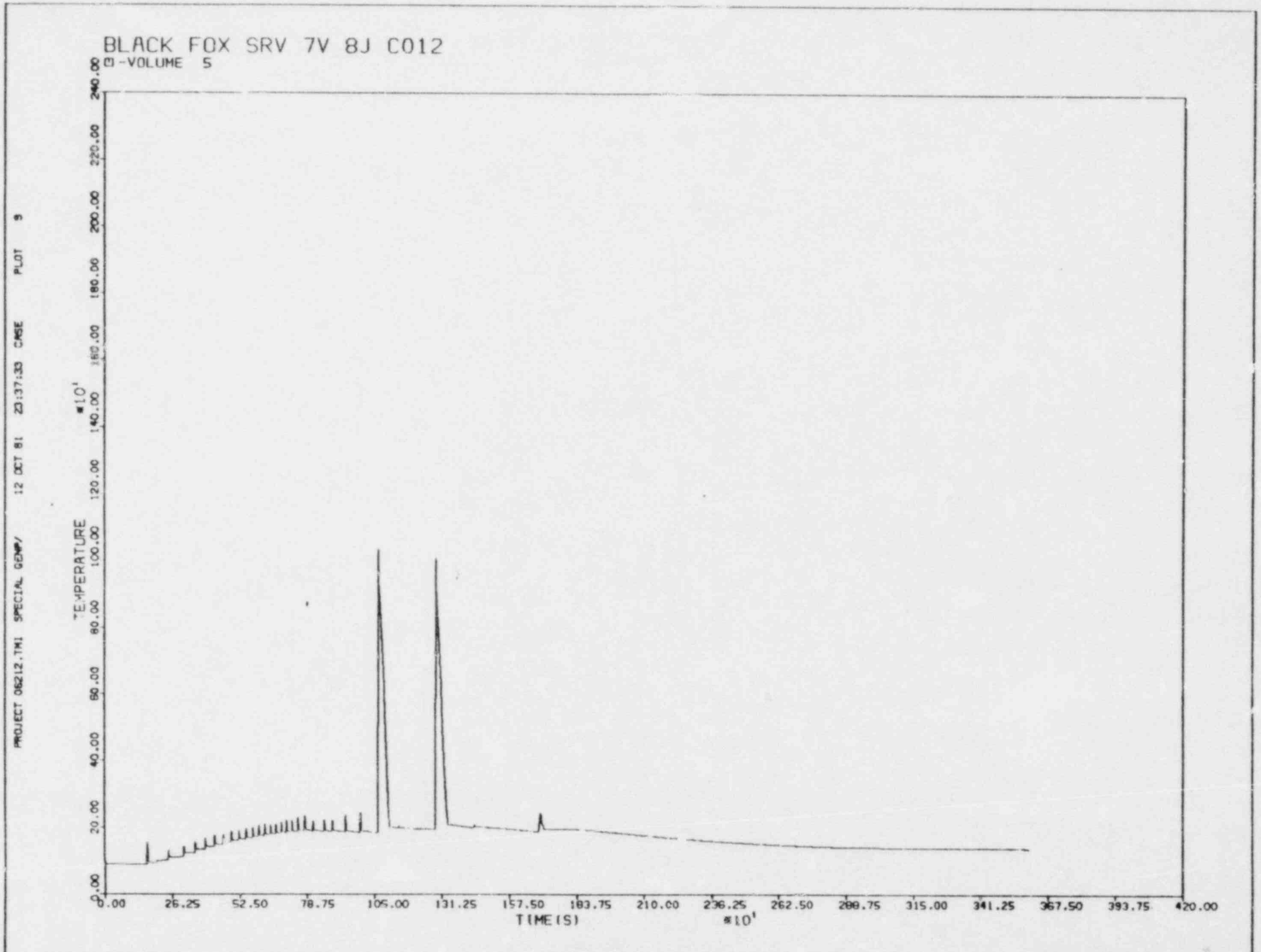
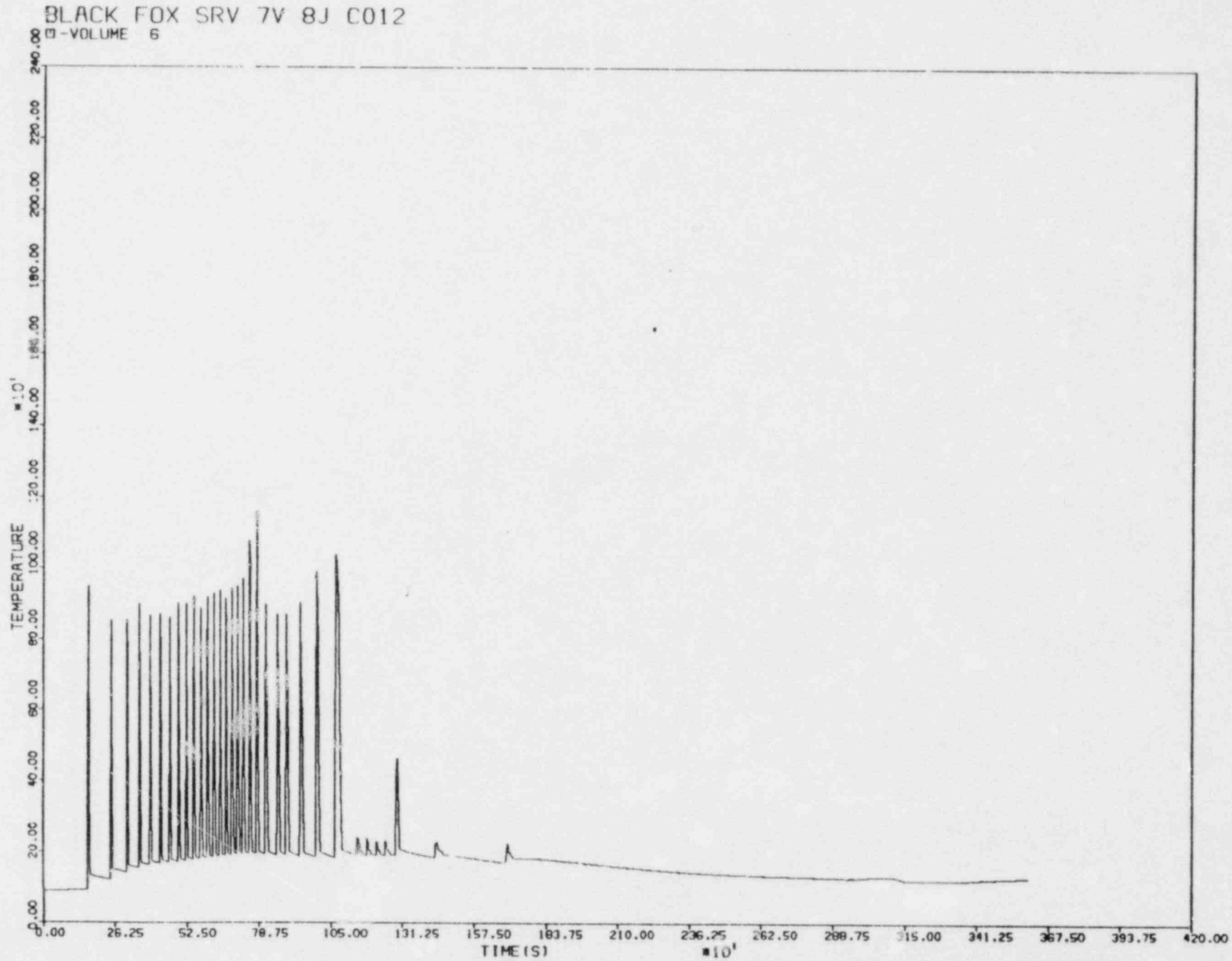


FIGURE (2)(1x)-46



USER: 03635SCHACE PLOT FILE: 03635SCHACE*PL8233733A.

FIGURE (2)(1x)-47

19-111381

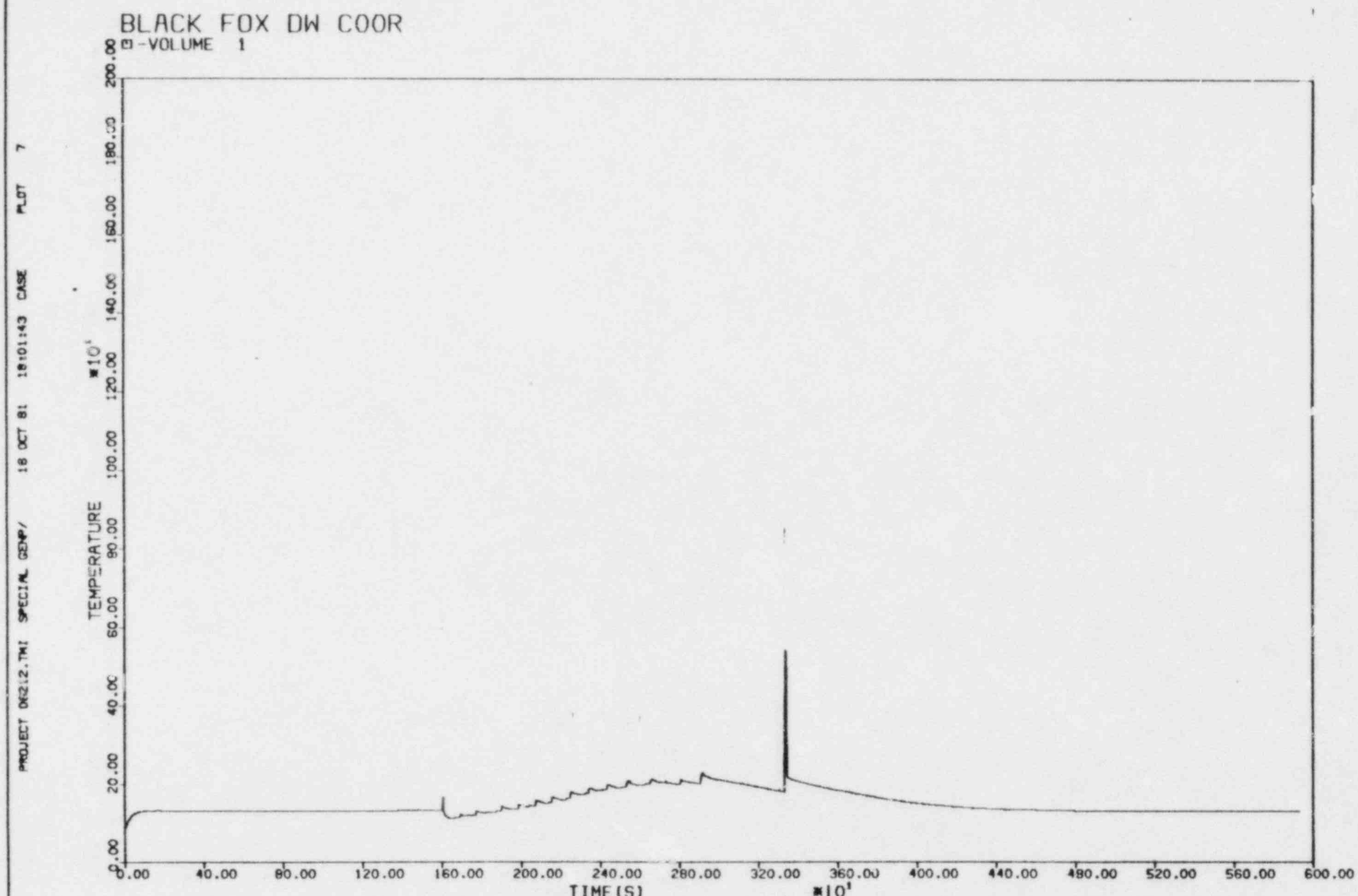


Figure (2)(1x)-48

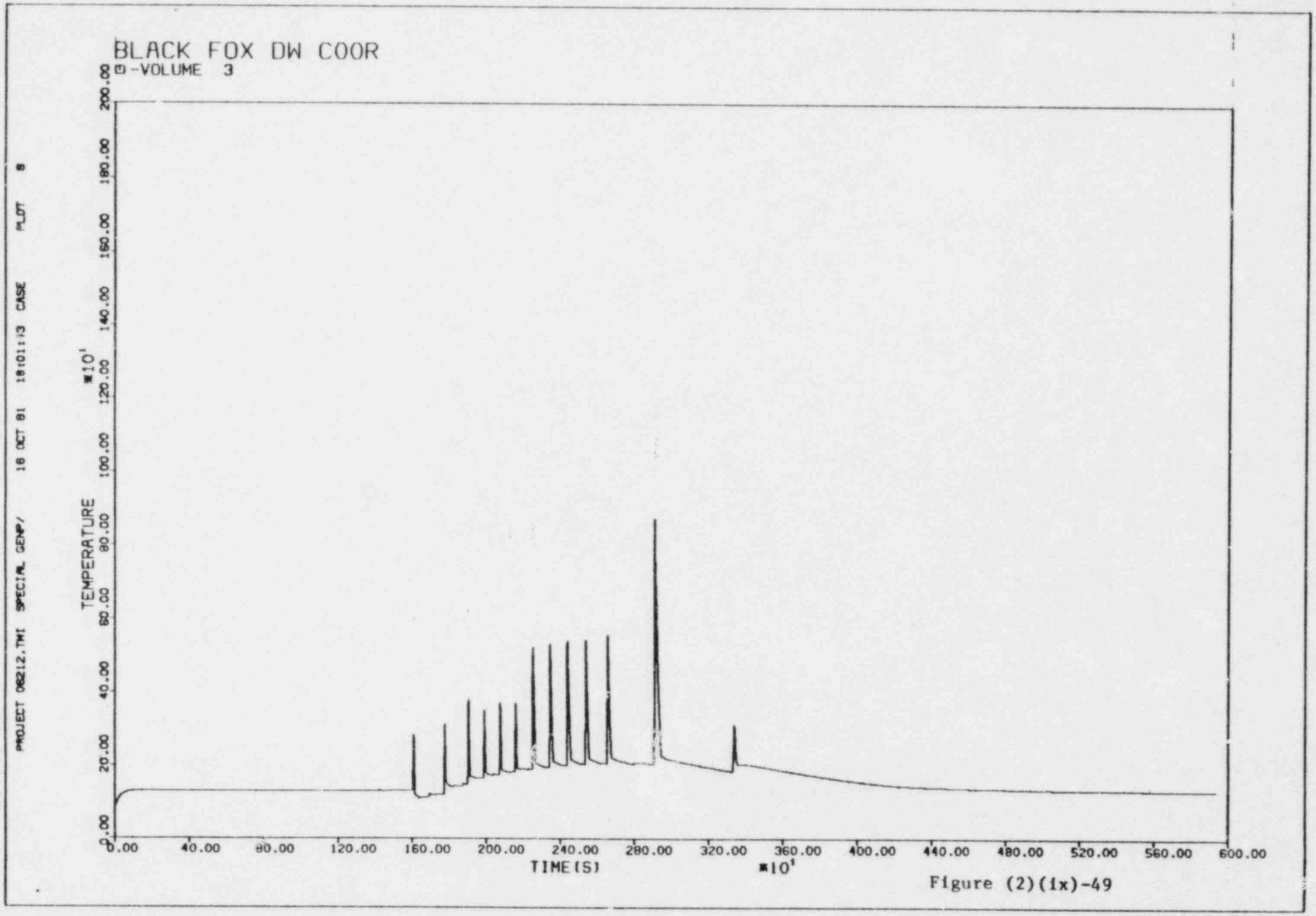


Figure (2)(ix)-49

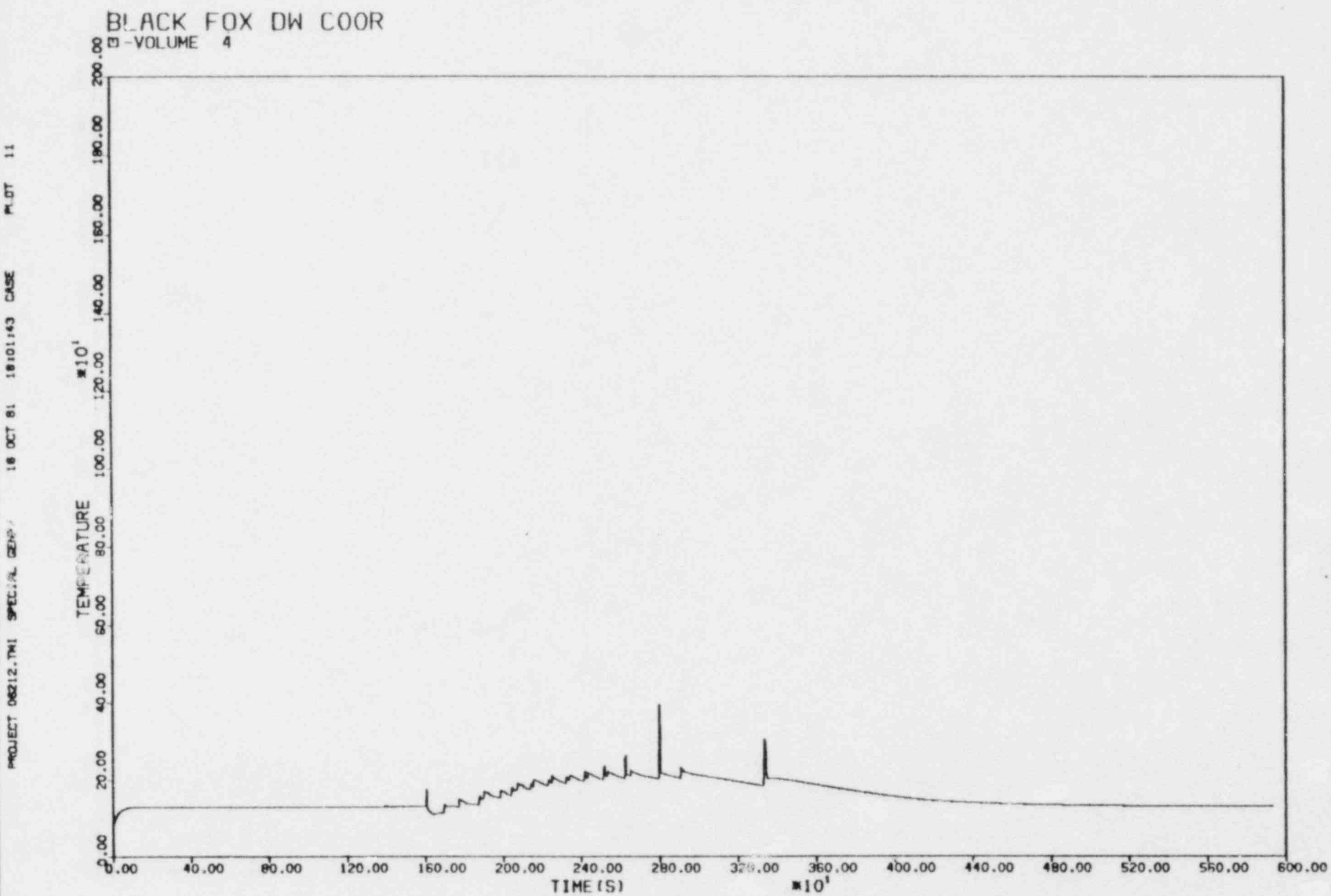


Figure (2)(ix)-50

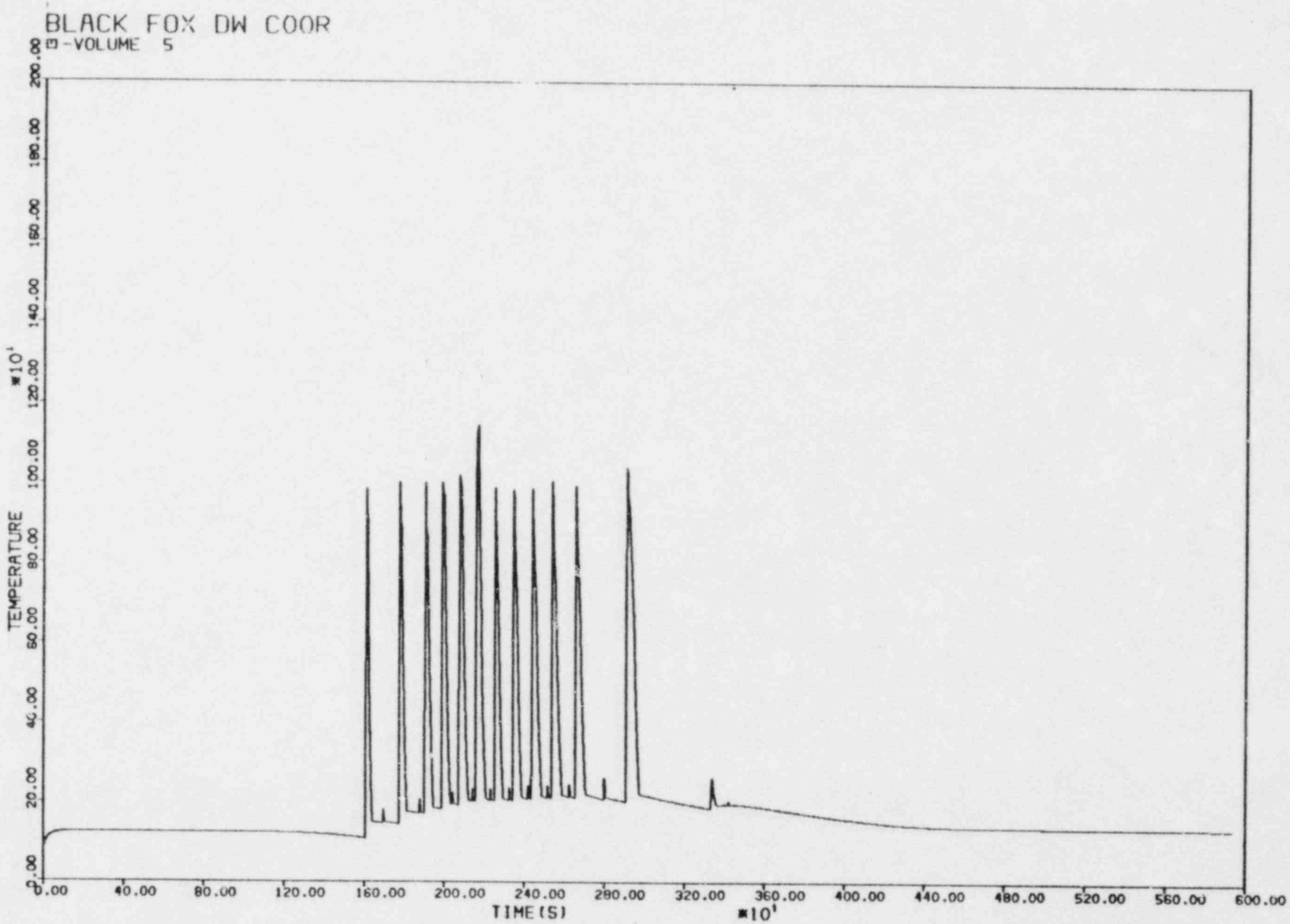


Figure (2)(1x)-51

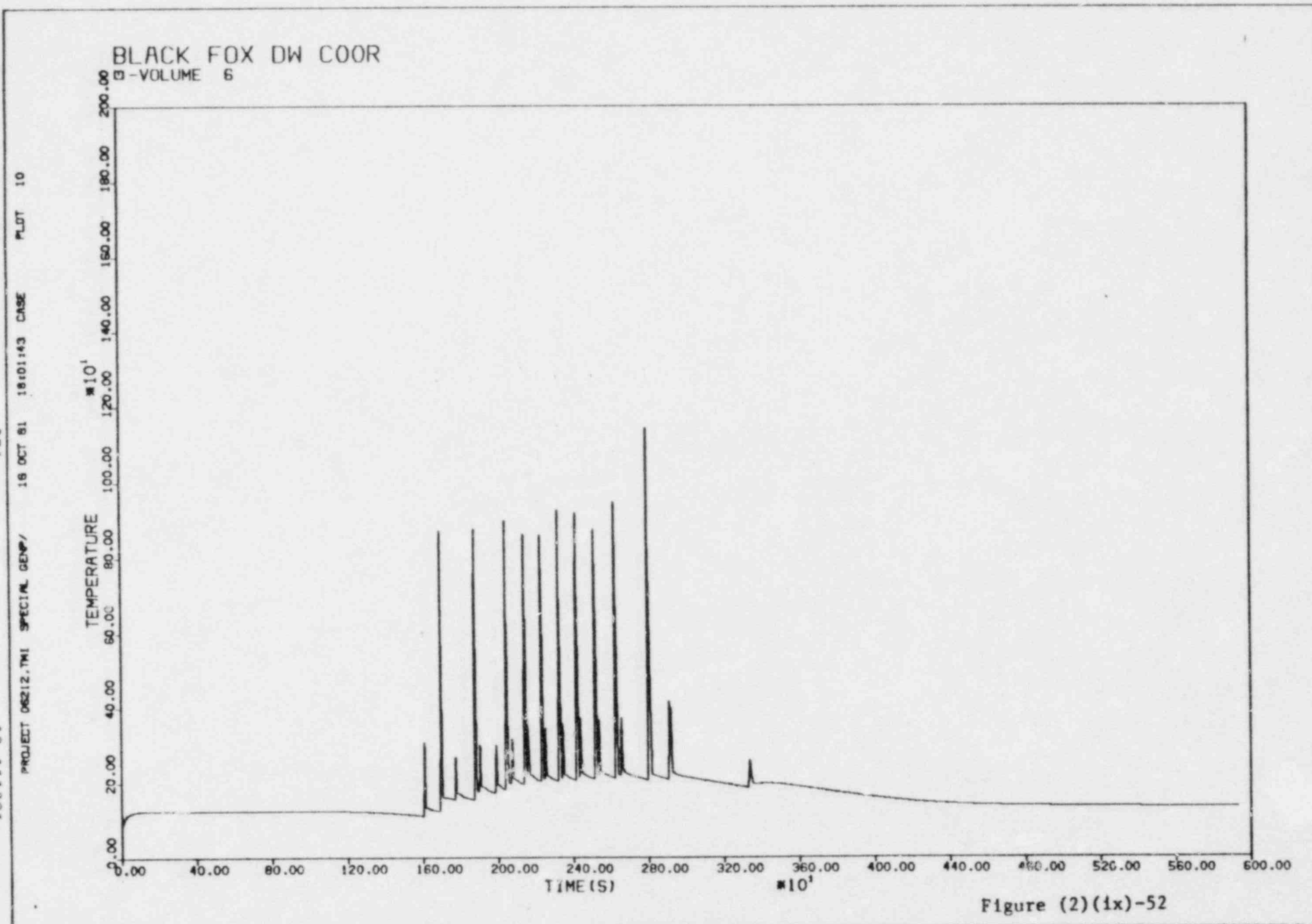
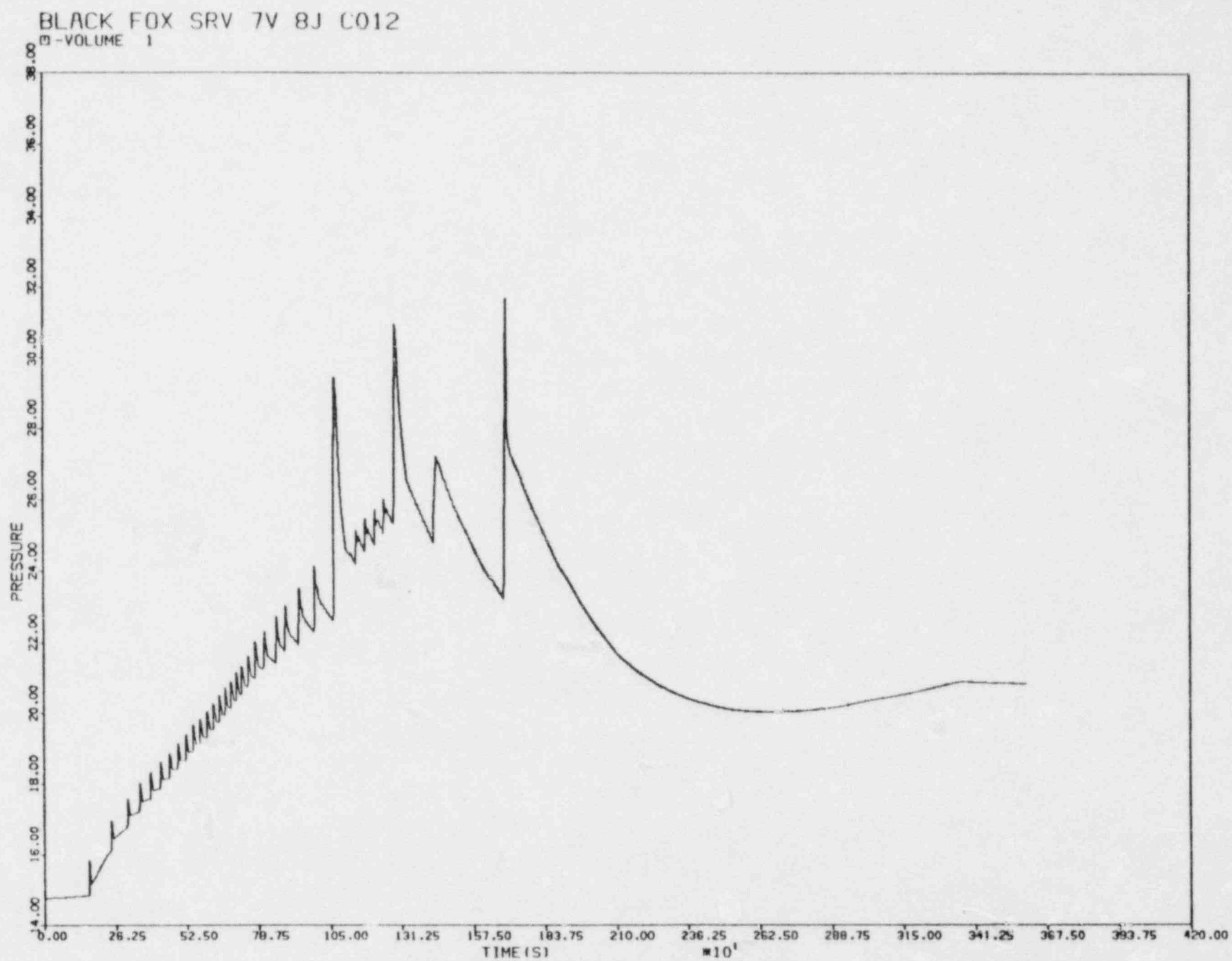


Figure (2)(ix)-52

19-111381

USER: 03638SCHADE PLOT FILE: 03638SCHADE*PL01801430.

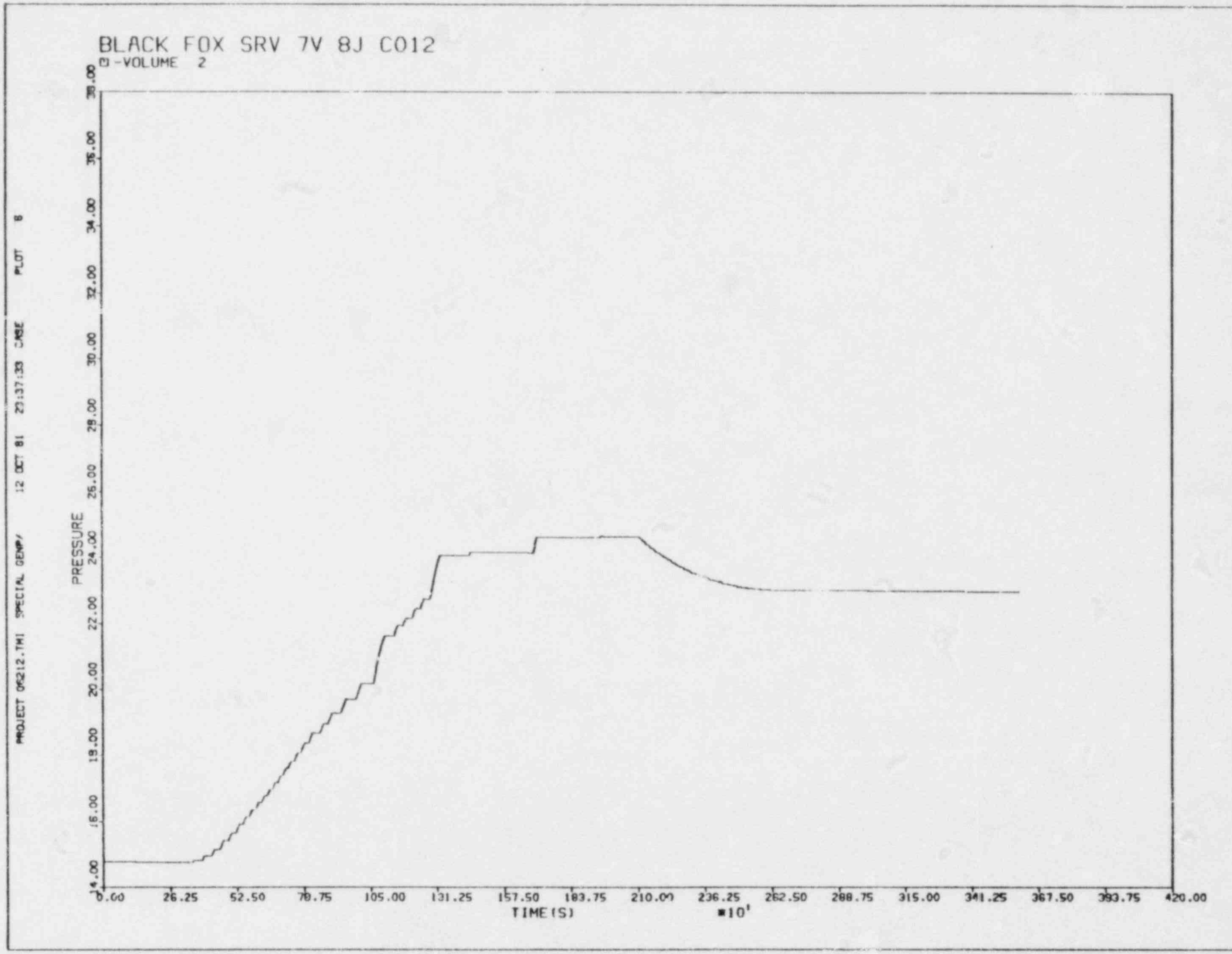


PROJECT 06212.TMI SPECIAL GENP/ 12 OCT 81 23:37.33 CASE PLOT 1

USER: 036355CHACE PLOT FILE: 036355CHACEMP.#2337733A.

FIGURE (2)(1x)-53

19-111381



USER: 036365SCHADE PLOT FILE: 036365SCHADE*PL.R233733A.

FIGURE (2)(1x)-54

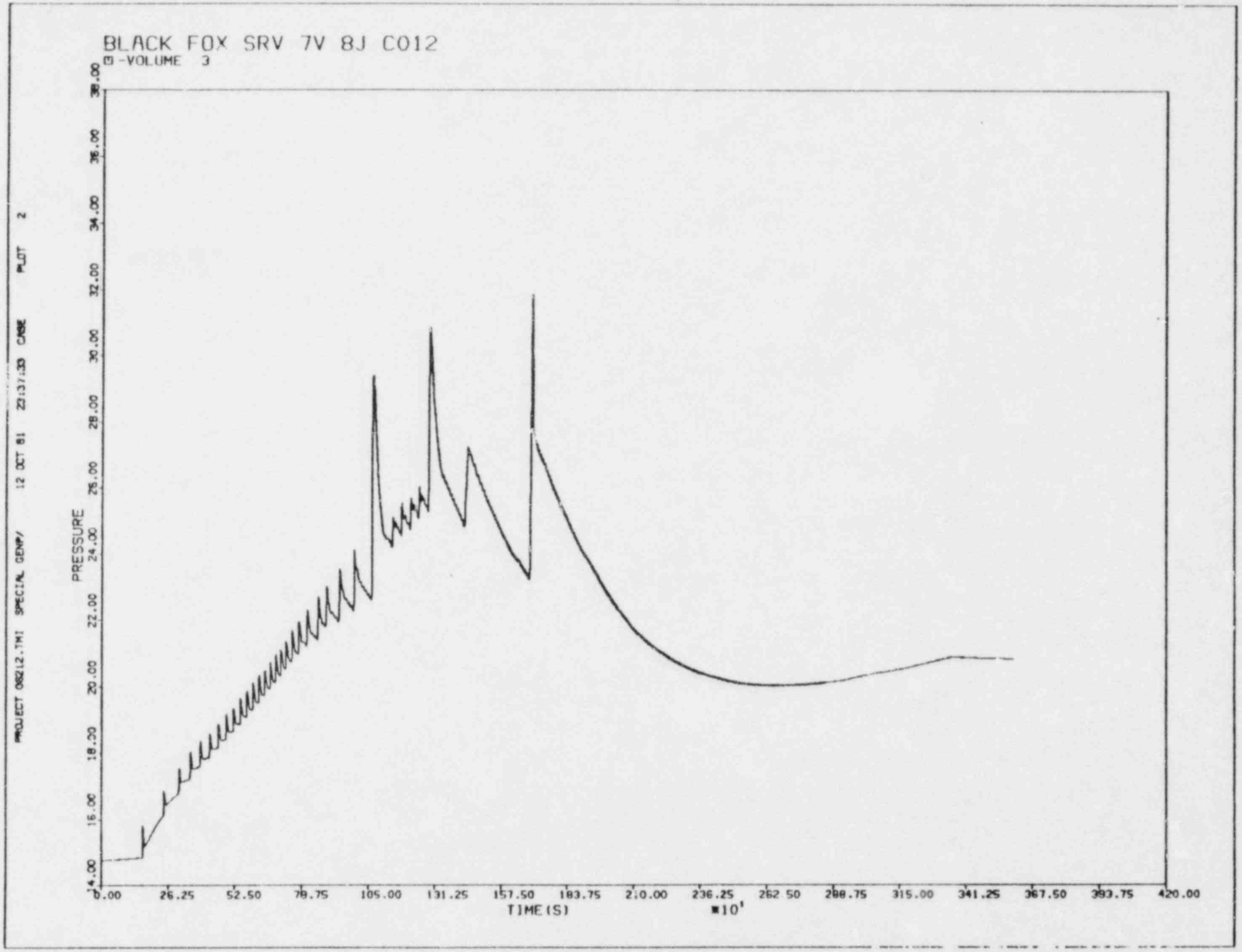


FIGURE (2)(1x)-55

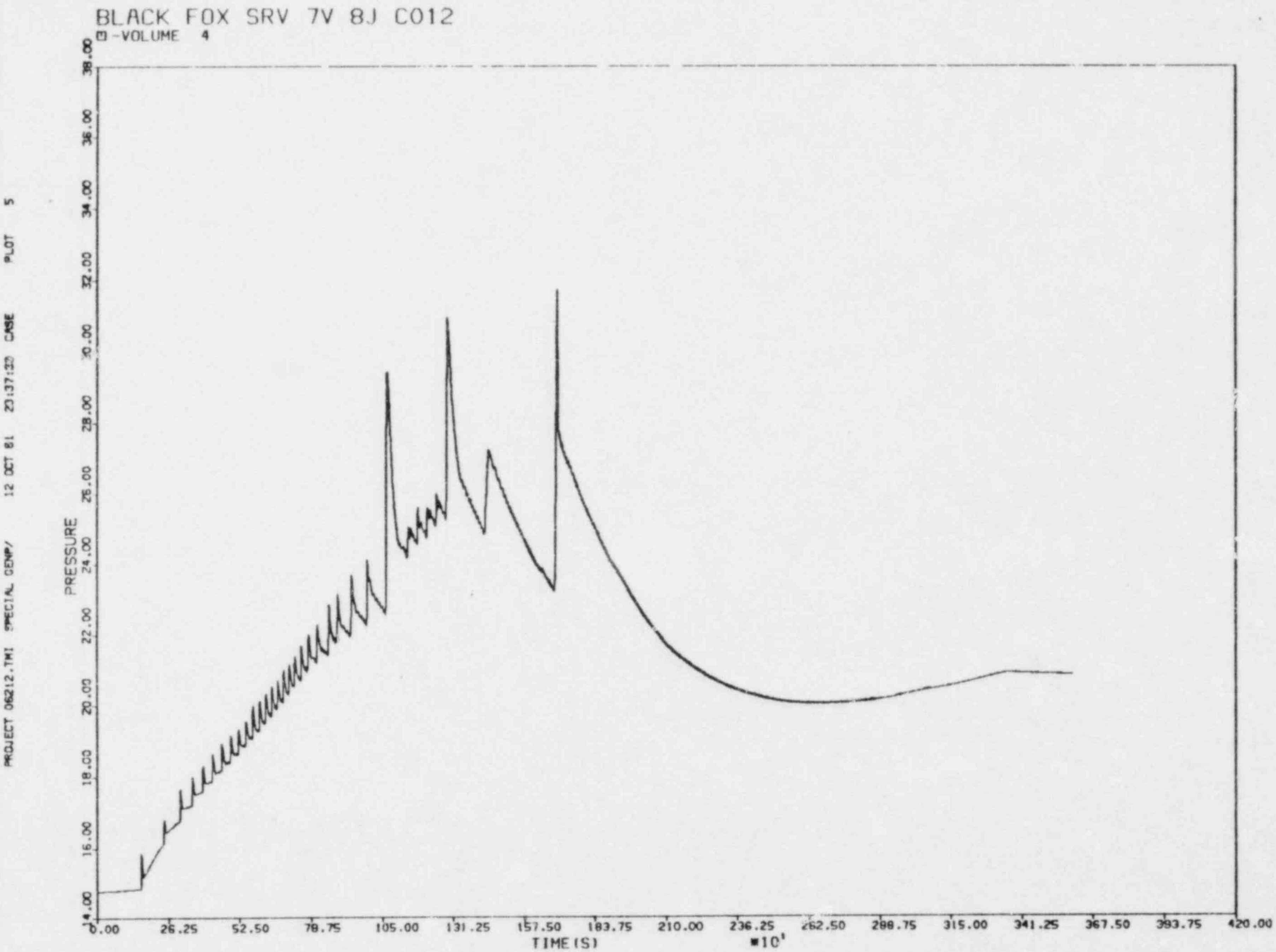
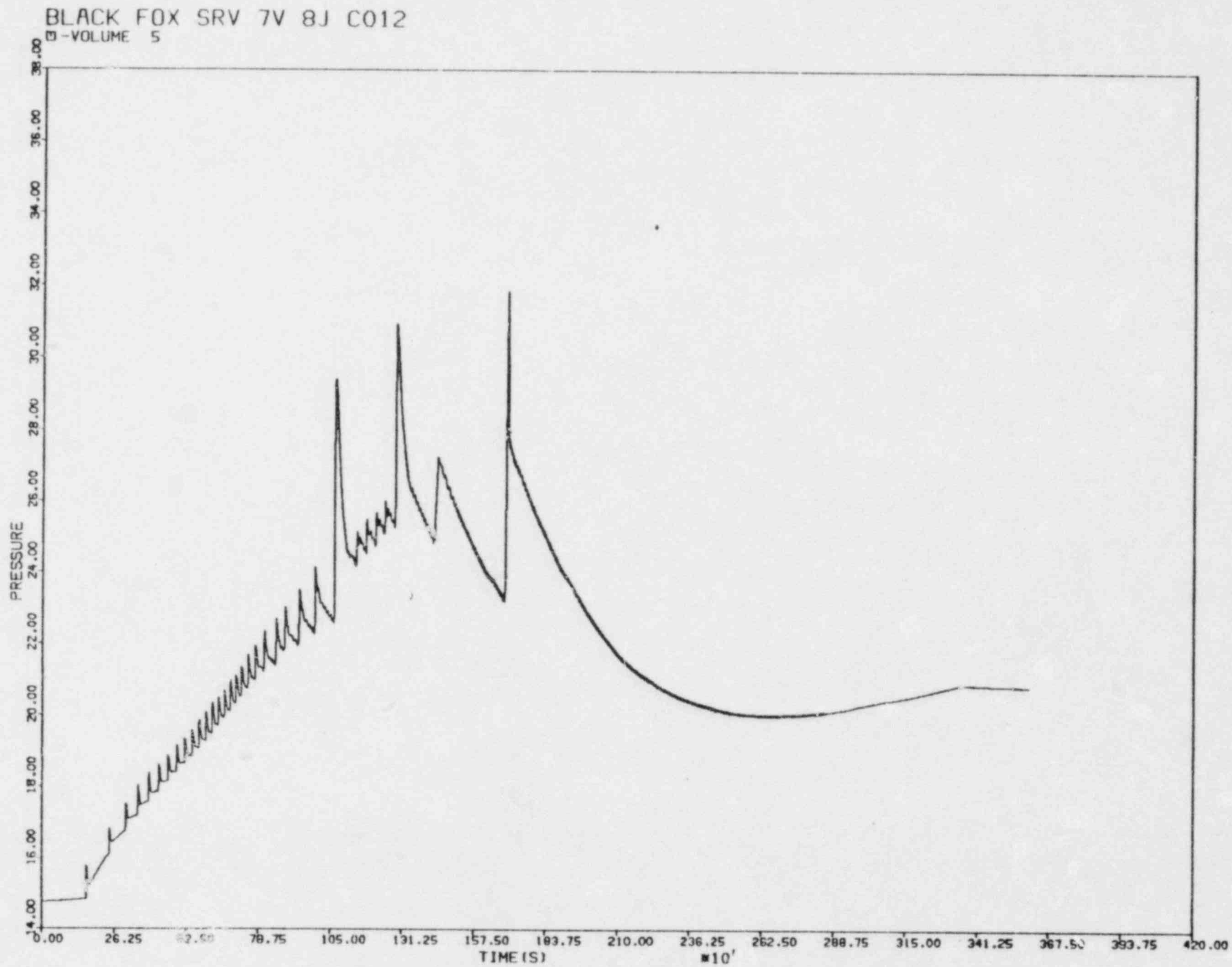


FIGURE (2)(1x)-56



PROJECT 06212.TMI SPECIAL GENP/ 12 OCT 81 23:37:33 CASE PLOT 3

USER: 03636SCHADE PLOT FILE: 03636SCHADE*PL#2337733A.

FIGURE (2)(ix)-57

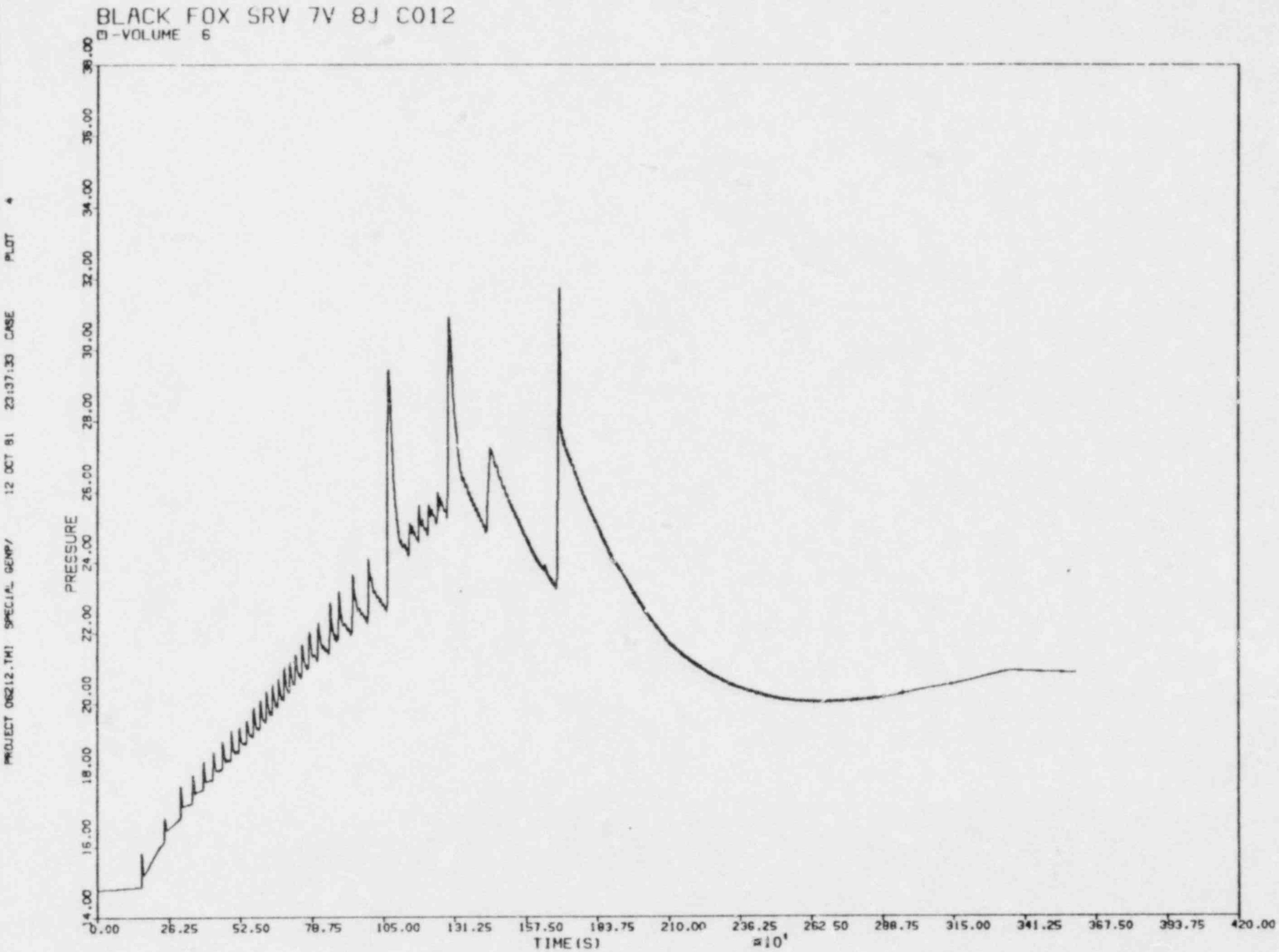
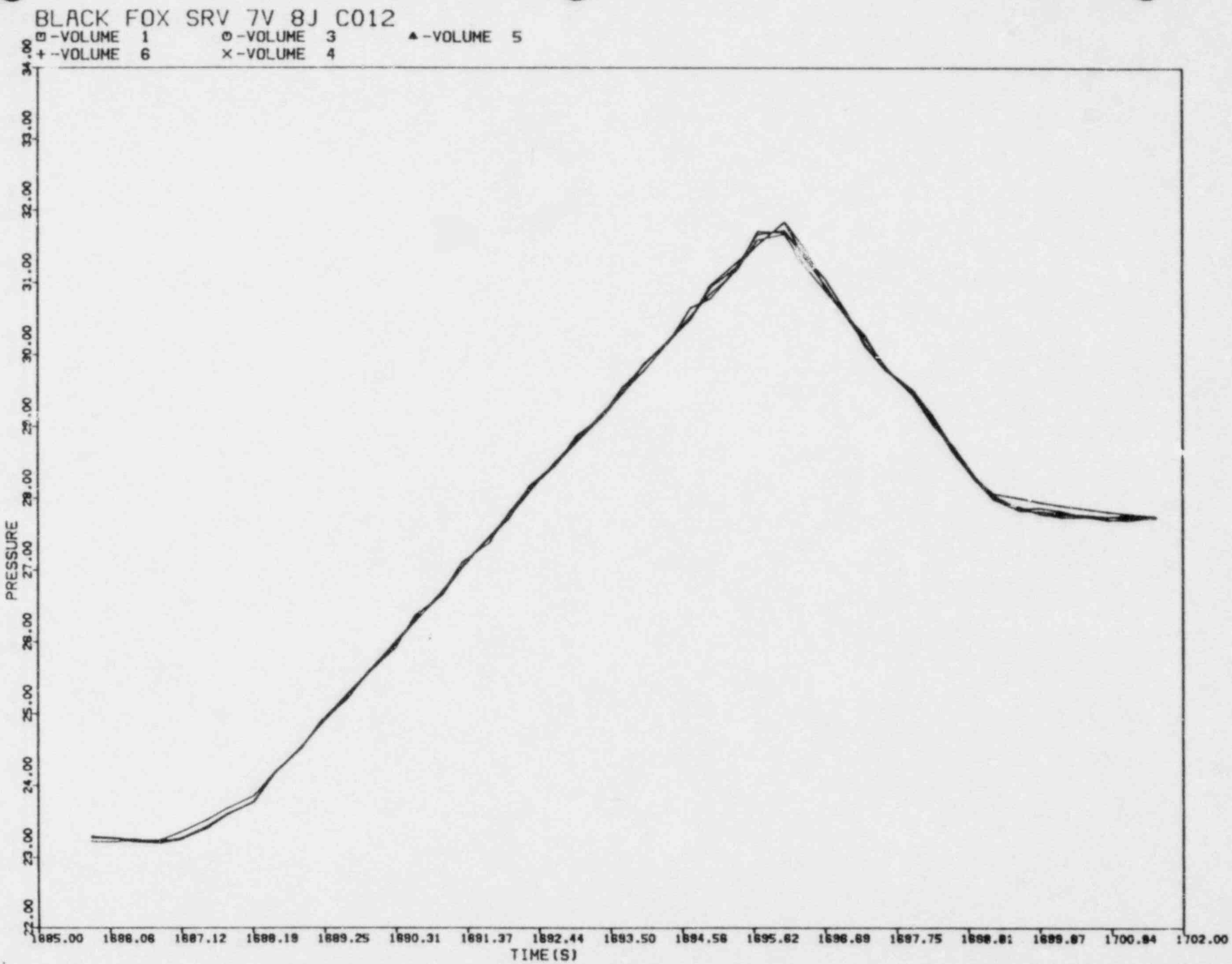


FIGURE (2)(1x)-58

USER: 03G3SSCHADE PLOT FILE: 03G3SSCHADE*PL#233733A.

PROJECT 06212.TMI SPECIAL 0000/ 15 OCT 81 13:32:17 CASE PLOT 1



(2)(4x)-59

19-111381

USER: 036JSSCHADE PLOT FILE: 036JSSCHADE#PL0133217A.

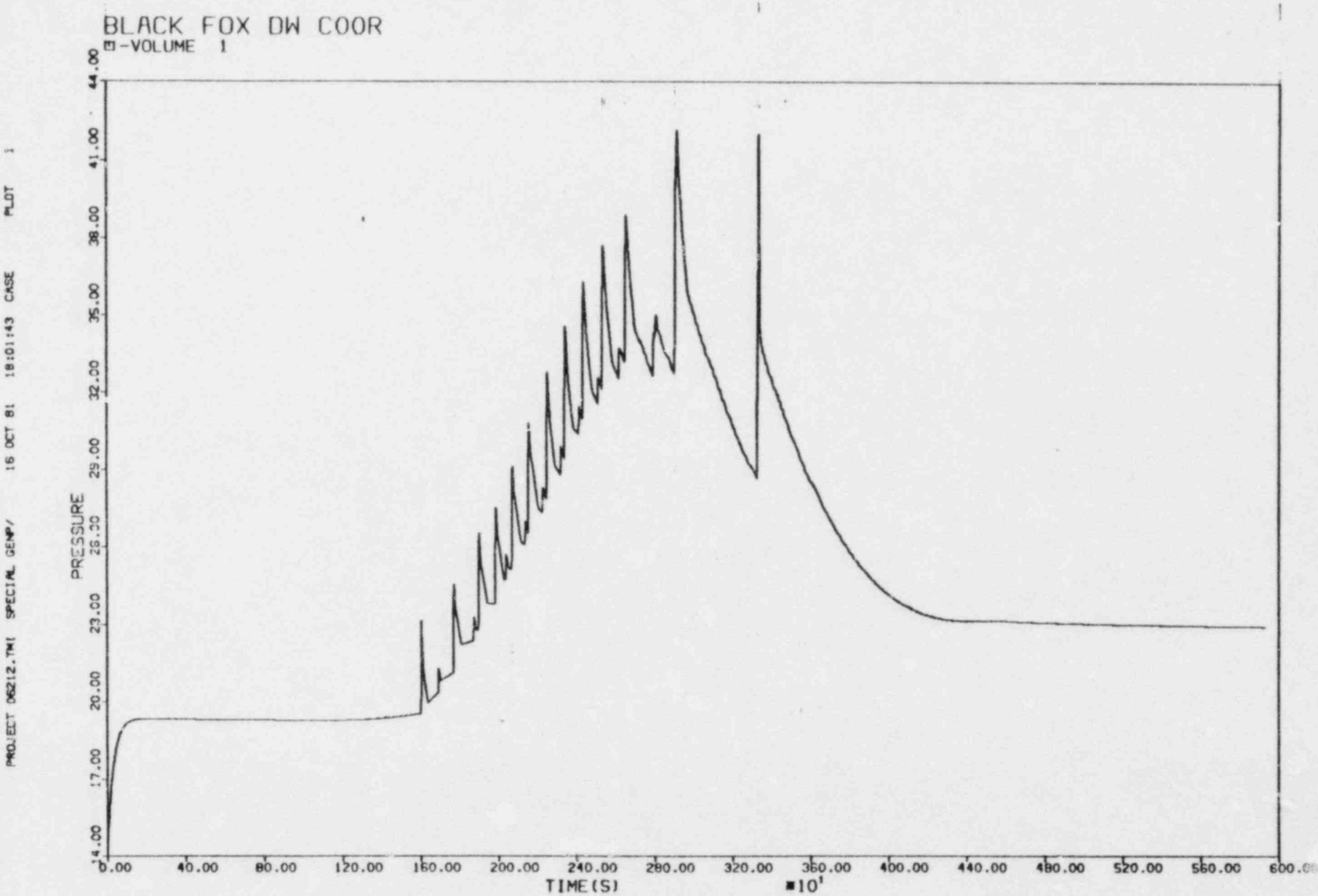


Figure (2)(ix)-60

19-111381

USER: 03636SCHADE PLOT FILE: 03636SCHADE*PL*180143A.

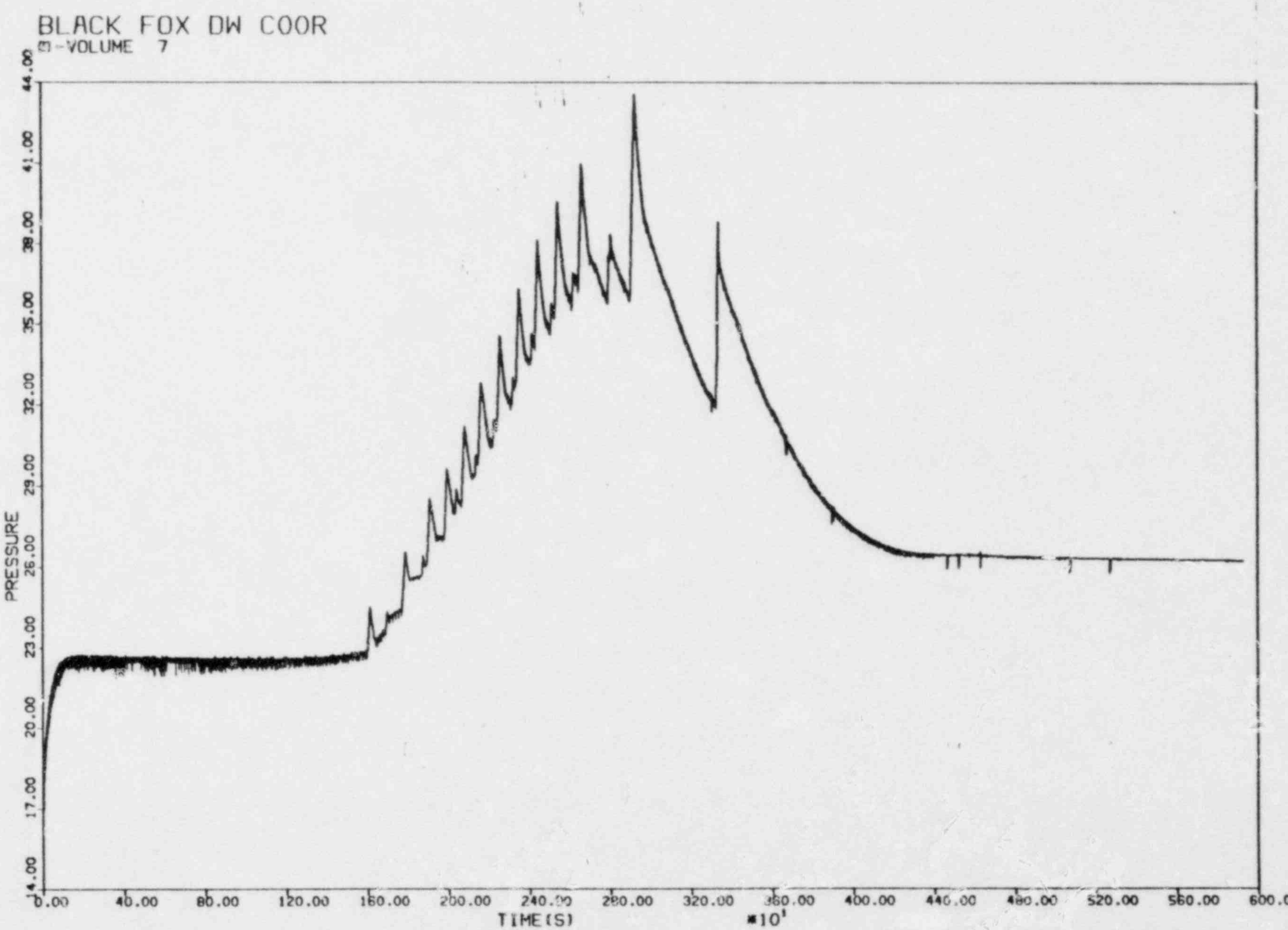


Figure (2)(1x)-61

PROJECT 06212.TMI SPECIAL GENP/ 16 OCT 81 18:01:43 CASE PLOT 8

USER: 03636SCHADE PLOT FILE: 03636SCHADE*PL8180143A.

19-111381

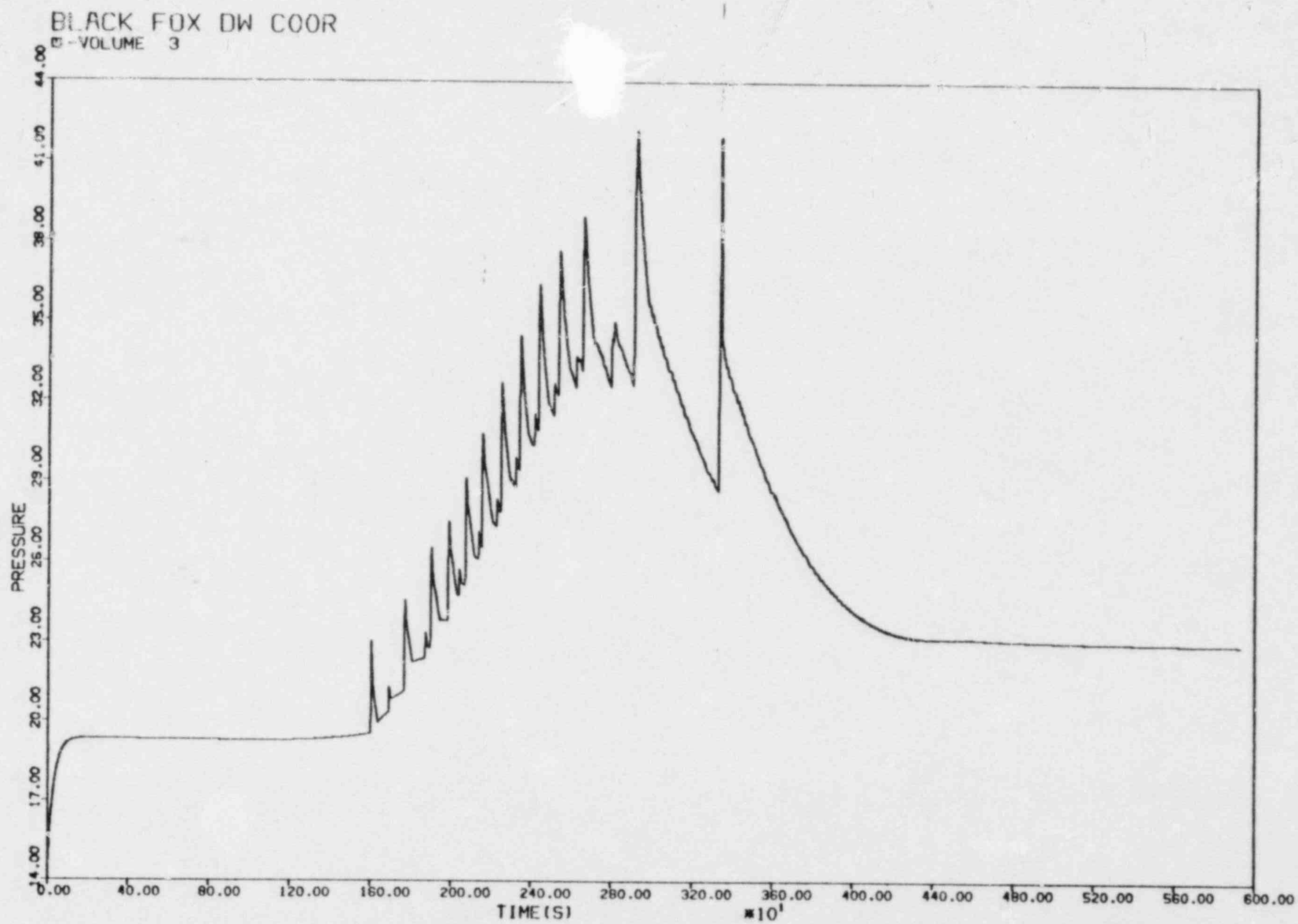


Figure (2)(1x)-62

PROJECT 06212.TMI SPECIAL GENP/ 16 OCT 81 18:01:43 CASE PLOT 2

USER: 03636SCHADE PLOT FILE: 03636SCHADE*PL*18C143A.

19-111381

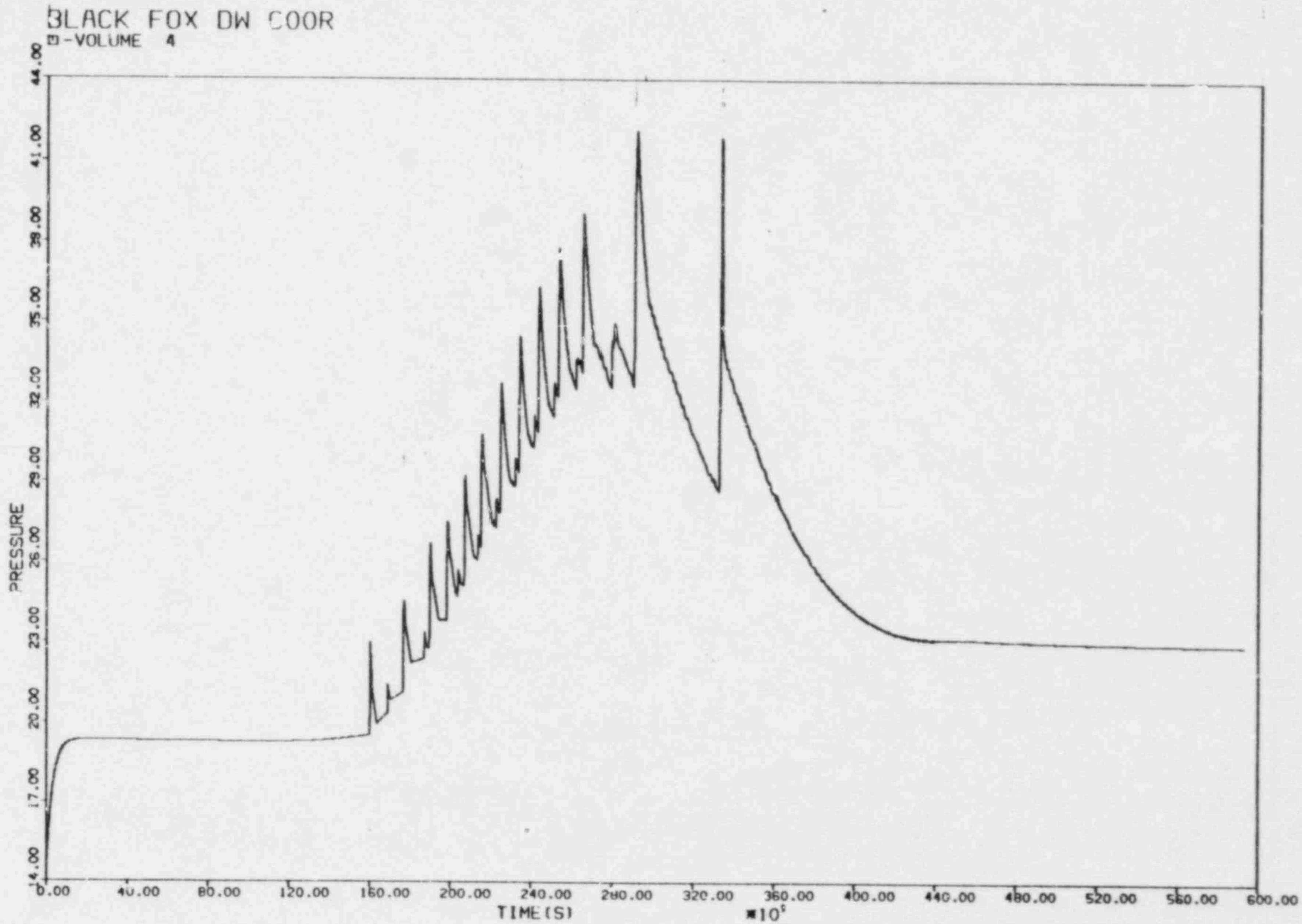


Figure (2)(1x)-63

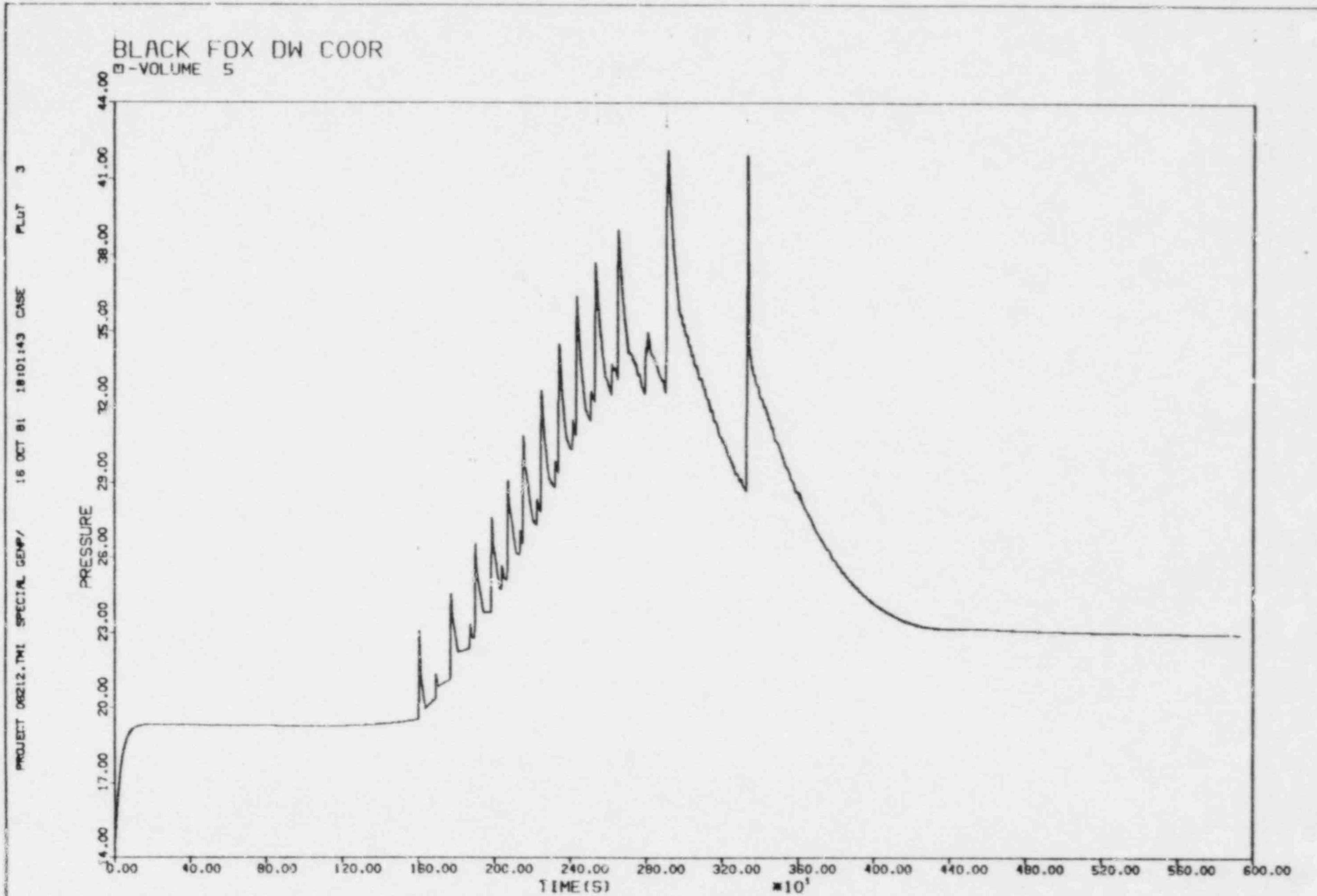
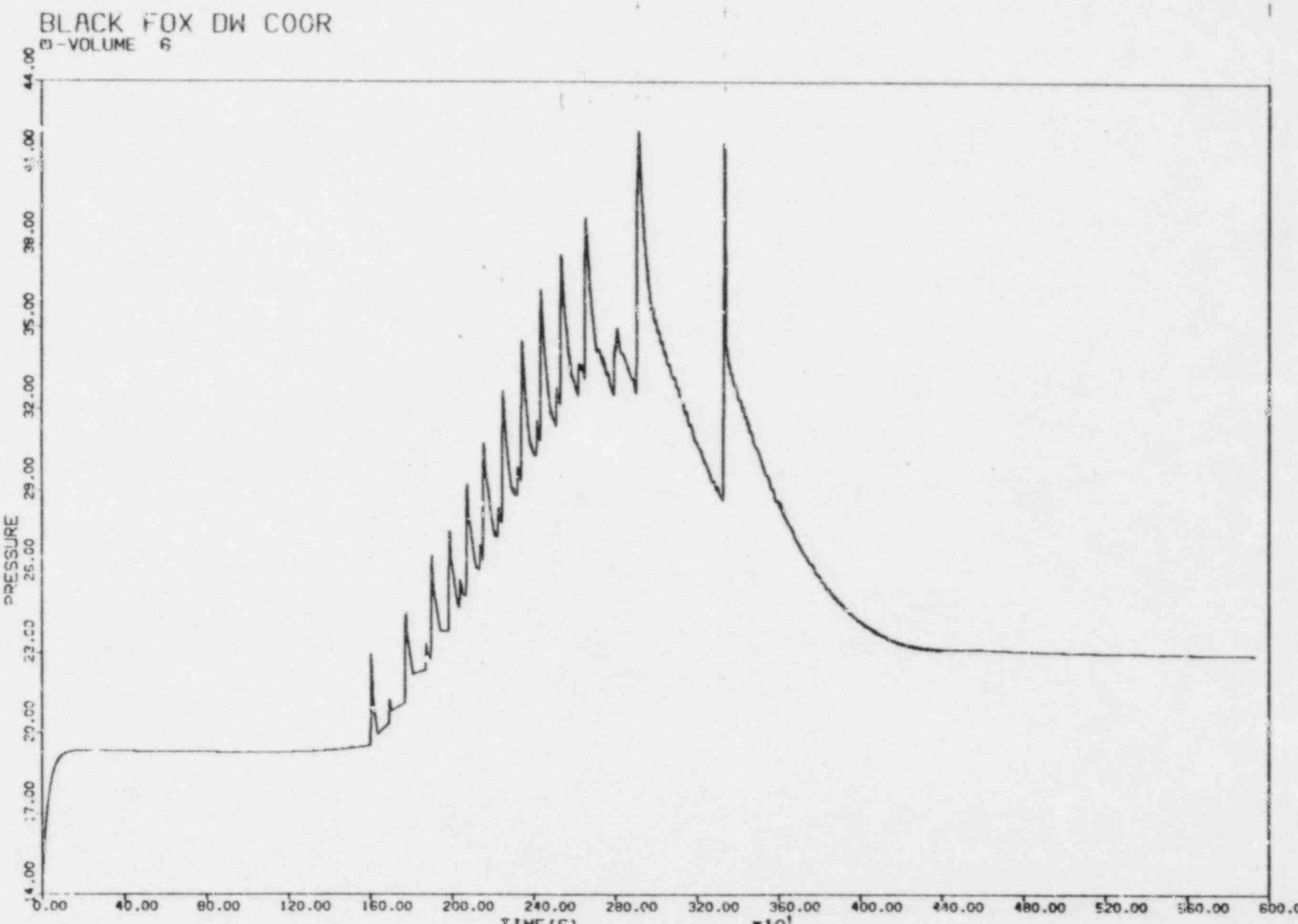


Figure (2)(1x)-64



PROJECT 06212.TMI SPECIAL GEMP/ 16 OCT 81 18:01:43 CASE PLOT 4

Figure (2)(1x)-65

19-111381

USER: 03636SCHADE PLOT FILE: 03636SCHADE*PL#180143A.

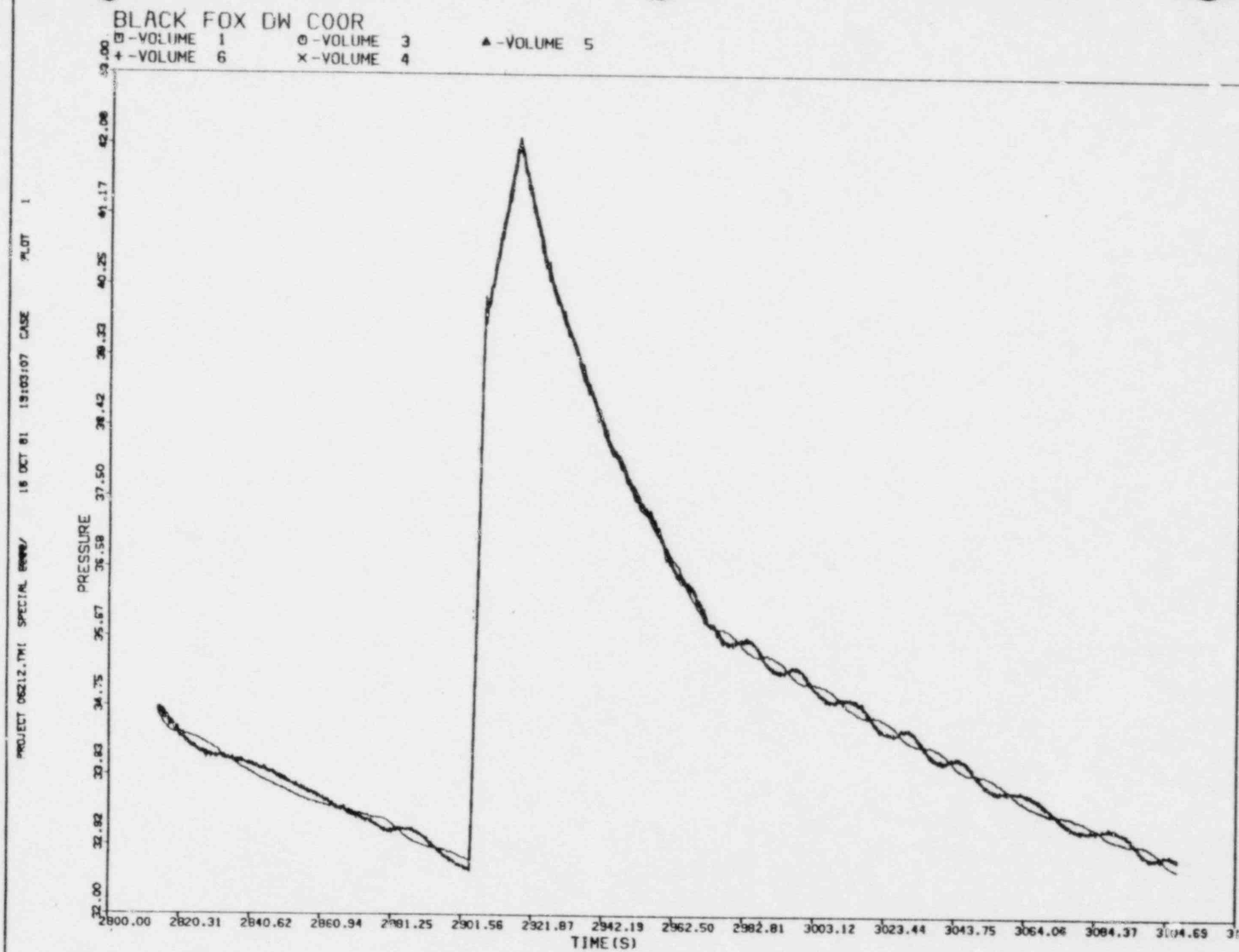


Figure (2)(1x)-66

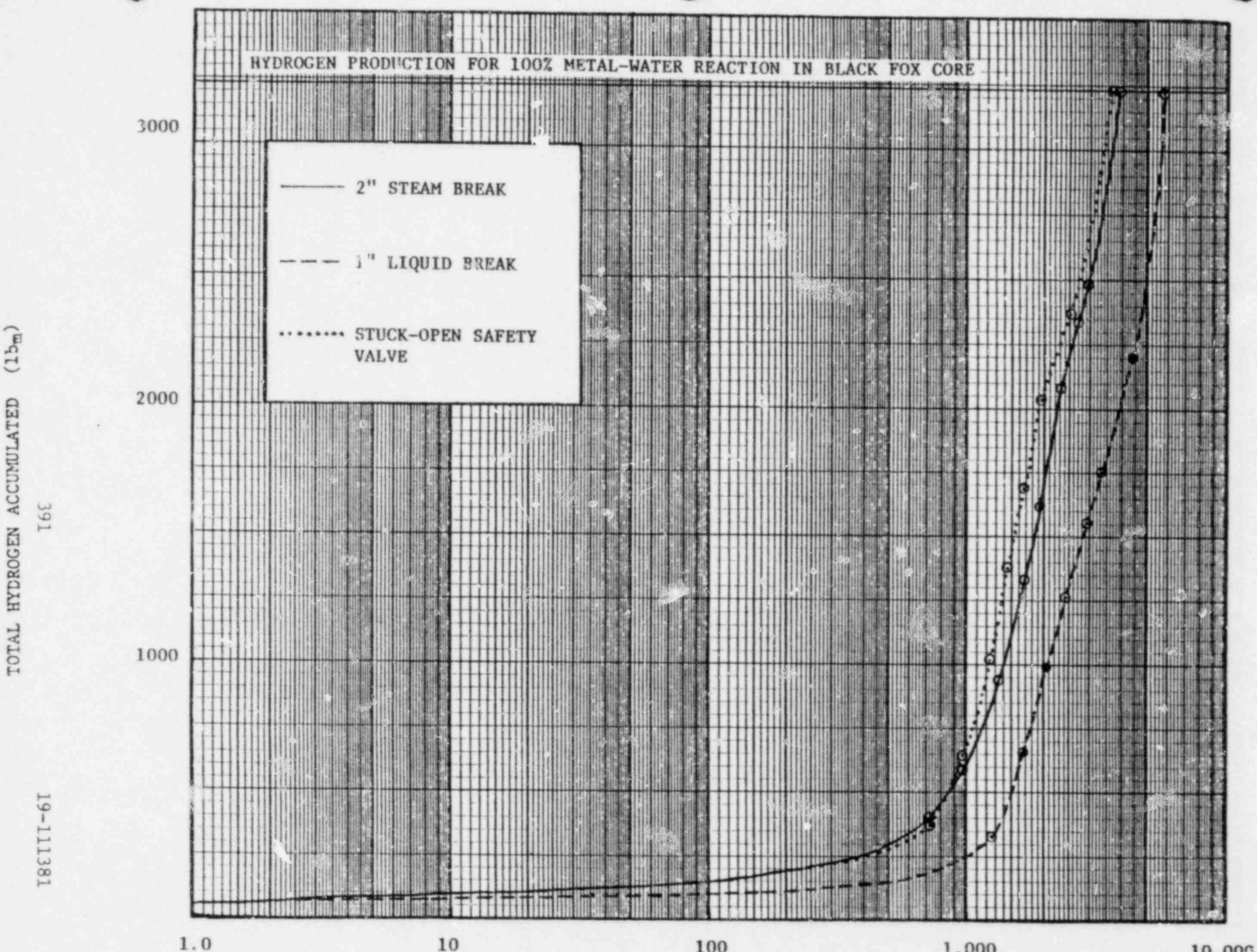


Figure (2)(ix)-67

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19-111381

BLACK FOX HYBRID NODAL DIAGRAM

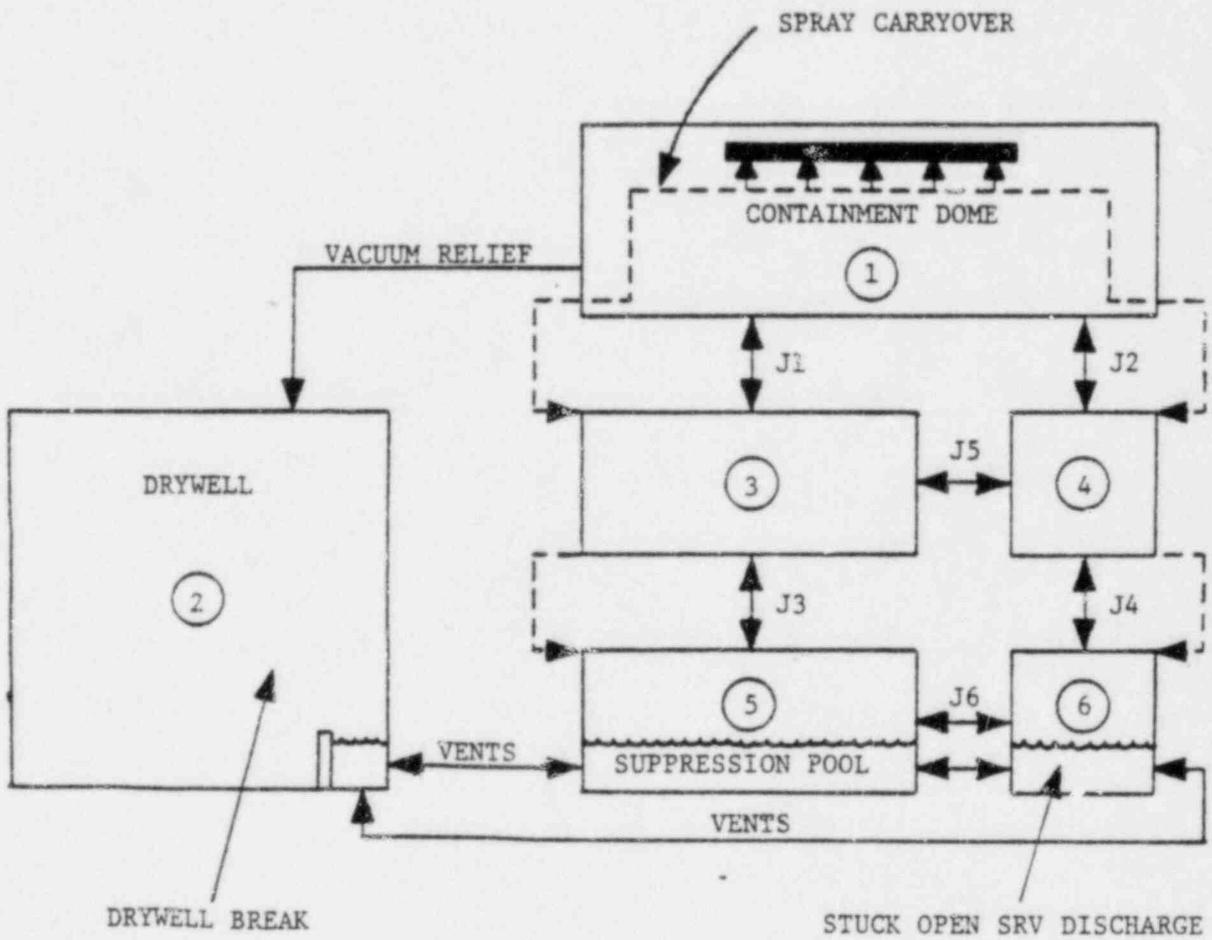

Figure (2)(ix)-68