# SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

NEMAHA COUNTY, NEBRASKA

DOCKET NO. 50-298



ISSUED: FEBRUARY 14, 1973

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 ections 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.

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failure of the main steamlines outside the containment or the turbine-condenser and because of the conservative nature of the staff's analysis of the dose consequences. Following our review and approval of the improved MSLIV surveillance test program, the appropriate portions of that test program will be included in the Technical Specifications.

6.2.3 Leakage Testing Program

The primary containment and components which will be subjected to containment test conditions were designed so that periodic integrated leakage rate testing can be conducted at peak calculated accident pressure and reduced pressures. We have reviewed the proposed test procedures for determination of the primary containment overall leakage, as well as penetration and isolation valve leakage, for both preservice and inservice containment leakage tests.

Penetrations, including personnel and equipment hatches and airlocks, and isolation valves, have generally been designed with the capability of being individually leak tested at peak calculated accident pressure. Large hatches have been strengthened structurally to sustain the pressures of individual leak tests. 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.

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

9/16/94

CNS ENFORCEMENT CONFERENCE

PREBRIEF

CONTAINMENT INTEGRITY

| | |
|---|---|
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| ATTACHMENT D | Systematic Assessment of Licensee Performance |
| ATTACHMENT E | Technical Specification and T.S. Basis |
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A/80

DRAFT
NOTICE OF VIOLATION
COOPER NUCLEAR STATION

9414-01

10 CFR Part 50, Appendix B, Criterion III, states, in part, that "[m]easures shall be established to assure that . . . the design basis . . . are correctly translated into . . . specifications, drawings . . . . These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled."

1. The Cooper Nuclear Station Updated Safety Analysis Report, Appendix F, "Conformance to AEC General Design Criteria," sates, in part, the " . . . the purpose of this appendix [is] to show that the design and construction of the Cooper Nuclear Station has been performed in accordance with these general design criteria."

   Contrary to the above, Flow Diagram No. 2028, "Reactor Building and Drywell Equipment Drain System," contained safety-related isolation valves but was not included on the safety-related drawing list as of July 1, 1994, and some safety-related components were not included on the drawing.

2. Draft General Design Criteria, Criterion 53, July 1967, in accordance with Appendix F to the USAR, states that "[a]ll lines which penetrate the primary containment and which communicate with the reactor vessel or the primary containment free space [were] provided with at least two isolation valves (or equivalent) in series."

   1. Contrary to the above, as of May 14, 1994, many penetrations were identified without redundant valving. These penetrations included, but were not limited to, penetrations X-21, X-22, X-25, X-29E, X-30E/F, X-33E/F, X-209A/B/C/D, and X-218.

   2. Contrary to the above, as of February 22, 1994, ten manual operated vents, drains, or test connections had single manual valves for containment isolation.

3. Draft General Design Criterion 1, in accordance with Appendix F to the Updated Safety Analysis Report, states that ". . . those systems and components of the station which [had] a vital role in the prevention or mitigation of consequences of accidents affecting the public health and safety [were] designed and constructed to high quality standards . . . ."

   General Electric Design Specification No. 22A1153, "Codes and Industrial Standard," Revision 1, states, in Note 3 of the Appendix, that "[p]iping, 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."

Contrary to the above, the licensee failed to design, fabricate and erect approximately 300 containment penetrations to the standards specified in USAS B31.7-1969.

9414-02

Technical Specification 4.7.A.2.f.1 states, in part, that "local leak rate tests (LLRT's) shall be performed on the primary containment testable penetrations and isolation valves . . . The total acceptable leakage for all valves and penetrations other than the MSIV's is 0.60 La."

1. Contrary to the above, as of May 14, 1994, the licensee failed to provide for Type C local leak rate testing of 68 components passing through 54 containment penetrations.

2. Contrary to the above, as of July 11, 1994, the total leakage for the valves and penetrations that had never been tested, with three tests remaining, exceeded the 0.60 La limit allowed by Technical Specifications. The 0.60 La limit was 5.37 scmh (189.60 scfh) and the leakage for the valves that had never been tested was in excess of 17.66 scmh (623.57 scfh).

3. Contrary to the above, several instrument pressure switches had not had local leak rate testing performed after being isolated from the containment integrated leak rate test.

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

Docket: 50-298
License: DPR-46
EA 94-165

Nebraska Public Power District
ATTN: Guy R. Horn, Vice President - Nuclear
P.O. Box 499
Columbus, Nebraska 68602-0499

SUBJECT: NRC INSPECTION REPORT 50-298/94-14

This refers to the inspection conducted by Ms. P. A. Goldberg and Mr. C. J. Paulk, of this office, and Mr. G. Cha, an NRC consultant, on June 13 through August 12, 1994. The inspection included a review of activities authorized for your Cooper Nuclear Station facility. At the conclusion of the inspection, the findings were discussed with you and those members of your staff identified in the enclosed report.

Areas examined during the inspection are identified in the report. Within these areas, the inspection consisted of selective examinations of procedures and representative records, interviews with personnel, and observation of activities in progress. The purpose of the inspection was to determine whether activities authorized by the license were conducted safely and in accordance with NRC requirements.

Based on the results of this inspection, two apparent violations were identified and are being considered for escalated enforcement action in accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions" (Enforcement Policy), 10 CFR Part 2, Appendix C. Accordingly, no Notice of Violation is presently being issued for these inspection findings. Please be advised that the number and characterization of apparent violations described in the enclosed inspection report may change as a result of further NRC review.

The apparent violations are of concern because it is apparent that the primary containment was inoperable for an undetermined period of time. Additionally, it is apparent that there was a breakdown in your design control program, dating back to initial construction, which you have had numerous opportunities to identify and correct. The apparent breakdown in design control contributed to the problems associated with the primary containment, as well as other recent problems identified at the Cooper Nuclear Station.

An enforcement conference to discuss these apparent violations has been scheduled for September 16, 1994. The decision to hold an enforcement conference does not mean that the NRC has determined that a violation has occurred or that enforcement action will be taken. The purposes of this conference are to discuss the apparent violations, their causes and safety significance; to provide you the opportunity to point out any errors in our inspection report; and to provide an opportunity for you to present your proposed corrective actions. In addition, this is an opportunity for you to provide any information concerning your perspectives on (1) the severity of the violation(s), (2) the application of the factors that the NRC considers when it determines the amount of a civil penalty that may be assessed in accordance with Section VI.B.2 of the Enforcement Policy, and (3) any other application of the Enforcement Policy to this case, including the exercise of discretion in accordance with Section VII. You will be advised by separate correspondence of the results of our deliberations on this matter. No response regarding these apparent violations is required at this time.

This enforcement conference, which will also address issues involving the control room filtration system (EA 94-164) and the electrical distribution system (EA 94-166), will be open to public observation in accordance with the Commission's continuing trial program as discussed in the enclosed Federal Register Notices (Enclosure 2). Although not required, we encourage you to provide your comments on how you believe holding this conference open to public observation affected your presentation and your communications with the NRC.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosures will be placed in the NRC Public Document Room.

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

Thomas P. Gwynn, Director
Division of Reactor Safety

Enclosures:
1. Appendix - NRC Inspection Report
   50-298/94-14
2. Federal Register Notices

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*Previously concurred

## APPENDIX

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Inspection Report: 50-298/94-14

EA No.: 94-165

License: DPR-46

Licensee: Nebraska Public Power District
P.O. Box 499
Columbus, Nebraska

Facility Name: Cooper Nuclear Station

Inspection At: Brownville, Nebraska

Inspection Conducted: June 13 through August 12, 1994

Inspectors: P. A. Goldberg, Reactor Inspector, Engineering Branch
Division of Reactor Safety

C. J. Paulk, Reactor Inspector, Engineering Branch
Division of Reactor Safety

Accompanied By: G. Cha, Consultant

Approved: ______________________________ __________
T. F. Westerman, Chief, Engineering Branch Date
Division of Reactor Safety


Inspection Summary

Areas Inspected: Reactive, announced inspection of the licensee's actions concerning containment penetration problems found as the result of reviews and inspections performed by the licensee. In addition, issues related to motor-operated valves and switch calibration for drywell instrumentation were reviewed.

Results:

- As a result of corrective actions for a previously identified violation, the licensee was reviewing the design function of all piping and equipment pressure parts to determine if they were properly classified. This effort was scheduled to be completed in October 1994 and will be evaluated during followup of Enforcement Action 93-137 (Section 2.1).

- The licensee prepared 15 design change packages to bring the containment penetrations into compliance with the draft General Design Criteria, Criterion 53, July 1967, as stated in Appendix F to the Updated Safety Analysis Report, and 10 CFR Part 50, Appendix J. Seven of these design change packages were reviewed and no concerns were identified (Section 2.1.1).

- During the inspection, the inspectors found that Flow Diagram No. 2028, which depicted 80 safety-related components, was not accurate since it failed to include some safety-related components. The failure to include containment isolation valves on the drawing and the failure to identify the drawing as safety-related was identified as an apparent violation of 10 CFR Part 50, Appendix B, Criterion III (Section 2.1.2).

- The licensee determined that the containment isolation valves in 54 penetrations had not had Type C local leak rate tests performed on 68 of the components passing through the penetrations. The systems associated with these valves were classified as nonessential. However, the containment isolation valves were required to function to prevent the release of the post-accident containment atmosphere. The failure to perform Type C local leak rate tests was identified as an apparent violation of Technical Specification 4.7.A.2.f.1 (Section 2.1.3).

- The total leakage of the local leak rate tests performed on components previously not tested exceeded the Technical Specification limit for leakage to ensure containment integrity. This was identified as an apparent violation of Technical Specification 4.7.A.2.f.1. (Section 2.1.3).

- The licensee identified a number of examples where penetrations were found to lack redundant containment isolation. The failure to have redundant containment isolation barriers was identified as an apparent violation of 10 CFR Part 50, Appendix B, Criterion III (Section 2.1.4).

- The licensee identified approximately 300 examples of components associated with containment penetrations which were not classified as essential. The failure to design, fabricate, and erect the containment isolation barriers to quality standards that reflected the importance of the safety function was identified as an apparent violation of 10 CFR Part 50, Appendix B, Criterion III (Section 2.1.5).

- The licensee determined that Containment Isolation Valve RHR-MOV-MO27B was not capable of passing its local leak rate test. The licensee decided to move the primary containment isolation function from the leaking valve to another valve. This change was accomplished by use of a safety evaluation that was performed in accordance with 10 CFR 50.59. It was concluded that the licensee's change of primary containment isolation boundary was adequately justified, and appropriate procedural controls were identified (Section 2.2).

- During a review of the licensee's actions concerning the lack of cleanliness inside motor-operated valve limit switch compartments, it was found that the licensee had not entered the recommended corrective actions into the corrective action tracking system. This was a concern because of the lengthy amount of time allowed to pass before the corrective actions were due which increased the chances for similar events to occur (Section 2.2).

- The failure to perform local leak rate testing for several instrument pressure switches was identified as an apparent violation of Technical Specification 4.7.A.2.f.1 (Section 2.3).

- Unresolved Item 298/9403-01, concerning ten valves used as single manual valves for containment isolation, was closed. These ten single isolation valves without a second barrier were identified as another example of an apparent violation of 10 CFR Part 50, Appendix B, Criterion III (Section 2.4).

Summary of Inspection Findings:

- Example 1 of apparent Violation 298/9414-01 was identified (Section 2.1.2).

- Example 2 of apparent Violation 298/9414-01 was identified (Sections 2.1.4 and 2.4).

- Example 3 of apparent Violation 298/9414-01 was identified (Section 2.1.5).

- Example 1 of apparent Violation 298/9414-02 was identified (Section 2.1.3).

- Example 2 of apparent Violation 298/9414-02 was identified (Section 2.1.3).

- Example 3 of apparent Violation 298/9414-02 was identified (Section 2.3).

- Inspection Followup Item 298/9414-03 was opened (Section 2.2.2).

- Unresolved Item 298/9403-01 was closed (Section 2.4).

Attachments:

- Attachment - Persons Contacted and Exit Meeting

## DETAILS

### 1 PLANT STATUS

During this inspection period, Cooper Nuclear Station was shutdown.

### 2 ENGINEERING (37550 and 92903)

This inspection was conducted to review Cooper Nuclear Station's actions concerning problems found with containment penetrations. In addition, licensee's actions concerning dirty torque switches on motor-operated valves and time-delay relays for the emergency diesel generators were reviewed.

The inspectors reviewed the licensing basis for the Cooper Nuclear Station in order to evaluate the problems associated with the containment penetrations against the appropriate criteria. The inspectors found that the licensee was committed to the draft "General Design Criteria for Nuclear Power Plant Construction Permits," issued in July 1967. This commitment was documented in the Updated Safety Analysis Report (USAR), Appendix F. The licensee was evaluated and licensed to the draft General Design Criteria, July 1971, and 10 CFR Part 50, Appendix J, as stated in Sections 3.1 and 6.2.3, respectively, of "Safety Evaluation by the Directorate of Licensing U. S. Atomic Energy Commission in the Matter of Nebraska Public Power District, Cooper Nuclear Station, Nemaha County, Nebraska, Docket No. 50-298," dated February 14, 1973. The inspectors also found that the licensee acknowledged the applicability of the draft General Design Criteria in the draft design criteria document prepared for the containment systems.

With regard to the applicability of 10 CFR Part 50, Appendix B, to Cooper Nuclear Station, 10 CFR 50.54(a)(1) requires, that each plant licensed subject to the quality assurance criteria in Appendix B shall implement pursuant to 10 CFR 50.34(b)(6)(ii), the quality assurance program described or referenced in the safety analysis report. The final 10 CFR Part 50, Appendix B, rule was issued on June 27, 1970, and the operating license for Cooper Nuclear Station was issued on January 18, 1974.

On the basis of the above, the inspectors reviewed the containment penetration issues against the draft General Design Criteria, July 1967, as described in the USAR, Appendix F; 10 CFR Part 50, Appendix B; 10 CFR Part 50, Appendix J; and, applicable licensee procedures, design specifications, and Technical Specifications.

#### 2.1 Containment Penetrations

The licensee prepared Special Procedure 94-202, dated May 17, 1994, "Containment Walkdown," to inspect each primary containment penetration and the piping to the outboard containment isolation barrier. The purpose of the inspection was to support development of the containment design criteria document: to comply with a commitment, made in response to a violation in NRC Inspection Report 50-298/93-17, to review all containment penetrations; and, to support the upgrade of the licensee's program for primary reactor containment leakage testing in accordance with 10 CFR Part 50, Appendix J.

The inspectors reviewed Special Procedure 94-202 and found that licensee inspection of each primary containment penetration, and components which were in the containment isolation system, was required. This inspection was also in support of the preparation of as-built drawings. The inspectors concluded that the procedure was adequate.

During the inspections, the licensee determined that 46, of the 255 primary containment penetrations inspected, had been incorrectly classified nonessential at the time of plant construction and were not contained in the inservice inspection program. In addition:

- The licensee determined that a number of penetrations had not had local leak rate tests performed in accordance with the requirements of 10 CFR Part 50, Appendix J.

- A number of penetrations did not have two containment barriers outside of the primary containment in accordance with draft General Design Criteria, Criterion 53, July 1971.

- A number of instrument lines and valves within the containment pressure boundary were classified as nonessential.

- 294 welds in the containment isolation barriers were found to either never have had nondestructive examinations performed or the qualification records could not be located.

Many penetrations were improperly classified during the construction of the plant. The inspectors attempted to determine how such a problem occurred. While no definite answer was provided, the licensee stated that the architect engineer apparently had missed a note in the General Electric design specification which resulted in the improper classification of containment penetrations and associated components.

The inspectors found that equipment and components classified as essential were designed, fabricated, installed, and tested in accordance with USAS B31.7-1969, "Nuclear Power Piping." Equipment and components classified as nonessential were designed, fabricated, installed, and tested in accordance with USAS B31.1.0-1967, "Power Piping."

On the basis of these codes, the architect engineer designed the equipment and components. The architect engineer, however, apparently missed a note in General Electric Design Specification 22A1153, "Codes and Industrial

Standard," Revision 1. Note 3 of five, to this specification, stated that "[p]iping, which [was] 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."

In Appendix A of the Updated Safety Analysis Report, the licensee provided definitions for the classification of piping and equipment pressure parts. Class C was assigned for "[p]iping and equipment pressure parts . . . for a high integrity system," such as the containment vessel. To meet this classification, the licensee applied the requirements of USAS B31.7-1969 for Class II piping. Therefore, the penetration piping and equipment pressure parts should have been designed, fabricated, installed, and inspected accordingly.

As a result of corrective actions for a previously identified violation, the licensee was reviewing the design function of all piping and equipment pressure parts to determine if they were properly classified. This effort was scheduled to be completed in October 1994 and will be evaluated during followup of Escalated Action 93-137 for violations cited in NRC Report 50-298/93-17.

The inspectors observed 17 liquid penetrant tests of welds that were originally designed, fabricated, installed, and tested in accordance with USAS B31.1.0-1967 rather than USAS B31.7-1969. The inspectors observed one weld that exhibited indication of weld slag. The licensee rejected that weld. Subsequently, the licensee chipped the weld slag off and retested the weld satisfactorily.

The licensee completed the liquid penetrant testing on 260 welds that had been improperly classified without identifying any other weld that was questionable. The inspectors concluded that the licensee had performed testing in accordance with USAS B31.7-1969 for the welds that had no documentation of such inspection during the construction of the plant.

2.1.1 Design Modifications

To address the concerns identified by the licensee's inspections of primary containment penetrations, design changes were prepared. The inspectors reviewed the 7 design change packages, discussed in the following sections, out of a total of 15 which the licensee was preparing to bring the penetrations into compliance with the draft General Design Criteria, Criterion 53, as stated in the USAR, Appendix F, and 10 CFR Part 50, Appendix J. During the licensee's verification and validation of the draft design criteria document for the primary containment, the identification of problems led to a complete scrutiny of all penetrations (approximately 300). As a result of the licensee's efforts, 99 penetrations were identified with problems other than classification. The problems were categorized into 11 types, which ranged from missing caps to inadequate design.

2.1.1.1 Design Change 94-212 Torus Penetration X-218 Modification

Penetration X-218, as-found, consisted of a ball valve on the torus shell with eight thermocouples routed through it. A sealant of unknown composition filled the void and acted as a containment barrier. The thermocouples were installed under Design Change 76-17, Revision 2, but were never placed in service. The design change was later voided because there were no provisions to calibrate the temperature elements and the equipment was abandoned in place. The penetration was not local leak rate testable, and was not on the local leak rate test list.

The design change consisted of removing all thermocouple hardware and the ball valve, and installing a 5.08 cm (2 in) socket welded cap, which would function as a primary containment boundary, hence the penetration would be restored to its original design. The design change was classified as essential and Seismic Class IS. The 5.08 cm (2 in) socket welded cap was purchased as essential material.

The applicable design code for fabrication and installation was USAS B31.7-1969. Weld integrity was checked by 100 percent liquid penetrant nondestructive examination and pneumatically tested to 1.25 of design pressure. The results of the liquid penetrant tests were discussed in Section 2.1 of this report.

The inspectors did not identify any concerns with this design change.

2.1.1.2 Design Change 94-212A Electrical Penetrations X-209A through D Modifications

Design Change 94-212A consisted of two parts: the first part, associated with Penetrations X-209A and X-209C, involved modifying the two thermocouple penetrations to permit periodic local leak rate testing as required by 10 CFR Part 50, Appendix J; and, the second part, associated with Penetrations X-209B and X209D, involved permanently capping the two penetrations.

The inspectors did not identify any concerns with this modification. The inspectors reviewed Design Change 94-212A in its entirety, verified the design changes during the walkdown, and concluded that it was acceptable.

2.1.1.3 Design Change 94-212B Penetrations X-43 and X-44 Testable Flanges

This design change replaced two flanged piping joints near Penetrations X-43 and X-44 with flanges incorporating a testable, double o-ring design. The new design permitted these joints, which were part of the primary containment boundary, to be periodically tested in accordance with 10 CFR Part 50, Appendix J.

The design change was classified essential and Seismic Class IS. All pressure retaining material was procured essential. The inspectors did not identify any concerns with this modification.

2.1.1.4 Design Change 94-212D Penetration X-21 and X-22 Upgrade

The purpose of Design Change 94-212D was to enhance the isolation capacity for both the service air and instrument air headers, upstream of Penetrations X-21 and X-22, respectively. Additionally, the modification provided test connections for periodically performing local leak rate tests of the containment isolation valves in accordance with 10 CFR Part 50, Appendix J requirements.

The inspectors did not identify any concerns with this modification.

2.1.1.5 Design Change 94-212E Primary Containment Integrity Issues

Design Change 94-212E consisted of three parts. The first part removed Swagelok caps and installed valves and caps at ten test connections for instruments which were in direct communication with primary containment. The ten test connections were PC-PT-1B2, -4B2, -5B2, -1A1, -4A1, -5A1, -2104A, -2104B; and PC-PI-2104AG, -2104BG. Also, at PC-DPT-3A1, Drain Valve PC-V-243 was missing and was reinstalled. The second part of the modification removed unnecessary tees located in instrument lines which communicated directly with primary containment and replaced them with unions, elbows or installed welded caps. The third part of the modification cut and capped 14 instrument lines which penetrated primary containment and had previously been spared out. The valves were removed and welded caps installed on the lines at the penetrations.

The inspectors did not identify any concerns with this modification.

2.1.1.6 Design Change 94-212H Post Accident Sampling System Modifications and Penetration X-51F Upgrade

The purpose of Design Change 94-212H was to replace the existing nonessential post-accident sampling system Containment Atmosphere Sample Isolation Valve PAS-AOV-3AV with two qualified 1.27 cm (0.5 in) air-operated valves, PC-AOV-247AV and PC-AOV-248AV, at Penetration X-51F. In addition, test connections with capped manual valves were provided.

The inspectors did not identify any concerns with this modification.

2.1.1.7 Maintenance Work Requests 94-2978 and 94-3116

These maintenance work requests installed caps and plugs to provide the second barrier for containment isolation. During the licensee's inspections, numerous vents, drains, and test connections, having direct access to the primary containment, were found to lack a second barrier. These were identified and a cap or plug was added, depending on the installation.

The inspectors reviewed Maintenance Work Requests 94-2978 and 94-3116 and concluded that both were acceptable.

2.1.2 Drawing Control

During a review of the penetration walkdown packages, the inspectors noted that some of the containment isolation valves identified on the penetration drawings, and existing in the plant, were not included on Flow Diagram No. 2028, "Reactor Building and Drywell Equipment Drain System," Revision N27. The inspectors found that Flow Diagram No. 2028 was not included on the safety-related drawing list in accordance with Cooper Nuclear Station Engineering Procedure 3.8, "Drawing Control Procedure," Revision 7. The inspectors concluded that the drawing was inaccurately classified as a result of the problems associated with classification of components as discussed in Section 2.1, above.

Cooper Nuclear Station Engineering Procedure 3.8, "Drawing Control Procedure," Revision 7, defined a safety-related drawing as "a drawing or schematic describing the features, characteristics, design or location of safety-related components, systems, or structures." The procedure also stated that any new drawing, or portion of a new drawing, classified as safety-related would be added to the safety-related drawing list.

During the inspection, the licensee initiated Condition Report 94-0309 in response to the inspectors' finding. The condition report stated that the subject drawing depicted a total of 80 safety-related components, but was not included on the safety-related drawing list. In response to this condition report, the licensee identified an additional 13 drawings, with safety-related components, that were not included on the safety-related drawing list.

Additionally, Draft General Design Criteria, Criterion 1, July 1967, in accordance with Appendix F to the USAR, stated that ". . . those systems and components of the station which [had] a vital role in the prevention or mitigation of consequences of accidents affecting the public health and safety [were] designed and constructed to high quality standards . . ."

The inspectors identified five missing valves on Flow Diagram 2028. These valves were associated with Penetrations X-18, X-30E, X-30F, X-33E, and X-33F. For Penetration X-18, an unlabeled vent isolation valve downstream of Valve RW-254 was not on the drawing. For Penetration X-30E, Valve NBI-502, the manual containment isolation valve for the air-to-vessel flange leak off detection air-operated valve, was not shown. For Penetration X-30F, Valve MS-900, the manual containment isolation valve for the air-to-reactor vessel head vent was not shown. For Penetration X-33E, Valve MS-501, the manual containment isolation valve for the air-to-vessel flange leak off detection air-operated valve, was not shown. For Penetration X-33F, Valve MS-899, the manual containment isolation valve for the air-to-vessel head vent, was not shown.

Appendix B to 10 CFR Part 50, Criterion III, requires that "[m]easures shall be established to assure that . . . the design basis . . . are correctly translated into . . . drawings . . . These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled."

The inspectors identified the licensee's failure to properly classify drawings as safety-related and the failure to include safety-related components on the drawing as Example 1 of an apparent violation of 10 CFR Part 50, Appendix B, Criterion III (298/9414-01).

2.1.3 Local Leak Rate Tests

The licensee determined that the containment isolation valves in 54 penetrations did not have Type C local leak rate tests performed on 68 of the components passing through the penetrations. The systems associated with these valves were classified as nonessential since they did not have to function post-accident.

Containment isolation valves, however, were required to function to prevent the release of the post-accident containment atmosphere. Containment isolation valves, as defined in 10 CFR Part 50, Appendix J, would be "any valve which [was] relied upon to perform a containment isolation function." Type C tests were required for containment isolation valves that "provide[d] a direct connection between the inside and outside of the primary containment under normal operation . . . ; [were] required to close automatically upon receipt of a containment isolation signal . . . ; and [were] required to operate intermittently under post-accident conditions."

In accordance with draft General Design Criteria, Criterion 57, July 1967, as stated in Appendix F to the USAR, the licensee was required to demonstrate the ". . . functional performance of containment system isolation valves and monitoring valve leakage."

Technical Specification 4.7.A.2.f.1 required that "local leak rate tests (LLRT's) shall be performed on the primary containment testable penetrations and isolation valves . . . ."

The inspectors identified the failure to perform local leak rate tests as Example 1 of an apparent violation of Technical Specification 4.7.A.2.f.1 (298/9414-02).

The licensee had begun performing local leak rate tests on the identified components. The inspectors attempted to review the results of this testing. The licensee had not developed a running total of the results of the as-found tests to determine the status of the primary containment and its ability to perform as designed. The inspectors were informed that one penetration (X-22), on June 13, 1994, had in excess of 17 scmn (600 scfh) leakage. This significantly exceeded Technical Specification 4.7.A.2.f.1 leakage limit of 0.60 La (5.37 scmn (169.6 scfh)).

The total leakage of the local leak rate tests being performed on containment isolation components not previously tested, with three remaining to be tested, was in excess of 17.66 scmh (623.57 scfh). This value did not include any leakage from those components previously tested, nor did it reflect the actual leakage through penetration X-22, which was listed only as greater than 17 scmh (600 scfh). As noted above, Technical Specification 4.7.A.2.f.1 established the limit for local leak rates to be 5.37 scmh (189.60 scfh). This limit was established to ensure containment integrity.

The licensee had initiated a licensee event report on July 5, 1994, to address the identification of penetrations that had not been tested as required by 10 CFR Part 50, Appendix J. The licensee stated that the causes would be addressed in a supplement to the report.

On the basis of the test results for the newly tested components, the inspectors concluded that the licensee had exceeded the Technical Specification limit for leakage to ensure containment integrity for an extended period without taking the required corrective actions. As such, this is identified as Example 2 of an apparent violation of Technical Specification 4.7.A.2.f.1 (298/9414-02).

2.1.4 Redundant Containment Isolation Barriers

The licensee inspected approximately 300 penetrations during the performance of Special Procedure 94-202. A number of those penetrations were found to lack redundant containment barriers.

The licensee identified some penetrations with both isolation valves located outside the primary containment. However, between the containment wall and the first isolation valve outside containment, there existed a single vent, drain, or test connection valve. Examples of this type of single barrier were Penetrations X-21, X-22, and X-25.

Some penetrations were identified by the licensee with only a single isolation valve outside of containment. Penetrations X-29E, X-30E/F and X-33E/F were examples.

Penetrations X-216 and X-209A/B/C/D had thermocouple wires routed in piping through the penetrations. On the outside of containment was an open valve, incapable of closing, with an unidentified sealant that could not be determined to be qualified. These penetrations were determined to have an unqualified barrier.

Appendix B to 10 CFR Part 50, Criterion III, requires that "[m]easures shall be established to assure that . . . the design basis . . . are correctly translated into specifications . . . . These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled."

Additionally, in accordance with draft General Design Criteria, Criterion 53, July 1967, as stated in Appendix F to the USAR, "[a]ll lines which penetrate the primary containment and which communicate with the reactor vessel or the primary containment free space [were] provided with at least two isolation valves (or equivalent) in series."

The inspectors identified the failure to have redundant barriers as Example 2 of an apparent violation of 10 CFR Part 50, Appendix B, Criterion III (298/9414-01).

2.1.5 Classification of Primary Containment Isolation Barriers

The licensee identified that approximately 300 examples of components associated with containment penetrations were not classified as essential. Draft General Design Criteria, Criterion 1, July 1967, as stated in Appendix F to the USAR, required ". . . those systems and components of the station which [had] a vital role in the prevention or mitigation of consequences of accidents affecting the public health and safety [were] designed and constructed to high quality standards . . ."

General Electric Design Specification No. 22A1153, "Codes and Industrial Standard," Revision 1, states, in Note 3 of the Appendix, that "[p]iping, 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."

In addition, 10 CFR Part 50, Appendix B, Criterion III, requires that "[m]easures shall be established to assure that . . . the design basis . . . are correctly translated into specifications . . . These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled."

The licensee concluded that those components not classified as essential were designed, fabricated, and erected to quality standards that did not reflect the importance of the safety function to be performed in accordance with 10 CFR Part 50, Appendix B, Criterion III; General Electric Design Specification No. 22A1153, Revision 1; or Appendix F to the USAR.

The failure to design, fabricate and erect the containment isolation barriers to quality standards that reflected the importance of the safety function was identified as Example 3 of an apparent violation of 10 CFR Part 50, Appendix B, Criterion III (298/9414-01).

2.1.6 Containment Penetration Inspections

The inspectors reviewed a number of primary containment penetrations previously inspected by the licensee. For those penetrations, the inspectors concluded that the licensee's inspection had been accurate and the marked-up drawings reflected the actual condition in the plant.

## 2.2 Motor-Operated Valve Issues

On December 20, 1993, as documented in NRC Inspection Report 50-298/93-29, Valve HPCI-MOV-MO17 failed to stroke. The licensee formed a problem resolution team to investigate that failure. The team issued a report on January 7, 1994, that documented the team's findings. Those findings were that the failure was the result of fiberglass fragments between the limit switch contacts. The team presented this report as the response to Nonconformance Report 93-270 in order to recommend corrective actions.

On March 14, 1994, as documented in NRC Inspection Report 50-298/94-09, excessive leakage was noted during the venting of piping between Valves RHR-MOV-MO25A and -MO27A. In this instance, the licensee determined that the problem was related to foreign material deposited on the valve seat after maintenance that breached the residual heat removal system boundary.

On May 27, 1994, the licensee reported that Valve MOV-MO16 was found "partially deenergized" after attempting to close the valve. The licensee's investigation led to the identification of "particles" stuck between the contacts of the torque switch.

On June 20, 1994, the licensee reported that Valve RHR-MOV-MO27B was not capable of passing its local leak rate test. At the time of this inspection, the licensee had not determined a root cause for the failure. The licensee had decided to move the primary containment isolation function from Valves RHR-MOV-MO25A(B) and -MO27A(B) to Valves RHR-CV-26CV(27CV), RHR-MOV-MO274A(B), and -MO25A(B). The licensee performed this change by use of a safety evaluation that was performed in accordance with 10 CFR 50.59.

### 2.2.1 Safety Evaluation Review

The inspectors reviewed the safety evaluation and found that the evaluation was thorough and in accordance with the requirements of 10 CFR 50.59. The inspectors noted that, in order to accomplish this change, the licensee had to change operating procedures to prevent the opening of either RHR-MOV-MO274 valve and to ensure that the motor operator will remain deenergized when the reactor coolant temperature was above 100°C (212°F). Another change to the procedures was that shutdown cooling could only be initiated when the reactor pressure was less than 344.7 kPa (50 psig). The inspectors concluded that the licensee's change of primary containment isolation boundary was adequately justified, and appropriate procedural controls were identified.

### 2.2.2 Limit Switch Compartment Cleanliness

During review of the licensee's actions related to the lack of cleanliness inside the limit switch compartments, the inspectors found that the licensee had proposed a completion date of September 1994 for the corrective actions related to the failure of Valve HPCI-MOV-MO17. The licensee had not entered the corrective actions into its tracking system.

This was a concern to the inspectors for two reasons. The lengthy amount of time allowed to pass before the corrective actions were due increased the chances for similar events to occur. In this case, a similar event did occur when Valve MOV-MO16 failed to operate properly. The other concern was the failure to timely incorporate the corrective actions into the tracking system to assure that management is provided with an appropriate status of corrective actions. The licensee had indicated that the failure to track was a backlog problem because of an administrative overload. In each case, a condition report had been issued and initial corrective action initiated. The inspectors were concerned that the licensee would have failed to perform these corrective actions without the NRC inspection into the motor-operated valve issues.

The licensee did form a condition resolution team to review the failure of Valve MOV-MO16. This team had not issued its report, therefore, the inspectors did not review the licensee's actions for that failure. The review of the licensee's actions is considered to be an inspection followup item (298/9414-03).

2.2.3 Analysis of Other Valve Concerns

The failures of Valves RHR-MOV-MO27A(B) presented other concerns. One concern was related to the control of foreign materials when systems were breached. The inspectors noted that corrective actions had not been approved for the March event when weld slag was determined to be the cause of the problem. When questioned by the inspectors, the responsible engineer stated that this issue had been given lower priority and, in essence, that there was a lack of personnel to ensure the corrective action process was timely. Another concern was that the licensee had not considered any interim actions to prevent foreign material to get into systems other than a memorandum to maintenance personnel informing them of recent problems and instructing them to be careful.

The inspectors concluded that management attention was warranted in the areas of foreign material exclusion and the corrective action programs. The corrective action program was considered to warrant the attention because of the fact that it had been implemented only recently and the inspectors noted these concerns.

## 2.3 Switch Calibration

The licensee identified that several instrument pressure switches in Racks 25-5 and -6, subject to drywell pressure, were isolated during the performance of the containment integrated leak rate tests performed in accordance with the ASME Boiler and Pressure Vessel Code. The licensee stated that these instruments were isolated because the licensee's staff thought the test pressure (approximately 400 kPa (58 psi)) would damage the instruments. Local leak rate testing had not been performed in lieu of opening the valves to the racks during integrated leak rate testing. On July 8, 1994, Surveillance Procedure 6.3.1.1.2, Revision 0, "Primary Containment Instrument

Local Leak Rate Tests," was issued to initiate local leak rate testing for these pressure switches. The pressure switches on Racks 25-5 and -6 include: PC-PS-12A, B, C, and D; PC-PS-101A, B, C, and D; PC-PS-119A, B, C, and D; PC-PS-16; and PC-PT-512A and B. The pressure switches perform scram, containment isolation, and emergency core cooling system initiation upon receiving a drywell pressure signal of 13.7 kPa (2 psig) or greater. The licensee contacted the instrument vendor and was notified that the instruments could withstand the test pressure, but should be calibrated after the test to ensure there was no shift in the operating characteristics of the instruments. The licensee stated that these instruments would be calibrated after being subjected to the pressure of the containment integrated leak rate test. The failure to perform local leak rate tests is identified as Example 3 of an apparent violation of Technical Specification 4.7.A.2.f.1 (298/9414-02).

2.4 <u>(Closed) Unresolved Item 50-298/9403-01: Use of Single Manual Valve for Containment Isolation</u>

NRC Inspection Report 50-298/94-03 summarized the inspection conducted during January 2 through February 12, 1994. The report discussed the use of a single manual valve for containment isolation, which was determined to be an Unresolved Item (298/9403-01) pending additional NRC review. The valves in question were all manual operated vents, drains, or test connections; a total of ten valves were affected.

During this inspection, the inspectors determined that the ten valves, identified in the earlier inspection, had been modified by means of a maintenance work request. The modification consisted of adding either a cap or plug, which acted as a second barrier for containment isolation. This design philosophy was consistent with draft General Design Criteria 53, as stated in Appendix F to the USAR. All material used in the maintenance work requests were classified essential, and their certification and traceability were available.

The licensee submitted its response to NRC Inspection Report 50-298/94-03 by letter dated May 31, 1994. The response stated that the licensee was reconstituting the design basis for the primary containment system and would evaluate the issue within that task. In addition, the licensee advised that it was pursuing efforts to resolve NRC concerns involving the identification and control of manual primary containment isolation valves, or more appropriately the administrative control of the valve and cap/plug combination. The licensee stated that it planned to complete this effort by August 1994.

In addition to the ten valves identified in Unresolved Item 298/9403-01, additional manual vents, drains and test valves were capped or plugged in accordance with Maintenance Work Request 94-2978 and its supplement 94-3116. This was discussed as a part of the design changes in Section 2.1.1 of this report.

In accordance with draft General Design Criteria, Criterion 53, July 1967, as stated in Appendix F to the USAR, "[a]ll lines which penetrate the primary containment and which communicate with the reactor vessel or the primary containment free space [were] provided with at least two isolation valves (or equivalent) in series."

The ten single isolation valves without a second barrier were identified as Example 2 of the apparent violation of 10 CFR Part 50, Appendix B, Criterion III, identified in Section 2.1.4 (298/9414-01).

ATTACHMENT

## 1 PERSONS CONTACTED

### 1.1 Licensee Personnel

*R. Gardner, Plant Manager
*R. Godley, Manager, Nuclear Licensing and Safety
*G. Horn, Vice President, Nuclear
*S. Jobe, Acting Senior Nuclear Division Manager, Safety Assessment
*J. Lynch, Manager, Engineering
*E. Mace, Senior Manager, Site Support
*J. Mueller, Site Manager
*J. Sayer, Technical Assistant to Plant Manager
*R. Wilbur, Division Manager
*V. Wolstenholm, Division Manager, Quality Assurance

### 1.2 Other Personnel

*H. Berchert, Director, Division of Radiological Health, State of Nebraska
*J. Parker, Midwest Power
*R. Stoddard, Lincoln Electric System
*W. Turnbull, Midwest Power

### 1.3 NRC Personnel

*A. Beach, Director, Division of Reactor Projects
*L. Callan, Regional Administrator, Region IV
*P. Goldberg, Reactor Inspector, Engineering Branch
*C. Hackney, State Liaison Officer
*P. Harrell, Chief, Reactor Projects Branch C
*R. Kopriva, Senior Resident Inspector
*W. Walker, Resident Inspector

In addition to the personnel listed above, the inspectors contacted other personnel during this inspection period.

* Denotes personnel that attended the exit meeting on August 12, 1994.

## 2 EXIT MEETING

An exit meeting was conducted on August 12, 1994. During this meeting, the scope and findings of the inspection were reviewed. The licensee acknowledged the inspection findings documented in this report. The licensee did not identify as proprietary any information provided to, or reviewed by, the inspectors.

SITE: CNS  COOPER STATION

| IT BR | REPORT NO (IFS NBR*) | SEQ NO | ITEM TYPE | EER/P.I/ EA-NUM | SEVERITY SUPLMNT | SALP AREA | REPORT TRANSMTL | STS | CREATE DATE | CLOSEOUT PRJ/ACT* | CLSOUT ORG NO | CLSOUT EMP | UPDATING INSPECTION REPORTS |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| S | 93-028 | 1 | EEI | 94-00 | | | 01/06/94 | C | 01/18/94 | 02/03/94 | 4204 | RKK | |
| TITLE: | INOPERABLE DIESEL GENERATOR | | | | | | | | | | | | |
| S | 93-028 | 2 | EEI | 94-00 | 1 | | 01/06/94 | O | 01/18/94 | | 4240 | RKK | |
| TITLE: | INADEQUATE PROCEDURES | | | | | | | | | | | | |
| S | 93-028 | 3 | EEI | 94-00 | 1 | | 01/06/94 | O | 01/18/94 | | 4240 | RKK | |
| TITLE: | FAILURE TO FOLLOW PROCEDURES | | | | | | | | | | | | |
| S | 93-028 | 4 | EEI | 94-00 | 1 | | 01/06/94 | O | 01/18/94 | | 4240 | RKK | |
| TITLE: | INADEQUATE CORRECTIVE ACTION | | | | | | | | | | | | |
| S | 93-028 | 5 | EEI | 94-00 | 1 | | 01/06/94 | O | 01/18/94 | | 4270 | RKK | |
| TITLE: | UNTIMELY DECLARATION OF UNUSUAL EVENT | | | | | | | | | | | | |

TOTAL OPEN ITEMS ======= ========> 4
TOTAL OPEN REPORT SEQUENCES =====> 4

*IF ITEM IS OPEN, PROJECTED CLOSEOUT DATE IS REPORTED
IF ITEM IS CLOSED, ACTUAL CLOSEOUT DATE IS REPORTED

SITE: CNS COOPER STATION

| IT BR | REPORT NO (IFS NBR*) | SEQ NO | ITEM TYPE | LER/P21/ EA-NUM | SEVERITY LEVEL/ ELEMENT | SALP AREA | REPORT TRANSMIT | STS | CREATE DATE | CLOSEOUT PRJ/ACT* | CLSOUT ORG NO | CLSOUT EMP | UPDATING INSPECTION REPORTS |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 15 | 92-004 | 1 | VIO | | 1 | SAQV | 0 05/21/92 | C | 07/30/92 | 07/24/92 | 4204 | RKK | |
| TITLE: | 250-VOLT DC BATTERY CELL VOLTAGE BELOW 2.13 VOLTS | | | | | | | | | | | | |
| 15 | 92-004 | 2 | VIO | | 1 | SAQV | 0 05/21/92 | C | 07/30/92 | 07/24/92 | 4204 | RKK | |
| TITLE: | FAILURE TO FOLLOW A PROCEDURE | | | | | | | | | | | | |
| 15 | 92-006 | 2 | VIO | | 4 | RADCON | 05/15/92 | C | 08/12/92 | 09/29/92 | 4304 | RKK | |
| TITLE: | FAILURE TO COMPLY WITH THE REQUIREMENTS OF SWP | | | | | | | | | | | | |
| 15 | 92-009 | 1 | VIO | | 1 | ETS-O | 08/01/92 | C | 09/15/92 | 05/31/94 | 4600 | JPQ | 94-009 |
| TITLE: | PROCEDURE DID NOT REQUIRE INDEPENDENT VERIFICATION | | | | | | | | | | | | |
| 15 | 92-011 | 1 | VIO | | 1 | SAQV | 0 08/28/92 | C | 09/15/92 | 06/29/93 | 4613 | TMG | 92-011 |
| TITLE: | PROCEDURE DEFICIENCIES WERE NOT CORRECTED BY THE LICENS | | | | | | | | | | | | |
| 15 | 92-015 | 1 | VIO | | 8 | EP | 09/29/92 | C | 10/15/92 | 02/05/93 | 4204 | RKK | |
| TITLE: | FAILURE TO COMPLY WITH NOUE EP TIME LIMIT | | | | | | | | | | | | |
| 15 | 92-019 | 3 | VIO | | 1 | OTHR | 0 11/03/92 | C | 11/30/92 | 04/29/94 | 4240 | RKK | |
| TITLE: | TEMPORARY STARTUP STRAINERS LEFT IN CORE SPRAY SYSTEM | | | | | | | | | | | | |
| 15 | 92-020 | 1 | VIO | | 3 | SEC | 11/04/92 | C | 11/12/92 | 03/18/93 | 4304 | | |
| TITLE: | FAILURE TO CHANGE LOCKS | | | | | | | | | | | | |
| 15 | 92-022 | 1 | VIO | | 1 | SAQV | 0 12/14/92 | C | 12/18/92 | 03/05/93 | 4204 | RKK | |
| TITLE: | FAILURE TO CONTROL FLAMMABLE COMBUSTIBLE MATERIAL | | | | | | | | | | | | |
| 15 | 92-022 | 2 | VIO | | 3 | SEC | 12/14/92 | C | 12/18/92 | 03/05/93 | 4204 | RKK | |
| TITLE: | FAILURE TO USE PROPER ESCORT PROCEDURES | | | | | | | | | | | | |
| 15 | 93-003 | 1 | VIO | | 8 | EP | 02/04/93 | C | 02/11/93 | 04/22/94 | 4340 | DSI | |
| TITLE: | FAILURE TO CONDUCT REQD TESTS OF PAGERS OF EMERG RESP | | | | | | | | | | | | |
| 15 | 93-003 | 2 | VIO | | 1/8 | EP | 02/04/93 | C | 02/11/93 | 04/22/94 | 4340 | DSI | |
| TITLE: | FAILURE TO CONDUCT DRILL CRITIQUE AND FOLLOWUP | | | | | | | | | | | | |
| 15 | 93-005 | 1 | VIO | | 1/1 | SAQV-D | 02/26/93 | O | 03/03/93 | 03/23/93 | 4610 | HHB | |
| TITLE: | FAILURE TO ASSIGN CONTINUOUS FIRE WATCH | | | | | | | | | | | | |
| 15 | 93-006 | 1 | VIO | 93-030 | 1/7 | | 03/30/93 | O | 04/14/93 | 04/29/93 | 4240 | RKK | |
| TITLE: | FAILURE TO COMPLY WITH 10 CFR 50.9 REQUIREMENT | | | | | | | | | | | | |
| 15 | 93-006 | 2 | VIO | 93-030 | 1/1 | | 03/30/93 | O | 04/14/93 | 04/29/93 | 4240 | RKK | |
| TITLE: | FAILURE TO COMPLY WITH 10 CFR PART 50 REQUIREMENTS | | | | | | | | | | | | |
| 15 | 93-008 | 4 | VIO | | 1/1 | | 10/08/93 | O | 10/26/93 | | 4610 | MOR | |
| TITLE: | FAILURE TO PROVIDE SAFETY RELATED MOV VALVE TESTING | | | | | | | | | | | | |

SITE: CNS COOPER STATION

| # | REPORT NO (IFS NBR*) | SEQ NO | ITEM TYPE | LER/P21/TA-NUM | SEVERITY UPLMNT | SALP AREA | REPORT TRANSMTL | STS | CREATE DATE | CLOSEOUT PRJ/ACT* | CLSOUT ORG NO | CLSOUT EMP | UPDATING INSPECTION REPORTS |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | 93-008 | 5 | VIO | | 4/1 | | 10/08/93 | O | 10/26/93 | | 4610 | MBR | |
| TITLE: | FAILURE TO PERFORM MOV TEST, OUTSIDE CALIBRATION RANGE | | | | | | | | | | | | |
| | 93-008 | 6 | VIO | | 4/1 | | 10/08/93 | O | 10/26/93 | | 4610 | MBR | |
| TITLE: | 29 MOV VALVES IMPROPERLY TESTED | | | | | | | | | | | | |
| | 93-008 | 7 | VIO | | 4/1 | | 10/08/93 | O | 10/26/93 | | 4610 | MBR | |
| TITLE: | VALVE CS-MOV-05A NOT EVALUATED | | | | | | | | | | | | |
| | 93-017 | 1 | VIO | 9 137 | 3/1 | | 06/17/93 | O | 10/21/93 | 10/12/93 | 4240 | RKK | |
| TITLE: | PROBLEM RESOLUTION AND LICENSEE CORRECTIVE ACTION-01013 | | | | | | | | | | | | |
| 5 | 93-017 | 2 | VIO | 9 137 | 3/1 | | 06/17/93 | O | 10/21/93 | | 4240 | RKK | |
| TITLE: | HISTORIC SECONDARY CONTAINMENT PERFORMANCE (01023) | | | | | | | | | | | | |
| 5 | 93-017 | 3 | VIO | 9 137 | 3/1 | | 06/17/93 | O | 10/21/93 | 10/12/93 | 4240 | RKK | |
| TITLE: | LEAK RATE TESTING OF ANALYZERS (01033) | | | | | | | | | | | | |
| 5 | 93-017 | 4 | VIO | 9 137 | 3/1 | | 06/17/93 | O | 10/21/93 | 10/12/93 | 4240 | RKK | |
| TITLE: | OPERABILITY OF HEAT TRACING IN ANALYZER CABINETS-01043 | | | | | | | | | | | | |
| 5 | 93-017 | 5 | VIO | 9 137 | 3/1 | | 06/17/93 | O | 10/21/93 | 10/12/93 | 4240 | RKK | |
| TITLE: | REVERSE DIRECTION 1ST CONT ISOLATION VALVES (01053) | | | | | | | | | | | | |
| 5 | 93-017 | 6 | VIO | 9 137 | 3/1 | | 06/17/93 | O | 10/21/93 | 10/12/93 | 4240 | RKK | |
| TITLE: | COMPLIANCE WITH 10 CFR 50.55A(G)(I) (01063) | | | | | | | | | | | | |
| 5 | 93-018 | 1 | VIO | | 4/1 | | 06/09/93 | O | 07/21/93 | | 4610 | RKK | |
| TITLE: | INADEQUATE CORRECTIVE ACTION OF RMV RM 1 PROBLEMS | | | | | | | | | | | | |
| 5 | 93-018 | 2 | VIO | | 4/1 | | 06/09/93 | O | 07/21/93 | | 4610 | RKK | |
| TITLE: | DESIGN CHANGE INVOLVED UNREVIEWED SAFETY ISSUE | | | | | | | | | | | | |
| 5 | 93-019 | 1 | VIO | | 4/1 | | 07/06/93 | C | 09/22/93 | 05/31/94 | 4240 | RKK | |
| TITLE: | HIGH ELECTROLYTE LEVELS IN 250-VOLT BATTERIES | | | | | | | | | | | | |
| | 93-019 | 2 | VIO | | 4/1 | OPS | 07/06/93 | O | 09/22/93 | 12/31/93 | 4240 | RKK | |
| TITLE: | UNABLE TO WITHDRAW CONTROL ROD | | | | | | | | | | | | |
| | 93-020 | 1 | VIO | | 1/8 | EP | 06/29/93 | C | 08/24/93 | 12/20/93 | 4600 | JFQ | |
| TITLE: | INAPPROPRIATE EMER. PLAN IMPLEMENTING PROCEDURE | | | | | | | | | | | | |
| 5 | 93-021 | 1 | VIO | | 1/3 | | 07/14/93 | C | 07/29/93 | 08/08/94 | 4340 | TFB | |
| TITLE: | FAILURE TO INFORM INDIVIDUALS | | | | | | | | | | | | |
| 5 | 93-022 | 1 | VIO | | 1/1 | OPS | 08/20/93 | O | 09/09/93 | 09/20/93 | 4240 | WEW | |
| TITLE: | FAILURE TO PREVENT THROUGH WALL LEAK | | | | | | | | | | | | |

SITE: LW  COOK STATION

| RESP UNIT | REPORT (IFS NEXT) NO | ITEM TYPE | EER/P21/EA-NUM | SEVERITY SUPLMNT | SALP AREA | REPORT TRANSMTL | STS | CREATE DATE | CLOSEOUT PRJ/ACT* | CLSOUT ORG NO | CLSOUT EMP | UPDATING INSPECTION REPORTS |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | 93-025 | VIO | | 4/1 | OPS | 11/23/93 | C | 12/21/93 | 04/29/94 | 4240 | RKK | |
| TITLE: | INADEQUATE ALARM RESPONSE PROCEDURE | | | | | | | | | | | |
| | 93-025 | VIO | | 4/1 | MS | 11/23/93 | O | 12/21/93 | | 4240 | RKK | |
| TITLE: | FAILURE TO INITIATE PROMPT CORRECTIVE ACTION | | | | | | | | | | | |
| | 93-026 | VIO | | 4/1 | OPS | 12/29/93 | O | 01/18/94 | | 4240 | WBW | |
| TITLE: | FAILURE TO MAINTAIN POSITIVE PRESSURE IN CR TEST | | | | | | | | | | | |
| | 93-0 | VIO | | 4/1 | MS | 02/03/94 | O | 02/15/94 | | 4240 | RKK | |
| TITLE: | FAILURE TO ANALYZE LUBE OIL SAMPLES | | | | | | | | | | | |
| | 93-202 | VIO | 93-018 | 4/1 | OPS | 01/03/94 | O | 06/21/94 | | 4610 | JVW | |
| TITLE: | URI #7 & #15 CHANGED TO VIO 93202-01 | | | | | | | | | | | |
| | 93-202 | VIO | 93-018 | 4/1 | OPS | 01/03/94 | O | 06/21/94 | | 4240 | RKK | |
| TITLE: | URI 93202-08 CHANGED TO VIO 93202-02 | | | | | | | | | | | |
| | 93-202 | VIO | 93-018 | 4/1 | MAINT | 01/03/94 | O | 06/21/94 | | 4240 | RKK | |
| TITLE: | URI 93202-09, 12 CHANGED TO VIO 93202-03 | | | | | | | | | | | |
| | 93-202 | VIO | 93-018 | 4/1 | OPS | 01/03/94 | O | 06/21/94 | | 4240 | RKK | |
| TITLE: | URI 93202-10 CHANGED TO VIO 93202-04 | | | | | | | | | | | |
| | 93-202 | VIO | 93-018 | 4/1 | PLTSUP | 01/03/94 | O | 06/21/94 | | 4240 | RKK | |
| TITLE: | URI 93202-10 CHANGED TO VIO 93202-05 | | | | | | | | | | | |
| | 93-202 | VIO | 93-018 | 4/1 | EIS.U | 01/03/94 | O | 06/21/94 | | 4610 | JVW | |
| TITLE: | URI 93202-11 CHANGED TO VIO 93202-06 | | | | | | | | | | | |
| | 93-202 | VIO | 93-018 | 4/1 | PLTSUP | 01/03/94 | O | 06/21/94 | | 4610 | JVW | |
| TITLE: | URI 93202-13 CHANGED TO VIO 93202-07 | | | | | | | | | | | |
| | 93-202 | VIO | 93-018 | 4/1 | EIS.U | 01/03/94 | O | 06/21/94 | | 4610 | JVW | |
| TITLE: | URI 93202-14 CHANGED TO VIO 93202-08 | | | | | | | | | | | |
| NS | 94-00 | VIO | | 4/1 | | 04/29/94 | O | 05/19/94 | | 4240 | RKK | |
| TITLE: | FAILURE TO CONTROL POSITION OF CONTAINMENT VALVES | | | | | | | | | | | |
| NS | 94-004 | VIO | | 4/1 | PLTSUP | 06/06/94 | O | 06/21/94 | | 4610 | | |
| TITLE: | FAILURE TO FOLLOW TEMP CHANGE PROCEDURE | | | | | | | | | | | |
| NS | 94-004 | VIO | | 4/1 | EIS.U | 06/06/94 | O | 06/23/94 | | 4610 | CJP | |
| TITLE: | FAILURE TO ASSURE CHANGE IN SW TEMP DURING FLOOD | | | | | | | | | | | |
| NS | 94-013 | VIO | | 4/1 | SEC | 07/29/94 | O | 08/17/94 | | 4240 | RKK | |
| TITLE: | ACCESS REQUIREMENTS NOT APPROPRIATELY FOLLOWED | | | | | | | | | | | |

SITE: IN COOPER STATION

| REPORT NO (IFS NBR*) | SEQ NO | ITEM TYPE | LER/P21/ EA-NUM | SEVERITY SUPLMNT | SALP AREA | REPORT TRANSMTL | STS | CREATE DATE | CLOSEOUT PRJ/ACT* | CLSOUT ORG NO | CLOUT EMP | UPDATING INSPECTION REPORTS |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 94-01 | | VIO | | 4/1 | MAINT | 07/29/94 | O | 08/17/94 | | 4240 | RH | |
| TITLE: FAILURE TO IMPLEMENT METAL-TO-METAL CONTACT | | | | | | | | | | | | |
| 94-01 | | VIO | | 4/1 | OPS | 07/29/94 | O | 08/17/94 | | 4240 | RH | |
| TITLE: REACTOR IN OPERATION AT GREATER THAN 2381 MEGAWATTS | | | | | | | | | | | | |
| 94-02 | 1 | VIO | | 4/3 | SEC | 08/08/84 | O | 08/17/94 | | 4370 | TKD | |
| TITLE: FAILURE TO PRACTICE CONTINUOUS BEHAVIORAL OBSERVATION | | | | | | | | | | | | |

TOTAL OPEN ITEMS ===> 35
TOTAL OPEN REPORT SEQUENCES ===> 35

*IF ITEM IS OPEN, PROJECTED CLOSEOUT DATE IS REPORTED
IF ITEM IS CLOSED, ACTUAL CLOSEOUT DATE IS REPORTED

SITE: COOPER STATION

| IE BR | REPORT NO (IFS NBR*) | SEQ NO | ITEM TYPE | TEH/P21/ EA-NUM | SEVERITY SUPLMNT | SALP AREA | REPORT TRANSMIT | STS | CREATE DATE | CLOSEOUT PRJ/ACT* | CLSOUT ORG NO | CLSOUT EMP | UPDATING INSPECTION REPORTS |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 4 | 94-06 | 1 | NEV | | | | 03/11/94 | C | 04/26/94 | 03/14/94 | 4610 | WHM | |

TITLE: FAILURE TO REVIEW PROPOSED PLANT MODIFICATIONS

TAL OPEN ITEMS ----------> 0
TAL OPEN REPORT SEQUENCES ---> 0

*IF ITEM IS OPEN, PROJECTED CLOSEOUT DATE IS REPORTED
IF ITEM IS CLOSED, ACTUAL CLOSEOUT DATE IS REPORTED

TS

## LIMITING CONDITIONS FOR OPERATION

### 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.

## SURVEILLANCE REQUIREMENTS

### 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.

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 a closed condition.

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

$\frac{L_{tm}}{L_{am}} \leq 1.0$

$$L_t = 0.635 \left(\frac{P_t}{P_a}\right)^{1/2}$$

| 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

LIMITING CONDITIONS FOR OPERATION

3.7.A (Cont'd)

SURVEILLANCE REQUIREMENTS

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.

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENTS |
|---|---|

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. Pressure Suppression Chamber - Reactor Building Vacuum Breakers | 3. Pressure Suppression Chamber - Reactor Building Vacuum Breakers |
| 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. Drywell-Pressure Suppression Chamber Vacuum Breakers | 4. Drywell-Pressure Suppression Chamber Vacuum Breakers |
| 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 drywell 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 operation of the position indication switch. The differential pressure setpoint 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. Oxygen Concentration | 5. Oxygen Concentration |
| 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 | SURVEILLANCE REQUIREMENT |
|---|---|
| 3.7.A (cont'd.) | 4.7.A (cont'd.) |
| 6. Low-Low Set Relief Function | 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. | a. The low-low set safety/relief valves shall be tested and calibrated as specified in Table 4.2.B. |
| 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.<br><br>NBI-PS-51A Open Low Valve<br>1015 ± 20 psig (Increasing)<br><br>NBI-PS-51B Close Low Valve<br>875 ± 20 psig (Decreasing)<br><br>NBI-PS-51C Open High Valve<br>1025 ± 20 psig (Increasing)<br><br>NBI-PS-51D Close High Valve<br>875 ± 20 psig (Decreasing) | |
| B. Standby Gas Treatment System | 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. | 1. At least once per operating cycle the following conditions shall be demonstrated.<br><br>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.<br><br>b. Inlet heater input is capable of reducing R.H. from 100 to 70% R.H. |
| 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. | 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. |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 3.7.B (cont'd) | 4.7.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. Secondary Containment

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. Secondary Containment

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.

LIMITING CONDITIONS FOR OPERATION

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.

SURVEILLANCE REQUIREMENTS

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 | 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<br>PC-246AV, PC-231MV | | 2 | 15 | C | SC |
| Primary Containment & $N_2$ Supply<br>PC-238AV, PC-232MV | | 2 | 15 | C | SC |
| Suppression Chamber Purge & Vent<br>PC-230MV Bypass (PC-305MV) | | 1 | 40 | C | SC(4) |
| Primary Containment Purge & Vent<br>PC-231MV Bypass (PC-306MV) | | 1 | 40 | C | SC(4) |
| Dilution Supply<br>PC-1303MV, PC-1304MV<br>PC-1305MV, PC-1306MV | | 2<br>2 | 15<br>15 | C<br>C | SC<br>SC |
| Dilution Supply<br>PC-1301MV, PC-1302MV<br>PC-1311MV, PC-1312MV | | 2<br>2 | 15<br>15 | O<br>O | GC<br>GC |
| Suppression Chamber Purge and Vent Exhaust<br>PC-1308MV | | 1 | 15 | C | SC |
| Primary Containment Purge and Vent Exhaust<br>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.

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 suggested in 10CFR100 in the event of a break in the primary system piping. Thus, containment 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 requirement 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 the 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 criteria 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

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

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.

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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 hy[illegible]gen 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 <u>BASES</u>(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 <u>LOW-LOW SET RELIEF FUNCTION</u>

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 ≤ 1040 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 <u>STANDBY GAS TREATMENT SYSTEM AND SECONDARY CONTAINMENT</u>

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

Initiating reactor building isolation and operation of the Standby Gas Treatment System to 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 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 sam 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 $1.1 \times 10^{-7}$ 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.

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

where,

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

(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 | ASME - Section III, Paragraph N-627 or ASME B&PV Code 1965 w/ Addenda thru Winter 1967 |

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<br>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.<br>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 Forgings -

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 Submittals

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


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## 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.

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.

rocess control systems and overrides all other controls to initiate the necessary safety 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.

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.

cipated 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 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.

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.