UNITED STATES NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

14-4(20)

Docket No. 50-298

August 5, 1975

Nebraska Public Power District
ATTN: Mr. J. M. Pilant, Manager
Licensing and Quality Assurance
Post Office Box 499
Columbus, Nebraska 68601

Re: Cooper Nuclear Station

Gentlemen:

10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," was published February 14, 1973. Since many nuclear plants had either received an operating license or their containments had reached advanced stages of design or construction at that time, some plants may not now be in full compliance with the requirements of this regulation.

You are requested to determine if you are conducting containment leakage testing in full compliance with Appendix J. This determination should include the identification of any design features that do not permit conformance with its requirements or existing technical specification requirements which are in conflict with Appendix J, (i.e. less restrictive than). It should be understood that while a containment leakage testing program may be in compliance with the technical specifications for your facility, the program may not be in conformance with Appendix J.

If you are not in full compliance, you should identify your planned actions and schedule to attain conformance to the Regulation. Possible courses of action include design modifications, amendments to the technical specifications, and requests for exemption pursuant to 10 CFR Part 50, Section 50.12.

Please submit the results of your study to us as soon as possible but no later than 30 days from receipt of this letter.

This request for generic information was approved by GAO under a blanket clearance number B-180225 (R0072); this clearance expires July 31, 1977.

Sincerely,

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Reactor Licensing

N/1 4/20

Enclosure:
Appendix J

An

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9512130209 951206
PDR FOIA
PATTERS95-262 PDR

8/11/75

cc w/enclosure:
Gene Watson, Attorney
Barlow, Watson & Johnson
P. O. Box #81686
Lincoln, Nebraska 68501

Mr. Arthur C. Gehr, Attorney
Snell & Wilmer
400 Security Building
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UNITED STATES [illegible] 0374
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1750 U3/2

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[signature]

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for Operating Reactors
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N/ /
4/20

Enclosure:
Appendix J

AH

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1750 6 0375

We conclude that design of the primary containment system will permit the conduct of a containment leakage testing program in compliance with the requirements set forth in proposed Appendix J to 10 CFR Part 50, "Reactor Containment Leakage Testing for Water Cooled Power Reactors" (36 Fed. Reg. 17053, Aug. 27, 1971).

6.2.4 Atmosphere Control

As an operational technique to preclude flammable gas concentrations, the primary containment will be operated with an inert nitroge atmosphere. The system will maintain the oxygen content of the containment atmosphere below 4 volume percent and we find it acceptab

Following a loss-of-coolant accident (LOCA), (a) hydrogen gas could be generated inside the primary containment from a chemical reaction between the fuel rod cladding and steam (metal-water reaction), and (b) both hydrogen and oxygen would be generated as a result of radiolytic decomposition of recirculating water. If a sufficient amount of the hydrogen is generated and oxygen is available in stoichiometric quantities, the subsequent reaction of hydrogen with oxygen can occur at rates rapid enough to lead to a significant pressure increase in the containment. This could cause damage to the containment and could lead to failure of the containmen to maintain low leakage integrity.

General Design Criterion 41 of Appendix A to 10 CFR Part 50 requires that systems to control hydrogen, oxygen and other substance

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COOPER NUCLEAR STATION
BACKUP DATA

ENFORCEMENT SUMMARY

SALP CYCLE 012

APRIL 25, 1993
THROUGH
OCTOBER 22, 1994

Updated

September 12, 1994

~~PREDECISIONAL INFORMATION~~
~~DO NOT RELEASE~~

N/1
ATT 4(4)

I. PLANT OPERATIONS

B. ENFORCEMENT AND REGULATORY ISSUES

1. Escalated Enforcement

2. Normal Enforcement

9319-01	IV	The licensee loaded two fuel bundles without having all control rods inserted
9326-02	IV	The shift supervisors in charge of the shifts on November 18 and 19, 1993, failed to adequately review the work documents involved in the repair and testing of Fire Door H-305 and did not identify that this work resulted in a breach of the control room isolation envelope.
9328-03	IV	On March 27 (EDG 2) and April 9 (EDG 1), the licensee failed to follow MP 7.3.1, in that, the manufacturer's recommended measurements of the wipe were not performed.
9328-05	IV	Untimely declaration of an unusual event and inadequate corrective action (Section 4).
93202-03	IV	Several examples were noted of operators improperly using procedures.
93-202-04	IV	Between October 14 and 21, 1993, with the plant in the Run mode, five shift technical advisors stood watch even though their training had expired.
93202-07	IV	Changes to the design and configuration of piping and equipment insulation were routinely made without the use of the design change process, so that reviews were not performed in a manner commensurate to those applied to the original insulation design.
93202-08	IV	Procedure 2.0.7 was determined to be inadequate for the control of temporary modifications. The procedure failed to provide measures to ensure that the necessary reviews associated with installed temporary modifications.

9415-03	IV	Authorized licensed power level exceeded due to error in the calibration of the pressure transmitters used for feedwater flowrate determination

MAINTENANCE/SURVEILLANCE

B. ENFORCEMENT AND REGULATORY ISSUES

1. Escalated Enforcement

2. Normal Enforcement

93-20	IV	Emergency Plan Implementing Procedure 5.7.16, Revision 16, implemented on May 7, 1993, was inappropriate to the circumstances because it contained erroneous instructions for determining the noble gas release rates using the primary containment monitor and the drywell curie content in Steps 8.3.5, 8.4.5, and Attachments 3 and 4. The licensee had created, then failed to identify, the error during the procedure revision process.
93-22	IV	July 3, 1993, a through-wall leak was discovered in a service water sample return piping, a significant condition adverse to quality, in which a previous leak affecting the same service water system sample return line had been identified on January 13, 1993. Measures had not been taken following the first event to assure that the cause for the condition was determined and corrective actions taken to preclude repetition.
9328-01	IV	EDG 2 became inoperable sometime between April and November 8, 1993, although, because of the noted procedure inadequacies, the inspectors could not eliminate that relay DG-REL-DG1(59) was also misadjusted during the refueling outage. This is an apparent violation of TS 3.9.A and 3.5.F
9329-01	IV	Six lube oil samples taken in the period of December 1992 through November 1993 from the safety-related Reactor Core Isolation Cooling system were not sent out for analysis of wear

products and, consequently, no system engineer review and trending was performed.

93202-06 IV On November 2 and November 13, fire doors R1 and R3, respectively, were found inoperable. Further inspection revealed that a total of twenty (20) fire doors were declared inoperable.

9415-02 IV The licensee used an inadequate special instruction to perform check valve maintenance in that no specific instructions were given for the use of a feeler gauge, which is required for the installation of a flexitallic gasket to verify proper gasket crush.

ENGINEERING/TECHNICAL SUPPORT

B. ENFORCEMENT AND REGULATORY ISSUES

1. Escalated Enforcement

94-14 Proposed Inoperable primary containment due to failure to LLRT penetrations

2. Normal Enforcement

93-25-01 IV A written procedure was not established and maintained for an alarm indicating a malfunction of a safety system in that Alarm Procedure 2.3.2.22, for Panel/Window Location 9-3-2/C-2, Section 2.0, did not appropriately specify the operator actions to be taken when the alarm was received during HPCI system surveillance testing. When the HPCI system is in operation, the condensate drain and steam trap are isolated and cannot be drained or verified to be operating correctly.

93-25-02 IV Measures did not assure that a deficiency was promptly corrected in that the licensee became aware that Alarm Procedure 2.3.2.22, Revision 16, "HPCI Turbine Inlet Drain Pot HI Level," was deficient after surveillance testing on September 1, 1993, and did not promptly

		correct the deficient procedure. On September 29, 1993, during surveillance testing, the deficient procedure caused the HPCI turbine to be unnecessarily secured.
9328-02	IV	Inadequate procedure resulted in relay misadjustment which represented a common mode failure mechanism for both EDGs.(2 examples) inadequate procedure.
9328-04	IV	The licensee did not effectively identify or address the relay out-of-tolerance conditions identified in March and April 1993, and the corrective actions taken did not preclude repetition.
9328-01	IV	New manually operated primary containment isolation valves, PC-V-506 and -507 were installed during the last refueling outage (March 1993 - July 1993), but the normal position of the manual primary containment isolation valves, an activity affecting quality, was not prescribed by any procedure
9406-01	IV	Conduct of Operations Procedure 2.0.7, "Plant Temporary Modifications Control", Revision 17, dated July 22, 1993, did not require that the Station Operations Review Committee review all proposed plant temporary modifications to station systems or equipment as described in the Updated Safety Analysis Report.
93202-01	IV	Identified several examples of procedures not appropriate for their intended purpose.
93202-02	IV	On January 30 and 31, 1993, with both suppression chamber/torus water level instruments (PC-LI-12 and PC-LI-13) inoperable, an orderly shutdown was not commenced after 6 hours and the reactor was not placed in hot shutdown within the following 6 hours. Instruments PC-LI-12 and PC-LI-13 were rendered inoperable on January 30, 1993, during the performance of Maintenance Work Request 92-0185 and were not declared inoperable until the following day at 1:19 p.m.
93202-03	IV	Several examples were noted of operators improperly using procedures.
93202-05	IV	In some cases the licensee failed to maintain configuration control.

PLANT SUPPORT

B. ENFORCEMENT AND REGULATORY ISSUES

1. Escalated Enforcement

2. Normal Enforcement

93-21	IV	The licensee failed to inform individuals that they had to report all arrests that could affect their trustworthiness
93-21	NCV	Unescorted access to an unauthorized contractor employee
9415-01	IV	On June 9, 1994, at the point of personnel access into a protected area, the licensee did not search a hand-carried package for devices such as firearms, explosives, and incendiary devices, or other items which could be used for radiological sabotage in that a security guard entered the protected area with a hand-carried package that had not been searched.
9420	IV	On July 14, 1994, the inspector identified one individual with a valid keycard that had not used the keycard since December 30, 1993. The individual had not been under the licensee's continuous behavioral observation program the entire period and, with a valid keycard on file, the individual could have entered the protected area at anytime of his choosing. In addition, the licensee informed the inspector on July 20, 1994, that at least 28 additional individuals were identified who also had not been under the continuous behavioral observation program for 31 days or more.

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As requested, I have gathered the following information for the DFIs being sent in regard to 4-93-020R.

NAME	CURRENT POSITION
Ricky L. Gardner	Maintenance Manager
Charles M. Estes	Retired
Eugene M. Mace	Senior Manager of Site Support
R. Brungardt	Staff, Operations Support Group
Michael F. Young	~~Planner~~ Maintenance Planner
J. V. Sayer	Radiological Manager
James R. Flaherty	Corrective Action Program Manager
C. R. Moeller	Technical Staff Manager
Paul L. Ballinger	Staff, Engineering
H. A. Jantzen	Instrumentation & Control Supervisor
G. E. Smith	Quality Assurance Operations Manager
John M. Meacham	Assistant to the VP - Nuclear

I got this information from Terry Reis. If I get anything on Meacham tomorrow, I will forward it to you. If you need anything else, let me know.

Virginia Van Cleave

M/1

14-4(5)

3.12.A (cont'd)

A. Control Room Emergency Filter System

1. Except as specified in Specification 3.12.A.3 below, the Control Room Emergency Filter system, the diesel generators required for operation of this system and the main control room air radiation monitor shall be operable at all times when containment integrity is required.

2.a. The results of the in-place cold DOP leak tests on the HEPA filters shall show ≥ 99% DOP removal. The results of the halogenated hydrocarbon leak tests on the charcoal adsorbers shall show ≥99% halogenated hydrocarbon removal. The DOP and halogenated hydrocarbon tests shall be performed at a flowrate of ≤ 341 CFM.

b. The results of laboratory carbon sample analysis shall show ≥99% radioactive methyl iodide removal with inlet conditions of: velocity ≥22 FPM, ≥1.75 mg/m^3 inlet iodide concentration, ≥ 95% R.H. and ≤30°C.

c. The emergency bypass fan shall be shown to provide 341 CFM ±10%.

3. From and after the date that the Control Room Emergency Filter system is made or found to be inoperable for any reason, reactor operations are permissible only during the succeeding seven days unless the system is sooner made operable. Refueling requirements are as specified in Specification 3.10.G.

4. If these conditions cannot be met, reactor shutdown shall be initiated and the reactor shall be in cold shutdown within 24 hours.

4.12.A (cont'd)

A. Control Room Emergency Filter System

1. At least once per operating cycle, the pressure drop across the combined HEPA filters and charcoal absorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate.

2.a. The tests and sample analysis of Specification 3.12.A.2 shall be performed at least once every 18 months for standby service or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal absorber bank or after any structural maintenance on the system housing.

d. The system shall be operated at least 10 hours every month.

At least once per operating cycle automatic initiation of the system shall be demonstrated.

N 11

ATT 4(5)

During normal operation two or three pumps will be required. Three pumps are used for a normal shutdown. The loss of all a-c power will trip all operating Service Water pumps. The automatic emergency diesel generator start system and emergency equipment starting sequence will then start one selected Service Water pump per division in 30-40 seconds. In the meantime, the drop in Service Water header pressure will isolate the non-critical services, ensuring adequate supply to the critical heat loads as described above.

Due to the redundance of pumps and the requirement of only one to meet the accident requirements, the 30 day repair time is justified.

D. Battery Room Ventilation

The temperature rise and hydrogen buildup in the battery rooms without adequate ventilation is such that continuous safe operation of equipment in these rooms cannot be assured.

4.12 BASES

A. Control Room Emergency Filter System

Pressure drop across the combined HEPA filters and charcoal absorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and absorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to show system performance capability.

Tests of the charcoal absorbers with halogenated hydrocarbon refrigerant should be performed in accordance with ANSI N510-1980.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal absorbers can perform as evaluated. The test canisters that are installed with the adsorber trays should be used for the charcoal adsorber efficiency test. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent qualified according to Table 5.1 of ANSI N509-1980. The replacement tray for the absorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1980. Any HEPA filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52.

Operation of the system for 10 hours every month will demonstrate operability of the filters and adsorber system and remove excessive moisture built up on the adsorber

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability is necessary to assure system performance capability.

B. Reactor Equipment Cooling System

Normal plant operation requires one heat exchanger and three pumps. Therefore, normal equipment rotation will demonstrate pump operability.

Pump rates will be demonstrated every three months as an indication of the pump condition.

C. Service Water System

The Service Water pumps shall be proven operable by their use during normal station operations. Since three pumps are continuously operating during normal operation and only one pump is required during accidents, the normal equipment rotation shall prove the pump operability.

Pump discharge head tests will be run every three months to verify the pumping ability.

Any silting problems caused by the Service Water system will be analyzed during and following the Preoperational Test Program. Any required changes in operating procedures, technical specifications or surveillance requirements will be made prior to CNS commercial operation.

D. Battery Room Ventilation

The ventilation fans will be rotated on a weekly basis to demonstrate operability.

III.D.3.4 CONTROL-ROOM HABITABILITY REQUIREMENTS

Position

In accordance with Task Action Plan item III.D.3.4 and control room habitability, licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50).

Changes to Previous Requirements and Guidance

There are no changes to the previous requirements.

Clarification

(1) All licensees must make a submittal to the NRC regardless of whether or not they met the criteria of the referenced Standard Review Plans (SRP) sections. The new clarification specifies that licensees that meet the criteria of the SRPs should provide the basis for their conclusion that SRP 6.4 requirements are met. Licensees may establish this basis by referencing past submittals to the NRC and/or providing new or additional information to supplement past submittals.

(2) All licensees with control rooms that meet the criteria of the following sections of the Standard Review Plan:

2.2.1-2.2.2	Identification of Potential Hazards in Site Vicinity
2.2.3	Evaluation of Potential Accidents;
6.4	Habitability Systems

shall report their findings regarding the specific SRP sections as explained below. The following documents should be used for guidance:

(a) Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of Regulatory Power Plant Control Room During a Postulated Hazardous Chemical Release";

(b) Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accident Chlorine Release"; and,

(c) K. G. Murphy and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," 13th AEC Air Cleaning Conference, August 1974.

Licensees shall submit the results of their findings as well as the basis for those findings by January 1, 1981. In providing the basis for the habitability finding, licensees may reference their past submittals. Licensees should, however, ensure that these submittals reflect the current facility design and that the information requested in Attachment 1 is provided.

(3) All licensees with control rooms that do not meet the criteria of the above-listed references, Standard Review Plans, Regulatory Guides, and other references.

These licensees shall perform the necessary evaluations and identify appropriate modifications.

Each licensee submittal shall include the results of the analyses of control room concentrations from postulated accidental release of toxic gases and control room operator radiation exposures from airborne radioactive material and direct radiation resulting from design-basis accidents. The toxic gas accident analysis should be performed for all potential hazardous chemical releases occurring either on the site or within 5 miles of the plant-site boundary. Regulatory Guide 1.78 lists the chemicals most commonly encountered in the evaluation of control room habitability but is not all inclusive.

The design-basis-accident (DBA) radiation source term should be for the loss-of-coolant accident LOCA containment leakage and engineered safety feature (ESF) leakage contribution outside containment as described in Appendix A and B of Standard Review Plan Chapter 15.6.5. In addition, boiling-water reactor (BWR) facility evaluations should add any leakage from the main steam isolation valves (MSIV) (i. e., valve-stem leakage, valve seat leakage, main steam isolation valve leakage control system release) to the containment leakage and ESF leakage following a LOCA. This should not be construed as altering the staff recommendations in Section D of Regulatory Guide 1.96 (Rev. 2) regarding MSIV leakage-control systems. Other DBAs should be reviewed to determine whether they might constitute a more-severe control-room hazard than the LOCA.

In addition to the accident-analysis results, which should either identify the possible need for control-room modifications or provide assurance that the habitability systems will operate under all postulated conditions to permit the control-room operators to remain in the control room to take appropriate actions required by General Design Criterion 19, the licensee should submit sufficient information needed for an independent evaluation of the adequacy of the habitability systems. Attachment 1 lists the information that should be provided along with the licensee's evaluation.

Applicability

This requirement applies to all operating reactors and operating license applicants.

Implementation

Licensees shall submit their responses to this request on or before January 1, 1981. Applicants for operating licenses shall submit their responses prior to issuance of a full-power license. Modifications needed for compliance with the control-room habitability requirements specified in this letter should be identified, and a schedule for completion of the modifications should be provided. Implementation of such modifications should be started without awaiting the results of the staff review. Additional needed modifications, if any, identified by the staff during its review will be specified to licensees

Type of Review

A postimplementation review will be performed.

Documentation Required

By January 1, 1981 licensees shall provide the information described in Attachment 1. Applicants for an operating license shall submit their responses prior to full-power licensing.

Technical Specification Changes Required

Changes to technical specifications will be required.

References

NUREG-0660, Item III.D.3.4.

Letter from D. G. Eisenhut, NRC, to All Operating Reactor Licensees, dated May 7, 1980.

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

NUREG-0800
(Formerly NUREG-75/087)

6.4 CONTROL ROOM HABITABILITY SYSTEM

REVIEW RESPONSIBILITIES

Primary - Accident Evaluation Branch (AEB)

Secondary - Effluent Treatment Systems Branch (ETSB)
Siting Analysis Branch (SAB)

I. AREAS OF REVIEW

The control room ventilation system and control building layout and structures, as described in the applicant's safety analysis report (SAR), are reviewed with the objective of assuring that plant operators are adequately protected against the effects of accidental releases of toxic and radioactive gases. A further objective is to assure that the control room can be maintained as the backup center from which technical support center personnel can safely operate in the case of an accident. To assure that these objectives are accomplished the following items are reviewed:

1. The zone serviced by the control room emergency ventilation system is examined to ascertain that all critical areas requiring access in the event of an accident are included within the zone (control room, kitchen, sanitary facilities, etc.) and to assure that those areas not requiring access are generally excluded from the zone.

2. The capacity of the control room in terms of the number of people it can accommodate for an extended period of time is reviewed to confirm the adequacy of self-contained breathing apparatus and to determine the length of time the control room can be isolated before CO(2) levels become excessive.

3. The control room ventilation system layout and functional design is reviewed to determine flow rates and filter efficiencies for input into the analyses of the buildup of radioactive or toxic gases inside the control room, assuming a design basis release. Basic deficiencies that might impair the

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Standard Review Plan - 6.4 - CONTROL ROOM HABITABILITY SYSTEM

U.S. NUCLEAR REGULATORY COMMISSION
STANDARD REVIEW PLAN
OFFICE OF NUCLEAR REACTOR REGULATION

6.4 CONTROL ROOM HABITABILITY SYSTEM

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2. The capacity of the control room in terms of the number of people it can accommodate for an extended period of time is reviewed to confirm the adequacy of self-contained breathing apparatus and to determine the length of time the control room can be isolated before CO_2 levels become excessive.

3. The control room ventilation system layout and functional design is reviewed to determine flow rates and filter efficiencies for input into the analyses of the buildup of radioactive or toxic gases inside the control room, assuming a design basis release. Basic deficiencies that might impair the

effectiveness of the system are examined. In addition, the system operation and procedures are reviewed.

4. The physical location of the control room with respect to potential release points of hazardous airborne materials is reviewed. The layout of the control building is reviewed to assure that airborne materials will not enter the

control room from corridors or ventilation ducts, etc.

5. Radiation shielding provided by structural concrete is analyzed to determine the effectiveness of shielding and structure surrounding the control room. The control building layouts are checked to see if radiation streaming through doors or other apertures or from equipment might be a problem.

6. Independent analyses are performed to determine the radiation doses and toxic gas concentrations. Estimates of dispersion of airborne contamination are made in conjunction with the assigned meteorologist.

A secondary review is performed by the Effluent Treatment Systems Branch (ETSB) and the Siting Analysis Branch (SAB) and the results are used by AEB in its overall evaluation of the control room habitability. ETSB reviews the iodine removal efficiencies of the control room atmosphere filtration system. The efficiencies are transmitted to AEB for use in the analysis and are referenced in the SER. The evaluation of the potential hazardous gas sources is performed by the SAB under SRP Section 2.2. The SAB will provide AEB with a description of the sources. In those cases where the identified sources are found to have the potential for incapacitating people in the vicinity of the control room building, the SAB will provide AEB with source location, estimated hazardous gas concentrations near the control room building, and probability for the releases with respect to transportation accidents.

In addition, AEB will coordinate the evaluation with other branches that interface with the review of the control room habitable system as follows: the Auxiliary System Branch (ASB) reviews the design of the control room ventilation system as part of its primary review responsibility for SRP Section 9.4.1. The Radiological Assessment Branch (RAB) reviews radiation shielding and exposures as part of the primary review responsibility for SRP Sections 12.1 through 12.5. The review for technical specifications are coordinated and performed by the Licensing Guidance Branch (LGB) as part of the primary review responsibility for SRP Section 16.0. The acceptance criteria necessary for the review and their application are contained in the above referenced SRP section of the corresponding primary branch.

II. ACCEPTANCE CRITERIA

The control room habitability system design is acceptable if the requirements of the following regulations are met:

a. General Design Criterion 4, "Environmental and Missile Design Bases" (Ref. 1), as it relates to accommodating the effects of and being compatible with postulated accidents, including the effects of the release of toxic gases.

b. General Design Criterion 5, "Sharing of Structures, Systems and Components" (Ref. 1), as it relates to facilities which have a single control room for more than one nuclear power unit and with respect to ensuring that such sharing will not significantly impair the ability to perform safety functions including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining unit(s).

c. General Design Criterion 19, "Control Room" (Ref. 3), as it relates to maintaining the control room in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases.

The specific criteria necessary to meet the relevant requirements of General Design Criteria 4, 5 and 19 and to assure that the control room habitability positions of item III.D.3.4 of NUREG 0737 (Ref. 4) are met are as follows:

1. Control Room Emergency Zone

The control room emergency zone should include the following:

a. instrumentation and controls necessary for a safe shutdown of the plant, i.e., the control room, including the critical document reference file,

b. computer room, if it is used as an integral part of the emergency response plan,

c. shift supervisor's office, and

d. operator wash room and the kitchen.

2. Ventilation System Criteria

The ventilation system is reviewed by ASB under SRP Section 9.4.1, "Control Room Area Ventilation System." The AEB reviewer ascertains from the ASB if the following system performance and availability criteria are met:

a. Isolation dampers - dampers used to isolate the control zone from adjacent zones or the outside should be leaktight. This may be accomplished by using low leakage dampers or valves. The degree of leaktightness should be documented in the SAR.

b. Single failure - a single failure of an active component should not result in loss of the systems functional performance. All the components of the control room emergency filter train should be considered active components. See Appendix A to this SRP for criteria regarding valve or damper repair.

3. Pressurization Systems

Ventilation systems that will pressurize the control room during a radiation emergency should meet the following requirements:

a. Systems having pressurization rates of greater than or equal to 0.5 volume changes per hour should be subject to periodic verification (every 18 months) that the makeup is + 10% of design value. During

plant construction or after any modification to the control room that might significantly affect its capability to maintain a positive pressure, measurements should be taken to verify that the control room is pressurized to at least 1/8-inch water gauge relative to all surrounding air spaces while applying makeup air at the design rate.

b. Systems having pressurization rates of less than 0.5 and equal to or greater than 0.25 volume changes per hour should have identical testing requirements as indicated in (1), above. In addition, at the CP stage an analysis Should be provided (based on the planned leaktight design features) that ensures the feasibility of maintaining 1/8-inch water gauge differential with the design makeup air flow rate.

c. Systems having pressurization rates of less than 0.25 volume changes per hour should meet all the requirements for (2), above, except that periodic verification of control room pressurization (every 18 months) should be specified.

4. Emergency Standby Atmosphere Filtration System

The atmosphere filtration system is reviewed by ETSB under SRP Section 6.5.1. The ETSB will determine the credit for iodine removal for this system in accordance with the guidelines of Regulatory Guide 1.52 (Ref. 5) and will advise the AEB accordingly. Efficiencies for systems not covered by Regulatory Guide 1.52 will be determined on a case-by case basis by ETSB.

5. Relative Location of Source and Control Room

The control room inlets should be located considering the potential release points of radioactive material and toxic gases. Specific criteria as to radiation and toxic gas sources are as follows:

a. Radiation sources - as a general rule the control room ventilation inlets should be separated from the major potential release points by at least 100 feet laterally and by 50 feet vertically. However, the actual minimum distances must be based on the dose analyses (Ref. 6).

b. Toxic gases - the minimum distance between the toxic gas source and the control room is dependent upon the amount and type of the gas in question, the container size, and the available control room protection provisions. The acceptance criteria for the control room habitability system are provided in the regulatory positions of Regulatory Guide 1.78 (Ref. 7) with respect to postulated hazardous chemical releases in general and in Regulatory Guide 1.95

(Ref. 8) with respect to accidental chlorine releases in particular.

6. Radiation Hazards

The dose guidelines for evaluating the emergency zone radiation protection provisions are as follows:

whole body gamma:	5 rem
thyroid:	30 rem
beta skin dose:	30 rem/*/

In accordance with GDC 19 (Ref. 3), these doses to an individual in the control room should not be exceeded for any postulated design basis accident. The whole body gamma dose consists of contributions from airborne radioactivity inside and outside the control room, as well as direct shine from all radiation sources.

7. Toxic Gas Hazards

Three exposure categories are defined: protective action exposure (2 minutes or less), short-term exposure (between 2 minutes and 1 hour), and long-term exposure (1 hour or greater). Because the physiological effects can vary widely from one toxic gas to another, the following general restrictions should be used as guidance: there should be no chronic effects from exposure; acute effects, if any, should be reversible within a short period of time (several minutes) without benefit of any measures other than the use of self-contained breathing apparatus.

The allowable limits should be established on the basis that the operators should be capable of carrying out their duties with a minimum of interference caused by the gas and subsequent protective measures. The limits for the three categories normally are set as follows:

a. Protective action limit (2 minutes or less): use a limit that will assure that the operators will quickly recover after breathing apparatus is in place. In determining this limit, it should be assumed that the concentration increases linearly with time from zero to two minutes and that the limit is attained at two minutes.

b. Short-term limit (2 minutes to 1 hour): use a limit that will assure that the operators will not suffer incapacitating effects after a 1-hour exposure.

c. Long-term limit (1 hour or greater): use a limit assigned for occupational exposure (40-hour week).

/*/Credit for the beta radiation shielding afforded by special protective clothing and eye protection is allowed if the applicant commits to their use during severe radiation releases. However, even though protective clothing is used, the calculated unprotected skin dose is not to exceed 75 rem. The skin and thyroid dose levels are to be used only for judging the acceptability of the design provisions for protecting control room operators under postulated design basis accident conditions. They are not to be interpreted as accept able emergency doses. The dose levels quoted here are derived for use in the controlled plant environment and should not be confused with the conservative dose computation assumptions used in evaluating exposures to the general public for the purposes of comparison with the guideline values of 10 CFR Part 100.

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the protective action limit is used to determine the acceptability of emergency zone protection provisions during the time personnel are in the process of fitting themselves with self-contained breathing apparatus. The other limits are used to determine whether the concentrations with breathing apparatus in place are applicable. They are also used in those cases where the toxic levels are such that emergency zone isolation with out use of protective gear is sufficient. Self-contained breathing apparatus for the control room personnel (at least 5 individuals) should be on hand. A six-hour onsite bottled air supply should be available with unlimited offsite replenishment capability from nearby location(s). As an example of appropriate limits, the following are the three levels for chlorine gas:

protective action:	15 ppm by volume
short-term:	4 ppm by volume
long-term:	1 ppm by volume

Regulatory Guide 1.18 (Ref. 7) provides a partial list for protective action levels for other toxic gases.

III. REVIEW PROCEDURES

The reviewer selects and emphasizes aspects of the areas covered by this review plan as appropriate for a particular case. The judgment on areas to be given attention and emphasis in the review is based on an inspection of the material presented to see whether it is similar to that recently reviewed for other plants and whether it is of special safety significance are Involved.

1. Control Room Emergency Zone

The reviewer verifies that the control room emergency zone includes the areas identified in Subsection II.1 of this SRP section. The emergency zone should be limited to those spaces requiring operator occupancy. Spaces such as battery rooms, cable spreading rooms, or other spaces not requiring continuous or frequent occupancy after a design basis accident (DBA) generally should be excluded from the emergency zone. Inclusion of these spaces may increase the probability of smoke or hazardous gases entering the emergency zone. They may also increase the possibility of infiltration into the emergency zone, thus decreasing the effectiveness of the ventilation system in excluding contamination. It is advantageous to have the emergency zone located on one floor, with the areas included in the zone being contiguous.

2. Control Room Personnel Capacity

A control room designed with complete isolation capability from the out side air to provide radiation and toxic gas protection is reviewed to determine if the buildup of carbon dioxide could present a problem. The air inside a 100,000 cubic foot control room would support five persons for at least six days. Thus, CO(2) buildup in an isolated emergency zone is not normally considered a limiting problem.

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3. Ventilation System Layout and Functional Design

The reviewer evaluates the control room ventilation system in order to establish appropriate parameters to be used In the control room dose calculations. The review is coordinated with the ASB which evaluates the control room ventilation system design and performance in accordance with SRP Section 9.4.1. the procedures are as follows:

a. The type of system proposed is determined. The following types of protection provisions are currently being employed for boiling water reactor (BWR) or pressurized water reactor (PWR) plants:

(1) Zone isolation, with the incoming air filtered and a positive pressure maintained by the ventilation system fans. This arrangement is often provided for BWRs having high stacks. Air flow rates are between 400 and 4000 cfm.

(2) Zone isolation, with filtered recirculated air. This arrangement is often provided for BWRs and PWRs with roof vents. Recirculation rates range from 2,000 to 30,000 cfm.

(3) Zone isolation, with filtered recirculated air, and with a positive pressure maintained in the zone. This arrangement is essentially the same as that in (2), with the addition of the positive pressure provision.

(4) Dual air inlets for the emergency zone. In this arrangement two widely spaced inlets are located outboard, on opposite sides of potential toxic and radioactive gas sources. The arrangement guarantees at least one inlet being free of contamination, except under extreme no-wind conditions. It can be used in all types of plants. Makeup air supplied from the contamination-free inlet provides a positive pressure in the emergency zone and thus minimizes infiltration.

(5) Bottled air supply for a limited time. In this arrangement a flow rate of 400 to 600 cfm is provided from compressed air containers for about one hour to prevent inleakage. It is used in systems having containments whose internal atmospheric pressure becomes negative within an hour after a DBA (subatmospheric containments).

b. the input parameters to the radiological dose model are determined (see Item 5 below). The parameters are emergency zone volume, filter efficiency, filtered makeup air flow rate, unfiltered inleakage (infiltration), and filtered recirculated air flow rate.

c. The ventilation system components and the system layout diagrams are examined. The review will be coordinated with the ASB in particular if there are questions pertaining to the system design. ASB will determine if the system meets the single failure criterion as well as other safety requirements under SRP Section 9.4.1. Damper failure and fan failure are especially important. The review should confirm that the failure of isolation dampers on the upstream side of fans will not result in too much unfiltered air entering the control room The radiation dose and toxic gas analysis results are used to determine how much unfiltered air can be tolerated.

d. The following information may be used in evaluating the specific system types (see Reference 6 for further discussion):

(1) Zone isolation with filtered incoming air and positive pressure. These systems may not be sufficiently effective in protecting against iodine. The staff allows an iodine protection factor (IPF), which is defined as the time-integrated concentration of iodine outside over the time-integrated cencentration within the emergency zone, of 20 to 100 for filters built, maintained, and operated according to Regulatory Guide 1.52 (Ref. 5). An IPF of 100 requires deep bed filters. Such systems are likely to provide a sufficient reduction in iodine concentration only if the source is at some distance from the inlets. Thus, in most cases only plants with high stacks

(about 100 meters) ... d meet GDC 19 (Ref. 3) with this system.

(2) Zone isolation ... th filtered recirculated air. These systems have a greater potential for controlling iodine than those having once-through filters. IPFs ranging from 20 to over 150 can be achieved. These are the usual designs for plants having vents located at containment roof level. A system having a recirculation rate of 5000 cfm and a filter efficiency of 95% would be rated as follows:

Infiltration (cfm)	IPF/*/
200	25
100	49
50	96
25	191

/*/Within the range of interest, the iodine protection factor is directly proportional to recirculation flow rate times efficiency.

Infiltration should be determined conservatively. The calculated or measured gross leakage is used to determine the infiltration rate that will be applied in the evaluation of the radiological consequences of postulated accidents. This rate is determined as follows:

(i) the leakage from the control room when pressurized to 1/8-inch water gauge is calculated on the basis of the gross leakage data. One-half of this value is used to represent the base infiltration rate. Component leak rates may be used to calculate gross leakage (see, for example, References 9 and 10).

(ii) The base infiltration rate is augmented by adding to it the estimated contribution from opening and closing of doors associated with such activities as required by the plant emergency plans and procedures. Normally 10 cfm is used for this additional contribution.

(iii) An additional factor that is used to modify the base infiltration rate is the enhancement of the infiltration

occurring at the dampers or valves upstream of recirculation fans. When closed, these dampers typically are exposed to a pressure differential of several-inches water gauge. This is accounted for by an additional infiltration contribution over the base infiltration of 1/8-inch water gauge.

The use of an infiltration rate that is based on calculation is acceptable except in the case where the applicant has assumed exceptionally low rates of infiltration. in these cases, more substantial verification or proof may be required. For instance, if an applicant submits an analysis that shows a gross leakage rate of less than 0.06 volume changes per hour, the reviewer would require that the gross leakage be verified by periodic tests as described in Regulatory Position C.5 of Regulatory Guide 1.95 (Ref. 8).

(3) Zone isolation with filtered recirculated air and a positive pressure. This system is essentially the same as the preceding one. However, an additional operational mode is possible. Makeup air for pressurization is admitted. It is filtered before entering the emergency zone. Pressurization reduces the unfiltered inleakage that is assumed to occur when the emergency tone is not pressurized. Assuming a filter fan capacity of 5000 cfm and a filter efficiency of 95%, the following protection factors result (flows in cfm):

Makeup Air	Recirculated Air	IPF (Assuming No Infiltration)	IPF (Assuming Infiltration/*/)
400	4600	238	159
750	4250	128	101
1000	4000	96	80

The makeup flow rate should have adequate margin to assure that the control room will be maintained at a pressure of at least 1/8-inch water gauge. The applicant should indicate that an acceptance test will be performed to verify adequate pressurization. If the makeup rate is less than 0.5 volume changes per hour, supporting calculations are required to verify adequate air flow. If the makeup rate is less than 0.25 volume changes per hour, periodic verification testing is required in addition to the calculations and the acceptance test.

A question that often arises is whether "pressurization" or "isolation and recirculation" of the control room is to be preferred. Which design gives the lowest doses depends upon the assumptions as to unfiltered inleakage. Isolation limits the entrance of noble gases (not filterable) and, in addition, it

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/*/Normally 10 cfm infiltration is assumed for conservatism. This flow could be reduced or eliminated if the applicant provides assurance that backflow (primarily as a result of ingress and egress) will not occur. This may mean installing two-door vestibules or equivalent.

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is a better approach when the accident involves a short-term "puff release. If infiltration is 25 cfm or less, "isolation" would be best in any event.

A second question related to the first involves the method of operation. The following possibilities have been considered:

(i) automatic isolation with subsequent manual control of pressurization.

(ii) automatic isolation with immediate automatic pressurization.

The first is advantageous in the case of external puff releases. Simple isolation would maintain the buildup of the unfilterable noble gases. It would also protect the filters from excessive concentrations in the case of a chlorine release. However, the second method does guarantee that infiltration (unfiltered) is reduced to near zero immediately upon accident detection. this would be beneficial in the case where the contamination transport path to the emergency zone is mainly inside the building. Method (i) should be used in the case of a toxic gas release and either method (i) or (ii) should be used in the case of a radiological release, provided GDC 19 (Ref. 3) can be satisfied. A substantial time delay should be assumed where manual isolation is assumed, e.g., 20 minutes for the purposes of dose calculations.

(4) Dual air inlets for the emergency zone. Several plants have utilized this concept. The viability of the dual inlet concept depends upon whether or not the placement of the inlets assures that one inlet will always be free from contamination. The assurance of a contamination-free inlet depends in part upon building wake effects, terrain, and the possibility of wind stagnation or reversal. For example, in a situation where the inlets are located at the extreme edges of the plant structures (e.g., one on the north side and one on the south side), it is possible under certain low probability conditions for both inlets to be contaminated from the same point source. Reference 6 presents the position for dealing with the evaluation of the atmospheric dispersion (X/Q values) for dual inlet systems.

With dual inlets placed on plant structures on opposite sides of potential radiation release points (e.g. containment building) and capable of functioning with an assumed single active failure in the inlet isolation system, the following considerations may be applied to the evaluation of the control room X/Q's:

(i) Dual inlet designs without manual or automatic selection control - equation (6) of Reference 6 may be used with respect to the least favorable Inlet location to estimate X/Q is. The estimated values can be reduced by a factor of 2 to account for dilution effects associated with a dual inlet configuration. This is based upon the dilution derived from drawing in equal amounts of clean and contaminated air through two open inlets.

(ii) Dual inlet designs limited to manual selection control equation (6) of Reference 5 may be used with respect to the more favorable inlet location to estimate the X/Q's. The estimated values can be reduced by a factor of 4 to account for dilution effects associated with a dual inlet configuration and the relative probability that the operator will make the proper Inlet selection. the reduction factor is contingent upon having redundant radiation detectors within each air inlet. the reduction factor is based on the judgment that trained control room operators, in conjunction with radiation alarm indication, will select and close the contaminated air Inlet.

(iii) Dual inlet designs with automatic selection control features - equation (6) of Reference 6 may be used with respect to the more favorable inlet location to estimate the X/Q's. The estimated values can be reduced by about a factor of 10 to account for the ability to select a "clean" air inlet. The actual factor may be somewhat higner if the inlet configuration begins to approach the remote air inlet concept such that the probability of having one clean air inlet is relatively high. Plant configuration and meteorological conditions should be used as the principal basis for reduction factors greater than 10. the reduction factor of 10 or more is contingent upon having redundant detectors in each inlet and the provisions of acceptable control logic which would be used in the automatic selection of a clean air Inlet.

Because damage to the ducting might seriously affect the system capability to protect the operators, the ducting should be seismic Category I and should be protected against tornado missiles. In addition, the number and placement of dampers must be such as to assure both flow and isolation in each inlet assuming one single active component-failure (see Appendix A for information on the damper repair alternative). The location of the Intakes with respect to the plant security fence should also be reviewed.

(5) Bottled air supply for a limited time. In some plant designs the containment pressure is reduced below atmospheric within one hour after a DBA. this generally assures that after one hour significant radioactive material will not be released from the containment. Such a design makes it feasible to maintain the control room above atmospheric pressure by use of bottled air. Periodic pressurization tests are required to determine that the rated flow (normally about 300 to 600 cfm) is sufficient to pressurize the control room to at least 1/8-inch water gauge. The system is also required to be composed of several separate circuits, one of which is assumed to be inoperative to account for a possible single failure. At least one non-redundant, once through filter system for pressurization as a standby for accidents of long duration should be provided.

Compressed air bottles should be protected from tornado missiles or internally-generated missiles and should be placed so as not to cause damage to vital equipment or interference with operation if they fail.

4. Atmosphere filtration Systems

ETSB evaluates the iodine removal efficiency of the atmosphere filtration systems under SRP Section 6.5.1, determines the appropriate credit to be given and advises the AEB reviewer.

5. Relative Location of Source and Control Room

The SAB will identify all potential sources of toxic or otherwise potentially hazardous gases as described in SRP Section 2.2. The SAB will provide to the AEB th· findings of its toxic gas estimates for use in the control room habi· lity analysis. There are three basic categories; Radioactive sources, tox . ses such as chlorine, and gases with the potential for being released inside confined areas adjacent to the control room.

a. Radiation Sources

The LOCA source terms determined from the AEB review in accordance with Appendix A to SRP Section 15.6.5 are routinely used to evaluate radiation levels external to the control room. The dispersal from the containment or the standby gas treatment vent is determined with a building wake diffusion model. This model is discussed in Reference 6. Cont mination pathways internal to the plant are examined to determine their impact on control room habitability. Other DBAs are reviewed to determine whether they might constitute a more severe hazard than the LOCA. If appropriate, an additional analysis is performed for the suspect DBAs.

b. Toxic Gases

The SAB will review and identify those toxic substances stored or transported in the vicinity of the site which way pose a threat to the plant operators upon a postulated accidental release. The method used to determine whether the quantity or location of the toxic material is such as to require closer study is described in Regulatory Guide 1.78 (Ref. 7). this guide also discusses the methods for analyzing the degree of risk and states, in general terms, the various protective measures that could be instituted if the hazard is found to be too great. In the case of chlorine, specific acceptable protective provisions have been determined (Ref. 8).

In summary, the following provisions or their equivalent are required for the emergency zone ventilation system:

(1) quick-acting toxic gas detectors,

(2) automatic emergency zone isolation,

(3) emergency zone leaktightness.

(4) limited fresh air makeup rates, and

(5) breathing apparatus and associated bottled air supply.

The best solution for a particular case will depend on the toxic gas in question and on the specific ventilation system design.

c. Confined Area Releases

The reviewer studies the control building layout in relation to potential sources of radiation and toxic gases inside the control building or adjacent connected buildings. The following is considered:

(1) Storage location of CO(2) or other fire fighting materials should be such as to eliminate the possibility of significant quantities of the gases entering the emergency zone. The review will be coordinated with the Chemical Engineering Branch (CMEB).

(2) The ventilation zones adjacent to the emergency zone should be configured and balanced to preclude air flow toward the emergency zone.

(3) All pressurized equipment and piping (e.g., main steam lines and turbines) that could cause significant pressure gradients when failed inside buildings should be isolated from the emergency zone by multiple barriers such as multiple door vestibules or their equivalent.

6. Radiation Shielding

Control room operators as well as other plant personnel are protected from radiation sources associated with normal plant operation by a combination of shielding and distance. The adequacy of this type of protection for normal operating conditions is coordinated with the RAB. To a large extent the same radiation shielding (and missile barriers) also provides protection from DBA radiation sources. This is especially true with respect to the control room walls which usually consist of at least 18 inches of concrete. In most cases, the radiation from external DBA radiation sources is attenuated to negligible levels. However, the following items should be considered qualitatively in assessing the adequacy of control room radiation shielding and should be coordinated with the RAB who will be requested to provide assistance as necessary.

a. Control room structure boundary. Wall, ceiling, and floor materials and thickness should be reviewed. Eighteen inches to two feet of concrete or its equivalent will be adequate in most cases.

b. Radiation streaming. The control room structure boundary should be reviewed with respect to penetrations (e.g., doors, ducts, stairways). the potential for radiation streaming from accident sources should be identified, and if deemed necessary, quantitatively evaluated.

c. Radiation shielding from internal sources. If sources internal to the control room complex are identified, protective measures against them should be reviewed typical sources in this category include contaminated filter trains, or airborne radioactivity in enclosures adjacent to the control room.

Evaluations of radiation shielding effectiveness with respect to the above items should be performed using simplified analytical models for point, line, or volume sources such as those presented in References 11 and 12. If more extended analysis is required, analytical support from the RAB should be requested. The applicant's coverage of the above items should also be reviewed in terms of completeness, method of analysis, and assumptions.

7. Independent Analyses

The applicant is required to calculate doses to the control room operators. Independent analyses are made by the AEB because of the diversity of control room habitability system designs and the engineering judgment Involved in their evaluation. Using the approach indicated in Reference 6, the source terms and doses due to a DBA are calculated. The source terms determined by the AEB's independent analysis of low population zone (LPZ) doses for a LOCA are used. The methods and assumptions for this calculation are presented in Appendix A to SRP Section 15.6.5. The control room doses are determined by estimating the X/Q from the source points to the emergency zone using meteorological input supplied by the assigned meteorologist, by determining the credit for the emergency zone's protection features, and by calculating the dose. the attached Table 6.4-1 is a form which may be used to summarize the information that is needed for the control room dose calculation. the effective X/Q's are used for calculating the doses. The dose is then compared with the guidelines of GDC 19. If the guideline values are exceeded, the applicant will be requested to improve the system. In the event that other DBAs are expected to result in doses comparable to or higher than the LOCA, additional analyses are performed. the limiting consequences of the accidents are compared with Criterion 19.

IV. EVALUATION FINDINGS

The reviewer verifies that sufficient information has been provided and that the review and calculations support conclusions of the following type, to be included in the staff's safety evaluation report (note: items 2 and 3 should be included only if appropriate):

We conclude that the control room habitability system of the (insert PLANT NAME) facility Is acceptable and meets the requirements of the following General Design Criteria:

1. GDC 19, "Control Room," with respect to maintaining the control room in a safe and habitable condition under accident conditions by providing adequate protection against radiation and toxic gases such that the radiological exposures are within the limits of this criterion, and

2. GDC 4, "Environmental and Missile Design Bases," with respect to the environmental effects of the release of toxic gases and

3. GDC 5, "Sharing of Structures, Systems and Components," with respect to ensuring that the control room, shared by Units and of the (insert PLANT NAME) facility will not significantly impair the ability of the control room personnel to perform safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the other unit(s)."

These conclusions are based on the staff review and evaluation that the control room habitability systems meet the regulatory positions of Regulatory Guide 1.52, "Design Testing and Maintenance Criteria for Engineered-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," and Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Release."

In meeting the positions of these regulatory guides, the applicant has demonstrated that the control room will adequately protect the control room operators and will remain habitable in accordance with Task Action Plan Item III.D.3.4 of NUREG-0737.

If the design is not adequate, the fact is stated. Alternatives such as an increase in the charcoal filter flow rate may be indicated as is given in the example below:

The staff has calculated the potential radiation doses to control room personnel following a LOCA. The resultant whole body doses are within the guidelines of General Design Criterion 19. The thyroid dose resulting from exposure to radioactive iodine exceeds the dose guidelines. A method of meeting GDC 19 would be to increase the filtration system size from 2000 cfm to 4000 cfm. This increased filtration will be sufficient to keep the estimated thyroid doses within the guidelines.

If special protection provisions for toxic gases are not required, the following statement or its equivalent is made:

The habitability of the control room was evaluated using the procedures described in Regulatory Guide 1.78. As indicated in Section 2.2, no offsite storage or transport of chemicals is close enough to the plant to be considered a hazard. There are no onsite chemicals that can be considered hazardous under Regulatory Guide 1.78. A sodium hypochlorite biocide system will be used, thus eliminating an onsite chlorine hazard Therefore, special provisions for protection against toxic gases will not be required. In accordance with plant emergency plans and procedures, self-contained breathing apparatus is provided for assurance of control room habitability in the event of occurrences such as smoke hazards.

If special protection provisions are required for toxic gases, compliance or noncompliance with the guidelines of Regulatory Guides 1.78 and 1.95 should be stated.

V. IMPLEMENTATION

The following provides guidance to applicants and licensees regarding the staff's plans for using this SRP section.

Except in those cases in which the applicant proposes an acceptable alter native method for complying with specified portions of the Commission's regulations, the method described herein will be used by the staff in its evaluation of conformance with Commission regulations.

Implementation schedules for conformance to parts of the method discussed herein are contained in the referenced regulatory guides.

VI. REFERENCES

1. 10 CFR Part 51 Appendix A, General Design Criterion 4, "Environmental and Missiles Design Bases."

2. 10 CfR Part 50, Appendix A, General Design Criterion 5, "Sharing of Structures, Systems and Components."

3. 10 CFR Part 50, Appendix A, General Design Criterion 19, "Control Room."

4. NUREG-0737, "Clarification of TMI Action Plan Requirements," Item III.D.3.4, "Control Room Habitability," November 1980.

5. Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

6. K.G. Murphy and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," 13th AEC Air Cleaning Conference, August 1974.

7. Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."

8. Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release."

9. "Leakage Characteristics of Openings for Reactor Housing Components," NM-SR-MEMO-5137, Atomics International, Div. of North American Aviation, Inc., June 20, 1960.

10. R.L. Koontz, et al., "Leakage Characteristics of Conventional Building Components for Reactor Housing Construction," Trans. Am. Nucl. Soc., November 1961.

11. R.G. Jaeger, et al., eds., "Engineering Compendium on Radiation Shielding," Vol. 1, "Shielding Fundamentals and Methods," Springer Verlag (1968).

12. N.M. Schaeffer, "Reactor Shielding for Nuclear Engineers," TID-75951, U.S. Atomic Energy Commission.

SECTION 6.4 APPENDIX A

ACCEPTANCE CRITERIA FOR VALVE OR DAMPER REPAIR ALTERNATIVE

The control room ventilation system must meet the criterion to function properly, even with a single failure of an active component. In certain cases, complex valve or damper configurations are required to meet the single failure criterion. For example, assurance of the isolation and operability of each leg of a dual inlet system at various times after a postulated accident could require a four-valve arrangement in which two pairs of series valves are connected in parallel. The mechanical, power, and control components of such arrangements combine to form a rather complex system. Credit will be allowed for an alternative system that allowed the failed valve to be manually repositioned so that it will not interfere with the operation of the system. For example, in the case of a dual inlet system, if credit for repair is given, then two valves in series in each leg of the dual inlet would be acceptable. Where a valve fails closed but meets the criteria given below, credit would be allowed for the valve to be repositioned and locked in an open position.

The approval of the repair option is contingent upon the intrinsic reliability of the internal components of the valve or damper and also upon the ease and ability to overcome the failure of the external actuating components (electrical relays, motors, hydraulic pistons, etc.). The following criteria or their equivalent will be required.

1. The valve or damper components must be listed as to which are considered internal (nonrepairable) and which external (repairable). These must be designed to meet the following criteria.

a. Internal valve components (i.e., components that are difficult to repair manually without opening the ductwork) must be judged to have a very low probability of failure. The component design details will be reviewed and characteristics such as simplicity, ruggedness, and susceptibility to postulated failure mechanisms will be considered in arriving at an engineered judgment of the acceptability of the Internal component design with respect to reliability. For example, a butterfly valve welded or keyed onto a pivot shaft would be considered a high reliability internal component. Conversely, multiple blade dampers, actuated by multi-element linkages or pneumatically operated components internal to the ducts, would be viewed as being subject to failure.

b. External valve components (i.e., components including motors and power supplies that are to be assumed repairable or removable) must be designed to ensure that the failed valve component can be bypassed easily and safely and that the valve can be manipulated into an acceptable position. The electronic components must be isolated from other equipment to assure that the repair operations do not result in further equipment failure.

2. The location and positioning of the valve or damper must permit easy access from the control room for convenient repair, especially under applicable DBA conditions.

3. Appropriate control room instrumentation should be provided for a clear indication and annunciation of valve or damper malfunction.

4. Periodic manipulation of the valve or damper by control room operators should be required for training purposes and to verify proper manual operability of the valve or damper.

5. The need for manual manipulations of the failed valve or damper should not be recurrent during the course of the accident. Manipulation should not occur more than once during the accident. Adjustment or realignment of other parts of the system should be possible from the control room with the failed valve in a fixed position.

6. The time for repair used in the computation of control room exposures should be taken as the time necessary to repair the valve plus a one-half hour margin. No manual correction will be credited during the first two hours of the accident.

7. Compliance with the above criteria should be documented in the SAR when ever the repair option is used.

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TABLE 6.4-1 Summary Sheet for Control Room Dose Analysis

MEMORANDUM TO:______________________________ , AEB Lead Reviewer
____, Meteorologist

cc: Meteorology Section, AEB
AEB Habitability Files

CONCERNING CONTROL ROOM DOSE ANALYSIS FOR (Insert Plant Name)

The following summarizes the X/Q's used in determining the control room operator dose for the subject plant:

A. VENTILATION SYSTEM DESCRIPTION

B. SKETCH OF SYSTEM (and inlets/sources if applicable)

C. SUMMATION OF X/Q ANALYSIS

Source/Receptor Type and Distance

S/D Ratio — K Factor

Number of 22-1/2x Wind Direction Sectors that Result in Exposure

Central Wind Sector — (sector wind is blowing from)

5% Wind Speed (m/sec) — 40% Wind Speed (m/sec)

Projected Area of Wake (m**2) — 5% X/Q (sec/m**3)

Time	Wind Speed Factor	Wind Direction Factor	Occupancy Factor	Effective X/Q's
0-8 hr	1	1		
8-24 hr		1	1	
1-4 day		0.6		
4-30 day		0.4		

D. ACTION REQUESTED

Assigned Reviewer

- For your information only
- Please use the effective X/Q's in TACT run and provide control room doses.

In addition, please summarize safety system assumptions and indicate their status (interim or final).

Meteorologist

- These are interim X/Q's. Please review to determine their reasonableness.
- These are final X/Q's. Please determine if they are accurate based on your analysis of site data.

Please Contact

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-8064

14-4(12)

Docket: 50-298
License: DPR-46

August 27, 1993

Nebraska Public Power District
ATTN: Guy R. Horn, Vice President, Nuclear
P.O. Box 98
Brownville, Nebraska 68321

SUBJECT: FINAL SYSTEMATIC ASSESSMENT OF LICENSEE PERFORMANCE (SALP) REPORT

This forwards the final SALP report (50-298/93-99) for the Cooper Nuclear Station for the period of January 19, 1992, through April 24, 1993. This final SALP Report includes:

1. The cover letter for the initial SALP report (no revisions to the initial SALP report were made).

2. A summary of our July 12, 1993, meeting at the Cooper Nuclear Station security building auditorium in Brownville, Nebraska.

3. Your August 11, 1993, response to the initial SALP report.

We have reviewed your letter dated August 11, 1993, in response to the NRC recommendations in each of the SALP functional areas. It was noted that your response has identified specific actions to improve performance in each of the SALP functional areas. We will review your progress to achieve these improvements in inspection efforts during this SALP period.

The next SALP period for Cooper Nuclear Station is scheduled to last approximately 18 months, from April 25, 1993, to October 22, 1994. As identified in our letter dated August 11, 1993, from Mr. A. B. Beach, Director, Division of Reactor Projects, to Mr. G. R. Horn, Vice President, Nuclear, the revised SALP program will be utilized.

Sincerely,

James L. Milhoan
Regional Administrator

Enclosures:
1. Cover letter for [illegible]tial SALP report
2. NRC Meeting Summary
3. Nebraska Public Power District response to the initial SALP report

cc: (see next page)

N 11
ATT 4 (12)

~~9309170112~~

cc w/enclosure:
Nebraska Public Power District
ATTN: G. D. Watson. General Counsel
P.O. Box 499
Columbus. Nebraska 68602-0499

Cooper Nuclear Station
ATTN: John M. Meacham. Site Manager
P.O. Box 98
Brownville. Nebraska 68321

Nebraska Department of Environmental
Control
ATTN: Randolph Wood. Director
P.O. Box 98922
Lincoln. Nebraska 68509-8922

Nemaha County Board of Commissioners
ATTN: Richard Moody. Chairman
Nemaha County Courthouse
1824 N Street
Auburn. Nebraska 68305

Nebraska Department of Health
ATTN: Harold Borchert. Director
Division of Radiological Health
301 Centennial Mall. South
P.O. Box 95007
Lincoln. Nebraska 68509-5007

Kansas Radiation Control Program Director

C: 0387

NEBRASKA PUBLIC POWER DISTRICT

14-4 (15)

CFM9400208

D: June 22, 1994

To: J. E. Lynch

From: R. E. Wilbur

Subject: Local Leak Rate Discrepancies

FOR INTER-DISTRICT BUSINESS ONLY

Reference: CFM9400193 from R. E. Wilbur to J. E. Lynch, dated June 7, 1994, Same Subject

The Nuclear Engineering and Construction Division provided a list, in the referenced memo, of the penetrations and valves that had been identified as of June 7th that had not been local leak rate tested in accordance with 10 CFR 50 Appendix J. The purpose of this memo is to update the list of valves and penetrations and to request your review as soon as possible to ensure that there are no modifications required to perform the LLRT on these valves.

Attachment 1 identifies the valves, instruments and penetrations that have either never been local leak rate tested or need an LLRT after modifications are complete. For each penetration, the valves or instruments that must be tested are identified. This testing must be completed prior to startup. This list includes those previously identified in the referenced memo (CFM940019).

Attachment 2 contains a penetration-by-penetration review identifying the status of LLRT testing. Those penetrations with an 'X' in the "Need Added to Procedure 6.3.1.1" have never been LLRT tested and will require an LLRT test prior to startup and will need to be incorporated into the LLRT program in the future. Those penetrations with an 'X' in the "Need One Time LLRT" will be modified such that they will not require future LLRT testing, but they will require one time pressure testing to prove zero leakage prior to startup (these penetrations will have welded caps and will be considered part of the containment liner).

It is important that all of these penetrations be reviewed for testability as soon as possible, to identify any additional design change work that must be accomplished before testing. In particular, please review Penetration X-35E, TIP N_2 Purge, for which it may be necessary to disassemble the check valve to test the outboard CIV. If this is not feasible, please notify NED immediately, so that test connections can be designed and installed.

Also, there are 13 pressure switches and 2 pressure transmitters (PS-12A-D,16,101A-D,119A-D and PT-512A/B; penetration X-40A-D) that are valved out during the ILRT and must be leak rate tested prior to startup. If there are any concerns with testing these, please notify NED as soon as possible.

Should you have any questions please call.

R. E. Wilbur

R. E. Wilbur
Division Manager
Nuclear Engineering & Construction

MTB llrtmemo rew

cc:		
	G. R. Horn	M. J. Spencer
	R. L. Gardner	F. A. Schizas
	J. V. Sayer	K. B. Curry
	J. M. Meacham	R. A. Jansky
	G. S. McClure	M. T. Boyce
	K. J. Done	File C5.3

N/1 Att 4 (15)

Powerful Pride in Nebraska

ATTACHMENT 1
PENETRATIONS REQUIRING LLRT

ITEM	PEN. NO.	DESCRIPTION	CIVs	MODIFICATIONS IN PROGRESS
1	X-20	Demin Water to D/W	DW-V-219 DW-V-133	None. Have never been LLRT tested
2	X-21	Service Air to D/W Ring Header	SA-V-647 SA-V-648	Two new qualified manual valves are being added (647 and 648) with test connections. These will be sealed closed. DC 94-212D SA-V-647 and 648 will require LLRT after mod
3	X-23	REC D/W Supply	REC-MOV-702MV	Adding test connections inside D/W. Will be LLRT tested after mod using freeze seal. DC 94-212C
4	X-24	REC Return from D/W	REC-MOV-709MV	Same as X-23
5	X-26	D/W Purge & Vent Exhaust	PC-MOV-306MV PC-MOV-1310 PC-MOV-231MV PC-AOV-246AV PC-PT-1B2 PC-PT-4B2 PC-PT-5B2	DC 94-212E will install qualified valves at pressure instrument test connections in addition to the caps. The valves and caps will be administratively controlled. Penetration must be LLRT tested after mod
6	X-27E	New Spare	Welded Cap	This penetration will be modified by removing the manual valve and installing a welded cap outside containment. A one time LLRT is required after mod. In future, will be considered part of Containment liner
7	X-27F	New Spare	Welded Cap	Same as X-27E
8	X-29E	Air to RR Sample Valve	Two new check valves (CIC not yet assigned)	DC 94-212Fwill move SOV directly outside Containment and install two new qualified check valves with test connections outboard of the SOV. The SOV will exhaust to containment. After the mod, the two new check valves will require LLRT.

ITEM	PEN. NO.	DESCRIPTION	CIVs	MODIFICATIONS IN PROGRESS
9	X-29F	New Spare	PC-PT-1A1 PC-PT-4A1 PC-PT-5A1	DC 94-212E will add qualified valves in addition to the caps on the test connections to the pressure instruments. After the mod, the penetration will require a one time LLRT. In future, the penetration will be part of containment tested per ILRT.
10	X-30E	Air to Reactor Vessel Flange Leakoff	Two new qualified manual valves (CIC not yet known)	DC 94-212F will install two new qualified manual valves with test connections. The valves will be sealed closed and administratively controlled. LLRT required on two new manual valves.
11	X-30F	Air to Reactor Vessel Head Vent	Same as X-30E	Same as X-30E
12	X-33E	Air to Reactor Vessel Flange Leakoff	Same as X-30E	Same as X-30E
13	X-33F	Air to Reactor Vessel Head Vent	Same as X-30E	Same as X-30E
14	X-34E	New Spare	Welded Cap	DC 94-212E removes manual valve IA-V-141, and installs welded cap outside containment. one time LLRT required. In future considered part of containment liner and included in ILRT.
15	X-34F	New Spare	Welded Cap	Same as X-34E
16	X-35A	TIP Probe	TIP Ball Valve	None. Has never been LLRT Tested
17	X-35B	TIP Probe	TIP Ball Valve	Same as X-35A
18	X-35C	TIP Probe	TIP Ball Valve	Same as X-35A
19	X-35D	TIP Probe	TIP Ball Valve	Same as X-35A
20	X-35E	TIP N_2 Purge	NM-CV-2CV NM-SOV-3SV	NONE. Has never been LLRT tested.

ITEM	PEN. NO.	DESCRIPTION	CIVs	MODIFICATIONS IN PROGRESS
21	X-37A	New Spare	Welded Cap	DC 94-212E will remove PC-V-502 and add welded cap outside containment. One time LLRT required after mod. In future, will be considered part of containment liner and included in ILRT
22	X-37B	New Spare	Welded Cap	Same as X-37A
23	X-38A	New Spare (One Line)	Welded Cap	Same as X-37A
24	X-38B	New Spare (One Line)	Welded Cap	Same as X-37A
25	X-40 A - D	Primary Containment Pressure	PS-12A - D PS-16 PS-101A - D PS-119A - D PT-512A/B	These pressure switches and transmitters were inadvertently valved out during the ILRT. Since they are containment boundary, they must be LLRT tested prior to startup and the ILRT changed to correct lineup in future.
26	X-43	Pump Floor Drains	Testable Flange	DC 94-212B replaced the single gasketed flange with a double o-ring flange. This new flange must be LLRT tested.
27	X-44	Pump Floor Drains	Testable Flange	Same as X-43
28	X-45D	SOV Air Exhaust to D/W	Two new Check valves (CIC not yet determined)	DC 94-212F will install two new qualified check valves outside containment with test connections. The two new check valves must be LLRT tested.
29	X-46A	New Spare	Welded Cap	DC 94-212F will remove the manual isolation valve and cut and add a welded cap outside containment. This penetration requires a one time pressure test to prove zero leakage. In future, penetration will be considered to be part of containment liner and ILRT tested.
30	X-46B	New Spare	Welded Cap	Same as X-46A
31	X-46C	New Spare	Welded Cap	Same as X-46A
32	X-46D	New Spare	Welded Cap	Same as X-46A

ITEM	PEN. NO.	DESCRIPTION	CIVs	MODIFICATIONS IN PROGRESS
33	X-46E	New Spare	Welded Cap	Same as X-46A
34	X-46F	New Spare	Welded Cap	DC 94-212F will remove the manual isolation valve and cut and add a welded cap outside containment.. This penetration requires a one time pressure test to prove zero leakage. In future, penetration will be considered to be part of containment liner and ILRT tested
35	X-47A	New Spare	Welded Cap	Same as X-46F
36	X-47C	New Spare	Welded Cap	Same as X-46F
37	X-47D	New Spare	Welded Cap	Same as X-46F
38	X-47E	New Spare	Welded Cap	Same as X-46F
39	X-47F	New Spare	Welded Cap	Same as X-46F
40	X-49E	New Spare	Welded Cap	DC 94-212F will remove the manual isolation valve and cut and add a welded cap outside containment. This penetration requires a one time pressure test to prove zero leakage. In future, penetration will be considered to be part of containment liner and ILRT tested
41	X-49F	New Spare	Welded Spare	Same as X-49E
42	X-51B	SOV Control Air to RR - AOV-741AV Exhaust	Two new check valves (CIC not yet determined)	DC 94-212F uses penetration X-51B to exhaust SOV control air back to the D/W from RR-AOV-741AV. This penetration must be LLRT tested in conjunction with X-29E.
43	X-51C	New Spare	Welded Cap	DC 94-212F will remove the manual isolation valve and cut and add a welded cap outside containment. This penetration requires a one time pressure test to prove zero leakage In future, penetration will be considered to be part of containment liner and ILRT tested

ITEM	PEN. NO.	DESCRIPTION	CIVs	MODIFICATIONS IN PROGRESS
44	X-51D	New Spare	Welded Cap	Same as X-51C
45	X-51F	PASS D/W Atmosphere	PAS-AOV-3AV PAS-AOV-12AV	DC 94-212H replaced the existing 3AV and 12AV with qualified valves, moved them closer to the penetration and seismically qualified the line out to the second CIV. Test connections were added to allow LLRT testing. The two AOVs must be LLRT tested prior to startup.
46	X-52E	New Spare	Welded Cap	DC 94-212F will remove the manual isolation valve and cut and add a welded cap outside containment. This penetration requires a one time pressure test to prove zero leakage. In future, penetration will be considered to be part of containment liner and ILRT tested
47	X-52F	New Spare	Welded Cap	Same as X-52E
48	X-100B	New Spare	Welded Caps	Same as X-52E (two lines)
49	X-203A	H_2O_2 Analyzer	Extension of Containment	DC 94-212E will either weld caps or replace "T" with straight pipe eliminating caps. After mod, must be LLRT tested.
50	X-203B	H_2O_2 Analyzer	Extension of Containment	Same as X-203A
51	X-206A	Torus Water Level Indication	Extension of Containment	MWR 94-2978 will add caps on instrument line valves. This will require a one time LLRT prior to startup. In future will be tested per ILRT
52	X-206B	Torus Water Level Indication	Extension of Containment	Same as X-206A
53	X-206C	Torus Water level Indication	Extension of Containment	Same as X-206A
54	X-206D	Torus Water Level Indication	Extension of Containment	Same as X-206A

ITEM	PEN. NO.	DESCRIPTION	CIVs	MODIFICATIONS IN PROGRESS
55	X-209A	Torus Air Temperature	Epoxy Seal	DC 94-212A will modify design of penetration to include qualified epoxy seal. Will require LLRT prior to startup
56	X-209B	Torus Water Temperature	Epoxy Seal	DC 94-212A will modify design of penetration to include qualified epoxy seal. Will require LLRT prior to startup
57	X-209C	Torus Air Temperature	Epoxy Seal	Same as X-209B
58	X-209D	Torus Water Temperature	Epoxy Seal	Same as X-209B
59	X-214	HPCI Turbine Exhaust Drain	HPCI-AO70,71 RHR-MO167A,166A RHR-MO167B,166B RV-18,19,20,21	MWR 942978 will add caps to pressure instruments and vent lines directly connected to containment. These will require LLRT. The RHR RVs have never been tested and require LLRT prior to startup.
60	X-215	Torus Air Pressure	Extension of Containment	DC94-212E will add a valve in addition to the cap for PI-20. Will require one time LLRT prior to startup. In future will be extension of containment tested per ILRT
61	X-218	New Spare	Welded Cap	DC 94-209 cut the line removed the thermocouples and welded a cap outside containment. Will require one time LLRT prior to startup. In future will be considered extension of containment tested per ILRT.
62	X-220	Torus Purge and Vent Exhaust	PC-MO230, AO245 PC-MO305, MO1308 PC-V-143 at Rack 137	DC 94-212E added a cap in addition to valve PC-V-43 at Local Rack 137. This requires an LLRT prior to startup.

ITEM	PEN. NO.	DESCRIPTION	CIVs	MODIFICATIONS IN PROGRESS
63	X-229A	Vacuum Breaker Actuating Air	two manual valves (CIC not yet determined)	DC 94-212F will qualify the existing manual valve and add a second manual valve and test connections. Both valves will be sealed closed and administratively controlled. Both manual valves must be LLRT tested prior to startup.
64	X-229B	Vacuum Breaker Actuating Air	Same as X-229A	Same as X-229A
65	X-229C	Vacuum Breaker Actuating Air	Same as X-229A	Same as X-229A
66	X-229D	Vacuum Breaker Actuating Air	two manual valves (CIC not yet determined)	DC 94-212F will qualify the existing manual valve and add a second manual valve and test connections. Both valves will be sealed closed and administratively controlled. Both manual valves must be LLRT tested prior to startup
67	X-229E	Vacuum Breaker Actuating Air	Same as X-229D	Same as X-229D
68	X-229F	Vacuum Breaker Actuating Air	Same as X-229D	Same as X-229D
69	X-229G	Vacuum Breaker Actuating Air	Same as X-229D	Same as X-229D
70	X-229H	Vacuum Breaker Actuating Air	Same as X-229D	Same as X-229D
71	X-229J	Vacuum Breaker Actuating Air	Same as X-229D	Same as X-229D
72	X-229K	Vacuum Breaker Actuating Air	Same as X-229D	Same as X-229D
73	X-229L	Vacuum Breaker Actuating Air	Same as X-229D	Same as X-229D

Pen. No.	Penetration Description	[illegible]			
		Currently in Procedure 6.3.1.1	Need Added to Procedure 6.3.1.1	Need One Time LLRT	Not Required
DWH	Drywell Head	X			
SIP1-8	Stabilizer Inspection Ports	X			
X-1A/B	Equipment Hatches	X			
X-2	Personnel Air Lock	X			
X-4	Access Hatch	X			
X-5A-H	Drywell Vent				X (Note 1)
X-6	CRD Hatch	X			
X-7A/D	Main Steam to Turbine	X			
X-7A/D Bellows	Main Steam to Turbine	X			
X-8	MSIVs Drain Line	X			
X-9A/B	Reactor Feedwater Supply	X			
X-9A/B Bellows	Reactor Feedwater Supply	X			
X-10	RCIC Steam Supply	X			
X-11	HPCI Steam Supply	X			
X-12	RHR Shutdown Cooling	X			
X-13A/B	RHR Loop Injection	X			
X-14	RWCU Supply	X			
X-15	Existing Spare				X (Note 2)
X-16A/B	Core Spray Loop Injection	X			
X-17	Existing Spare				X (Note 2)
X-18	Drywell Equipment Sump Discharge	X			
X-19	Drywell Floor Sump Discharge	X			
X-20	Demineralized Water Supply for Drywell		X		
X-21	Service Air Containment Isolation Valves		X		
X-22	Instrument Air Containment Isolation Valves		X		
X-23	RBCCW System Supply to Drywell		X		
X-24	RBCCW System Return from Drywell		X		

Pen. No.	Penetration Description	LLRT STATUS			
		Currently in Procedure 6.3.1.1	Need Added to Procedure 6.3.1.1	Need One Time LLRT	Not Required
X-25	Drywell Purge and Vent Supply & Dilution Supply Valves	X			
X-26	Drywell Purge and Vent Exhaust	X		X (Note 3)	
X-27A	Pressure Above Core Plate				X (Note 4)
X-27B	Pressure Below Core Plate				X (Note 4)
X-27C	Turbine Steam Line Pressure				X (Note 4)
X-27D	Turbine Steam Line Pressure				X (Note 4)
X-27E	New Spare			X (Note 5)	
X-27F	New Spare			X (Note 5)	
X-28A	RPV Level & Pressure Instrumentation				X (Note 4)
.8B	RPV Level & Pressure Instrumentation				X (Note 4)
X-28C	RPV Level & Pressure Instrumentation				X (Note 4)
X-28D	RPV Level & Pressure Instrumentation				X (Note 4)
X-28E	RPV Level & Pressure Instrumentation				X (Note 4)
X-28F	RPV Flange Seal Leak Detection				X (Note 4)
X-29A/D	RPV Level & Pressure Instrumentation				X (Note 4)
X-29E	Air to RR Sample Valve		X		
X-29F	New Spare			X (Note 5)	
X-30A/D	Main Steam Line Flow Measurement				X (Note 4)
X-30E	Air to Reactor Vessel Head Vent		X		
0F	Air to Reactor Vessel Head Vent		X		
X-31A,B	Reactor Recirc Loop 1A Pressure				X (Note 4)

[illegible]	[illegible]	[illegible]			
		Currently in Procedure 6.3.1.1	Need Added to Procedure 6.3.1.1	Need One Time LLRT	Not Required
X-31C/D	Reactor Recirc. Loop ΔP				X (Note 4)
X-31E/F	Reactor Recirc. Pump Seal Pressure				X (Note 4)
X-32A/D	Reactor Recirc. Loop 1A Flow				X (Note 4)
X-32E/F	Reactor Recirc. Pump Seal Leakage				X (Note 4)
X-33A/D	Reactor Recirc. Loop 1A/B ΔP				X (Note 4)
X-33E/F	Air to Vessel Flange Leakoff		X		
X-34A/D	Main Steam Line Flow Measurement				X (Note 4)
X-34E	New Spare			X (Note 5)	
X-34F	New Spare			X (Note 5)	
A/E	Traveling In-Core Probes		X		
X-35A/E Flanges	Traveling In-Core Probes	X			
X-36	Drywell H_2/O_2 Monitors (3 lines)	X (Procedure 6.3.1.1.1)			
X-37A (31 lines)	Control Rod Drive Water Insert				X (Note 6)
X-37A (1 line)	New Spare			X (Note 5)	
X-37B (37 lines)	Control Rod Drive Water Insert				X (Note 6)
X-37B (1 line)	New Spare			X (Note 5)	
X-37C (38 lines)	Control Rod Drive Water Insert				X (Note 6)
X-37C (1 line)	CRD Mini-Purge to RR Pump A	X			
X-37D (31 [illegible])	Control Rod Drive Water Insert				X (Note 6)
[illegible]37D (1 line)	Existing Spare				X (Note [illegible]

Pen. No.	Penetration Description	LLRT STATUS [illegible]			
		Currently in Procedure 6.3.1.1	Need Added to Procedure 6.3.1.1	Need One Time LLRT	Not Required
X-38A (31 lines)	Control Rod Drive Water Withdraw				X (Note 6)
X-38A (1 line)	New Spare			X (Note 5)	
X-38B (37 lines)	Control Rod Drive Water Withdraw				X (Note 6)
X-38B (1 line)	New Spare			X (Note 5)	
X-38C (38 lines)	Control Rod Drive Water Withdraw				X (Note 6)
X-38C (1 line)	CRD Mini-Purge to RR Pump B	X			
X-38D (31 lines)	Control Rod Drive Water Withdraw				X (Note 6)
X-38D (1 line)	Existing Spare				X (Note 2)
X-39A/B	Drywell Spray Loop/Dilution Supply	X			
[illegible]0A-D	Primary Containment Pressure			X (Note 3)	
X-40A-D a-f	Jet Pump Instrumentation				X (Note 4)
X-41	Reactor Water Sample	X			
X-42	SLC Injection	X			
X-43	Pump Floor Drains		X		
X-44	Pump Floor Drains		X		
X-45A	Existing Spare				X (Note 2)
X-45B	Existing Spare				X (Note 2)
X-45C	Atmosphere Radiation Monitor	X			
X-45D	SOV Air Exhaust to Drywell		X		
X-46A/F	New Spare			X	
X-47A	New Spare			X	
X-47B	Nitrogen Inerting Sys			X (Note 3)	
[illegible]-47C F	New Spare			X (Note 5)	

[illegible]	[illegible]	[illegible]			
		Currently in Procedure 6.3.1.1	Need Added to Procedure 6.3.1.1	Need One Time LLRT	Not Required
X-46	Existing Spare				X (Note 2)
X-49A/B	Existing Spare				X (Note 2)
X-49C	Electrical	X			
X-49D	Electrical	X			
X-49E/F	New Spare			X	
X-50A	Electrical	X			
X-50B	Electrical	X			
X-50C	Existing Spare				X (Note 2)
X-50D	Existing Spare				X (Note 2)
X-50E	Turbine Steam Line Pressure				X (Note 4)
X-50F	Turbine Steam Line Pressure				X (Note 4)
X-[illegible]	Pressure Below Core Plate				X (Note [illegible]
X-51B	Solenoid valve Exhaust Return		X		
X-51C	New Spare			X	
X-51D	New Spare			X	
X-51E	Atmosphere Radiation Monitor	X			
X-51F	PASS		X		
X-52A/B	RCIC System Diff Press				X (Note 4)
X-52C/D	Core Spray System Diff Press				X (Note 4)
X-52E/F	New Spare			X (Note 4)	
X-53	Existing Spare				X (Note 2)
X-100A	Instrumentation Circuits	X			
[illegible]B	New Spare			X	~~X (Note 2)~~
X-100C/D	Existing Spare				X (Note 2)

Pen. No.	Penetration Description	[illegible]			
		Currently in Procedure 6.3.1.1	Need Added to Procedure 6.3.1.1	Need One Time LLRT	Not Required
X-100E	Electrical	X			
X-100F	Instrumentation Circuits	X			
X-100G	Instrumentation Circuits	X			
X-100H	Instrumentation Circuits	X			
X-101A	5kV Power Feeders	X			
X-101B	Instrumentation Circuits	X			
X-101C	5kV Power Feeders	X			
X-101D	5kV Power Feeders	X			
X-101E	480V & 120VAC Circuits	X			
X-101F	5kV Power Feeders	X			
X-102	Instrumentation Circuits	X			
X-103	Instrumentation Circuits	X			
X-104A	Instrumentation Circuits	X			
X-104B	Instrumentation Circuits	X			
X-104C	Existing Spare				X (Note 2)
X-104D	Instrumentation Circuits	X			
X-104E	Instrumentation Circuits	X			
X-105A	480V & 120VAC Circuits	X			
X-105B/C	Existing Spare				X (Note 2)
X-105D	480V & 120VAC Circuits	X			
X-106	Instrumentation Circuits	X			
X-200A B	Torus Hatches	X			
X-201A H	DW Vent Line to Suppression Chamber				X (Note 1)
X-202A M	Vacuum Breakers				X (Note [illegible])
X-203A B	H.O. Monitors	X (Procedure 6.3.1.1.1)		X	
X-205	Torus Purge and Vent, Vacuum Relief, Dilution Supply	X			
X-206A B	Torus Water Level Indication			X (Note 3)	~~X~~ (Note --)

[illegible]	Penetration Description	[illegible]			
		Currently in Procedure 6.3.1.1	Need Added to Procedure 6.3.1.1	Need One Time LLRT	Not Required
X-206C/D	Torus Water Level Indication			X (Note 3)	~~X (Note 3)~~
X-207A/H	Drywell Vent Line to Torus Drain				X (Note 7)
X-208A/H	MS SRV Discharge				X (Note 7)
X-209A/D	Suppression Chamber Air Temperature		X		
X-210A/B	RCIC Min Flow	X			X (Note 8)
X-211A	RHR Loop A to Torus	X			
X-211B	RHR Loop B to Torus, Torus Dilution Supply	X			
X-212	RCIC Turbine Exhaust	X			
X-213A/B	Torus Drain Connection	X			X (Note 8)
X-214	HPCI Turbine Exhaust Drain	X	X (Add RV-18,19,20,21)		
X-215	Torus Air Pressure			X (Note 3)	
X-216	Existing Spare				X (Note 2)
X-217	Existing Spare				X (Note 2)
X-218	New Spare			X (Note 5)	
X-219	Existing Spare				X (Note 2)
X-220	Torus Purge and Vent Exhaust	X		X (Note 3)	
X-221	RCIC Vacuum Pump Discharge	X			X (Note 8)
X-222	HPCI Turbine Drain	X			X (Note 8)
X-223A/B	CS Pump Min Flow Line	X			X (Note 8)
X-224	RCIC Torus Suction	X			X (Note [illegible])
X-225	RHR Pump Suction	X			X (Note [illegible])

[illegible]	[illegible]	[illegible]			
		Currently in Procedure 6.3.1.1	Need Added to Procedure 6.3.1.1	Need One Time LLRT	Not Required
X-226	HPCI Torus Suction	X			X (Note 8)
X-227A/B	CS Torus Suction	X			X (Note 8)
X-228	Existing Spare				X (Note 2)
X-229A/L	Vacuum Breaker Actuating Air		X		
X-229M	Existing Spare				X (Note 2)
X-230	Electrical	X			

NOTES

The Drywell to Torus Vent Lines are considered an extension of the containment boundary and are included in the Type A test.

Existing spare penetrations are considered as part of the containment liner and are included in the Type A test.

Instrumentation associated with these penetrations is considered an extension of the containment boundary and is included with the Type A test. Modifications to the instrumentation or isolation of the instrumentation during the Type A test will require a one time local leak rate test (LLRT).

These penetrations are designed in accordance with Safety Guide 11 (Regulatory Guide 1.11) and are included in the Type A test.

Modifications are being made to make these penetrations spares. A one time local leak rate test (LLRT) is required. Subsequently, these penetrations will be considered part of the containment liner and included in the Type A test.

The Cooper Nuclear Station Safety Evaluation, dated February 14, 1973, Section 6.2.3, states "Systems designed prior to the implementation of Appendix J, such as the control rod drive penetrations and standby liquid control system, do not have design provisions for individual leak tests; however, the normal functional testing of these systems ensure their operability and thence the necessary containment integrity."

These penetrations are entirely contained in the torus and do not represent a potential post-accident atmospheric release path.

These penetrations are currently in the local leak rate test (LLRT) program. However, these penetrations are water sealed by the torus and can be removed from the LLRT program. IST program requirements for these penetrations should be reviewed prior to removal from the LLRT program.

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GENERAL ELECTRIC
ATOMIC POWER EQUIPMENT DEPARTMENT

DESIGN SPECIFICATION

No. 22A1153, Rev. 1
File Index No. 0000/5A
F.P.C. Reference No.
Project STANDARD

22A1153, Rev. 1
CONT ON SHEET 2 SH NO. 1

TITLE CODES AND INDUSTRIAL STANDARD

1.0 SCOPE

1.1 This document specifies the Codes and Industrial Standards applicable to (a) the systems and items of equipment which make up the nuclear boiler system and (b) components related to the nuclear boiler system of a boiling water type nuclear power plant.

1.2 The Codes and Industrial Standards listed are those which apply. They may be supplemented to satisfy the requirements of the application. This supplementing information will be specified by design, purchasing or installation specifications issued by the General Electric Company.

1.3 Reference should be made to any drawings or specifications issued by the General Electric Company as being applicable to the specific project for the items included herein. Where differences exist between this standard design specification and the requirements of the above mentioned documents, the project documents shall be used.

1.4 This design specification was prepared based upon no specific division of work assignment or responsibility between the General Electric Company and its customers or contractors. The division of work assignment shall be specified elsewhere.

2.0 OBJECTIVE

The objective is to conform with the requirements of the various regulatory bodies having jurisdiction at the site of the nuclear power plant. It is a further objective to establish the Codes and Industrial Standards which best meet the level of quality and assurance consistent with the nature of the application of the system or item of equipment included herein.

3.0 REQUIREMENTS

3.1 Unless otherwise specified, the design of lines, systems and equipment and their subsequent fabrication and installation, shall be in accordance with the Standards and Industrial Codes specified in the Appendix.

3.2 Items of piping and equipment which are not within the scope of the nuclear boiler systems and not listed in the Appendix shall be designed, fabricated and installed in accordance with the standard codes which apply.

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GENERAL ELECTRIC
ATOMIC POWER EQUIPMENT DEPARTMENT

No. 22A1153, Rev. 1
File Index No. 0002/96
F.P.C. Reference No.
Project STANDARDS

DESIGN SPECIFICATION

22A1153, Rev. 1 CONT ON SHEET 3 SH NO. 2

TITLE CODES AND INDUSTRIAL STANDARD - APPENDIX

SCHEDULE A (SEE NOTE 5)

Item	Description	Codes and Standards
1	Reactor Pressure Vessel	ASME Section III, Class A
2	In-Core Ion Chamber Pressure Parts	ASME Section III, Class A
3.	Control Rod Drive Pressure Parts	ASME Section III, Class A
4	Control Rod Drive Hydraulic System	
	Pump Casing	ASME Section VIII, (See Note 2)
	Accumulators	ASME Section VIII
5	Reactor Water Recirculation System	
	Pump Casing	ASME Section III, Class C (See Note 2)
6	Residual Heat Removal System	
	Heat Exchangers	ASME Section III, Class C & TEMA, Class C
	Pump Casing	ASME Section III, Class C (See Note 2)
7	Standby Liquid Control System	
	Pump Casing	ASME Section III, Class C (See Note 2)
8	Core Spray System	
	Pump Casing	ASME Section III, Class C (See Note 2)
9	Reactor Core Isolation Cooling System (RCIC)	
	Pump Casing	ASME Section III, Class C (See Note 2)
	Turbine Casing	NEMA Standards for Mechanical Drive Steam Turbine

(Refer to Sheet 5 for Notes)

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TITLE CODES AND INDUSTRIAL STANDARDS - APPENDIX

SCHEDULE A (CONTINUED)

Item	Description	Codes and Standards
10	High Pressure Cooling Injection System (HPCI)	
	Pump Casing	ASME Section III, Class C (See Note 2)
	Turbine Casing	NEMA Standards for Mechanical Drive Steam Turbine
11	Reactor Water Clean-Up System	
	Regenerative Heat Exchangers	ASME Section III, Class C
	Non-Regenerative Heat Exchangers	
	Primary Side	ASME Section III, Class C
	Secondary Side (Cooling Water Side)	ASME Section VIII
	Pressurized Tanks (Filter-Demineralizers or Deep-Bed Demineralizers as applicable)	ASME Section III, Class C
	Filters (See Note 4)	ASME Section III, Class C
12	Primary Steam System	
	Safety Valves	ASME Section III, Article 9
13	Turbine	
	Turbine External Moisture Separator	ASME Section VIII
♠	Steam Packing Exhauster Condenser	Heat Exchanger Institute
14	Main Condenser	Heat Exchanger Institute
	Steam Jet Air Ejector	Heat Exchanger Institute
	Inter and After Condensers	Heat Exchanger Institute
	Off Gas Piping	ASA B31.1 & Code Case N-12 (See Note 1)

(Refer to Sheet 5 for Notes)

♠ Changed

TITLE CODES AND INDUSTRIAL STANDARD - APPENDIX

SCHEDULE A (CONTINUED)

Item	Description	Codes and Standards
15	Feedwater System	
	Heaters (including drain coolers)	ASME Section VIII & Feedwater Heaters Manufactures Association Standards
	Piping from the Reactor Pressure Vessel thru first Shut-Off Valve	ASA B31.1 and ASME Section I Paragraph PG-58 (3) (See Note 1)
16	Condensate Filter-Demineralizer	
	Pressurized Tanks	ASME Section VIII
	Filters (See Note 4)	ASME Section VIII
17	Closed Loop Cooling System - Reactor Building	
	Heat Exchanger	ASME Section VIII
18	Fuel Pool Cooling & Filtering Systems	
	Pump Casing	ASME Section VIII (See Note 2)
	Heat Exchanger	ASME Section VIII
19	Radioactive Waste Disposal System	
	Waste System Pressure Vessels	ASME Section VIII
20	Primary Containment	
	Containment Vessel (including dry-well, wetwell, and interconnecting piping)	ASME Section III, Class B
	Containment Vacuum Breakers	ASME Section III, Class B and ASA B31.1
	Containment Auxiliary Process Piping	ASA B31.1 (See Note 3)
	Containment Penetrations	ASME Section III, Class B

(Refer to Sheet 5 for Notes)

GENERAL ELECTRIC
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DESIGN SPECIFICATION

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Project STANDARD

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TITLE CODES AND INDUSTRIAL STANDARD - APPENDIX

SCHEDULE A (CONTINUED)

Item	Description	Codes and Standards
21	Piping (Unless otherwise noted such as items 15 and 20)	ASA B31.1 (See Note 1)
22	Filters (Except Item 11)	ASME Section VIII

NOTES

1 -American Standards Association Specification B31.1 shall be supplemented by General Electric Company design, purchasing or installation specifications where applicable, based upon the specific requirements for the specific system involved and/or the requirements of local agencies having jurisdiction at the plant site location.

2 -Pump casings shall be designed to requirements of specified code. They are not required to be stamped. (Pumps are classified as machinery and therefore, are outside the scope of the code).

3 -Piping, which is an integral part of the primary containment for isolation purposes, shall have at least the same quality and levels of assurance as the primary containment.

4 -Applicable for plants employing deep-bed demineralizers in lieu of Filter-Demineralizers (Powdex).

5 -SEE FIGURE 1 for illustration of systems and equipment included herein. See Paragraph 1.3.

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CONT. ON SHEET 6 SHT. NO. 5

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No. 22A1153, Rev. 1
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Project STANDARD

DESIGN SPECIFICATION

22A1153, Revision 1
CON'T ON SHEET FINAL SH. NO. 6

TITLE CODES AND INDUSTRIAL STANDARD - APPENDIX

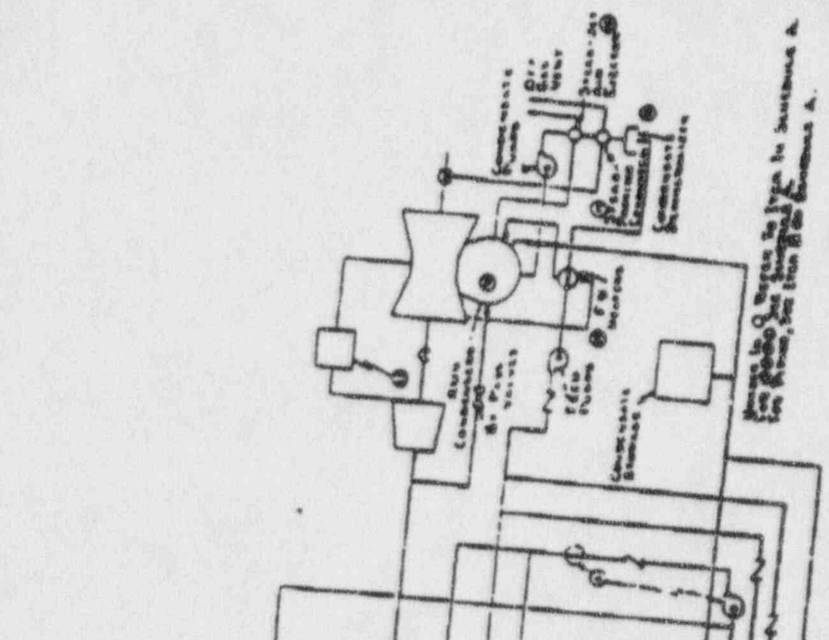

FIGURE I
(ILLUSTRATION OF CODE APPLICATION ONLY)

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CON'T. ON SHEET FINAL SH. NO. 6

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. Suppression Pool

At any time that the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the suppression pool water volume and temperature shall be maintained within the following limits except as specified in 3.7.A.2. and 3.5.F.5.

a. Minimum water volume - 87,650 ft^3

b. Maximum water volume - 91,100 ft^3

c. Maximum suppression pool temperature during normal power operation - 95°F.

d. During testing which adds heat to the suppression pool, the water temperature shall not exceed 10°F above the normal power operation limit specified in c. above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in c. above within 24 hours.

e. The reactor shall be scrammed from any operating condition if the pool temperature reaches 110°F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in c. above.

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

1. Suppression Pool

a. The suppression pool water level and temperature shall be checked once per day.

b. Whenever there is indication of relief valve operation or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

c. Whenever there is indication of relief valve operation with the temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

d. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

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LIMITING CONDITIONS FOR OPERATION

3.7.A.1 (cont'd)

f. During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cooldown rates if the pool temperature reaches 120°F.

2. Containment Integrity

a. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics tests at power levels not to exceed 5 MW(t).

b. When Coolant Temperature is above 212°F, the drywell and suppression chamber purge and vent system may be in operation for up to 90 hours per calendar year with the supply and exhaust 24-inch isolation valves in one supply line and one exhaust line open for containment inerting, deinerting, or pressure control.

If venting or purging is through Standby Gas for such operations, then both Standby Gas Treatment Systems shall be operable and only one Standby Gas Treatment System is to be used.

Not applicable to valves open during venting or purging provided such venting or purging utilizes the 2-inch bypass line(s) around the applicable inboard purge exhaust isolation valve(s) with the inboard valve(s) in closed condition.

SURVEILLANCE REQUIREMENTS

4.7.A (cont'd)

2. Leak Rate Testing

a. Integrated leak rate test (ILRT's) shall be performed to verify primary containment integrity. Primary containment integrity is confirmed if the leakage rate does not exceed the equivalent of 0.635 percent of the primary containment volume per 24 hours at 58 psig.

b. Integrated leak rate tests may be performed at either 58 psig or 29 psig, the leakage rate test period, extending to 24 hours of retained internal pressure. If it can be demonstrated to the satisfaction of those responsible for the acceptance of the containment structure that the leakage rate can be accurately determined during a shorter test period, the agreed-upon shorter period may be used.

Prior to initial operation, integrated leak rate tests must be performed at 58 and 29 psig (with the 29 psig test being performed prior to the 58 psig test) to establish the allowable leak rate, L_t (in percent of containment volume per 24 hours) at 29 psig as the lesser of the following values.

L_a (is 0.635 percent)

$$L_t = 0.635 \frac{L_{tm}}{L_{am}}$$

for $\frac{L_{tm}}{L_{am}} \leq 0.7$

where

L_{tm} = measured ILR at 29 psig

L_{am} = measured ILR at 58 psig, and

$\frac{L_{tm}}{L_{am}}$ [illegible]

[illegible]

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.7.A (cont'd.)

4.7.A.2.b. (cont'd.)

where

P_a = peak accident pressure, 58 psig

P_t = appropriately measured test pressures (psig)

for $\frac{L_{tm}}{L_{am}} > 0.7$

c. The ILRT's shall be performed at the following minimum frequency:

1. Prior to initial unit operation.

2. At approximately three and one-third year intervals so that any ten-year interval would include four ILRT's. These intervals may be extended up to eight months if necessary to coincide with refueling outage.

d. The measured leakage rates, L_{tm} and L_{am}, shall be less than 0.75 L_t and 0.75 L_a for the reduced pressure tests and peak pressure test respectively.

e. Except for the initial ILRT, all ILRT's shall be performed without any preliminary leak detection surveys and leak repairs immediately prior to the test. If an ILRT has to be terminated due to excessive leakage through identified leakage paths, the leakage through such paths shall be determined by a local leakage test and recorded. After repairs are made another ILRT shall be conducted.

If an ILRT is completed but the acceptance criteria of Specification 4.7.A.2.d is not satisfied and repairs are necessary, the ILRT need not be

3.7.A (Cont'd)

4.7.A.2.e (cont'd)

repeated provided locally measured leakage reductions, achieved by repairs, reduce the containment's overall measured leakage rate sufficiently to meet the acceptance criteria.

f. Local Leak Rate Tests

1. With the exceptions specified below, local leak rate tests (LLRT's) shall be performed on the primary containment testable penetrations and isolation valves at a pressure of 58 psig during each reactor shutdown for refueling, or other convenient intervals, but in no case at intervals greater than two years. The test duration of all valves and penetrations shall be of sufficient length to determine repeatable results. The total acceptable leakage for all valves and penetrations other than the MSIV's is 0.60 La.

2. Bolted double-gasket seals shall be tested after each opening and during each reactor shutdown for refueling, or other convenient intervals but in no case at intervals greater than two years.

3. The main steam isolation valves (MSIV's) shall be tested at a pressure of 29 psig. If a total leakage rate of 11.5 scf/hr for any one MSIV is exceeded, repairs and retest shall be performed to correct the condition. This is an exemption to Appendix J of 10CFR50.

3.7.A (Cont'd)

4.7.A.2.f (cont'd)

4. Main steam line and feedwater line expansion bellows shall be tested by pressurizing between the laminations of the bellows at a pressure of 5 psig. This is an exemption to Appendix J of 10CFR50.

5. The personnel airlock shall be tested at 58 psig at intervals no longer than six months. This testing may be extended to the next refueling outage (not to exceed 24 months) provided that there have been no airlock openings since the last successful test at 58 psig. In the event the personnel airlock is not opened between refueling outages, it shall be leak checked at 3 psig at intervals no longer than six months. Within three days of opening (or every three days during periods of frequent opening) when containment integrity is required, test the personnel airlock at 3 psig. This is an exemption to Appendix J of 10CFR50.

   The maximum allowable leakage at a test pressure of 58 psig is 12 scfh. Leakage measured at test pressure less than 58 psig is adjusted to the equivalent value at 58 psig.

g. Deleted

h. Drywell Surfaces

   The interior surfaces of the drywell and torus shall be visually inspected each operating cycle for evidence of torus corrosion or leakage.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.A (cont'd.)	4.7.A (cont'd.)
3. <u>Pressure Suppression Chamber - Reactor Building Vacuum Breakers</u>	3. <u>Pressure Suppression Chamber - Reactor Building Vacuum Breakers</u>
a. Except as specified in 3.7.A.3.b below, two pressure suppression chamber-reactor building vacuum breakers shall be operable at all times when primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the pressure suppression chamber-reactor building air actuated vacuum breakers shall be 0.5 psid. The self actuated vacuum breakers shall open fully when subjected to a force equivalent to 0.5 psid acting on the valve disc.	a. The pressure suppression chamber-reactor building vacuum breakers and associated instrumentation, including set points shall be checked for proper operation every three months.
b. From and after the date that one of the pressure suppression chamber-reactor building vacuum breakers is made or found to be inoperable for any reason, the vacuum breaker switch shall be secured in the closed position and reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is sooner made operable, provided that the repair procedure does not violate primary containment integrity.	b. During each refueling outage each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker does not exceed the force specified in Specifications 3.7.A.3.a and each vacuum breaker shall be inspected and verified to meet design requirements.
4. <u>Drywell-Pressure Suppression Chamber Vacuum Breakers</u>	4. <u>Drywell-Pressure Suppression Chamber Vacuum Breakers</u>
a. When primary containment is required, all drywell-suppression chamber vacuum breakers shall be operable at the 0.5 psid setpoint and positioned in the fully closed position as indicated by the position indicating system except during testing and except as specified in 3.7.A.4.b and .c below.	a. Each drywell-suppression chamber vacuum breaker shall be exercised through an opening-closing cycle every 30 days.
b. Three drywell-suppression chamber vacuum breakers may be determined to be inoperable for opening provided they are secured in the fully closed position or that the requirement of 3.7.A.4.c is demonstrated to be met.	b. When it is determined that a vacuum breaker valve is inoperable for opening at a time when operability is required all other vacuum breaker valves shall be exercised immediately and every 15 days thereafter until the inoperable valve has been returned to normal service.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.A.4 (cont'd.)	4.7.A.4 (cont'd.)
c. The total leakage between the dry-well and suppression chamber shall be less than the equivalent leakage through a 1" diameter orifice.	c. Once each operating cycle, each vacuum breaker valve shall be visually inspected to insure proper maintenance and cperation of the position indicati switch. The differential pressure set-point shall be verified.
d. If specifications 3.7.A.4.a, b or c, cannot be met, the situation shall be corrected within 24 hours or the reactor will be placed in a cold shutdown condition within the subsequent 24 hours.	d. Prior to reactor startup after each refueling, a leak test of the drywell to suppression chamber structure shall be conducted to demonstrate that the requirement of 3.7.A.4.c is met.
5. <u>Oxygen Concentration</u>	5. <u>Oxygen Concentration</u>
a. After completion of the startup test program and demonstration of plant electrical output, the primay containment atmosphere shall be reduced to less than 4% oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.5.b.	a. The primary containment oxygen concentration shall be measured and recorded at least twice weekly.
b. Within the 24-hour period subsequent to placing the reactor in the Run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. De-inerting may commence 24 hours prior to a shutdown.	b. The quantity of liquid nitrogen in the liquid nitrogen storage tank shall be determined twice per week when the volume requirements of 3.7.A.5.c are in effect.
c. When the containment atmosphere oxygen concentration is required to be less than 4%, the minimum quantity of liquid nitrogen in the liquid nitrogen storage tank shall be 500 gallons.	
d. If the specifications of 3.7.A.5.a thru c cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.	
e. The specifications of 3.7.A.5.a thru d are not applicable during a 48 hour continuous period between the dates of March 22, 1982 and March 25, 1982.	

LIMITING CONDITION FOR OPERATION

3.7.A (cont'd.)

6. Low-Low Set Relief Function

a. The low-low set function of the safety-relief valves shall be operable when there is irradiated fuel in the reactor vessel and the reactor coolant temperature is ≥ 212°F, except as specified in 3.7.A.6.a.1 and 2 below.

1. With the low-low function of one safety/relief valve (S/RV) inoperable, restore the inoperable LLS S/RV to OPERABLE within 14 days or be in the HOT STANDBY mode within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the low-low set function of both S/RVs inoperable, be in at least HOT STANDBY within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. The pressure switches which control the low-low set safety/relief valves shall have the following settings.

NBI-PS-51A Open Low Valve
1015 ± 20 psig (Increasing)

NBI-PS-51B Close Low Valve
875 ± 20 psig (Decreasing)

NBI-PS-51C Open High Valve
1025 ± 20 psig (Increasing)

NBI-PS-51D Close High Valve
875 ± 20 psig (Decreasing)

B. Standby Gas Treatment System

1. Except as specified in 3.7.B.3 below, both Standby Gas Treatment subsystems shall be operable at all times when secondary containment integrity is required.

2.a. The results of the in-place cold DOP leak tests on the HEPA filters shall show ≥99% DOP removal. The results of the halogenated hydrocarbon leak tests on the charcoal adsorbers shall show ≥99% halogenated hydrocarbon removal. The DOP and halogenated hydrocarbon tests shall be performed at a Standby Gas Treatment flowrate of ≤1780 CFM and at a Reactor Building pressure of ≤-.25" Wg.

SURVEILLANCE REQUIREMENT

4.7.A (cont'd.)

6. Low-Low Set Relief Function

a. The low-low set safety/relief valves shall be tested and calibrated as specified in Table 4.2.B.

B. Standby Gas Treatment System

1. At least once per operating cycle the following conditions shall be demonstrated.

a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at the system design flow rate.

b. Inlet heater input is capable of reducing R.H. from 100 to 70% R.H.

2.a. The tests and sample analysis of Specification 3.7.B.2 shall be performed at least once every 18 months for standby service or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.

3.7.B (cont'd)

b. The results of laboratory carbon sample analysis shall show ≥99% radioactive methyl iodide removal with inlet conditions of: velocity ≥27 FPM, ≥1.75 mg/m^3 inlet methyl iodide concentration, ≥70% R.H. and ≤30°C.

c. Each fan shall be shown to provide 1780 CMF ±10%.

3. From and after the date that one Standby Gas Treatment subsystem is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such subsystem is sooner made operable, provided that during such seven days all active components that affect operability of the operable Standby Gas Treatment subsystem, and its associated diesel generator, shall be operable.

Fuel handling requirements are specified in Specification 3.10.E.

4. If these conditions cannot be met, procedures shall be initiated immediately to establish reactor conditions for which the Standby Gas Treatment System is not required.

C. <u>Secondary Containment</u>

1. Secondary containment integrity shall be maintained during all modes of plant operation except when all of the following conditions are met.

4.7.B (cont'd)

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

d. Each subsystem shall be operated with the heaters on at least 10 hours every month.

e. Test sealing of gaskets for housing doors downstream of the HEPA filters and charcoal adsorbers shall be performed at, and in conformance with, each test performed for compliance with Specification 4.7.B.2.a and Specification 3.7.B.2.a.

3. System drains where present shall be inspected quarterly for adequate water level in loop-seals.

4.a. At least once per operating cycle automatic initiation of each Standby Gas Treatment subsystem shall be demonstrated.

b. At least once per operating cycle manual operability of the bypass valve for filter cooling shall be demonstrated.

c. When one Standby Gas Treatment subsystem becomes inoperable, the operable Standby Gas Treatment subsystem shall be verified to be operable immediately and daily thereafter. A demonstration of diesel generator operability is not required by this specification.

C. <u>Secondary Containment</u>

1. Secondary containment surveillance shall be performed as indicated below:

3.7.C (cont'd.)

a. The reactor is subcritical and Specification 3.3.A is met.

b. The reactor water temperature is below 212°F and the reactor coolant system is vented.

c. No activity is being performed which can reduce the shutdown margin below that specified in Specification 3.3.A.

d. No irradiated fuel is being handled in the secondary containment and no loads which could potentially damage irradiated fuel are being moved in the secondary containment.

e. If secondary containment integrity cannot be maintained, restore secondary containment integrity within 4 hours or:

a. Be in at least Hot Shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

b. Suspend irradiated fuel handling operations in the secondary containment, movement of loads which could potentially damage irradiated fuel in the secondary containment, and all core alterations and activities which could reduce the shutdown margin. The provisions of Specification 1.0.J are not applicable.

D. Primary Containment Isolation Valves

1. During reactor power operating conditions, all isolation valves listed in Table 3.7.1 and all instrument line flow check valves shall be operable except as specified in 3.7.D.2.

4.7.C (cont'd.)

a. A preoperational secondary containment capability test shall be conducted after isolating the reactor building and placing either Standby Gas Treatment subsystem filter train in operation. Such tests shall demonstrate the capability to maintain 1/4 inch of water vacuum under calm wind ($2<\bar{\mu}<5$ mph) conditions with a filter train flow rate of not more than 100% of building volume per day. ($\bar{\mu}$= wind speed)

b. Additional tests shall be performed during the first operating cycle under an adequate number of different environmental wind conditions to enable valid extrapolation of the test results.

c. Secondary containment capability to maintain 1/4 inch of water vacuum under calm wind ($2<\bar{\mu}<5$ mph) conditions with a filter train flow rate of not more than 100% of building volume per day, shall be demonstrated at each refueling outage prior to refueling.

d. After a secondary containment violation is determined, the Standby Gas Treatment System will be operated immediately after the affected zones are isolated from the remainder of the secondary containment to confirm its ability to maintain the remainder of the secondary containment at 1/4 inch of water negative pressure under calm wind conditions.

D. Primary Containment Isolation Valves

1. The primary containment isolation valves surveillance shall be performed as follows:

a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.

3.7.D (cont'd.)

2. In the event any isolation valve specified in Table 3.7.1 becomes inoperable, reactor power operation may continue provided at least one valve in each line having an inoperable valve shall be in the mode corresponding to the isolated condition.*

3. If Specification 3.7.D.1 and 3.7.D.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within 24 hours.

*Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative control.

4.7.D (cont'd.)

b. At least once per quarter:

(1) All normally open power operated isolation valves (except for the main steam line power-operated isolation valves) shall be fully closed and reopened.

(2) With the reactor power less than 75%, trip main steam isolation valves individually and verify closure time.

c. At least once per operating cycle the operability of the reactor coolant system instrument line flow check valves shall be verified.

d. At least once per operating cycle, while shutdown, the devices that limit the maximum opening angle to 60° shall be verified functional for the following valves: PC-230MV, PC-231MV, PC-232MV, and PC-233MV.

2. Whenever an isolation valve listed in Table 3.7.1 is inoperable, the position of at least one other valve in each line having an inoperable valve shall be recorded daily.

COOPER NUCLEAR STATION
TABLE 3.7.1 (Page 1)
PRIMARY CONTAINMENT ISOLATION VALVES

Valve & Steam	Number of Power Operated Valves Inboard	Number of Power Operated Valves Outboard	Maximum Operating Time (Sec) (1)	Normal Position (2)	Action On Initiating Signal (3)
Main Steam Isolation Valves					
MS-AO-80- A,B,C, & D	4		3 < T < 5	O	GC
MS-AO-86- A,B,C, & D		4	3 < T < 5	O	GC
Drywell Floor Drain Iso. Valves RW-AO-82, RW-AO-83		2	15	O	GC
Drywell Equipment Drain Iso. Valves RW-AO-94, RW-AO-95		2	15	O	GC
Main Steam Line Drain Valves MS-MO-74, MS-MO-77	1	1	30	O	GC
Reactor Water Sample Valves RR-740AV, RR-741AV	1	1	15	O	GC
Reactor Water Cleanup System Iso. Valves RWCU-MO-15, RWCU-MO-18	1	1	60	O	GC
RHR Suction Cooling Iso. Valve RHR-MO-17, RHR-MO-18	1	1	40	C	SC
RHR Discharge to Radwaste Iso. Valves RHR-MO-57, RHR-MO-67		2	20	C	SC
Suppression Chamber Purge & Vent PC-245AV, PC-230MV		2	15	C	SC
Suppression Chamber N_2 Supply PC-237AV, PC-233MV		2	15	C	SC

COOPER NUCLEAR STATION
TABLE 3.7.1 (Page 2)
PRIMARY CONTAINMENT ISOLATION VALVES

Valve & Steam	Number of Power Operated Valves Inboard	Outboard	Maximum Operating Time (Sec) (1)	Normal Position (2)	Action On Initiating Signal (3)
Primary Containment Purge & Vent PC-246AV, PC-231MV		2	15	C	SC
Primary Containment & N_2 Supply PC-238AV, PC-232MV		2	15	C	SC
Suppression Chamber Purge & Vent PC-230MV Bypass (PC-305MV)		1	40	C	SC(4)
Primary Containment Purge & Vent PC-231MV Bypass (PC-306MV)		1	40	C	SC(4)
Dilution Supply					
PC-1303MV, PC-1304MV		2	15	C	SC
PC-1305MV, PC-1306MV		2	15	C	SC
Dilution Supply					
PC-1301MV, PC-1302MV		2	15	O	GC
PC-1311MV, PC-1312MV		2	15	O	GC
Suppression Chamber Purge and Vent Exhaust PC-1308MV		1	15	C	SC
Primary Containment Purge and Vent Exhaust PC-1310MV		1	15	C	SC

NOTES FOR TABLE 3.7.1

1. Maximum valve operating times in seconds in the closed direction. This is the direction required for Primary Containment isolation.

2. Normal position indicates the normal valve position during power operations.

   O = Open
   C = Closed

3. Action on initiating signal indicates the valve operation after the signal initiation.

   GC = Goes Closed
   SC = Stays Closed

4. PC-305MV & PC-306MV have override switches (key operated) which can be used to open valves when isolation signals are in.

Pages [illegible] - [illegible]5 are not used

3.7 & 4.7 BASES

3.7.A & 4.7.A PRIMARY CONTAINMENT

3.7.A.1 & 4.7.A.1 SUPPRESSION POOL

The integrity of the primary containment and operation of the core standby cooling system, in combination, limit the off-site doses to values less than those suggeste in 10CFR100 in the event of a break in the primary system piping. Thus, containmen integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requiremen during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on th system at this time, thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep off-site doses well below 10CFR100 limits.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1035 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

As a result of the Mark I Containment Program, the District has completed the evaluation and requalification of the various containment structures and components at CNS. As a result of the requalification work, significant modifications were designed in accordance with the NRC acceptance crite.'a and installed. The Plant Unique Analysis Report, which was submitted on April 29,1982, and accepted on January 20, 1984, contains a detailed summary of the modifications installed. The maximum and minimum water volumes of 91,100 and 87,650 were not altered, but the downcomers were shortened by 1' 0½", so that their nominal submergence is now 3 feet and the initial volume of water in them is decreased proportionately. The acceptability of this is proven in "Mark I Containment Program Downcomer Submergence Functional Assessment Report", Task 6.6, NEDE - 21885-P, Class III, June, 1978.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for core standby cooling systems operability as explained in bases 3.5.F.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally change very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

The maximum suppression pool temperature of 95°F is based on not exceeding the 200°F Mark I temperature limit as contained in NUREG-0661. This 95°F limit also prevents exceeding LOCA considerations, or ECCS pump NPSH requirements. The basis for these limits are contained in NEDC-24360-P.

3.7.A.2 & 4.7.A.2 CONTAINMENT INTEGRITY

The maximum allowable test leak rate is 0.635%/day at a pressure of 58 psig, the peak calculated accident pressure. Experience has shown that there is negligible difference between the leakage rates of air at normal temperature and a steam-hot air mixture.

Establishing the test limit of 0.635%/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak-tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as built" condition is achieved by establishing the allowable operational leak rate. The allowable operational leak rate is derived by multiplying the maximum allowable leak rate, La, or the allowable test leak rate, Lt, by 0.75 thereby providing a 25% margin to allow for leakage deterioration which may occur during the period between leak rate tests.

The primary containment leak rate test frequency is based on maintaining adequate assurance that the leak rate remains within the specification. The leak rate test frequency is based on the NRC guide for developing leak rate testing and surveillance of reactor containment vessels. Allowing the test intervals to be extended up to 8 months permits some flexibility needed to have the tests coincide with scheduled or unscheduled shutdown periods.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a bolted double-gasketed penetration is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized.

Certain isolation valves are tested by pressurizing the volume between the inboard and outboard isolation valves. This results in conservative test results since the inboard valve, if a globe valve, will be tested such that the test pressure is tending to lift the globe off its seat. Additionally, the measured leak rate for such a test is conservatively assigned to both of the valves equally and not divided between the two.

The main steam and feedwater testable penetrations consist of a double layered metal bellows. The inboard high pressure side of the bellows is subjected to drywell pressure. Therefore, the bellows is tested in its entirety when the drywell is tested. The bellows layers are tested for the integrity of both layers by pressurizing the void between the layers to 5 psig. Any higher pressure could cause permanent deformation, damage and possible ruptures of the bellows.

Surveillance requirements for integrity of the personnel air lock are specified in Enclosure 1 (Exemption) to the letter, D. G. Eisenhut to J. M. Pilant, September 3, 1982. When the Personnel Air Lock Leakage Test is performed at a test pressure less than 58 psig, the measured leakage must be adjusted to reflect the expected leakage at 58 psig. Equation A-3 of Enclosure 3 (Franklin Research Center Technical Evaluation Report) to the letter, D. G. Eisenhut to J. M. Pilant, September 3, 1982, defines the method of adjustment.

The primary containment pre-operational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The peak drywell pressure would be about 58 psig which would rapidly reduce to 29 psig following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig, equalizes with drywell pressure and therefore rapidly decays with the drywell pressure decay. The design pressure of the drywell and suppression chamber is 56 psig. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressure was chosen. Also, based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 0.635%/day at 58 psig. Calculations made by the NRC staff with leak rate and a standby gas treatment system filter efficiency of 90% for halogens and assuming the fission product release fractions stated in NRC Regulatory Guide 1.3, show that the maximum total whole body passing cloud dose is about 1.0 REM and the maximum total thyroid dose is about 12 REM at 1100 meters from the stack over an exposure duration of two hours. The resultant doses reported are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines.

The water in the suppression chamber is used for cooling in the event of an accident, i.e., it is not used for normal operation; therefore, a daily trends. Whenever a bolted double-gasketed penetration is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized.

Table 3.7.4 identifies certain isolation valves that are tested by pressurizing the volume between the inboard and outboard isolation valves. This results in conservative test results since the inboard valve, if a globe valve, will be tested such that the test pressure is tending to lift the globe off its seat. Additionally, the measured leak rate for such a test is conservatively assigned to both of the valves equally and not divided between the two.

The main steam and feedwater testable penetrations consist of a double layered metal bellows. The inboard high pressure side of the bellows is subjected to drywell pressure. Therefore, the bellows is tested in its entirety when the drywell is tested. The bellows layers are tested for the integrity of both layers by pressurizing the void between the layers to 5 psig. Any higher pressure could cause permanent deformation, damage and possible ruptures of the bellows.

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The primary containment pre-operational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The peak drywell pressure would be about 58 psig which would rapidly reduce to 29 psig following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig, equalizes with drywell pressure and therefore rapidly decays with the drywell pressure decay. The design pressure of the drywell and suppression chamber is 56 psig. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressure was chosen. Also, based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 0.635%/day at 58 psig. Calculations made by the NRC staff with leak rate and a standby gas treatment system filter efficiency of 90% for halogens and assuming the fission product release fractions stated in NRC Regulatory Guide 1.3, show that the maximum total whole body passing cloud dose is about 1.0 REM and the maximum total thyroid dose is about 12 REM at 1100 meters from the stack over an exposure duration of two hours. The resultant doses reported are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines.

The water in the suppression chamber is used for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, approximately once per year, assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

The intent of Specification 3.7.A.2.b is to reduce the probability of a LOCA occurrence when the 24-inch purge and vent valves are open in series. These valves are normally closed during power operation to minimize reliance on the valve operators to ensure containment integrity. The requirements for Standby Gas is due to the damage the filters would experience from excessive difference pressure caused by a LOCA with the 24-inch exhaust valves open in series from the drywell or suppression chamber. This specification does allow venting with the inboard exhaust bypass valve and the outboard exhaust valve both open in series and the time does not count against the yearly limit. The NRC has accepted the determination that due to the small size of the bypass valve, there is no chance of damage to the filters if a LOCA occurs while venting the containment through the bypass with a SBGT system on line. The term "calendar year" is a period of time beginning on January 1 and ending on December 31 for each numbered year.

3.7.A.3 & 4 and 4.7.A.3 & 4 VACUUM BREAKERS

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and reactor building so that the structural integrity of the containment is maintained. The vacuum relief system from the pressure suppression chamber to reactor building consists of two 100% vacuum relief breakers (2 parallel sets of 2 valves in series). Operation of either system will maintain a pressure differential of less than 2 psi, the external design pressure. One valve may be out of service for repairs for a period of 7 days. If repairs cannot be completed within 7 days the reactor coolant system is brought to a condition where vacuum relief is no longer required.

The capacity of the 12 drywell vacuum relief valves are sized to limit the pressure differential between the suppression chamber and drywell during post-accident dry-well cooling operations to well under the design limit of 2 psi. They are sized on the basis of the Bodega Bay pressure suppression system tests. The ASME Boiler and Pressure Vessel Code, Section III, Subsection B, for this vessel allows a 2 psi differential; therefore, with three vacuum relief valves secured in the closed position and 9 operable valves, containment integrity is not impaired.

3.7.A.5 and 4.7.A.5 OXYGEN CONCENTRATION

Safety Guide 7 assumptions for Metal-Water reaction result in hydrogen concentration in excess of the Safety Guide 7 flammability limit. By keeping the oxygen concentration less than 4% by volume the requirements of Safety Guide 7 are satisfied.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss-of-coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended period of time with significant leaks in the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration.

3.7.A & 4.7.A BASES (cont'd)

The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least twice a week the oxygen concentration will be determined as added assurance.

The 500 gallon conservative limit on the nitrogen storage tank assures that adequate time is available to get the tank refilled assuming normal plant operation. The estimated maximum makeup rate is 1500 SCFD which would require about 160 gallons for a 10 day makeup requirement. The normal leak rate should be about 200 SCFD.

3.7.A.6 & 4.7.A.6 LOW-LOW SET RELIEF FUNCTION

The low-low set relief logic is an automatic safety relief valve (SRV) control system designed to mitigate the postulated thrust load concern of subsequent actuations of SRV's during certain transients (such as inadvertent MSIV closure) and small and intermediate break loss-of-coolant accident (LOCA) events. The setpoints used in Section 3.7.A.6.b are based upon a minimum blowdown range to provide adequate time between valve actuations to allow the SRV discharge line high water leg to clear, coupled with consideration of instrument inaccuracy and the main steam isolation valve isolation setpoint.

The as-found setpoint for NBI-PS-51A, the pressure switch controlling the opening of RV-71D, must be ≤ 1050 psig. The as-found closing setpoint for NBI-PS-51B must be at least 90 psig less than 51A, and must be ≥ 850 psig. The as-found setpoint for NBI-PS-51C, pressure switch controlling the opening of RV-71F must be ≤ 1050 psig. The as-found closing setpoint for NBI-PS-51D must be at least 90 psig below 51C, and must be ≥ 850 psig. This ensures that the analytical upper limit for the opening setpoint (1050 psig), the analytical lower limit on the closing setpoint (850 psig) and the analytical limit on the blowdown range (≥ 90 psig) for the Low-Low Set Relief Function are not exceeded. Although the specified instrument setpoint tolerance is ± 20 psig, an instrument drift of ± 25 psig was used in the analysis to ensure adequate margin in determining the valve opening and closing setpoints. The opening setpoint is set such that, if both the lowest set non-LLS S/RV and the highest set of the two LLS S/RVs drift 25 psig in the worst case directions, the LLS S/RVs will still control subsequent S/RV actuations. Likewise, the closing setpoint is set to ensure the LLS S/RV closing setpoint remains above the MSIV low pressure trip. The 90 psig blowdown provides adequate energy release from the vessel to ensure time for the water leg to clear between subsequent S/RV actuations.

3.7.B & 3.7.C STANDBY GAS TREATMENT SYSTEM AND SECONDARY CONTAINMENT

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation when the drywell is sealed and in service. The reactor building provides primary containment when the reactor is shut down and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system secondary containment is required at all times that primary containment is required as well as during refueling, and during movement of loads which could potentially damage irradiated fuel in the secondary containment. Secondary containment may be broken for short periods of time to allow access to the reactor building roof to perform necessary inspections and maintenance.

The Standby Gas Treatment System consists of two, distinct subsystems, each containing one exhaust fan and associated filter train, which is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions. Both Standby Gas Treatment System fans are designed to automatically start upon containment isolation and to maintain the reactor building pressure to the design negative pressure so that all leakage should be in-leakage. Should one subsystem fail to start, the redundant subsystem is designed to start automatically. Each of the two fans has 100 percent capacity.

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3.7.B & 3.7.C BASES (cont'd)

High efficiency particulate absolute (HEPA) filters are installed before and after the charcoal adsorbers to minimize potential release of particulates to the environment and to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and HEPA filters. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 99 percent for expected accident conditions. If the performance of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10 CFR 100 guidelines for the accidents analyzed.

Only one of the two Standby Gas Treatment subsystems is needed to cleanup the reactor building atmosphere upon containment isolation. If one subsystem is found to be inoperable, there is no immediate threat to the containment system performance and reactor operation or refueling operation may continue while repairs are being made. If both subsystems are inoperable, the plant is brought to a condition where the Standby Gas Treatment System is not required.

4.7.B & 4.7.C BASES

Standby Gas Treatment System and Secondary Containment

Initiati[illegible] reactor building isolation and operation of the Standby Gas Treatment System t[illegible] maintain at least a 1/4 inch of water vacuum within the secondary containment provides an adequate test of the operation of the reactor building isolation valves, leak tightness of the reactor building and performance of the Standby Gas Treatment System. Functionally testing the initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Periodic testing gives sufficient confidence of reactor building integrity and Standby Gas Treatment System performance capability.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A 7.8 kw heater is capable of maintaining relative humidity below 70%. Heater capacity and pressure drop should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant shall be performed in accordance with ANSI N510-1980. The test canisters that are installed with the adsorber trays should be used for the charcoal adsorber efficiency test. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. If test results are unacceptable, all adsorbent in the system shall be replaced

4.7.B & 4.7.C BASES

with an adsorbent qualified according to Table 5.1 of ANSI N509-1980. The replacement tray for the adsorber tray removed for the test should meet the same adsorbent quality. Tests of the HEPA filters with DOP aerosol shall be performed in accordance to ANSI N510-1980. Any filters found defective shall be replaced with filters qualified pursuant to Regulatory Position C.3.d. of Regulatory Guide 1.52, Revision 2, March, 1978.

All elements of the heater should be demonstrated to be functional and operable during the test of heater capacity. Operation of the heaters will prevent moisture buildup in the filters and adsorber system.

With doors closed and fan in operation, DOP aerosol shall be sprayed externally along the full linear periphery of each respective door to check the gasket seal. Any detection of DOP in the fan exhaust shall be considered an unacceptable test result and the gaskets repaired and test repeated.

If system drains are present in the filter/adsorber banks, loop-seals must be used with adequate water level to prevent by-pass leakage from the banks.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significance shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

Demonstration of the automatic initiation capability and operability of filter cooling is necessary to assure system performance capability. If one Standby Gas Treatment subsystem is inoperable, the operable subsystem's operability is verified daily. This substantiates the availability of the operable subsystem and thus reactor operation or refueling operation can continue for a limited period of time.

3.7.D & 4.7.D BASES

Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident.

The maximum closure times for the automatic isolation valves of the primary containment and reactor vessel isolation control system have been selected in consideration of the design intent to prevent core uncovering following pipe breaks outside the primary containment and the need to contain released fission products following pipe breaks inside the primary containment.

The USAR identifies those testable primary containment valves that perform an isolation function, and testable penetrations with Double O-Ring Seals, and testable penetrations with testable Bellows ensuring that any changes thereto receive a 10CFR50.59 review. In addition, plant procedures also identify containment isolation valves, and testable penetrations with Double O-Ring Seals, and testable penetrations with testable Bellows changes to these procedures and the USAR are controlled by Technical Specification 6.2.1.A.4 (Administrative Controls).

These valves are highly reliable, have a low service requirement, and most are normally closed. The initiating sensors and associated trip channels are also checked to demonstrate the capability for automatic isolation. The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve operability results in a greater assurance that the valve will be operable when needed.

In order to assure that the doses that may result from a steam line break do not exceed the 10CFR100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds. The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25 inch restricting orifice inside the primary containment and an excess flow check valve outside the primary containment. A program for periodic testing and examination of the excess flow check valves is performed as follows:

1. Vessel at pressure sufficient to actuate valves. This could be at time of vessel hydro following a refueling outage.

2. Isolate sensing line from its instrument at the instrument manifold.

3. Provide means for observing and collecting the instrument drain or vent valve flow.

4. Open vent or drain valve.

    a. Observe flow cessation and any leakage rate.

    b. Reset valve after test completion.

5. The head seal leak detection line cannot be tested in this manner. This valve will not be exposed to primary system pressure except under unlikely conditions of seal failure where it could be partially pressurized to reactor pressure. Any leakage path is restricted at the source and therefore this valve need not be tested. This valve is in a sensing line that is not safety related.

6. Valves will be accepted if a marked decrease in flow rate is observed and the leakage rate is acceptable.

The operators for containment vent/purge valves PC-230MV, PC-231MV, PC-232MV, and PC-233MV have devices in place to limit the maximum opening angle to 60 degrees. This has been done to ensure these valves are able to close against the maximum differential pressure expected to occur during a design basis LOCA.

APPENDIX A
PRESSURE INTEGRITY OF PIPING AND EQUIPMENT PRESSURE PARTS

1.0 SCOPE

This appendix provides additional information pertinent to the preceding sections concerning the pressure integrity of piping and equipment parts.

Piping and equipment pressure parts are classified according to service and location. The design, fabrication, inspection, and testing requirements which are defined for the equipment of each classification assure the proper pressure integrity. This Appendix describes the requirements in effect at the time of the original installation of the piping and equipment pressure parts. The evolution of industry codes and standards, regulatory requirements, fabrication, testing, and erection procedures; and supplementary requirements has resulted in parts of these requirements being superseded. The new requirements generally result in an improvement in quality and overall margins over the original requirement. Upgrades or replacement of piping and equipment pressure parts are performed to these new requirements provided the safety design bases described in the USAR are maintained.

For the purpose of this appendix, the pressure boundary of the process fluid includes but is not necessarily limited to: branch outlet nozzles or nipples, instrument wells, reservoirs, pump casing closures, blind flanges and similar pressure closures, studs, nuts and fasteners in flanged joints between pressure parts and bodies and pressure parts of in-line components such as traps and strainers.

Specifically excluded from the scope of this appendix are pressure parts such as vessels and heat exchangers or any components which are within the scope of the ASME Pressure Vessel Code, Section III and VIII; and nonpressure parts such as pump motors, shafts, seals, impellers, wear rings, valve stems, gland followers, seat rings, guides, yokes, and operators; any nonmetallic material such as packing and gaskets; fasteners not in pressure part joints such as yoke studs and gland follower studs; and washers of any kind.

1.1 Codes and Specifications

The piping and equipment pressure parts in this station are designed, fabricated, inspected, and tested in accordance with recognized industrial codes and specifications. In some cases supplementary requirements are applied to increase safety and operational reliability. The application of the industrial codes and specifications is defined in this appendix as well as the application of the supplementary requirements. Where conflicts occur between the industrial codes and specifications and the supplementary requirements, the supplementary requirements take precedence.

United States of America Standards (USAS) referenced herein have been superceded by ANSI standards. The edition of the USA standards in effect when bids were made for supplying and installing piping was:

USAS-B31.1.0 - Power Piping (1967)
USAS-B31.7 - Nuclear Power Piping (Feb. 1968) w/
Draft and Errata (June 1968)

2.0 CLASSIFICATION OF PIPING AND EQUIPMENT PRESSURE PARTS

For the purpose of identification and association of requirements, piping and equipment pressure parts are classified in accordance with one of two basic principles.

2.1 GE Company Classification and Pressure Integrity Requirements

Class A — Piping and equipment pressure parts which cannot be isolated from the reactor vessel.

Class B — Piping and equipment pressure parts, which can be isolated from the reactor vessel by only a single isolation valve.

Class C — Piping and equipment pressure parts other than included in Classes A and B, for a high integrity system.

Class D — Piping and equipment pressure parts which serve as an extension of containment and which operate at either pressures greater than 150 psig or temperatures greater than 212°F.

Class E — Piping and equipment pressure parts which serve as an extension of containment and which operate at pressures equal to or less than 150 psig or temperatures equal to or less than 212°F.

Class F — Piping and equipment pressure parts which transport fibrous or particulate materials such as resins or filter aids and which operate at pressures equal to or less than 150 psig and temperatures equal to or less than 212°F.

Class G — Piping and equipment pressure parts used for acids in concentrations of 60 to 100 percent at ambient temperatures or caustics in concentrations of 50 percent or less at temperatures less than 150°F.

Class H — Piping and equipment pressure parts used for acids in concentrations of 10 percent or less.

Class L — Piping and equipment pressure parts which require materials considerations to maintain deionized water purity.

Class M — Power piping and equipment pressure parts not otherwise classified and which are considered within the scope of USAS B31.1.0, Code for Power Piping.

Class N — Miscellaneous piping and equipment not otherwise classified and not considered within the scope of USAS B31.1.0, Code for Power Piping.

2.2 Engineer - Constructor's Classification and Definition of Piping and In-Line Pressure Parts

For this project, all piping systems or subsystems and all in-line pressure parts are functionally classified as IN, IIN, IIIN, or IVP, and seismically classified as IS or IIS.

2.2.1 Functional Piping and Equipment Pressure Part Classifications

1. Class IN nuclear piping and in-line pressure parts are those, whose loss or failure could cause or increase the severity of a nuclear incident.

2. Class IIN nuclear piping and in-line pressure parts are those, whose loss or failure could cause a hazard to plant personnel, but would represent no hazard to the public.

3. Class IIIN nuclear piping and in-line pressure parts, are those that normally would be Class IIN, except that the operating pressure does not exceed 150 psig and the operating temperature is below 212°F.

4. Class IVP power piping and in-line pressure parts are those, which are conventional steam and service piping and equipment pressure parts.

2.2.2 Seismic Piping Classifications

1. Class IS seismic piping and in-line pressure parts are those, whose failure would cause significant release of radioactivity or which are vital to a safe shutdown of the plant and removal of decay and sensible heat.

2. Class IIS seismic piping and in-line pressure parts are those, which may be essential to the operation of the station, but which are not essential to a safe shutdown.

2.3 Tabulation of Classification Equivalencies

Classification in Accordance with Definitions of:	
GE Company	Engineer-Constructor
A and B	IN/IS
C and D	IIN/IS and IIN/IIS
E and F	IIIN/IS and IIIN/IIS
F,G,H,L,M and N	IVP/IS and IVP/IIS

2.4 Engineer-Constructor's Classification and Definition of Equipment

Equipment is classified by seismic requirements as follows:

1. Class I equipment is that whose failure would cause significant release of radioactivity or which is vital to a safe shutdown of the plant and removal of decay and sensible heat.

2. Class II equipment is that which may be essential to the operation of the station, but which is not essential to a safe shutdown.

3.0 DESIGN REQUIREMENTS

3.1 Piping Design

All piping is designed in accordance with USAS B31.1.0,"Power Piping". Class IN/IS piping is also designed to meet the requirements of Appendix C which outlines loading criteria to be met for high reliability for piping designed to rational stress analysis techniques. All other Class IS piping is designed to meet the supplementary requirements included in this Appendix, Subsection A-3.1.1. The terms utilized in this Subsection A-3.1 are either defined in the text, or pertain to definitions of USAS B31.1.0. Selection of design earthquakes is discussed in Appendix A of the Cooper Nuclear Station PSAR.

3.1.1 Analysis

3.1.1.1 Primary Stresses (Sp)

Primary stresses are as follows:

1. Circumferential Primary Stress (S_R)
Circumferential primary stresses are below the allowable stress (S_h) at the design pressure and temperature.

2. Longitudinal Primary Stresses (S_L)
The following loads are considered as producing longitudinal primary stresses: internal or external pressures; weight loads including valves, insulation, fluids, and equipment; hanger loads; static external loads and reactions; and the inertia load portion of seismic loads.

When the seismic load is due to the maximum probable earthquake (0.1g), the vectorial combination of all longitudinal primary stresses (S_L) does not exceed 1.2 times the allowable stress (S_h).

When the seismic load is due to the hypothetical maximum possible earthquake (0.20g), the vectorial combination of all longitudinal primary stresses does not exceed 1.8 times the allowable stress (S_h).

3.1.1.2 Secondary Stresses (S_E)

Secondary stresses are determined by use of the maximum shearing stress theory.

$$T\ Max = 1/2 \sqrt{S_b^2 + 4S_t^2} = 1/2\ S_E,$$

where,

$$S_E = \sqrt{S_b^2 + 4S_t^2}$$

(See USAS B31.1.0)

The following loads are considered in determining longitudinal secondary stresses: (a) thermal expansion of piping, (b) movement of attachments due to thermal expansion, (c) forces applied by other piping systems as a result of their expansion, (d) any variations in pipe hanger loads resulting from expansion of the system.

5.0 FABRICATION AND INSTALLATION REQUIREMENTS

Fabrication and erection of piping and equipment pressure parts are in accordance with USAS B31.1.0, "Power Piping", and the supplementary requirements in schedules FIN, FIIN, FIIIN, and FIVP included herein. These schedules are applied as follows:

Piping and Equipment Pressure Parts Classification	Fabrication and Erection Schedules
IN	FIN
IIN	FIIN
IIIN	FIIIN
IVP	FIVP

6.0 TESTING AND INSPECTION REQUIREMENTS

Testing and inspection of piping and equipment pressure parts are in accordance with USAS B31.1.0, "Power Piping" and the supplementary requirements in schedules TIN, TIIN, TIIIN, and TIVP included herein. These schedules are applied as follows:

Piping and Equipment Pressure Parts Classification	Inspection and Test Schedule
IN	TIN
IIN	TIIN
IIIN	TIIIN
IVP	TIVP

6.1 Methods, Techniques and Acceptance Standards

6.1.1 Radiography

6.1.1.1 Welds

The radiography of welds, including acceptability standards, are in accordance with the following:

Classification	Criteria & Acceptance Standards
IN & IIN	ASME Boiler and Pressure Vessel Code, Section III, Paragraph N-624
IIIN & IVP	ASME B&PV Code, Section I, para. PW-51 and Section VIII, para. VW-51 (a through k).

6.1.1.2 Castings

Methods and Techniques

The radiography of castings employ methods and techniques in accordance with ASTM E94, "Tentative Recommended Practices for Radiographic Testing", to the quality level in accordance with ASTM E142, "Standard Method for Controlling Quality of Radiographic Testing".

Acceptance Standards

Discontinuities are judged by comparison with ASTM E71, E186, and E280 as appropriate for section thickness. Discontinuity types A through C of severity level 2 are acceptable; discontinuity types beyond C are not acceptable.

6.1.2 Ultrasonic Testing

Ultrasonic examination of forgings in Class IN and IIN systems is done in accordance with the following:

6.1.2.1 Ultrasonic Examination

Ultrasonic examination of pipe, plate and forgings shall be performed, and acceptance standards shall comply with the following applicable specifications:

(a) Pipe, (Seamless) ASTM E213. Ultrasonic inspection of pipe and tubing for longitudinal discontinuities.

(b) Pipe Welded Without Filler Metal, ASTM E273. Ultrasonic inspection of longitudinal and spiral welds of welded pipe and tubing.

(c) Forgings, Bars, Bolting Materials and Plate, ASTM A388. Ultrasonic testing and inspection of heavy steel forging. In examination of plate or bars where the words "forging" or "forgings" appear they are considered to mean plate or bar material.

6.1.2.2 Normal Beam Examination General Acceptance Standards

The materials shall be considered unacceptable based on the following test indications unless eliminated or repaired:

(a) Indications of discontinuities in the material that produce a complete loss of back reflection not associated with the geometric configuration of the piece. (Complete loss in back reflection is assumed when the back reflection falls below 5 percent of full screen height.)

(b) Traveling indications of discontinuities 10 percent or more of the back reflection. (A traveling indication is defined as an indication which displays sweep movement of the oscilloscope pattern at a relatively constant amplitude as the search unit is moved along the part being examined.)

6.1.3 Liquid Penetrant Testing

Methods, techniques and acceptance standards for liquid penetrant testing are in accordance with the following:

Classification	Criteria & Acceptance Standards
IN, IIN, IIIN	1965 w/ Addenda thru Winter 1967 ASME - Section III, Paragraph N-627 or ASME B&PV Code

6.1.4 Magnetic Particle Testing

Methods, techniques and acceptance standards for magnetic particle testing are in accordance with the following:

Classification	Criteria & Acceptance Standards
IN, IIN, IIIN	ASME Section III, Paragraph N-626, Paragraph 1-724 for pipe and fittings.
IVP	ASME B&PV Code, Section VIII, Appendix VI on MS-1, RF-1 systems and 20% random testing on IS (seismic) portion of RCC-1 system.

6.1.5 Hydrostatic Testing

Hydrostatic tests of piping and equipment pressure parts are conducted in accordance with the following:

Classification	Criteria & Acceptance Standards
IN, IIN IIIN, IVP	USAS B31.1.0 and the applicable sections of other published piping codes referenced in ASME Section III and applicable to nuclear power piping. USAS B31.1.0, "Section 137".

6.2 Personnel Qualification Requirements

(Pressure containing components in General Electric BWR System Classifications A, B, C, D, E, and F.) The manufacturer of pressure containing components shall be responsible to ensure that personnel who perform nondestructive examinations of pressure containing components meet the qualification requirements of Appendix IX, Paragraph IX-325, Section III of the ASME Boiler and Pressure Vessel Code. This shall apply to both the manufacturer's own employees and those of his subvendors.

8.0 FABRICATION AND ERECTION SCHEDULE FIN & FIIN

Paragraphs apply to both Schedule FIN and FIIN unless noted otherwise:

8.1 Welding

Welding of piping and equipment pressure parts is accomplished according to the following requirements:

8.1.1 Qualification

All welding, including fillet, seal, repair, and attachment welds, is performed in accordance with written welding procedures. Procedure qualification and welder performance qualification are in accordance with Section IX of the ASME Boiler and Pressure Vessel Code.

8.1.2 Qualification Records

Qualification records and application of welder's identification symbols are in accordance with Section 127.6 of USAS B31.1.0.

8.1.3 Butt Joints

Joint design and welding procedures for longitudinal and girth butt joints larger than 2 inches in nominal pipe size are in accordance with General Electric Dwg. 209A4280.

8.1.4 Branch Connections

Branch connections are made using fittings to USAS B16.9.

8.1.5 Socket Welds

Socket welds are employed for nominal pipe size 2 inches and smaller and are in accordance with USAS B31.1.0, Paragraph 127.4.4.

8.1.6 Attachment Welds

Attachment of nonpressure-containing parts (such as supports and hangers) to pressure-containing components shall be by full penetration welds with inspection, heat treatment and welding per requirements for butt welds.

8.1.7 Fabrication Reinforcement for Openings

Reinforcement is in accordance with the requirements of the applicable sections of published piping codes referenced in ASME Section III applicable to nuclear piping systems.

8.1.8 Welding Procedures and Processes(1)

(1) See Subsection A-8.8.1 on specific limitations on welding austenitic stainless steel.

1. Welding procedures
2. Repair procedures
3. Heat treatment procedures
4. Cleaning procedures
5. Quality Assurance Control Plan (as specified in Appendix D)

8.9 Inspection and Testing

Inspection and testing of piping and equipment pressure parts, including completed welds, assemblies, and subassemblies, is performed as shown in the applicable schedule for the specific classification of piping and equipment pressure parts (see Subsection A-6.0).

13.0 INSPECTION AND TESTING SCHEDULE TIN

Refer to Subsection A-6.0 for application of this schedule and for test methods, techniques, and acceptance standards.

13.1 Certification

The manufacturer of the materials or components certifies that the requirements for which he is responsible, including those of this appendix as well as those of the specific material specification, are fully satisfied.

13.2 Hydrostatic Tests

Piping and equipment pressure parts are hydrostatically tested. If any repairs are made, the piping or equipment pressure part is retested. If any omissions or modifications of the test requirement are made, the deviation is shown valid before approval.

13.3 Nondestructive Testing

13.3.1 Welds

Girth and longitudinal pressure containing complete penetration groove butt welds are 100% examined by radiography. Accessible surfaces of the weld and adjacent base metal are examined by either liquid penetrant or magnetic particle methods.

Fillet welds, socket welds, and nonpressure containing attachment welds such as supports, lugs, anchors, and guides are examined on all accessible surfaces by either liquid penetrant or magnetic particle methods. Radiography is not required.

Welds attaching branch connections larger than 4 inches in pipe size are 100% examined by radiography, and accessible surfaces of the weld and adjacent base metal are examined by either liquid penetrant or magnetic particle methods. Welds attaching branch connections 4 inches and smaller are examined by either liquid penetrant or magnetic particle methods on the accessible surfaces of the weld and adjacent base metal.

Ultrasonic examination is performed whenever required in accordance with Subsection A-6.1.2.

13.3.2 Double-Welded Joints

The back of the first side welded shall be ground or chipped to sound metal and visually inspected prior to welding the second side.

13.3.3 Castings

Castings for pressure containing components larger than 4 inches are 100% examined by radiography and all accessible surfaces, including machined surfaces and castings 4 inches and smaller are examined by either the magnetic particle or the liquid penetrant method.

13.3.4 Forgings

Forgings for pressure containing components over 4 inches nominal diameter are examined in the finished condition by ultrasonic inspection; components 4 inches and smaller on all accessible surfaces including machined surfaces, by either the liquid penetrant or the magnetic particle method.

13.4 Submittals

Approval is required for the following inspection and test procedures:

1. Radiography
2. Ultrasonic testing
3. Liquid penetrant testing
4. Magnetic particle testing

14.0 INSPECTION AND TESTING SCHEDULE TIIN

Refer to Subsection A-6.0 for application of this schedule and for test methods, techniques and acceptance standards.

14.1 Certification

The manufacturer of the materials or components certifies that the requirements for which he is responsible including those included in this appendix as well as those of the specific material specification, are fully satisfied.

14.2 Hydrostatic Tests

Piping and equipment pressure parts are hydrostatically tested. If any repairs are made, the piping or equipment pressure part is retested. If any omissions or modifications of the test requirement are made, the deviation is shown valid before approval.

14.3 Nondestructive Testing

14.3.1 Welds

Girth and longitudinal pressure containing complete penetration groove butt welds are 100% examined by radiography.

Fillet welds, socket welds, and nonpressure-containing attachment welds such as supports, lugs, anchors, and guides are examined on all accessible surfaces by either the liquid penetrant or the magnetic particle method. Radiography is not required.

Welds attaching branch connections larger than 4 inches in pipe size are 100% examined by radiography, except where configuration does not permit effective radiography; then the root and final pass is examined by liquid penetrant or magnetic particle methods.

Accessible surfaces of the weld and adjacent base metal of branch connections 4 inches and less in pipe size are examined by either the liquid penetrant or the magnetic particle method.

Ultrasonic examination is not required.

14.3.1.1 Double-Welded Joints

The back of the first side welded is ground or chipped to sound metal and visually inspected prior to welding the second side.

14.3.2 Castings

Castings for pressure containing components larger than 4 inches are 100% examined by radiography and in the finished condition on all accessible machined surfaces by either the liquid penetrant or the magnetic particle method.

Castings for pressure containing components 4 inches nominal size and smaller do not require special non-destructive testing beyond non-destructive testing per materials specification.

14.3.3 <u>Forgings</u>-

Forgings for pressure containing components larger than 4 inches in nominal pipe size are examined in the finished condition on all accessible surfaces including machined surfaces by either the liquid penetrant or the magnetic particle method.

14.4 <u>Submittals</u>

Approval is required for the following inspection and test procedures:

1. Radiography
2. Ultrasonic testing
3. Liquid penetrant testing
4. Magnetic particle testing

APPENDIX F

CONFORMANCE TO AEC GENERAL DESIGN CRITERIA

APPENDIX F

CONFORMANCE TO AEC GENERAL DESIGN CRITERIA

1.0 SUMMARY DESCRIPTION

The proposed 70 General Design Criteria for Nuclear Power Plant Construction Permits were issued in July of 1967 to serve as a guide in the establishment of design criteria and bases for the design and construction of a nuclear power station. It is the purpose of this appendix to show that the design and construction of the Cooper Nuclear Station has been performed in accordance with these general design criteria.

It should be recognized that these criteria, which appeared in the July 11, 1967 issue of the Federal Register, were issued in order to secure comments from the nuclear industry, and at that time had not yet been adopted as regulatory requirements. It was anticipated that revisions and clarifications would take place prior to such adoption. The comparison which follows is presented to show that the concerns expressed by those criteria, as interpreted by the applicant, have been fully considered in the design of the station.

The method of presentation is to consider the criteria in nine groups. The grouping of the criteria is that given in the above referenced draft. For each group, a statement of the applicant's then current understanding of the intent of the criteria in the group is given along with discussion of conformance which is applicable to all of the criteria within the group. Each criterion in the group is then discussed as necessary to enlarge upon the general statements and a list of references where the subject material of the individual criterion is found in the original CNS-SAR is presented. The statements of the criteria are not presented but are referenced by number to the criteria statements presented in the July 11, 1967 Federal Register.

The following discussion was extracted directly from the original FSAR and left in its original form for historical purposes.

2.0 CRITERION CONFORMANCE

2.1 Group I -- Overall Plant Requirements (Criteria 1-5)

The purpose of these criteria is to insure that those systems and components of the station which have a vital role in the prevention or mitigation of consequences of accidents affecting public health and safety are designed and constructed to high quality standards which include consideration of natural phenomena and fire. Also, there must be sufficient surveillance and record keeping during fabrication and construction to ensure that these high quality standards have been met. As the station consists of a single nuclear plant, Criterion 4, Sharing of Systems, is not applicable. It will be seen that the concerns of these criteria have been properly considered throughout the design of the station.

Criterion 1 -- Quality Standards

A thorough quality assurance program has been undertaken during design and construction of the station to ensure that highest quality standards were used. Applicable codes were used where they were sufficient and more stringent requirements were placed on the design, where available codes were not sufficient. The quality assurance program is presented in Appendix D. The description of the various systems and components includes the codes and standards that are met in the design and their adequacy.

References: Subsections I-5, I-10, III-2 through III-8, IV-1 through IV-8, VII-2 through VII-5, Sections V, VI, VIII, and Appendix D.

Criterion 2 -- Performance Standards

Conformance to the structural loading criteria presented in Appendix C insures that those systems and components affected by this criterion are designed and built to withstand the forces that might be imposed by the occurrence of the various natural phenomena mentioned in the criterion, and this presents no risk to the health and safety of the public. The phenomena considered and margins of safety are also given.

References: Subsections I-5, XII-2 and Appendix C.

Criterion 3 -- Fire Protection

As described in Subsection X-9, the materials and layout used in the station design have been chosen to minimize the possibility and to mitigate the effects of fire. Sufficient fire protection equipment is provided in the unlikely event of a fire, and in no case will the ability of the station to be shutdown be compromised by fire.

References: Subsection X-9, Section XII.

Criterion 5 -- Records Requirement

Complete records of the as-built design of the station, changes during operation and quality assurance records will be maintained throughout the life of the station.

References: Subsection XIII-8, XIII-9, and Appendix D.

2.2 Group II -- Protection by Multiple Fission Barriers (Criteria 6-10)

Conformance to these criteria assures, through proper design, that the station has been provided with multiple barriers against the release of, or means for, the mitigation of the consequences of the release of fission products to the environs and that these barriers remain intact during abnormal operational transients. These criteria also provide for proper containment and barrier against the release of fission products in the event of design basis accidents.

To provide the required protection, the reactor design provides six means of containing, preventing, or mitigating the release of fission products. These are: the fuel barrier consisting of highly compacted UO_2 fuel sealed in high integrity Zircaloy cladding, the nuclear process system, the primary containment, the reactor building (secondary containment), the reactor building standby gas treatment system, and the plant stack (ERP).

Criterion 6 -- Reactor Core Design

The basis of the reactor core design, in combination with the station equipment characteristics and nuclear safety systems, is to provide sufficient margins to ensure that fuel damage does not occur during normal operation or as a result of abnormal operational transients. The core design is described in Section III and analysis of abnormal operational transients is given in Section XIV. The residual heat removal system and the reactor core isolation cooling system which remove decay heat during normal shutdowns and when the core is isolated from the condenser, are discussed in Section IV.

References: Subsections I-5, III-2, III-6, III-7, IV-3, IV-7, IV-8, VII-2, XIV-2, XIV-4, and XIV-5.

Criterion 7 -- Suppression of Power Oscillations

The core design alone and the design of the nuclear system including the core have been analyzed to determine if power oscillations could occur. This analysis, which is presented in Section VII-17 "Nuclear System Stability Analysis", shows that all power oscillations are suppressed and no fuel damage would occur.

References: Subsections I-5, III-4, III-6, III-7, IV-4, VII-2, VII-5, VII-7, VII-17, and XIV-5.

Criterion 8 -- Overall Power Coefficient

As indicated in Sections III and VII-17, the core is designed to be self-limiting; i.e., an arbitrary increase in core power over the power operating range results in a negative feedback. Thus, the overall power coefficient is negative.

References: Subsections I-5, III-6, III-7, and VII-17.

Criterion 9 -- Reactor Coolant Pressure Boundary (Nuclear System Process Barrier)

The nuclear system process barrier consists of the vessels, pipes, pumps, tubes and similar process components that contain steam, water, gases, and radioactive materials coming from, going to, or in communication with the reactor core. These are described primarily in Section IV "Reactor Coolant System". The reactor coolant system is designed to carry its dead weight and specified live loads separately or concurrently; these include pressure and temperature stresses, vibrations, and seismic loads prescribed for the station. Provisions are made to control or shutdown the reactor coolant system in the event of malfunction of operating equipment or leakage of coolant from the system. The reactor vessel and support structures are designed, within the limits of applicable criteria for low probability accident conditions, to withstand the forces that would be created by a full area flow of any vessel nozzle to the containment atmosphere with the reactor vessel at design pressure concurrent with the station maximum earthquake loads.

References: Subsections I-5, IV-2, IV-3, IV-4, IV-10, VII-8, XII-2, XIV-5, XIV-6, Appendix A and Appendix C.

Criterion 10 -- Containment

Two containment systems are provided; the drywell suppression chamber primary containment and the reactor building (secondary containment). These are described in Section V.

The primary containment system is designed, fabricated, and erected to accommodate without failure the pressures and temperatures resulting from or subsequent to the double-ended rupture or equivalent failure of any coolant pipe within the primary containment. The reactor building, encompassing the primary containment system, provides secondary containment when the primary containment is closed and in service, and provides for primary containment when the primary containment is open. The two containment systems and such other associated engineered safeguards as may be necessary are designed and maintained so that off-site doses resulting from postulated design basis accidents are below the values stated in 10CFR100.

References: Subsections V-2, V-3, XIV-4, and XIV-6.

2.3 Group III -- Nuclear and Radiation Controls (Criteria 11-18)

These criteria identify and define the station instrumentation and control systems necessary for maintaining the station in a safe operational status. This also includes determining the adequacy of radiation shielding, effluent monitoring, and fission process controls, and providing for the effective sensing of abnormal conditions and initiation of nuclear safety systems and engineered safeguards.

To satisfy the intent of these criteria the station is provided with a comprehensive control and instrumentation system, most of which is described in Section VII. Control of the station is from a central control room. Shielding and radiation protection are discussed in Subsection XII-3.

Criterion 11 -- Control Room

The station is provided with a centralized control room having adequate shielding to permit access and continuous occupancy under 10CFR20 dose limits during any of the design basis accident situations. This allows the station to be shut down when necessary and allows safe control of the station to be maintained following shutdown. The station design does not contemplate the necessity for evacuation of the control room. However, if it is necessary to evacuate the control room, the station can be brought to a safe, cold shutdown from outside the control room.

References: Subsections I-5, VII-2 through VII-5, VII-7 through VII-10, VII-12, X-10, XII-2, XII-3, and Section XIV.

Criterion 12 -- Instrumentation and Control Systems

The necessary station controls, instrumentation, and alarms for safe and orderly operation are located in the control room. These instruments and systems allow complete monitoring control of the facility throughout the normal operating range and through startup and shutdown. Sufficient instrumentation is provided to allow monitoring of all variables necessary for effective station control.

References: Subsections I-5, III-4, III-8, IV-10, VII-2 through VII-5, VII-7 through VII-10, VII-12 through VII-14, VII-17, and IX-2 through IX-4.

Criterion 13 -- Fission Process Monitors and Control

Continuous monitoring of the performance of the reactor and the reactor power level are provided by the nuclear instrumentation system as described in Subsection VII-5. Control of core reactivity is through the use of control rods, the positions of which are continuously available on the control board.

References: Subsections I-5, III-4, III-8, VII-2, VII-5, VII-7 through VII-9, and VII-17.

Criterion 14 -- Core Protection Systems

The reactor protection system, described in Subsection VII-2 in association with other safety systems, automatically senses and limits conditions which could lead to unacceptable fuel damage. This system acts independently of, and overrides, all other controls to initiate the necessary protective action. Evaluation of the protective action is given in the safety analysis.

References: Subsections I-5, III-4, III-5, IV-4 through IV-8, VI-1 through VI-7, VII-2 through VII-5, VII-7, VII-12, and XIV-1 through XIV-7.

Criterion 15 -- Engineered Safety Features Protection Systems

The reactor core standby cooling system control and instrumentation system description in Subsection VII-4 details the instrumentation provided to monitor the necessary variables and to automatically initiate the proper safety action in the event of an accident. This system also acts independently of the station

-ocess control systems and overrides all other controls to initiate the necessary ıfety actions.

References: Subsections I-5, VI, VII-2 through VII-5, and VII-12.

Criterion 16 -- Monitoring Reactor Coolant Pressure Boundary

The methods of detecting leakage through the reactor coolant pressure boundary, and the limits imposed on this leakage, are discussed in Subsection IV-10.

References: Subsections I-4, IV-10, V-2, VII-8, and X-14.

Criterion 17 -- Monitoring Radioactive Releases

The station process and area radiation monitoring systems and station sampling procedures are provided for monitoring significant parameters from specific station process systems and specific areas including the station effluents to the site environs and to provide alarms and signals for appropriate corrective actions. These are described in Subsections VII-12 and VII-13.

References: Subsections I-4, VII-12, VII-13, IX-2 and IX-4.

Criterion 18 -- Monitoring Fuel and Waste Storage

The new and spent fuel storage areas have been analyzed to determine their safety, and instrumentation is provided for monitoring where needed. Control and monitoring of waste storage is provided as described in Section IX, Subsection VII-12 and X-5.

References: Subsections I-5, VII-12, VII-13, IX-2, IX-4, and X-5.

2.4 Group IV -- Reliability and Testability of Protection Systems (Criteria 19-26)

The purpose of these criteria is to ensure that the systems used to prevent breach of the clad barrier will: (1) function when needed in spite of the failure of a component within the system, (2) be designed such that a condition requiring a protection system will not prevent the proper functioning of that system, and (3) be designed so that each channel of a protection system is independent of other channels within that system and the control systems. Protection system testability and detection of failures within the protection systems are necessary to ensure the reliability of these systems. As seen in the design bases and descriptions of these systems, sufficient attention has been paid to component reliability, system testability and alarms, independence and power supply, to ensure that the protection systems are adequate with respect to these criteria. The description of these systems appears largely in Section VII of the CNS-SAR.

Criterion 19 -- Protection Systems Reliability

The components of the protection systems are designed to a high standard of reliability. Each system is designed with provisions for testing which approximate very closely the functioning of the system under design conditions of that system.

References: Subsections I-5, III-4, VI, VII-2 through VII-5, VII-12, and Section XIV.

Criterion 20 -- Protection Systems Redundancy and Independence

Protection system design includes the capability of providing the required protection, even with a component or channel inoperative due to failure or removal. Each of the protection function actions are initiated by a variety of sensed station conditions and by at least two instrument channels. This protection action is not dependent on a single channel.

References: Subsections I-5, III-4, VI, VII-2 through VII-5, VII-12, and Section XIV.

Criterion 21 -- Single Failure Definition

This definition is used in the design throughout the CNS-SAR for safety systems.

References: Subsections I-2, and XIV-4.

Criterion 22 -- Separation of Protection and Control Instrumentation Systems

The systems which initiate the scram, isolation, and core standby cooling actions are designed to automatically override normal operational controls whenever station conditions monitored by these systems exceed pre-established limits. Removal from service of a control instrumentation system cannot compromise any reactor protection function. Thus, protection action is independent of the state in normal operational process control actions.

References: Subsections I-4, III-4, VI-5, VII-2 through VII-5, and VII-12.

Criterion 23 -- Protection Against Multiple Disability for Protection Systems

These systems are designed to provide the required protection as long as necessary and in the presence of the most severe conditions which would be encountered. This includes all conditions resulting from transients and accidents for which the protective action is required.

References: Subsections I-5, III-4, VII-2 through VII-5, VII-12, and Section XIV.

Criterion 24 -- Emergency Power for Protection Systems

In the event of a loss of offsite power, the station auxiliary power system, the standby diesel generators, and the 125 volt battery system provide adequate power and redundancy to permit the required functioning of the protection systems. In addition, the 100% capacity redundant halves of each system are adequately separated to prevent the loss of power to the protection system resulting from any single active or passive failure.

References: Sections VI, VII, Subsections I-5, and VIII-4 through VIII-6 criterion is not interpreted to require a fast scram capability of both systems but only the stated shutdown capability.

References: Subsection I-5, III-4, III-6, III-9, VII-7, and Section XIV.

Criterion 29 -- Reactivity Shutdown Capability

Reactor shutdown by the control rod drive system is sufficiently rapid to prevent violation of fuel damage limits for normal operation and all abnormal operational transients, even with the most reactive control rod fully withdrawn. The nuclear design assures that sufficient reactivity compensation is always available to make the reactor subcritical from its most reactive condition including compensation for positive and negative reactivity changes resulting from nuclear coefficients, fuel depletion and fission product transients and buildup.

References: Subsections I-5, III-4, III-6, VII-2, and Section XIV.

Criterion 30 -- Reactivity Holddown Capability

As indicated in the previous criterion response, the operational control system is designed to make and hold the reactor subcritical from its most reactive condition under all normal credible operating conditions.

References: Subsections I-5, III-4, III-6 and III-9.

Criterion 31 -- Reactivity Control Systems Malfunction

Reactivity control systems designs (in conjunction with the reactor protection systems) ensure that acceptable fuel damage limits will not be exceeded for any credible reactivity transient resulting from a single equipment malfunction or a single operator error.

References: Subsections I-5, III-4, III-6, III-7, VII-2, VII-7, and Section XIV.

Criterion 32 -- Maximum Reactivity Worth of Control Rods

The system design is such that control rod worths and the rate at which reactivity can be added are sufficiently limited to assure that the design basis reactivity accident is not capable of damaging the reactor coolant system or disrupting the reactor core, its support structures, or other vessel internals sufficiently to impair the core, standby cooling systems' effectiveness, if these systems are needed.

References: Subsections I-5, III-4, III-6, III-7, VI, VII-7 and Section XIV.

2.6 Group VI -- Reactor Coolant Pressure Boundary (Criteria 33-36)

The intent of this group of proposed criteria is to establish the reactor coolant pressure boundary design requirements and to identify the means used to satisfy these design requirements. The "reactor coolant pressure boundary" is referred to in the safety analysis report as the "nuclear system primary barrier" (see "Definitions" in "Introduction and Summary", Subsection I-2). The react-

Criterion 25 -- Demonstration of Functional Operability of Protection Systems

All of the protection systems contain sufficient test signals, bypasses and indicators to allow testing of the system under simulated conditions closely approximating the actual condition for which the protective action is required. Provisions are also included to automatically override any testing being carried on, should the channel under test be needed for a protective action.

References: Subsections I-5, VI-7, VII-2 through VII-5, and VII-12.

Criterion 26 -- Protection Systems Fail-Safe Design

Systems essential to the protection functions are designed to fail-safe in their most probable failure modes. Thus, a systematic or environmentally caused failure will be indicated and will not compromise the protective function of the system.

References: Subsections I-5, VI-1 through VI-6, VII-2 through VII-5, VIII-4 and VIII-5.

2.5 Group V -- Reactivity Control (Criteria 27-32)

Conformance to these six criteria provides assurance that the reactor core can be made and held subcritical from normal operation or from normal anticipated operational transients, by at least two reactivity control systems and that malfunction of a reactivity control system will not result in unacceptable damage to the fuel, rupture of the reactor coolant pressure boundary, or disrupt the core to the point of preventing core standby cooling if needed. Two systems, an operational control system, consisting of moveable control rods, and control by recirculation flow control; and a standby liquid control system are provided to meet the intent of these criteria. The moveable control rod system design is given in Subsection III-4 and control of the moveable rod system is described in Subsection VII-7; the nuclear design, including the control rod reactivity worths, is given in Subsection III-6; reactor coolant recirculation system flow control is described in Subsection VII-9; and the standby liquid control system is described in Subsection III-8.

Criterion 27 -- Redundancy of Reactivity Control

The two reactivity control systems provided are completely independent and of different principal. The operational control system accommodates fuel burnup, load changes and long-term reactivity changes. The standby liquid control system provides independent shutdown capability if it is needed.

References: Subsection I-5, III-4, III-9, and VII-7.

Criterion 28 -- Reactivity Hot Shutdown Capability

Both the control rod system and the standby liquid control system are capable of making and holding the core subcritical from any hot standby or hot operating condition up through full power. Consistent with current practice, this coolant system design, described in Section IV and Subsection III-3, together with the quality assurance program (Appendix D), show that these criteria have been properly considered. In-service inspection of components and parts inside this boundary is discussed in Appendix J.

Criterion 33 -- Reactor Coolant Pressure Boundary Capability

As shown in Section XIV, the consequences of the design basis rod drop accident cannot result in damage (either by motion or rupture) to the nuclear system process barrier. This is due to the inherent safety features of the reactor core design combined with the control rod velocity limiter.

References: Subsections I-5, III-3 through III-6, IV-2, IV-5, IV-6, and XIV-4 through XIV-6.

Criterion 34 -- Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention

The ASME and USASI Codes are used as the established and acceptable criteria for design, fabrication, and operation of components of the nuclear system primary barrier. The nuclear system primary barrier is designed and fabricated to meet the following, as a minimum:

1. Reactor Vessel--ASME Boiler and Pressure Vessel Code, Section III Nuclear Vessels, Subsection A.

2. Pumps--ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Subsection C.

3. Piping and Valves--USAS B.31.1, Code for Pressure Power Piping.

The brittle fracture failure mode of the nuclear system primary barrier components is prevented by control of the notch toughness properties of ferritic steel. This control is exercised in the selection of materials and fabrication of equipment and components. In the design, appropriate consideration is given to the different notch toughness requirements of each of the various ferritic steel product forms, including weld and heat-affected zones. In this way, assurance is provided that brittle fracture is prevented under all potential service loading temperatures.

References: Subsections III-3, IV-2, IV-3, VII-8, Appendix A and Appendix D.

Criterion 35 -- Reactor Coolant Pressure Boundary Brittle Fracture Prevention

The applicant's selected approach to brittle fracture prevention is to use a temperature based rule with modifications drawn from fracture mechanics technology. The approach, which is generally accepted by materials specialists, establishes the requirements for brittle fracture prevention. These requirements are less stringent, when measured in terms of NDTT requirement, for thin section materials than the thick section materials assumed in the first draft of this criterion.

The toughness properties of ferritic material and the service temperature of the reactor coolant pressure boundary shall assure:

1. Fully ductile behavior (e.g., in the energy absorption region of 100 percent shear fracture) whenever the boundary can be pressurized beyond the systems safety valve setting by operational transients in postulated accidents; and

2. A ductile to brittle fracture transition temperature at least 60°F below the service temperature whenever the boundary can be pressurized beyond 20 percent of its design pressure by operational transients, hydro tests, and postulated accidents.

The response of the reactor system pressure to postulated accidents is discussed in the General Electric Company reply to Comment 3.8.1 of Amendment 1 to Bell Station Unit 1, Docket No. 50-319. There are no operational transients which can pressurize the system boundary beyond the system's safety valve setting (1250 psig), so requirements for fully ductile behavior in pressure boundary materials are not anticipated.

It is believed that Criterion 35 should be applicable only to those components or systems whose failure would result in a loss of coolant in excess of the normal make-up capability of the reactor coolant system. On this ground small lines such as instrument lines have been excluded; certain other lines, such as the main steam lines, have been exempted from temperature control during hydrostatic test conditions in which failure would not affect core cooling.

Reference: Subsection IV-2.

Criterion 36 -- Reactor Coolant Pressure Boundary Surveillance

The reactor coolant system is given a final hydrostatic test at 1560 psig in accordance with code requirements prior to initial reactor startup. A hydrostatic test, not to exceed system operating pressure, is made on the reactor coolant system following each removal and replacement of the reactor vessel head. The system is checked for leaks, and abnormal conditions are corrected before reactor startup. The minimum vessel temperature during hydrostatic tests is at least 60°F above the calculated NDTT prior to pressurizing the vessel. Extensive quality control assurance programs are also followed during the entire fabrication of the reactor coolant system. Vessel material surveillance samples are used to enable periodic monitoring of material properties with exposure. The program includes specimens of the base metal, heat affected zone metal and standards specimens. Also, leakage from the reactor coolant system is monitored during reactor operation. The material surveillance program will conform to ASTM E 18566.

References: Subsections IV-2, IV-3, and IV-10.

2.7 Group VII -- Engineered Safety Features (Criteria 37-65)

The intent of this group of proposed criteria is (1) to identify the nuclear safety systems and engineered safeguards, (2) to examine each one for independency, functional redundancy, capability, testability, inspectability, and reliability, (3) to determine the suitability of each for its intended duty, and (4) justify that each safety feature's capability-scope encompasses all the anticipated and credible phenomena associated with the station operational transients or design basis accidents being considered. While the first seven criteria are applicable to all of the engineered safety features, the remaining criteria fall into four groups: emergency core cooling systems (Criteria 44-48); containment (Criteria 49-57); containment pressure reducing systems (Criteria 58-61); and air cleanup systems (Criteria 62-65). Examination of each of these safety features will show that their design conforms to the Group VII Criteria.

2.7.1 General Requirements for Engineered Safety Features (Criteria 37-43)

Criterion 37 -- Engineered Safety Features Basis for Design

The normal station control systems maintain station variables within operating limits. These systems are thoroughly engineered and backed up by a significant amount of experience in system design and operation. Even if an improbable maloperation or equipment failure occurs (including a nuclear system process barrier break up to and including the circumferential rupture of any pipe in that barrier), the nuclear safety systems and engineered safeguards limit the effects to levels well below those which are of public safety concern. These engineered safety features include those systems which are essential to the containment, isolation, and core standby cooling functions.

References: Subsections I-5, III-3, III-4, IV-2, IV-4, IV-6, V-2, V-3, VI-1 through VI-7, VII-2 through VII-4, VIII-4 through VIII-6, and VIV-1 through XIV-7.

Criterion 38 -- Reliability and Testability of Engineered Safety Features

The design of each of the systems essential to the engineered safety features includes the use of highly reliable components and provides for ready testability of these systems. Extensive analytical and experimental programs have shown that these systems are capable of performing their designated tasks.

References: Subsections I-5, III-4, III-5, IV-6, V-2, V-3, VI-6, VII-2, VII-4, VII-5, VII-12, and VIII-4 through VIII-6.

Criterion 39 -- Emergency Power for Engineered Safety Features

With the redundant, full capacity diesel generators and batteries and redundant sources of offsite power, adequate power sources to accomplish all required safety functions under postulated design basis accident conditions is assured. Furthermore, each power source can be periodically tested for availability.

References: Subsections VII-2, VII-3, VII-4, and VIII-2 through VIII-6.

Criterion 40 -- Missile Protection

The systems and equipment which are required to function after design basis accidents or abnormal operational transients are designed to withstand the most severe forces and environmental effects, including missiles from station equipment failures anticipated from the accidents and missiles generated by tornadoes, without impairment of their performance capability.

References: Subsections V-2, XII-2, and Appendix C.

Criterion 41 -- Engineered Safety Features Performance Capability

Those systems that comprise the engineered safety features are designed with sufficient redundancy and independence to fulfill their integrated required safety functions even with failure or a single component. However, as with Criterion 39, the applicant considers high availability and reliability of the engineered safety features to be the intent of this criterion. Designs based on meeting overall reliability and availability goals will lead to improved safety as it gives proper weighting not only to single failures, but also to combined failures with a high probability of occurrence.

References: Subsections VI-1 through VI-5, VII-4, and XIV-6.

Criterion 42 -- Engineered Safety Features Components Capability

The components which are required to function following a design basis loss-of-coolant accident are designed to withstand the most severe forces and environmental effects resulting from the accident.

References: Subsections III-4 V-2, V-3, VI-1 through VI-5, VII-2, VII-3, VII-4, VIII-2 through VIII-6, and XIV-6.

Criterion 43 -- Accident Aggravation Prevention

The systems comprising the engineered safety features are all designed to act in a positive manner in reducing the consequences of a loss-of-coolant accident.

References: Subsections III-4, V-2, V-3, VI-1 through VI-5, VII-3, VII-4, and VIII-2 through VIII-6.

2.7.2 Emergency Core Cooling Systems (Criteria 44-48)

Criterion 44 - Emergency Core Cooling Systems Capability

The core standby cooling systems (CSCS) are designed to limit clad temperature to below 2700°F over the entire credible spectrum of postulated design basis reactor coolant system breaks. Such capability is available concurrently with the loss of all offsite a-c power. The CSCS themselves are designed to various levels of component redundancy such that no single active component failure in addition to the accident can prevent adequate core cooling.

References: Subsections VI-1 through VI-5, VII-4, and XIV-6.

Criterion 45 -- Inspection of Emergency Core Cooling Systems

The CSCS design includes provisions to enable physical and visual inspection of the CSCS components. All components are inspected prior to installation. In-service inspection is discussed in Appendix J.

References: Subsectior- III-3, IV-2, and VI-6.

Criterion 46 -- Testing of Emergency Core Cooling System Components

To assure that the CSCS functions properly, if needed, specific provisions have been made for testing the operability and functional performance of each active component of each system.

References: Subsections I-5, VI-6, and VII-4.

Criterion 47 -- Testing of Emergency Core Cooling Systems

Specific provisions such as recirculation loops have been provided in the CSCS design to allow periodic testing of the delivery capability of these systems with conditions as close to accident conditions as possible.

References: Subsections VI-6, and VII-4.

Criterion 48 -- Testing of Operational Sequence of Emergency Core Cooling Systems

To assure that the CSCS functions properly, if needed, specific provisions have been made for testing the sequential operability and functional performance of each individual system. Testing of the systems is done in parts rather than testing of the entire operational sequence. This is due to the unavailability of these systems during a complete operational test as described, particularly since it may be extremely difficult to perform such a test during reactor operation. The design complications which will be required in order to permit such a test complicates an already complex system, which may be detrimental to safety.

References: Subsections I-5, VI-4, VI-6, VII-4, VIII-5, VIII-6, and X-8.

2.7.3 Containment (Criteria 49-57)

Criterion 49 -- Containment Design Basis

The primary containment structure, including access openings and penetrations, is designed to withstand the peak accident pressure and temperatures which could occur due to the postulated design basis loss-of-coolant accident. The containment design includes considerable allowance for energy addition from metal-water or other chemical reactions beyond conditions that could exist during the accident.

References: Subsections I-5, IV-6, V-2, V-3, VI-1, VI-2, VI-5, VII-3, VII-4, XIV-2 through XIV-7, and Appendix C.

Criterion 50 -- NDTT Requirement for Containment Material

The design of the containment and its material are described in Subsection V-2. The criterion as stated is considered to be overly specific, considering the general nature of the other criteria. In keeping with the intent of these criteria to serve as a general guide, this criterion is interpreted to mean that the containment will be designed in accordance with applicable engineering codes.

References: Subsections V-2 and V-3.

Criterion 51 -- Reactor Coolant Pressure Boundary Outside Containment

Analyses have been made to ensure that a rupture of a pipe which is part of the reactor coolant pressure boundary will not jeopardize the health and safety of the public according to the requirements established by 10CFR100. When needed, isolation valves are provided. The largest of these pipes are the steam lines. The analysis of a circumferential rupture of the steam line is discussed in Section XIV. The definition of the coolant primary pressure boundary is given in Subsection I-2.

References: Subsections I-5, II-2, II-3, IV-6, V-2, VII-3, and XIV-6.

Criterion 52 -- Containment Heat Removal Systems

Provisions are made for the removal of heat from within the primary containment as necessary to maintain the integrity of the containment for as long as necessary following the various postulated design basis accidents. Pressure suppression phenomena and the containment spray cooling system provide two different means to rapidly condense the steam portion of the flow from the postulated design basis loss-of-coolant accident.

References: Subsections I-5, IV-8, V-2, VI-1 through VI-5, VII-4, X-8, XIV-6 and XIV-7.

Criterion 53 -- Containment Isolation Valves

All lines which penetrate the primary containment and which communicate with the reactor vessel or the primary containment free space are provided with at least two isolation valves (or equivalent) in series.

References: Subsections I-5, IV-6, V-2, and VII-3.

Criterion 54 -- Containment Leakage Rate Testing

After completion and installation of all penetrations, an integrated leakage rate test is performed at design pressure to verify that the containment design does meet the required maximum leakage rate. The test is performed over a 24 hour interval or longer as required, to show conformance to the required performance.

Reference: Subsection V-2.

Criterion 55 -- Containment Periodic Leakage Rate Testing

Leakage rate testing of the containment at design pressure is not now an AEC requirement for stations after they have been placed in service. The severe burden which this would impose has been recognized and a modified procedure adopted. This procedure uses the relationship between leakage rates measured initially at design pressure, and at some reduced pressure. Such a relationship is then employed to extrapolate subsequent test values for leakage at reduced pressure to the full design pressure of interest.

This criterion is therefore interpreted as follows: "The containment shall be designed so that integrated leakage rate testing can be done periodically during station lifetime. Such tests will be made at a pressure which permits extrapolation of results to the design pressure condition, using relationships established initially for comparative leakage at these low conditions."

Provisions have been included in the station design for periodic leakage rate testing as described above.

Reference: Subsection V-2.

Criterion 56 -- Provisions for Testing of Penetrations

Provisions are made to demonstrate leak tightness at design pressure of all resilient seals and expansion bellows on containment penetrations on an individual basis.

Reference: Subsections V-2 and V-3.

Criterion 57 -- Provisions for Testing of Isolation Valves

Provisions are also made for demonstrating the functional performance of containment system isolation valves and monitoring valve leakage.

References: Subsections IV-6, IV-10, V-2, VII-3, and VII-12.

2.7.4 Containment Pressure Reducing Systems

Criterion 58 -- Inspection of Containment Pressure Reducing Systems

The containment spray cooling system, an integral part of the residual heat removal system, is designed to allow periodic inspection of the pumps, pump motors, valves, heat exchangers, and piping of this system. The torus and torus water and the spray nozzles may also be periodically inspected.

References: Subsections IV-8, V-2, V-3, VI-4, VI-6, X-6, X-8, and XII-2.

Criterion 59 -- Testing of Containment Pressure Reducing Systems Components

All of the valves and pumps of these systems can be tested periodically for operability and capability to perform as required.

References: Subsections IV-8, V-2, VI-4, VI-6, VII-3, VII-4, X-6, and X-8.

Criterion 60 -- Testing of Containment Spray Systems

The capability to test the functional performance of the containment spray cooling system is provided by inclusion in the design of appropriate test connections.

References: Subsections IV-8, VI-4, VI-6, and VII-7.

Criterion 61 -- Testing of Operational Sequence of Containment Pressure-Reducing Systems

Those concerns expressed in the similar criterion for core standby cooling systems (Criterion 48) are reiterated here. These concerns are the detrimental effects on safety of the unavailability of the systems during test and the system complications required to carryout the test. Because of these concerns, this criterion is also interpreted to mean testing of such systems in parts rather than testing of the entire operational sequence. Such testing is provided in the design of the containment pressure-reducing systems, including the transfer to alternate power sources.

References: Subsections V-2, V-3, VI-4, VI-6, VII-4, VIII-4, VIII-5, and VIII-7.

2.7.5 Air Cleanup Systems

Criterion 62 -- Inspection of Air Cleanup Systems

The standby gas treatment system (see Subsection V-3) which is located in the reactor building, may be physically inspected. This includes all ducting, fans, filters, valves and heaters.

References: Subsections V-2, V-3, and X-10.

Criterion 63 -- Testing of Air Cleanup Systems Components

Fans and dampers for the standby gas treatment system can be tested periodically for operability and required functional performance.

References: Subsections V-2, V-3, and X-10.

Criterion 64 -- Testing of Air Cleanup Systems

The standby gas treatment system includes provisions for periodic testing and surveillance to verify that degradation of the system has not occurred.

References: Subsections V-2, V-3, and X-10.

Criterion 65 -- Testing of Operational Sequence of Air Cleanup Systems

The standby gas treatment system can be periodically tested for system performance by tracer injection and sampling under full flow conditions.

References: Subsections V-3, VII-12, and XIII-4.

2.8 Group VIII -- Fuel and Waste Storage Systems (Criteria 66-69)

The intent of this group of criteria is to ensure that fuel and waste storage systems are designed to minimize the probability of radioactivity release to station operating areas or public environs. A review of the new and spent fuel storage systems (Subsections X-2 and X-3) and the radwaste systems (Section IX) shows that the intent of these criteria has been met.

Criterion 66 -- Prevention of Fuel Storage Criticality

Appropriate station fuel handling and storage facilities are provided to preclude accidental criticality for spent fuel. The new fuel storage vault racks (located inside the reactor building) are top entry, and are geometrically designed to prevent an accidental critical array, even in the event the vault becomes flooded. Vault drainage is provided to prevent possible water collection.

References: Subsections VII-6, X-2 and X-3.

Criterion 67 -- Fuel and Waste Storage Decay Heat

The spent fuel pool cooling system is designed to remove decay heat to maintain the pool water temperature. The fuel storage pool contains sufficient water so that in the event of the failure of an active system component, sufficient time is available to either repair the component or provide alternate means of cooling the storage pool.

References: Subsection X-5.

Criterion 68 -- Fuel and Waste Storage Radiation Shielding

The handling and storage of spent fuel is done in the spent fuel storage pool. Water depth in the pool is maintained at a level to provide sufficient shielding for normal reactor building occupancy (10CFR20) by operating personnel. The spent fuel pool cooling and demineralizer system is designed to control water clarity (to allow safe fuel movement) and to reduce water radioactivity. Accessible portions of the reactor and radwaste buildings have sufficient shielding to maintain dose rates within the limits of 10CFR20.

References: Subsections IX-1 through IX-4, X-3, X-5, XII-2 and XII-3.

Criterion 69 -- Protection Against Radioactivity Release From Spent Fuel and Waste Storage

The consequences of a fuel handling accident are presented in Subsection XIV-6 of the CNS-SAR. In this analysis, it is demonstrated that undue amounts of radioactivity are not released to the public.

All spent fuel and waste storage systems are conservatively designed with ample margin, to prevent the possibility of gross mechanical failure which could release significant amounts of radioactivity. Backup systems such as floor and trench drains are provided to collect potential leakages. The fuel handling and waste disposal systems are described in Sections X and IX, respectively. Operators are rigorously trained and administrative procedures are strictly followed to reduce the potential for human error.

The radiation monitoring system as described in Subsections VII-12 and VII-13 of the CNS-SAR is designed to provide station personnel with early indication of possible station malfunctions.

References: Subsections V-1, V-2, V-3, IX-2 through IX-4, X-2, X-3, X-5, X-14, XII-1, XII-2, and XIV-6.

2.9 Group IX -- Plant Effluents (Criterion 70)

The intent of this criterion is to establish the station effluent release limits as defined by applicable regulations and to ensure that the station design provides means of controlling the releases within these limits. The various systems provided for radioactive effluent control are all designed to meet the intent of this criterion.

Criterion 70 -- Control of Releases of Radioactivity to the Environment

The station radioactive waste control systems (which include the liquid, gaseous and solid radwaste systems) are designed to limit the off-site radiation exposure to levels below limits set forth in 10CFR20. The station engineered safeguards (including the containment barriers) are designed to limit the off-site dose under various postulated design basis accidents to levels significantly below the limits of 10CFR100. The air ejector off-gas system is designed with sufficient holdup retention capacity so that during planned station operation the controlled release of radioactive materials does not exceed the established release limits at the ERP.

References: Subsections I-5, V-2, V-3, VII-12, VII-13, IX-2 through IX-4, XIV-2 through XIV-7, and Appendix E.

14-4(19)

September 10, 1975

Mr. Karl R. Goller
Assistant Director for Operating Reactors
Division of Reactor Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Cooper Nuclear Station Compliance With 10CFR50, Appendix J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors

Dear Mr. Goller:

This letter is in response to your letter to the District dated August 5, 1975 which requests information concerning the compliance with 10CFR50, Appendix J.

A review of the Cooper Nuclear Station Technical Specifications and 10CFR50, Appendix J indicates that the District is in full compliance with Appendix J with the following exceptions:

1.) The Main Steam Isolation Valves (MSIV's) are tested at 29 psig (P_t) instead of the required 58 psig (Pa).

2.) The personnel air lock door is tested at intervals no longer than one year at 58 psig (Pa) and at 3 psig after each opening during the one year interval between the 58 psig tests.

3.) The void between the bellows located in the main steam line and feedwater line penetrations are tested at 5 psig instead of the required 58 psig (Pa).

4.) The feedwater checkvalves are tested with water.

The present method of testing the MSIV's calls for pressurization between the inboard and outboard isolation valves. This procedure loads the inboard valve in the opposite direction as the valve design and therefore requires the reduced test pressure. Additional testing required by the NRC (Letter from Vone A. Moore to Ronald [illegible], dated July 27, [illegible]) specifies the testing options of single valves, pairs of valves or manifold leak testing of all eight valves.

Present plans for MSIV testing call for leakage testing to be done in October 1975 with an approved method using a 29 psig test pressure and the program to develop a new procedure allowing pressurization from the outside. [illegible]

N/I
ATT 4 (19)

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Mr. Karl R. Goller
September 10, 1975
Page 2

procedure will use a 58 psig test pressure. After this procedure has been perfected and proven operational, a request for a Technical Specification change will be submitted. The allowance for midcycle testing at 29 psig by use of any of the three prescribed options will be retained.

The existing personnel air lock door opens inwardly and seats with accident pressure. The present technical specifications require testing the personnel air lock at 58 psig at intervals no longer than one year. This test requires that a strongback be installed on the containment side of the inside door. The air lock doors are tested for air leakage at 3 psig after each opening during the test interval between the annual 58 psig test.

No changes are contemplated from the existing testing requirements. Testing at an increased pressure of 58 psig (Pa) would require drywell entry for strongback installation. This can only be done during a shutdown condition. The pressurization of the door in the wrong direction also entails some risk of permanent deformation which would be greatly increased if all tests were run at 58 psig. It is the opinion of the District that a yearly test at 58 psig is sufficient to show physical integrity and the 3 psig test after door operation shows the seal condition. An increase to a 6 month test frequency also places an additional operational restraint on the plant which should ideally operate for periods longer than this without required shutdown and drywell entry.

The main steam and feedwater testable penetrations consist of a double layered metal bellows. The inboard high pressure side of the bellows sees the existing drywell pressure. Therefore the bellows is tested in its entirety when the drywell is tested. The bellows layers are tested for integrity of both layers by pressurizing the void between the layers to 5 psig. Any higher pressure could cause permanent deformation, damage and possible ruptures of the bellows. While this testing between the bellows does not meet the 58 psig requirement of Appendix J, it is felt that the testing of the penetration as a complete unit with the drywell pressurization test and the indication of the integrity of both bellows layers provides reasonable assurance that the penetration will withstand the accident conditions.

The feedwater check valves, which are of a tilting disc design, are currently leak tested with water instead of air as required by Appendix J. Water at a constant 58 psig pressure is supplied to the system downstream side of the valve. After the leak rate has stabilized to produce a constant rate of flow on the upstream side of the valve, the flow is measured over a given time span. This information is used to calculate leak rate for these valves which will provide an adequate indication of the valve's condition.

The Cooper Nuclear Station Radiological Technical Specifications were issued with the facility operating license on January 18, 1974. Up until that time, meetings were held with the AEC licensing staff to assure that the CNS Technical Specifications were in compliance with the 10CFR50, Appendix J. The four areas of non-compliance discussed in this letter were discussed at length with the

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14-4(21)

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Title 10—Atomic Energy

CHAPTER I—ATOMIC ENERGY COMMISSION

PART 50—LICENSING OF PRODUCTION AND UTILIZATION FACILITIES

Reactor Containment Leakage Testing for Water-Cooled Power Reactors

On August 27, 1971, the Atomic Energy Commission published in the FEDERAL REGISTER (36 FR 17053) a proposed amendment to its regulations in 10 CFR Part 50 which would specify the minimum containment leakage test requirements for water-cooled power reactors.

Interested parties were invited to submit written comments and suggestions for consideration in connection with the proposed amendment within 60 days after publication in the FEDERAL REGISTER. Upon consideration of the comments received, and other factors involved, the Commission has adopted the proposed amendment, with certain modifications in the form set forth below.

Significant differences from the amendment published for comment are: (1) Modification of procedures governing containment inspection and leak detection surveys, as a prerequisite to conducting formal leakage tests, and clarification of the basis for reporting preleak leakage values to the Commission, (2) establishment of criteria for defeating certain safety-related systems from routinely scheduled Type A containment leakage tests, (3) incorporation by reference of the recently-issued American National Standard for leakage rate testing of containment structures for nuclear reactors into the regulation, (4) inclusion of nitrogen gas as a suitable testing medium for testing the leaktightness of valves, and (5) inclusion of water-leakage test and acceptance criteria for containment isolation valves which are sealed against containment atmosphere outleakage during a design basis accident condition by means of a seal-water system. In addition, editorial and format changes were made.

With regard to item (1) above, the rule set forth below requires the licensee to identify specifically those components whose initial poor leak-tightness performance precluded completion of a Type A containment leakage test and to report this information to the Commission. The proposed rule would not have required the reporting of such information unless attempts to reduce the leakage rate of poor leak-tight components failed to meet minimum leak-tightness acceptance criteria. Thus, components which required frequent adjustments or repair in order to meet allowable leakage limits will be identified and the specific reductions in leakage rate values resulting from such adjustments, will be reported to the Commission. The identification of such components will provide the AEC with a sounder basis for judging whether or not containment leakage rates could have been exceeded in the unlikely event a design basis accident were to occur. In addition, such identification may provide insight into the frequency and kinds of adjustments being made to components to meet the minimum acceptable leakage limits and a basis for either establishing a more frequent containment leakage test schedule, or modifying or replacing components.

With regard to item (2) above, the rule set forth below specifies criteria in which the licensee may for certain safety-related systems temporarily dispense with drainage and venting to containment atmosphere during Type A containment leakage tests. The proposed rule had specified that all systems which would connect directly with the containment atmosphere and would become an extension of the containment boundary should be vented to containment. Strict compliance with this rule would have required removing certain safety-related systems from service for the duration of the test and would limit the performance of the overall integrated containment leakage tests to those times when there would be no fuel in the reactor. This procedure is considered to be unnecessarily conservative.

The inclusion of all safety-related systems in the overall integrated containment leakage test can be accomplished while the reactor is fueled, and in a state of potential criticality, by maintaining the minimum number of safety-related systems in an operable state until all systems are tested. Another option is to periodically test the containment isolation valves in these safety-related systems in accordance with the rule set forth below. This would also assure that the requisite level of plant safety will be provided during the containment leakage test program without compromising the requirements for including all systems which penetrate the containment boundary in the leakage test.

The proposed rule required the use of test methods described in proposed American Nuclear Society Standard ANS 7.60 by referencing a portion of the proposed standard. On March 16, 1972, the American National Standards Institute approved ANS 7.60 and officially issued it for use as ANSI N45.4-1972, American National Standard, "Leakage Rate Testing of Containment Structures for Nuclear Reactors." The standard has been reviewed for compatibility with the proposed rule and it was concluded that incorporation of the requirements of ANSI

N/1
ATT 4 (21)

1-1972 by reference would enhance quality of containment leak testing. Accordingly, the rule set forth below now specifies that the Type A containment leakage tests shall be conducted in accordance with the provisions of ANSI N45.4-1972.

The proposed rule limited the leakage testing medium for reactor containment isolation valves to air, which is widely used in the containment leakage testing program. However, the use of nitrogen gas for valve leakage testing is also technically satisfactory. Accordingly, the rule set forth below specifies that either air or nitrogen may be used as the testing medium in the conduct of the valve leakage tests.

The rule set forth below expands upon the requirements contained in the proposed rule for testing valves, sealed with water from a seal-water system, by including minimum water test pressure and test acceptance criteria.

The proposed rule required that the valves be subjected to a seal-water system operability test to establish that the valves could be satisfactorily pressurized with seal-water. There was no requirement to measure the actual rate of water leakage past the valve. It had been assumed that the seal-water inventory would be adequate to seal the valves against outleakage of containment atmosphere during the design basis accident condition. However, the lack of a specific water inventory criterion against which actual valve leakage rates would be measured, could result in an inadequate supply of seal-water for valve sealing with attendant loss of the containment isolation function. Accordingly, a provision has been incorporated into the rule set forth below which requires that the valve leakage rate shall not exceed the seal-water inventory, on the assumption that the seal-water system will be pressurized for 30 days at 110 percent of the calculated peak containment internal pressure related to the design basis accident. With the inclusion of this requirement, the requirements for conducting only a seal-water system operability test were eliminated.

Containment is provided for water-cooled power reactors to prevent the uncontrolled release of radioactive materials to the environment [illegible] reactor coolant pressure boundary should be breached. Testing the reactor containment for leakage helps to assure that:

(a) Leakage of the primary reactor containment and associated systems is held within allowable leakage rate limits as specified in the technical specifications or associated bases of the license;

(b) Periodic surveillance is performed to assure proper maintenance and leak repair during the life of the containment; and

(c) The containment will continue to perform its function throughout the life of the plant.

The amendment which follows provides uniform requirements for containment leakage testing. It specifies the minimum requirements for periodic verification by tests of the leak-tight integrity of the primary reactor containment and associated systems for water-cooled power reactors, and the acceptance criteria for such tests.

Pursuant to the Atomic Energy Act of 1954, as amended, and sections 552 and 553 of title 5 of the United States Code, the following amendment to Title 10, Chapter I, Code of Federal Regulations, Part 50, is published as a document subject to codification to be effective on March 16, 1973.

1. A new paragraph (o) is added to § 50.54 to read as follows:

§ 50.54 Conditions of licenses.

* * * * *

(o) Primary reactor containments for water cooled power reactors shall be subject to the requirements set forth in Appendix J.

2. A new Appendix J is added to read as follows:

APPENDIX J

PRIMARY REACTOR CONTAINMENT LEAKAGE TESTING FOR WATER-COOLED POWER REACTORS

I. Introduction.
II. Explanation of terms.
III. Leakage test requirements.
A. Type A test.
B. Type B test.
C. Type C test.
D. Periodic retest schedule.
IV. Special test requirements.
A. Containment modification.
B. Multiple leakage-barrier containments.
V. Inspection and reporting of tests.
A. Containment inspection.
B. Report of test results.

I. INTRODUCTION

One of the conditions of all operating licenses for water-cooled power reactors as specified in § 50.54(o) is that primary reactor containments shall meet the containment leakage test requirements set forth in this appendix. These test requirements provide for preoperational and periodic verification by tests of the leak-tight integrity of the primary reactor containment, and systems and components which penetrate containment of water-cooled power reactors, and establish the acceptance criteria for such tests. [illegible] These test requirements may also be used for guidance in establishing appropriate containment leakage test requirements in technical specifications or associated bases for other types of nuclear power reactors.

II. EXPLANATION OF TERMS

A. "Primary reactor containment" means the structure or vessel that encloses the components of the reactor coolant pressure boundary, as defined in § 50.2(v), and serves as an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.

B. "Containment isolation valve" means any valve which is relied upon to perform a containment isolation function.

C. "Reactor containment leakage test program" includes the performance of Type A, Type B, and Type C tests, described in II.F, II.G, and II.H, respectively.

D. "Leakage rate" for test purposes is that leakage which occurs in a unit of time, stated as a percentage of weight of the original content of containment air at the leakage rate test pressure that escapes to the outside atmosphere during a 24-hour test period.

E. "Overall integrated leakage rate" means that leakage rate which obtains from a summation of leakage through all potential leakage paths including containment welds, valves, fittings, and components which penetrate containment.

F. "Type A Tests" means tests intended to measure the primary reactor containment overall integrated leakage rate (1) after the containment has been completed and is ready for operation, and (2) at periodic intervals thereafter.

G. "Type B Tests" means tests intended to detect local leaks and to measure leakage across each pressure-containing or leakage-limiting boundary for the following primary reactor containment penetrations:

1. Containment penetrations whose design incorporates resilient seals, gaskets, or sealant compounds, piping penetrations fitted with expansion bellows, and electrical penetrations fitted with flexible metal seal assemblies.

2. Air lock door seals, including door operating mechanism penetrations which are part of the containment pressure boundary.

3. Doors with resilient seals or gaskets except for seal-welded doors.

4. Components other than those listed in II.G.1, II.G.2, or II.G.3 which must meet the acceptance criteria in III.B.3.

H. "Type C Tests" means tests intended to measure containment isolation valve leakage rates. The containment isolation valves included are those that:

1. Provide a direct connection between the inside and outside atmospheres of the primary reactor containment under normal operation, such as purge and ventilation, vacuum relief, and instrument valves.

2. Are required to close automatically upon receipt of a containment isolation signal in response to controls intended to effect containment isolation;

3. Are required to operate intermittently under postaccident conditions; and

4. Are in main steam and feedwater piping and other systems which penetrate containment of direct-cycle boiling water power reactors.

I. P_a (p.s.i.g.) means the calculated peak containment internal pressure related to the design basis accident and specified either in the technical specifications or associated bases.

J. P_t (p.s.i.g.) means the containment vessel reduced test pressure selected to measure the integrated leakage rate during periodic Type A tests.

K. L_a (percent/24 hours) means the maximum allowable leakage rate at pressure P_a as specified for preoperational tests in the technical specifications or associated bases, and as specified for periodic tests in the operating license.

L. L_d (percent/24 hours) means the design leakage rate at pressure, P_a, as specified in the technical specifications or associated bases.

M. L_t (percent/24 hours) means the maximum allowable leakage rate at pressure P_t derived from the preoperational test data as specified in III.A.4.(a)(iii).

N. L_{am}, L_{tm} (percent/24 hours) means the total measured containment leakage rates at pressure P_a and P_t, respectively, obtained

14-4 (26)

trends. Whenever a bolted double-gasketed penetration is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized.

Table 3.7.4 identifies certain isolation valves that are tested by pressurizing the volume between the inboard and outboard isolation valves. This results in conservative test results since the inboard valve, if a globe valve, will be tested such that the test pressure is tending to lift the globe off its seat. Additionally, the measured leak rate for such a test is conservatively assigned to both of the valves equally and not divided between the two.

The main steam and feedwater testable penetrations consist of a double layered metal bellows. The inboard high pressure side of the bellows is subjected to drywell pressure. Therefore the bellows is tested in its entirety when the drywell is tested. The bellows layers are tested for the integrity of both layers by pressurizing the void between the layers to 5 psig. Any higher pressure could cause permanent deformation, damage and possible reptures of the bellows.

The primary containment pre-operational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The peak drywell pressure would be about 58 psig which would rapidly reduce to 29 psig following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig, equalizes with drywell pressure and therefore rapidly decays with the drywell pressure decay. The design pressure of the drywell and suppression chamber is 56 psig. Based on the calculated containment pressure response discussed above, the primary containment pre-operational test pressure was chosen. Also, based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 0.635%/day at 58 psig. Calculations made by the NRC staff with leak rate and a standby gas treatment system filter efficiency of 90% for halogens and assuming the fission product release fractions stated in NRC Regulatory Guide 1.3, show that the maximum total whole body passing cloud dose is about 1.0 REM and the maximum total thyroid dose is about 12 REM at 1100 meters from the stack over an exposure duration of two hours. The resultant doses reported are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site doses and 10 CFR 100 guidelines.

The water in the suppression chamber is used only for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily

N/1 ATT 4 (26)

3.7.A (cont'd)

4.7.A (cont'd)

repeated provided locally measured leakage reductions, achieved by repairs, reduced the containment's overall measured leakage rate sufficiently to meet the acceptance criteria.

f. With the exception of main steam isolation valves and main steam line and feedwater line bellows, (see below) local leak rate tests (LLRT's) shall be performed on the primary containment testable penetrations and isolation valves at a pressure of 58 psig during each reactor shutdown for refueling but in no case at intervals greater than two years. Bolted double-gasket seals shall be tested after each opening and during each reactor shutdown for refueling but in no case at intervals greater than two years.

* The main steam isolation valves (MSIV's) shall be tested a pressure of 29 psig. If a total leakage rate of 11.5 scf/hr for any one MSIV is exceeded, repair and retest shall be performed to correct the condition.

* Main steam line and feedwater line expansion bellows shall be tested at a pressure of 5 psig.

g. Continuous Leak Rate Monitor

When the primary containment is inerted, the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

h. Drywell Surfaces

The interior surfaces of the drywell and torus shall be visually inspected each operating cycle for evidence

* Exemption to Appendix J of 10 CFR 50.

14-4 (28)

PENETRATION NO. X-18

CHECKLIST of PACKAGE CONTENTS

#	Item	Initial By	Date
1.	Completed Design Basis Sheet	RES	5/20/94
2.	Sketch of Containment Barriers/Pathway	RES	5/20/94
3.	ISO # JELCO 2713-12 Rev. N01 (as applicable) JELCO X-2713-222	RES	5/20/94
4.	P & ID Instrument/System Drawings (as applicable) 2028 Rev. N26	RES	5/20/94
5.	Walkdown Instruction and Acceptance Criteria	RES	5/20/94
6.	Drawings Verified to be Latest Version	Kdy	5-21-94
7.	Other Contents: Iso Key 2028 Rev. N03 G8-2211 30 Rev N01 JELCO PT-2-B Rev. 5	RES	5/20/94

N || ATT 4 (28)

PENE. NO. X-18 CIV NO. RW-A094

DESIGN BASIS

VALVE FUNCTION: MO / AO / CV / MAN	LOCATION: AZ. / ELEV. - DRYWELL / REA. BLDG / TORUS 32° / 898' 9"	
DIV. SEPARATION: CCP1A 120VAC Div I	PCIS SIGNAL: YES / NO	GDC REQUIREMENTS: 1967 - 53 / 54 / 55 / 56 / 57 1971 - 54 / 55 / 56 / 57
STANDARDS: ANSI/ANS-52.1-1983 ANSI/ANS-56.2-1984 Section 3.6.5, Fig 1 ANSI/ANS-56.8-1987 Note: Not classic valve configuration		USAR KEY SECTIONS: V Section 2.0, Tab. V-2-2, V-2-7 VII Section 3.0 Tab. VII 3-1 Note: shows testable check valve ASME XI SAFETY CLASS: I / II / III / NA
APP. J-TYPE C TEST REQUIREMENTS & BASES: FROM CONTAINMENT / PARALLEL / REVERSE		COMMITMENTS: Tech Spec Table 3.7.1
NORMAL OPERATING POSITION: OPEN / CLOSED / NA	REFERENCES: GE 22A1132AB, Rev. 0, Section 3.2.1, App. A -Classified as 'B-1' Note: App. A shows normal ops open but B&R 2028 shows closed.	
FAIL POSITION: OPEN / CLOSED / NA		
DBA POSITION: OPEN / CLOSED / NA	CHECK BY: ____________ DATE ________ VERIFIED BY: ____________ DATE ________	

PENE. NO. X-18 CIV NO. RW-A095

DESIGN BASIS

VALVE FUNCTION:

MO / AO / CV / MAN

LOCATION:

AZ. / ELEV. - DRYWELL / REA. BLDG / TORUS
32° / 898' 9"

DIV. SEPARATION:

CCP1B 120VAC Div II

PCIS SIGNAL:

YES / NO

GDC REQUIREMENTS:

1967 - 53 / 54 / 55 / 56 / 57

1971 - 54 / 55 / 56 / 57

STANDARDS:

ANSI/ANS-52.1-1983
ANSI/ANS-56.2-1984 Section 3.6.5, Fig 1
ANSI/ANS-56.8-1987

Note: Not classic valve configuration

USAR KEY SECTIONS:

V Section 2.0, Tab. V-2-2, V-2-7
VII Section 3.0 Tab. VII 3-1
Note: shows testable check valve

ASME XI SAFETY CLASS:

I / II / III / NA

APP. J-TYPE C TEST REQUIREMENTS & BASES:

FROM CONTAINMENT / PARALLEL / REVERSE

COMMITMENTS:

Tech Spec Table 3.7.1

NORMAL OPERATING POSITION:

OPEN / CLOSED / NA

REFERENCES:

GE 22A1132AB, Rev. 0, Section 3.2.1, App. A
-Classified as "B"

Note: App. A shows normal ops open but B&R 2028 shows closed.

FAIL POSITION:

OPEN / CLOSED / NA

DBA POSITION:

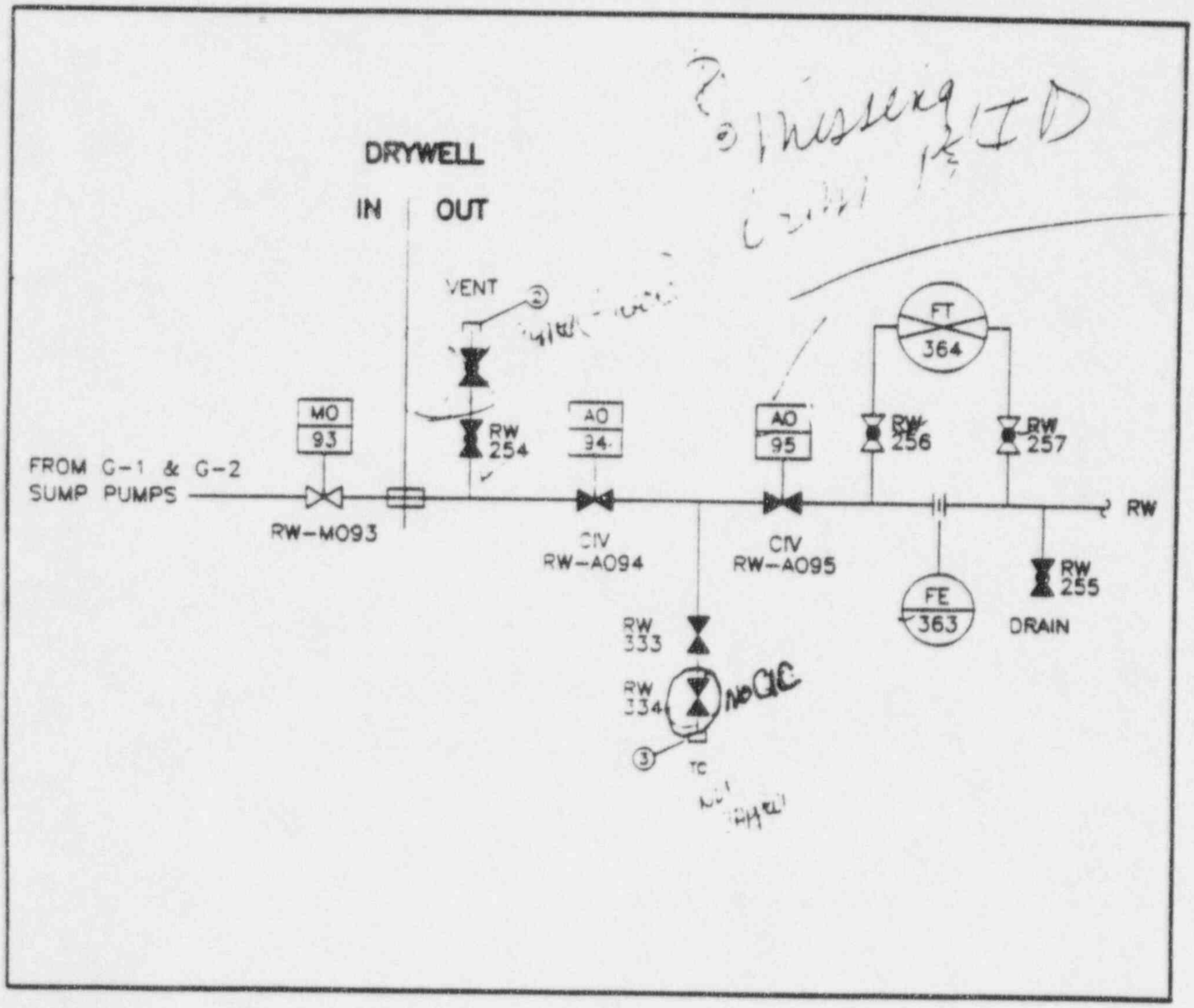
OPEN / CLOSED / NA

CHECK BY: ______________________ DATE ________

VERIFIED BY: ______________________ DATE ________

Procedure No.	Rev.	Page

PENETRATION X-18
DRYWELL EQUIPMENT SUMP DISCHARGE
CONTAINMENT ISOLATION VALVES
RW-AO94 & RW-AO95



NOTE	DESCRIPTION
1	FOR MORE INFORMATION SEE BURNS & ROE #2028 & 4262 (TYPE 3)
2	VENT POINT
3	TEST CONNECTION

PENETRATION NO. X-30E

CHECKLIST of PACKAGE CONTENTS

#	Item	Initial By	Date
1.	Completed Design Basis Sheet	JLS	5-23-94
2.	Sketch of Containment Barriers/Pathway	JLS	5-23-94
3.	ISO # X2714-200 R4 (as applicable) X2507-201 RN01	JLS	5-23-94
4.	P & ID Instrument/System Drawings (as applicable) 2028 RN27	JLS	5-23-94
5.	Walkdown Instruction and Acceptance Criteria	JLS	5-23-94
6.	Drawings Verified to be Latest Version	FEV	5-23-94
7.	Other Contents: * IL-E-70-3 Sht. 24 (I.D.18) R3 B&R 2028 R3	JLS	5-23-94.

~~* REV. NO 3 NOT AVAILABLE IN THE DRAWING SYSTEM AT CNS.~~ FEV 5/27/94

PENE. NO. X-30E CIV NO. NBI-502

DESIGN BASIS

VALVE FUNCTION:

MO / AO / CV / MAN

LOCATION:

AZ. / ELEV. - DRYWELL / REA. BLDG / TORUS
268° / 911' 6"

DIV. SEPARATION:

N/A

PCIS SIGNAL:

YES / NO

GDC REQUIREMENTS:

1967 - 53 / 54 / 55 / 56 / 57

1971 - 54 / 55 / 56 / 57
Note: Manual exterior valve closed, not in compliance

STANDARDS:

ANSI/ANS-56.2-1984, Section 3.6.2, & 4.8
Note: Isolation valves not needed but available manually

USAR KEY SECTIONS:

V Section 2.0, Table V-2-2

ASME XI SAFETY CLASS:

I / II / III / NA

APP. J-TYPE C TEST REQUIREMENTS & BASES:

FROM CONTAINMENT / PARALLEL / REVERSE

Note: Not Type C testable.

COMMITMENTS:

Tech Spec 3.7.A.3

NORMAL OPERATING POSITION:

OPEN / CLOSED / NA

FAIL POSITION:

OPEN / CLOSED / NA

REFERENCES:

Reg. Guide 1.11

DBA POSITION:

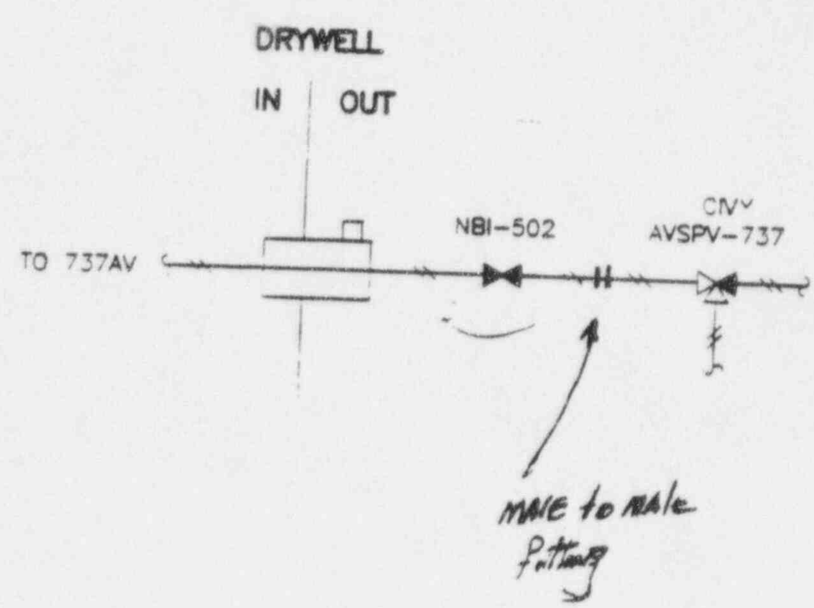
OPEN / CLOSED / NA

CHECK BY: ____________________ DATE __________

VERIFIED BY: ____________________ DATE __________

Procedure No.	Rev.	Page

PENETRATION X-30E
AIR TO VESSEL FLANGE LEAK-OFF DETECTION AOV CONTAINMENT ISOLATION VALVE
AVSPV-737



NOTE	DESCRIPTION
1	FOR MORE INFORMATION SEE BURNS & ROE #2028 & 4262 (TYPE 9)
	& IL-E-70-3 SHT 24
2	VENT POINT
3	TEST CONNECTION

PENETRATION NO. X-30F

CHECKLIST of PACKAGE CONTENTS

#	Item	Initial By	Date
1.	Completed Design Basis Sheet	JLS	5-23-94
2.	Sketch of Containment Barriers/Pathway	JLS	5-23-94
3.	ISO # (as applicable) X2506-204 R1	JLS	5-23-94
4.	P & ID Instrument/System Drawings (as applicable) 2028, RN27	JLS	5-23-94
5.	Walkdown Instruction and Acceptance Criteria	JLS	5-23-94
6.	Drawings Verified to be Latest Version	JLS	5-23-94
7.	Other Contents: B&R 2028 RN03 * IC-E-70-3 Sht. 24 (I.D. 18) R3	JLS	5-23-94

* ~~R3 not available in the bldg. Got from an ENS @ this time~~ JLS 5-23-94

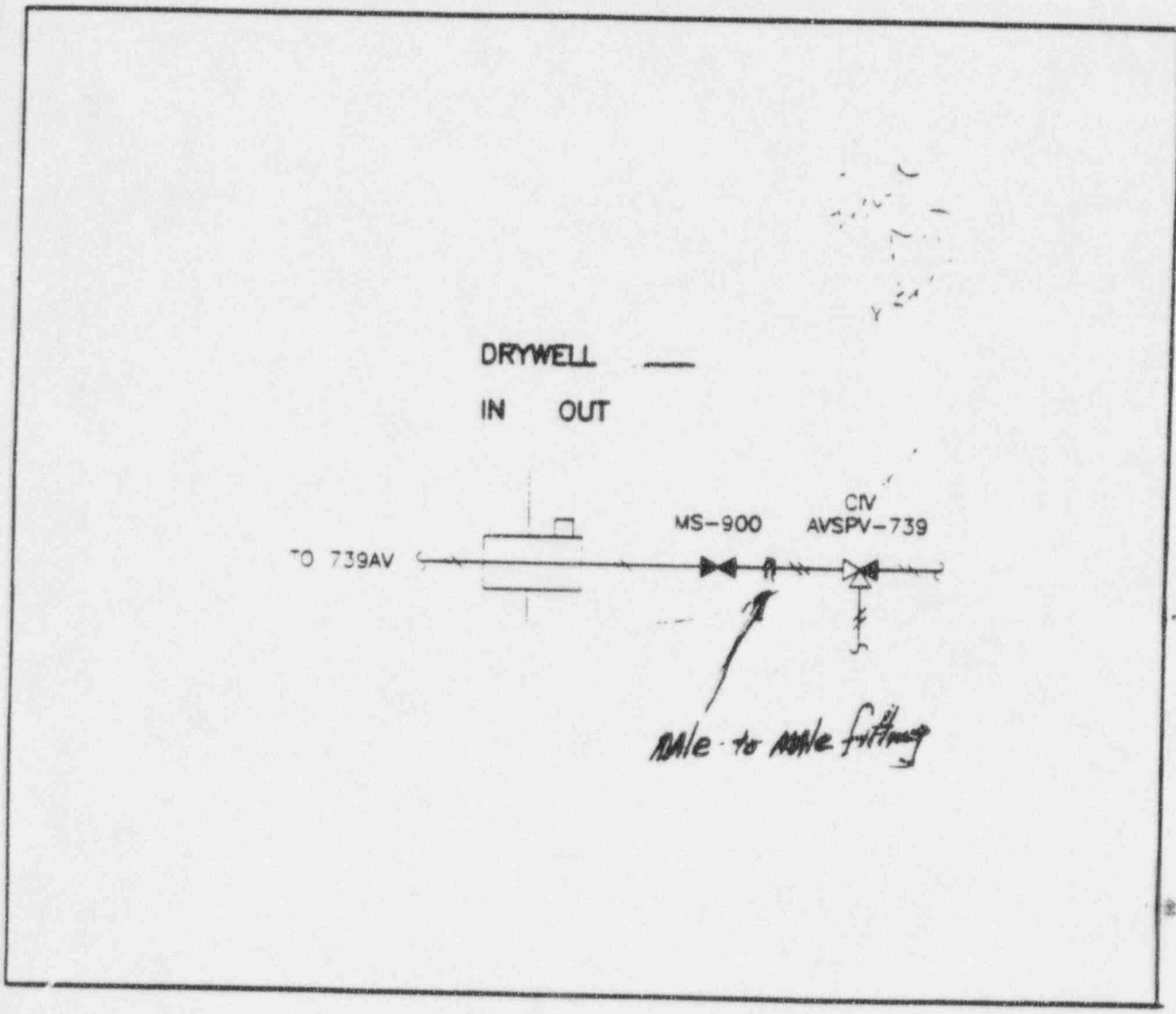
PENE. NO. X-30F CIV NO. MS-900

DESIGN BASIS

VALVE FUNCTION: MO / AO / CV / MAN	LOCATION: AZ. / ELEV. - DRYWELL / REA. BLDG / TORUS 268° / 911' 6"	
DIV. SEPARATION: N/A	PCIS SIGNAL: YES / NO	GDC REQUIREMENTS: 1967 - 53 / 54 / 55 / 56 / 57 1971 - 54 / 55 / 56 / 57 Note: Manual exterior valve closed, not in compliance
STANDARDS: ANSI/ANS-56.2-1984, Section 3.6.2, & 4.8 Note: Isolation valves not needed but available manually		USAR KEY SECTIONS: V Section 2.0, Table V-2-2 ASME XI SAFETY CLASS: I / II / III / NA
APP. J-TYPE C TEST REQUIREMENTS & BASES: FROM CONTAINMENT / PARALLEL / REVERSE Note: Not Type C testable.		COMMITMENTS: Tech Spec 3.7.A.3
NORMAL OPERATING POSITION: OPEN / CLOSED / NA	REFERENCES: Reg. Guide 1.11	
FAIL POSITION: OPEN / CLOSED / NA		
DBA POSITION: OPEN / CLOSED / NA	CHECK BY: ____________ DATE ______ VERIFIED BY: ____________ DATE ______	

Procedure No.	Rev.	Page

PENETRATION X-30F
AIR TO REACTOR VESSEL HEAD VENT
CONTAINMENT ISOLATION VALVE
ASPV-739



NOTE	DESCRIPTION
1	FOR MORE INFORMATION SEE BURNS & ROE #2028 & 4262 (TYPE 9)
	& IL-E-70-3 SHT 24
2	VENT POINT
3	TEST CONNECTION

PENETRATION NO. X-33E

CHECKLIST of PACKAGE CONTENTS

#	Item	Initial By	Date
1.	Completed Design Basis Sheet	RES	5/21/94
2.	Sketch of Containment Barriers/Pathway	RES	5/21/94
3.	ISO # JELCO X-2507-201 Rev. 001 (as applicable) JELCO X-2714-200 Rev 4	RES	5/21/94
4.	P & ID Instrument/System Drawings (as applicable) 2028 Rev 026	RES	5/21/94
5.	Walkdown Instruction and Acceptance Criteria	RES	5/21/94
6.	Drawings Verified to be Latest Version	PGA	5/23/94
7.	Other Contents: Iso Key 2028 003 IL-E-70-3 ID14 Rev. 001	RES	5/21/94

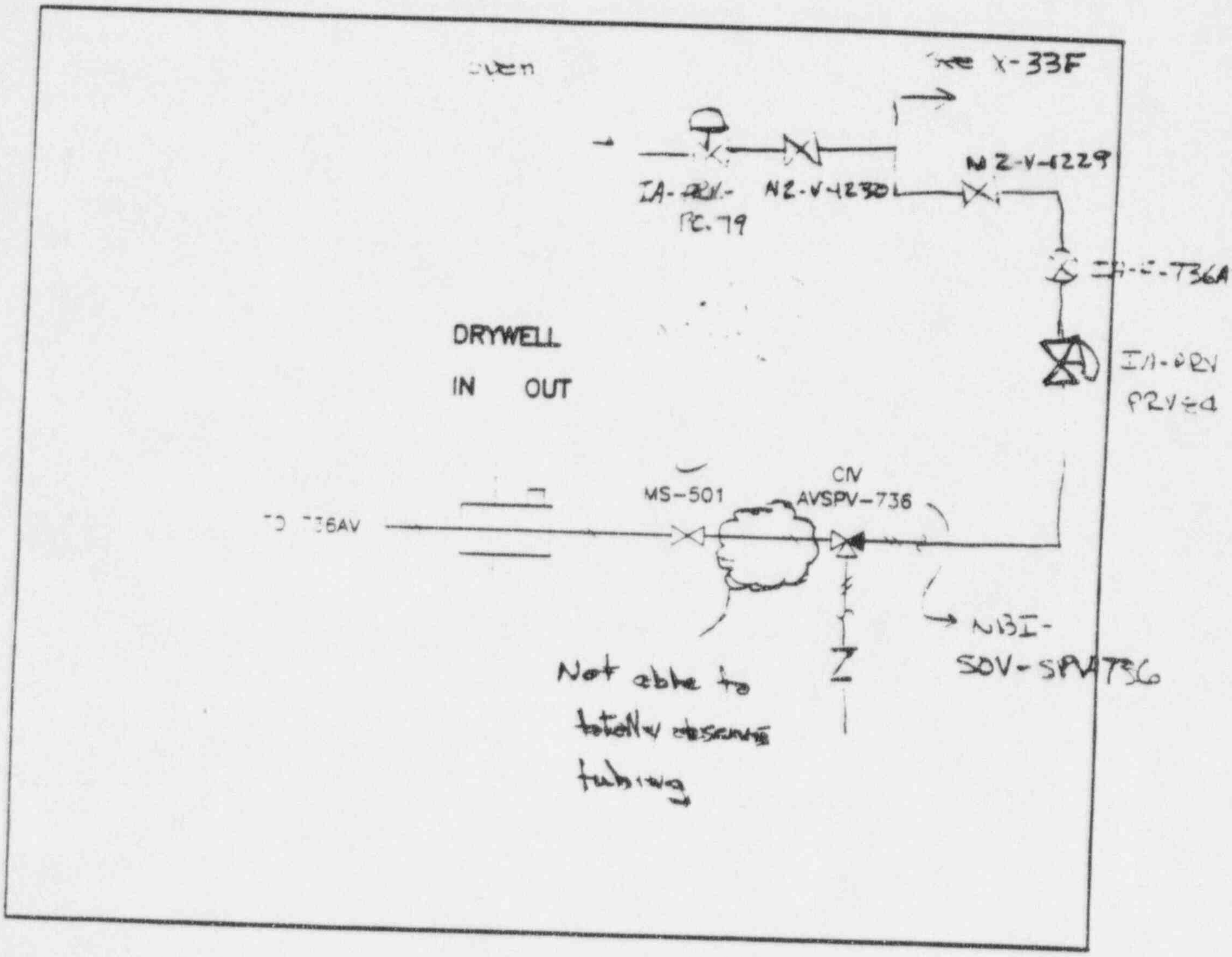
PENE. NO. X-33E CIV NO. MS-501

DESIGN BASIS

VALVE FUNCTION: MO / AO / CV / **MAN**	LOCATION: AZ. 55° ELEV. 898' 9" - DRYWELL / **REA. BLDG** / TORUS	
DIV. SEPARATION: N/A	PCIS SIGNAL: YES **NO**	**GDC REQUIREMENTS:** 1967 - **53 / 54 / 55** / 56 / 57 1971 - 54 / 55 / **56** / 57
STANDARDS: **ANSI/ANS-56.2-1984, Section 3.6.2. & 4.8** **Note: Isolation valves not needed but available manually**		USAR KEY SECTIONS: **V Section 2.0, Table V-2-2** ASME XI SAFETY CLASS: I / **II** / III / NA
APP. J-TYPE C TEST REQUIREMENTS & BASES: FROM CONTAINMENT / PARALLEL REVERSE **Note: Not Type C testable.**		COMMITMENTS: **Tech Spec 3.7.A.3**
NORMAL OPERATING POSITION: **OPEN** / CLOSED NA FAIL POSITION: **OPEN** / CLOSED NA	REFERENCES: **Reg. Guide 1.11**	
DBA POSITION: **OPEN** CLOSED NA	CHECK BY: ____________ DATE ________ VERIFIED BY: ____________ DATE ________	

Procedure No.	Rev.	Page

PENETRATION X-33E
AIR TO VESSEL FLANGE LEAK-OFF DETECTION AOV CONTAINMENT ISOLATION VALVE
AVSPV-736



NOTE	DESCRIPTION
1	FOR MORE INFORMATION SEE BURNS & ROE #2028 & 4262 (TYPE 9)
	& IL-E-70-3 SHT 20
2	VENT POINT
3	TEST CONNECTION

PENETRATION NO. X-33F

CHECKLIST of PACKAGE CONTENTS

#	Item	Initial By	Date
1.	Completed Design Basis Sheet	RES	5/21/94
2.	Sketch of Containment Barriers/Pathway	RES	5/21/94
3.	ISO # JELCO X-2506-204 Rev L01 (as applicable)	RES	5/21/94
4.	P & ID Instrument/System Drawings (as applicable) 2028 Rev N26	RES	5/21/94
5.	Walkdown Instruction and Acceptance Criteria	RES	5/21/94
6.	Drawings Verified to be Latest Version	RES	5/23/94
7.	Other Contents: Iso Key 2028 Rev N03 IL-70-3 ID 14 Rev N01	RES	5/21/94

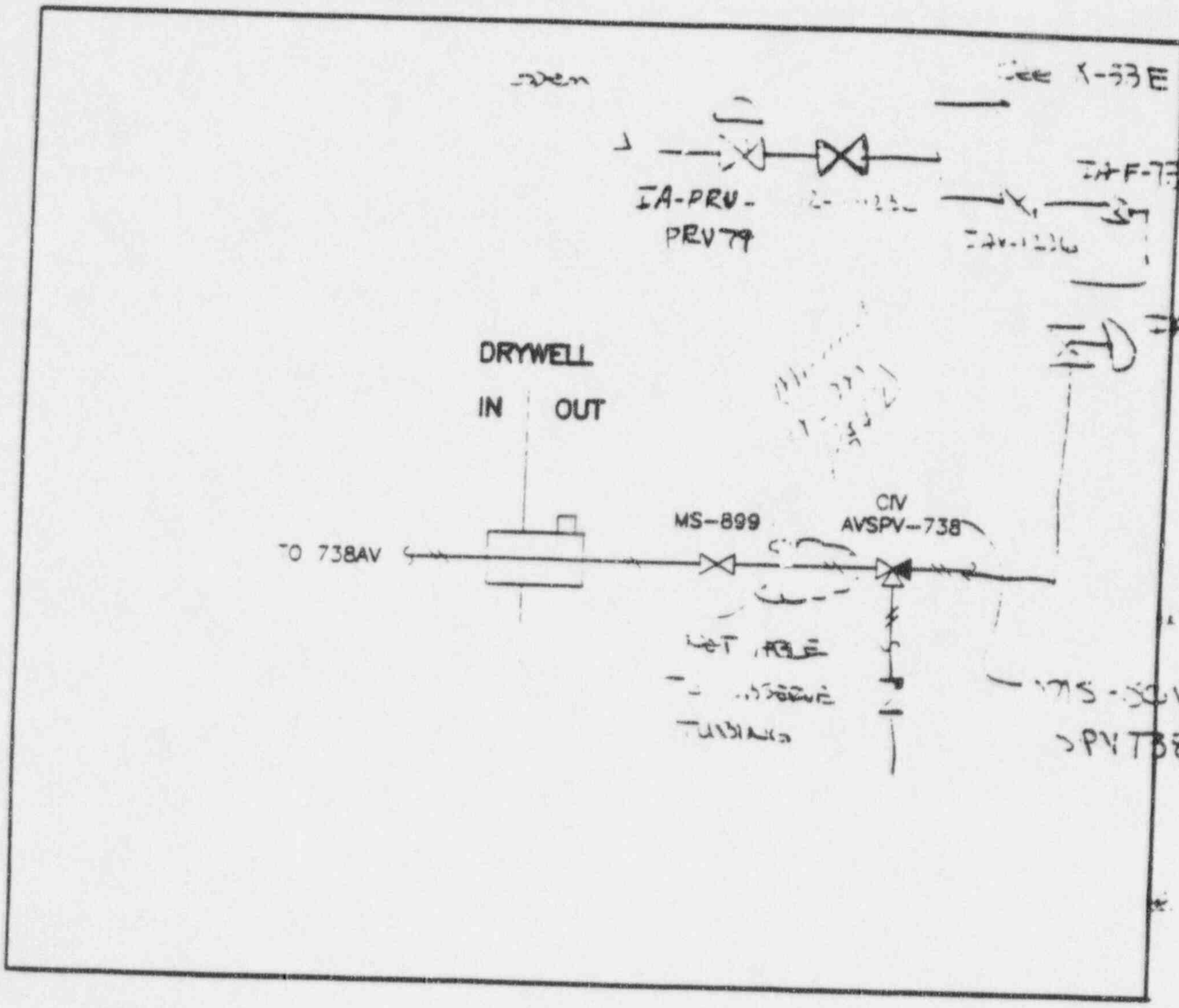
PENE. NO. X-33F CIV NO. MS-899

DESIGN BASIS

VALVE FUNCTION: MO / AO / CV / MAN	LOCATION: AZ. / ELEV. - DRYWELL / REA. BLDG / TORUS 55° / 898' 9"	
DIV. SEPARATION: N/A	PCIS SIGNAL: YES / NO	GDC REQUIREMENTS: 1967 - 53 / 54 / 55 / 56 / 57 1971 - 54 / 55 / 56 / 57
STANDARDS: **ANSI/ANS-56.2-1984, Section 3.6.2, & 4.8** **Note: Isolation valves not needed but available manually**		USAR KEY SECTIONS: **V Section 2.0, Table V-2-2** ASME XI SAFETY CLASS: I / II / III / NA
APP. J-TYPE C TEST REQUIREMENTS & BASES: FROM CONTAINMENT / PARALLEL / REVERSE **Note: Not Type C testable.**		COMMITMENTS: **Tech Spec 3.7.A.3**
NORMAL OPERATING POSITION: OPEN / CLOSED / NA	REFERENCES: **Reg. Guide 1.11**	
FAIL POSITION: OPEN / CLOSED / NA		
DBA POSITION: OPEN / CLOSED / NA	CHECK BY: ____________ DATE ______ VERIFIED BY: ____________ DATE ______	

Procedure No.	Rev.	Page

PENETRATION X-33F
AIR TO REACTOR VESSEL HEAD VENT
CONTAINMENT ISOLATION VALVE
AVSPV-738



NOTE	DESCRIPTION
1	FOR MORE INFORMATION SEE BURNS & ROE #2028 & 4262 (TYPE 9)
	& IL-E-70-3 SHT 20
2	VENT POINT
3	TEST CONNECTION

x33F

PENETRATION NO. X-45D

CHECKLIST of PACKAGE CONTENTS

#	Item	Initial By	Date
1.	Completed Design Basis Sheet	KRG	5-24-94
2.	Sketch of Containment Barriers/Pathway	KRG	5-24-94
3.	ISO # ________ (as applicable) ________	N/A	N/A
4.	P & ID Instrument/System Drawings (as applicable) 2028 Rev ~~N26~~ N27	KRG	5-24-94
5.	Walkdown Instruction and Acceptance Criteria	KRG	5-24-94
6.	Drawings Verified to be Latest Version	KRG	5-24-94
7.	Other Contents: ISO Key #2028 Rev. 3	KRG.	5-24-94

PENE. NO. <u>X-45D</u> CIV NO. <u>UNKNOWN</u>

DESIGN BASIS

VALVE FUNCTION:

MO / AO / <u>CV</u> / MAN

LOCATION:

AZ. / ELEV. - DRYWELL / <u>REA. BLDG.</u> / TORUS
250° / 919' 1"

DIV. SEPARATION:

N/A

PCIS SIGNAL:

YES / <u>NO</u>

REV. FLOW

GDC REQUIREMENTS:

1967 - <u>53 / 54 / 55</u> / 56 / <u>57</u>

1971 - 54 / 55 / <u>56</u> / 57

STANDARDS:

56.2, Not in compliance, need solenoids.
56.8 - 1987

USAR KEY SECTIONS:
V 5A 2.0, Tab. V-2-2

ASME XI SAFETY CLASS:

I / II / III / <u>NA</u>
UNKNOWN/ NOT SHOWN

APP. J-TYPE C TEST REQUIREMENTS & BASES:

FROM CONTAINMENT / PARALLEL / REVERSE
Note: Not in LLRT Program but should be.

COMMITMENTS:

NOT AVAILABLE

NORMAL OPERATING POSITION:

OPEN / <u>CLOSED</u> / NA

REFERENCES:

NONE

FAIL POSITION:

OPEN / <u>CLOSED</u> / NA

DBA POSITION:

OPEN / <u>CLOSED</u> / NA

CHECK BY: ____________________ DATE ________

VERIFIED BY: ____________________ DATE ________

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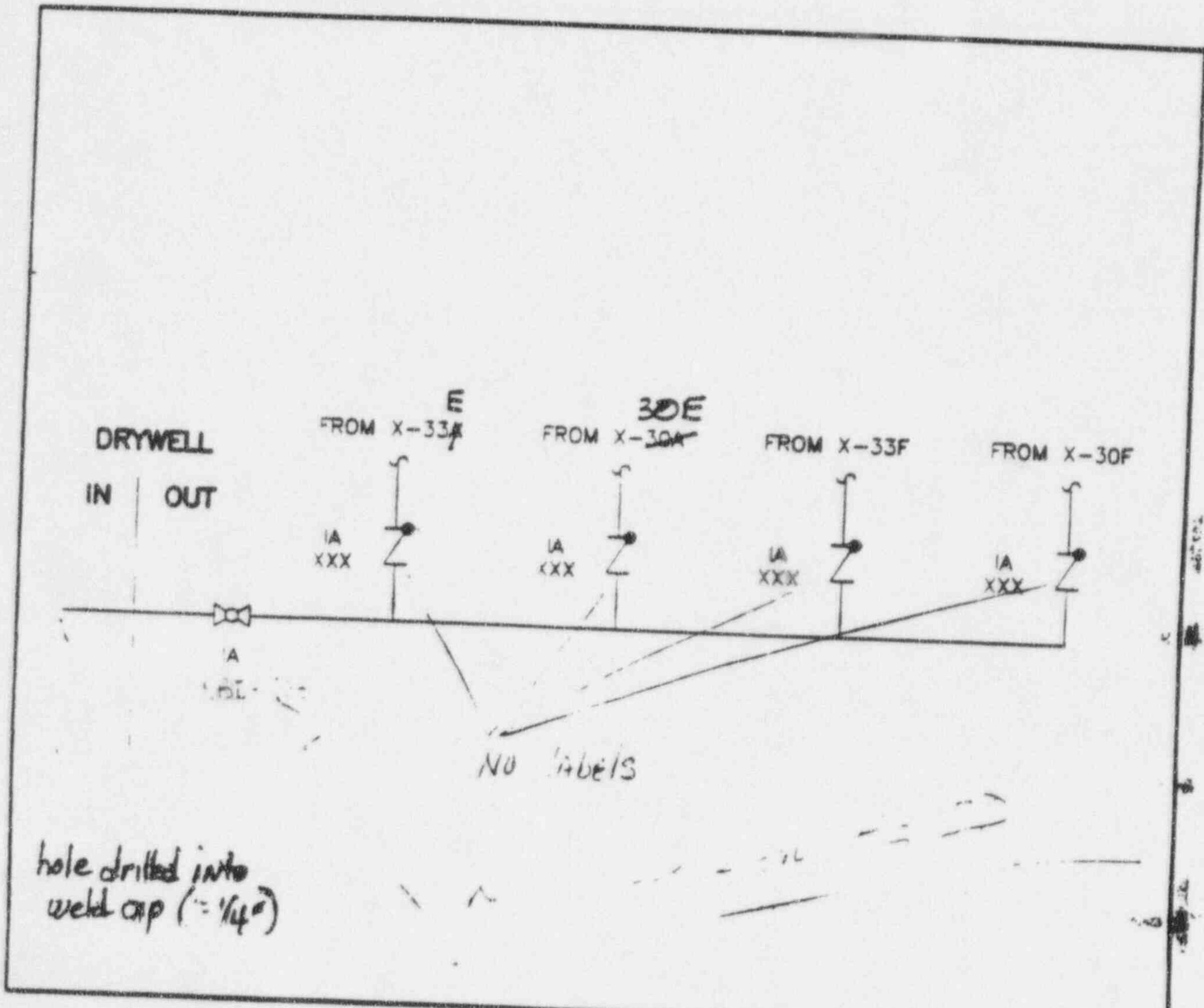
PENETRATION X-45D
SOV AIR EXHAUST TO DRYWELL
CONTAINMENT ISOLATION VALVES
IA



NOTE	DESCRIPTION
1	FOR MORE INFORMATION SEE BURNS & ROE #2028
2	VENT POINT
3	TEST CONNECTION

AS FOUND TOTALS → (NRW PENETRATIONS)

	MAX PTH	MIN PTH (WHERE APPLICABLE)	
X-20	0.284	0.24	
X-21	0.986	0.986	
X-22	< 600	< 600	
X-23	2.67	NA	
X-24	0.2	NA	
X-35A			
X-35B	0.3	0.3	
X-35C	0.18	0.12	
X-35D			
X-35E		0.0 *	1 ASFOUND LEFT
X-214	0.63	0.63	18RV
X-214	0.0	0.0	19RV
X-214	1.3	1.3	20RV
X-214	0.0	0.0	21RV
X-51F	0.53	0.0	
X-43	0.0	0.0	} – AS LEFT FROM DEC. OUTAGE (AS-FOUND IN DEC WAS 10 SCFH TOTAL FOR BOTH
X-44	0.0	0.0	
X-209A	0.0	0.0	
X-209B	0.0	0.0	
X-209C	0.0	0.0	
X-209D	~~0.0~~ ~~0.032~~ 0.032	~~0.0~~ 0.032	
X-218	0.078	0.078	
X-29E	0.0	0.0	
X-51B	0.0	0.0	
X-30E	2.7	2.7	
X-30F	2.14	2.14	
X-33E	2.03	2.03	
X-33F	2.41	2.41	
X-45D	0.0	0.0	
X-46A	0.0	0.0	
X-46B	0.0	0.0	
X-46C	0.0	0.0	
X-46D	0.0	0.0	
X-46E	0.0	0.0	
X-46F	0.0	0.0	
X-47A	0.0	0.0	
X-47C	0.0	0.0	
X-47D	0.0	0.0	
X-47F	0.0	0.0	
X-47E	0.0	0.0	
X-100B	0.0		

	MX PTH	MIN PT	
X-229A	0.08	0.08	
X-229B	0.08	0.08	
X-229C	0.16	0.16	
X-229D	0.08	0.08	
X-229E	0.08	0.08	
X-229F	0.16	0.16	
X-229G	0.16	0.16	
X-229H	0.08	0.08	
X-229J	0.08	0.08	
X-229K	0.08	0.08	
X-229L	0.16	0.16	
X-229M	0.0	0.0	
X-40A	0.034	0.034	12A
X-40B	0.102	0.102	12B
X-40C	0.0	0.0	12C
X-40D	0.0	0.0	12D
X-40A	0.47	0.47	101A
X-40B	0.27	0.27	101B
X-40C	0.068	0.068	101C
X-40D	0.34	0.34	101D
X-40D	0.1	0.1	119D
X-40B	3.24	3.24	119A
X-40B	1.21	1.21	119C
X-40D	0.0	0.0	119B
X-40A	0.068	0.068	16
X-40A	0.0	0.0	512A
X-40C	0.0	0.0	512B
TOTAL	623.57 SCFH	NA	

AS OF CLOSE OF BUSINESS 7/11/94

CURRENT AS LEFT TOTAL - (MAX PATH)

181.50 SCFH (FROM 3/24 START UP)
- 38.501 SCFH (PENETRATIONS DELETED FROM APP J TOTALS)
142.998 SCFH
- 1.46 SCFH - OLD X-212 TEST
141.538 SCFH
+ 0.22 SCFH - NEW X-212 TEST
141.758 SCFH
- 17.1 SCFH - OLD RHR-MO39B TEST
124.658 SCFH
+ 23.3 SCFH - NEW RHR-MO39B TEST
147.958 SCFH
- 5.51 SCFH - REMOVING RHR-MO27A (OLD X-18B MAX PATH)
142.448 SCFH
+ 1.80 SCFH - ADD RHR-27CV (NEW MAX PATH)
144.248 SCFH
- 15.45 SCFH - OLD X-100A TEST
128.798 SCFH
+ 19.56 - NEW X-100A TEST
148.35 SCFH
+ 0.284 SCFH - NEW X-20 MAX PATH
0.18 SCFH - NEW X-21 MAX PATH
0.24 SCFH - NEW X-22 MAX PATH
2.67 SCFH - NEW X-23 MAX PATH
0.2 SCFH - NEW X-24 MAX PATH
0.2 SCFH - NEW X-214 MAX PATH (18RV)
0.0 SCFH NEW X-214 MAX PATH (19RV)
0.0 SCFH NEW X-214 MAX PATH (20RV)
0.0 SCFH NEW X-214 MAX PATH (21RV)
0.13 SCFH NEW X-29E MAX PATH
0.085 SCFH NEW X-51F MAX PATH
0.0 SCFH NEW X-30E MAX PATH
0.0 SCFH NEW X-30F MAX PATH
0.17 SCFH NEW X-33E MAX PATH
0.0 SCFH NEW X-45D MAX PATH
0.0 SCFH NEW X-229A MAX PATH
0.0 SCFH NEW X-229B MAX PATH
0.0 SCFH NEW X-229C MAX PATH
0.27 SCFH NEW X-229D MAX PATH
0.0 SCFH NEW X-229E MAX PATH
0.0 SCFH NEW X-229F MAX PATH
0.0 SCFH NEW X-229G MAX PATH
0.0 SCFH NEW X-229H MAX PATH
0.0 SCFH NEW X-229J MAX PATH
152.779 SCFH

→ CONTINUED ON NEXT PG.

152.779 SCFH

+ 0.0	SCFH	NEW	X-229J	MAX PATH	
0.0	SCFH	NEW	X-229K	MAX PATH	
0.0	SCFH	NEW	X-229L	MAX PATH	
0.182	SCFH	NEW	X-35A	MAX PATH	
0.182	SCFH	NEW	X-35B	MAX PATH	
0.182	SCFH	NEW	X-35C	MAX PATH	
0.193	SCFH	NEW	X-35D	MAX PATH	
0.0	SCFH	NEW	X-43	MAX PATH	
0.0	SCFH	NEW	X-44	MAX PATH	
0.06	SCFH	NEW	X-51B	MAX PATH	
0.0	SCFH	NEW	X-209A	MAX PATH	
0.0	SCFH	NEW	X-209C	MAX PATH	
0.034	SCFH	NEW	X-40A	MAX PATH	PC-12A
0.102	SCFH	NEW	X-40B	MAX PATH	PC-12B
0.0	SCFH	NEW	X-40C	MAX PATH	PC-12C
0.0	SCFH	NEW	X-40D	MAX PATH	PC-12D
0.47	SCFH	NEW	X-40A	MAX PATH	PC-101A
0.27	SCFH	NEW	40B	MAX PATH	PC-101B
0.068	SCFH	NEW	X-40C	MAX PATH	PC-101C
0.34	SCFH	NEW	X-40D	MAX PATH	PC-101D
0.10	SCFH	NEW	X-40D	MAX PATH	PC-119D
0.0	SCFH	NEW	X-40B	MAX PATH	PC-119A
0.0	SCFH	NEW	X-40B	MAX PATH	PC-119C
0.0	SCFH	NEW	X-40D	MAX PATH	PC-119B
0.068	SCFH	NEW	X-40A	MAX PATH	PC-16
0.0	SCFH	NEW	X-40A	MAX PATH	PT-512A
0.0	SCFH	NEW	X-40C	MAX PATH	PT-512B

155.03 SCFH - CURRENT AS LEFT TOTAL AS OF 1915 hrs 7/11/84

TESTING LEFT -

X-214	-	HPCI - 44
X-211B	-	RHR - MO35B ?
X-35E	-	NM - SOV - SPV2
X-6	-	CRD HATCH
X-200A	-	TORUS HATCH
X-2	-	OW AIRLOCK
X-33F	-	PC-565
X-33F	-	PC-566

TABLE 1 - TYPE C LLRT PENETRATION TESTS

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft³	AS FOUND LEAKAGE scfh	OUTBD /INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh		INITIAL /DATE
							RECOMMENDED	TECH SPEC	
X-8	MS-MOV-MO74 MS-MOV-MO77	Main Steam Line Drain	1.12	[redacted]		[redacted]	≤ 1.875	[redacted]	
X-9A	RF-CV-16CV**	Reactor Feedwater - Inboard	29.39	[redacted]		[redacted]	≤ 11.25	[redacted]	
X-9A	RF-CV-15CV** RCIC-CV-26CV RWCU-CV-15CV**	RCIC And RWCU Connection To Reactor Feedwater	59.16	[redacted]		[redacted]	≤ 11.25	[redacted]	
X-9B	RF-CV-14CV**	Reactor Feedwater - Inboard	29.34	[redacted]		[redacted]	≤ 11.25	[redacted]	
X-9B	RF-CV-13CV** HPCI-CV-29CV	HPCI Connection To Reactor Feedwater	63.93	[redacted]		[redacted]	≤ 11.25	[redacted]	
X-10	RCIC-MOV-MO15 RCIC-MOV-MO16	RCIC Steam Supply	2.18	[redacted]		[redacted]	≤ 1.875	[redacted]	
X-11	HPCI-MOV-MO15 HPCI-MOV-MO16	HPCI Steam Supply	18.3	[redacted]		[redacted]	≤ 6.25	[redacted]	
X-12	RHR-MOV-MO17	RHR Shutdown Cooling	208.19	[redacted]		[redacted]	≤ 12.5	[redacted]	
X-12	RHR-MOV-MO18	RHR Shutdown Cooling	34.3(1) 155.87(2)	[redacted]		[redacted]	≤ 12.5	[redacted]	

1 Method 1
2 Method 2
* If determined.
** Satisfactory completion of leak testing also satisfies closure test requirements of the CNS IST Program.

14-4 (29)

ATT 4 N/ (29)

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft³	AS FOUND LEAKAGE scfh	OUTBD /INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh		INITIAL /DATE
							RECOMMENDED	TECH SPEC	
X-13A	RHR-MOV-MO25A	RHR Loop A Injection	355.51	[redacted]		[redacted]	≤ 15.0	[redacted]	
X-13A	RHR-MOV-MO27A	RHR Loop A Injection	15.2(1) 350.33(2)	[redacted]		[redacted]	≤ 15.0	[redacted]	
X-13B	RHR-MOV-MO25B	RHR Loop B Injection	338.92	[redacted]		[redacted]	≤ 15.0	[redacted]	
X-13B	RHR-MOV-MO27B	RHR Loop B Injection	15.2(1) 353.74(2)	[redacted]	NA	[redacted]	≤ 15.0	[redacted]	[illegible] 6/21/94
X-14	RWCU-MOV-MO15	RWCU System Supply	1.577	[redacted]		[redacted]	≤ 3.75	[redacted]	
X-14	RWCU-MOV-MO18	RWCU System Supply	9.522	[redacted]		[redacted]	≤ 3.75	[redacted]	
X-16A	CS-MOV-MO11A CS-MOV-MO12A	CS Loop A Injection	3.0	[redacted]		[redacted]	≤ 6.25	[redacted]	
X-16B	CS-MOV-MO11B CS-MOV-MO12B	CS Loop B Injection	3.0	[redacted]		[redacted]	≤ 6.25	[redacted]	
X-18	RW-AOV-AO94	Drywell Equipment Drain Sump Discharge	3.054	[redacted]		[redacted]	≤ 1.875	[redacted]	
X-18	RW-AOV-AO95	Drywell Equipment Drain Sump Discharge	0.195	[redacted]		[redacted]	≤ 1.875	[redacted]	

TPCN 94-15

1 Method 1
2 Method 2
* If determined.
** Satisfactory completion of leak testing also satisfies closure test requirements of the CNS IST Program.

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft³	AS FOUND LEAKAGE scfh	OUTBD /INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh RECOMMENDED	ALLOWABLE LIMITS scfh TECH SPEC	INITIAL /DATE
X-19	RW-AOV-AO82	Drywell Equipment Drain Sump Discharge	1.470	[redacted]		[redacted]	≤ 1.875	[redacted]	
X-19	RW-AOV-AO83	Drywell Equipment Drain Sump Discharge	0.196	[redacted]		[redacted]	≤ 1.875	[redacted]	
X-25	PC-MOV-232MV PC-AOV-238AV	Drywell Purge And Vent Supply	8.49	[redacted]		[redacted]	≤ 15.0	[redacted]	
X-25	PC-MOV-1305MV PC-MOV-1306MV	Drywell Dilution Supply Isolation Valves - Train A	0.028	[redacted]		[redacted]	≤ 0.625	[redacted]	
X-26	PC-MOV-231MV PC-AOV-246AV PC-MOV-306MV PC-MOV-1310MV	Drywell Purge And Vent Exhaust	9.27	[redacted]		[redacted]	≤ 15.0	[redacted]	
X-39A	RHR-MOV-MO26A RHR-MOV-MO31A	Drywell Spray Loop A	36.7	[redacted]		[redacted]	≤ 6.25	[redacted]	
X-39B	RHR-MOV-MO26B RHR-MO-MO31B	Drywell Spray Loop B	36.7	[redacted]		[redacted]	≤ 6.25	[redacted]	
X-39B	PC-MOV-1311MV PC-MOV-1312MV	Drywell Dilution Supply Isolation Valves - Train B	0.038	[redacted]		[redacted]	≤ 0.625	[redacted]	
X-41	RR-AOV-740AV RR-AOV-741AV	Reactor Water Sample	0.07	[redacted]		[redacted]	≤ 0.469	[redacted]	
X-42	SLC-CV-12CV**	Standby Liquid Control Injection	0.03	[redacted]		[redacted]	≤ 0.9375	[redacted]	

* If determined.

** Satisfactory completion of leak testing also satisfies closure test requirements of the CNS IST Program.

PRIMARY CONTAINMENT LLRT TEST RESULTS

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft^3	AS FOUND LEAKAGE scfh	OUTBD /INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh RECOMMENDED	ALLOWABLE LIMITS scfh TECH SPEC	INITIAL /DATE
	SLC-CV-13CV**		0.88	[redacted]		[redacted]	≤ 0.9375	[redacted]	
X-37C	CRD-CV-13CV**	CRD Mini-Purge To RR Pump A	0.003	[redacted]		[redacted]	≤ 0.469	[redacted]	
	CRD-CV-14CV**		0.200	[redacted]		[redacted]	≤ 0.469	[redacted]	
X-38C	CRD-CV-15CV**	CRD Mini-Purge to RR Pump B	0.003	[redacted]		[redacted]	≤ 0.469	[redacted]	
	CRD-CV-16CV**		0.223	[redacted]		[redacted]	≤ 0.469	[redacted]	
X-51E	RMV-AOV-10AV	Drywell Vent Monitor Supply Isolation Valves	0.0683	[redacted]		[redacted]	≤ 0.469	[redacted]	
	RMV-AOV-11AV		0.0951	[redacted]		[redacted]	≤ 0.469	[redacted]	
X-45C	RMV-AOV-12AV	Drywell Vent Monitor Return Isolation Valves	0.0763	[redacted]		[redacted]	≤ 0.469	[redacted]	
	RMV-AOV-13AV		0.0983	[redacted]		[redacted]	≤ 0.469	[redacted]	
X-205	PC-MOV-233MV PC-AOV-237AV	Suppression Chamber Purge And Vent Supply	2.53	[redacted]		[redacted]	≤ 15.0	[redacted]	
X-205	PC-AOV-243AV PC-CV-13CV	Suppression Chamber Vacuum Relief	2.53	[redacted]		[redacted]	≤ 12.5	[redacted]	
X-205	PC-AOV-244AV PC-CV-14CV	Suppression Chamber Vacuum Relief	2.53	[redacted]		[redacted]	≤ 12.5	[redacted]	

* If determined.

** Satisfactory completion of leak testing also satisfies closure test requirements of the CNS IST Program.

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft³	AS FOUND LEAKAGE scfh	OUTBD /INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh		INITIAL /DATE	
							RECOMMENDED	TECH SPEC		
X-205	PC-MOV-1303MV PC-MOV-1304MV	Suppression Chamber Dilution Supply Isolation Valves - Train A	0.033	[redacted]		[redacted]	≤ 0.625	[redacted]		
X-210A	RCIC-MOV-MO27 RCIC-CV-13CV**	RCIC Minimum Flow	0.065	[redacted]		[redacted]	≤ 1.25	[redacted]		
X-210A	RHR-MOV-MO21A	RHR HX A Drain To Suppression Chamber	0.09	[redacted]		[redacted]	≤ 1.25	[redacted]		
X-210A	RHR-MOV-MO16A RHR-CV-10CV RHR-CV-12CV	RHR Loop A Minimum Flow	3.6	[redacted]		[redacted]	≤ 2.5	[redacted]		
X-210B	HPCI-MOV-MO25 HPCI-CV-17CV**	HPCI Minimum Flow	0.17	[redacted]		[redacted]	≤ 2.5	[redacted]		
X-210B	RHR-MOV-MO21B	RHR HX B Drain To Suppression Chamber	0.09	[redacted]		[redacted]	≤ 1.25	[redacted]		
X-210B	RHR-MOV-MO16B RHR-CV-11CV RHR-CV-13CV	RHR Loop B Minimum Flow	3.48	[redacted]	NA	[redacted]	≤ 2.5	[redacted]	DAS 6/28/94	TPCN 94-13
X-210A X-211A	DAS 6/28/94 ~~RHR-MOV-MO34A~~ RHR-MOV-MO38A ~~RHR-MOV-MO39A~~	RHR Loop A To Suppression Chamber	10.3	[redacted]	NA	[redacted]	≤ 11.25	[redacted]	DAS 6/28/94	TPCN 94-
X-210B X-211B	DAS 6/28/94 ~~RHR-MOV-MO34B~~ RHR-MOV-MO38B ~~RHR-MOV-MO39B~~	RHR Loop B To Suppression Chamber	10.3	[redacted]	NA	[redacted]	≤ 11.25	[redacted]	DAS 6/28/94 DAS 7/8/94 TPCN 94-161	TPCN 94-150 TPCN 94-156 CMD394

* If determined.

** Satisfactory completion of leak testing also satisfies closure test requirements of the GNS IST Program.

ATTACHMENT 5

PRIMARY CONTAINMENT LLRT TEST RESULTS

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft³	AS FOUND LEAKAGE scfh	OUTBD /INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh RECOMMENDED	ALLOWABLE LIMITS scfh TECH SPEC	INITIAL /DATE
X-211B	PC-MOV-1301MV PC-MOV-1302MV	Suppression Chamber Dilution Supply Isolation Valves - Train B	0.028	[redacted]		[redacted]	≤ 0.625	[redacted]	
X-210A & 211A X-210B & 211B	RHR-MOV-M067	RHR Discharge to Radwaste Surge Tank	0.033	[redacted]		[redacted]	≤ 1.25	[redacted]	
X-210A & 211A X-210B & 211B	RHR-MOV-M057	RHR Discharge to Radwaste Surge Tank	0.075	[redacted]		[redacted]	≤ 1.25	[redacted]	
X-212	RCIC-CV-15CV** RCIC-V-37**	RCIC Turbine Exhaust	1.26	[redacted]	NA	[redacted]	≤ 10.0	[redacted]	[illegible] 94-159
X-214	HPCI-CV-15CV** HPCI-V-44**	HPCI Turbine Exhaust	17.7	[redacted]	NA	[redacted]	≤ 25.0	[redacted]	[illegible] 94-159
X-214	RHR-MOV-MO166A RHR-MOV-MO167A	RHR HX A Vent	0.044	[redacted]		[redacted]	≤ 0.625	[redacted]	
X-214	RHR-MOV-MO166B RHR-MOV-MO167B	RHR HX B Vent	0.039	[redacted]		[redacted]	≤ 0.625	[redacted]	
X-214	HPCI-AOV-AO70 HPCI-AOV-AO71	HPCI Turbine Exhaust Drain	0.02	[redacted]		[redacted]	≤ 0.625	[redacted]	
X-220	PC-MOV-230MV PC-AOV-245AV PC-MOV-305MV PC-MOV-1308MV	Suppression Chamber Purge And Vent Exhaust	13.94	[redacted]		[redacted]	≤ 15.0	[redacted]	
X-221	RCIC-CV-12CV** RCIC-V-42**	RCIC Vacuum Pump Discharge To Suppression Chamber	0.0456	[redacted]		[redacted]	≤ 2.5	[redacted]	

* If determined.

** Satisfactory completion of leak testing also satisfies closure test requirements of the CNS IST Program.

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft³	AS FOUND LEAKAGE scfh	OUTBD /INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh RECOMMENDED	TECH SPEC	INITIAL /DATE
X-222	HPCI-CV-16CV** HPCI-V-50**	HPCI Turbine Drain To Suppression Chamber	0.027	[redacted]		[redacted]	≤ 2.5	[redacted]	
X-223A	CS-MOV-MO26A	CS A Test	2.19	[redacted]		[redacted]	≤ 3.125	[redacted]	
X-223A	CS-MOV-MO5A	CS A Minimum Flow	0.395	[redacted]		[redacted]	≤ 0.9375	[redacted]	
X-223B	CS-MOV-MO26B	CS B Test	2.46	[redacted]		[redacted]	≤ 3.125	[redacted]	
X-223B	CS-MOV-MO5B	CS B Minimum Flow	0.30	[redacted]		[redacted]	≤ 0.9375	[redacted]	
X-224	RCIC-MOV-MO41	RCIC Suction From Suppression Chamber	2.01	[redacted]		[redacted]	≤ 1.875	[redacted]	
X-225A	RHR-MOV-MO13A	RHR Pump A Suction From Suppression Chamber	11.3	[redacted]		[redacted]	≤ 6.25	[redacted]	
X-225B	RHR-MOV-MO13C	RHR Pump C Suction From Suppression Chamber	10.95	[redacted]		[redacted]	≤ 6.25	[redacted]	
X-225C	RHR-MOV-MO13B	RHR Pump B Suction From Suppression Chamber	11.3	[redacted]		[redacted]	≤ 6.25	[redacted]	
X-225D	RHR-MOV-MO13D	RHR Pump D Suction From Suppression Chamber	10.95	[redacted]		[redacted]	≤ 6.25	[redacted]	
X-226	HPCI-MOV-MO58	HPCI Suction From Suppression Chamber	4.6	[redacted]		[redacted]	≤ 5.0	[redacted]	

* If determined.
** Satisfactory completion of leak testing also satisfies closure test requirements of the CNS IST Program.

PRIMARY CONTAINMENT LLRT TEST RESULTS

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft^3	AS FOUND LEAKAGE scfh	OUTBD /INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh		INITIAL /DATE
							RECOMMENDED	TECH SPEC	
X-227A	CS-MOV-MO7A	CS Pump A Suction From Suppression Chamber	3.11	[redacted]		[redacted]	≤ 4.375	[redacted]	
X-227B	CS-MOV-MO7B	CS Pump B Suction From Suppression Chamber	3.11	[redacted]		[redacted]	≤ 4.375	[redacted]	

* If determined.

ATTACHMENT 5	PRIMARY CONTAINMENT LLRT TEST RESULTS

TABLE 1 - TYPE C LLRT PENETRATION TESTS

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft³	AS FOUND LEAKAGE scfh	OUTBD /INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh RECOMMENDED	ALLOWABLE LIMITS scfh TECH SPEC	INITIAL /DATE	
X-13B	RHR-MOV-MO27MO RHR-CV-27CV	RHR LOOP A INJECTION CHECK VALVE AND BYPASS	MAKE UP FLOW TEST ONLY	[redacted]	NA	[redacted]	≤15.0	[redacted]	JAS 6/21/94	TPCN 94-140
X-20	~~DW-V-133~~ DW-V-219	DEMIN WATER SUPPLY TO DW	NA	[redacted]	NA	[redacted]	0.5	[redacted]	JAS 6/23/94	TPCN 94-140
X-20	DW-V-133	DEMIN WATER SUPPLY TO DW	NA	[redacted]	NA	[redacted]	0.5	[redacted]	JAS 6/23/94	TPCN 94-140
X-22	IA-SV-65CV	IA SUPPLY TO MSIV'S	~~NA~~ 0.2	[redacted]	NA	[redacted]	0.5	[redacted]	JAT 6/23/94 JAS 7/5/94	TPCN 94-140 TPCN 94-161
X-23	REC-MOV-702MV	REC INLET TO DW	NA	[redacted]	NA	[redacted]	2.5	[redacted]	JAS 7/6/94	TPCN 94-159
X-24	REC-MOV-709MV	REC OUTLET FROM DW	NA	[redacted]	NA	[redacted]	2.5	[redacted]	JAS 7/6/94	TPCN 94-159
X-35A	NMT-NVA-104A	TIP VALVE A	NA	[redacted]	NA	[redacted]	0.25	[redacted]	JAS 7/6/94	TPCN 94-16
X-35B	NMT-NVA-104B	TIP VALVE B	NA	[redacted]	NA	[redacted]	0.25	[redacted]	JAS 6/23/94	TPCN 94-140 TPCN 94-161
X-35C	NMT-NVA-104C	TIP VALVE C	NA	[redacted]	NA	[redacted]	0.25	[redacted]	JAS 6/23/94	TPCN 94-14 TPCN 94-161
X-22	IA-CV-78CV	IA SUPPLY TO MSIV'S (OUTBD)	0.2	NA	NA	0.2	0.5	1.5	JAS 7/5/94	TPCN 94-161

1 Method 1
2 Method 2
* If determined.
** Satisfactory completion of leak testing also satisfies closure test requirements of the CNS IST Program.

TABLE 1 - TYPE C LLRT PENETRATION TESTS

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft³	AS FOUND LEAKAGE scfh	OUTBD /INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh RECOMMENDED	ALLOWABLE LIMITS scfh TECH SPEC	INITIAL /DATE	
X-35D	NMT-NVA-104D	TIP VALVE D	NA	[redacted]	NA	[redacted]	0.25	[redacted]	DAS 7/11/94	TPCN 94-161
X-35E	NM-CV-CV2	TIP PURGE SUPPLY CV	NA	[redacted]	NA	[redacted]	7/11/94 0.25	[redacted]	DAS 6/15/94 DAS 7/11/94	TPCN 94-140 TPCN 94-161
X-35E	NM-SOV-SPV3	TIP PURGE SUPPLY SOLENOID	NA	[redacted]	NA	[redacted]	0.25	[redacted]		
X-211	RHR-RV-18RV	STEAM SUPPLY TO RHR Hx A	NA	[redacted]	NA	[redacted]	0.5	[redacted]	DAS [illegible] DAS [illegible]	TPCN 94-156 TPCN 94-156
X-214	RHR-RV-19RV	STEAM SUPPLY TO RHR Hx B	NA	[redacted]	NA	[redacted]	0.5	[redacted]	DAS 6/21/94	TPCN 94-132
X-214	RHR-RV-20RV	RHR Hx A SHELL SIDE RELIEF	NA	[redacted]	NA	[redacted]	0.5	[redacted]	DAS 6/20/94 DAS 6/20/94	TPCN 94-156 TPCN 94-156
X-214	RH-RV-21RV	RHR Hx B SHELL SIDE RELIEF	NA	[redacted]	NA	[redacted]	0.5	[redacted]	DAS 6/21/94	TPCN 94-132
X-29E	PC-CV-33CV	RR-SOV-SPV741 INBOARD SUPPLY	0.001 **** ~~0.6~~ [illegible]	[redacted]	NA	[redacted]	0.1	[redacted]	DAS 7/14/94	TPCN 94-[illegible]
X-29E	PC-CV-34CV	RR-SOV-SPV741 OUTBOARD SUPPLY	0.002 **** ~~0.6~~ [illegible]	[redacted]	NA	[redacted]	0.1	[redacted]	DAS 7/14/94	TPCN 94-[illegible]
X-51F	PC-AOV-247AV	PAS SYSTEM ISOLATION VALVE	*** 0.013	NA	NA	0.085	1.0	3.0	DAS 7/14/94	TPCN 94-159

1 Method 1
2 Method 2
* If determined.
** Satisfactory completion of leak testing also satisfies closure test requirements of the CNS IST Program.
*** - USE 0.263 FT³ IF USING PORTABLE TEST VOLUME.
**** - ADD 0.25 FT³ IF USING PORTABLE TEST VOLUME.

TABLE 1 - TYPE C LLRT PENETRATION TESTS

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft³	AS FOUND LEAKAGE scfh	OUTBD /INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh RECOMMENDED	ALLOWABLE LIMITS scfh TECH SPEC	INITIAL /DATE	
X-30E	PC-V-559	NAI-SOV-SPV737 INBD SUPPLY	0.085 **** ~~0.6~~	[redacted]	NA	[redacted]	0.1	[redacted]	7/1/94	TCN 94-161
X-30E	PC-V-560	NAI-SOV-SPV737 OUTBD SUPPLY	0.115 **** ~~0.6~~	[redacted]	NA	[redacted]	0.1	[redacted]	7/1/94	TCN 94-161
X-30F	PC-V-561	MS-SOV-SPV739 INBD SUPPLY	0.085 **** ~~0.6~~	[redacted]	NA	[redacted]	0.1	[redacted]	7/1/94	TCN 94-161
X-30F	PC-V-562	MS-SOV-SPV739 OUTBD SUPPLY	0.115 **** 0.6	[redacted]	NA	[redacted]	0.1	[redacted]	7/1/94	TCN 94-161
X-33E	PC-V-563	NAI-SOV-SPV736 ~~SPV738~~ INBD SUPPLY	0.085 **** ~~0.6~~	[redacted]	NA	[redacted]	0.1	[redacted]	7/1/94	TCN 94-161
X-33E	PC-V-564	NAI-SOV-SPV736 ~~SPV738~~ OUTBD SUPPLY	0.115 **** ~~0.6~~	[redacted]	NA	[redacted]	0.1	[redacted]	7/1/94	TCN 94-161
X-33F	PC-V-565 ~~566~~	NAI-SOV-SPV738 INBD SUPPLY	0.085 **** ~~0.6~~	[redacted]	NA	[redacted]	0.1	[redacted]		
X-33F	PC-V-566	~~NAI~~-SOV-SPV738 OUTBD SUPPLY	0.115 **** ~~0.6~~	[redacted]	NA	[redacted]	0.1	[redacted]		
X-45D	PC-CV-35CV	NAI SOLENOID EXH TO AV CV (INBD)	0.002 **** ~~0.25~~	[redacted]	NA	[redacted]	0.25	[redacted]	7/1/94	TCN 94-161
X-51F	PC-AOV-248AV	PAS SYSTEM ISOLATION VALVE	*** 0.0011	NA	NA	0.068	1.0	3.0	7/4/94	TCN 94-159

1 Method 1
2 Method 2
* If determined.
** Satisfactory completion of leak testing also satisfies closure test requirements of the CNS IST Program.
*** USE 0.015 FT³ IF TESTING BETWEEN PC-V-44 AS PC-AOV-248AV, USE 0.265 FT³ IF TESTING BETWEEN PC-V-44 AND PC-AOV-248AV AND USING PORTABLE TEST VOLUME, USE 0.26 FT³ IF TESTING BETWEEN PC-AOV-247AV AND PC-AOV-248AV AND USING PORTABLE TEST VOLUME.
**** ADD 0.55 FT³ IF USING PORTABLE TEST VOLUME.

TABLE 1 - TYPE C LLRT PENETRATION TESTS

PENETRATION NUMBER	CIV	PENETRATION DESCRIPTION	VOLUME ft³	AS FOUND LEAKAGE scfh	OUTBD /INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh RECOMMENDED	ALLOWABLE LIMITS scfh TECH SPEC	INITIAL /DATE	
X-45D	PC-CV-36CV	N2 SOLENOID AIR TO ON (OUTBD)	0.003 **** ~~0.75~~ [illegible]	[redacted]	NA	[redacted]	0.25	[redacted]	[illegible] 7/6/94	TCN 94-159
X-229A	PC-V-569	AIR TO NRV-20 (INBD)	0.014 **** *** ~~0.6~~ [illegible]	[redacted]	NA	[redacted]	0.1	[redacted]	JAS 7/6/94	TCN 94-159
X-229A	PC-V-570	AIR TO NRV-20 (OUTBD)	0.015 **** *** ~~0.6~~ [illegible]	[redacted]	NA	[redacted]	0.1	[redacted]	JAS 7/6/94	TCN 94-159
X-229B	PC-V-571	AIR TO NRV-21 (INB)	0.014 **** *** ~~0.6~~ [illegible]	[redacted]	NA	[redacted]	0.1	[redacted]	JAS 7/6/94	TCN 94-159
X-229B	PC-V-572	AIR TO NRV-21 (OUTBD)	0.015 **** *** ~~0.6~~ [illegible]	[redacted]	NA	[redacted]	0.1	[redacted]	JAS 7/6/94	TCN 94-159
X-229C	PC-V-573	AIR TO NRV-22 (INB)	0.014 **** *** ~~0.6~~ [illegible]	[redacted]	NA	[redacted]	0.1	[redacted]	JAS 7/6/94	TCN 94-159
X-229C	PC-V-574	AIR TO NRV-22 (OUTBD)	0.015 **** *** ~~0.6~~ [illegible]	[redacted]	NA	[redacted]	0.1	[redacted]	JAS 7/6/94	TCN 94-159
X-229D	PC-V-575	AIR TO NRV-23 (INB)	0.014 **** *** ~~0.6~~ [illegible]	[redacted]	NA	[redacted]	0.1	[redacted]	JAS 7/6/94	TCN 94-159
X-229D	PC-V-576	AIR TO NRV-23 (OUTBD)	0.015 **** *** ~~0.6~~ [illegible]	[redacted]	NA	[redacted]	0.1	[redacted]	JAS 7/6/94	TCN 94-159

1 Method 1
2 Method 2
* If determined.
** Satisfactory completion of leak testing also satisfies closure test requirements of the CNS IST Program.
*** - ADD 0.20 ft³ IF USING RIG WITH 200 FT OF 1/2" TUBING. (200 FT ASSUMED LENGTH, AS THIS EXCEEDS MAX AMOUNT REQUIRED FOR TESTING).
**** - ADD 0.25 ft³ IF USING PORTABLE TEST VOLUME.

TABLE 1 - TYPE C LLRT PENETRATION TESTS

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft³	AS FOUND LEAKAGE scfh	OUTBD /INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh		INITIAL /DATE	
							RECOMMENDED	TECH SPEC		
X-229E	PC-V-577	AIR TO NRV-24 (INBD)	0.014 *** **** ~~0.6~~ [illegible]	[redacted]	NA	[redacted]	0.1	[redacted]	JAS 7/6/94	[illegible] 94-159
X-229E	PC-V-578	AIR TO NRV-24 (OUTBD)	0.015 *** **** ~~0.6~~ [illegible]	[redacted]	NA	[redacted]	0.1	[redacted]	JAS 7/6/94	[illegible] 94-159
X-229F	PC-V-579	AIR TO NRV-25 (INBD)	0.014 *** **** ~~0.6~~ [illegible]	[redacted]	NA	[redacted]	0.1	[redacted]	JAS 7/6/94	[illegible] 94-159
X-229F	PC-V-580	AIR TO NRV-25 (OUTBD)	0.015 *** **** ~~0.6~~ [illegible]	[redacted]	NA	[redacted]	0.1	[redacted]	JAS 7/6/94	[illegible] 94-159
X-229G	PC-V-581	AIR TO NRV-26 (INBD)	0.014 *** **** ~~0.6~~ [illegible]	[redacted]	NA	[redacted]	0.1	[redacted]	JAS 7/6/94	[illegible] 94-159
X-229G	PC-V-582	AIR TO NRV-26 (OUTBD)	0.015 *** **** ~~0.6~~ [illegible]	[redacted]	NA	[redacted]	0.1	[redacted]	JAS 7/6/94	[illegible] 94-159
X-229H	PC-V-583	AIR TO NRV-27 (INBD)	0.014 *** **** ~~0.6~~ [illegible]	[redacted]	NA	[redacted]	0.1	[redacted]	JAS 7/6/94	[illegible] 94-159
X-229H	PC-V-584	AIR TO NRV-27 (OUTBD)	0.015 *** **** ~~0.6~~ [illegible]	[redacted]	NA	[redacted]	0.1	[redacted]	JAS 7/6/94	[illegible] 94-159
X-229J	PC-V-585	AIR TO NRV-28 (INBD)	0.014 *** **** 0.6 [illegible]	[redacted]	NA	[redacted]	0.1	[redacted]	JAS 7/6/94	[illegible] 94-159

1 Method 1
2 Method 2
* If determined.
** Satisfactory completion of leak testing also satisfies closure test requirements of the CNS IST Program.
*** - ADD 0.10 ft³ if using air with adapter of ½" tubing (adapter assumed as this reflects max amount required for testing)
**** - ADD 0.55 ft³ if using portable test volume.

TABLE 1 TYPE C LLRT PENETRATION TESTS

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft^3	AS FOUND LEAKAGE scfh	OUTBD /INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh RECOMMENDED	ALLOWABLE LIMITS scfh TECH SPEC	INITIAL /DATE	
X-229 J	PC-V-586	AIR TO NRV-28 (OUTBD)	0.015 *** **** ~~0.6~~	[redacted]	NA	[redacted]	0.1	[redacted]	7/6/94	TPCN 94-159
X-229 K	PC-V-587	AIR TO NRV-29 (INBD)	0.014 *** **** ~~0.6~~	[redacted]	NA	[redacted]	0.1	[redacted]	7/6/94	TPCN 94-159
X-229 K	PC-V-588	AIR TO NRV-29 (OUTBD)	0.015 *** **** ~~0.6~~	[redacted]	NA	[redacted]	0.1	[redacted]	7/6/94	TPCN 94-159
X-229 L	PC-V-589	AIR TO NRV-30 (INBD)	0.014 *** **** ~~0.6~~	[redacted]	NA	[redacted]	0.1	[redacted]	7/6/94	TPCN 94-159
X-229 L	PC-V-590	AIR TO NRV-30 (OUTBD)	0.015 *** **** ~~0.6~~	[redacted]	NA	[redacted]	0.1	[redacted]	7/6/94	TPCN 94-159
X-21	SA-CV-15CV	SA TO DW SUPPLY CV	2.5	[redacted]	NA	[redacted]	0.5	[redacted]	7/6/94	TPCN 94-156
X-21	SA-V-647	SA TO DW SUPPLY VALVE (IB)	2.5 ~~2.5~~	[redacted]	NA	[redacted]	0.5	[redacted]	7/6/94	TPCN 94-156
X-21	SA-V-648	SA TO DW SUPPLY VALVE (OB)	2.5 ~~2.5~~	[redacted]	NA	[redacted]	0.5	[redacted]	7/6/94	TPCN 94-156
X-51F	PAS-AOV-3AV ~~PAS-AOV-12AV~~	CONTAINMENT ATMOSPHERE SAMPLE ISOLATION	.013 ~~.223~~	[redacted]	NA	[redacted]	1.0	[redacted]	6/30/94	TPCN 94-156
X-51F	PAS-AOV-12AV	VOLUME X VENT	.210	0.0	NA	NA	1.0	3.0	6/30/94	TPCN 94-156

1 Method 1
2 Method 2
* If determined.
** Satisfactory completion of leak testing also satisfies closure test requirements of the CNS IST Program.
*** - ADD 0.2 FT^3 IF USING AIR WITH 200 FT OF 1/4" TUBING (200 FT ASSUMED, AS MAY LENGTH REQUIRED FOR TESTING)
**** - ADD 0.35 FT^3 IF USING PORTABLE TEST VOLUME.

TABLE 2 - TYPE B PENETRATION LLRT TESTS

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft³	AS FOUND LEAKAGE scfh	OUTBD/INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh	INITIAL/ DATE	
X-1A**	PC-PENT-X1A	Northeast Drywell Equipment Hatch	0.25				≤ 0.1		
X-1B**	PC-PENT-X1B	Southwest Drywell Equipment Hatch	0.25				≤ 0.1		
X-4**	PC-PENT-X4	Drywell Head Access Hatch	0.25				≤ 0.1		
X-6**	PC-PENT-X6	CRD Removal Hatch	0.25	0.0	NA		≤ 0.1	[illegible]	TPCN [illegible]
X-7A***	PC-PENT-X7A	Main Steam Line A Expansion Bellows	0.25				≤ 1.0		
X-7B***	PC-PENT-X7B	Main Steam Line B Expansion Bellows	0.25				≤ 1.0		
X-7C***	PC-PENT-X7C	Main Steam Line C Expansion Bellows	0.25				≤ 1.0		
X-7D***	PC-PENT-X7D	Main Steam Line D Expansion Bellows	0.25				≤ 1.0		
X-9A***	PC-PENT-X9A	Reactor Feedwater Line A Expansion Bellows	0.25				≤ 1.0		
X-9B***	PC-PENT-X9B	Reactor Feedwater Line B Expansion Bellows	0.25				≤ 1.0		
X-35A**	PC-PENT-X35A	TIP D	0.25	0.0	NA	0.0	≤ 0.1	[illegible] 7/3/94 [illegible] 7/4/94	TPCN 94- TPCN 94-/

* If determined.

** Denotes bolted double gasket seal or testable gasket.

*** Denotes testable expansion bellows.

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PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft³	AS FOUND LEAKAGE scfh	OUTBD/INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh	INITIAL/ DATE
X-35B**	PC-PENT-X35B	TIP A	0.25	0.0	NA	0.0	≤ 0.1	[illegible]
X-35C**	PC-PENT-X35C	TIP C	0.25	0.0	NA	0.0	≤ 0.1	[illegible]
X-35D**	PC-PENT-X35D	TIP B	0.25	0.0	NA	0.0	≤ 0.1	[illegible]
X-35E**	PC-PENT-X35E	TIP Nitrogen Purge	0.25	0.0	NA	0.0	≤ 0.1	[illegible]
X-36	PC-AN-H_2/O_2I	Division I H_2/O_2 Analyzer	0.33				≤ 5.0	
X-36	PC-AN-H_2/O_2II	Division II H_2/O_2 Analyzer	0.32				≤ 5.0	
X-49C	PC-PENT-X49C	Instrumentation And Control	0.023				≤ 0.1	
X-49D	PC-PENT-X49D	Instrumentation And Control	0.023				≤ 0.1	
X-50A	PC-PENT-X50A	Instrumentation And Control	0.023				≤ 0.1	
X-50B	PC-PENT-X50B	Instrumentation and Control	0.023				≤ 0.1	
X-100A	PC-PENT-X100A	Low Voltage Power	5.7				≤ 0.1	

* If determined.
** Denotes bolted double gasket seal or testable gasket.
*** Denotes testable expansion bellows.

ATTACHMENT 5	PRIMARY CONTAINMENT LLRT TEST RESULTS

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft^3	AS FOUND LEAKAGE scfh	OUTBD/INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh	INITIAL/ DATE
X-100E	PC-PENT-X100E	Thermocouple	5.5				≤ 0.1	
X-100F	PC-PENT-X100F	Neutron Monitoring Signals	5.5				≤ 0.1	
X-100G	PC-PENT-X100G	Low Voltage Power	5.9				≤ 0.1	
X-100H	PC-PENT-X100H	Low Voltage Power	5.56				≤ 0.1	
X-101A	PC-PENT-X101A	Medium Voltage Power	5.7				≤ 0.1	
X-101B	PC-PENT-X101B	Neutron Monitoring Signals	5.56				≤ 0.1	
X-101C	PC-PENT-X101C	Medium Voltage Power	5.9				≤ 0.1	
X-101D	PC-PENT-X101D	Medium Voltage Power	5.9				≤ 0.1	
X-101E	PC-PENT-X101E	Low Voltage Power And Instrumentation	0.25				≤ 0.1	
X-101F	PC-PENT-X101F	Medium Voltage Power	6.11				≤ 0.1	
X-102	PC-PENT-X102	Low Voltage Power	5.9				≤ 0.1	

* If determined.

** Denotes bolted double gasket seal or testable gasket.

*** Denotes testable expansion bellows.

PRIMARY CONTAINMENT LLRT TEST RESULTS

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft³	AS FOUND LEAKAGE scfh	OUTBD/INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh	INITIAL/ DATE
X-103	PC-PENT-X103	Neutron Monitoring Signals	5.09				≤ 0.1	
X-104A	PC-PENT-X104A	Instrumentation And Control	5.97				≤ 0.1	
X-104B	PC-PENT-X104B	Instrumentation And Control	5.56				≤ 0.1	
X-104D	PC-PENT-X104D	Instrumentation And Control	4.33				≤ 0.1	
X-104E	PC-PENT-X104E	Instrumentation And Control	4.58				≤ 0.1	
X-105A	PC-PENT-X105A	Low Voltage Power	5.84				≤ 0.1	
X-105D	PC-PENT-X105D	Medium Voltage Power	6.24				≤ 0.1	
X-106	PC-PENT-X106	Neutron Monitoring Signals	5.42				≤ 0.1	
X-230	PC-PENT-X230	Low Voltage Power	2.39				≤ 0.1	
X-200A**	PC-PENT-X200A	Northwest Suppression Chamber Access Hatch	0.25				≤ 0.1	
X-200B**	PC-PENT-X200B	Southeast Suppression Chamber Access Hatch	0.25	0.0	NA		≤ 0.1	[illegible] 6.3.1.1

* If determined.
** Denotes bolted double gasket seal or testable gasket.
*** Denotes testable expansion bellows.

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft^3	AS FOUND LEAKAGE scfh	OUTBD/INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh	INITIAL/DATE
X-213B**	PC-PENT-X213B	Suppression Chamber Drain Flange	0.25				≤ 0.1	
------**	PC-PENT-DWH	Drywell Head	0.25				≤ 0.1	
------**	PC-PENT-SIP1	Stabilizer Inspection Port 1	0.25				≤ 0.1	
------**	PC-PENT-SIP2	Stabilizer Inspection Port 2	0.25				≤ 0.1	
------**	PC-PENT-SIP3	Stabilizer Inspection Port 3	0.25				≤ 0.1	
------**	PC-PENT-SIP4	Stabilizer Inspection Port 4	0.25				≤ 0.1	
------**	PC-PENT-SIP5	Stabilizer Inspection Port 5	0.25				≤ 0.1	
------**	PC-PENT-SIP6	Stabilizer Inspection Port 6	0.25				≤ 0.1	
------**	PC-PENT-SIP7	Stabilizer Inspection Port 7	0.25				≤ 0.1	
------**	PC-PENT-SIP8	Stabilizer Inspection Port 8	0.25				≤ 0.1	
X-220**	PC-FLG-230MV	PC-230MV Testable Flange	0.25				≤ 0.1	
X-26**	PC-FLG-231MV	PC-231MV Testable Flange	0.25				≤ 0.1	

* If determined.

** Denotes bolted double gasket seal or testable gasket.

*** Denotes testable expansion bellows.

PRIMARY CONTAINMENT LLRT TEST RESULTS

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft³	AS FOUND LEAKAGE scfh	OUTBD/INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh	INITIAL/ DATE
X-25**	PC-FLG-232MV	PC-232MV Testable Flange	0.25				≤ 0.1	
X-205**	PC-FLG-233MV	PC-233MV Testable Flange	0.25				≤ 0.1	
X-205**	PC-FLG-243AV	PC-243AV Testable Flange	0.25				≤ 0.1	
X-205**	PC-FLG-244AV	PC-244AV Testable Flange	0.25				≤ 0.1	
X-225A**	RHR-FLG-10RV	Testable Flange	0.25				≤ 0.1	
X-225C**	RHR-FLG-11RV	Testable Flange	0.25				≤ 0.1	
X-225B**	RHR-FLG-12RV	Testable Flange	0.25				≤ 0.1	
X-225D**	RHR-FLG-13RV	Testable Flange	0.25				≤ 0.1	
X-210A**	RHR-FLG-14RV	Testable Flange	0.25				≤ 0.1	
X-210B**	RHR-FLG-15RV	Testable Flange	0.25				≤ 0.1	
X-210A**	RHR-FLG-17RV	Testable Flange	0.25				≤ 0.1	
X-214**	RHR-FLG-18RV	Testable Flange	0.25	0.0	NA	0.0	≤ 0.1	[illegible]

* If determined.

** Denotes bolted double gasket seal or testable gasket.

*** Denotes testable expansion bellows.

TABLE 2 - TYPE B PENETRATION LLRT TESTS

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft^3	AS FOUND LEAKAGE scfh	OUTBD/INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh	INITIAL/ DATE	
X-43	PC-PENT-X43	ILRT TEST CONNECTION	0.25	0.0*	NA	0.0	0.1	DAS [illegible] 6/13/94	TPCN 94-[illegible] TPCN 94-[illegible]
X-44	PC-PENT-X44	ILRT TEST CONNECTION	0.25	0.0*	NA	0.0	0.1	DAS [illegible] DAS 6/13/94	TPCN 94-[illegible] TPCN 94-[illegible]
X-209A	PC-PENT-X209A	TORUS AIR TEMPERATURE	0.5	0.0	NA	0.0	0.5	DAS 6/[illegible]/94 [illegible] 6/[illegible]/94	TPCN 94-133 94-140
X-209B	PC-PENT-X209B	TORUS WATER TEMPERATURE	0.5	0.0	NA	0.0	0.5	DAS 6/[illegible]/94	TPCN 94-132 94-140
X-209C	PC-PENT-X209C	TORUS AIR TEMPERATURE	0.5	0.0	NA	0.0	0.5	[illegible] 6/[illegible]/94	TPCN 94-136 94-140
X-209D	PC-PENT-X209D	TORUS WATER TEMPERATURE	0.5	0.032 ~~0.0~~ [illegible]	NA	0.032 ~~0.0~~ [illegible]	0.5	DAS 6/[illegible]/94	TPCN 94-132 94-140
X-218	PC-PENT-X218	TORUS TEMPERATURE	1.2	0.078	NA	0.0	1.0	DAS 6/[illegible]/94	TPCN 94-132
X-29E	PC-PENT-X29E	SUPPLY PIPING TO PR-SOV-SPV741	0.5	0.0	NA	NA	0.5	DAS 8/1/94	TPCN 94-176
X-51B	PC-PENT-X51B	EXHAUST PIPING TO PR-SOV-SPV741	0.5	0.0	NA	0.06 ~~0.0~~ [illegible]	0.5	DAS 6/[illegible]/94 [illegible]	TPCN 94-186 [illegible]
X-30E	PC-PENT-X30E	PIPING TO MS-SOV-SPV737	0.5	2.7	NA	NA	0.5	DAS 6/[illegible]/94	TPCN 94-158
X-30F	PC-PENT-X30F	PIPING TO MS-SOV-SPV739	0.5	2.14	NA	NA	0.5	DAS 6/[illegible]/94	TPCN 94-158

* PERFORMED 12/18/93 TPCN 93-961 DAS 6/13/94

* If determined.
** Denotes bolted double gasket seal or testable gasket.
*** Denotes testable expansion bellows.

TABLE 2 - TYPE B PENETRATION LLRT TESTS

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft^3	AS FOUND LEAKAGE scfh	OUTBD/INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh	INITIAL/ DATE	
X-33E	PC-PENT-X33E	PIPING TO NBI-SOV-SPV736	0.5	2.03	NA	NA	0.5	JAS 6/29/94	TPCN 94-150
X-33F	PC-PENT-X33F	PIPING TO NBI-SOV-SPV738	0.5	2.41	NA	NA	0.5	JAS 6/29/94	TPCN 94-156
X-45D	PC-PENT-X45D PC-V-265	NBI & MS SOV EXHAUST LOOP	0.75	0.0	NA	NA	0.5	JAS 6/30/94	TPCN 94-156
X-46A	PC-PENT-X46A PC-V-266	SPARE	0.35	0.0	NA	NA	0.1	JAS 6/28/94	TPCN 94-150
X-46B	PC-PENT-X46B PC-V-267	SPARE	0.35	0.0	NA	NA	0.1	JAS 6/28/94	
X-46C	PC-PENT-X46C PC-V-268	SPARE	0.35	0.0	NA	NA	0.1	JAS 6/28/94	
X-46D	PC-PENT-X46D PC-V-269	SPARE	0.35	0.0	NA	NA	0.1	JAS 6/28/94	
X-46E	PC-PENT-X46E PC-V-270	SPARE	0.35	0.0	NA	NA	0.1	JAS 6/28/94	
X-46F	PC-PENT-X46F PC-V-271	SPARE	0.35	0.0	NA	NA	0.1	JAS 6/28/94	
X-47A	PC-PENT-X47A	SPARE	0.35	0.0	NA	NA	0.1	JAS 6/28/94	
X-47C JAS 6/28/94 ~~X-47A~~	PC-V-272 JAS 6/28/94 X47C PC-PENT-X47A	SPARE	0.35	0.0	NA	NA	0.1	JAS 6/28/94	

* If determined.
** Denotes bolted double gasket seal or testable gasket.
*** Denotes testable expansion bellows.

TABLE 2 - TYPE B PENETRATION LLRT TESTS

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft³	AS FOUND LEAKAGE scfh	OUTBD/INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh	INITIAL/ DATE	
X-47D	PC-V-273 PC-PENT-X47D	SPARE	0.35	0.0	NA	NA	0.1	JAS 6/28/94	TPCN 94-15
X-47F	PC-V-275 PC-PENT-X47F	SPARE	0.35	0.0	NA	NA	0.1	JAS 6/28/94	TPCN 94-15
X-47R JAS 6/28/94 ~~X-48~~	PC-V-274 JAS 6/28/94 PC-PENT-X47R	SPARE	0.35	0.0	NA	NA	0.1	JAS 6/28/94	TPCN 94-15
X-100B	PC-V-230 PC-PENT-X100B	SPARE	0.35	0.0	NA	NA	0.1	JAS 6/28/94	TPCN 94-15
X-229A	IA-V-339	AIR TO NRV-20	0.5	0.08	NA	NA	0.25	JAS 6/28/94	TPCN 94-15
X-229B	IA-V-341	AIR TO NRV-21	0.5	0.08	NA	NA	0.25	JAS 6/28/94	
X-229C	IA-V-343	AIR TO NRV-22	0.5	0.16	NA	NA	0.25	JAS 6/28/94	
X-229D	IA-V-345	AIR TO NRV-23	0.5	0.08	NA	NA	0.25	JAS 6/28/94	
X-229E	IA-V-347	AIR TO NRV-24	0.5	0.08	NA	NA	0.25	JAS 6/28/94	
X-229F	IA-V-349	AIR TO NRV-25	0.5	0.16	NA	NA	0.25	JAS 6/28/94	
X-229G	IA-V-351	AIR TO NRV-26	0.5	0.16	NA	NA	0.25	JAS 6/28/94	

* If determined.
** Denotes bolted double gasket seal or testable gasket.
*** Denotes testable expansion bellows.

TABLE 2 - TYPE B PENETRATION LLRT TESTS

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft³	AS FOUND LEAKAGE scfh	OUTBD/INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh	INITIAL/ DATE
X-229H	IA-V-353	AIR TO NRV-27	0.5	0.08	NA	NA	0.25	JAS 6/20/94
X-229J	IA-V-355	AIR TO NRV-28	0.5	0.08	NA	NA	0.25	JAS 6/22/94
X-229K	IA-V-357	AIR TO NRV-29	0.5	0.08	NA	NA	0.65	JAS 6/20/94
X-229L	IA-V-361	AIR TO NRV-30	0.5	0.16	NA	NA	0.55	JAS 6/20/94
X-40A	PC-PS-12A	DW INSTRUMENTATION	0.35	0.034	NA	0.034	0.1	JAS 7/10/94
X-40B	PC-PS-12B	DW INSTRUMENTATION	0.35	0.102	NA	0.102	0.1	JAS 7/10/94
X-40C	PC-PS-12C	DW INSTRUMENTATION	0.35	0.0	NA	0.0	0.1	JAS 7/10/94
X-40D	PC-PS-12D	DW INSTRUMENTATION	0.35	0.0	NA	0.0	0.1	JAS 7/10/94
X-40A	PC-PS-101A	DW INSTRUMENTATION	0.35	0.47	NA	0.47	0.1	JAS 7/10/94
X-40B	PC-PS-101B	DW INSTRUMENTATION	0.35	0.27 ~~2.7~~ JAS 7/10/94	NA	0.27 ~~2.7~~	0.1	JAS 7/10/94
X-40C	PC-PS-101C	DW INSTRUMENTATION	0.35	0.068	NA	0.065	0.1	JAS 7/10/94

TPCN 94-190

TPCN 94-161

* If determined.
** Denotes bolted double gasket seal or testable gasket.
*** Denotes testable expansion bellows.

TABLE 2 - TYPE B PENETRATION LLRT TESTS

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft^3	AS FOUND LEAKAGE scfh	OUTBD/INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh	INITIAL/ DATE
X-40D	PC-PS-101D	DW INSTRUMENTATION	0.35	0.34	NA	0.34	0.1	[illegible] 7/10/94
X-40D ~~X-40A~~ [illegible]	PC-PS-119D	DW INSTRUMENTATION	0.35	0.1	NA	0.1	0.1	[illegible] 7/10/94
X-40B	PC-PS-119A	DW INSTRUMENTATION	0.35	3.24	NA	0.0	0.1	[illegible] 7/10/94 [illegible] 7/10/94
X-40B	PC-PS-119C	DW INSTRUMENTATION	0.35	1.21	NA	0.0	0.1	[illegible] 7/10/94 [illegible] 7/10/94
X-40D [illegible] ~~X-40B~~	PC-PS-119B	DW INSTRUMENTATION	0.35	0.0	NA	0.0	0.1	[illegible] 7/10/94
X-40A	PC-PS-16	DW INSTRUMENTATION	0.35	0.068	NA	0.068	0.1	[illegible] 7/10/94
X-40A	PC-PT-512A	DW INSTRUMENTATION	0.35	0.0	NA	0.0	0.1	[illegible] 7/10/94
X-40C	PC-PT-512B	DW INSTRUMENTATION	0.35	0.0	NA	0.0	0.1	[illegible] 7/10/94
X-229M	PC-PANT-X229M	SPARE	0.35	0.0	NA	NA	0.1	[illegible] 7/7/94

TPCN 94-161

* If determined.
** Denotes bolted double gasket seal or testable gasket.
*** Denotes testable expansion bellows.

PRIMARY CONTAINMENT LLRT TEST RESULTS

PENETRATION NUMBER	CIC	PENETRATION DESCRIPTION	VOLUME ft³	AS FOUND LEAKAGE scfh	OUTBD/INBD LEAKAGE scfh*	AS LEFT LEAKAGE scfh	ALLOWABLE LIMITS scfh	INITIAL/ DATE	
X-214**	RHR-FLG-19RV	Testable Flange	0.25	0.0	NA	0.0	≤ 0.1	JAS 6/21/94	TPCN 94-192
X-214**	RHR-FLG-20RV	Testable Flange	0.25	0.0	NA	0.0	≤ 0.1	JAS 6/21/94 JAS 6/30/94	TPCN 94-[illegible] TPCN 94-[illegible]
X-214**	RHR-FLG-21RV	Testable Flange	0.25	0.0	NA	0.0	≤ 0.1	JAS 6/21/94	TPCN 94-[illegible]
X-227A**	CS-FLG-10RV	Testable Flange	0.25				≤ 0.1		
X-223A**	CS-FLG-11RV	Testable Flange	0.25				≤ 0.1		
X-227B**	CS-FLG-12RV	Testable Flange	0.25				≤ 0.1		
X-223B**	CS-FLG-13RV	Testable Flange	0.25				≤ 0.1		
X-213A**	PC-FLG-DL1	Testable Flange	0.25				≤ 0.1		
X-213A**	PC-FLG-DL2	Testable Flange	0.25				≤ 0.1		

* If determined.
** Denotes bolted double gasket seals or testable flanges.
*** Denotes testable expansion bellows.

IST Engineer Review (Table 1 only): ______________________ Date: ____________

LLRT Engineer Review: ______________________ Date: ____________

I. OVERALL PLANT REQUIREMENTS

CRITERION 1 - QUALITY STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

CRITERION 2 - PERFORMANCE STANDARDS (Category A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

CRITERION 3 - FIRE PROTECTION (Category A)

The reactor facility shall be designed (1) to minimize the probability of events such as fires and explosions and (2) to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

CRITERION 4 - SHARING OF SYSTEMS (Category A)

Reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing.

CRITERION 5 - RECORDS REQUIREMENTS (Category A)

Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator or under its control throughout the life of the reactor.

II. PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS

CRITERION 6 - REACTOR CORE DESIGN (Category A)

The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all off-site power.

CRITERION 7 - SUPPRESSION OF POWER OSCILLATIONS (Category B)

The core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed.

CRITERION 8 - OVERALL POWER COEFFICIENT (Category B)

The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive.

CRITERION 9 - REACTOR COOLANT PRESSURE BOUNDARY (Category A)

The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

CRITERION 10 - CONTAINMENT (Category A)

Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

III. NUCLEAR AND RADIATION CONTROLS

CRITERION 11 - CONTROL ROOM (Category B)

The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR 20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

CRITERION 12 - INSTRUMENTATION AND CONTROL SYSTEMS (Category B)

Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

CRITERION 13 - FISSION PROCESS MONITORS AND CONTROLS (Category B)

Means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.

CRITERION 14 - CORE PROTECTION SYSTEMS (Category B)

Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

CRITERION 15 - ENGINEERED SAFETY FEATURES PROTECTION SYSTEMS (Category B)

Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

CRITERION 16 - MONITORING REACTOR COOLANT PRESSURE BOUNDARY (Category B)

Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.

CRITERION 17 - MONITORING RADIOACTIVITY RELEASES (Category B)

Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

CRITERION 18 - MONITORING FUEL AND WASTE STORAGE (Category B)

Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

IV. RELIABILITY AND TESTABILITY OF PROTECTION SYSTEMS

CRITERION 19 - PROTECTION SYSTEMS RELIABILITY (Category B)

Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

CRITERION 20 - PROTECTION SYSTEMS REDUNDANCY AND INDEPENDENCE (Category B

Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served. Differe-- principles shall be used where necessary to achieve true independence of redundant instrumentation components.

CRITERION 21 - SINGLE FAILURE DEFINITION (Category B)

Multiple failures resulting from a single event shall be treated as a single failure.

CRITERION 22 - SEPARATION OF PROTECTION AND CONTROL INSTRUMENTATION SYSTEMS (Category B)

Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.

CRITERION 23 - PROTECTION AGAINST MULTIPLE DISABILITY FOR PROTECTION SYSTEMS (Category B)

The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function.

CRITERION 24 - EMERGENCY POWER FOR PROTECTION SYSTEMS (Category B)

In the event of loss of all offsite power, sufficient alternate sources of power shall be provided to permit the required functioning of the protection systems.

CRITERION 25 - DEMONSTRATION OF FUNCTIONAL OPERABILITY OF PROTECTION SYSTEMS (Category B)

Means shall be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

CRITERION 26 - PROTECTION SYSTEMS FAIL-SAFE DESIGN (Category B)

The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

V. REACTIVITY CONTROL

CRITERION 27 - REDUNDANCY OF REACTIVITY CONTROL (Category A)

At least two independent reactivity control systems, preferably of different principles, shall be provided.

CRITERION 28 - REACTIVITY HOT SHUTDOWN CAPABILITY (Category A)

At least two of the reactivity control systems provided shall independently be capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits.

CRITERION 29 - REACTIVITY SHUTDOWN CAPABILITY (Category A)

At least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided.

CRITERION 30 - REACTIVITY HOLDDOWN CAPABILITY (Category B)

At least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

CRITERION 31 - REACTIVITY CONTROL SYSTEMS MALFUNCTION (Category B)

The reactivity control systems shall be capable of sustaining any single malfunction, such as, unplanned continuous withdrawal (not ejection) of a control rod, without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

CRITERION 32 - MAXIMUM REACTIVITY WORTH OF CONTROL RODS (Category A)

Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

VI. REACTOR COOLANT PRESSURE BOUNDARY

CRITERION 33 - REACTOR COOLANT PRESSURE BOUNDARY CAPABILITY (Category A)

The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

CRITERION 34 - REACTOR COOLANT PRESSURE BOUNDARY RAPID PROPAGATION FAILURE PREVENTION (Category A)

The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given (a) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

CRITERION 35 - REACTOR COOLANT PRESSURE BOUNDARY BRITTLE FRACTURE PREVENTION (Category A)

Under conditions where reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperatures shall be at least 120°F above the nil ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation or 60°F above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

CRITERION 36 - REACTOR COOLANT PRESSURE BOUNDARY SURVEILLANCE (Category A)

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with ASTM-E-185-66 shall be provided.

VII. ENGINEERED SAFETY FEATURES

CRITERION 37 - ENGINEERED SAFETY FEATURES BASIS FOR DESIGN (Category A)

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

CRITERION 38 - RELIABILITY AND TESTABILITY OF ENGINEERED SAFETY FEATURES (Category A)

All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

CRITERION 39 - EMERGENCY POWER FOR ENGINEERED SAFETY FEATURES (Category A)

Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the onsite power system and the offsite power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.

CRITERION 40 - MISSILE PROTECTION (Category A)

Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

CRITERION 41 - ENGINEERED SAFETY FEATURES PERFORMANCE CAPABILITY (Category A)

Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.

CRITERION 42 - ENGINEERED SAFETY FEATURES COMPONENTS CAPABILITY (Category A)

Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident.

CRITERION 43 - ACCIDENT AGGRAVATION PREVENTION (Category A)

Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse after-effects of the loss of normal cooling is avoided.

CRITERION 44 - EMERGENCY CORE COOLING SYSTEMS CAPABILITY (Category A)

At least two emergency core cooling systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident

CRITERION 45 - INSPECTION OF EMERGENCY CORE COOLING SYSTEMS (Category A)

Design provisions shall be made to facilitate physical inspection of all critical parts of the emergency core cooling systems, including reactor vessel internals and water injection nozzles.

CRITERION 46 - TESTING OF EMERGENCY CORE COOLING SYSTEMS COMPONENTS (Category A)

Design provisions shall be made so that active components of the emergency core cooling systems, such as pumps and valves, can be tested periodically for operability and required functional performance.

CRITERION 47 - TESTING OF EMERGENCY CORE COOLING SYSTEMS (Category A)

A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as is practical.

CRITERION 48 - TESTING OF OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEMS (Category A)

A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the emergency core cooling systems into action, including the transfer to alternate power sources.

CRITERION 49 - CONTAINMENT DESIGN BASIS (Category A)

The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

CRITERION 50 - NDT REQUIREMENT FOR CONTAINMENT MATERIAL (Category A)

Principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures under noroperating and testing conditions are not less than 30°F above nil ductility transition (NDT) temperature.

CRITERION 51 - REACTOR COOLANT PRESSURE BOUNDARY OUTSIDE CONTAINMENT (Category A)

If part of the reactor coolant pressure boundary is outside the containment appropriate features as necessary shall be provided to protect the health and safety of the public in case of an accidental rupture in that part. Determination of the appropriateness of features such as isolation valves and additional containment shall include consideration of the environmental and population conditions surrounding the site.

CRITERION 52 - CONTAINMENT HEAT REMOVAL SYSTEMS (Category A)

Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.

CRITERION 53 - CONTAINMENT ISOLATION VALVES (Category A)

Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

CRITERION 54 - CONTAINMENT LEAKAGE RATE TESTING (Category A)

Containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period of time to verify its conformance with required performance.

CRITERION 55 - CONTAINMENT PERIODIC LEAKAGE RATE TESTING (Category A)

The containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime.

CRITERION 56 - PROVISIONS FOR TESTING OF PENETRATIONS (Category A)

Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leaktightness to be demonstrated at design pressure at any time.

CRITERION 57 - PROVISIONS FOR TESTING OF ISOLATION VALVES (Category A)

Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

CRITERION 58 - INSPECTION OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (Category A)

Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as, pumps, valves, spray nozzles, torus, and sumps.

CRITERION 59 - TESTING OF CONTAINMENT PRESSURE-REDUCING SYSTEMS COMPONENTS (Category A)

The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance.

CRITERION 60 - TESTING OF CONTAINMENT SPRAY SYSTEMS (Category A)

A capability shall be provided to test periodically the delivery capability of the containment spray system at a position as close to the spray nozzles as is practical.

CRITERION 61 - TESTING OF OPERATIONAL SEQUENCE OF CONTAINMENT PRESSURE-REDUCING SYSTEMS (Category A)

A capability shall be provided to test under conditions as close to the design as practical the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources.

CRITERION 62 - INSPECTION OF AIR CLEANUP SYSTEMS (Category A)

Design provisions shall be made to facilitate physical inspection of all critical parts of containment air cleanup systems, such as, ducts, filters, fans, and dampers.

CRITERION 63 - TESTING OF AIR CLEANUP SYSTEMS COMPONENTS (Category A)

Design provisions shall be made so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance.

CRITERION 64 - TESTING OF AIR CLEANUP SYSTEMS (Category A)

A capability shall be provided for in situ periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits.

CRITERION 65 - TESTING OF OPERATIONAL SEQUENCE OF AIR CLEANUP SYSTEMS (Category A)

A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability.

VIII. FUEL AND WASTE STORAGE SYSTEMS

CRITERION 66 - PREVENTION OF FUEL STORAGE CRITICALITY (Category 3)

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

CRITERION 67 - FUEL AND WASTE STORAGE DECAY HEAT (Category 3)

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

CRITERION 68 - FUEL AND WASTE STORAGE RADIATION SHIELDING (Category 3)

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10 CFR 20.

CRITERION 69 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE (Category 3)

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

IX. PLANT EFFLUENTS

CRITERION 70 - CONTROL OF RELEASES OF RADIOACTIVITY TO THE ENVIRONMENT (Category 3)

The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

(Sec. 161, 68 Stat. 948; 42 U.S.C. 2201)

Dated at Washington, D. C. this twenty-eighth day of June, 1967.

For the Atomic Energy Commission.

W. B. McCool
Secretary

AEC PUBLISHES GENERAL DESIGN CRITERIA
FOR NUCLEAR POWER PLANT CONSTRUCTION PERMITS

The AEC is publishing for public comment a revised set of proposed General Design Criteria which have been developed to assist in the preparation of applications for nuclear power plant construction permits.

In November 1965, the AEC issued an announcement requesting comments on General Design Criteria developed by its regulatory staff. These criteria were statements of design principles and objectives which have evolved over the years in licensing nuclear power plants by the AEC.

It was recognized at the time the criteria were first issued for comment that further efforts were needed to develop them more fully. The revision being published today reflects extensive public comments received from twenty groups or individuals, suggestions made at meetings with the Atomic Industrial Forum, and review within the AEC.

The regulatory staff has worked closely with the Commission's Advisory Committee on Reactor Safeguards on the development of the criteria and the revision of the proposed criteria reflects ACRS review and comment.

The General Design Criteria reflect the predominating experience to date with water reactors, but they are considered to be generally applicable to all power reactors. The proposed criteria are intended to be used as guidance to an applicant in establishing the principal design criteria for a nuclear power plant. The framework within which the criteria are presented provides sufficient flexibility to permit applicants to establish design requirements using alternate and/or additional criteria. In particular, additional criteria will be needed for unusual sites and environmental conditions and for new or advanced types of reactors. In each case an applicant will be required to identify its principal design criteria and provide assurance that they encompass all those facility design features required in the interest of public health and safety.

The criteria are designated as "General Design Criteria for Nuclear Power Plant Construction Permits" to emphasize the key role they assume at this stage of the licensing process. The criteria have been categorized as Category A or Category B. Experience has shown that more definitive information has been needed at the construction permit stage for certain of the criteria; these have been designated as Category A.

Development of these criteria is part of a longer-range Commission program to develop criteria, standards, and codes for nuclear reactor plants. This includes codes and standards that industry is developing with AEC participation. The ultimate goal is the evolution of industry codes and standards based on accumulated knowledge and experience as has occurred in various fields of engineering and construction.

The provisions of the proposed amendment relating to General Design Criteria are expected to be useful as interim guidance until such time as the Commission takes further action on them.

The proposed criteria, which would become Appendix A to Part 50 of the AEC's regulations, will be published in the Federal Register on ______________. Interested persons may submit written comments or suggestions to the Secretary, U. S. Atomic Energy Commission, Washington, D. C. 20545, within 60 days. A copy of the proposed "General Design Criteria for Nuclear Power Plant Construction Permits" is attached.

3.9 AUXILIARY ELECTRICAL SYSTEM

Applicability:

Applies to the auxiliary electrical power system.

Objective:

To assure an adequate supply of electrical power for operation of those systems required for safety.

Specification:

A. Auxiliary Electrical Equipment

1. The reactor shall not be made critical from a Cold Shutdown Condition unless all of the following conditions are satisfied:

a. Both off-site sources (345 KV and 69 KV) and the startup transformer and emergency transformer are available and capable of automatically supplying power to the 4160 Volt emergency buses 1F and 1G.

b. Both diesel generators shall be operable and there shall be a minimum of 48,000 gal. of diesel fuel in the fuel oil storage tanks.

c. The 4160V critical buses 1F and 1G and the 480V critical buses 1F and 1G are energized.

1. The loss of voltage relays and their auxiliary relays are operable.

2. The undervoltage relays and their auxiliary relays are operable.

d. The four unit 125V/250V batteries and their chargers shall be operable.

The power monitoring system for the inservice RPS MG set or alternate source shall be operable.

4.9 AUXILIARY ELECTRICAL SYSTEM

Applicability:

Applies to the periodic testing requirements of the auxiliary electrical systems.

Objective:

Verify the operability of the auxiliary electrical system.

Specification:

A. Auxiliary Electrical Equipment

1. Emergency Buses Undervoltage Relays

a. Loss of voltage relays

Once every 18 months, loss of voltage on emergency buses is simulated to demonstrate the load shedding from emergency buses and the automatic start of diesel generators.

b. Undervoltage relays

Once every 18 months, low voltage on emergency buses is simulated to demonstrate disconnection of the emergency buses from the offsite power source. The undervoltage relays shall be calibrated once every 18 months.

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3.9.B

B. Operation with Inoperable Equipment

Whenever the reactor is in Run Mode or Startup Mode with the reactor not in a Cold Condition, the availability of electric power shall be as specified in 3.9.A.1, except as specified in 3.9.B.1.

1. Incoming Power

a. From and after the date incoming power is not available from a startup or emergency transformer, continued reactor operation is permissible under this condition for seven days. At the end of this period, provided the second source of incoming power has not been made immediately available, the NRC must be notified of the event and the plan to restore this second source. During this period, the two diesel generators and associated critical buses must be demonstrated to be operable.

b. From and after the date that incoming power is not available from both start-up and emergency transformers (i.e., both failed), continued operation is permissible, provided the two diesel generators and associated critical buses are demonstrated to be operable, all core and containment cooling systems are operable, reactor power level is reduced to 25% of the rated and NRC is notified within 24 hours of the situation, the precautions to be taken during this period and the plans for prompt restoration of incoming power.

4.9.A (cont'd.)

3.9.B (cont'd.)

2. Diesel Generators

a. From and after the date that one of the diesel generators or an associated critical bus is made or found to be inoperable for any reason, continued reactor operation is permissible in accordance with Specification 3.5.F.1 if Specification 3.9.A.1 is satisfied.

b. From and after the date that both diesel generators are made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 24 hours in accordance with Specification 3.5.F.2 if Specification 3.9.A.1 is satisfied.

c. From and after the date that one of the diesel generators or associated critical buses and either the emergency or startup transformer power source are made or found to be inoperable for any reason, continued reactor operation is permissible in accordance with Specification 3.5.F.1, provided the other off-site source, startup transformer or emergency transformer is available and capable of automatically supplying power to the 4160V critical buses and the NRC is notified within 24 hours of the occurrence and the plans for restoration of the inoperable components.

d. From and after the date that the diesel fuel oil particulate concentration level defined in Surveillance Requirement 4.9.A.2.d cannot be met, restore the diesel fuel oil total particulate concentration to within the acceptable limits within 7 days, or declare the associated Diesel Generator inoperable.

e. From and after the date that the new diesel fuel oil properties defined in Surveillance Requirement 4.9.A.2.e.2 cannot be met, restore the stored diesel fuel oil properties to within acceptable limits within 30 days, or declare the associated Diesel Generator inoperable.

4.9.A (cont'd)

2. Diesel Generators

a. Each diesel-generator shall be started manually and loaded to not less than 50% of rated load for no less than 2 hours once each month to demonstrate operational readiness.

During the monthly generator test the diesel generator starting air compressor shall be checked for operation and its ability to recharge air receivers. The operation of the diesel fuel oil transfer pumps and fuel oil day tank level switches shall be demonstrated, and the diesel starting time to reach rated voltage and frequency shall be logged.

b. Once every 18 months the condition under which the diesel generator is required will be simulated and a test conducted to demonstrate that it will start and accept the emergency load within the specified time sequence. The results shall be logged.

c. Once a month the quantity of diesel fuel available shall be logged.

d. At least once per month the particulate concentration level of the Diesel Fuel Oil Storage Tanks shall be determined in accordance with ASTM D2276-1989, Method A. The total particulate concentration in the diesel fuel oil storage tanks, shall have a limit of less than 10 mg/liter when checked in accordance with ASTM-D2276-1989, Method A.

e. New fuel oil sampling will be performed in accordance with ASTM-D4057-1988 within 30 days upon delivery. Fuel oil testing will be performed in accordance with the following:

1. By verifying in accordance with the tests specified in ASTM-D975-1989a prior to addition to the storage tank that the sample has:

a) An API Gravity of within 0.3 degrees at 60°F, or specific gravity of within 0.0016 at 60/60°F, when

LIMITING CONDITIONS FOR OPERATION

3.5.E (cont'd)

2. From and after the date that one valve in the Automatic Depressurization System is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding seven days unless such valve is sooner made operable, provided that during such seven days the HPCI System is operable.

3. With the surveillance requirements of 4.6.D.5 not performed at the required intervals due to reactor shutdown, a reactor startup may be conducted provided the appropriate surveillance is performed within 12 hours of achieving 113 psig reactor steam pressure.

4. If the requirements of 3.5.E.1 or 3.5.E.2 cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to at least 113 psig within 24 hours.

F. Minimum Low Pressure Cooling and Diesel Generator Availability

1. During any period when one diesel generator is inoperable, continued reactor operation is permissible only during the succeeding seven days unless such diesel generator is sooner made operable, provided that the operable diesel generator and its associated LPCI, Core Spray, and RHR Service Water subsystems are operable and the requirements of 3.5.A.1 are met. If this requirement cannot be met, the requirements of 3.5.F.2 shall be met.

SURVEILLANCE REQUIREMENTS

4.5.E (cont'd)

2. When it is determined that one valve of the ADS is inoperable, the ADS actuation logic for the other ADS valves shall be demonstrated to be operable immediately. In addition, the HPCI System shall be verified to be operable immediately.

F. Minimum Low Pressure Cooling and Diesel Generator Availability

1. When it is determined that one diesel generator is inoperable, the LPCI, Core Spray, and RHR Service Water subsystems associated with the operable diesel generator shall be verified to be operable immediately and daily thereafter. In addition, the operable diesel generator shall be demonstrated to be operable immediately and every three days thereafter.

3.5.F (cont'd.)

2. During any period when both diesel generators are inoperable, continued reactor operation is permissible only during the succeeding 24 hours unless one diesel generator is sooner made operable, provided that both LPCI subsystems, both Core Spray subsystems, and both RHR Service Water subsystems are operable and the reactor power level is reduced to 25% of rated power and the requirements of 3.9.A.1 are met. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor placed in the cold shutdown condition within 24 hours.

3. Any combination of inoperable components in the LPCI, RHR Service Water, and Core Spray systems shall not defeat the capability of the remaining operable components to fulfill the cooling functions.

4. When irradiated fuel [illegible] the reactor vessel and the [illegible] is in the Cold Shutdown Condition, both Core Spray subsystems both LPCI subsystems, and both RHR Service Water subsystems may be inoperable, provided no work is being done which has the potential for draining the reactor vessel. Refueling requirements are as specified in Specification 3.10.F.

5. With irradiated fuel in the reactor vessel, one control rod drive housing may be open while the suppression chamber is completely drained provided that:

   a. The reactor vessel head is removed.

   b. The spent fuel pool gates are open and the fuel pool water level is maintained at a level ≥ 33 feet.

   c. The condensate transfer system is operable and a minimum of 330.000 gallons of water is in the condensate storage tank.

4.5.F (cont'd.)

of 4160 volt switchgear contacts for normal station service transformer the breakers for start-up transformer supply automatically close on fast er. Testability can be demonstrated by control switch operation to open the station service transformer breaker.

2. The emergency station service transformer provides the second for preferred a-c power. Control circuitry is arranged so that upon opening 160 volt switchgear contacts for incoming normal or start-up supply to the volt critical (emergency) bus, the breaker for emergency transformer atically closes. Testability can be demonstrated by control switch operation en the incoming breaker to the critical bus.

Provision is also made in the under-voltage protective scheme for 4160 volt switchgear on critical buses 1F and 1G for automatic assumption of ical loads by the emergency transformer before standby a-c power is used as her described below. This test can be repeated during normal operation.

.2 Standby A-C Power (Diesel Generators) Test Capability[5]

1. The undervoltage protective schemes for 4160 volt switchgear on tical buses 1F and 1G provide for automatic start of the diesel generators and mption of load upon loss of voltage on bus 1F or 1G. The protective scheme ides the following functions on each critical bus:
er.

a. Clear the bus of all motor loads excepting supply to the volt critical unit substation.

b. Isolate the bus by opening all incoming breakers.

c. Start the diesel generator on emergency basis bringing it to full speed and voltage.

d. Close the generator breaker to the critical bus when the chine is at rated speed and voltage.

e. Signal to GE logic for Residual Heat Removal (RHR) and core ray systems that diesel power is available for timed starts.

f. Start the station service water standby pumps after pproximately 15 second delay.

g. Remain in running status until manual shutdown.

2. Testability of the protective scheme can be demonstrated during normal station operation by opening incoming breaker 1FE (1GE) to prevent actual standby a-c connection to the bus, and by using test switches at either switchgear buses 1F or 1G in proper sequence to allow all protective relays to operate as required. The test switches provide the following functions:

a. Temporary removal of undervoltage trips from all motors on the critical bus.

b. Simulate opening of incoming breaker 1FA (1GB) cutting off normal and start-up supply.