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Nuclear Management Company, LLC
Prairie Island Nuclear Generating Plant
1717 Wakonade Dr. East • Welch MN 55089

September 12, 2000 10 CFR 50.71(e)

U S Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Submittal of Revision No. 22 to the
Updated Safety Analysis Report (USAR)

Pursuant to 10 CFR 50.71(e) we are submitting one original and 10 copies of Revision No. 22 to the Updated Safety Analysis Report (USAR) for the Prairie Island Nuclear Generating Plant. This revision brings the USAR up-to-date as of August 1, 2000.

Attachment 1 contains descriptions and summaries of safety evaluations for changes, tests, and experiments made under the provisions of 10 CFR 50.59 during the period since the last update. Attachment 1 also contains discussion of a change to a regulatory commitment made within our Regulatory Commitment Change Process.

Attachment 2 contains the USAR page changes and instructions for entering the pages.

In this letter we have made no new Nuclear Regulatory Commission commitments.

Rec'd at NRC DCD UPS. on 10/10/00 by mail.

A053 1/14

I certify that the information presented herein accurately presents changes made since the last updating submittal of the Prairie Island USAR.

Please contact Arne Hunstad (651-388-1121, Ext. 4152) if you have any questions related to this letter.

Joel P. Sorensen

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Site General Manager
Prairie Island Nuclear Generating Plant

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Attachments: 1. Safety Evaluation Summaries
2. USAR page changes

Mfst Num: 2000 - 0643 Date : 09/29/00
FROM : Bruce Loesch/Mary Gadient Loc : Prairie Island
TO : US NRC DOC CONTROL DESK
Copy Num: 486 Holder : US NRC DOC CONTROL DESK
SUBJECT : Revisions to CONTROLLED DOCUMENTS

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Procedure #	Rev	Title
Revisions:		
USAR	22	UNDATED SAFETY ANALYSIS REPORT

UPDATING INSTRUCTIONS

Place this material in your Prairie Island Controlled Manual or File. Remove revised or cancelled material and recycle it. Sign and date this letter in the space provided below within ten working days and return to Bruce Loesch or Mary Gadient, Prairie Island Nuclear Plant, 1717 Wakonade Drive E., Welch, MN 55089.
Contact Bruce Loesch (ext 4664) or Mary Gadient (ext 4478) if you have any questions.

Received the material stated above and complied with the updating instructions

____ Date ______________

ATTACHMENT 1

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
REPORT OF CHANGES, TESTS AND EXPERIMENTS – MARCH 2000

Below are a brief description and a summary of the safety evaluation for each of those changes, tests, and experiments which were carried out without prior NRC approval, pursuant to the requirements of 10 CFR Part 50, Section 50.59(b).

Modification 90L215 – Replace EOF Communications Link

Description of Change

This modification replaced components in the communications link between plant computers and terminals in the Emergency Operations Facility (EOF). Spare parts no longer exist for some of the equipment.

Summary of Safety Evaluation

The Safety Evaluation concluded that this modification would not result in an unreviewed safety question. This modification does not affect or change Plant Technical Specifications or USAR.

Modification 96EM01 – Containment Hydrogen Monitor Processor Replacement

Description of Change

This modification replaced the microprocessor system with a new one which will utilize commercial grade hardware and software that is qualified for safety-related use. The existing microprocessor is no longer manufactured and spare parts are not available. The chassis of the new system is designed to be installed in the same rack as the existing system.

Summary of Safety Evaluation

The Safety Evaluation concluded that this modification would not result in an unreviewed safety question. This modification does not affect or change Plant Technical Specifications or USAR.

Modification 96FP02 – D5/D6 Fire Alarm Control Panel Relocation

Description of Change

This modification relocated FACP Panel 70466 to the control room, which is a controlled environment, to eliminate problems caused by high ambient temperatures. In addition, a new cabinet and upgraded electronics replaced the existing fire alarm control panel.

Summary of Safety Evaluation

This modification improved the operability of the system by relocating the panel into a controlled temperature environment. Furthermore, the new electronics more suitably adapts itself to the newer style of fire/smoke detector heads.

The safety evaluation concluded that this modification will not result in an unreviewed safety question and does not affect or change Plant Technical Specifications. This modification affects only the non-safety related fire protection system.

Modification 98CC01 – CC Cross-Leakage Modification

Description of Change

This modification eliminates Component Cooling (CC) cross leakage attributable to Spent Fuel Pool Cooling. This modification split the CC supply and return headers to the Spent Fuel Pool Heat Exchangers. The splitting was accomplished by installing new CC supply and return lines to the 122 Spent Fuel Pool Heat Exchanger. Also, the CC supply motor valves to the SFP Heat Exchangers were replaced.

Summary of Safety Evaluation

The Safety Evaluation concluded that the modification and associated procedure changes would not have an adverse impact on the safe operation of the plant and the associated licensing basis, nor would it present any new accident scenarios that would need to be analyzed.

Modification 98RV06 – Reactor Vessel Head Penetrations Seal Welds

Description of Change

This modification governed repair and preventive weld overlays of canopy seal welds on reactor vessel head penetrations. This modification affected only the leakage barrier provided by the lower canopy seal weld. The pressure retaining (structural) integrity of the joint between the head penetration and CRDM Latch Housing is provided by the threaded joint which joins the two components.

Summary of Safety Evaluation

Failure modes of the CRDM housings, Reactor Coolant pressure boundary and RCC Assembly Ejection/Stuck Rod are discussed in the USAR. Weld buildup of the lower canopy seal does not affect the status of the F/L CRDMs or RCCAs. The weld buildup does not affect the structural integrity of the joint provided by the threads (i.e. potential for CRDM housing failure and RCPB failure is not affected). Measures taken during welding ensured the displacement of the rod travel housing did not affect RCCA travel. The seismic plates located at the top of the rod travel housings position the housing in the proper position. Finally, rod drop testing during unit startup verified proper operation of the rod.

Modification 99FH03 – Unit 2 Cycle 20 Reload

Description of Change

This modification will replace depleted Unit 2 fuel assemblies with a fresh reload of Westinghouse VANTAGE+ fuel assemblies allowing another cycle of power operation. All applicable documents and analyses have been reviewed and performed for Unit 2 Cycle 20 assuring safe operations. The core design is verified by the performance of post-refueling startup physics testing.

The Unit 2 Cycle 20 reload was developed by NSP Nuclear Analysis & Design (NSPNAD) using approved methodology addressed in NSPNAD-8101-A, Qualifications of Reactor Physics Methods for Application to PI Units. The Unit 2 Cycle 20 core safety analysis was performed by NSPNAD using approved methodologies.

Reload modification 99FH03 was revised due to problems encountered while attempting to replace top nozzles on Q region fuel assemblies scheduled to be used in Unit 2 Cycle 20. The core was redesigned using S region fuel assemblies instead of Q region fuel assemblies. These S region fuel assemblies have been renozzled with the replacement top nozzles for the Q region fuel assemblies and have been renumbered as Q##R fuel assemblies. The 2 core designs are so similar that none of the boron concentrations or physics testing parameters have changed.

Summary of Safety Evaluation

The following safety concerns and their resolutions are the basis for demonstrating that the Unit 2 Cycle 20 reload modification does not represent an unreviewed safety question.

A. Thermal Hydraulic Design Analysis
B. Transient and Accident Analysis
C. Uncontrolled Boron Dilution
D. Main Steam Line Break/Containment Response Analysis

E. LOCA-ECCS Analysis
F. Rod Ejection Analysis
H. Refueling Shutdown Margin
I. Heatup/Cooldown Curves - Reactor Vessel Radiation Surveillance Program
J. Fuel Rod Design Performance
M. Core Exposure Limits/Off-site Dose Calculations
N. Peak Linear Heat Generation Rate
O. Fuel Assembly Design Change
P. Startup and Operations
Q. Validity of Safety Evaluation

All results were acceptable and are presented in NSPNAD-00004, Rev. 1, Prairie Island Unit 2 Cycle 20 Final Reload Design Report (Reload Safety Evaluation) and USAR Update.

Modification 99FH04 – Unit 2 Manipulator Crane Upgrade

Description of Change

The modification provided the following:

Upgrade the manipulator crane load system.
Relocate the remote Z-Z axis readout.
Upgrade the Toshiba Inverter Drive system.

The Dillon load system on the manipulator crane is original equipment. The load cell and readout are out-dated and parts replacement are no longer supported by the manufacturer. If the Dillon load cell or readout were to fail, neither part could be replaced without doing a modification. Since the manipulator crane is not operable without the load system, refueling operations would be delayed until the system could be replaced. This project upgraded the Dillon Load System with a new Sensotec Load system that is recommended by the manipulator crane vendor (Raytheon/Stearns-Roger).

The second issue is the manipulator crane inching circuit. The inching circuit is used to fine position the manipulator crane bridge and trolley in the core and transfer basket regions. This circuit has been troublesome in the past and has only been operational approximately 50% of the time. This is an operator distraction and also causes delays during the refueling process. This project upgraded the Toshiba Inverter Drive System to provide more reliable fine positioning of the manipulator crane.

The final issue pertains to the Z-Z axis digital readout. The Z-Z axis readout informs the crane operator of the position of the inner mast and grapple and also a fuel assembly when attached. The readout is currently located on the Motor Control Center (MCC). The operator is forced to turn his head away from the Control Console in order to obtain and verify the inner mast position. The Z-Z axis readout will be relocated such that it is in the same field of view as the Control Console during normal operation of the manipulator crane.

Summary of Safety Evaluation

The Safety Evaluation concluded that the modification and associated procedure changes would not have an adverse impact on the safe operation of the plant and the associated licensing basis, nor would it present any new accident scenarios that would need to be analyzed.

Modification 99SF02 – Replace Spent Fuel Cooling System Heat Exchanger

Description of Change

This modification replaced the existing 122 Spent Fuel Pool Heat Exchanger with a heat exchanger which has a higher heat removal capability. This modification was performed in conjunction with the CC Cross Leakage Modification 98CC01.

Summary of Safety Evaluation

The Safety Evaluation concluded that the modification would not cause any system to be operated outside of its design basis. Personnel and environmental safety would be controlled by following the plant Work Order process and following the Site Safety Manual.

Modification 99SG04 – Steam Generator Tube/Sleeve Sample Removal

Description of Change

This modification removed sleeves and tubes from Unit 1 steam generators as needed in order to meet the voltage based repair criteria.

Summary of Safety Evaluation

Removal of the tube and sleeve samples and plugging the tubesheet holes and tube ends maintains the integrity of the steam generator required by Technical Specifications and the ASME Code. The tube plugs maintain the primary to secondary pressure boundary under normal and postulated accident conditions. The post maintenance leak check provided verification that the steam generator was not returned to service with significant leakage paths due to the installation of the plugs. There are no safety concerns created by the removal of tube and sleeve samples and the installation of the welded tubesheet plugs and mechanical plugs.

Modification 99SI01 – SI Test Line Orifice Installation

Description of Change

In response to NRC Generic Letter 96-06, NSP prepared calculation ENG-ME-299 "Piping Internal Pressurization". This calculation identified the SI Test Line as potentially becoming overpressurized while high temperature conditions (≈268°F) exist inside containment. The line is normally isolated by the four accumulator test control valves, two SI test return check valves, and outside containment manual isolation valves (penetration 35). During an accident inside containment, the water in the line is postulated to heat up to ≈268°F, which causes the pressure in the line to increase, far exceeding the design pressure of the piping system. As a result of this analysis, NCR 19983428 was written; the corrective action suggested was to evaluate and install an appropriate method to prevent line over-pressurization.

Several options were evaluated to prevent the line from becoming overpressurized; it was determined that the installation of an orifice in place of the SI Test Return Check Valve SI-21-2 [2SI-21-2] was the best solution due to its passive design.

Summary of Safety Evaluation

The modification creates no adverse effects on the ability of the Safety Injection System to perform its function to mitigate an accident. The ability to provide containment integrity is also maintained.

Modification 99SI02 – Repower RHR Sump B Suction Valves

Description of Change

During the NRC Fire Protection Functional Inspection at PINGP in 1998, a question was raised concerning the possibility of having the RWST drain into the Containment Sump B due to the spurious opening of series valves MV-32075 and MV-32077 (Unit 1 Train A), MV-32076 and MV-32078 (Unit 1 Train B), MV-32178 and MV-32180 (Unit 2 Train A), or MV-32179 and MV-32181 (Unit 2 Train B). These valves were not on the Appendix R equipment list. Each pair of series valves is powered from the same MCC, and the two valves in each pair are susceptible to spurious operation due to a fire in the ground floor of the Aux Building or in the Control/Relay Room. In addition, manual action to prevent or mitigate a second spurious operation would not be possible for the Aux Building fire because the valves and the MCCs that power them are in the same fire area as the fire causing the spurious operations, thereby preventing entry to perform timely manual actions. This created a situation where there may not be adequate volume of water in the RWST for safe shutdown. Reference Condition Report 19982352 and LER 1-98-15.

When it was identified that these eight valves should be addressed for Appendix R safe shutdown, they were then evaluated for fire induced circuit failure caused valve damage and were identified in Supplement 1 to LER 1-98-10 as being susceptible to damage by fire induced circuit failures as discussed in NRC Information Notice 92-18 "Potential for Loss of Remote Shutdown Capability during a Control Room Fire".

This modification was implemented to address fire induced flow diversion and circuit failures causing valve damage. One valve in each pair (the outside valve) is now powered from an MCC outside of the 695 elevation of the Aux Building. The new power source is the "A" MCCs in the Aux Feed Water Pump Rooms. Also, the order of connection of the control circuit for that valve was changed to eliminate the susceptibility to circuit failure induced damage in fire areas where the valve (or flowpath) is credited.

Summary of Safety Evaluation

The modification does not present an Unreviewed Safety Question because it does not affect the operation of the valves for any design basis accident, malfunction, or event. This design change limits fire damage to equipment and systems required by the Appendix R Safe Shutdown Analysis to achieve and maintain safe shutdown conditions.

Modification 99SI03 – Filters for Nitrogen Supply to Accumulators

Description of Change

This modification installed a filter in the Nitrogen supply line to each accumulator nitrogen supply isolation control valve. This modification is the result of a Nonconformance Report 19983613 to eliminate the galling of CV-31440 [CV-31554].

Summary of Safety Evaluation

The installed components do not perform any Safety Related functions and do not change how the accumulators are pressurized. Therefore, the installed filter does not have any associated safety concerns.

Modification 00HD01 – Remove Feedwater Heater High Level Turbine Trips

Description of Change

Unit 2 had experienced a turbine trip from low power during each of the last two shutdowns for refueling. A review following the trips determined that these were spurious trips caused by the Hi-Hi water level trip on the low pressure feedwater heaters without an actual high water level.

While reviewing the trip, a task force determined from Westinghouse technical manuals that an automatic trip with a water level at the current 90% level setpoint could result in damage to the main turbine. The preferred response would be an operator action at a lower level to drain water, bypass the heater, or manually trip the turbine while the water level remains low. This modification deletes the high-high water level turbine trip from the 1A & 1B, 2A & 2B, and 3A & 3B low pressure feedwater heaters for both Unit 1 and Unit 2. The high-high water level switches are reused as inputs to an annunciator in the main Control Room. The six level switches on each unit are moved from the existing 90% level to approximately 70% level to assure that the level is below the highest tube in the feedwater heater.

Summary of Safety Evaluation

The safety evaluation concludes that this design change is not an Unreviewed Safety Question and does not affect or require a change to the Plant Technical Specifications. A USAR change to Figures showing the Feedwater Heater Hi-Hi- level instruments is required.

Safety Evaluation 137, Addendum 2 – Upgrade Refueling Cavity Seal

Description of Change

This safety evaluation justifies use of an upgraded refueling cavity to reactor vessel seal.

Summary of Safety Evaluation

The upgraded seal is suitable for service for normal and postulated faulted conditions and will not adversely impact core cooling or inventory makeup capability. Severe leakage of the seal could affect personnel exposure, but the effects are dependent on the leakage rate and makeup capabilities. In all cases the upgraded seal will outperform the original seal.

Safety Evaluation 478-AI-01 – USAR Update Appendix I, Miscellaneous Topics

Description of Change

This non-modification safety evaluation revises those portions of Appendix I that are not addressed by Safety Evaluations #478-AI-02 through 478-AI-07.

* Revised the format of the entire appendix to eliminate extensive duplication and combine information into more appropriate topical sections.

* Revised the introductory paragraph to correctly cite the AEC letter and added reference to subsequent clarification letters.

* Corrected the discussion concerning encapsulation sleeve vent area to reflect the as-built conditions.

* Deleted product specific information and invoked the plant's equipment environmental qualification program.

Summary of Safety Evaluation

The proposed changes have no effect on any of the methods, inputs or assumptions used in any accident analysis. Thus, there is no potential increase in consequences or a reduction in margin of safety. The proposed changes do not affect any assumptions or precursors which could lead to any different types of accidents. The proposed changes do not adversely affect the design or operating assumptions used in any accident analyses for any structures, systems or components important to safety. The assumptions regarding component performance are consistent with their design basis. Thus, there is no increase in the probability of an accident or equipment malfunction previously evaluated, nor is there the possibility of creating an accident or equipment malfunction of a different type. Therefore, the proposed changes do not constitute an unreviewed safety question.

Safety Evaluation 478-AI-02 – USAR Update Appendix I, Required Equipment Lists

Description of Change

Revised the required equipment lists for responding to postulated HELB events in the five identified systems.

Replaced Tables I.3-1 through I.7-1 and the lists on Figures I.3-3, I.4-1, I.5-1, I.6-1 and I.7-1 with a new Table I.1.4-1.

Revised the electrical and mechanical components to reflect the modifications and additions made during the station blackout and electrical system upgrade projects.

Added appropriate Reg Guide 1.97, Category 1 & 2 events monitoring instrumentation.

Revised or added various valves based on previously performed modifications.

Added the manual reactor trip, safety injection and MSIV closure control switches.

Summary of Safety Evaluation

The proposed changes have no effect on any of the methods, inputs or assumptions used in any accident analysis. Thus, there is no potential increase in consequences or a reduction in margin of safety. The proposed changes do not affect any assumptions or precursors which could lead to any different types of accidents. The proposed changes do not adversely affect the design or operating assumptions used in any accident analyses for any structures, systems or components important to safety. The assumptions regarding component performance are consistent with their design basis. Thus, there is no increase in the probability of an accident or equipment malfunction previously evaluated, nor is there the possibility of creating an accident or equipment malfunction of a different type. Therefore, the proposed changes do not constitute an unreviewed safety question.

Safety Evaluation 478-AI-03 – USAR Update Appendix I, Pipe Stress and Pipe Whip

Description of Change

This non-modification safety evaluation revises portions (Pipe Stress and Pipe Whip) of Appendix I of the USAR as follows:

- Abbreviated the discussion concerning the content of the Giambusso letters (now historical information) and added a discussion of GL 87-11 break and crack location criteria.
- Deleted the discussion of why breaks were not assumed in the Main Steam safety valve and steam dump headers.
- Created new tables that summarized all identified design basis break and leakage crack locations for the five high energy systems and the Auxiliary Building compartment in which they occur.
- Revised the pipe whip discussion to properly reflect the content of the final version of the NSC Topical Reports.

Summary of Safety Evaluation

The proposed changes have no effect on any of the methods, inputs or assumptions used in any accident analysis. Thus, there is no potential increase in consequences or a reduction in margin of safety. The proposed changes do not affect any assumptions or precursors which could lead to any different types of accidents. The proposed changes do not adversely affect the design or operating assumptions used in any accident analyses for any structures, systems or components important to safety. The assumptions regarding component performance are consistent with their design basis. Thus, there is no increase in the probability of an accident or equipment malfunction previously evaluated, nor is there the possibility of creating an accident or equipment malfunction of a different type. Therefore, the proposed changes do not constitute an unreviewed safety question.

Safety Evaluation 478-AI-04 – USAR Update Appendix I, Compartment Pressure and Temperature

Description of Change

Replaced the discussion of the methodology for determining pressure and temperature transients in the Auxiliary Building as the result of a high energy line break event. The discussion and values cited reflect the results of a completely new analysis, which closely reflect the original analysis results.

Summary of Safety Evaluation

The proposed changes have no effect on any of the methods, inputs or assumptions used in any accident analysis. Thus, there is no potential increase in consequences or a reduction in margin of safety. The proposed changes do not affect any assumptions or precursors which could lead to any different types of accidents. The proposed changes do not adversely affect the design or operating assumptions used in any accident analyses for any structures, systems or components important to safety. The assumptions regarding component performance are consistent with their design basis. Thus, there is no increase in the probability of an accident or equipment malfunction previously evaluated, nor is there the possibility of creating an accident or equipment malfunction of a different type. Therefore, the proposed changes do not constitute an unreviewed safety question.

Safety Evaluation 478-AI-05 – USAR Update Appendix I, Jet Impingement

Description of Change

Replaced the discussion of the methodology for determining jet impingement pressure and temperature versus distance for various high energy line break events. The discussion, values presented and the curves reflect the results of a completely new analysis based on currently accepted industry methodology as contained in ANSI/ANS-58.2-1988.

Summary of Safety Evaluation

The proposed changes have no effect on any of the methods, inputs or assumptions used in any accident analysis. Thus, there is no potential increase in consequences or a reduction in margin of safety. The proposed changes do not affect any assumptions or precursors which could lead to any different types of accidents. The proposed changes do not adversely affect the design or operating assumptions used in any accident analyses for any structures, systems or components important to safety. The assumptions regarding component performance are consistent with their design basis. Thus, there is no increase in the probability of an accident or equipment malfunction previously evaluated, nor is there the possibility of creating an accident or equipment malfunction of a different type. Therefore, the proposed changes do not constitute an unreviewed safety question.

Safety Evaluation 478-AI-06 – USAR Update Appendix I, Steam Exclusion and Ventilation

Description of Change

This non-modification safety evaluation revised the discussions concerning steam exclusion areas and ventilation systems in Appendix I of the USAR as follows:

- Expanded the description of the steam exclusion areas and boundaries to include the floors, walls, doors, penetrations, etc.
- Expanded the discussion concerning ventilation damper leakage testing to include replacement dampers.
- Up-dated the discussion concerning the peak pressure and temperature the shield building seals will experience to reflect the latest compartment GOTHIC analysis.

Summary of Safety Evaluation

The proposed changes have no effect on any of the methods, inputs or assumptions used in any accident analysis. Thus, there is no potential increase in consequences or a reduction in margin of safety. The proposed changes do not affect any assumptions or precursors which could lead to any different types of accidents. The proposed changes do not adversely affect the design or operating assumptions used in any accident analyses for any structures, systems or components important to safety. The assumptions regarding component performance are consistent with their design basis. Thus, there is no increase in the probability of an accident or equipment malfunction previously evaluated, nor is there the possibility of creating an accident or equipment malfunction of a different type. Therefore, the proposed changes do not constitute an unreviewed safety question.

Safety Evaluation 478-AI-07 – USAR Update Appendix I, Operating Procedures

Description of Change

This non-modification safety evaluation revises USAR Appendix I.8 (Emergency Procedures) as follows:

Renamed this section "Operating Procedures" to be consistent with plant terminology.

Added closure of the feedwater containment isolation valves as a means of reducing flow to a faulted steam generator.

Replaced the term "safe" shutdown with "hot" or "cold" shutdown as appropriate.

Revised the discussion concerning plant response to small steam line breaks causing depletion of condenser hotwell inventory.

Revised discussions concerning safety injection to reflect all functions, deleted being initiated by low-low steam generator level, steam flow-feed flow mismatch and coincident pressurizer pressure and level.

Summary of Safety Evaluation

The proposed changes have no effect on any of the methods, inputs or assumptions used in any accident analysis. Thus, there is no potential increase in consequences or a reduction in margin of safety. The proposed changes do not affect any assumptions or precursors which could lead to any different types of accidents. The proposed changes do not adversely affect the design or operating assumptions used in any accident analyses for any structures, systems or components important to safety. The assumptions regarding component performance are consistent with their design basis. Thus, there is no increase in the probability of an accident or equipment malfunction previously evaluated, nor is there the possibility of creating an accident or equipment malfunction of a different type. Therefore, the proposed changes do not constitute an unreviewed safety question.

Safety Evaluation 527-10-01 – USAR Update, Reactor Makeup Controls

Description of Change

The USAR states in several places that the reactor makeup pumps stop upon completion of makeup activities. This is contrary to the control scheme for these pumps, which have no automatic functions. The pumps have a manual start and stop function.

Summary of Safety Evaluation

The reactor makeup system is not connected with the mitigation of any accident. The cause of the dilution accident analyzed in the USAR is operator error; the mitigative action is operator action to stop the dilution.

Safety Evaluation 527-10-03, USAR Update, CVCS Components

Description of Change

This safety evaluation provides the justification for a revision to the USAR to reflect clarification and deletion of portions of these sections. Specific types and forms of resin, as well as specific performance and operational practices were deleted or changed, to allow improved performance and operational flexibility. None of these changes degrade or compromise system performance, but rather more accurately reflect current operational improvements to the system.

Summary of Safety Evaluation

The proposed changes allow the system to function as originally designed. These components of the CVCS system have no effect on the initiation of design based accidents or their severity. Thus, there is no potential increase in consequences or a reduction in the margin of safety. These changes do not affect any assumptions or precursors which could lead to any different types of accidents. These changes do not adversely affect the design or operating assumptions used in any accident or transient analyses for any structures, systems, or components important to safety. Thus, there is no increase in the probability of an accident or equipment malfunction previously evaluated, nor is there the possibility of creating an accident or equipment malfunction of a different type. Therefore, these changes do not constitute an unreviewed safety question.

Safety Evaluation 527-10-04 – USAR Update, Tritium Measurement

Description of Change

This safety evaluation provides the justification for a revision to the USAR to reflect clarification and deletions of portions of these sections. The requirement to base the tritium concentration on humidity measurements is removed. This change does not degrade or compromise system performance or the health and safety of the general public.

Summary of Safety Evaluation

The change proposed allows the calculation of airborne tritium in containment to be completed utilizing a methodology equivalent of the actual tritium concentration as that which is described in the USAR. The containment tritium concentration has no effect on the initiation of design based accidents or their severity. Thus, there is no potential increase in consequences or a reduction in margin of safety. This change does not affect any assumptions or precursors which could lead to any different types of accidents. This change does not adversely affect the design or operating assumptions used in any accident or transient analyses for any structures, systems or components important to safety. Thus, there is no increase in the probability of an accident or equipment malfunction previously evaluated, nor is there the possibility of creating an accident or equipment malfunction of a different type. Therefore, this change does not constitute an unreviewed safety question.

Safety Evaluation 527-10-05 – USAR Update, CVCS Holdup Tanks

Description of Change

The text of Sections 10.2.3,10.2.3.2, and 10.2.3.2.4 were revised to more accurately describe the processing of holdup tank liquids. Customary practice for processing CVCS holdup tank liquids is through filtration and ion exchange. Though the gas stripper/boric acid evaporator packages are piped into the system, they are generally not employed to process the liquid effluent. The USAR descriptions of the effluent processing and components employed have been updated to more closely describe current practice and procedures.

Summary of Safety Evaluation

The consequences of an accident previously evaluated in the SAR, the probability of occurrence of an equipment malfunction or its consequences remain unaffected by these USAR changes. Similarly, these USAR changes will not create the possibility of an accident of a different type now the possibility of a different type of equipment malfunction. The margin of safety as defined in the basis for any Technical Specification remains unchanged by these USAR text changes.

Safety Evaluation 555 – USAR Update, EDG Loading during an SBO Event

Description of Change

Section 8.4 of the USAR was reviewed when calculation ENG-EE-045 for EDG SBO loading was revised as part of the review for SE 549 to support Two Charging Pump Operation. Several changes were made to section 8.4 as a result of SE 549. However, it was found in the course of the review that the USAR summary of the general criteria for this calculation were incorrect. This SE documents the basis for a change to USAR section 8.4 to correct the explanation of EDG loading criteria for a Station Blackout (SBO) event.

Summary of Safety Evaluation

The existing USAR statement "This guidance also specifies that power be available for operating an RHR pump (160 KW) in the non-SBO unit" is a misinterpretation of the NUMARC guidance document and the PI NRC SER statements that the non-SBO unit has the capability for hot shutdown/hot standby forced cooling, cooldown and depressurization as required. The current methodology for calculating EDG capability satisfies this requirement.

Other changes made to the USAR section clarify the actual load requirements for an EDG and the assumptions in the calculations for the SBO and the non-SBO unit.

The changes in the wording of the USAR do not change any system, structure, or component in the plant. These changes add details and make corrections regarding SBO event EDG loading for the SBO and the non-SBO units in the USAR. This information is described in the regulations and implemented in the calculations for Prairie Island. These changes are consistent with the regulations, regulatory guidance, the NRC SER, NSP commitments, and the plant analysis and calculations which document plant compliance with the regulations. Therefore, they do not alter the plant's response to events or its design basis.

Safety Evaluation 557 – Recirculation – Passive Failure

Description of Change

To preclude concerns for water draining into the RHR pits, boot seals were previously installed on the lines (in the Containment Spray Room) from the containment sump to the RHR pumps. These boot seals also prevent unwanted debris from making its way into the RHR pits. The USAR previously credited this flow path to allow the operator to detect a passive failure during post-LOCA recirculation operation. With the boot seals installed, this flow path is no longer available. This safety evaluation looks at this discrepancy between the configuration and the description and concludes that the configuration with the boot seals installed is acceptable. The USAR will be revised accordingly. This discrepancy was identified during the USAR review project.

Summary of Safety Evaluation

These changes have no effect on any of the methods, inputs, or assumptions used in any analysis. As the intent is to maintain the RHR flow during long term post-accident mitigation, there is reasonable assurance of adequate core cooling. Thus, there is no potential increase in consequences or a reduction in the margin of safety. These changes do not affect any assumptions or precursors which could lead to any different types of accidents. These changes do not adversely affect the design or operating assumptions used in any accident or transient analyses for any structures, systems, or components important to safety. Thus, there is no increase in the probability of an accident or equipment malfunction previously evaluated, nor is there the possibility of creating an accident or equipment malfunction of a different type. Therefore, these changes do not constitute an unreviewed safety question.

Safety Evaluation 558 – Containment Vacuum Relief System

Description of Change

This safety evaluation provides the justification for a revision to the USAR to reflect a new calculation of the capability of the Containment Vacuum Relief System (vacuum breakers). The new calculation employs the same methodology as the previous analysis to reflect changes in operating parameters for the systems involved and to correct apparent non-conservatisms in the original analysis.

Summary of Safety Evaluation

The updated evaluation shows that the design containment vacuum condition will not be exceeded assuming that one of the two redundant vacuum breaker assemblies is functioning. As the design values for the containment are not exceeded, there is no potential increase in consequences or a reduction in margin of safety. These changes do not affect any assumptions or precursors which could lead to any different types of accidents. These changes do not adversely affect the design or operating assumptions used in any accident or transient analyses for any structures, systems, or components important to safety. Thus, there is no increase in the probability of an accident or equipment malfunction previously evaluated, nor is there the possibility of creating an accident or equipment malfunction of a different type. Therefore, these changes do not constitute an unreviewed safety question.

Safety Evaluation 559 – Emergency Lighting Quality Classification

Description of Change

During the industry's initial response to the Browns Ferry fire, fire protection and quality assurance came under scrutiny. An early NSP response was to docket the Operational Quality Assurance Plan. In response to the fire protection/quality assurance concerns, emergency lighting was placed on the OQAP Appendix B, "Prairie Island Structures, Systems, and Components Subject to Appendix B of 10CFR50." As more complete NRC guidance became available, the OQAP Appendix C for fire protection was developed. When the OQAP Appendix C, "Nuclear Plant Fire Protection Program," was developed, "emergency lighting" fell under the auspices of the fire protection program but "emergency lighting" was inadvertently left on the Appendix B list also.

This safety evaluation traces the design and regulatory history of emergency lighting at PINGP and concludes that Appendix R/fire protection lighting is the only emergency lighting system that meets current regulatory requirements. Therefore "emergency lighting" is most appropriately removed from the OQAP Appendix B list and kept under the auspices of the OQAP Appendix C for fire protection purposes.

Summary of Safety Evaluation

The safety evaluation does not cause a physical change to the plant nor does it change the response of the plant to any accident. The safety evaluation merely clarifies our existing design basis.

Safety Evaluation 561 – Alternate Method to Cool RCP Seals

Description of Change

Recent work in developing the most effective method to restore seal cooling is to not restore CC or seal injection, but to use the RCS leakage past the seal during a plant cooldown as an alternate method to cool the seal. The purpose of this evaluation is to provide the basis for a revision to the procedures and associated USAR description for using this method of recovery from a complete loss of seal cooling to the Reactor Coolant Pumps. This evaluation is not applicable to a loss of seal cooling caused by an Appendix R event.

Summary of Safety Evaluation

These changes have no effect on any of the methods, inputs, or assumptions used in any analysis. As the intent is to maintain the RCS cooling and these methods of restoring seal cooling have no adverse effect on this ability, there is reasonable assurance of adequate core cooling. Thus, there is no potential increase in consequences or a reduction in the margin of safety. These changes do not affect any assumptions or precursors which could lead to any different types of accidents. These changes do not adversely affect the design or operating assumptions used in any accident or transient analyses for any structures, systems, or components important to safety. Thus, there is no increase in the probability of an accident or equipment malfunction previously evaluated, nor is there the possibility of creating an accident or equipment malfunction of a different type. Therefore, these changes do not constitute an unreviewed safety question.

Safety Evaluation 563 – P219 and P120 MSLB using NSPNAD-97002 Rev. 1

Description of Change

This SE changes the Main Steam Line Break Analysis of Record for Unit 2 cycle 19 and Unit 1 cycle 20. The Analysis used the new methodology described in Topical Report NSPNAD-97002 Rev. 1.

Summary of Safety Evaluation

There are no unreviewed safety questions since the analysis used NRC approved methodology and met all the acceptance criteria.

Safety Evaluation 564 – Correct Deficiencies in USAR Description of Integrated SI Testing

Description of Change

Various sections of the USAR describe the sequential and functional testing of the ESFAS. Some sections contain errors and language that is confusing. In addition, these statements conflict with the current testing methodology of the plant as contained in the Tech Specs Table 4.1-1B. Also, sections of the USAR that deal with component, system and sequence testing will be clarified to better make these distinctions within the USAR. The purpose of this SE is to correct the USAR to be consistent with the plant Tech Specs and plant procedures.

Summary of Safety Evaluation

No unreviewed safety questions were encountered during this review.

Safety Evaluation 565 – RHR Pump Pit Leak Detection

Description of Change

The as found/installed condition of the RHR Pit Sump level instrumentation is not in agreement with portions of the USAR. This safety evaluation evaluates the effect that the current configuration of the RHR Pit Sump level instrumentation has on its function of reactor coolant leak detection, and the ability of the RHR System to perform its safety function.

Summary of Safety Evaluation

These changes have no effect on any of the methods, inputs, or assumptions used in any analysis. As the intent is to maintain the RHR flow during long term post accident mitigation or normal cooling operations, there is reasonable assurance of adequate core cooling. Thus, there is no potential increase in consequences or a reduction in the margin of safety. These changes do not affect any assumptions or precursors which could lead to any different types of accidents. These changes do not adversely affect the design or operating assumptions used in any accident or transient analyses for any structures, systems, or components important to safety. Thus, there is no increase in the probability of an accident or equipment malfunction previously evaluated, nor is there the possibility of creating an accident or equipment malfunction of a different type. Therefore, these changes do not constitute an unreviewed safety question.

Safety Evaluation 567 – Correct Deficiencies in USAR Description of Response to Single Dropped RCCA

Description of Change

This safety evaluation was developed as a corrective action associated with NCR 19992982 which evaluated a change to plant procedures which was performed without a safety evaluation. The plant procedure directed the plant to be tripped on a single dropped Rod Cluster Control Assembly (RCCA), while the USAR directed the RCCA to be retrieved with no reactor trip required. The purpose of this safety evaluation is to perform an evaluation of the procedure changes and correct the USAR to be consistent with the plant procedures.

Summary of Safety Evaluation

The accident analysis for a dropped RCCA was not modified by this change. Rather this change is in how the plant would respond if a reactor trip does not occur. Therefore, no changes in consequences or probability or type of an accident could occur. No malfunction of equipment important to safety could occur, because there is no requirement for equipment important to safety to actuate based on the accident analysis. Therefore, there is no unreviewed safety question associated with this change.

Safety Evaluation 568 – Containment Spray Nozzle Test

Description of Change

The USAR describes the method used to verify that the containment spray nozzles are not obstructed. Due to system configuration changes, it is necessary to deviate from this description. The purpose of this evaluation is to provide the basis for a revision to the method for this test and to revise the USAR accordingly.

Summary of Safety Evaluation

This testing is performed with the unit shut down and cooled down to less than 200F. Prior to plant startup, system restoration and testing ensures that the containment spray system can perform its design functions. These changes have no effect on any of the methods, inputs, or assumptions used in any analysis. There is no potential increase in consequences or a reduction in the margin of safety. These changes do not affect any assumptions or precursors which could lead to any different types of accidents. These changes do not adversely affect the design or operating assumptions used in any accident or transient analyses for any structures, systems, or components important to safety. Thus, there is no increase in the probability of an accident or equipment malfunction previously evaluated, nor is there the possibility of creating an accident or equipment malfunction of a different type. Therefore, these changes do not constitute an unreviewed safety question.

Safety Evaluation 570 – Install Jumper during Bus 25 Load Sequencer Testing

Description of Change

This safety evaluation addresses the interim condition while the D5 breaker control circuit remains as designed and installed by 97FP26 Rev. 0 and a jumper is required to test the load sequencer in SP 2094. A procedure change for SP 2094 will reflect installing a jumper around the RTLR relay contact in the D5 breaker close control circuit at the beginning of the SP and removal of the jumper at the end of the SP. A precaution will also be added to address Appendix R concerns for limiting the time that the jumper is installed. Applying the jumper across the RTLR contact during SP 2094 is safe because it does not affect the ability of the Load Sequencer or D5 to perform their design basis functions.

If the sequencer receives an initiating signal while it is in test, the test aborts and the sequencer returns to normal operating mode. When the sequencer returns to operational status, the auto close portion of the circuit which contains the sequencer 52C contact would be used to close the EDG breaker. If the jumper is installed across the RTLR contact, there is no change in design basis function of the sequencer because the jumper is across a single contact on the RTLR relay. This contact is only used in the EDG breaker close circuit; this contact is not used by the Load Sequencer. Furthermore, there is no change in design basis function of the breaker auto close circuit because this RTLR contact is a duplication of the RTLR function in the auto close logic and the RTLR input to the sequencer, which generates the 52C contact in the breaker close circuit, is maintained.

If the sequencer fails during a scenario where an event occurs while the sequencer is in test, the remote manual closure method could be used to restore voltage to the bus from D5 manually per the EOP's. If the jumper is still in place, the remote manual close circuit will perform as it did prior to design change 97FP26 with no RTLR contact. The sync check relaying provides the voltage and frequency checking function for this closure method. Since the RTLR contact is only in the circuit for the Appendix R control/relay room fire, there is no change in design basis function for the remote manual close circuit either.

The jumper will be installed in accordance with 5AWI 3.9.0 which directs use of proper jumpers, installation, removal, and independent verification. If the jumper fails, the auto close circuit will remain intact and will function as designed. If the jumper is only half installed at the time of an event, the circuit will remain intact and the RTLR contact will still be in the circuit, and again there will be no change in function of the circuit.

The SP will direct the installation and removal of the jumper so the jumper will only be installed during the performance of SP 2094. During all other times, the circuit will function properly as designed, installed, and evaluated by design change 97FP26.

SP 2094 does not take any components out of service; however, the Appendix R function of D5 is impaired with the jumper in place. The jumper will normally be installed for less than 30 minutes. This is significantly less than the 30 day Appendix R component out-of-service limitation of 5AWI 3.15.0 or the 7 day Tech Spec LCO for the EDG. The note added to the SP ensures that the jumper will not impair the Appendix R function of D5 beyond that normally allowed by Tech Specs and 5AWI 3.15.0.

Installation of the jumper during testing does not change the Load Sequencer or D5 performance during Design Basis Accidents and Events. Installation of the jumper during the test does not change the sequencer test or the automatic abort of the test as described in the NRC SER for the Load Sequencer.

Summary of Safety Evaluation

This procedure change only changes the way the Bus 25 Load Sequencer is tested. The Load Sequencer and the EDG are not accident initiators. Installation of the jumper during Load Sequencer testing does not change the Load Sequencer or D5 performance during Design Basis Accidents and Events. Installation of the jumper during the test does not change the sequencer test or the automatic abort of the test as described in the NRC SER for the Load Sequencers. Installation of the jumper allows the Load Sequencer to be tested monthly to satisfy Tech Specs requirements. This safety evaluation concludes that this procedure change is not an Unreviewed Safety Question and does not affect or require a change to the Plant Technical Specifications or the SAR.

CHANGE TO REGULATORY COMMITMENTS

Regulatory Commitment Change 00-01

In our response to GL 88-17 dated 1/6/1989, NSP committed to recording RCS water level from the tygon tube hourly. The tygon tube will no longer be used for RCS draindown operations at Prairie Island. The tygon tube is being replaced by utilizing the existing Refueling Canal Level Transmitter which has dedicated indication on the main control board. Redundant indication is also available from permanently installed RCS level transmitters that read out on ERCS terminals in the control room. Frequency of logging indication is controlled in draining procedures.

ATTACHMENT 2

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

Revision 22 to the Updated Safety Analysis Report

Instructions:

1. Remove and discard individual USAR pages, tables, and figures and replace with the new pages provided. Special instructions, where applicable, are included with the replacement pages.

2. When page removal/replacement is complete, review the USAR List of Effective Pages to ensure your copy of the USAR is current and complete. Contact Nuclear Licensing at 651-388-1121, Extension 4152 if you require additional assistance.

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1	12/82	6/30/83	
2	12/83	6/29/84	
3	12/84	6/28/85	
4	12/85	6/30/86	
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6	12/87	6/30/88	
7	12/88	6/30/89	
8	12/89	6/30/90	
9	12/90	6/25/91	
10	12/91	6/29/92	
11	9/93	9/30/93	Not a general update, but virtually only covers the changes associated with the SBO/ESU project (addition of D5 and D6 diesel generators and upgrade of the electrical distribution system)
12	12/93	6/15/94	
13	6/95	12/27/95	
14	3/97	9/30/97	
15	3/97	10/23/97	Update to correct errors in the Rev. 14 submittal.
16	8/98	2/8/99	
17	5/99	5/21/99	
18	7/99	7/22/99	
19	8/99	8/31/99	
20	12/99	12/17/99	
21	4/00	4/21/00	
22	8/00	9/7/00	

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Joel P. Sorensen — Approved by Site General Manager or Designee

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1.2 PRINCIPAL DESIGN CRITERIA

The Prairie Island Nuclear Generating Plant was designed and constructed to comply with NSP's understanding of the intent of the AEC General Design Criteria for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967. Since the construction of the plant was significantly completed prior to the issuance of the February 20, 1971, 10CFR50, Appendix A General Design Criteria, the plant was not reanalyzed and the FSAR was not revised to reflect these later criteria. However, the AEC Safety Evaluation Report acknowledged that the AEC staff assessed the plant, as described in the FSAR, against the Appendix A design criteria and "... are satisfied that the plant design generally conforms to the intent of these criteria."

Section 1.2 of the USAR presents a brief description of related plant features which are provided to meet the design objectives reflected in groups of the proposed general design criteria. Section 1.5 of the USAR presents a brief description of related plant features which are provided to meet the design objectives reflected in each of the 70 proposed (July 1967) general design criteria. The description of plant structures, systems and components is more fully developed in those succeeding sections of the USAR as indicated by the references. These individual sections state the licensee's understanding of the intent of the criterion and describe how the plant design complies with those requirements.

For those structures, systems and components that have been added to the plant or other licensing commitments made, the appropriate vintage general design criteria has been identified in the applicable section of the USAR.

In Section 1.5, those criterion which were originally designated in parentheses as Category "A" required that more definitive information be provided to the AEC at the construction permit stage. All other criterion were designated as Category "B." However, these categories are no longer applicable and are not included.

1.2.1 Overall Plant Requirements (GDC 1 - GDC 5)

1. Quality Standards
2. Performance Standards
3. Fire Protection
4. Sharing of Systems
5. Records Requirements

All systems and components of the facility are classified according to their importance. Those items vital to safe shutdown whose failure might cause or increase the severity of an accident or result in an uncontrolled release of substantial amount of radioactivity are designated Design Class I.

Design Class I systems and components are essential to the protection of the health and safety of the public. Quality standards of material selection, design, fabrication and inspection conform to the applicable provisions of recognized codes, and good nuclear practice.

All systems and components designated Design Class I are designed so that there is no loss of function in the event of the Design Basis Earthquake acting in the horizontal (0.12g) and vertical (0.08g) directions simultaneously. In addition, Design Class I structures and equipment are designed to withstand all environmental factors including tornadoes. The working stress for both Design Class I and Design Class II items are kept within code allowable values for the operating basis earthquake. Similarly, measures are taken in the plant design to protect against high winds, flooding, and other natural phenomena.

Fire prevention in all areas of the nuclear unit is provided by structure and component design which maximizes the use of fire-resistant materials, optimizes the containment of combustible materials and maintains exposed combustible materials below their ignition temperature in the design atmosphere. Fixed or portable fire fighting equipment is provided with capacities proportional to the energy that might credibly be released by fire.

The Fire Protection System provided has the design capability to extinguish any fire which might occur at the plant.

Those systems of components which are shared, between the two units or functionally within a single unit, are designed in such a manner that plant safety is not impaired by the sharing.

A complete set of as-built facility plant and system diagrams, including arrangement plans and structural plans, and records of initial tests and operation are maintained throughout the life of the plant. A set of all the quality assurance data generated during fabrication and erection of the essential components of the plant, as defined by the quality assurance program, is retained.

Reference sections:

Section Title	Section
Methodology, Hydrology & Seismology	2.3, 2.4, 2.6
Reactor Coolant System	4.1
Containment System	5.1
Engineered Safety Features	6.1
Plant Instrumentation and Control Systems	7.1
Fire Prevention Design	7.8.4
Plant Electrical Systems	8.1
Plant Fire Protection Program	10.3.1
Plant Principal Structures and Equipment	12.2
Initial Tests and Operation	13.4.1, Appendix J
Quality Assurance	
Design & Construction (FSAR)	Appendix C
Operation (USAR)	13.4.5, Appendix C

1.2.2 Protection by Multiple Fission Product Barriers (GDC 6-GDC 10)

6. Reactor Core Design
7. Suppression of Power Oscillations
8. Overall Power Coefficient
9. Reactor Coolant Pressure Boundary
10. Containment

The reactor core, with its related control and protection system, is designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The core design, together with reliable process and decay heat removal systems, provides for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations.

Each Reactor Control and Protection System is designed to actuate a reactor trip for any anticipated combination of plant conditions, when necessary, to ensure a minimum Departure from Nucleate Boiling (DNB) ratio equal to or greater than the applicable limit.

The design of the reactor core and related protection systems ensures that power oscillations which could cause fuel damage in excess of acceptable limits are not possible or can be readily suppressed.

The potential for possible spatial oscillation of core power distribution has been reviewed. It is concluded that tolerable low frequency xenon oscillations may occur in the axial dimension. Control systems (control rods and boron) are available to suppress these oscillations. The core is stable to xenon oscillations in the X-Y dimension.

Out-of-core instrumentation is provided to obtain necessary information concerning power distribution. This instrumentation is adequate to enable the operator to monitor and control xenon induced oscillations.

The Reactor Coolant System in conjunction with its control and protective provisions is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions, and maintain the stresses within applicable code stress limits.

The materials of construction of the pressure boundary of the Reactor Coolant System are protected, by control of coolant chemistry, from corrosion phenomena which might otherwise reduce the system structural integrity during its service lifetime.

System conditions resulting from anticipated transients or malfunctions are monitored, and appropriate action is automatically initiated to maintain the required cooling capability and to limit system conditions to a safe level.

The system is protected from overpressure by means of pressure relieving devices, as required by Section III of the ASME Boiler and Pressure Vessel Code.

Isolable sections of the system containing components designed in conformance with Section III of the ASME Boiler and Pressure Vessel Code are provided with overpressure relieving devices discharging to closed systems, such that the system code allowable relief pressure within the protected section is not exceeded.

The containment design pressure and temperature exceeds the peak pressure and temperature occurring as the result of the complete blowdown of the reactor coolant through any pipe rupture of the Reactor Coolant System up to and including the hypothetical severance of a reactor coolant pipe.

The penetration for the main steam, feedwater, blowdown and sample lines are designed so that the penetration is stronger than the piping system and the containment will not be breached due to a hypothesized pipe rupture. All lines connected to the Reactor Coolant System that penetrate the containment are also anchored in the loop compartment shield walls and are each provided with at least one valve between the anchor and the coolant system. These anchors are designed to withstand the thrust moment and torque resulting from a hypothesized rupture of the attached pipe or the loads induced by the Design Basis Earthquake.

All isolation valves are supported to withstand, without impairment of valve operability, the loading of the design basis accident coincident with the Design Basis Earthquake.

Reference section:

Section Title	Section
Reactor	3.1, 3.2
Reactor Coolant System	4
Containment System	5
Plant Protection Systems	7.4
Safety Analysis	14

1.2.3 Nuclear and Radiation Controls (GDC 11 - GDC 18)

11. Control Room
12. Instrumentation and Control System
13. Fission Process Monitors and Controls
14. Core Protection Systems
15. Engineered Safety Features Protection Systems
16. Monitoring Reactor Coolant Pressure Boundary
17. Monitoring Radioactivity Releases
18. Monitoring Fuel and Waste Storage

The plant is equipped with a control room which contains the controls and instrumentation necessary for operation of both reactors and turbine generators under normal and accident conditions.

Sufficient shielding, distance, and containment integrity are provided to assure that control room personnel shall not be subjected to doses under postulated accident conditions during occupancy of, ingress to and egress from the control room which, in the aggregate, would exceed 5 Rem to the whole body or its equivalent to any part of the body, for the duration of the accident.

For each unit, instrumentation and controls essential to avoid undue risk to the health and safety of the public are provided to monitor and maintain neutron flux, reactor coolant pressure, flow rate, temperature, and control rod positions within prescribed operating ranges.

Other instrumentation and control systems are provided to monitor and maintain within prescribed operating ranges the temperatures, pressures, flows, and levels in the Reactor Coolant Systems, Steam Systems, Containments and other Auxiliary Systems. The quantity and types of instrumentation provided are adequate for safe and orderly operation of all systems and processes over the full operating range of the plant.

The operational status of each reactor is monitored from the control room. When the reactor is subcritical the neutron source multiplication is continuously monitored and indicated by proportional counters located in instrument wells in the primary shield adjacent to the reactor vessel. The source range detector channels can be checked prior to operations in which criticality may be approached. Any appreciable increase in the neutron source multiplication, including that caused by the maximum physical boron dilution rate, is slow enough to give ample time to start corrective action (boron dilution stop and/or emergency boron injection) to prevent the core from becoming critical.

Means for showing the relative reactivity status of each reactor is provided by control bank positions displayed in the control room. Periodic samples of coolant boron concentration can be taken. The variation in concentration during core life provides a further check on the reactivity status of the reactor including core depletion.

Instrumentation and controls provided for the protection systems are designed to trip the reactors when necessary to prevent or limit fission product release from the cores and to limit energy release; to signal containment isolation; and to control the operation of engineered safety features equipment.

During reactor operation in the startup and power modes, redundant safety limit signals will automatically actuate two reactor trip breakers which are in series with the rod drive mechanism coils. This action would interrupt rod drive power and initiate reactor trip.

If the reactor protection system receives signals which are indicative of an approach to an unsafe operating condition, the system actuates alarms, prevents control rod motion, initiates load cutback, and/or opens the reactor trip breakers.

The basic reactor tripping philosophy is to define an allowable region of power and coolant temperature conditions. This allowable range is defined by the primary tripping functions, the overpower ΔT trip, overtemperature ΔT trip, and the nuclear power range high flux trip. The operating region below these trip settings is designed so that no combination of power, temperatures and pressure could result in DNBR less than the applicable limit. Additional tripping functions such as power range high positive and negative neutron flux rate, power range neutron flux (low setpoint), intermediate range high neutron flux, source range high neutron flux, pressurizer high pressure, pressurizer low pressure, pressurizer high level, RCP breaker open, RCP bus undervoltage, RCP bus underfrequency, steam generator low-low level, safety injection initiation, turbine trip and manual trip are provided to backup the primary tripping functions for specific accident conditions and mechanical failures.

Rod stops from nuclear intermediate and power range high flux, overpower ΔT and overtemperature ΔT deviation are provided to prevent abnormal power conditions which could result from excessive control rod withdrawal initiated by a malfunction of the reactor control system or by operator violation of administrative procedures.

Positive indication in the control room of leakage of coolant from the Reactor Coolant System to the containment is provided by equipment which permits continuous monitoring of the containment air activity and humidity, and is provided by the runoff from the condensate collecting pans under the cooling coils of the containment air cooling (fan coil) units. The basic design criterion is the detection of deviations from normal containment environmental conditions including air particulate activity, radiogas activity, humidity, condensate runoff and in addition, in the case of gross leakage, the liquid inventory in the process systems and containment sump.

The containment atmosphere, the plant vents, the containment cooling water discharges, the condenser air ejectors, the steam generator blowdown effluents, and the Waste Disposal System liquid effluent are monitored for radioactivity concentration during all normal operations, anticipated transients and accident conditions.

For the case of leakage from the reactor containment under accident conditions the plant area radiation monitoring system supplemented by portable survey equipment provides adequate monitoring of releases during an accident.

Monitoring and alarm instrumentation are provided for fuel and waste storage and handling areas to detect excessive radiation levels. Monitoring and alarms are also provided to detect inadequate cooling of spent fuel. Radiation monitors are provided to maintain surveillance over the release of radioactive gases and liquids.

Controlled ventilation systems remove airborne radioactivity from the atmosphere of the fuel storage and waste treatment areas of the auxiliary building and discharge it through filters to the atmosphere via the vents. Radiation monitors are in continuous service in these areas to actuate high-activity alarms on the control board annunciator.

Reference sections:

Section Title	**Section**
Engineered Safety Features	6
Plant Instrumentation and Control Systems	7
Shielding and Radiation Protection	12.3

1.2.4 Reliability and Testability of Protection Systems (GDC 19-GDC 26)

19. Protection System Reliability
20. Protection Systems Redundancy and Independence
21. Single Failure Definition
22. Separation of Protection and Control Instrumentation Systems
23. Protection Against Multiple Disability for Protection Systems
24. Emergency Power for Protection Systems
25. Demonstration of Functional Operability of Protection Systems
26. Protection Systems Fail-safe Design

Upon a loss of power to the coils, the rod cluster control (RCC) assemblies are released and fall by gravity into the core. The reactor internals, fuel assemblies, RCC assemblies and drive system components are designed as Class I equipment. The RCC assemblies are fully guided through the fuel assembly and for the maximum travel of the control rod into the guide tube. Furthermore, the RCC assemblies are never fully withdrawn from their guide thimbles in the fuel assembly while in the core. As a result of these design safeguards and the flexibility designed into the RCC assemblies, abnormal loadings and misalignments can be sustained without impairing operation of the RCC assemblies.

Protection channels are designed with sufficient redundancy for individual channel calibration and testing to be made during operation without degrading the reactor protection system. Bypass removal of one trip circuit is accomplished by placing that channel in a partial-tripped mode, i.e., a two-out-of-three channel becomes a one-out-of-two channel. Testing does not cause a trip unless a trip condition exists in a concurrent channel. The trip signal furnished by the remaining channels would be unimpaired in this event.

In the Reactor Protection System two reactor trip breakers are provided to interrupt power to the RCCA drive mechanisms. The breaker main contacts are connected in series (with the power supply) so that opening either breaker interrupts power to all RCC assemblies permitting them to fall by gravity into the core. Each breaker is opened through an undervoltage trip coil. Each protection channel actuates two separate trip logic trains, one for each reactor trip breaker undervoltage trip coil. The protection system is thus inherently safe in the event of a loss of rod control power.

Channel independence is carried throughout the system extending from the sensor to the relay actuating the protective function. The protective and control functions when combined are combined only at the sensor. A failure in the control circuit does not affect the protection channel.

The power supplied to the channels is fed from four instrument buses for each unit. Each of the buses is normally powered through an inverter by a 480v safeguards bus, which can be connected to one of the plant's emergency generators. In the event of the loss of its associated 480v safeguards bus, each of the instrument buses is powered by one of the plant's 125v safeguards batteries.

The initiation of the engineered safety features provided for loss-of-coolant accidents, e.g., high head safety injection and residual heat removal pumps, and containment spray systems, is accomplished from redundant signals derived from Reactor Coolant System and containment pressure instrumentation. The initiation signal for containment spray comes from coincidence of three sets of one-out-of-two high-high containment pressure signals. On loss of voltage of a safety features equipment bus, the diesel generator will be automatically started and connected to the bus provided no other source of power is available to the bus. The signals for initiation of safety injection are main steam line low pressure, pressurizer low pressure, containment high pressure and manual from the control room. A safety injection initiation will then cause a reactor trip, isolate main feedwater, start the diesel generators, start the auxiliary feedwater pumps, safety injection pumps, containment fan coil units and safeguards cooling water pumps, initiate containment isolation, containment ventilation isolation and control room ventilation isolation. The main steam isolation valves on both loops will be closed by a high-high containment pressure signal. The main steam isolation valve will be closed by a high-high steam flow in that loop coincident with a safety injection signal or high steam flow coincident with low-low T-average and a safety injection signal.

The components of the protection system are designed and laid out so that the mechanical and thermal environment accompanying any emergency situation in which the components are required to function does not interfere with that function.

Each protection channel in service at power is capable of being calibrated and tripped independently by simulated signals to verify its operation without tripping the plant.

Each reactor trip circuit is designed so that trip occurs when the circuit is de-energized; an open circuit or loss of channel power therefore, causes the system to go into its trip mode. In a two-out-of-three circuit, the three channels are equipped with separate primary sensors and each channel is energized from independent electrical buses. In addition, the reactor protection system will energize the normally de-energized shunt trip device, which in turn trips the reactor trip breaker.

Redundancy in emergency power is provided in that there are two diesel-generator sets dedicated to each unit, and capable of supplying separate 4160 volt buses. One complete set of safety features equipment for the associated unit is therefore independently supplied from each diesel generator.

Diesel engine cranking is accomplished by a stored energy system supplied solely for the associated diesel-generator. The undervoltage relay scheme is designed so that loss of power does not prevent the relay scheme from functioning properly.

The ability of the diesel-generator sets to start within the prescribed time and to carry load is checked during the integrated SI test. The diesel-generator breaker is closed automatically after starting during this testing. The generator may also be manually synchronized to the 4160 volt bus for loading.

Reference section:

Section Title	Section
Plant Protection System	7.4
Plant Electrical Systems	8

1.2.5 Reactivity Control (GDC 27 - GDC 32)

27. Redundancy of Reactivity Control
28. Reactivity Hot Shutdown Capability
29. Reactivity Shutdown Capability
30. Reactivity Holddown Capability
31. Reactivity Control Systems Malfunction
32. Maximum Reactivity Worth of Control Rods

In addition to the reactivity control achieved by the RCC assemblies as detailed in Section 7, reactivity control is provided by the Chemical and Volume Control System which regulates the concentration of boric acid solution neutron absorber in the Reactor Coolant System. The system is designed to prevent uncontrolled or inadvertent reactivity changes which might cause system parameters to exceed design limits.

The reactivity control systems provided are capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes.

The RCC assemblies are divided into two categories comprised of control and shutdown rod groups. The control group of RCC assemblies is used to compensate for short term reactivity changes at power such as those produced due to variations in reactor power requirements or in coolant temperature. The soluble poison control is used to compensate for the more slowly occurring changes in reactivity throughout core life such as those due to fuel depletion, fission product buildup and decay, and load follow.

The shutdown groups are provided to supplement the control groups of RCC assemblies to make the reactor at least one per cent subcritical (keff = 0.99) following trip from any credible operating condition to the hot zero power condition assuming the most reactive RCC assembly remains in the fully withdrawn position.

Any time that the plant is at power, the quantity of boric acid retained in one of the boric acid tanks and ready for injection always exceeds that quantity required for normal cold shutdown of one unit.

For each unit, boric acid is pumped from the boric acid tanks by one of the boric acid transfer pumps to the suction of the charging pumps which inject boric acid into the reactor coolant system. Each charging pump and any boric acid transfer pump can be operated from diesel generator power on loss of offsite power. Boric acid can be injected by one pump at a rate which takes the reactor to hot shutdown, with no rods inserted, in less than 30 minutes. In 45 additional minutes, enough boric acid can be injected to compensate for xenon decay although xenon decay below the equilibrium operating level does not begin immediately, but could occur up to 26 hours after shutdown, depending upon power history. If two boric acid transfer pumps and two charging pumps are available, these time periods are reduced. Additional boric acid injection is employed if it is desired to bring the reactor to cold shutdown conditions.

In the event that injection using the Charging Pumps is not available, the Safety Injection system can perform this function using borated water from either the boric acid tanks and/or the refueling water storage tank. If necessary, the RCS can be sufficiently depressurized to allow injection with the Safety Injection Pumps. Since the CVCS is normally used for responding to slower reactivity transients, crediting the Safety Injection Pumps in this event is considered acceptable.

The Reactor Protection Systems are designed to limit reactivity transients to DNBR equal to or greater than the applicable limit due to any single malfunction in the deboration controls.

The maximum reactivity worth of control rods and the maximum rates of reactivity of insertion employing both control rods and boron removal are limited to values for which acceptable transient analysis results are obtained in terms of preventing rupture of the reactor coolant pressure boundary or disruption of the core or vessel internals to a degree so as to lose capability to cool the core.

The rod cluster drive mechanisms are wired into preselected groups and are normally prevented from being withdrawn in other than their respective groups. The control and shutdown rod drive mechanisms are of the magnetic latch type and the coil actuation is programmed to provide variable speed rod travel. The insertion rate is analyzed in the detailed plant analysis. It is assumed that two of the highest worth groups are accidentally withdrawn at maximum speed. This is to insure that the reactivity insertion rates are well within the capability of the reactor protection circuits. Thus core damage is prevented.

Reference sections:

Section Title	Section
Reactor; Principal Design Criteria	3.1.2
Plant Protection System	7.4
Regulating Systems	7.2
Chemical and Volume Control System	10.2.3

1.2.6 Reactor Coolant Pressure Boundary (GDC 33 - GDC 36)

33. Reactor Coolant Pressure Boundary Capability
34. Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention
35. Reactor Coolant Pressure Boundary Brittle Fracture Prevention
36. Reactor Coolant Pressure Boundary Surveillance

The reactor coolant boundary is shown to be capable of accommodating without rupture, the static and dynamic loads imposed as a result of a sudden reactivity insertion such as a rod ejection.

The operation of the reactor is such that the severity of a rod ejection accident is inherently limited. Since rod cluster control assemblies are used to control load variations only and boron dilution is used to compensate for core depletion, only the RCCA in the controlling groups are inserted in the core at power, and at full power these rods are only partially inserted. This condition can be verified by a rod insertion limit monitor.

By using the flexibility in the selection of control groupings, radial locations and position as a function of load, the design limits the maximum fuel temperature for the highest worth ejected rod to a value which precludes any resultant damage to the Reactor Coolant System pressure boundary, from possible excessive pressure surges.

The Reactor Vessel Material Surveillance Program monitors the effects of radiation on reactor vessel materials, and establishes operating limits to assure that brittle fracture of the reactor vessel will not occur. The program is in accordance with ASTM-E-185.

Reference sections:

Section Title	Section
Reactor Coolant System	4.1, 4.6
Vessel NDTT	4.7.2 Appendix 4A of FSAR

1.2.7 Engineered Safety Features (GDC 37 - GDC 65)

37. Engineered Safety Features Basis for Design
38. Reliability and Testability of Engineered Safety Features
39. Emergency Power for Engineered Safety Features
40. Missile Protection
41. Engineered Safety Features Performance Capability
42. Engineered Safety Features Components Capability
43. Accident Aggravation Prevention
44. Emergency Core Cooling Systems Capability
45. Inspection of Emergency Core Cooling Systems
46. Testing of Emergency Core Cooling Systems Components
47. Testing of Emergency Core Cooling Systems
48. Testing of Operational Sequence of Emergency Core Cooling Systems

49. Containment Design Basis
50. NDT Requirement for Containment Material
51. Reactor Coolant Pressure Boundary Outside Containment
52. Containment Heat Removal Systems
53. Containment Isolation Valves
54. Containment Leakage Rate Testing
55. Containment Periodic Leakage Rate Testing
56. Provisions for Testing of Penetrations
57. Provisions for Testing of Isolation Valves
58. Inspection of Containment Pressure-reducing Systems
59. Testing of Containment Pressure-reducing Systems
60. Testing of Containment Spray Systems
61. Testing of Operational Sequence of Containment Pressure-reducing Systems
62. Inspection of Air Cleanup Systems
63. Testing of Air Cleanup Systems Components
64. Testing of Air Cleanup Systems
65. Testing of Operational Sequence of Air Cleanup Systems

The design, fabrication, testing and inspection of the core, reactor coolant pressure boundary and their protection systems give assurance of safe and reliable operation under all anticipated normal, transient, and accident conditions. However, engineered safety features are provided in the facility to back up the safety provided by these components.

These engineered safety features have been designed to cope with any size reactor coolant pipe break up to and including the circumferential rupture of any pipe assuming unobstructed discharge from both ends, and to cope with any steam or feedwater line break. The total loss of all offsite power is assumed concurrent with these accidents.

The primary purpose of the Safety Injection System is to automatically deliver cooling water to the reactor core in the event of a loss-of-coolant accident. This limits the fuel clad temperature and ensures that the core will remain intact and in place, with its heat transfer geometry preserved. This protection is afforded for:

a. All pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant loop, assuming unobstructed discharge from both ends.

b. A loss of coolant associated with the rod ejection accident.

c. A steam generator tube rupture.

The principal design criteria for loss-of-coolant accident evaluations are given in Section 14.6.

These criteria assure the core geometry is retained to such an extent that effective cooling of the core is not impaired.

For any rupture of a steam pipe and the associated uncontrolled heat removal from the core, the Emergency Core Cooling System adds shutdown reactivity so that with a stuck rod, no off-site power, and minimum engineered safety features, there is no consequential damage to the primary Reactor Coolant System and the core remains in place and intact. With no stuck rod, no-off-site power and all equipment operating at design capacity, there is insignificant cladding rupture.

The Safety Injection System consists of centrifugal safety injection pumps driven by electric motors, and passive accumulator tanks which are self energized and which act independently of any actuation signal or power source.

The release of fission products from the containment is limited in three ways:

a. Blocking the potential leakage paths from the containment. This is accomplished by:

    1. A steel, leak-tight containment vessel with testable penetrations;

    2. Isolation of process lines which imposes double barriers for each line penetrating the containment;

    3. A shield building surrounding the containment vessel with an associated ventilation system containing particulate, absolute and charcoal filters;

    4. A special zone ventilation system, collecting leakage from the auxiliary building and discharging it through particulate, absolute and charcoal filters.

b. Reducing the fission product concentration in the containment atmosphere. This is accomplished by spraying water which removes airborne elemental iodine vapor by washing action.

c. Reducing the containment pressure and thereby limiting the driving potential for fission product leakage by cooling the containment atmosphere using the following systems:

    1. Containment Spray System

    2. Containment Air Cooling System

A comprehensive program of plant testing is formulated for all equipment systems and system control vital to the functioning of engineered safety features. The program consists of performance tests of individual pieces of equipment in the manufacturer's shop, integrated tests of the system as a whole, and periodic tests of the actuation circuitry and mechanical components to assure reliable performance upon demand, throughout the plant lifetime.

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The plant is supplied with normal, standby and emergency power sources as follows:

a. The normal source of auxiliary power for safeguards equipment is the off-site power source. Power is supplied via the reserve auxiliary transformer or the cooling tower substation transformer.

b. Two emergency diesel-generators for each unit are connected to the emergency buses to supply power in the event of loss of all other a-c auxiliary power. Each of the two emergency diesel generators per unit is capable of supplying automatically the engineered safety features load required for an acceptable post-blowdown containment pressure transient for any loss-of-coolant accident, or for shutdown of the unit.

c. Emergency power supply for vital instruments, for control and for emergency lighting, is supplied from the 125V DC systems.

For such engineered safety features as are required to ensure safety in the event of such an accident or equipment failure, protection from these dynamic effects or missiles is considered in the layout of plant equipment and missile barriers.

Layout and structural design specifically protect injection paths leading to unbroken reactor coolant loops against damage as a result of the maximum reactor coolant pipe rupture. Injection lines penetrate the main compartment walls which act as missile barriers. The injection headers are located in the missile-protected area between the compartment walls and the containment outside wall. Individual injection lines are connected to the injection header, pass through the compartment walls and then connect to the loops. Movement of the injection line associated with rupture of a reactor coolant loop is accommodated by line flexibility and by the design of the pipe supports such that no damage outside the missile barrier is possible.

Each engineered safety feature provides sufficient performance capability to accommodate any single failure of an active component and still function in a manner to avoid undue risk to the health and safety of the public.

Under the hypothetical accident conditions, the Containment Air Cooling System and the Containment Spray System are designed to supply the post-accident cooling capacity to rapidly reduce the containment pressure following blowdown. 00049

All active components of the Safety Injection System (with the exception of injection line isolation valves) and the Containment Spray System are located outside the containment and not subjected to containment accident conditions.

Instrumentation, motors, cables and penetrations located inside the containment are selected to meet the most adverse accident conditions to which they may be subjected. These items are either protected from containment accident conditions or are designed to withstand, without failure, exposure to the worst combination of temperature, pressure, and humidity expected during the required operational period.

The reactor is maintained subcritical following a Reactor Coolant System pipe rupture accident. Introduction of borated cooling water into the core results in a net negative reactivity addition. The control rods insert and remain inserted.

The delivery of cold safety injection water to the reactor vessel following accidental expulsion of reactor coolant has been analyzed. The results indicate that no further loss of integrity of the Reactor Coolant System boundary occurs as explained in Section 4.1.

Design provisions are made to facilitate access to the critical parts of the reactor vessel internals, injection nozzles, pipes, valves and safety injection pumps for visual or boroscopic inspection for erosion, corrosion and vibration wear evidence, and for non-destructive inspection where such techniques are desirable and appropriate.

The design provides for periodic testing of active components of the Safety Injection System for operability and functional performance. The safety injection and residual heat removal pumps are tested periodically during plant operation using the minimum flow recirculation lines provided.

An integrated system test can be performed during each reactor refueling shutdown when the residual heat removal loop is in service. This test would not introduce flow into the Reactor Coolant System but would demonstrate the operation of the valves, pump circuit breakers, and automatic circuitry upon initiation of safety injection.

The design provides for continuously monitoring the accumulator tank pressure and level during plant operation.

The accumulators and the safety injection piping up to the final isolation valve are maintained full of borated water while the plant is in operation. Flow in each of the high head injection header lines and in the main flow line for the residual heat removal pumps is monitored by a flow indicator.

The design provides for capability to test initially, to the extent practical, the full operational sequence up to the design conditions for the Safety Injection System to demonstrate the state of readiness and capability of the system.

These functional tests provide information to confirm valve operating times, pump motor starting times, the proper automatic sequencing of load addition to the diesel-generators, and delivery rates of injection water to the Reactor Coolant System.

The following general criteria are followed to assure conservatism in computing the required containment structural load capacity:

a. In calculating the containment pressure, rupture sizes up to and including a double-ended severance of reactor coolant pipe are considered.

b. In considering post-accident pressure effects, various malfunctions of the emergency systems are evaluated. Contingent mechanical or electrical failures are assumed to disable one of the diesel generators, such that only two of the four fan-coil units and one of the two containment spray pumps operate.

c. The pressure and temperature loadings obtained by analyzing various loss-of-coolant accidents, when combined with operating loads and maximum wind or seismic forces, do not exceed the load-carrying capacity of the structure, its access opening or penetrations.

Discharge of reactor coolant through a double-ended rupture of the main loop piping, followed by operation of only those engineered safety features which can run simultaneously with power from one emergency on-site diesel generator results in a sufficiently low radioactive material leakage from the containment structure that there is not undue risk to the health and safety of the public.

The reinforced concrete shield building containment is not susceptible to a low temperature brittle fracture. The containment vessel is enclosed within the shield building and thus is not exposed to the temperature extremes of the environs. The containment ambient temperature during operation is between 50°F and 120°F which is well above the NDT temperature + 30°F for the liner material. Containment penetrations which can be exposed to the environment are also designed to the NDT + 30°F criterion.

Isolation valves are provided as necessary for all fluid system lines penetrating the containment to assure at least two barriers for redundancy against leakage of radioactive fluids to the environment in the event of a loss-of-coolant accident. These barriers, in the form of isolation valves or closed systems, are defined on an individual line basis. In addition to satisfying containment isolation criteria, the valving is designed to facilitate normal operation and maintenance of the systems and to ensure reliable operation of other engineered safety features.

After completion of the containment structure and installation of all penetrations, an initial integrated leakage rate test was conducted at the peak calculated accident pressure, maintained for a minimum of 24 hours, to verify that the leakage rate was well below the Technical Specification Limit.

Periodic leak rate tests are performed as required in accordance with the Appendix J leak rate testing program.

Penetrations are designed with double seals so as to permit test pressurization of the interior of the penetration. To accomplish this, a supply of clean, dry, compressed air or nitrogen is connected to the penetrations raising the internal pressure to the containment internal design pressure. Leakage from the system is checked by measurement of the pressure decay or metering of flow rate required to maintain the test pressure. In the event excessive leakage is discovered, penetration groups can then be checked separately.

Capability is provided to the extent practical for testing the functional operability of valves and associated apparatus during periods of reactor shutdown.

Initiation of containment isolation employs coincidence circuits which allow checking of the operability and calibration of one channel at a time.

The main steam and feedwater piping and isolation valves in systems which connect to the Reactor Coolant System are hydrostatically tested to detect leakage. The steam line isolation valves are tested periodically for operability.

Design provisions are made to the extent practical to facilitate access for periodic visual inspection of important components of the Containment Air Cooling and Containment Spray Systems. 00049

The containment pressure reducing systems are designed to the extent practical so that the spray pumps, spray injection valves and spray nozzles can be tested periodically and after any component maintenance for operability and functional performance.

Permanent test lines for all the containment spray loops are located so that all components up to the isolation valves at the containment may be tested. These isolation valves are checked separately.

Periodic testing is performed to verify that spray nozzles are not obstructed. 00115

Capability is provided to test initially, to the extent practical, the operational startup sequence beginning with transfer to alternate power sources.

Reference sections:

<u>Section Title</u>	<u>Section</u>
Containment System	5
Engineered Safety Features	6
Plant Electrical Systems	8

1.2.8 <u>Fuel and Waste Storage Systems (GDC 66-GDC 69)</u>

66. Prevention of Fuel Storage Criticality
67. Fuel and Waste Storage Decay Heat
68. Fuel and Waste Storage Radiation Shielding
69. Protection Against Radioactivity Release From Spent Fuel and Waste Storage

Borated water is used to fill the spent fuel storage pit at a concentration to maintain K_{eff} <0.95 and to prevent dilution of the reactor cavity and refueling canal during refueling operations. The fuel is stored vertically in an array with sufficient center-to-center distance between assemblies to assure keff < 1.0 even if unborated water were used to fill the pit.

During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration is maintained at not less than that required to shutdown the core to a keff $\leqq$ 0.95 with all control rods are withdrawn from the core.

The design of the fuel handling equipment incorporates built-in interlocks and safety features.

The refueling water provides a reliable and adequate cooling medium for spent fuel transfer. Heat removal is accomplished with an auxiliary cooling system.

Adequate shielding for radiation protection is provided during reactor refueling by conducting all spent fuel transfer and storage operations under water. This permits visual control of the operation at all times while maintaining radiation levels less than 2.5 mrem/hr at or near the water surface. Pit water level is indicated, and water to be removed from the pit must be pumped out as there are no gravity drains. Shielding is provided for waste handling and storage facilities to permit operation within requirements of 10CFR20.

Gamma radiation is continuously monitored at various locations in the Auxiliary Building. A high level signal is alarmed locally and is annunciated in the control room.

Auxiliary shielding for the Waste Disposal System and its storage components is designed to limit the dose rate to levels not exceeding 1 mrem/hr in normally occupied areas, to levels not exceeding 2.5 mrem/hr in intermittently occupied areas and to levels not exceeding 15 mrem/hr in controlled occupancy areas.

All waste handling and storage facilities are contained and equipment designed so that accidental releases directly to the atmosphere are monitored and will not exceed the guidelines of 10CFR100.

The reactor cavity, refueling canal and spent fuel storage pit are reinforced concrete structures with seam-welded stainless steel plate liners. These structures are designed to withstand the anticipated earthquake loadings as Design Class I structures so that the liner will prevent leakage.

Reference sections:

Section Title	Section
Fuel Storage and Fuel Handling Systems	10.2.1
Plant Radioactive Waste Control Systems	9
Shielding and Radiation Protection	12.3
Standby Safety Features Analysis	14.5

1.2.9 Plant Effluents (GDC 70)

70. Control of Releases of Radioactivity to the Environment

Liquid, gaseous, and solid waste disposal facilities are designed so that discharge of effluents and off-site shipments are in accordance with applicable governmental regulations.

Radioactive fluids entering the Waste Disposal System are collected in sumps and tanks until determination of subsequent treatment can be made. They are sampled and analyzed to determine the quantity of radioactivity, with an isotopic identification if necessary. Before discharge, radioactive fluids are processed as required and then released under controlled conditions. The system design and operation are characteristically directed toward minimizing releases to unrestricted areas in accordance with Appendix I to 10CFR Part 50. Discharge streams are appropriately monitored and safety features are incorporated to preclude release rates in excess of the limits of 10CFR20.

Radioactive gases are transferred to an augmented gaseous radwaste system. The gases are segregated, recombined, and then pumped by compressors through a manifold to one of the gas decay tanks where they are held a suitable period of time for decay. Cover gases in the nitrogen blanketing system are re-used to minimize gaseous wastes. During normal operation, gases are discharged intermittently at a controlled rate from these tanks through the monitored plant vent.

Liquid wastes are processed to remove radioactive materials. Filter cartridges, the spent resins from the demineralizers, and the concentrates from the evaporators are packaged and stored on-site until shipment off-site for disposal. Miscellaneous solid wastes, such as paper, rags and glassware, are compressed for storage, disposal or further processing.

Reference section:

Section Title	Section
Plant Radioactive Waste Control Systems	9

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1.3 SUMMARY DESIGN DESCRIPTION AND SAFETY ANALYSIS

The inherent design of the pressurized water, closed-cycle reactor significantly reduces the quantities of fission products which are released to the atmosphere. Four barriers exist between fission product accumulation and the environment. These are the uranium dioxide fuel matrix, the fuel cladding, the reactor vessel and coolant loops, and the reactor containment. The consequences of a breach of the fuel cladding are greatly reduced by the ability of the uranium dioxide lattice to retain fission products. Escape of fission products through fuel cladding defects would be contained within the pressure vessel, loops and auxiliary systems. Breach of these systems or equipment would release the fission products to the reactor containment where they would be retained. The reactor containment is designed to adequately retain these fission products under the most severe accident conditions, as analyzed in Section 14.

Several engineered safety features have been incorporated into the plant design to reduce the consequences of a loss of coolant accident. These safety features include a Safety Injection System. This system automatically delivers borated water to the reactor vessel for cooling the core under high and low reactor coolant pressure conditions. The Safety Injection System also serves to insert negative reactivity into the core in the form of borated water during an uncontrolled plant cooldown following a steam line break or an accidental steam release. Other safety features which have been included in the reactor containment design are a Containment Air Cooling System which acts to effect a depressurization of the containment following a loss of coolant accident, and a Containment Spray System which acts to depressurize the containment and remove elemental iodine from the atmosphere by washing action. 00049

1.3.1 Plant Site and Environs

Section 2 of this report provides detailed information on the site and environs of the Prairie Island Nuclear Generating Plant Units 1 and 2 which confirms the suitability of the site. This section summarizes the principal design characteristics of the site and environs.

The plant site is located in southeastern Minnesota on the west bank of the Mississippi River about 26 miles SE of the Twin City Metropolitan Area. The nearest population center is Eagen, Minnesota. Cooling water is drawn from the Mississippi River. Farming is the predominant activity in this moderately-populated area of the state. The plant is situated in a productive dairy farming and vegetable canning region; however, there is heavy industrialization to the northwest in the Twin Cities and to the south in Red Wing.

The sub-surface soils at the site consist of permeable sandy alluvium which are generally suitable from a bearing capacity standpoint for support of the structures. However, settlement restrictions and a low margin of safety against liquefaction of the upper 50 feet (above elevation 645) of alluvium required that certain critical structures be supported on densified sand. Several hundred feet of sound sandstone underlie the alluvial soils.

River flows vary widely through the year. Generally, maximum flows occur in the spring and minimum flows occur in late summer (July, August, September) or mid-winter

(January, February). The low flow of record is 2100 cfs (1936) and the average flow is 15,020 cfs. The plant design, construction and operation, including the radioactive waste control system, take into consideration the extremes of river flow and stage. The cooling towers are operated in accordance with the NPDES permit.

The finished plant grade (695 feet MSL) is about 20 feet above mean river level (674.5 feet MSL), 7 feet above the record (688 feet MSL-1965), and 1 ft above the predicted 1,000 year flood (693.5 feet MSL). The plant is designed to withstand the effects of the probable maximum flood (703.6 feet MSL)

The meteorology of the site area is basically that of a continental location with favorable atmospheric dilution conditions prevailing. Diffusion climatology comparisons with other locations indicate that the site is typical of midwestern United States. All structures are designed to withstand the maximum potential loadings resulting from a wind speed of 100 mph. The design is in accordance with standard codes and normal engineering practices. It is estimated that the probability of experiencing tornadic forces at the site is of the order of one chance per 220 years. In spite of this low probability, features of the plant important to the integrity of reactor core cooling are designed to withstand the forces of short-term tornadoes.

There is no evidence of even ancient inactive faulting closer than six miles to the site. Inactive faults are located approximately 6 and 13 miles from the site. No activity has occurred along either of these faults in recent geologic times. The seismic design for critical structures and equipment for this plant is based on dynamic analyses of acceleration or velocity-response spectrum curves, based on a horizontal ground acceleration of 0.06g. Earthquake design is based on ordinary allowable stresses as set forth in the applicable codes. As an additional requirement, the design is such that a safe shutdown can be made during a horizontal ground acceleration of 0.12g. Seismic design criteria and the safety classification of important components are described in Section 12.

An environmental radiation monitoring program was initiated in May 1970. Measurements are made of the radioactivity present in air, surface and well water, raw milk, vegetation, fish and other selected specimens. An ecological study of the Mississippi River in the areas of the plant was also begun in May 1970. Meteorological and water quality data has been gathered since May 1968.

1.3.2 <u>Structures</u>

The major structures are the reactor containment vessels, the shield buildings, the turbine building, the auxiliary building, D5/D6 diesel generator building, administration and service buildings, intake structures, and radwaste buildings. General equipment and plant layouts appear in Figures 1.1-3 through 1.1-24. All structures housing the reactors, their essential auxiliaries, and engineered safeguards systems are designed and rigorously analyzed to meet the most severe environmental conditions. These conditions and the applicable structural design criteria are described in Section 12.

Each reactor containment consists of a cylindrical steel shell with a hemispherical dome and ellipsoidal bottom designed to withstand the internal pressure accompanying a loss-of-coolant accident. Each containment vessel is surrounded by a cylindrical shield building constructed of reinforced concrete which serves as a radiation shielding for normal operation and for the loss-of-coolant condition. In addition, the shield building acts as a secondary containment structure for control of containment leakage.

The auxiliary building housing the essential auxiliaries, control room and spent fuel storage facilities for both units is located adjacent to the reactor buildings. The turbine building housing the turbine-generators and technical support center for both units is located adjacent to the auxiliary building. The D5/D6 diesel generator building housing the Unit 2 emergency diesel generators and electrical safeguards buses is located adjacent to the auxiliary and turbine buildings. The administration and service buildings housing general offices and computer facilities is located adjacent to the turbine building. The radwaste building, resin disposal building, and drum storage enclosure which house the radioactive waste handling, treatment, storage and disposal facilities for both units are all located adjacent to the auxiliary building.

The plant screenhouse houses the cooling water pumps, fire pumps, circulating water pumps, trash racks and traveling screens. The intake screenhouse contains trash racks and traveling screens.

1.3.3 Nuclear Steam Supply System

The Nuclear Steam Supply System for each unit consists of a pressurized water reactor, Reactor Coolant System, and associated auxiliary fluid systems. The Reactor Coolant System is arranged as two closed reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump and a steam generator. An electrically heated pressurizer is connected to one of the loops.

The reactor core is composed of uranium dioxide pellets enclosed in ZIRLO/Zircaloy tubes with welded end plugs. The tubes are supported in assemblies by a spring clip grid structure. The mechanical control rods consist of clusters of stainless steel clad absorber rods and ZIRLO/Zircaloy guide tubes located within the fuel assembly. The core fuel load for each unit's fuel cycle is described in Sections 14B and 14C.

The steam generators are vertical U-tube units utilizing Inconel tubes. Integral separating equipment reduces the moisture content of the steam at the turbine throttle to 1/4 percent or less.

The reactor coolant pumps are vertical, single stage, centrifugal pumps equipped with controlled leakage shaft seals.

Auxiliary systems are provided to charge the Reactor Coolant System and to add makeup water, purify reactor coolant water, provide chemicals for corrosion inhibition and reactor control, cool system components, remove residual heat when the reactor is shutdown, cool the spent fuel storage pool, sample reactor coolant water, provide for emergency safety injection, and vent and drain the Reactor Coolant System.

1.3.4 Reactor and Plant Control

The reactor is controlled by a coordinated combination of chemical shim and mechanical control rods. The control system allows the plant to accept step load changes of 10% and ramp load changes of 5% per minute over the load range of 15 to 95% power under nominal operating conditions. It is also designed to sustain reactor operation following a step nominal full load rejection up to 47.5% power.

Complete supervision of both the reactor and turbine generator is accomplished from the control room. Units 1 and 2 share the control room located in the auxiliary building. The control room layout including location of control boards for each unit is shown in Figure 7.8-1.

Annunciators for alarms on the two units are on different control boards and have different audible tones which make them distinguishable.

The waste disposal control board is located in the Auxiliary Building. This board permits the control and monitoring of the processing of wastes from a central location in the same general area where equipment is located.

1.3.5 Waste Disposal System

The Waste Disposal System, common to both units, provides all equipment necessary to collect, process, and prepare for disposal all potentially radioactive liquid, gaseous, and solid wastes produced as a result of reactor operation.

Liquid wastes are collected and processed as required. The waste evaporator condensate is sampled to determine residual activity and monitored during discharge to the river via the condenser circulating water discharge to assure concentrations as low as practicable below 10CFR20 limits. The evaporator residues are solidfied, drummed and shipped from the site for ultimate disposal in an authorized location.

Gaseous wastes are collected and stored until their radioactivity level is low enough so that discharge to the environment will be as low as practicable below 10CFR20 limits.

1.3.6 Fuel Handling System

Each reactor is refueled with equipment designed to handle spent fuel under water from the time it leaves either reactor vessel until it is placed in a cask for shipment from the site. Underwater transfer of spent fuel provides an optically transparent radiation shield, as well as a reliable source of coolant for removal of decay heat. This system also provides capability for receiving, handling and storage of new fuel. Both the new fuel storage facility and the spent fuel storage facility are shared by the two units.

1.3.7 Turbine and Auxiliaries

The turbine is a three-element, tandem-compound, four-flow exhaust, 1800 rpm unit that has moisture separation and reheating between the HP and LP elements.

Multi-pressure radial flow surface condensers with deaerating hotwells, steam-jet air ejector, three 50% capacity condensate pumps, two 50% capacity motor-driven feedwater pumps, and five stages of feedwater heaters are provided. One steam-driven auxiliary feedwater pump per unit and one motor-driven auxiliary feedwater pump per unit are available to remove residual heat in case of a complete loss of off-site power.

1.3.8 Electrical System

The main generator is an 1,800 rpm, 3 phase, 60 cycle, hydrogen innercooled unit. One three phase main step-up transformer on each unit delivers power to the 345 KV switchyard.

The Station Service System consists of auxiliary transformers, 4160 V. switchgear, 480 V. motor control centers, and 125 V. d-c and 120 V. a-c equipment.

Emergency power, supplied by alternate sources including two emergency diesel generators for each unit, is capable of operating post-accident containment cooling equipment as well as both high head and low head safety injection pumps to ensure an acceptable post-loss-of-coolant containment pressure transient. Sufficient power capacity is provided to safely shut down the second (non-accident) unit with its emergency diesel generators at the same time adequate power is provided to the engineered safety features of the unit having the accident.

1.3.9 Engineered Safety Features

The Engineered Safety Features provided for this plant have redundancy of component and power sources such that under the conditions of a hypothetical loss-of-coolant accident as well as all other accidents analyzed in Section 14, the system does, including the effects of a single failure, maintain the integrity of the containment and keep the exposure of the public below the guidelines of 10CFR100.

The systems provided are summarized below:

a. The Containment System structure, together with the Containment Isolation, provides a highly reliable, essentially leak-tight barrier against the escape of fission products to the environment.

b. The Safety Injection System provides borated water to cool the core by injection into the core outlet plenum and cold legs of the reactor coolant loops.

c. The Containment Air Cooling System provides a dynamic heat sink to cool the containment atmosphere. The system utilizes the normal containment ventilation and cooling equipment.

d. Each Containment Spray System provides a spray of cool water to the containment atmosphere to work in parallel with the Containment Air Cooling System during the injection phase of LOCA mitigation. In addition to heat removal, the spray system is also effective in scrubbing fission products from the containment atmosphere.

e. The Auxiliary Feedwater System provides high-pressure feedwater to the steam generators in order to maintain water inventory for removal of heat energy from the Reactor Coolant System in the event the main feedwater system is not available. Redundant water supplies and power sources are provided to motor and steam operated pumps.

f. The following redundant ventilation systems are provided to assist in handling activity releases in important areas of the plant:

   1. The Auxiliary Building Special (Category 1 Ventilation Zone) Ventilation System is designed to process high airborne-activities in important areas of the auxiliary building. Air from this ventilation system is passed through particulate, absolute and charcoal filters before release to the environment;

   2. The Shield Building Special Ventilation System provides pressure control in the annulus between the Containment Vessel and the Shield Building, and recirculation of annulus air through particulate, absolute and charcoal filters during accident conditions;

   3. The Control Room Air Ventilation System processes control room air through particulate, absolute and charcoal filters during conditions of high airborne activity in the environs of the control room.

g. Two quick-start diesel generators are provided for each unit to supply adequate power for plant safety in the event of loss of station and off-site a-c power. Each generator has adequate capacity to supply the engineered safety features for the design basis accident in one unit, or to allow the unit to be placed in a safe shutdown condition in the event of loss of outside electrical power.

h. Two 125-V Station Batteries are provided for each unit to supply plant controls, d-c motors, inverters serving non-interruptable a-c buses and emergency lighting. Redundant safety controls, normal controls and nuclear instrument inverters are divided between the two batteries associated with each unit.

i. The Component Cooling System is provided to remove heat from major components in the Nuclear Steam Supply System under normal conditions and from all components associated with the removal of reactor core decay heat under accident conditions.

j. The Cooling Water System provides a water supply for normal plant equipment heat loads, and to safeguards equipment during normal and emergency operating conditions.

1.3.10 Shared Facilities and Equipment

Separate and similar systems and equipment are provided for each unit except for those systems listed in Tables 1.3-1, 1.3-2 and 1.3-3. A functional evaluation of the components of the systems which are required for normal plant operation and are shared by the two units is provided in Table 1.3-2 and for engineered safeguard related systems in the appropriate section as referenced in Table 1.3-1. Table 1.3-3 is a functional evaluation of those shared components not required for normal plant operation.

Those structures and buildings which are shared by the two units are listed below. The related equipment and floorplan layouts are given in Figures 1.1-3 through 1.1-24. A discussion of control room sharing is contained in Section 7.

Auxiliary Building
Radwaste Building
Resin Disposal Building
Drum Transfer and Storage Building
Barrel Storage Building
Turbine Building
Administration and Service Buildings
Control Room
Screenhouse
Spent Fuel Pool Structure and Enclosure
Circulating Water External Structures

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1.5 GENERAL DESIGN CRITERIA

The Prairie Island Nuclear Generating Plant was designed and constructed to comply with NSP's understanding of the intent of the AEC General Design Criteria for Nuclear Power Plant Construction Permits, as proposed on July 10, 1967. Since the construction of the plant was significantly completed prior to the issuance of the February 20, 1971, Appendix A General Design Criteria, the plant was not reanalyzed and the FSAR was not revised to reflect these later criteria. However, the AEC Safety Evaluation Report acknowledged that the AEC staff assessed the plant, as described in the FSAR, against the Appendix A design criteria and "...are satisfied that the plant design generally conforms to the intent of these criteria."

Section 1.2 of the USAR presents a brief description of related plant features which are provided to meet the design objectives reflected in groups of the proposed general design criteria. Section 1.5 of the USAR presents a brief description of related plant features which are provided to meet the design objectives reflected in each of the 70 proposed (July 1967) general design criteria. The description of plant structures, systems and components is more fully developed in those succeeding sections of the USAR as indicated by the references. These individual sections state the licensee's understanding of the intent of the criteria and describe how the plant design complies with those requirements.

For those structures, systems and components that have been added to the plant or other licensing commitments made, the appropriate vintage general design criteria have been identified in the applicable section of the USAR.

In Section 1.5, those criteria which were originally designated in parentheses as Category "A" required that more definitive information be provided to the AEC at the construction permit stage. All other criteria were designated as Category "B." However, these categories are no longer applicable and are not included.

I. OVERALL PLANT REQUIREMENTS

CRITERION 1 - QUALITY STANDARDS

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety functions, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability or codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

ANSWER

The systems and components of the facility have been classified according to their importance in the prevention and mitigation of accidents which could cause undue risk to the health and safety of the public. These classifications are described in Section 12 and Appendix C. A discussion of the codes and standards, quality assurance programs, test provisions, etc., applying to each system is included in that portion of the USAR describing that system. A listing of the applicable sections is included in Section 1.2.

CRITERION 2 - PERFORMANCE STANDARDS

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

ANSWER

The systems and components designated Class I in Section 12 are designed to withstand, without loss of capability to protect the public, the most severe environmental phenomena ever experienced at the site with appropriate margins included in the design for uncertainties in historical data. Potential environmental hazards are discussed and analyzed in Sections 2 and 14 of the report and the influence of these hazards on various aspects of the plant design is discussed in the sections covering the specific systems and components concerned. An outline of the design philosophy for Class I systems and components and a listing of the applicable report sections describing the systems and components covered by this criterion are included in Section 1.2.

CRITERION 3 - FIRE PREVENTION

The reactor facility shall be designed (1) to minimize the probability of events such as fires and explosions and (2) to minimize the potential effects of such events to safety. Noncombustible and fire resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

ANSWER

Through the use of noncombustible and fire resistant materials wherever practical in the facility and the limitation of combustible supplies (e.g., logs, records, manuals, etc.) in such areas as the control rooms to amounts required for current operation, the probability of such events as fire and explosion and the effects of such events should they occur are minimized. Fire protection criteria are discussed in Section 1.2 and specific means of meeting these criteria are described in Sections 7 and 10.

CRITERION 4 - SHARING OF SYSTEMS

Reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing.

ANSWER

As noted in Section 1.2, those systems or components which are shared, either between the two units or functionally within a single unit, are designed in such a manner that plant safety is not impaired by the sharing. Specific instances of component or system sharing are described in the appropriate sections of the report as listed in Section 1.2. A functional evaluation of safety related shared systems is presented in Table 1.3-1 and 1.3-2.

CRITERION 5 - RECORDS REQUIREMENTS

Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator or under its control throughout the life of the reactor.

ANSWER

The applicant maintains, either in its possession or under its control, a complete set of records of the design, fabrication, construction and testing of Class I plant components throughout the life of the plant. Section 13 presents summary of records requirements for plant operation, maintenance, modification and review of procedures.

II. PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS

CRITERION 6 - REACTOR CORE DESIGN

The reactor core shall be designed to function throughout its design lifetime, without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all offsite power.

ANSWER

The ability of the core to function throughout its lifetime without exceeding acceptable fuel damage limits is discussed in Section 3. Detailed information on core design and performance is included in Section 3. The instrumentation and controls associated with the reactor are described in Section 7 while decay heat removal systems are discussed in Sections 6 and 10. Section 14 demonstrates that adequate fuel integrity is maintained under those postulated abnormal situations which could ultimately lead to problems in this area.

CRITERION 7 - SUPPRESSION OF POWER OSCILLATIONS

The core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed.

ANSWER

The inherent ability of the core to prevent and suppress power oscillations and the instrumentation and controls provided to assist in this function is discussed in Sections 3 and 7, respectively.

CRITERION 8 - OVERALL POWER COEFFICIENT

The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive.

ANSWER

The overall power coefficient is discussed in Section 3 and the core reload safety analysis for each fuel cycle, which is contained in Sections 14.B (Unit 1) and 14.C (Unit 2).

CRITERION 9 - REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

ANSWER

As discussed in detail in Section 4, the reactor coolant pressure boundary materials, design, analysis, fabrication and testing preclude the possibility of gross rupture or significant leakage throughout its design lifetime.

CRITERION 10 - CONTAINMENT

Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

ANSWER

The design of the containment structure, and associated auxiliary systems is described in Section 5. Other Engineered Safety Features required to suppress pressure inside the containment are described in Sections 6 and 10. Section 14 demonstrates the adequacy of such systems under various accident conditions including a rupture of the largest reactor coolant pipe.

III. NUCLEAR AND RADIATION CONTROLS

CRITERION 11 - CONTROL ROOM

The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident condition, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10CFR20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

ANSWER

A common control room contains all controls and instrumentation necessary for operation of each unit's reactor, turbine generator, and auxiliary and emergency systems under normal or accident conditions.

The control room is designed and equipped to minimize the possibility of events which might preclude occupancy. In addition, provisions were made for bringing both units to and maintaining a safe shutdown condition for an extended period of time from locations outside the control room.

Safe shutdown is a reactor condition that requires the ability to maintain the reactor sub-critical, remove core decay heat, assure reactor coolant pressure boundary integrity for an extended time period and maintain the integrity of components whose failure could result in excessive offsite release. These conditions may be achieved by operator actions or by automatic reactor protection functions.

The reactor conditions stated above are consistent with those conditions described in the Technical Specifications as Operational Mode 3, except as otherwise defined by 10CFR50, Appendix R.

The employment of non-combustible and fire retardant materials in the construction of the control room contained equipment and furnishings, the limitation of combustible supplies to the minimum consistent with safe and efficient operation of the plant, the location of fire fighting equipment in the control room, and the continuous presence of an operator minimize the probability that the control room will become uninhabitable. In addition, the control room ventilation system is designed to keep the control room at a positive pressure and can be operated in a recirculating mode to prevent fire originating outside the control room from spreading to the control area.

Sufficient shielding, distance, and containment integrity are provided to assure that control room personnel shall not be subjected to doses under postulated accident conditions during occupancy, ingress or egress of control room, which in the aggregate, would not exceed 5 Rem to the whole body or its equivalent to any part of the body,, for the duration of the accident. The control room ventilation consists of a system having a large percentage of recirculated air. After the postulated accident, makeup air is automatically rerouted through a system of HEPA and charcoal filters.

<u>CRITERION 12 - INSTRUMENTATION AND CONTROL SYSTEMS</u>

Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

ANSWER

As discussed in detail in Section 7, sufficient instrumentation and controls are provided for safe and efficient operation of the facility. Additional details on instrumentation and controls are included in sections relating to specific systems and components.

<u>CRITERION 13 - FISSION PROCESS MONITORS AND CONTROLS</u>

Means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.

ANSWER

The means provided for monitoring the fission and the means of determining control rod position are described in Section 7 while the means of control and determination of boron concentration are detailed in Section 10.

<u>CRITERION 14 - CORE PROTECTION SYSTEMS</u>

Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

ANSWER

The instrumentation and controls provided to prevent or suppress conditions which could result in exceeding acceptable fuel damage limits are described in Section 7.

CRITERION 15 - ENGINEERED SAFETY FEATURES PROTECTION SYSTEMS

Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

ANSWER

The facility is provided with adequate instrumentation and controls to sense accident situations and initiate the operation of necessary engineered safeguards systems. This protection system is presented in detail in Sections 6, 7, 10 and 11.

CRITERION 16 - MONITORING REACTOR COOLANT PRESSURE BOUNDARY

Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.

ANSWER

Means of detecting leakage from the Reactor Coolant System is provided by measuring the airborne activity and humidity of the lower containment compartment, condensate collected by the fan coil units and indicating changes in makeup requirements and containment sump levels. These leakage detection methods are presented in detail in Section 6.

CRITERION 17 - MONITORING RADIOACTIVITY RELEASES

Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths, and the facility environs for radioactivity that could be released from normal operations, from anticipated transients, and from accident conditions.

ANSWER

The facility contains means for monitoring the containment atmosphere, effluent discharge paths, and the facility environs for radioactivity which could be released under any conditions. The details of the effluent discharge path and containment monitoring methods are contained in Sections 7 and 9 while the Radiological Environmental Monitoring Program is described in Section 2.

CRITERION 18 - MONITORING FUEL AND WASTE STORAGE

Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

ANSWER

Sufficient monitoring and alarm instrumentation is provided in waste and fuel storage areas to detect conditions which might contribute to loss of cooling for decay heat removal or abnormal radiation releases. Details of the monitoring systems are included in Sections 7, 9 and 10.

IV. RELIABILITY AND TESTABILITY OF PROTECTION SYSTEMS

CRITERION 19 - PROTECTION SYSTEMS RELIABILITY

Protection systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed.

ANSWER

All protection systems are designed for the utmost in reliability based on extensive testing in the shop and many years of actual operating experience. Sufficient redundancy of such systems is provided to enable test of instrumentation channels during plant operation without jeopardizing reactor safety. Detailed description of various portions of the systems are included in Section 7.

CRITERION 20 - PROTECTION SYSTEMS REDUNDANCY AND INDEPENDENCE

Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection channel. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served.

ANSWER

As detailed in Section 7, sufficient redundancy and independence is designed into the protection systems to assure that no single failure nor removal from service of any component or channel results in loss of the protection function. In addition, the "Proposed IEEE Criteria for Nuclear Power Plant Protection Systems" of the Institute of Electrical and Electronic Engineer, IEEE No. 279, August 30, 1968, was employed in the detailed design of the protection systems.

CRITERION 21 - SINGLE FAILURE DEFINITION

Multiple failures resulting from a single event shall be treated as a single failure.

ANSWER

When evaluating the control, protection, engineered safeguards and other systems of the facility, multiple failures resulting from a single event are treated as a single failure. The ability of each system to perform its function with a single failure is discussed in the sections describing the individual systems.

A single failure is described as:

> A random failure and its consequential effects, in addition to an initiating occurrence, that results in the loss of capability of a component to perform its intended safety function(s).
>
> Fluid and electrical systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly, nor (2) a single failure of any passive component (assuming active components function properly) results in a loss of the capability of the system to perform its nuclear safety function.
>
> During the short term, the single failure considered may be limited to an active failure. During the long term, assuming no prior failure during the short term, the limiting single failure considered can be either active or passive.
>
> The short term is defined as that period of operation up to 24 hours following an initiating event, but for purposes of design of the emergency core cooling and containment spray systems, the short term shall be considered to terminate upon transfer of these systems to the long term cooling mode. The long term is defined as that period of safety related fluid system operation following the short term, during which the safety function of the system is required.
>
> For electrical systems, no distinction is made between failures of active and passive components and all such failures must be considered in applying the single failure criterion.
>
> Active failure in a fluid system is (1) the failure of a component which relies on mechanical movement for its operation to complete its intended function on demand or (2) an unintentional movement of the component. A passive failure in a fluid system is a breach in the fluid pressure boundary or a mechanical failure which adversely affect a flow path.

CRITERION 22 - SEPARATION OF PROTECTION AND CONTROL INSTRUMENTATION SYSTEM

Protection systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.

ANSWER

Protection and control channels in the facility protection systems were designed in accordance with the IEEE-279, "Proposed IEEE Criteria For Nuclear Power Plant Protection Systems."

The coincident trip philosophy was employed to prevent a single failure from causing a spurious trip or from defeating the function of any channel.

Each reactor trip is designed so that the trip occurs upon deenergization of the circuit; and open circuit or loss of power to a channel will, therefore, result in that channel going into its trip mode. In addition, the reactor protection system will energize the normally de-energized shunt trip device, which in turn trips the reactor trip breaker. Redundancy within each channel provides reliability and independence of operation. Channel independence is carried throughout the system from the sensor to the relay providing the logic. In some cases, however, it is desirable to employ a common sensor for both a control and protection channel. Both functions are fully isolated in the remainder of the channel, control being derived from the primary safety signal path through an isolation amplifier. As such, failure in the control circuitry does not adversely affect the safety channel.

CRITERION 23 - PROTECTION AGAINST MULTIPLE DISABILITY FOR PROTECTION SYSTEMS

The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function.

ANSWER

Protection system components are being designed and arranged so that the mechanical and thermal environment accompanying any emergency situation in which the components are required to function does not interfere with that function. Details of this protection are provided in the appropriate portions of Section 7.

CRITERION 24 - EMERGENCY POWER FOR PROTECTION SYSTEMS

In the event of loss of all offsite power, sufficient alternate sources of power shall be provided to permit the required functioning of the protection systems.

ANSWER

The facility is supplied with normal, reserve and emergency power to provide for the required functioning of the protection systems.

In the event of a reactor and turbine trip, emergency power is supplied by 2 diesel generators per unit, as described in Section 8. Any one diesel is capable of supplying the emergency power requirements for that unit.

The instrumentation and controls portions of the protection systems is supplied from the 125-VDC station batteries during the diesel startup period, as described in Section 8.

CRITERION 25 - DEMONSTRATION OF FUNCTIONAL OPERABILITY OF PROTECTION SYSTEMS

Means shall be included for testing protection systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

ANSWER

Each protection channel in service at power is capable of being calibrated and tested at power to verify its operation. Details of the means used to test protection system instrumentation are included in Section 7.

CRITERION 26 - PROTECTION SYSTEMS FAIL-SAFE DESIGN

The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electrical power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.

ANSWER

The details of the design and failure modes of the various protection channels are found in portions of Section 7 concerned with those channels.

V. REACTIVITY CONTROL

CRITERION 27 - REDUNDANCY OF REACTIVITY CONTROL

At least two independent reactivity control systems, preferably of different principles, shall be provided.

ANSWER

Two independent reactivity control systems, rod cluster control assemblies and boric acid dissolved in the reactor coolant, are employed in the facility.

Details of the construction and operation of the rod cluster control system are included in Sections 3 and 7. Means of controlling the boric acid concentration are included in Section 10.

CRITERION 28 - REACTIVITY HOT SHUTDOWN CAPABILITY

At least two of the reactivity control systems provided shall independently be capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits.

ANSWER

The rod cluster control system is capable of making and holding the core subcritical from all operating and hot shutdown conditions and sufficiently fast to prevent exceeding acceptable fuel damage limits. The chemical shim control is also capable of making and holding the core subcritical, but at a slower rate, and is not employed as a means of compensating for rapid reactivity transients. The rod cluster control system is, therefore, used in protecting the core from such transients. Details of the operation and effectiveness of these systems are included in Sections 3, 7 and 10.

CRITERION 29 - REACTIVITY SHUTDOWN CAPABILITY

At least one of the reactivity control systems provided shall be capable of making the core subcritical under any condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided.

ANSWER

As detailed in Section 3, the reactor may be made subcritical by the rod cluster control system sufficiently fast to prevent exceeding acceptable fuel damage limits, under all anticipated conditions even with the most reactive rod control cluster fully withdrawn.

CRITERION 30 - REACTIVITY HOLDDOWN CAPABILITY

At least one of the reactivity control systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

ANSWER

The facility is provided with the means of making and holding the core subcritical under any anticipated conditions and with appropriate margin for contingencies. These means are discussed in detail in Sections 3, 7 and 10. Combined use of the rod cluster control system and the chemical shim control system permit the necessary shutdown margin to be maintained during long term xenon decay and plant cooldown.

CRITERION 31 - REACTIVITY CONTROL SYSTEMS MALFUNCTION

The reactivity control systems shall be capable of sustaining any single malfunction, such as, unplanned continuous withdrawal (not ejection) of a control rod, without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

ANSWER

The facility reactivity control systems are such that acceptable fuel damage limits are not exceeded even in the event of a single malfunction of either system. An analysis of the effects of possible malfunction is presented in Chapters 3, 7 and 14.

CRITERION 32 - MAXIMUM REACTIVITY WORTH OF CONTROL RODS

Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

ANSWER

The maximum reactivity worth of control rods and the maximum rates of reactivity insertion employing both control rods and boron removal are limited to values which prevent rupture of the coolant pressure boundary or disrupt the core or vessel internals to a degree which could impair the effectiveness of emergency core cooling. Details of rod worths, reactivity insertion rates and their relationship to plant safety are included in Sections 3 and 14.

VI. REACTOR COOLANT PRESSURE BOUNDARY

CRITERION 33 - REACTOR COOLANT PRESSURE BOUNDARY CAPABILITY

The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

ANSWER

The reactor coolant boundary is designed to accommodate static and dynamic loads associated with sudden reactivity insertions (e.g., rod ejection) without failure. Details of the design can be found in Sections 3 and 4 and an analysis of the effects of such incidents as rod ejection is included in Section 14.

CRITERION 34 - REACTOR COOLANT PRESSURE BOUNDARY RAPID PROPAGATION FAILURE PREVENTION

The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given (a) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

ANSWER

The reactor coolant pressure boundary is designed to minimize the probability of rapidly propagating type failures. To fulfill these requirements, the selection of materials for the systems and the fabrication of components are closely controlled and inspected. The details of the material selection and inspection procedures are contained in Section 4.

CRITERION 35 - REACTOR COOLANT PRESSURE BOUNDARY BRITTLE FRACTURE PREVENTION

Under conditions where reactor coolant pressure boundary system components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperature shall be at least 120°F above the nil ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation or 60°F above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

ANSWER

Sufficient testing and analysis of materials employed in reactor coolant system components was performed to insure that the required NDT limits specified in the criterion are met. Removable test capsules are installed in the reactor vessel and are removed and tested at various times in the plant lifetime to determine the effects of operation on system materials. Details of the testing and analysis programs are included in Section 4.

CRITERION 36 - REACTOR COOLANT PRESSURE BOUNDARY SURVEILLANCE

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leaktight integrity of the boundary components during service lifetime. For the reactor vessel, a material surveillance program conforming with ASTM-E-185-66 shall be provided.

ANSWER

Provision has been made in the reactor coolant system design for adequate inspection testing and surveillance during the facility's service lifetime. The reactor coolant system inservice inspection program is discussed in Section 4.7. The vessel material surveillance inspection program conforms to ASTM-E-185. These provisions are also discussed in detail in Section 4.

VII. ENGINEERED SAFETY FEATURES

CRITERION 37 - ENGINEERED SAFETY FEATURES BASIS FOR DESIGN

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

ANSWER

The containment systems, containment air cooling system, the safety injection system, the special zone ventilation systems, the containment vessel internal spray system, the auxiliary feedwater system and the diesel generators comprise the engineered safety features for the facility. These systems and their supporting systems (component cooling system and cooling water system) are designed to cope with any size reactor coolant pressure boundary break up to and including rupture of the largest reactor coolant pipe. The design bases for each system are included in the appropriate portions of Sections 5, 6, 8 and 10. An analysis of the performance of the safeguards is presented in Section 14. 00049

CRITERION 38 - RELIABILITY AND TESTABILITY OF ENGINEERED SAFETY FEATURES

All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

ANSWER

All engineered safety features components were tested in the manufacturers shop and after installation at the facility to demonstrate their reliability. Provision has also been made in the system design for periodic testing of engineered safety features during the plant lifetime. Details of the tests to be performed and the basis for the determination of system reliability are included in Section 5 for the containment and containment isolation system, and in Sections 6, 8 and 10 for the remaining engineered safety features.

CRITERION 39 - EMERGENCY POWER FOR ENGINEERED SAFETY FEATURES

Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the onsite power system and the offsite power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.

ANSWER

Reliability of electric power supply is insured through two independent connections to the system grid, and a redundant source of emergency power from four diesel generators installed in the facility. Power to the engineered safety features is assured even with the failure of a single active component in each system. The facility electrical systems, including network interconnections and the emergency power system, are described in Section 8.

CRITERION 40 - MISSILE PROTECTION

Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failure.

ANSWER

All engineered safety features are protected against dynamic effects and missiles resulting from equipment failures. The means for accomplishing this protection are described in Sections 5, 6 and 12.

CRITERION 41 - ENGINEERED SAFETY FEATURES PERFORMANCE CAPABILITY

Engineered safety features such as emergency core cooling and containment heat removal systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety features shall provide this required safety function assuming a failure of a single active component.

ANSWER

Sufficient redundancy and duplication is incorporated into the design of the engineered safety features to insure that they may perform their function adequately even with the loss of a single active component. Details of the capability of these systems under normal and component malfunction conditions are included in Section 6 and 10. An analysis of the adequacy of these systems to perform their functions is included in Section 14.

CRITERION 42 - ENGINEERED SAFETY FEATURES COMPONENTS CAPABILITY

Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident.

ANSWER

The design of the engineered safety features, the materials selected for fabrication of these systems, and the layout of the various portions of the systems combine to insure that the performance of the engineered safety features is not impaired by the effects of a loss-of-coolant accident. Details of the design and construction of the engineered safety features are included in Sections 5, 6, 8 and 10. The ability of these features to perform their functions is analyzed in Section 14.

CRITERION 43 - ACCIDENT AGGRAVATION PREVENTION

Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse after-effects of the loss of normal cooling is avoided.

ANSWER

The operation of the engineered safety features will not accentuate the after effects of a loss-of-coolant accident. These considerations are detailed in Sections 5, 6, 8, 10 and 14.

CRITERION 44 - EMERGENCY CORE COOLING SYSTEMS CAPABILITY

At least two emergency core cooling systems, preferably of different design principles, each with a capability of accomplishing abundant emergency core cooling, shall be provided. Each emergency core cooling system and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each emergency core cooling system shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that (a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, (b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and (c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident.

ANSWER

By combining the use of passive accumulators with two independent high pressure pumping systems and two independent low pressure pumping systems abundant emergency core cooling is provided even if there should be a failure of any component in any system. A description of the system and its operation is contained in Section 6 and an analysis of the operation of the system under accident conditions is included in Section 14.

CRITERION 45 - INSPECTION OF EMERGENCY CORE COOLING SYSTEMS

Design provisions shall be made to facilitate physical inspection of all critical parts of the emergency core cooling systems, including reactor vessel internals and water injection nozzles.

ANSWER

The design of the emergency core cooling system is such that critical portions are accessible for examination by visual, optical or other nondestructive means. Details of the inspection program for the reactor vessel internals are included in Section 4 while inspection of the remaining portions of the system is discussed in Section 6.

CRITERION 46 - TESTING OF EMERGENCY CORE COOLING SYSTEM COMPONENTS

Design provisions shall be made so that active components of the emergency core cooling systems, such as pumps and valves, can be tested periodically for operability and required functional performance.

ANSWER

The emergency core cooling system design permits periodic testing of active components for operability and required functional performance. The test procedures are described in Section 6.

CRITERION 47 - TESTING OF EMERGENCY CORE COOLING SYSTEMS

A capability shall be provided to test periodically the delivery capability of the emergency core cooling systems at a location as close to the core as is practical.

ANSWER

By recirculation to the refueling water storage tank, the emergency core cooling system delivery capability can be tested periodically. The system can be so tested to the last valve before the piping enters the reactor coolant piping. Details of the system tests are included in Section 6.

CRITERION 48 - TESTING OF OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEMS

A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the emergency core cooling systems into action, including the transfer to alternate power sources.

ANSWER

Provision has been made in the emergency core cooling system design for testing the sequence of operation including transfer to alternate power sources. The details of these tests are included in Section 6, and the switching sequence from normal to emergency power is described in Section 8.

CRITERION 49 - CONTAINMENT DESIGN BASIS

The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions, that could occur as a consequence of failure of emergency core cooling systems.

ANSWER

The reactor containment vessel and its contained heat removal system are designed to accommodate the pressures and temperatures associated with a loss of coolant accident without exceeding the design leak rate. A considerable margin for unidentified energy sources has been included in the design. The loadings and energy sources considered in the design and the stress and loading criteria are described in Section 12. An analysis of the performance of the containment during a loss-of-coolant accident is included in Section 14. The heat removal systems are described in Section 6 (Containment Vessel Internal Spray System and Containment Air Cooling System). Design of the concrete shield building is given in Section 12.

CRITERION 50 - NDT REQUIREMENTS FOR CONTAINMENT VESSEL

Principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures under normal operating and testing conditions are not less than 30°F above nil ductility transition (NDT) temperature.

ANSWER

As stated in Section 5, all containment ferritic materials are selected to ensure that their temperature under normal operating and testing conditions will be at least 30°F above nil ductility transition temperature.

CRITERION 51 - REACTOR COOLANT PRESSURE BOUNDARY OUTSIDE CONTAINMENT

If part of the reactor coolant pressure boundary is outside the containment, appropriate features as necessary shall be provided to protect the health and safety of the public in case of an accidental rupture in that part. Determination of the appropriateness of features such as isolation valves and additional containment shall include consideration of the environmental and population conditions surrounding the site.

ANSWER

The reactor coolant pressure boundary is defined as those piping systems and components which contain reactor coolant at design pressure and temperature. With the exception of the reactor coolant sampling lines, the entire reactor coolant pressure boundary, as defined above, is located entirely within the reactor containment vessel. All sampling lines are provided with remotely operated valves for isolation in the event of a failure. These valves also close automatically on a containment isolation signal. Sampling lines are only used during infrequent sampling and can be readily isolated.

All other piping and components which may contain reactor coolant are low pressure, low temperature systems which would yield minimal environmental doses in the event of failure.

The Sampling System and low pressure systems are described in Section 10.

CRITERION 52 - CONTAINMENT HEAT REMOVAL SYSTEM

Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.

ANSWER

Heat capability for the Containment is provided by two separate, engineered safety features systems. These are the Containment Vessel Internal Spray System, whose components are described in Section 6.4 and the Containment Air Cooling System whose components operate as described in Section 6.3. 00049

CRITERION 53 - CONTAINMENT ISOLATION VALVES

Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.

ANSWER

At least two barriers are provided between the atmosphere outside the containment and the containment atmosphere, the reactor coolant system, or closed systems which are assumed vulnerable to accident forces. The valving installed on the various systems penetrating the containment and the other barriers employed in the design are described in Sections 5, 6 and 10.

CRITERION 54 - CONTAINMENT LEAKAGE RATE TESTING

Containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period of time to verify its conformance with required performance.

ANSWER

Provision is included in the containment vessel design for integrated leak rate testing after completion of construction. The test procedure is described in Section 5 and is formulated to demonstrate that leakage is below the Technical Specification limits.

CRITERION 55 - CONTAINMENT PERIODIC LEAKAGE RATE TESTING

The containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime.

ANSWER

Provision for full integrated leak rate testing of the containment is incorporated in the design. The testing procedures are discussed in Section 5.

CRITERION 56 - PROVISIONS FOR TESTING OF PENETRATIONS

Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leaktightness to be demonstrated at design pressure at any time.

ANSWER

Each containment penetration includes a means to test its leaktightness at any time. This system is described in Section 5.

CRITERION 57 - PROVISIONS FOR TESTING OF ISOLATION VALVES

Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.

ANSWER

The containment isolation system, including test provisions, is described in Section 5.

CRITERION 58 - INSPECTION OF CONTAINMENT PRESSURE-REDUCING SYSTEMS

Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as, pumps, valves, spray nozzles, torus, and sumps.

ANSWER

The design of the Containment Vessel Internal Spray Systems includes provision for physical inspection of vital components. The inspectability of the spray systems is discussed in Section 6.

CRITERION 59 - TESTING OF CONTAINMENT PRESSURE-REDUCING SYSTEMS COMPONENTS

The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves, can be tested periodically for operability and required functional performance.

ANSWER

Component testing of the Containment Vessel Internal Spray Systems is discussed in detail in Section 6.

CRITERION 60 - TESTING OF CONTAINMENT SPRAY SYSTEMS

A capability shall be provided to test periodically the delivery capability of the containment spray system at a position as close to the spray nozzles as is practical.

ANSWER

All portions of the Containment Vessel Internal Spray Systems may be tested. The delivery capacity may be tested up to the last valve before the system enters the containment. Details of the Containment Vessel Internal Spray System are included in Section 6.

CRITERION 61 - TESTING OF OPERATIONAL SEQUENCE OF CONTAINMENT PRESSURE-REDUCING SYSTEMS

A capability shall be provided to test under conditions as close to the design as practical the full operational sequence that would bring the containment pressure-reducing systems into action, including the transfer to alternate power sources.

ANSWER

Capability for testing of the operational sequence of the Containment Vessel Internal Spray System is incorporated into the system design. Details of the Containment Vessel Internal Spray System are included in Section 6. The switching sequence from normal to emergency power is described in Section 6.

CRITERION 62 - INSPECTION OF AIR CLEANUP SYSTEMS

Design provisions shall be made to facilitate physical inspection of all critical parts of the containment air cleanup systems, such as ducts, filters, fans and dampers.

ANSWER

The inspection of the special zone ventilation systems and their components is discussed in Section 10.

CRITERION 63 - TESTING OF AIR CLEANUP SYSTEMS COMPONENTS

Design provisions shall be made so that active components of the air cleanup systems, such as fans and damper, can be tested periodically for operability and required functional performance.

ANSWER

Testing of special zone ventilation system components is discussed in Section 10.

CRITERION 64 - TESTING OF AIR CLEANUP SYSTEMS

A capability shall be provided for in situ periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed and (b) filter and trapping materials have not deteriorated beyond acceptable limits.

ANSWER

In situ testing of the special zone ventilation system is discussed in Section 10.

CRITERION 65 - TESTING OF OPERATIONAL SEQUENCE OF AIR CLEANUP SYSTEMS

A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the designing air flow delivery capability.

ANSWER

The operational sequence testing of the special zone ventilation system is discussed in Section 10.

CRITERION 66 - PREVENTION OF FUEL STORAGE CRITICALITY

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

ANSWER

Criticality in new and spent fuel storage areas is prevented both by physical separation of new and spent fuel elements and the presence of borated water in the spent fuel storage pit. Criticality prevention is discussed in detail in Section 10.

CRITERION 67 - FUEL AND WASTE STORAGE DECAY HEAT

Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

ANSWER

The Spent Fuel Pool Cooling system provides decay heat removal for the spent fuel pool. The system is capable of handling a maximum heat load corresponding to both pools being filled with a combined total of 1362 normally discharged fuel assemblies plus a freshly off loaded core consisting of 121 fuel assemblies. Details of the Spent Fuel Pool Cooling System and fuel handling facilities are described in Section 10.

CRITERION 68 - FUEL AND WASTE STORAGE RADIATION SHIELDING

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities as required to meet the requirements of 10CFR20.

ANSWER

Shielding is provided for fuel handling and waste storage areas to lower radiation doses to levels below limits specified in 10CFR20. Shielding for these areas and other plant shielding requirements and criteria are included in Sections 9 and 12.

CRITERION 69 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

ANSWER

All fuel storage and waste storage facilities are designed to prevent the release of undue radioactivity to the public. Fuel storage facilities are described in Section 10, waste storage facilities are described in Sections 9 and 12 and analysis of potential accidents in these systems is included in Section 14.

CRITERION 70 - CONTROL OF RELEASES OF RADIOACTIVITY TO THE ENVIRONMENT

The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10CFR20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10CFR100 dosage level guidelines for potential reactor accidents of exceeding low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities of very large cities can be affected by the radioactive effluents.

ANSWER

Provision is included in the facility design for storage and processing of radioactive waste and the release of such wastes under controls adequate to prevent exceeding the limits of 10CFR20. The facility also includes provision to prevent radioactivity releases during accidents from exceeding the guidelines of 10CFR100. A description of the Radioactive Waste Disposal System is included in Section 9. The effects of potential accidents, including a loss-of-coolant accident, are analyzed in Section 14.

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TABLE 1.3-1
Auxiliary, Emergency, and Waste Disposal System
Shared Systems

System	Section Reference	Engineered Safeguard Related
CVCS Boron Makeup and Recovery Subsystem	10.2.3	Yes
Component Cooling System	10.4.2	Yes
Spent Fuel Pool Cleanup and Cooling System	10.2.2	No
Fuel Handling System	10.2.1	No
Cooling Water System	10.4.1	Yes
Radioactive Waste Control System	9.1	No
Auxiliary Building Special Ventilation System	10.3.4	Yes
Fire Protection System for Other Than Class I Areas	10.3.1	No
Condensate Polishing System	11.8	No
Circulating Water System	11.5	No
Station Air System	10.3.10	No
Control Room Air Conditioning System	10.3.3	Yes
Steam Exclusion System	App. I	No
Safeguards Chilled Water System	10.4.3	No

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TABLE 1.3-2 SHARED COMPONENTS REQUIRED FOR NORMAL PLANT OPERATION

System	Components Shared	Function	Quantity Provided	Explanation	Serves Shutdown Function	Serves Emergency Function	Emergency (and Shutdown where Associated) Conditions Which Make the Maximum Demands on the System	Quantity Required to Meet the Maximum Demand	Ability to Tolerate Under Emergency Conditions Either Maintenance of a Single Item of Equipment or Failure of one Active Component
Chemical and Volume Control System	Boric Acid Tanks	Storage of boric acid for refueling and emergency shutdown	3	Three tanks are provided with one tank aligned to each unit and the third tank as a spare. Each tank has sufficient boric acid solution to achieve cold shutdown for that unit. These tanks also supply the suction of the safety injection pumps for emergency conditions.	Yes (See Note 1)	Yes (See Note 1)	Simultaneous shutdown of both units.	1	Yes (See Note 1)
	Batching Tank	Makeup of fresh concentrated boric acid solution.	1	One tank is provided for the two units.	No	No	N/A (See Note 2)	N/A	N/A
	Hold-up Tanks	Storage of dilute boric acid prior to recycle processing.	3	Three tanks are provided to handle the rejected chemical shim solution from all expected operating and start-up transients for two unit plant operation	No	No	N/A	N/A	N/A
	Recirculation Pump	Handling of tank inventory	1	Serves the common hold-up tanks.	No	No	N/A	N/A	N/A

TABLE 1.3-2 SHARED COMPONENTS REQUIRED FOR NORMAL PLANT OPERATION

System	Components Shared	Function	Quantity Provided	Explanation	Serves Shutdown Function	Serves Emergency Function	Emergency (and Shutdown where Associated) Conditions Which Make the Maximum Demands on the System	Quantity Required to Meet the Maximum Demand	Ability to Tolerate Under Emergency Conditions Either Maintenance of a Single Item of Equipment or Failure of one Active Component
	Gas Stripper Feed Pumps	Pumping of chemical shim solution to be processed using ion exchangers and filtration or evaporation.	3	Three pumps are provided, each with sufficient capacity to supply water for processing.	No	No	N/A	N/A	N/A
	Evaporator Feed Ion Exchanger	Remove significant contaminants from the process stream.	4	Cation and anion demineralizers are operated as necessary to achieve the desired contaminant removal efficiencies.	No	No	N/A	N/A	N/A
	Gas Stripper Boric Acid Evaporator Packages	When operating, process used chemical shim solution to produce concentrated boric solution and distillate for reuse or release.	2	Two processing packages serve as common equipment for the two units. When in service, each package will normally be operated separately. Capability exists for processing either unit with one package.	No	No	N/A	N/A	N/A

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TABLE 1.3-2 SHARED COMPONENTS REQUIRED FOR NORMAL PLANT OPERATION

System	Components Shared	Function	Quantity Provided	Explanation	Serves Shutdown Function	Serves Emergency Function	Emergency (and Shutdown where Associated) Conditions Which Make the Maximum Demands on the System	Quantity Required to Meet the Maximum Demand	Ability to Tolerate Under Emergency Conditions Either Maintenance of a Single Item of Equipment or Failure of one Active Component
	Monitor Tanks	Reservoirs for processed water for analysis prior to release or reuse.	3	Three monitor tanks are provided to collect the water processed from the CVCS Holdup Tanks. Each tank is sized to hold the condensate produced by one evaporator in approximately eight hours.	No	No	N/A	N/A	N/A
	Monitor Tank Pumps	Pump water from the monitor tanks to the river or for reuse.	2	Two pumps are provided with adequate capacity to handle both units. One pump serves as a spare to the other.	No	No	N/A	N/A	N/A
	Evaporator Condensate Demineralizers	Remove impurities from processed water.	2	Two demineralizers are provided, each with sufficient capacity to serve both units. One resin bed serves as a spare to the other.	No	No	N/A	N/A	N/A
	Reactor Makeup Water Storage Tank	Storage of clean makeup water	4	Four tanks are provided, each adequately sized to serve one unit.	No	No	N/A	2	N/A
	Reactor Makeup Pumps	Supply Miscellaneous reactor makeup	4	Two pumps are provided for each unit, each with sufficient capacity to serve needs of one unit. The other two pumps serve as backups to the first two.	No	No	N/A	2	Yes

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TABLE 1.3-2 SHARED COMPONENTS REQUIRED FOR NORMAL PLANT OPERATION

System	Components Shared	Function	Quantity Provided	Explanation	Serves Shutdown Function	Serves Emergency Function	Emergency (and Shutdown where Associated) Conditions Which Make the Maximum Demands on the System	Quantity Required to Meet the Maximum Demand	Ability to Tolerate Under Emergency Conditions Either Maintenance of a Single Item of Equipment or Failure of one Active Component
Auxiliary Coolant system	Component Cooling Heat Exchangers	Intermediate heat exchanger between cooling water and component cooling water.	4	Four exchangers are provided to serve both units. Except to speed cooldown, only one exchanger is required per unit. Normally, each units' component cooling system operates independently although provision to cross-tie is made.	Yes	Yes	The recirculation phase of the post LOCA condition in one unit with a simultaneous hot shutdown condition in the second unit.	2	Yes
	Component Cooling Water Pumps	Circulate component cooling water for miscellaneous services in both units.	4	Four pumps are provided. One pump will provide adequate circulation to cool each unit. One additional pump is provided for each unit to serve as a spare.	Yes	Yes	The recirculation phase of the post LOCA condition in one unit with a simultaneous hot shutdown condition in the second unit.	2	Yes
	Component Cooling Surge Tanks	Surge and head tanks for component cooling water loop.	2	One tank is provided for each unit. These tanks can be isolated if required.	No	No	N/A	N/A	N/A

TABLE 1.3-2 SHARED COMPONENTS REQUIRED FOR NORMAL PLANT OPERATION

System	Components Shared	Function	Quantity Provided	Explanation	Serves Shutdown Function	Serves Emergency Function	Emergency (and Shutdown where Associated) Conditions Which Make the Maximum Demands on the System	Quantity Required to Meet the Maximum Demand	Ability to Tolerate Under Emergency Conditions Either Maintenance of a Single Item of Equipment or Failure of one Active Component
Waste Disposal	Each containment structure has its own reactor coolant drain tank with 2 pumps, and containment sump. All other waste disposal equipment is in the common auxiliary and services buildings. This shared equipment includes:								
	Laundry and Hot Shower Tank, Chemical Drain Tank, Sump Tank, Waste Holdup Tank, Gas Decay Tanks, Waste Condensate Tanks, Waste Condensate Pumps, Waste Gas Compressor, Waste Evaporator Train, Drumming Station, Baling Station, Gas manifolds, Gas Analyzer, and Decontamination Area		1		Yes	No	N/A	N/A	N/A
Cooling Water System	Screen House and Headers	Environment for Cooling Water Pumping Equipment	1	A common screen house is provided for the two units	Yes	Yes	The recirculation phase of the post LOCA condition in one unit with a simultaneous hot shutdown condition in the second unit.	See Cooling Water Pumps	Yes

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TABLE 1.3-2 SHARED COMPONENTS REQUIRED FOR NORMAL PLANT OPERATION

System	Components Shared	Function	Quantity Provided	Explanation	Serves Shutdown Function	Serves Emergency Function	Emergency (and Shutdown where Associated) Conditions Which Make the Maximum Demands on the System	Quantity Required to Meet the Maximum Demand	Ability to Tolerate Under Emergency Conditions Either Maintenance of a Single Item of Equipment or Failure of one Active Component
	Cooling Water Pumps	Provide cooling water for common component cooling loop, the containment ventilation cooling fans, and miscellaneous loads in the Turbine and Auxiliary Building.	5	Five cooling water pumps, two direct diesel engine driven and three electric motor driven, are provided to supply water to the dual, common loop piped system for the two units. Normally, two motor driven pumps will supply both units; the additional pumps provide increased capacity when required. In the loss of auxiliary A.C. case, the diesel engine driven pumps and/or the vertical electric motor driven pump connected to the diesel generator are the source of cooling water	Yes	Yes	The recirculation phase of the post LOCA condition in one unit with a simultaneous hot shutdown condition in the second unit	1	Yes
Fire Protection	The Fire Protection System, utilizes water spray, cardox, halon, hose lines and sprinklers which are actuated by fusible heads, rate of rise detectors, thermal detectors, smoke detectors or ionization detectors to combat fire. Portable extinguishers are also provided extensively throughout the plant facilities. This system is designed to extinguish any probable combination of simultaneous fires which might occur at the station. The shared equipment includes: Fire Pumps (121 and 122) Jockey Pump Sprinkler System Screen Wash Pump				No	N/A	N/A	N/A	N/A
Aerated Drains Treatment (Liquid Waste Disposal)	The Aerated Drains Treatment system receives radioactive, aerated, liquid waste and treats it so that it can be returned to the plant as make-up water or be discharged to the river. The shared equipment includes:				N/A	N/A	N/A	N/A	N/A

TABLE 1.3-2 SHARED COMPONENTS REQUIRED FOR NORMAL PLANT OPERATION

System	Components Shared / Function / Quantity Provided / Explanation	Serves Shutdown Function	Serves Emergency Function	Emergency (and Shutdown where Associated) Conditions Which Make the Maximum Demands on the System	Quantity Required to Meet the Maximum Demand	Ability to Tolerate Under Emergency Conditions Either Maintenance of a Single Item of Equipment or Failure of one Active Component
	ADT Collection Tanks and Pumps ADT Condensate Receiver Tanks and Pumps ADT Monitor Tanks and Pumps ADT Miscellaneous Drains Collection Tank and Pump ADT Ion Exchangers ADT Sump Tank and Pump Cask Wash Down Area Sump Pump ADT Filters ADT Evaporators					
Condensate Polishing	A condensate polishing system is provided for both units. Each unit has its own filter/demineralizers. The backwash and flush water subsystem is cross-tied, and the backwash air supply , spent resin disposal, and resin disposal building (RDB) sump equipment is sized to service both units. The shared equipment includes: Backwash Waste Clamshell Filters Backwash Air Compressors and Receivers Spent Resin Transfer Tank Spent Resin Transfer Pump RDB and Truck Area Sumps	N/A	N/A	N/A	N/A	N/A
Station Air (Instrument and Service Air)	A common station air system supplies the instrument and service air requirements for both units. Each unit has its own instrument air dryer and instrument air header. The IA headers are cross-tied. The shared equipment includes: Air Compressors Moisture Separators Aftercoolers Air Receivers	N/A	N/A	N/A	N/A	N/A

TABLE 1.3-2 SHARED COMPONENTS REQUIRED FOR NORMAL PLANT OPERATION

System	Components Shared	Function	Quantity Provided	Explanation	Serves Shutdown Function	Serves Emergency Function	Emergency (and Shutdown where Associated) Conditions Which Make the Maximum Demands on the System	Quantity Required to Meet the Maximum Demand	Ability to Tolerate Under Emergency Conditions Either Maintenance of a Single Item of Equipment or Failure of one Active Component
Safeguard Chilled Water System	Chillers	Supply chilled water to unit coolers in safety related compartments for localized heat removal	2		Yes	Yes	LOOP with concurrent SBO in other unit	1 Train	Yes

Notes for Table 1.3-2

(1) Boric acid injection affords back up reactivity shutdown capability, independent of control rod clusters which normally serve this function in the short term situation.

(2) N/A Not Applicable, i.e., Serves No Emergency Function.

TABLE 1.3-3 SHARED COMPONENTS NOT REQUIRED FOR NORMAL PLANT OPERATION

System	Components Shared	Function	Quantity Provided	Explanation	Serves Shutdown Function	Serves Emergency Function	Emergency (and Shutdown where Associated) Conditions Which Make the Maximum Demands on the System	Quantity Required to Meet the Maximum Demand	Ability to Tolerate Under Emergency Conditions Either Maintenance of a Single Item of Equipment or Failure of one Active Component	
Spent Fuel Pool Cooling	Spent Fuel Pool Demineralizer	Purification of the spent fuel pool water and refueling water	1	One demineralizer is provided.	Yes See Note (1)	No	N/A	N/A	N/A	98119
	Spent Fuel Pool Filter	Purification of the spent fuel pool water and refueling water.	3	Three filters are provided.	Yes See Note (1)	No	N/A	N/A	N/A	98119
	Spent Fuel Pool Heat Exchanger	Cooling Spent Fuel Pool Water	2	One heat exchanger has sufficient capacity to maintain reasonable pool temperatures when handling the design basis normal heat load See Note (2).	Yes See Note (1)	No	See Note (2)	See Note (3)	See Note (4)	98119

2.6 SEISMOLOGY

2.6.1 General

A seismological investigation of the site was performed by Dames and Moore. The seismological program consisted of:

a. An evaluation of the seismicity of the area.

b. A study of geologic structure as related to earthquake activity.

c. The postulation of "operational" and "design" earthquake accelerations, and the preparation of recommended response spectra.

d. Field and laboratory measurements of the dynamic response characteristics of the soil and rock strata underlying the site.

The results of the seismological program are reported in Appendix E.

The State of Minnesota has experienced only a few moderate earthquake shocks in the relatively short period since 1860 during which earthquakes have been recorded in the State. A tabulation of earthquakes having epicenters in Minnesota, together with certain out-of-state earthquakes felt in Minnesota, is presented in Table 2.6-1 and in Figure 2.6-1. Earthquake intensities are described in terms of the Modified Mercalli Intensity Scale of 1931, which is explained in Table 2.6-2.

Based on the seismic history and the regional tectonics, it is anticipated that the site will not experience any significant earthquake motion during the economic life of the nuclear facility. Historically, there is no basis for expecting ground motion of more than a few percent of gravity. However, for conservatism, the plant is designed to respond elastically to earthquake ground motion as high as 6 percent gravity, with no loss of function. Provisions have also been made for safe shutdown of the reactor if ground motions reach as high as 12 percent of gravity in the overburden soils at the site. In the event of an earthquake, plant operating procedures identify the action thresholds for plant shutdown and post event physical inspection of the facility.

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Because of the possibility of liquefaction which may occur during a design basis earthquake, all critical structures at elevation 645 or higher have been supported on densified sand. All foundations are within the sand above the bedrock.

The design of the structures and their foundations took into account the dynamic effects of earthquake motion. Consideration was given in the design to maximum expected ground motions, response spectra, and elastic moduli and damping values of the various soil and rock. Seismic design criteria are provided in Section 12.

2.6.2 Seismic History

Southeastern Minnesota is considered one of the least active seismic zones of the United States. King's distribution of epicenters contours the area as having less than one epicenter per 10,000 sq. km, the "least active" classification (Ref. 26). However, earthquakes are not unknown in Minnesota. At least six (1860, 1865-70, 1917, 1928, 1939, and 1950) have had local origins, and certain others, with epicenters outside the state, have been felt within the borders of Minnesota. These events are discussed in Appendix E, Section 4. There has been no seismic activity of any consequence in recent years in the vicinity of the plant.

2.6.3 Recent Seismic History

An extensive seismic monitoring system is installed at Prairie Island. Only one seismic event has been recorded at the plant. An earthquake triggered the seismic alarm about 0650 on June 10, 1987. The seismic acceleration was measured at about 0.01 g, the lower limit of detectability for the installed instrumentation. The quake was centered in southeastern Illinois and caused tremors in fifteen states and Canada. The quake was not detectable at the Monticello Nuclear Generating Plant.

2.8 ECOLOGICAL AND BIOLOGICAL STUDIES

On January 19, 1981, the Minnesota Pollution Control Agency, the permitting agency under the U. S. Environmental Protection Agency, issued the National Pollution Discharge Elimination System (NPDES) Permit No. MN0004006 [Ref. 30] covering the Prairie Island Nuclear Generating Plant. This permit is reissued with any modifications every 5 years. The NPDES effluent limitations and monitoring requirements, thermal studies and ecological monitoring requirements provide appropriate protection for the environment. There are no ecological or biological monitoring requirements under NRC jurisdiction. Pre-operational and early operational ecological and biological studies are described in the FSAR. 00252

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2.9 CONSEQUENCES OF HYPOTHETICAL LOCAL CATASTROPHES

2.9.1 Effects of Oil Spillage

The plant is fully protected from possible effects of oil spillage on the river by the design of the intake screenhouse and earthen dikes which prevent floating oil from entering the plant. The only water intake that bypasses these barriers is the emergency intake to the cooling water system and this intake is located in a crib at the bottom of the river branch beyond the intake canal.

In addition, the suction intakes for the circulating water pumps, the cooling water pumps, and the fire pumps are submerged in bays within the screenhouse.

Another purpose of these barriers is to prevent uncontrolled return of hot surface water to the river when the plant is operated with cooling towers, the return flow from which is by a return canal that merges with the plant intake canal.

In conclusion, the operation of the circulating water system, the engineered safety feature cooling water system, and other systems is unaffected by the occurrence of an oil spill.

2.9.2 Postulated Explosion of Munitions Barge

In the absence of reported bulk shipments of munitions or other explosives this far north on the Mississippi River, the question of a barge explosion can be relevant only in the sense that such cargo is not restricted.

The size and nature of a hypothetical cargo that might be postulated to explode is entirely speculative; therefore it is assumed conservatively that a jumbo barge (195 ft. length, 35 ft. width, 8-1/2 ft. draft), the largest hauler of dry cargo, is completely laden with 1400 tons of TNT, and that this cargo detonates in mid-channel directly opposite the plant. The resulting 1.4 kiloton explosion would be comparable to the Texas City disaster (April 1947) which resulted from detonation of 2 to 4 kilotons of equivalent explosive.

A mid-channel location opposite the plant would be along the Minnesota-Wisconsin line, which is a minimum of 2600 feet distance from the control room, at a point nearly due east from it. An overestimate of the blast effect is given by assuming the occurrence of a surface blast at this point.

Surface detonation of 1.4 kilotons would result in a peak overpressure of 2-1/4 psi at 2600 feet distance, plus a minor dynamic pressure due to a 78 mph transient wind, as determined directly by use of "The Nuclear Bomb Effects Computer" (Ref. 12). The overpressure would actually be substantially less because of the attenuation and vertical blast deflection that would occur due to the fact that much of the full cargo would necessarily be located below the waterline.

The effects of this pressure on either of the shield buildings can be scaled conservatively from the results of previous calculations of tornado loading. The tornado-induced stresses were based on a 1.56 psi frontal overpressure with resultant areas of depression up to minus 3 psi on the sides of the structure. The maximum local tensile stress in any steel reinforcement member was determined to be 50,000 psi, relative to a minimum yield strength of 60,000 psi and a minimum ultimate strength of 90,000 psi. This occurred for certain more highly loaded members at mid-elevation in the structure. The concrete was nowhere compressively stressed to more than about one-half its compressive strength.

The calculated frontal overpressure of the blast, and the overall blast loading, will momentarily be 1.44 times as great as for the tornado, and should correspondingly cause a maximum local tensile stress of 72,000 psi. This is between the yield strength and the ultimate tensile strength, indicating that some local deformation and concrete cracking could possibly occur for sustained loading of such magnitude, depending on the extent to which the actual yield strength of the affected members exceeded the minimum specified value, but that no extensive structural failure and no structural collapse would occur.

Both the yield strength of the steel and the compressive strength of the concrete will actually be much greater for the pulse loading of the blast, which falls nearly linearly to zero from the initial overpressure over a time duration of 0.5 seconds, as determined by the referenced computer. It can be inferred from Figures 6-2 and 6-5 of the Air Force Design Manual (Ref. 13) that the short-term minimum yield strength would not be exceeded under these conditions. Considering also the attenuation effects due to partial submergence of the explosion source at the time of detonation, it may be concluded that there would actually be no local deformation whatever.

In any case, the free-standing containment vessel within the shield building would be unaffected, as would the components of the reactor system within the containment.

The control room should readily survive the postulated blast without injury to its occupants. The entire room is enclosed with two-foot thick concrete walls, except for the north wall which is 18 inches thick, and it is surrounded by other structures. Conservative application of the linear and rotational components of tornado velocities for those areas of the structure that would be exposed to the blast has effectively resulted in design for a 2-1/4 psi internal loading, plus allowance for missiles and earthquakes. The reinforcement in the structure is symmetrical and it can be concluded that the design is also adequate for such pressure loading applied externally.

Similarly, it can be concluded that the massive structure of the spent fuel storage area would be unaffected.

Damage may be expected to occur to light external structures that are exposed to the blast. In particular, the metal siding and roof decking of certain structures such as the turbine building would be blown off, consistent with the intent of their design with regard to tornado forces. Extensive minor damage would be expected to occur throughout the plant and switchyard.

Despite such superficial and repairable damage to the plant from an occurrence for which it was not specifically designed, no reduction in effectiveness would be expected for either the containment system or the engineered safety features that are provided to respond to a nuclear accident. The only realistic concern is that the blast could cause a nuclear accident or incapacitate the operators before they can accomplish an orderly shutdown.

No consequence of the postulated explosion is foreseen that would either initiate a nuclear accident or prevent safe shutdown of the plant.

2.9.3 Vulnerability of Cooling Water Intakes to Barge Collision

Plant safety with regard to continued availability of cooling water supply requires only that there be sufficient flow to satisfy normal shutdown and post-accident requirements. Such assurance is provided by redundancy and reliability of adequate sources of cooling water supply.

The possible consequences of a storm-driven or flood-driven barge colliding with any structure or earthwork related to the plant are therefore of interest only to the extent that such collision might conceivably disable all redundant sources of supply.

The post-shutdown or post-accident supply paths of interest are those to the five cooling water pumps, any one of which can accommodate the total demand for both units with an accident having occurred in one and hot shutdown in the other. Two means of supply of intake canal water are provided for the safeguards pumps. Two horizontal pumps take suction from the main intake bays in the screenhouse, and three vertical pumps take suction from a safeguards bay in the screenhouse. The safeguards bay is a concrete structure enclosed on all sides and normally supplied by underwater ports on each side of the structure which are open to the water in the other bays. This island structure is located well back in the screenhouse and is protected on the canal side by the massive concrete piers that define the other bays.

The safeguards bay has further redundancy of supply in that it can also be fed by the emergency intake line described in Section 10.4. This source of supply is delivered through underground piping which becomes embedded within the piers, and is particularly invulnerable to any barge accident condition. The pipe is buried approximately 40 feet below the canal and emerges at a submerged intake in the branch channel of the river between the intake canal and the approach canal (see Figure 10.4-3). The intake terminal protrudes four feet upward from its crib structure, which is depressed relative to the two canals such that the highest elevation of the intake is below the bottom of the two canals.

To disable all supplies of cooling water, an accident would have to result in concurrently blocking the intake screenhouse structure screens and totally damaging or blocking the emergency intake structure. Screen bypass gates are provided in the intake screenhouse and the emergency intake structure is designed and located to preclude total blocking by the postulated barge accident. There is no credible way in which an uncontrolled barge could cause total loss of necessary cooling water supply capability.

2.9.4 Toxic Chemical Study

Due to the toxicity of commonly used chemicals, which may be transported near the Prairie Island Nuclear Generating Plant by railroad, highway or the Mississippi River, a survey was performed to predict which chemicals may become hazardous in the event of a spill. The analysis was performed in conformance with the guidance set forth by the Nuclear Regulatory Guide 1.78 and NUREG 0570.

Due to recent design changes, chlorine is no longer stored onsite and regulations requiring early warning of onsite chlorine releases no longer apply. Recognizing that removal of onsite chlorine may eliminate the need for the control room HVAC chlorine detectors and also realizing that the detectors were installed in response to a Control Room Habitability Study based on survey results which were ten years old, it was decided to revise the survey and reassess the need for toxic chemical detectors.

Surveys (Ref. 23) were performed which identified toxic chemicals either stored onsite in sufficient quantities or shipped near the plant at sufficient frequencies to warrant further evaluation. These toxic chemicals were evaluated in accordance with applicable regulatory requirements. It was deterministically concluded that all chemicals stored onsite or transported near the plant, with the exception of Soo Railroad Line chlorine and anhydrous ammonia shipments, do not pose a significant threat to control room operators. No early detection equipment is required for postulated chemical releases as sufficient time (at least two minutes) is available for the control room operators to don protective breathing equipment. A breathing air system consisting of three independent banks of high pressure air cylinders is located adjacent to the Control Room. The air cylinder banks are provided with Quick Fill stations for refilling the self contained breathing apparatus (SCBA) provided in the Control Room for the operating staff. Each SCBA provides ½ hour of breathing air capacity. Any combination of two air cylinder banks provides an additional six hours of breathing air for fourteen control room operating personnel.

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For the case of Soo Line railcar releases of chlorine or anhydrous ammonia, sufficient time could not be demonstrated using the conservative regulatory guidance for all possible combinations of weather conditions and distances from the spill to the control room ventilation outside air intake. For such releases, a probabilistic model was developed which accounts for the frequency of various weather conditions and the likelihood of a chlorine or anhydrous ammonia railcar accident which results in a toxic chemical release. Calculated probabilities were compared to the criteria of Standard Review Plan, July 1981 (SRP) Section 2.2.3 and Regulatory Guide 1.70, November 1978. Regulatory Guide 1.70, Section 2.2.3.1 states: "Design basis events external to the nuclear plant are defined as those accidents that have a probability of occurrence on the order of about 10^{-7} per year or greater and have potential consequences serious enough to affect the safety of the plant to the extent that Part 100 guidelines could be exceeded." The SRP indicates that offsite hazardous releases need not be considered if "a conservative calculation showing that the probability of occurrence of potential exposures in excess of the 10 CFR Part 100 guidelines is approximately 10^{-6} per year . . ." This "is acceptable if when combined with reasonable qualitative arguments, the realistic probability can be shown to be lower." The probability of either a chlorine or anhydrous ammonia spill resulting in control room operator incapacitation to the extent that Part 100 guidelines could be exceeded is approximately 10^{-7} per year. This probability was determined with multiple conservatisms in the analysis approach. Therefore, the acceptance criteria of both the Regulatory Guide and SRP were demonstrated.

Potential hazardous releases from all identified sources need not be considered in the design of the plant and no special control room HVAC detectors are required. Table 2.9-1 lists the chemicals which were considered and the basis for excluding them from plant design concern. Table 2.9-1 is updated periodically to evaluate any changes in stored (Reference 24) or transported toxic chemicals.

2.9.5 Summary of Analysis of Effects of Local Disasters

The possible consequences of various hypothetical local disasters have been investigated and it is concluded, for the conditions or assumptions specified for each occurrence except the toxic chemical study (Reference 23), that the plant would either be relatively unaffected in its operation or that it could safely be shutdown without initiation of a nuclear accident.

These conclusions are not entirely surprising in view of the plant design requirements which include the effects of earthquakes, floods, tornadoes, and gross release of radioactivity. Once designed consistent with these requirements, the plant is found to be relatively invulnerable to lesser local disasters, even though its design has not specifically anticipated their occurrence.

It is noted with some concern, however, that the relevance of vulnerability of any nuclear plant to the more severe offsite catastrophes that can be postulated is questionable. As in the Texas City-type explosion, the role of the nuclear plant would essentially be that of victim rather than potential offender. The direct consequences of such an accident would reasonably be expected to exceed greatly in severity any likely secondary consequences of a nuclear accident that might be provoked by the blast, provided such secondary consequences were evaluated consistent with the assumed reality of the primary event. With engineered safety features presumed to be effective unless they are directly affected by the disaster, and with activity release, if any, considered on the basis of reasonable expectation, the predicted nuclear consequences would be relatively minor.

The more severe consequences of a nuclear accident that are predicted on the traditional basis of sequential conservative assumptions, and which we apply to the nuclear plant when it is regarded by itself as a potential offender, would not logically be applied to overall evaluation of a composite accident, the severe primary consequences of which are directly predictable and not a matter of consistently conservative assumption.

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TABLE 2.9-1 CONTROL ROOM HABITABILITY TOXIC CHEMICALS

Chemical	Not Listed in ACGIH, NIOSH or 29CFR 1910?	Eliminated Based on Original Study?	Eliminated Based on Chemical Properties?	TLV Not Exceed?	IDLH Not Exceeded?	Greater Than 2 Minutes Using NUREG/CR-1741 Model? [Model Type]	Probability per year of incapacitation
Toxic Chemicals Stored Onsite (Note 2)							
1. Boric Acid	Yes						
2. Diesel Fuel #1	Yes						
3. Diesel Fuel #2	Yes						
4. Nitrogen (Liquid)	Yes						
5. Oil, Diesel Lube	Yes						
6. Oil, Turbine Lube	Yes						
7. Hydrazine	No	Yes					
8. Sodium Hydroxide (50%)	No	Yes					
9. Sulfuric Acid	No	Yes					
10. Ethylene Glycol	No	No	Yes				
11. Anion/Cation Resin	No	No	Yes				
12. Carbon Dioxide	No	No	No	Yes			
13. Propane	No	No	No	Yes			
Toxic Chemicals Stored Within Five Miles of Plant:							
None (Note 1)							
Toxic Chemicals Shipped by Truck:							
None (Note 1)							

TABLE 2.9-1 CONTROL ROOM HABITABILITY TOXIC CHEMICALS

Chemical	Not Listed in ACGIH, NIOSH or 29CFR 1910?	Eliminated Based on Original Study?	Eliminated Based on Chemical Properties?	TLV Not Exceed?	IDLH Not Exceeded?	Greater Than 2 Minutes Using NUREG/CR-1741 Model? [Model Type]	Probability per year of incapacitation
Toxic Chemicals Shipped by Rail (SOO Line):							
1. Chlorine	No	No	No	No	No	No	1.16×10^{-7} (Note 1)
2. Ammonia Anhydrous	No	No	No	No	No	No	1.47×10^{-7} (Note 1)
3. Isobutane	No	No	No	No	Yes (Note 1)		
4. LPB	No	No	No	No	Yes (Note 1)		
5. Styrene, Monomer	No	No	No	Yes (Note 1)			
6. Vinyl Acetate	No	No	No	No	Not Available	Yes (Note 1) [Model E]	
7. Benzene	No	No	No	No	Yes (Note 1)		
8. Denatured Alcohol	Yes (Note 1)						
9. Ethyl Alcohol	No	No	No	Yes (Note 1)			
10. Ethyl Acetate	No	No	No	Yes (Note 1)			
11. Methanol	No	No	No	No	Yes (Note 1)		

TABLE 2.9-1 CONTROL ROOM HABITABILITY TOXIC CHEMICALS

Chemical	Not Listed in ACGIH, NIOSH or 29CFR 1910?	Eliminated Based on Original Study?	Eliminated Based on Chemical Properties?	TLV Not Exceed?	IDLH Not Exceeded?	Greater Than 2 Minutes Using NUREG/CR-1741 Model? [Model Type]	Probability per year of Incapacitation
Toxic Chemicals Shipped by Rail (SOO Line):							
12. Toluene	No	No	No	Yes (Note 1)			
13. Flammable Liquid, N.D.S., (Pulp Mill Liquid)	Yes (Note 1)						
14. Petroleum Naptha	No	No	No	No	Yes (Note 1)		
15. Ammonium Nitrate Fertilizer	No	Yes (Note 1)					
16. Hydrogen Peroxide	No	No	No	No	Yes (Note 1)		
17. Phenol	No	No	No	Yes (Note 1)			
18. Phosphoric Acid	No	No	No	No	Yes (Note 1)		
19. Benzene Phosphorous Dichloride	Yes (Note 1)						
20. Molten Sulfur	-	No	Yes (Note 1)				
21. Nickel Sulfate	Yes (Note 1)						

TABLE 2.9-1 CONTROL ROOM HABITABILITY TOXIC CHEMICALS

Chemical	Not Listed in ACGIH, NIOSH or 29CFR 1910?	Eliminated Based on Original Study?	Eliminated Based on Chemical Properties?	TLV Not Exceed?	IDLH Not Exceeded?	Greater Than 2 Minutes Using NUREG/CR-1741 Model? [Model Type]	Probability per year of Incapacitation
Toxic Chemicals Shipped by Rail (BN Line):							
1. Chlorine	No	No	No	No	No	Yes (Note 1)	
2. Sulfur Dioxide	No	No	No	No	No	Yes (Note 1) [Model A]	
3. Carbon Dioxide	No	No	No	No	Yes (Note 1)		
4. Hydrogen Sulfide	No	No	No	No	No	Yes (Note 1)	
5. Butane	(Enveloped by SOO Line results)						
6. LPG	(Enveloped by SOO Line results)						
7. Vinyl Chloride	No	No	No	No	No	Yes (Note 1) [Model D]	
8. Ethylene Oxide	No	No	No	No	Yes (Note 1)		
9. Styrene Monomer	(Enveloped by SOO Line results.)						

TABLE 2.9-1 CONTROL ROOM HABITABILITY TOXIC CHEMICALS

Chemical	Not Listed in ACGIH, NIOSH or 29CFR 1910?	Eliminated Based on Original Study?	Eliminated Based on Chemical Properties?	TLV Not Exceed?	IDLH Not Exceeded?	Greater Than 2 Minutes Using NUREG/CR-1741 Model? [Model Type]	Probability per year of incapacitation
Toxic Chemicals Shipped by Rail (BN Line):							
10. Benzene	(Enveloped by SOO Line results)						
11. Denatured Alcohol	Yes (Note 1)						
12. Ethyl Alcohol	(Enveloped by SOO Line results)						
13. Methyl Alcohol	(Enveloped by SOO Line results)						
14. Paint	Yes (Note 1)						
15. Resin Solution	Yes (Note 1)						
16. Aromatic Concentrates	Yes (Note 1)						
17. Diesel Fuel Oil	No	Yes (Note 1)					
18. Petroleum Naptha	(Enveloped by SOO Line results)						
19. Calcium Carbide (Flammable Solid)	No	No	Yes (Note 1)				
20. Sodium Metal (Flammable Solid)	No	No	Yes (Note 1)				

TABLE 2.9-1 CONTROL ROOM HABITABILITY TOXIC CHEMICALS

Chemical	Not Listed in ACGIH, NIOSH or 29CFR 1910?	Eliminated Based on Original Study?	Eliminated Based on Chemical Properties?	TLV Not Exceed?	IDLH Not Exceeded?	Greater Than 2 Minutes Using NUREG/CR-1741 Model? [Model Type]	Probability per year of Incapacitation
Toxic Chemicals Shipped by Rail (BN Line):							
21. Sodium Chlorate	No	No	Yes (Note 1)				
22. Chloropierin Mixture	No	No	No	No	No	Yes (Note 1)	
23. Sodium Cyanide (Solid)	No	No	Yes (Note 1)				
24. Sulfuric Acid	No	No	No	Yes (Note 1)			
25. Phosphoric Acid	(Enveloped by SOO Line results)						
26. Acetic Anhydride	No	No	No	Yes (Note 1)			
27. Ferric Chloride Solution	Yes (Note 1)						
28. Silicon Chloride	No	No	Yes (Note 1)				
29. Titanium Tetrachloride	Yes (Note 1)						
30. Potassium Hydroxide	Yes (Note 1)						
31. Sodium Hydroxide	No	Yes	Yes (Note 1)				
32. Molten Sulfur	-	Yes (Note 1)					

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TABLE 2.9-1 CONTROL ROOM HABITABILITY TOXIC CHEMICALS

Chemical	Not Listed in ACGIH, NIOSH or 29CFR 1910?	Eliminated Based on Original Study?	Eliminated Based on Chemical Properties?	TLV Not Exceed?	IDLH Not Exceeded?	Greater Than 2 Minutes Using NUREG/CR-1741 Model? [Model Type]	Probability per year of Incapacitation
Toxic Chemicals Shipped by Barge:							
1. Chemical Fertilizers	No	Yes (Note 1)					

NOTE 1: See Reference 23 for details.

NOTE 2: See Reference 24 for details.

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SECTION 4

REACTOR COOLANT SYSTEM

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SECTION 4 REACTOR COOLANT SYSTEM

4.1 SUMMARY DESCRIPTION

The Reactor Coolant Systems of the two nuclear power units are essentially identical, and do not share any components. The following description applies to either unit.

The Reactor Coolant System, shown in the Flow Diagrams, Figure 4.1-1A for Unit 1 and Figure 4.1-1B for Unit 2, consists of two identical heat transfer loops connected in parallel to the reactor vessel. Each loop contains a circulating pump and a steam generator. The system also includes a pressurizer, pressurizer relief tank, connecting piping, and instrumentation necessary for operational control. The pressurizer surge line is connected to one of the loops.

The containment boundary shown on the flow diagrams indicates those major components which are to be located inside containment. The intersection of a process line with this boundary indicates a containment penetration.

Reactor Coolant System and components design data are listed in Tables 4.1-1 through 4.1-7.

The Reactor Coolant System transfers the heat generated in the core to the steam generators where steam is generated to drive the turbine generator. Borated demineralized water is circulated at the flow rate and temperature consistent with achieving the reactor core thermal hydraulic performance presented in Section 3. The water also acts as a neutron moderator and reflector, and as a solvent for the neutron absorber used in chemical shim control.

The Reactor Coolant System provides a boundary for containing the coolant under operating temperature and pressure conditions. It serves to confine radioactive material and limits to acceptable values any release of radioactive material to the secondary system and to other parts of the plant under conditions of either normal or abnormal reactor operation. During transient operation the system's heat capacity attenuates thermal transients. The Reactor Coolant System accommodates coolant volume changes within the bounds of the protection system criteria.

By appropriate selection of the inertia of the reactor coolant pumps, the thermal hydraulic effects which result from a loss-of-flow situation are reduced to a safe level during the pump coastdown. The layout of the system assures the natural circulation capability following a loss of flow to permit plant cooldown without overheating the core.

Pressure in the system is controlled by the pressurizer, where water and steam pressure is maintained through the use of electrical heaters and sprays. Steam can either be formed by the heaters, or condensed by pressurizer spray to minimize pressure variations due to contraction and expansion of the coolant. Instrumentation used in the pressure control system is described in Section 7. Spring-loaded steam safety valves and power-operated relief valves are connected to the pressurizer and discharge to the pressurizer relief tank, where the discharged steam is condensed and cooled by mixing with water.

Maximum Heating and Cooling Rates

The reactor system operating cycles used for design purposes are given in Table 4.1-8 and described in Section 4.1.4. The normal system heatup rate limit is 100°F/hr and the cooldown rate is 100°F/hr. These limits are discussed in the Pressure-Temperature Limits Report (PTLR) in the back of the Technical Specifications. The pressurizer heatup rate will not exceed 100°F/hr and the pressurizer cooldown rate will not exceed 200°F/hr. The original capacity of the pressurizer heaters permitted a heat up rate of 55°F/hr, starting with a minimum water level. This rate takes into account the small continuous bypass spray flow provided around the pressurizer spray valves to maintain the pressurizer liquid boron concentration homogeneous with that in the reactor coolant. The capacity of the heaters may be reduced below the original design; which translates into a reduced heat up rate.

The spray is not used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F. The fastest cooldown rates which result from the hypothetical case of a break of a main steam line are addressed in Section 4.1.4.13.

In January 1990, it was determined that the pressurizer cooldown rate limit of 200°F/hr and the temperature difference limit of 320°F between the pressurizer and the spray fluid was exceeded during the cooldown for the Unit 1 refueling outage that month. Subsequent investigation revealed that the cooldown rate limit had been previously exceeded during unit cooldowns due to procedural inadequacy. Westinghouse performed an analysis to determine the effects of exceeding these limits. The analysis (Reference 91) concluded that the transients did not compromise the structural integrity of the pressurizer. Measures have been taken to ensure that the limits will not be exceeded in the future (References 92 and 97).

Materials and Design Control

Each of the materials used in the Reactor Coolant System was selected for the expected environment and service conditions. The major component materials are listed in Table 4.1-1.

The safety of the reactor vessel and all other Reactor Coolant System pressure containing components and piping is dependent on several major factors including design and stress analysis, material selection and fabrication, quality control and operations control.

The phenomena of stress corrosion cracking and corrosion fatigue are not encountered unless a specific combination of conditions is present. The necessary conditions are a susceptible alloy, an aggressive environment, a stress, and time.

A complete stress analysis of the Reactor Coolant System which reflects consideration of all design loadings detailed in the design specification has been prepared by the designer. The analysis shows that the reactor vessel, steam generator, pump casing and pressurizer comply with the stress limits of Section III of the ASME Code. A similar analysis of the piping shows that it complies with the stress limits of the applicable USAS Code.

As part of the design control on materials, Charpy V-notch toughness tests were run on all ferritic material used in fabricating pressure parts of the reactor vessel, steam generator and pressurizer to provide assurance for hydrotesting and operation in the ductile region at all times. In addition, dropweight tests and Charpy V-notch transition temperature curves were performed on the reactor vessel materials.

As an assurance of system integrity, all components in the system were hydrotested at a nominal test pressure of 3107 psig prior to initial operation.

All Reactor Coolant System materials which are exposed to the coolant are corrosion resistant. They consist of stainless steels and Inconel, and they are chosen for specific purposes at various locations within the system for their superior compatibility with the reactor coolant.

Water Chemistry

The water chemistry is selected to provide the necessary boron content for reactivity control and to minimize corrosion of Reactor Coolant System surfaces.

All Reactor Coolant System materials which are exposed to the coolant are corrosion resistant. Maintenance of the water quality to minimize corrosion is accomplished using the Chemical and Volume Control System and Sampling System which are described in Section 10.

Typical reactor coolant chemistry compositions are given in Table 4.1-9. This subject is also discussed in the Technical Requirements Manual.

A typical condition of operation could include any combination of chemical elements, as long as none of the specified limits are exceeded.

Galvanic Corrosion

The only types of materials which are in contact with each other in borated water are stainless steels, Inconel, Stellite valve materials and Zircaloy fuel element coating. These materials have been shown (Reference 121) to exhibit only an insignificant degree of galvanic corrosion when coupled to each other.

For example, the galvanic corrosion of Inconel versus 304 stainless steel resulting from high temperature tests (575°F) in lithiated, boric acid solution was found to be less than -20.9 mg/dm^2 for the test period of 9 days. Further galvanic corrosion would be trivial since the cell currents at the conclusion of the tests were approaching polarization. Zircaloy versus 304 stainless steel was shown to polarize in 180°F lithiated, boric acid solution in less than 8 days with a total galvanic attack of -3.0 mg/dm^2. Stellite versus 304 stainless steel was polarized in 7 days at 575°F in lithiated, boric acid solution. The total galvanic corrosion for this couple was -0.97 mg/dm^2.

As can be seen from the tests, the effects of galvanic corrosion are insignificant to systems containing borated water.

ZIRLO material properties are essentially identical to the Zircaloy alloy; therefore, the effect of galvanic corrosion on this new zirconium based fuel rod clad and guide thimble tube alloy is insignificant. (Reference 122)

Protection Against Proliferation of Dynamic Effects

Engineered Safety Features and associated systems are protected from loss of function due to dynamic effects and missiles which might result from a loss-of-coolant accident. Protection is provided by missile shielding and/or separation of redundant components. Further discussion of missile protection is given in Sections 6 and 12.

The Reactor Coolant System is surrounded by concrete shield walls. These walls provide shielding to permit access into the containment during full power operation for inspection and maintenance of miscellaneous equipment. These shielding walls also provide missile protection for the reactor containment building.

The steam generator is provided with hydraulic shock suppressors as part of the upper lateral support near the center of gravity to resist lateral loads, including those resulting from seismic forces and pipe rupture forces. Additional bracing is also provided at a lower elevation to resist pipe rupture loads.

Missile protection afforded by the arrangement of the Reactor Coolant System is illustrated in the containment structure drawings which are given in Section 12.

4.1.1 General Design Basis

4.1.1.1 Quality Standards

Criterion: Those systems and components of reactor facilities which are essential to the prevention, or the mitigation of the consequences, of nuclear accidents which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required. (GDC 1)

The Reactor Coolant System is of primary importance with respect to its safety function in protecting the health and safety of the public.

Quality standards of material selection, design, fabrication and inspection conform to the applicable provisions of recognized codes and good nuclear practice (Section 4.1.6). Details of the quality assurance programs, test procedures and inspection acceptance levels are given in Section 4.7. Particular emphasis is placed on the assurance of quality of the reactor vessel to obtain material whose properties are uniformly within code specifications.

4.1.1.2 Performance Standards

Criterion: Those systems and components of reactor facilities which are essential to the prevention or to the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that will enable such systems and components to withstand, without undue risk to the health and safety of the public the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind or heavy ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design. (GDC 2)

All piping, components and supporting structures of the Reactor Coolant System, except for those components provided for in paragraph (c)(2) of Section 50.55a of 10CFR50, are designed as Class I equipment, i.e. they are capable of withstanding:

a. The Operational Basis Earthquake (OBE) ground acceleration within allowable working stresses.

b. The Design Basis Earthquake (DBE) ground acceleration acting in the horizontal and vertical direction simultaneously with no loss of function.

Allowable limits for the above are given in Section 12.

The Reactor Coolant System is located in the containment building whose design, in addition to being a Class I structure, also considers accidents or other applicable natural phenomena. Details of the containment design are given in Section 12.

4.1.1.3 Records Requirements

Criterion: The reactor licensee shall be responsible for assuring the maintenance throughout the life of the reactor of records of the design, fabrication, and construction of major components of the plant essential to avoid undue risk to the health and safety of the public. (GDC 5)

Records of the design, of the major Reactor Coolant System components and the related engineered safety features components are maintained in the offices of the Licensee and will be retained there throughout the life of the plant.

Records of fabrication are maintained in the manufacturers' plants as required by the appropriate Code, or other requirements. Construction records will be retained by the Licensee for the life of the plant.

4.1.1.4 Missile Protection

Criterion: Adequate protection for those engineered safety features, the failures of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures. (GDC 40)

The dynamic effects during blowdown following a loss-of-coolant accident are evaluated in the detailed layout and design of the high pressure equipment and barriers which afford missile protection. Fluid and mechanical driving forces are calculated, and consideration is given to possible damage due to fluid jets and secondary missiles which might be produced.

The steam generators are supported, guided and restrained in a manner which prevents rupture of the steam side of a generator, the steam lines and the feedwater piping as a result of forces created by a Reactor Coolant System pipe rupture. These supports, guides and restraints also prevent rupture of the primary side of a steam generator as a result of forces created by a steam or feedwater line rupture.

The mechanical consequences of a pipe rupture are restricted by design such that the functional capability of the engineered safety features is not impaired.

A further discussion of missile protection is given in Section 12.

4.1.2 Principal Design Basis

4.1.2.1 Reactor Coolant Pressure Boundary

Criterion: The reactor coolant pressure boundary shall be design, fabricated and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime. (GDC 9)

The Reactor Coolant System in conjunction with its control and protective provisions is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions, and maintain the stresses within applicable code stress limits.

Fabrication of the components which constitute the pressure boundary of the Reactor Coolant System is carried out in strict accordance with the applicable codes. In addition there are areas where equipment specifications for Reactor Coolant System components go beyond the applicable codes. Details are given in Section 4.7.1.

The materials of construction of the pressure boundary of the Reactor Coolant System are protected by control of coolant chemistry from corrosion phenomena which might otherwise reduce the system structural integrity during its service lifetime.

System conditions resulting from anticipated transients or malfunctions are monitored and appropriate action is automatically initiated to maintain the required cooling capability and to limit system conditions to a safe level.

The system is protected from overpressure by means of pressure relieving devices, as required by Section III of the ASME Boiler and Pressure Vessel Code.

Isolable sections of the system containing components designed in conformance with Section III of the ASME Boiler and Pressure Vessel Code are provided with overpressure relieving devices discharging to closed systems such that the system code allowable relief pressure within the protected section is not exceeded.

4.1.2.2 Monitoring Reactor Coolant Leakage

Criterion: Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary. (GDC 16)

Positive indications in the control room of leakage of coolant from the Reactor Coolant System to the containment are provided by equipment which permits continuous monitoring of containment air radioactivity and humidity, and runoff from the condensate collecting pans under the cooling coils of the containment air cooling units. This equipment provides indication of normal background which is indicative of a basic level of leakage from Reactor Coolant System pressure boundary. Any increase in the observed parameters is an indication of change within the containment, and the equipment provided is capable of monitoring this change. The basic design criterion is the detection of deviations from normal containment environmental conditions including air particulate radioactivity, radiogas activity, humidity, condensate runoff and in addition, in the case of gross leakage, the liquid inventory in the process systems and containment sump.

Further details are supplied in Section 6.5. The maximum permitted reactor coolant leakage rates for uncontrolled sources are stated in the Technical Specifications.

4.1.2.3 Reactor Coolant Pressure Boundary Capability

Criterion: The reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition. (GDC 33)

The reactor coolant boundary is shown to be capable of accommodating without rupture, the static and dynamic loads imposed as a result of a sudden reactivity insertion such as a rod ejection. Details of this analysis are provided in Section 14.

The operation of the reactor is such that the severity of a rod ejection accident is inherently limited. Since rod cluster control assemblies (RCCA) are used to control load variations only and boron dilution is used to compensate for core depletion, only the RCCA in the controlling groups are inserted in the core at power, and at full power these rods are only partially inserted. A rod insertion limit monitor is provided as an administrative aid to insure that this condition is met.

By using the flexibility in the selection of control rod groupings, radial locations and position as a function of load, the design limits the maximum fuel temperature for the highest worth ejected control rod accident to a value which precludes any resultant damage to the primary system pressure boundary due to excessive pressure surges.

4.1.2.4 Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention

Criterion: The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failure. Consideration is given (a) to the provisions for control over service temperature and irradiation effects which may require operational restrictions, (b) to the design and construction of the reactor pressure vessel in accordance with applicable codes, including those which establish requirements for absorption of energy within the elastic strain energy range and for absorption of energy by plastic deformation and (c) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes. (GDC 34)

The ability of the large steel pressure vessel containing the reactor core and its primary coolant to resist fracture constitutes an important factor in insuring safety in the nuclear industry. The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to significant fast neutron bombardment. The overall effects of fast neutron irradiation on the mechanical properties of low alloy ferritic pressure vessel steels such as SA508 Class 3 (base material of the Prairie Island Units 1 and 2 reactor pressure vessel beltlines) are well documented in the literature. Generally, low alloy ferritic materials show an increase in hardness and tensile properties and decrease in ductility and toughness under certain conditions of irradiation.

The Reactor Vessel Material Surveillance Program, described in Section 4.7.2, monitors the effects of radiation on reactor vessel materials, and establishes operating limits to assure that brittle fracture of the reactor vessel will not occur. The program is in accordance with ASTM-E-185.

The special case of low temperature overpressurizations has been addressed by installing the Low Temperature Overpressure Protection System (OPPS) described in Section 4.4.3.3. The design criteria for this system is detailed in References 89 and 107.

4.1.3 Design Characteristics

4.1.3.1 Design Pressure

The Reactor Coolant System design and operating pressure together with the safety, power relief and pressurizer spray valve set points, and the protection system set point pressures are listed in Table 4.1-2. The design pressure allows for operating transient pressure changes. The selected design margin considers core thermal lag, coolant transport times and pressure drops, instrumentation and control response characteristics, and system relief valve characteristics. The design pressures and data for the respective system components are listed in Tables 4.1-3 through 4.1-7. Table 4.1-10 gives the design pressure drop of the system components.

4.1.3.2 Design Temperature

The design temperature for each component is selected to be above the maximum coolant temperature in that component under all normal and anticipated transient load conditions. The design and operating temperatures of the system components are listed in Tables 4.1-3 through 4.1-7.

4.1.3.3 Seismic Loads

The seismic loading conditions are established by the Operational Basis Earthquake (OBE) and the Design Basis Earthquake (DBE). The former is selected to be typical of the largest probable ground motion based on the site seismic history. The latter is selected to be the largest potential ground motion at the site based on seismic and geological factors and their uncertainties.

For the OBE loading condition, the nuclear steam supply system is designed to be capable of continued safe operation. Therefore, for this loading condition, critical structures and equipment needed for this purpose are required to operate within normal design limits. The seismic design for the DBE is intended to provide a margin in design that assures capability to shut down and maintain the nuclear facility in a safe condition. In this case, it is only necessary to ensure that the Reactor Coolant System components do not lose their capability to perform their safety function. This has come to be referred to as the "no-loss-of-function" criteria and the loading condition as the "no-loss-of-function earthquake" loading condition.

For the combination of normal plus OBE loadings, the stresses in the support structures are kept within the limits of the applicable codes.

For the combination of normal plus DBE loadings, the stresses in the support structures are limited to values necessary to ensure their integrity, and to keep the stresses in the Reactor Coolant System components within the allowable limits as given in Section 12. Shock suppressors are installed on the RCS system to prevent the unrestrained motion of the RCS pipes and components under dynamic loads such as earthquakes and other severe transients. Shock suppressors do not restrain the normal thermal movements during startup and shutdown.

4.1.4 Cyclic Loads

To provide the necessary high degree of integrity for the components in the Reactor Coolant System, transient conditions are selected for fatigue evaluation based on a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from normal operation, normal and abnormal load transients and accident conditions. To a large extent, the specific transient operating conditions to be considered for equipment fatigue analyses are based upon engineering judgment and experience. Those transients are chosen which are representative of transients to be expected during plant operation and which are sufficiently severe or frequent to be of possible significance to component cyclic behavior. The number of thermal and loading cycles used for fatigue evaluation are given in Table 4.1- 8.

Clearly it is difficult to discuss in absolute terms the transients that the plant will actually experience during the 40 years operating life. For clarity, however, each transient condition is discussed in order to make clear the nature and basis for the various transients.

4.1.4.1 Heatup and Cooldown

The normal heatup or cooldown cases are conservatively represented by a continuous operation performed at a uniform temperature rate of 100°F per hour (except for a pressurizer cooldown rate of 200°F per hour).

For these cases, the heatup occurs from ambient to the no-load temperature and pressure condition and the cooldown represents the reverse situation. In actual practice, the rate of temperature change of 100°F per hour will not be attained because of other limitations such as:

a. Slower initial heatup rates when using pumping energy only.

b. Interruptions in the heatup and cooldown cycles due to such factors as drawing a pressurizer steam bubble, rod withdrawal, sampling, water chemistry and gas adjustments.

The number of such complete heatup and cooldown operations is specified at 200 times each which corresponds to five such occurrences per year for the 40-year plant design life. In practice, experience at Prairie Island over a period of more than 20 years indicates that the number of complete heatup and cooldown operations on each unit will be much less than 200 over its 40-year plant design life.

4.1.4.2 Unit Loading and Unloading

The unit loading and unloading cases are conservatively represented by a continuous and uniform ramp power change of 5% of nominal full load per minute between 15% and 100% of nominal full load. This load swing is the maximum possible without reactor trip subject to possible xenon limitations consistent with operation with automatic reactor control. The reactor coolant temperature will vary with load as prescribed by the temperature control system. The number of each operation is specified at 18,300 times or 1 time per day with approximately 40% margin for the 40 year design life.

4.1.4.3 Step Load Increase and Decrease of 10%

The 10% of nominal full load step change, increase or reduction, in load demand is a control transient which is assumed to be a change in turbine control valve opening which might be occasioned by disturbances in the electrical network into which the plant output is tied. The Reactor Control System is designed to restore plant equilibrium without reactor trip following a 10% of nominal full load step change, increase or reduction, in turbine load demand initiated from nuclear plant equilibrium conditions in the range between 15% and 100% of nominal full load, the range for automatic reactor control. In effect, during load change conditions, the Reactor Control System attempts to match turbine and reactor outputs in such a manner that peak reactor coolant temperature is minimized and reactor coolant temperature is restored to its programmed set point at a sufficiently slow rate to prevent excessive pressurizer pressure decrease.

Following a step decrease in turbine load, the secondary side steam pressure and temperature initially increase since the decrease in nuclear power lags behind the step decrease in turbine load. During the same increment of time, the Reactor Coolant System average temperature and pressurizer pressure also initially increase. Because of the power mismatch between the turbine and reactor and the increase in reactor coolant temperature, the control system automatically inserts the control rods to reduce core power. With load decrease, the reactor coolant temperature will be ultimately reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average temperature set point change is made as a function of turbine-generator load as determined by first stage turbine pressure measurement. The pressurizer pressure will also decrease from its peak pressure value and follow the reactor coolant decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash which reduces the rate of pressure decrease. Subsequently, the pressurizer heaters come on to restore the plant pressure to its normal value.

Following a step load increase in turbine load, the reverse situation occurs, i.e., the secondary side steam pressure and temperature initially decrease and the reactor coolant average temperature and pressure initially decrease. The control system automatically withdraws the control rods to increase core power. The decreasing primary pressure transient is reversed by actuation of the pressurizer heaters and eventually the system pressure is restored to its normal value. The reactor coolant average temperature will be raised to a value above its initial equilibrium value at the beginning of the transient.

The number of each operation is specified at 2000 times or 50 per year for the 40-year plant design life.

4.1.4.4 Large Step Decrease in Load

This transient applies to a step decrease in turbine load from full power of such magnitude that the resultant rapid increase in reactor coolant average temperature and secondary side steam pressure and temperature will automatically initiate a secondary side steam dump system that will prevent a reactor shutdown or lifting of steam generator safety valves. The plant is designed to accept a step decrease of 47.5% of nominal full load. This signifies that a steam dump system will provide a heat sink to accept 37.5% of nominal full load. The remaining 10% of the total step change is assumed by the Reactor Rod Control System. If a steam dump system was not provided to cope with this transient, there would be such a strong mismatch between turbine and reactor power that a reactor trip and lifting of steam generator safety valves would occur.

The number of occurrences of this transient is specified at 200 times or 5 per year for the 40-year plant design life.

4.1.4.5 Loss of Load

This transient applies to a step decrease in turbine load from full power brought about by a loss of turbine load without immediately initiating a reactor trip and represents the most severe transient on the Reactor Coolant System. The reactor and turbine eventually trip as a consequence of a high pressurizer level trip initiated by the Reactor Protection System.

The number of occurrences of this transient is specified at 80 times or 2 per year for the 40-year plant design life. Since redundant means of tripping the reactor upon turbine trip are provided as part of the Reactor Protection System, transients of this nature are not expected.

4.1.4.6 Loss of Offsite Power

This transient applies to the loss of outside electrical power to the station and a reactor and turbine trip, on low reactor coolant flow, culminating in a complete loss of plant AC electrical power. Under these circumstances, the emergency diesel generators are started, the reactor coolant pumps are de-energized and following the coastdown of the reactor coolant pumps, natural circulation builds up in the system to some equilibrium value. This condition permits removal of core residual heat through the steam generators which at this time are receiving feedwater from the Auxiliary Feedwater System operating from diesel generator power or steam driven auxiliary feedwater pumps. Steam is removed for reactor cooldown through atmospheric power operated relief valves provided for this purpose.

The number of occurrences of this transient is specified at 40 times or 1 per year for the 40-year plant design life.

4.1.4.7 Loss of Flow

This transient applies to a partial loss of flow accident from full power in which a reactor coolant pump is tripped out of service as a result of a loss of power to that pump. The consequences of such an accident at high power level are a reactor and turbine trip, on low reactor coolant flow, followed by automatic opening of the steam dump system and flow reversal in the affected loop. The flow reversal results in reactor coolant at cold leg temperature, being passed through the steam generator and cooled still further. This cooler water then passes through the hot leg piping and enters the reactor vessel outlet nozzles. The net result of the flow reversal is a sizeable reduction in the hot leg coolant temperature of the affected loop.

The number of occurrences of this transient is specified at 80 times or 2 per year for the 40-year plant design life.

4.1.4.8 Reactor Trip From Full Power

A reactor trip from full power may occur for a variety of causes resulting in temperature and pressure transients in the Reactor Coolant System and in the secondary side of the steam generator. This is the result of continued heat transfer from the reactor coolant in the steam generator. The transient continues until the reactor coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. A continued supply of feedwater and controlled dumping of secondary steam remove the core residual heat and prevent the steam generator safety valves from lifting. The reactor coolant temperature and pressure undergo a rapid decrease from full power values as the Reactor Protection System causes the control rods to move into the core.

The number of occurrences of this transient is specified at 400 times or 10 per year for the 40-year plant design life.

4.1.4.9 Turbine Roll Test

This transient is imposed upon a plant during the hot functional test period for turbine cycle checkout. Reactor coolant pump power is used to heat the reactor coolant to operating temperature and the steam generated is used to perform a turbine roll test. However, the plant cooldown during this test may exceed the normal 100°F per hour maximum rate.

The number of such test cycles is specified at 10 times to be performed at the beginning of plant operating life prior to irradiation. Two such cycles were performed at Prairie Island on Unit 1; none, on Unit 2.

4.1.4.10 Hydrostatic Test Conditions

The pressure tests are outlined below:

a. Primary Side Hydrostatic Test Before Initial Startup at 3107 psig

   The pressure tests covered by this section included both shop and field hydrostatic tests which occurred as a result of component or system testing. This hydro test was performed at a water temperature which was compatible with reactor vessel material design transition temperature (DTT) requirements which shift with lifetime and a maximum test pressure of 3107 psig. In this test, the primary side of the steam generator was pressurized to 3107 psig coincident with the secondary side pressure of 0 psig. The Reactor Coolant System was analyzed for 5 cycles of this hydro test.

b. Secondary Side Hydrostatic Test Before Initial Startup

   The secondary side of the steam generator was pressurized to 1356 psig with a minimum water temperature of 70°F coincident with the primary side at 0 psig.

   The steam generator was analyzed for 5 cycles of this test. Normally only one test would be made to satisfy the code requirements and this was made at the site after installation.

c. Primary Side Leak Test

   Subsequent to each time the primary system is opened, a leak test will be performed. During this test the primary system pressure, for design purposes, is assumed to be raised to 2500 psia, with the system temperature above Design Transition Temperature, while the system is checked for leaks.

   For design purposes it was assumed that the primary side experienced 50 cycles of this test during the 40-year design life of the plant. In actual practice, the primary system is pressurized to the nominal operating pressure associated with 100% rated reactor power with the test pressure and temperature attained at a rate in accordance with DTT considerations.

During this leak test, the secondary side of the steam generator must be pressurized so that the pressure differential across the tube sheet does not exceed 1600 psi. This is accomplished by closing off the steam lines.

4.1.4.11 Pressurizer Surge and Spray Line Connections

The surge and spray nozzle connections at the pressurizer vessel are subject to cyclic temperature changes resulting from the transient conditions described previously. The various transients are characterized by variations in reactor coolant temperature which in turn result in water surges into or out of the pressurizer. The surges manifest themselves as changes in system pressure which, depending upon whether an increase or decrease in pressure occurs, result in introducing spray water into the pressurizer to reduce pressure or actuating the pressurizer heaters to increase pressure to the equilibrium value. To illustrate a load change cycle as it affects the pressurizer, consider a design step increase in load. The pressurizer initially experiences an outsurge with a drop in system pressure which actuates the pressurizer heaters to restore system pressure. As the Reactor Control System reacts, the reactor coolant temperature is increased which causes an insurge into the pressurizer raising system pressure. As pressure is increased, the heaters go off and at some pressure setpoint, the spray valves open to limit the pressure rise and restore system pressure. Thus the pressurizer surge nozzle is subjected to a temperature increase on the outsurge followed by a temperature decrease on the insurge during this load transient. The pressurizer spray nozzle is subjected to a temperature decrease when the spray valve opens to admit reactor coolant cold leg water into the pressurizer. The pressurizer experiences a reverse situation during a load decrease transient, i.e., an insurge followed by an outsurge. It is assumed that the spray valve opens to admit spray water into the pressurizer once at the design flowrate for each design step change in plant load. Thus the number of occurrences for the spray nozzle corresponds to that shown for the step changes in plant load in Table 4.1-8.

During plant cooldown, spray water is introduced into the pressurizer to cool down the pressurizer and to remove gas from the reactor coolant. The maximum pressurizer cooldown rate is specified at 200°F per hour which is twice the rate specified for the other Reactor Coolant System components.

4.1.4.12 Classification of RCS Transients

Transients shown in Table 4.1-8 are classified by the following conditions:

Normal Condition	-	Items 1-4, 13
Upset Condition	-	Items 5-8
Test Condition	-	Items 9-11
Faulted Condition	-	Items 12a,b,c

4.1.4.13 Accident Conditions

The effect of the accident loading was evaluated in combination with normal loads to demonstrate the adequacy to meet the stated plant safety criteria.

A brief description of each accident transient which was considered is listed below.

a. Reactor Coolant Pipe Break

   This accident involves the rupture of a Reactor Coolant System pipe resulting in a loss of primary coolant. It is conservatively assumed that the system pressure and temperature are reduced rapidly and the Safety Injection System is initiated to introduce 70°F water into the Reactor Coolant System. The safety injection signal results in a turbine and reactor trip. Because of the rapid blowdown of coolant from the system and the comparatively large heat capacity of the metal sections of the components, it is likely that the metal is still at no-load temperature conditions when the 70°F safety injection water is introduced into the system. One occurrence has been evaluated for this case.

b. Steam Line Break

   For Reactor Coolant System component evaluation, the following conservative conditions were considered:

   1. The reactor is initially in a hot, zero-load, just critical condition assuming all rods in except the most reactive rod which is assumed to be stuck in its fully withdrawn position.

   2. A steam line break occurs inside the containment resulting in a reactor trip.

   3. Subsequent to the break, there is no return to power and the reactor coolant temperature cools down to 212°F.

   4. The Safety Injection System pumps restore the reactor coolant pressure.

   The above conditions result in the most severe temperature and pressure variations which the Reactor Coolant System components will encounter during a steam break accident. One occurrence has been evaluated for this case.

c. Steam Generator Tube Rupture

This accident postulates the double-ended rupture of a steam generator tube resulting in a decrease in pressurizer level and reactor coolant pressure. Reactor trip will occur due to low pressurizer pressure. Shortly after this, a low pressurizer pressure safety injection will occur. This safety injection signal will close the feedwater regulating valves. After the rupture, the primary system pressure is reduced below the secondary system design pressure (1100 psia). The planned procedure for recovery from this accident calls for isolation of the steam line leading from the affected steam generator at this time. Therefore, this accident will result in a transient which is no more severe than that associated with a reactor trip. For this reason, it requires no special treatment in so far as fatigue evaluation is concerned, so no occurrences have been evaluated.

4.1.5 Service Life

The service life of Reactor Coolant System pressure components depends upon the end-of-life material radiation damage, unit operational thermal cycles, quality manufacturing standards, environmental protection, and adherence to established operating procedures.

The reactor vessel is the only component of the Reactor Coolant System which is exposed to a significant level of neutron irradiation and it is therefore the only component which is subject to any appreciable material radiation damage effects.

The nil ductility transition temperature (NDTT) shift of the vessel material and welds, during service due to radiation damage effects is monitored by a radiation damage surveillance program which conforms with ASTM E185 and Appendix H of 10CFR50.

Reactor vessel design is based on the transition temperature method of evaluating the possibility of brittle fracture of the vessel material, as a result of operations such as leak testing and plant heatup and cooldown.

To establish the service life of the Reactor Coolant System components as required by the ASME (Part III), Boiler and Pressure Vessel Code for Class "A" Vessels, the unit operating conditions have been established for the 40 year design life. These operating conditions include the cyclic application of pressure loadings and thermal transients.

The number of thermal and loading cycles used for design purposes are listed in Table 4.1-8.

4.1.6 Codes and Classifications

All pressure-containing components of the Reactor Coolant System are designed, fabricated, inspected and tested in conformance with the applicable codes listed in Table 4.1-11.

The Reactor Coolant System is classified as Class I as detailed in Section 12, except for those components provided for in paragraph (c)(2) of Section 50.55a of 10CFR50.

4.1.7 Materials of Construction

All core structural load bearing members were made from annealed type 304 stainless steel, so there is no possibility of sensitization, with the exception of the core barrel itself, which required stress relief during manufacturing at temperatures over 750°F. The stress relieving operation was conducted in a manner to minimize the possibility of severe sensitization, while maintaining the necessary conditions for relieving residual fabrication stresses. This consisted of heating to 1650°F, holding at this temperature for several hours, then cooling very slowly in the furnace. This treatment results in massive carbide precipitation at the grain boundaries, and agglomeration of the carbides, instead of the formation of detrimental continuous carbide films. Further, the long times at high temperatures cause diffusion of chromium into the grain boundary areas that were depleted in chromium by the precipitation of chromium carbides. This combination of formation of massive carbides, plus diffusion of chromium back into the depleted zone is referred to as "desensitization", and is commonly used to prevent severe sensitization of parts requiring heat treatments that otherwise would cause severe sensitization of the material. Strauss tests run according to ASTM A393 were performed on core barrel material given this heat treatment, and results verified that severe sensitization is prevented.

It is characteristic of stress corrosion that combinations of alloy and environment which result in cracking are usually quite specific. Environments which have been shown to cause stress corrosion cracking of stainless steels are free alkalinity in the presence of a concentrating mechanism and the presence of chlorides, fluorides, and free oxygen. With regard to the former, experience has shown that deposition of chemicals on the surface of tubes can occur in a steam blanketed area within a steam generator.

In the presence of this environment under very specific conditions, stress-corrosion cracking can occur in stainless steels having the nominal residual stresses resulting from normal manufacturing procedures. However, the steam generator contains Inconel tubes. Testing to investigate the susceptibility of heat exchanger construction materials to stress corrosion in caustic and chloride aqueous solutions has indicated that Inconel Alloy has excellent resistance to general and pitting-type corrosion in severe operating water conditions.

The use of lead in the materials of the secondary side of this plant has been minimized to the practical limit of that occurring as trace elements in metallurgical alloys and as such is insignificant.

All external insulation of Reactor Coolant System components is compatible with the component materials. The cylindrical shell exterior and closure flanges to the reactor vessel, the reactor vessel closure head, and all other external corrosion resistant surfaces in the Reactor Coolant System are insulated with metallic reflective insulation as required.

The nil ductility transition (NDT) temperature of the reactor vessel material opposite the core is established at a Charpy V-notch impact energy of 30 ft-lb or greater. The material is tested to verify conformity to specified requirements and to determine the actual NDT temperature value. In addition, this material is 100 percent volumetrically inspected by ultrasonic test using both straight beam and angle beam methods.

The remaining material in the reactor vessel, and other Reactor Coolant System components, meets the appropriate design code requirements and specific component function.

The reactor vessel material is heat-treated specifically to obtain good Charpy V-notch ductility which ensures a low NDT temperature and thereby gives assurance that the finished vessel can be initially hydrostatically tested and operated as near to room temperature as possible without restrictions. The stress limits established for the reactor vessel are dependent upon the temperatures at which the stresses are applied. As a result of fast neutron irradiation in the region of the core, the material properties will change, including an increase in the NDT temperature. A nominal maximum value of NDT temperature was established during fabrication.

The shift of the NDT is affected by neutron fluence. The methodology used to provide the best estimate neutron exposure evaluation of the vessel wall is based upon a technique where an analytical model of the irradiation capsule exposure is compared with measured data producing a bias. This bias is projected into the analytical model of exposure in the vessel wall. The techniques used to measure and predict the integrated fast neutron ($E > 1$ Mev) fluxes at the sample locations and the analytical method used to obtain the maximum neutron ($E > 1$ Mev) exposure of the reactor vessel are described in References 105, 106 and 107.

The maximum integrated fast neutron ($E > 1$ Mev) exposure of the vessel at the 1/4 T location has most recently been computed to be 2.64×10^{19} n/cm^2 for Unit 1 and 2.80×10^{19} n/cm^2 for Unit 2 for 40 years operation at 1650 MWt at 87.50 per cent load factor (35 Effective Full Power Years). The computed exposure at the clad/metal interface is 3.95×10^{19} n/cm^2 for Unit 1 and 4.18×10^{19} n/cm^2 for Unit 2 (References 105 and 106).

The predicted bounding $RT_{NDT(PTS)}$ at the end of life ($E > 1$ Mev fluence of $3.95 \times 10^{19}/4.18 \times 10^{19}$ n/cm^2) is 162°F for Unit 1 and 143°F for Unit 2 as computed in References 108 and 109.

To evaluate the NDT temperature shift of welds, heat affected zones and base material for the vessel, test coupons of these material types have been included in the reactor vessel surveillance program described in Section 4.7.2. The methods used to measure the initial NDT temperature of the reactor vessel base plate material are also given in Section 4.7.2.

4.1.7.1 Effect of Aging on Cast Stainless Steel

As a result of investigations (References 118 and 119) conducted both by Westinghouse NES in the United States and by a Westinghouse Licensee in France, it was found that long-time thermal service could severely degrade the Charpy V-notch impact properties of cast AISI 316 stainless steels. Since the Charpy test has long been used as a measure of structural performance for carbon and low alloy steels, this degradation was cause for some concern as to the integrity of PWR primary coolant piping and some reactor internals components made from this type stainless steel.

The AISI 316 cast stainless steel has a duplex microstructure consisting of ferrite islands in an austenite matrix. Thermal aging embrittles the ferrite, but has little effect on the austenite. Therefore, the degree of thermal aging degradation which occurs is proportional to the percentage of ferrite in the material's microstructure.

A preliminary evaluation showed that the ductility of the stainless steel was so high initially that the thermal aging phenomenon did not affect the way it would fail, and therefore the integrity of these components was not of concern even after a lifetime of thermal exposure. These conclusions were examined in more depth by a combined analytical-experimental program completed by Westinghouse. The program was designed to quantify the effect of aging time and temperature on the material behavior, and verify the conclusions reached in the preliminary assessment.

The findings of this program demonstrate that thermal aging is not a problem with regard to the integrity of cast 316 stainless steel piping or other components.

The program has also demonstrated that even the most severe loading condition is unlikely to cause a failure in reactor coolant piping. This finding is in direct support of Westinghouse's position that the combination of earthquake loads and accident loads in the overall evaluation of primary system piping integrity results in undue conservatism and is not warranted.

4.1.8 Reliance on Interconnected Systems

The principal heat removal systems which are interconnected with the Reactor Coolant System are the Steam and Feedwater Systems and the Safety Injection and Residual Heat Removal Systems. The Reactor Coolant System is dependent upon the steam generators, and the steam, feedwater, and condensate systems for decay heat removal from normal operating conditions to a reactor coolant temperature of approximately 350°F. The layout of the system ensures the natural circulation capability to permit plant cooldown following a loss of all reactor coolant pumps.

The flow diagrams of the Steam and Power Conversion System are shown in Figures 11.1-1 through 11.1-8. In the event that the condensers are not available to receive the steam generated by residual heat, the water stored in the condensate and feedwater system may be pumped into the steam generator and the resultant steam vented to the atmosphere. The Auxiliary Feedwater System will supply water to the steam generators in the event that the main feedwater pumps are inoperative. Indication of auxiliary feedwater flow to each steam generator is provided in the control room. The system is described in Section 11.9.

The Safety Injection System is described in Section 6. The Residual Heat Removal System is described in Section 10.

4.3 STEAM GENERATOR AND REACTOR COOLANT PUMPS

4.3.1 Design Basis

The design bases for the Steam Generators and Reactor Coolant Pumps are discussed in Section 4.1.

4.3.2 Steam Generators

4.3.2.1 General Description

4.3.2.1.1 Steam Generators

Each loop of the Reactor Coolant System contains a vertical shell and U-tube steam generator. A steam generator of this type is shown in Figure 4.3-1. Principal (original) 00054
design parameters are listed in Table 4.1-5.

Reactor coolant enters the inlet side of the channel head at the bottom of the steam generator through the inlet nozzle, flows through the U-tubes to an outlet channel and leaves the generator through another bottom nozzle.

The inlet and outlet channels are separated by a partition. Manways are provided to permit access to the U-tubes and moisture separating equipment.

Feedwater to the steam generator enters just above the top of the U-tubes through a feedwater ring. The water flows downward through an annulus between the tube wrapper and the shell and then upward through the tube bundle where part of it is converted to steam.

The steam-water mixture from the tube bundle passes through a steam swirl vane assembly which imparts a centrifugal motion to the mixture and separates the water droplets from the steam. The water spills over the edge of the swirl vane housing and combines with the feedwater for another pass through the tube bundle.

Modifications to the secondary sides of the steam generators were made to eliminate the excessive moisture carryover. Orifice rings were installed on the outlet of each swirl vane assembly and a demister was added in order to limit the moisture carryover to below 0.25% at power levels above 75%. The modifications to the feedwater ring and the blowdown pipe at the steam generator tube sheet also increased the velocity of the water across the tubesheet. The increase in velocity will help in the blowdown of foreign materials and reduce the area where sludge build up occurred in the past.

The steam generator is constructed primarily of low alloy steel. The heat transfer tubes are Inconel. The interior surfaces of the channel heads and nozzles are clad with austenitic stainless steel, and the side of the tube sheet in contact with the reactor coolant is clad with Inconel. The tube to tube sheet joint is welded.

4.3.2.1.2 Steam Generator Support Structure

The steam generator is supported on a structural system consisting of four vertical columns fitted at the top and bottom with a double clevis and pin assembly. The vertical column clevis base plates are bolted to the steam generator support feet and permit movement in the horizontal plane to accommodate reactor coolant pipe thermal expansion. Horizontal restraint is accomplished at two locations. The lower lateral support is located at the support feet and the upper lateral support is located near the center of gravity below the transition cone. This combination of upper and lower supports and included stops and hydraulic shock suppressors limit and control horizontal movement for pipe rupture and seismic effects. The steam generator support structures are further described in Section 12 and shown in Figure 12.2-26.

4.3.2.2 Performance Evaluation

Calculations confirmed that the steam generator tube sheet withstands the loading (which is a quasi-static rather than a shock loading) by loss of reactor coolant. The maximum primary membrane plus primary bending stress in the tube sheet under these conditions is 23,853 psi. This is well below ASME Section III yield strength of 41,112 psi at 660°F. Because the pressure in the primary channel head drops to zero under the condition postulated, no damage results to the channel head.

The rupture of primary or secondary piping has been assumed to impose a maximum pressure differential of 2485 psig across the tubes and tube sheet from the primary side or a maximum pressure differential of 1100 psi across the tubes and tube sheet from the secondary side, respectively. A criterion was established from these conditions under which there was no rupture of the primary to secondary boundary (tubes and tube sheet). This criterion prevents any violation of the containment boundary.

To meet this criterion, it has been established that, under the postulated accident conditions where a primary to secondary side differential pressure of 2485 psig exists, the primary membrane stresses in the tube sheet ligaments, averaged across the ligament and through the tube sheet thickness do not exceed 90% of the material yield stress at the operating temperature. Furthermore, the primary membrane plus primary bending stress in the tube sheet ligaments, averaged across the ligament width at the tube sheet surface location giving maximum stress, do not exceed 135% of the material yield stress at the operating temperature. This criterion is felt to be applicable to abnormal operating circumstances in that it is consistent with the ASME, Nuclear Pressure Vessel Code, Section III rules, Para. N-712.2 for hydrotest limitations.

An examination of stresses under these conditions show that for the case of a 2485 psig maximum tube sheet pressure differential the stresses are within acceptable limits. These stresses together with the corresponding stress limits are given in Table 4.3-1.

The tubes have been designed to the requirements (including stress limitation) of Section III for normal operation, assuming 2485 psig as the normal operation pressure differential.

Hence, the secondary pressure loss accident condition imposes no extraordinary stress on the tubes beyond that normally expected and considered in Section III requirements.

No significant corrosion of the Inconel tubing was expected during the lifetime of the plant. Operating experience has shown that Inconel 600 tubing is susceptible to several degradation mechanisms such as primary water stress corrosion cracking (PWSCC) and secondary side intergranular and stress corrosion cracking and wear. These active and potential corrosion mechanisms are monitored by periodic inspections. 00118

Design Change 00SG03 implemented a heat treatment of the Unit 2 steam generator Rows 1&2 U-tubes to minimize the propensity for PWSCC by reducing the residual stress in the regions of the tubes most prone to attack. Unit 2 Rows 1&2 tubes plugged prior to implementation of this Design Change were not heat treated. 00118

In the case of a primary pressure loss accident, the secondary-primary pressure differential can reach 1100 psi. This pressure differential is less than the primary-secondary design pressure differential (1520 psi) for normal operating conditions. Hence, no stresses in excess of those covered in Section III rules for normal operation are experienced on the tube sheet for this accident case.

ASME Section VIII design curves for iron-chromium-nickel steel cylinders under external pressure indicate a collapse pressure of 2310 psi for tubes having the minimum properties required by the ASTM specification. This indicates a minimum factor of safety of 2.1 against collapse. Collapse tests of 7/8-.050 wall straight tubes at room temperature indicate actual tube strengths are significantly higher than specification and a collapse pressure of 6,000 psi was recorded for the straight tube. The Code charts indicate a collapse pressure of 2740 psi for this tube. The difference is attributed to the fact that the yield strength of the tube tested was 44,000 psi and the Code charts are based on a yield strength of approximately 29,000 psi at room temperature.

Consideration has been given to the superimposed effects of secondary side pressure loss and the DBE loading. The fluid dynamic forces on the internal components affecting the primary-secondary boundary (tubes) have been considered as well. For this condition, the criterion is that no rupture of primary to secondary boundary (tubes and tube sheet) occurs.

For the case of the tube sheet, the DBE loading contributes an equivalent static pressure loading over the tube sheet of less than 10 psi (for vertical shock). Such an increase is small when compared to the pressure differentials (up to 2485 psig) for which the tube sheet is designed. Under horizontal shock loading of the DBE the stresses are less than those for 1.0g gravity loading experienced in a horizontal position, which the design can readily accept.

The fluid dynamic forces on the internals under secondary steam break accident conditions indicate, in the most severe case, that the tubes are adequate to constrain the motion of the baffle plates with some plastic deformation, but boundary integrity is maintained.

The ratios of the allowable stresses (based on an allowable membrane stress of 0.9 of the nominal yield stress of the material) to the computed stresses are summarized in Table 4.3-2.

The evaluation of Westinghouse steam generator tube sheets was performed according to rules of the ASME Boiler and Pressure Vessel Code for Nuclear Vessels, Section III, Article 4 - Design. The design criteria considered encompassed consideration of both steady state, transient and emergency operations specified in the Equipment Specification. Due to the complex nature of the tube-tubesheet-shell-head structure, the analysis of the tubesheet requires the application of results of related research programs (such as the design data on perforated plates resulting from PVRC programs) and the utilization of current techniques in computer analysis, the application of which is verified by comparison of analytical and experimental results for related equipment.

Examination of the introductory paragraph I-900 of the ASME Boiler and Pressure Vessel Code, Section III - Nuclear Vessels, reveals a precise explanation that consideration may be given to the stiffening effect of tubes in perforations, and staying action of the tubes if applicable, effect of stiffening on the plate stress levels, etc. Furthermore, it is noted that the stress analysis methods in Appendix I of Section III are described as accepted techniques for obtaining solutions to problems for which these procedures are applicable. It allows and requires use of other valid analytical or experimental techniques where necessary.

Although the Nuclear Pressure Vessel Code Article I-9 provides for rules and techniques in analysis of perforated plates, it should be noted that the stress intensity levels for perforated plate are given for triangular perforation arrays. Westinghouse tube sheets contain square hole arrays. Hence, Westinghouse utilizes its own data and that obtained from PVRC research in square array perforation patterns for development of similar charts for stress intensity factors and elastic constants. The resulting stress intensity levels and fatigue stress ranges are evaluated according to the stress limitation of the Code.

The Westinghouse analysis of the steam generator tubesheets is included as part of the Stress Report requirement for Class A Nuclear Pressure vessels. The evaluation was based on the stress and fatigue limitations outlined in Article 4-Design of Section III. The stress analysis techniques utilized include all factors considered appropriate to conservative determination of the stress levels utilized in evaluation of the tubesheet complex. The analysis of the tubesheet complex includes the effect of all appurtenances attached to the perforated region of the tubesheet considered appropriate to conservative analysis of stress for evaluation of the basis of Section III stress limitations. The evaluation involves the heat conduction and stress analysis of the tubesheet, channel head, secondary shell structure for particular steady design conditions for which Code stress limitations are to be satisfied and for discrete points during transient operation for which the temperature/pressure conditions must be known to evaluate stress maxima and minima for fatigue life usage. In addition, limit analyses are performed to determine tubesheet capability to sustain emergency operating conditions for which elastic analysis does not suffice. The analytic techniques utilized are computerized and significant stress problems are verified experimentally to justify the techniques where possible.

Generally, the analytic treatment of the tube-tubesheet complex includes determination of elastic equivalent plate stress within the perforated region from an interaction analysis utilizing effective elastic constants appropriate to the nature of the perforation array. For the perforated region of the tubesheet the flexural rigidity is based on studies of behavior of plates with square hole arrays utilizing techniques such as those reported by O'Donnell (Reference 23), Mahoney (Reference 24), Lemcoe (Reference 25), and others. Similarly, stress intensity factors are determined for square hole arrays using the combined equivalent plate interaction forces and moments applied to results of photo-elastic tests of model coupons of such arrays as well as verification using computer analysis techniques such as "Point Matching" or "Collocation". The stress analysis considers stress due to symmetric temperature and pressure distribution as well as asymmetric temperature distribution due to temperature drop across the tubesheet divider lane.

The fatigue analysis of the complex is performed at potentially critical regions in the complex such as the junction between tubesheet and channel head or secondary shell as well as at many locations throughout the perforated region of the tubesheet. For the holes for which fatigue evaluation is done, several points around the hole periphery are considered to assure that the maximum stress excursion has been considered. The fatigue evaluation is computerized to include stress maxima-minima excursions considered on the intra-transient basis. Under Modification 96SG01, tubesheet bore hole sizes were increased slightly in 12 Steam Generator in order to remove 5 sleeve samples. This resulted in a small but acceptable increase in fatigue usage factors for 12 Steam Generator. Under Modification 96SG04, tubesheet bore hole size was increased slightly in 12 Steam Generator to remove 1 sleeve sample which resulted in an acceptable increase in fatigue usage factors for 12 Steam Generator

The evaluation of the tube-to tubesheet juncture of Westinghouse PWR System steam generators is based on a stress analysis of the interaction between tube and tubesheet hole for the significant thermal and pressure transients that are applied to the steam generator in its predicted histogram of cyclic operation. The evaluation is based on the numerical limits specified in the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

Of importance in the analysis of the interaction system is the behavior of the tube hole, where it is recognized that the hole behavior is a function of the behavior of the entire tubesheet complex with attached head and shell. Hence, the output of the tubesheet analysis giving equivalent plate stresses in the perforated region is utilized in determining the free boundary displacements of the perforation to which the tube is attached.

Analysis of the juncture for the fillet-type weld utilized in the Westinghouse steam generator design has been made with consideration of the effect of the rolled-in joint in the weld region as well as with the conservative assumption that the tube flexure relative to the perforation is not inhibited with the rolled-in effect.

The major concern in fatigue evaluation of the tube weld is the fatigue strength reduction factor to be assigned to the weld root notch. For this reason, Westinghouse has conducted low-cycle fatigue tests of tube material samples to determine the fatigue strength reduction factor and applied them to the analytic interaction analysis results in accordance with the accepted techniques in the Nuclear Pressure Vessel Code for Experimental Stress Analysis. The fatigue strength reduction factor determined therefrom are not different from that reported in the well known paper on the subject by O'Donnell and Purdy (Reference 26). An actual tubesheet joint contained in a tubesheet has been successfully tested experimentally under thermal transient conditions much more severe than that achieved in anticipated power plant operation.

The above statement refers to tests of actual tubesheet joints (fillet welded) under thermal fatigue conditions which exercise the weld root notch. The tubesheet joint will be exposed under LOCA to the maximum pressure possible on the secondary side with loss of primary pressure. As explained before, the secondary-primary pressure differential can reach 1100 psi. This differential is far below that differential 2485 psig under loss of secondary pressure for which the tube sheet joint is designed.

A wide range of computational tools are utilized in these solutions including finite element, heat conduction and thin shell computer solutions. In addition, analysis techniques have been verified by photoeleastic model tests and strain gaging of prototype models of an actual steam generator tubesheet.

Finally, in order to evaluate the ultimate safety of structural complex, a computer program for determining a lower-bound pressure limit for the complex based on elastic-plastic analysis has been developed and applied to the structure. This was verified by a strain gage steel model of the complex tested to failure.

In all cases evaluated, the Westinghouse steam generator tubesheet complex met the stress limitations and fatigue criteria specified in Article 4 of the Code as well as emergency condition limitations specified in the Equipment Specifications or anticipated otherwise.

In this way, the tube-tubesheet integrity of a Westinghouse steam generator is demonstrated under the most adverse conceivable conditions resulting from a major breach in either the primary or secondary system piping.

Tabulations of significant results of the tubesheet complex are shown in Tables 4.3-3 through 4.3-10 and Figures 4.3-8 through 4.3-10. Figure 4.3-11 denotes the primary-secondary boundary components shell locations.

Following an incident of a steam generator tube rupture at the North Anna plant, the USNRC issued NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes." This Bulletin requested licensees operating plants with Westinghouse steam generators employing carbon steel support plates to take actions to minimize the potential for steam generator tube rupture caused by a rapidly propagating fatigue crack. Prairie Island initiated enhanced methods of leak detection to recognize potential fatigue cracking problems in a more timely manner, and initiated a study of the potential for fatigue cracking of steam generator tubes in the area of the steam generator top tube support plate. The results of this analysis have been reported in WCAP-11787, (Reference 77). "Prairie Island Units 1 and 2: Evaluation for Tube Vibration Induced Fatigue." This analysis showed based on the design of the Prairie Island steam generators, operating conditions and the design and placement of the steam generators anti-vibration bars, no modifications or precautionary tube plugging of any kind is required for the Prairie Island steam generators. By letter dated December 8, 1988, (Reference 78) the USNRC concluded in a safety evaluation that the analysis presented in WCAP 11787 fully resolves the issues identified in NRC Bulletin 88-02. The NRC Safety Evaluation is based on an NSP commitment to update stress ratio and fatigue usage calculations in the event of any significant changes to the steam generator operating parameters.

Under Modification 95L486 and an amendment to the Prairie Island Technical Specifications (Reference 102), analysis and testing were completed which justified shifting the primary-to-secondary boundary from the tube to tubesheet weld to a hard roll expansion meeting the F* alternate repair criteria.

4.3.2.3 Steam Generator Tube Sleeving

The NRC approved an amendment to the Prairie Island Technical Specifications (Reference 56) which allowed tube sleeving to be used as a method for repairing steam generator tubes. Three methods of tube sleeving were approved for use; mechanical hard rolled sleeves, brazed sleeves and welded sleeves. These methods are described in References 57, 58 and 59. Only the Combustion Engineering leak tight tubesheet sleeve has been installed using the 1985 Technical Specification change. Improvements in sleeve installation and inspection technology were implemented under modification 95SG01 (Reference 104). In 1997, the NRC approved an amendment to the Prairie Island Technical Specification (Reference 113) which included additional improvements in sleeve installation technology and examination acceptance criteria as well as alternative sleeving configurations. These methods and configurations are described in References 114 and 115. Implementation of the tubesheet sleeve with lower hard roll joint was done under design change 97SG04. In 1999, additional sleeve cleaning techniques were approved and the repair criteria were decreased by amendment to the Prairie Island Technical Specification (Reference 123). The sleeving process and repair criteria are described in Reference 124.

4.3.2.4 F-Star Alternate Repair Criteria

The NRC has approved an amendment to the Prairie Island Technical Specifications (Reference 102) which allows tubes to remain in service if the required length of hard roll expansion is intact above the highest degradation in the tubesheet crevice region.

4.3.2.5 Elevated F-Star Alternate Repair Criteria

License Amendments 137 and 128 (Reference 116) allow tubes to remain in service if the required length of an elevated hard roll expansion is intact above the highest degradation in the tubesheet crevice region. Elevated F-Star can be used above the mid-plane of the tubesheet.

4.3.2.6 Voltage Based Repair Criteria for Steam Generator Tubes

License Amendments No. 133 and 125 approved application of voltage-based repair criteria in accordance with NRC Generic Letter 95-05 for steam generator tubes with degradation due to predominantly axially oriented outside diameter stress corrosion cracking confined within the tube to tube support plate locations. The amendment also reduced the reactor coolant system secondary leakage limit through any one steam generator to 150 gallons per day. The leakage limit is applicable to both units.

In accordance with the guidance provided in Generic Letter 95-05, radiological dose calculations were performed at the Exclusion Area Boundary (EAB), for the Low Population Zone (LPZ) and in the control room for the MSLB (outside of containment and upstream of the main steamline isolation valve). The limiting acceptance criteria were for the MSLB with an accident-initiated iodine spike and are General Design Criteria 19 (1971) guideline values. The more conservative NRC calculation documented in the Safety Evaluation for the license amendments established a limiting leak rate due to the control room dose associated with the 30 rem thyroid limit of 1.42 gallons per minute (gpm) at 578 degrees F which is 1.0 gpm at 70 degrees Fahrenheit. Leakage in the intact loops is equal to the new Technical Specification normal operation leakage limit of 150 gallons per day. The Technical Specification reactor coolant and secondary coolant dose equivalent Iodine-131 activity limits of 1.0 microcuries per gram and 0.1 microcuries per gram, respectively, are used in the analysis for establishing the initial radioactivity conditions. This analysis is specific to a Main Steam Line Break outside of containment to support the voltage based repair criteria. The dose analysis uses the methodology associated with the Standard Review Plan 15.1.5, Appendix A. Both the pre-existing iodine spike and the accident-initiated iodine spike cases were evaluated.

4.3.3 Reactor Coolant Pumps

4.3.3.1 General Description

4.3.3.1.1 Reactor Coolant Pumps

Each reactor coolant loop contains a vertical single stage centrifugal pump which employs a controlled leakage seal assembly. A view of a controlled leakage pump is shown in Figure 4.3-2 and the principal design parameters for the pumps are listed in Table 4.1-6. The reactor coolant pump performance and NPSH characteristics are shown in Figure 4.3-3. The performance characteristic is common to all of the higher specific speed centrifugal pumps and the 'knee' at about 45% design flow introduces no operational restrictions since the pumps operate at full flow.

Both reactor coolant pumps will be in operation when the reactor is critical (except during low power physics test) to provide core cooling in the event that a loss of flow occurs. Cladding damage and release of fission products to the reactor coolant will not occur in the event of loss of both pumps from 100% power since the minimum calculated DNBR remains above the applicable limit (see Section 14.4.8). At power above 10%, an automatic reactor trip will occur if flow from either pump is lost. Below 10% power, a shutdown under administrative control will be made if flow from either pump is lost.

All the pressure bearing parts of the reactor coolant pump were analyzed in accordance with Article 4 of the ASME B&PV Code, Section III. This included the casing, the main flange and the main flange bolts. The analysis included pressure, thermal and cyclic stresses, and these were compared with the allowable stresses in the Code.

Mathematical models of the parts were prepared and used in the analysis which proceeds in two phases.

a. In the first phase, the design was checked against the design criteria of the ASME Code, with stress calculations using the allowable stress at design temperature. By this procedure, the shells were profiled to attain optimum metal distribution with stress levels adequate to meet the more exacting requirements of the second phase.

b. In the second phase, the interacting forces needed to maintain geometric capability between the various components were determined, and applied to the components along with the external load, to determine the final stress state of the components. This stress was also used in the fatigue analyses. These results were finally compared with the Code allowable values.

There were no other sections of the Code which were specified as areas of compliance, but where Code methods, allowable stresses, fabrication methods, etc., were applicable to a particular component, these were used to give a rigorous analysis and conservative design.

Stress Analysis Reports were prepared on these components as described in Section 4.1. These reports include the calculation of stress intensities and a summary of fatigue usage factors. These reports are a part of the plant documentation on file with the Licensee.

Reactor coolant is pumped by the impeller attached to the bottom of the rotor shaft. The coolant is drawn up through the impeller, discharged through passages in the diffuser and out through a discharge nozzle in the side of the casing. The rotor-impeller can be removed from the casing for maintenance or inspection without removing the casing from the piping. All parts of the pump in contact with the reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials.

The pump employs a controlled leakage seal assembly to restrict leakage along the pump shaft, as well as a secondary seal which directs the controlled leakage out of the pump, and a third seal which minimizes the leakage of water and vapor from the pump into the containment atmosphere.

A portion of the high pressure water flow from the charging pumps is injected into the reactor coolant pump (RCP) between the impeller and the controlled leakage seal. Part of the flow enters the Reactor Coolant System through a labyrinth seal in the lower pump shaft to serve as a buffer to keep reactor coolant from entering the upper portion of the pump. The remainder of the injection water flows along the drive shaft, through the controlled leakage seal, and finally out of the pump. A very small amount which leaks through the secondary seal is also collected and removed from the pump.

Component cooling water is supplied to the RCP thermal barrier heat exchanger and the motor bearing cooler. RCP operation is permitted with loss of seal injection provided component cooling water is available to the RCP Thermal Barrier Heat Exchanger (WCAP 10541, Revision 2).

The squirrel cage induction motor driving the pump is air cooled and has oil lubricated thrust and radial bearings. A water lubricated bearing provides radial support for the pump shaft. In addition, pump vibration is monitored as a means of early detection of mechanical abnormalities.

4.3.3.1.2 Pump Support Structure

The reactor coolant pump is supported by a structural system consisting of three vertical columns fitted at the top and bottom with double clevis and pin assembly and a system of stops. The vertical column clevis base plates are bolted to the pump support feet and permit movement in the horizontal plane to accommodate reactor coolant pipe thermal expansion. Horizontal restraint is accomplished by a combination of the tie rods and stops which limit horizontal movement for pipe rupture and seismic effects. The reactor coolant pump support structures are further described in Section 12.

The reactor coolant pumps and other components are bolted down to foundations by means of high strength bolts and nuts. Double nuts or lock nuts are furnished to prevent loosening of the nuts due to vibration.

Pinned or bolted parts of support components that are subject to pivotal action or articulation due to temperature movements are designed as non-loosening devices.

The reactor coolant pump is mounted and anchored at the three pump casing support brackets to the support column pedestal by means of high strength threaded rods at each support point.

All clevis pins are held in place by means of retainer plates bolted to each end of each pin with four high strength bolts to prevent the pins from becoming dislodged.

With the positive bolting devices provided, procedures for the surveillance of loose bolts during normal operation is not required.

4.3.3.2 Performance Evaluation

4.3.3.2.1 Reactor Coolant Pump

An extensive test program has been conducted for several years to develop the controlled leakage shaft seal for pressurized water reactor applications. Long term tests have been conducted on less than full scale prototype seals as well as on full size seals.

For conditions of the loss-of-coolant accident it is not considered likely that the reactor coolant pump will accelerate significantly. A program to determine with certainty the behavior of reactor coolant pumps for breaks in either the suction or discharge piping near the pumps was conducted by Westinghouse in cooperation with Purdue University. Westinghouse's analytical study program was confirmed by evaluating obtained data from the test program conducted at Purdue University.

The test program established pump head loss and torque under locked rotor, free spinning and reverse flow conditions. The test was conducted with a scale model of the 93A pump with air as fluid at a pressure of 15 to 60 psia.

Flow, simulating blowdown conditions range from 500% to -100% of normal. The results of the analytical study and test program was discussed with the NRC on a generic basis.

Precautionary measures, taken to preclude missile formation from reactor coolant pump components, assure that the pumps will not produce missiles under any anticipated accident condition.

The reactor coolant pumps run at about 1200 rpm and may operate briefly at overspeeds of 109% during loss of load. For conservatism the motors were designed in accordance with NEMA standards for operation at a maximum speed of 125% of rated speed.

Each component of the reactor coolant pump motors was analyzed for missile generation. Any fragments would be contained by the heavy stator. The same conclusion applies to the pump impeller because the small fragments that might be ejected would be contained by the heavy casing.

The reactor coolant pump flywheels are shown in Figure 4.3-4. As for the pump motors, the most adverse operating condition of the flywheels is visualized to be the loss-of-load situation. The following conservative design-operation conditions preclude missile production by the pump flywheels. The wheels are fabricated from rolled, vacuum-degassed, ASTM A-533 Grade B Class 1 steel plates. Flywheel blanks are flame-cut from the plate, with allowance for exclusion of flame-affected metal. A minimum of 3 Charpy tests are made from each plate parallel (RW, longitudinal) and normal (WR, transverse) to the rolling direction. An NDTT less than + 10°F is specified. Westinghouse has a great deal of experience and data in determining fracture toughness of A533 Grade B Class 1 steel utilizing fracture mechanics specimens as well as Charpy-V specimens. Fracture mechanics specimens up to 12-inches in thickness have been tested to characterize A533 Grade B material. From Westinghouse's experience and those of others found in the literature, an empirical relationship can be established for Charpy-V data and fracture toughness data. The finished flywheels are subjected to 100% volumetric ultrasonic inspection. The finished machined bores are also subjected to magnetic particle, or liquid penetrant examination.

Acceptability of flywheel material for NSP, in comparison to Safety Guide 14 toughness criteria, can be determined by the following two steps:

a. Establish a reference curve describing the lower bound fracture toughness behavior for the material in question.

b. Use Charpy impact energy values obtained in certification tests at 10°F to fix position of the heat in question on the reference curve.

The following supplier certification data shows the Charpy V-notch test results at +10°F for NSP flywheels:

5 in. Thick Plates
Heat No. 06458

	1	2	3
Slab 2C Transverse direction (ft-lbs)	44	44	50
(2 plates) Longitudinal direction (ft-lbs)	69	65	53
Slab 2E Transverse direction (ft-lbs)	74	74	62
(2 plates) Longitudinal direction (ft-lbs)	80	83	77

8 in. Thick Plates
Heat No. 07090 Slab No. 3 (one plate)

	1	2	3
Transverse direction (ft-lbs)	65	35	44
Longitudinal direction (ft-lbs)	58	57	69

Heat No. 7442 Slab No. 1 (3 plates)

	1	2	3
Transverse direction (ft-lbs)	53	58	52
Longitudinal direction (ft-lbs)	93	79	71

A lower bound fracture toughness reference curve (see Figure 4.3-5) was constructed from dynamic fracture toughness data generated by Westinghouse (Reference 15) on A-533 Grade B Class I steel. All data points are plotted on the temperature scale relative to the NDT temperature. The construction of the lower bound below which no single test point falls, combined with the use of dynamic data when flywheel loading is essentially static, together represent a large degree of conservatism.

The applicability of a 30 ft-lb Charpy energy reference value was derived from sections on Special Mechanical Property Requirements and Tests in Article 3, Section III of the ASME Boiler and Pressure Vessel Code. The implication is that the test temperature lies a safe margin above NDTT. The NSP flywheel plates exhibit an average value greater than 30 ft-lbs in the weak direction and, therefore, met the specific requirement C.1.a stated in Safety Guide 14 that NDTT must be no higher than 10°F. Making the conservative assumption that all materials in compliance with the Code requirements are characterized by an NDT temperature of 10°F, one is able to reassign the "zero" reference temperature position in Figure 4.3-5 a value of 10°F

Flywheel operating temperature at the surface is 120°F. The lower bound toughness curve indicates a value of 116 ksi-in$^{1/2}$ at the (NDTT + 110) position corresponding to operating temperature. Safety Guide 14 requirement C.1.c is fulfilled with considerable margin for safety.

By assuming a minimum toughness at operating temperature in excess of 100 ksi- in$^{1/2}$, it can be seen by examination of the Corten and Sailors correlation in Figure 4.3-6 (Reference 117) that the Cy upper shelf energy must be in excess of 50 ft-lb, therefore, the Safety Guide 14 requirement C.1.b, that the upper shelf energy must be at least 50 ft-lb, is satisfied.

Based on the above discussion, the flywheel materials meet the Safety Guide 14 toughness criteria on the basis of supplier certification data.

Justification for the 125% overspeed has been given above. The overspeed test was conducted in accordance with the NEMA Standards Publication for Motors and Generators, Part 20, Paragraph MG 1-20.44 with the flywheel installed on the motor.

The Safety Guide 14 requirements for inservice inspection of reactor coolant pump flywheels include the following information:

a. An in-place ultrasonic volumetric examination of the areas of higher stress concentration at the bore and key way at approximately 3 year intervals, during the refueling or maintenance shutdown coinciding with the inservice inspection schedule as required by the ASME Boiler and Pressure Vessel Code Section XI.

b. A surface examination of all exposed surfaces and complete ultrasonic volumetric examination at approximately 10 year intervals, during the plant shutdown coinciding with the inservice inspection schedule as required by the ASME Boiler and Pressure Vessel Code Section XI. Removal of the flywheel is not required to perform these examinations.

c. Examination procedure and acceptance criteria in conformance with the requirements specified in Safety Guide 14 requirement C.1.d.

To perform an inspection of this nature the only required disassembly would be the removal of the motor cover plate and removal of four plugs from the flywheel. This would permit the partial ultrasonic examination of the keyways for evidence of cracking developing at the corners by the insertion of a special ultrasonic search unit into four holes drilled through the flywheel.

These design-fabrication techniques yield flywheels with primary stress at operating speed (shown in Figure 4.3-7) less than 50% of the minimum specified material yield strength at room temperature (100 to 150°F). The stress resulting from the press fit of the flywheel on the shaft is less than 2000 psi at zero speed, but this stress becomes zero at approximately 600 rpm because of radial expansion of the hub. Bursting speed of the flywheels has been calculated on the basis of Griffith-Irwin's results (References 16 and 17) to be 3900 rpm, more than three times the operating speed.

A fracture mechanics evaluation was made on the reactor coolant pump flywheel. This evaluation considered the following assumptions:

a. Maximum tangential stress at an assumed overspeed of 125%.

b. A crack through the thickness of the flywheel at the bore.

c. 400 cycles of startup operation in 40 years.

Using critical stress intensity factors and crack growth data attained on flywheel material, the critical crack size for failure was greater than 17 inches radially and the crack growth rate was 0.030″ to 0.060″ per 1000 cycles.

An ultrasonic inspection capable of detecting at least 1/2″ deep cracks from the ends of the flywheel is more than adequate as part of a plant surveillance program. The inservice inspection program of the flywheel is given in the Technical Specifications.

Installation of New RCP Internals in Unit 2 #21 RCP Casing

The #21 Reactor Coolant Pump internals were replaced with internals obtained from D. C. Cook Nuclear Plant. The D. C. Cook Nuclear Plant RCP has the same style as the original #21 RCP but with slightly different design than the original and therefore had slightly different flow characteristics.

Westinghouse has determined the potential flow imbalance by replacing the #21 RCP with a spare from D. C. Cook Nuclear Plant. It was found that no significant flow imbalance will occur as a result of this pump replacement. Before replacement, the Loop A (#21 RCP) pump flow was approximately 0.6% below Loop B (#22 RCP). The D. C. Cook spare will shift flow in Loop A to approximately 1.0% greater than Loop B.

RCS Flow Measurement Testing was performed to verify that the new #21 RCP could meet the design requirements for flow and for flow coastdown following a Reactor Coolant Pump trip (See Section 14.4.8). All the data collected from the test is summarized in Table 4.3-11. The flow results collected compared very closely with the Westinghouse predicted flows for the replacement pump. The data clearly shows the change in the flows from previous measurements (Table 4.3-12) taken in previous years. The data was also checked for difference in flow from Loop A to Loop B at the total core flow. In both cases the acceptance criteria were met. The increase in core pressure drive, which is the driving force for baffle jetting, is on the order of 0.2 psi. This increase is on a total pressure drop of 24.6 psi so that any increase in baffle jetting could be considered small.

The use of the D. C. Cook impeller in the #21 RCP has no effect on the LOCA analysis input parameters.

4.3.3.2.2 Trip of Reactor Coolant Pumps During LOCA

In response to NUREG-0737, ITEM II.K.3.5, Westinghouse, in support of the Westinghouse Owners' Group, has performed 1) an analysis of delayed reactor coolant pump trip during small-break LOCA's and 2) test predictions of LOFT experiments L3-1 and L3-6. This analysis and test predictions are documented in References 18, 19, 20 and 21. Based on the Westinghouse analysis, the prediction of the LOFT experiment L3-6 results using the Westinghouse analytical model, and Westinghouse simulator data related to operator response time, the Westinghouse and NSP position is that automatic reactor coolant pump trip is not necessary since sufficient time is available for manual tripping of the pumps.

Generic Letters 83-10 c and d contained NRC staff requirements for resolution of NUREG-0737, Item II.K.3.5. Two Westinghouse Owners Group letters OG-117 dated March 12, 1984 entitled "Justification of Manual RCP Trip for Small Break LOCA Events," and OG-110 dated December 1, 1983 entitled "Evaluation of Alternate RCP Trip Criteria," fulfilled the requirements of the Generic Letters. This methodology was approved by the NRC (Reference 53). Revision 1 to the WOG Emergency Response Guidelines contains associated procedure revisions which have been incorporated into Prairie Island procedures.

Procedures based on the Westinghouse Owners Group Emergency Response Guidelines have been implemented at Prairie Island (see Section 13.7). The RCP trip criteria adopted in the Prairie Island procedures not only assures RCP trip for all losses of primary coolant for which trip is considered necessary but also permits RCP operation to continue during most non-LOCA accidents, including steam generator tube rupture events up to the design basis double-ended steam generator tube rupture. The RCP trip criteria is based on

Reactor Coolant System Pressure. Two setpoints have been determined, one for normal containment conditions and one for adverse containment conditions. The use of two setpoints permits the setpoint to be lower (less transmitter uncertainty) for the accidents that don't affect the containment environment. This assures that RCPs will not need to be tripped for non-LOCA events like steam generator tube ruptures, which don't require a RCP trip but will be tripped for small break LOCA's requiring a RCP trip (References 53 and 54).

The NRC staff found the treatment of the RCP trip criteria to be acceptable in a Safety Evaluation Report (SER) dated October 8, 1986. The SER discussed the uncertainties associated with the setpoint selection and operator training including recommendations in detail.

4.3.3.2.3 Effect of Loss of AC Power on Pump Seals

During normal operation, seal injection flow from the chemical and volume control system is provided to cool the RCP seals and the component cooling water system provides flow to the thermal barrier heat exchanger to limit the heat transfer from the reactor coolant to the RCP internals. In the event of loss of offsite power (LOOP), the RCP motor is deenergized and both of these cooling supplies are terminated. However, the diesel generators are automatically started and component cooling water to the thermal barrier heat exchanger is automatically restored. The load rejection and delayed reapplication logic will restart the component cooling water pumps within 15 seconds after the loading sequence starts. Charging pumps will be manually restarted. The NRC has concluded that this arrangement, as described in the submittal of December 29, 1981, and subsequent telecom of June 11, 1982, adequately responds to the requirements of NUREG-0737, Item II.K.3.25 (Reference 33). 00091

Westinghouse has completed a detailed investigation of the Westinghouse reactor coolant pump (RCP) seal performance following a postulated loss of all seal cooling event (WCAP 10541 Revision 2). Since the number and duration of the loss of all seal cooling events experienced to date is very limited, an integrated testing and analysis program was undertaken. Analyses of the seal assembly structural response, the seal system thermal-hydraulic response, the Reactor Coolant System thermal-hydraulic response, and the containment response to a small break LOCA resulting from seal failure were performed. The following are the major results and findings of the analyses:

- Detailed thermal/stress structural analysis of the complete cartridge seal design and of the No. 2 seal of the 8-inch seal design were performed. The analysis results provided information on the conditions to which the secondary sealing elastomers will be exposed and the mechanical deformation of the seal system components.

- Detailed thermal hydraulic analyses were performed which demonstrated that the change in fluid properties flowing through the seal system, the mechanical change in the seal system components due to higher temperature water, and the fluid flow through the leakoff line portion of the seal cooling support systems all interact to result in expected leakage rates which are higher than normal, but which are acceptably low.

- Assuming the integrity of the secondary sealing elastomers, the results of detailed thermal hydraulic two-phase flow analyses indicated that the leakage flow rate through the RCP seals and support systems would be limited to 21.1 gpm/pump or less. Thus, the total maximum leakage for two RCPs is within the capacity of one charging pump to make up to the Reactor Coolant System. This calculated leak rate provides more than sufficient time 00091
to restore normal make up to the Reactor Coolant System. Subsequent cooling of the seals, following an extended loss of cooling, is performed using leakage past the seals and cooldown of the Reactor Coolant System.

- The effect of RCS fluid discharge from the seal system to containment significantly greater than best estimate leakage rates does not approach the containment design limits. The effect of RCS temperatures on the pump internals in conjunction with the containment heatup was evaluated and determined to have no detrimental effects on the seal leakage rate during the event.

- The forces on the hydrostatic No. 1 RCP seal were shown to provide extremely large net restoring forces which would reestablish the seal ring at an equilibrium gap position for any hypothesized displacement. No large discontinuities or variation in the restoring force or fluid quality distribution were observed as the inlet flow conditions varied from subcooled flow to high quality two-phase flow.

Based significantly on the results of the analytical efforts, two test programs were conducted to determine the response of the seal elastomers and the integrated response of a total full scale seal system to a loss of all seal cooling resulting from a loss of all AC. The significant testing results and findings were:

- Extensive extrusion testing of the secondary sealing elastomers has been performed. The testing indicated that the previous O-ring material had the capability to survive, but could not always be relied upon to survive the loss 00091
of all seal cooling conditions. Subsequently, alternate materials with better properties have been installed which significantly improve seal survivability at these conditions.

- A full scale demonstration test of a 7-inch RCP seal package was conducted in France with transient boundary conditions simulating the fluid properties at the seal inlet during a loss of all seal cooling resulting from the loss of all AC power. The 7-inch seal design performed satisfactorily and limited the leakage to less than 17 gpm following the initial heatup transient.

- During the test of the 7-inch seal system, the seals exhibited the generic tendencies postulated by the analysis of the 8-inch and 8-inch cartridge seal packages by limiting the leakage flow. The 8-inch and 8-inch cartridge seal No. 1 seals would be expected to exhibit a less severe initial heatup 00091 transient, which would remain brief in duration, than the 7-inch No. 1 seal.

Based on the results of the analyses and testing programs the Westinghouse hydrostatic seal design was found to provide for a stable and acceptable response following a loss of all seal cooling event. Specifically;

- The previous seal system components were shown to have considerable capacity to survive the low probability loss of all seal cooling event, which is 00091 beyond the design basis.

- Improved secondary sealing elastomers, which have significant capability to survive the conditions associated with this event, were demonstrated to be feasible and have been developed. Replacement seals using improved elastomer materials have been installed and are used during normal pump 00091 seal maintenance

- The use of improved elastomers significantly increases the probability that RCS leakage through the seal will be sufficiently small so that core uncovery will not occur prior to the time required to recover from the event. 00091

- Quantification of the leakage resulting from the loss of all seal cooling has shown that the leakage rates are skewed toward lower values which greatly increase the time to core uncovery and dramatically reduces the core melt frequency when compared to earlier models. 00091

The results and consequences of the loss of all seal cooling to a Westinghouse reactor coolant pump will result in low and acceptable pump seal leakage rates. These low leakage rates will enable adequate time to restore RCS makeup capability and therefore 00091 prevent core uncovery.

4.6 PIPING, INSTRUMENTATION AND VALVES

4.6.1 Description

4.6.1.1 Piping

The general arrangement of the Reactor Coolant System piping is shown on the plant layout drawings in Section 12. Piping design data are presented in Table 4.1-7.

The reactor coolant piping layout is designed on the basis of providing "floating" supports for the steam generator and reactor coolant pump in order to absorb the thermal expansion from the fixed or anchored reactor vessel.

The austenitic stainless steel reactor coolant piping and fittings which make up the loops are 29 in. ID in the hot legs, 27-1/2 in. ID in the cold legs and 31 in. ID between each loop's steam generator outlet and its reactor coolant pump suction. Nitrogen has been added to enhance the strength of only the Unit 1 reactor coolant loop pipe. Unit 2 piping is centrifugally cast ASTM A351 CPF8M material. The Unit 1 reactor coolant loop pipe material is seamless, forged, ASTM A376 Type 316. To improve the mechanical properties of this material, controlled nitrogen was added in conformance with ASME Code Case 1423. Based on tests performed on similar material, it is concluded that the nitrogen addition does not adversely affect the corrosion resistance of this material in the PWR coolant environment. This material is not "furnace sensitized."

Smaller piping, including the pressurizer spray and relief lines, drains and connections to other systems are austenitic stainless steel. Unisolable sections of piping connected to the Reactor Coolant System have been evaluated for potential temperature distributions or oscillations which could cause unacceptable thermal stresses. This evaluation determined that leakage past the pressurizer auxiliary spray control valves could result in unacceptable thermal stresses in the downstream pressurizer auxiliary spray piping. Temperature monitoring has been installed on the auxiliary spray lines for both Unit 1 and Unit 2 for detection of piping thermal cycling due to valve leakage into the Reactor Coolant System. The monitoring program, with a discussion of the exceedance criteria, are described in NSP's response to Bulletin 88-08 (Reference 94) and was found acceptable by the NRC (Reference 101). The evaluation and the results of the initial NDE examinations are described in NSP's responses to Bulletin 88-08 (References 80 and 83).

All joints and connections are welded except for stainless steel flange connections to the pressurizer relief tank and the connections at the safety valves.

Thermal sleeves are installed at the following locations where high thermal stresses could otherwise develop due to rapid changes in fluid temperature during normal operational transients:

a. Return line from the residual heat removal loop.

b. Both ends of the pressurizer surge line.

c. Pressurizer spray line connection to the pressurizer.

d. Charging line connections.

4.6.1.2 Valves

All valve surfaces in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials. Connections to stainless steel piping are welded.

The potential for valve stem leakage is reduced by monitoring valves for leakage, early detection, and repacking without leakoff per current maintenance standards. Post maintenance testing, inservice system leak tests, periodic walkdown surveillances, radiation monitoring, fluid inventory surveillances, and leakage control programs address early leak detection and repair.

The only valves manufactured outside the United States were valves MV32195 (MV32197)/8000A and MV32196(MV32198)/8000B on the Pressurizer Relief Lines and RC-2-1(2RC-2-1)/8001A and RC-2-2(2RC-2-2)/8001B on the Bypass Manifold. These were fabricated by Velan Engineering Company, Montreal, Canada.

The manufacturer's qualifications are as follows:

a. Velan's capabilities are evaluated in the same manner as domestic plants to assure that they are able to manufacture quality materials.

b. All specifications used in procurement are identical to the specifications utilized for domestic procurement.

c. Extensive quality assurance coverage is maintained to assure compliance with specifications. Valves are manufactured within the quality assurance program as established by Velan to the satisfaction of Westinghouse.

d. Velan supplies valves for nuclear power plant suppliers other than Westinghouse.

e. Velan has supplied valves for other Westinghouse units and has an "N" stamp.

f. Velan obtains materials from domestic (U.S.) suppliers.

4.6.1.2.1 Pressure Isolation Valves (PIV)

RCS PIVs are two normally closed valves in series within the reactor coolant pressure boundary, which separate the high pressure RCS from an attached low pressure system. The purpose of the PIVs is to prevent overpressure failure of the low pressure system. To assure that this purpose is met, the leakage through the PIVs is limited by the Technical Specifications. The following valves are the PIVs required by Technical Specifications: 00073

<u>RHR to Loop B accumulator injection line</u>

SI-6-2 (2SI-6-2) 00073

SI to Upper Plenum

SI-9-3 (2SI-9-3)
SI-9-4 (2SI-9-4)
SI-9-5 (2SI-9-5)
SI-9-6 (2SI-9-6)

00073

4.6.1.3 Reactor Coolant Flow Measurements

Elbow taps are used in the Reactor Coolant System as an instrument device that indicates the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred. The correlation between flow reduction and elbow tap read out has been well established by the following equation (Reference 29):

$$\frac{\Delta P}{\Delta P_0} = \left[\frac{w}{w_0}\right]^2$$

where ΔP_0 is the pressure differential with the corresponding referenced flow rate w_0 and ΔP is the pressure differential with the corresponding flow rate w. The full flow reference point is established during initial plant startup. The low flow trip point is then established by extrapolating along the correlation curve. The technique has been well established in providing core protection against low coolant flow in Westinghouse PWR plants. The expected absolute accuracy of the channel is within ±10% and field results have shown the repeatability of the trip point to be within ±1%. As a result of the calibration techniques used, the absolute accuracy of the coolant flow measurement is not relevant. As indicated in Section 14, the limiting trip setpoint assumed for analysis was 87% loop flow. This represents a 3% of flow allowance below the setpoint of ≥90% which is specified in the Technical Specifications. Since the trip point is calibrated as a function of full flow output of the instrument and since the flow rate of the reactor is verified during startup testing to be equal or greater than the design flow rate listed in Table 4.1-2 which is the initial flow used for the safety analysis, the actual trip point would be 89% based on the 1% repeatability. Westinghouse has concluded that a more accurate measurement of Reactor Coolant System flow is not required for either plant operation or safety.

Startup tests provided a means for verifying that reactor coolant flow is equal to or greater than the design flow rate. The core flow rate was verified with an accuracy better than 10% by correlating a secondary system heat balance and the inlet and outlet core temperatures. In addition measurements of pump input power and loop ΔP were made at hot shutdown condition for various configurations of running pumps (A pump running, B pump running and both pumps running), the absolute flow rate of each pump is verified to be greater than the design flow.

A blocked or plugged common instrument line to the three redundant reactor coolant flow instruments will produce a low flow indication on the control board for the affected RCS loop. If reactor power is above permissive seven (P-7), that is, 10% full power, a reactor trip will occur. If the power level is below P-7, the operator will bring the plant into the hot shutdown condition as per stipulated in Table 3.5-2A of the Technical Specifications. No safety problem occurs as a result of this condition; the core is maintained well within all operational safety limits.

A rupture of the common instrument line to the reactor coolant loop flow instrument will also be indicated as a low flow condition in the affected loop. Reactor trip will occur if power level is above 10%; manual shutdown below 10% will be provided exactly as in the case of a blocked line. A ruptured instrument line is a type of loss-of-coolant accident discussed in Section 14.7.

Reactor coolant system flow anomalies resulting in simultaneous changes to other reactor coolant system parameters and nuclear instrumentation system parameters have been reported at other Westinghouse plants. These anomalies were first observed in November, 1986. Westinghouse, in conjunction with owners of several operating plants, took test data to determine the nature and magnitude of these flow disturbances. Analysis of the data resulted in the conclusion that these flow disturbances do not occur in the Prairie Island units. Westinghouse prepared an investigation report on this subject (Reference 79).

4.6.1.4 Pump Power-Differential Pressure

This procedure has been used experimentally in an existing plant. The results have produced calculated flowrates in close agreement with the analytically predicted most probable flow and consistent flowrates to within ±.3% for a number of pumps. It is a refinement of the pump power method that utilizes a procedure to establish the actual operating curve from its known shape, determined from model tests, by interrelating pump input power and a relative change in system pressure drop under conditions of one and two pumps running. This procedure reduces the uncertainties associated with the absolute relation of the pump input power curve and flow. This procedure is described in more detail than the more familiar mentioned previously. Figure 4.6-1 is an example of a typical pump input power curve and is included to describe the procedure which is as follows:

a. With the reactor coolant system pressurized, all pumps are started. The flow within the loop to be measured is assumed to be equal to the design (represented by line 1 on Figure 4.6-1) and pump power (represented by line 2 on Figure 4.6-1) and a reference differential pressure is measured. The intersection of lines 1 and 2 establishes a point on the assumed pump power input curve. This allows construction of the assumed curve by shifting the model test curve vertically until it intersects this point.

b. The other pump is stopped. The flow within the active loop increases because of the reduced flow through the reactor vessel. This increased flow above the assumed design flow is determined from the relative increase in the measured differential pressure.

c. This increased flow is then plotted on Figure 4.6-1 (line 3). Its intersection with the previously assumed pump curve will yield the amount of anticipated input power (line 4). If the anticipated input power equals the measured input power with one pump running, the originally assumed flowrate was correct.

The above procedure is all that is necessary to establish whether actual flow is less than, equal to, or greater than design flow. The sense of the difference between anticipated one loop operation input power and measured one loop input power will indicate this. If anticipated power is greater than measured power, the actual flow rate was greater than design. (This can be seen by following the construction of lines 5, 6 and 7 on Figure 4.6-1.) If it is desired to know the actual flowrate, the flow with all pumps operating must again be assumed and the construction of the lines repeated until anticipated one loop power equals measured one loop input power.

This procedure makes use of elbow tap (or steam generator) differential pressure readings. These readings are not used as absolute quantities but only in reference to each other in order to determine the magnitude of the change in flow from one point to another. Therefore, calibration or accurate knowledge of elbow characteristics and dimensions are not required.

The accuracy of this procedure is affected by the accuracy of measured input power, the accuracy of determining the relative change in flow, and the accuracy of the shape of the input power curve. From a review of data from full scale tests of smaller earlier model pumps and the accuracies associated with model tests and hydraulic scaling theory it has been judged that an accuracy of .5% is a conservative tolerance to apply to the accuracy of the shape of the curve. The relative change in flow between the two pump running condition and the one pump running condition can be determined to an accuracy of .5% by the use of pre-test deadweight tester calibrated differential pressure cells and a digital voltmeter. Pump input power can be measured to an accuracy of 0.5% by use of procedures and instrumentation available from a test organization at the Westinghouse Large Rotating Apparatus Division. Typical instrumentation that would be used consists of a wattmeter, and volt and ammeters. These accuracies result in an expected total flowrate measurement accuracy of ±2.5%.

4.6.1.5 Reactor Coolant System Temperature Measurements

Resistance Temperature Detectors (RTD's) are located in bypass loops for each hot and cold leg to develop signals used as part of the reactor control and protection systems. The RTD bypass design improves the capability to perform maintenance without sacrificing accuracy.

In addition to the bypass loop RTD's, one well type RTD is located in each hot and cold leg to provide loop temperature signals independent of the bypass loops. However, these temperature signals are not used in the control or protection of the reactor. Figures 4.1-1A and 4.1-1B show the various RTD locations.

The hot and cold leg RTD's are inserted into reactor coolant bypass loops. A bypass loop from upstream of the steam generator to downstream of the steam generator is used for the hot leg RTD's and a bypass loop from downstream of the reactor coolant pump to upstream of the pump is used for the cold leg RTD's. The RTD's are located in manifolds within the containment and are directly inserted into the reactor coolant bypass loop without thermowells. Direct immersion in the RCS piping is not used in order to keep the detector thermal lag small. Also the bypass arrangement permits replacement of defective temperature elements while the plant is at hot shutdown without draining or depressurizing the reactor coolant loops.

To obtain a representative hot leg temperature, three sampling probe connections are installed 120° apart on the same cross-sectional plane of the Reactor Coolant System piping and extend into the Reactor Coolant System pipe. The hot leg RTD bypass flow from the three connections joins a common line upstream of the hot leg bypass loop isolation valves.

Each of the sampling probes, which extend several inches into the hot leg coolant stream, contains five inlet orifices distributed along its length. In this way a total of fifteen locations in the hot leg stream are sampled providing a representative coolant temperature measurement. The two inch diameter pipe leading to the manifold containing the temperature measuring elements (RTD's) provides mixing of the samples to give an accurate temperature measurement.

Care has been taken to distribute the flow evenly among the five orifices of each probe by effectively restricting the flow through the orifices. This has been done by designing a smaller overall flow area than that of the common flow channel within the probe.

This arrangement has also been applied to the flow transition from the three probe flow channels to the pipe leading to the temperature element manifold. The total flow area of the three probe channels has therefore been designed to be less than that of the two inch pipe connecting the probes to the manifold.

Flow for the cold leg RTD bypass originates downstream of the reactor coolant pump discharge. Because of the mixing action of the pump, only one connection is required for the cold leg bypass. This connection is located in the same relative position for each loop.

The accuracy of the RTD bypass loop temperature measurements was demonstrated during plant startup tests by comparing temperature measurements from all bypass loop RTD's with one another as well as with the temperature measurements obtained from the RTD's located in the hot leg and cold leg piping of each loop. The comparisons are done with the Reactor Coolant System in an isothermal condition. The linearity of the ΔT measurements obtained from the hot leg and cold leg bypass loop RTD's as a function of plant power was also checked during plant startup tests. As part of the plant startup tests, the loop RTD signals were compared with the core-exit thermocouple signals.

Low flow is to be avoided since it could result in an overall time delay in the temperature measurement greater than that assumed in the safety analysis (see Section 14.3). Loss of flow or reduced flow in a single bypass loop would result in an increase in the time response of the coolant loop temperature measurement.

An alarm occurs if the flow in a bypass loop is reduced below the full power flow by 10% or more. If redundancy conditions on the ΔT trips are not met with the reactor at power, the Technical Specifications require proceeding to hot shutdown. However, bypass flow is not a direct input quantity to either the protection or control systems.

The use of more flow instruments in each bypass loop does not enhance the plant safety design. Failure of the flow instrument in a bypass line does not by itself result in any adverse behavior or loss of either protection or control system function associated with the RTD's.

An actual occurrence of reduced flow or loss of flow will tend to cause the RTD to read a lower T_{avg} for the affected loop. Bypass loop low flow will not cause control system behavior requiring protection system countermeasure action. Low flow in one bypass loop, even if undetected as a result of a faulted instrument would not negate the capability of the protection system to function properly. However, the coincidence of a low flow condition with failure of the flow instrument is considered by Westinghouse to be an extremely unlikely situation. Further, aberrant readings and inconsistencies in expected behavior of RTD's in a bypass line will provide additional indication of reduced flow. Periodic inspection of the bypass loop flow indicators in accordance with Technical Specifications is performed to check against malfunctions.

A blocked or plugged common instrument line to the four redundant reactor coolant temperature instruments will produce a low flow indication on the control board for the affected RCS loop, and a reactor trip will occur.

A rupture of the common instrument line to the reactor coolant temperature instruments will also be indicated as a low flow condition in the affected loop, and a reactor trip will occur. A ruptured instrument line is a type of loss-of-coolant accident discussed in Section 14.7.

Sufficient alarms, indicators and recorders are available on the control board for the operator to monitor the status of both RCS loops with regard to all operating variables and reactor trips, including RCS pump operation, flow, ΔT, Tavg, pressurizer pressure and water level.

Reactor coolant system pressure and temperature are continuously recorded on both units 1 and 2 by permanently installed strip-chart recorders in the control room.

4.6.2 Design Evaluation

4.6.2.1 System Incident Potential

The potential of the Reactor Coolant System as a cause of accidents is evaluated by investigating the consequences of certain credible types of components and control failures as discussed in Section 14. Reactor coolant pipe rupture is evaluated in Sections 14.6 and 14.7.

4.6.2.2 Blowdown Jet Forces & Pipe Whip

All piping systems were routed, barriers installed, or the piping otherwise restrained such that all vital equipment is shielded from potential damage due to pipe whip caused by jet reaction loads. Individual lines and components of all engineered safety features are separated to the maximum extent practicable and restrained where necessary to prevent interaction with redundant lines and components, as well as with other systems.

Pipe restraint design requirements are such that restraints were located such that plastic hinges were prevented, unless formation of the plastic hinge did not allow pipe whip to impair containment integrity or a safety system function. The pipe rupture restraints are designed with an allowable stress less than 0.9 times the yield strength of the restraint material.

Pipe rupture analyses were performed on all high pressure piping (including lines with diameters less than 3/4"). The analyses establish that the containment vessel and all essential equipment within and without the containment (system and equipment defined as Class I in Section 12) are adequately protected against the effects of potential pipe ruptures.

Method of Analysis:

Pipe ruptures were postulated in the portions of piping systems pressurized during normal plant operations, the resulting forces were determined in accordance with the criteria as specified below, and potential damage to the system under consideration and to other Class I systems and equipment was evaluated.

a. Ruptures were postulated in adjacent Class I, II and III high pressure piping and potential damage to Class I systems or equipment under consideration was evaluated.

b. Potential damage to Class I Systems and Equipment, including the containment vessel, resulting from pipe rupture was evaluated to assure that their minimum required performance is not reduced below that specified in the USAR. As part of the evaluation it is assumed that the failure of any single active component could occur coincident with the assumed pipe rupture.

Pipe rupture constitutes potential sources of damage as a result of:

a. Jet impingement - the force loading of the jet issuing from the break.

b. Pipe whip - the unrestrained movement of a length of pipe caused by the reaction loading at point of break.

The evaluation of damage propagation from these forces was based upon piping configuration, location of barriers and supports, locations of postulated breaks, separation of redundant parts of the system, and location of other systems and equipment in relation to the system under consideration. Location of breaks were assumed such that the most critical results would occur.

Rupture Forces:

The initial force at the point of rupture is

$$F = 1.2\,PA$$

where:

P = static pressure at point of rupture

A = flow area of the pipe

For breaks in compressible fluid systems and in liquid systems connected to reservoirs that are large relative to the pipe size, the pressure is the maximum normal operating pressure at the point of rupture. For breaks in liquid systems not connected directly to reservoirs, the pressure is based on the saturation pressure at the maximum normal operating temperature.

Break Size:

The area of any postulated rupture is assumed equal to the flow area of the ruptured pipe. A longitudinal break is assumed to be rectangular in shape with length equal to two times the inside diameter of the ruptured pipe.

Jet Impingement Load:

The jet impingement load is defined as the load on a component (piping or equipment) of the undeflected jet from an instantaneous circumferential or longitudinal break of an adjacent pipe.

At the point of rupture, the jet pressure is assumed equal to the rupture pressure (P), and the effective loading area is assumed to be the break area (A). As the flow progresses away from the point of rupture the jet is assumed to diverge at an inclined angle of 45°. Hence, the effective loading area at some distance from the point of a longitudinal break is,

$$A_i = [L_1 + 2L_3 \ \text{Tan } 22\text{-}1/2°] \ [L_2 + 2L_3 \ \text{Tan } 22\text{-}1/2°]$$

where:

L_1 = width of break

L_2 = length of break

L_3 = distance to target

and the jet pressure at some distant target object with the effective load area is,

$$P_i = P \left[\frac{A}{A_i} \right]$$

The effective load on a distant target is then,

$$F_e = f_i P A_e$$

where:

A_e = projected area of the target object

f = shape factor of the target object

Criteria For Pipe Whip:

The evaluation of the effects of pipe whip is based upon a review of the physical arrangement of the piping system under consideration in conjunction with the evaluation of the ultimate load carrying capability of the piping. At any point in the pipe where the load resulting from a rupture exceeds the ultimate load carrying capability, it is assumed that a plastic hinge is formed and that the pipe will rotate freely about this point unless restrained at another point.

Discussion and derivations of the ultimate loads, bending strengths and bending moments for carbon steel and stainless steel piping are given in Reference 30.

For conservatism, the lower limit value is used to determine the location of a plastic hinge in stainless steel piping.

If either of the two following conditions are found to exist, it is assumed that pipe whip would result:

a. A circumferential break such that a section of pipe is subjected to a cantilever type loading in a manner which produces a bending moment greater than the ultimate bending moment.

b. A longitudinal break resulting in bending moments greater than the ultimate bending moment at the point of break and at the restraint points on either side of the break.

A whipping section of pipe is assumed to move freely until striking an object capable of stopping it and no recurring plastic hinges are assumed to develop. Further, it is assumed that the pipe will neither rebound nor change directions.

It is assumed that a whipping pipe will not damage another pipe of equal or greater size and schedule.

Restraint/Anchor Loading:

Determination of the maximum loads on pipe rupture restraints and anchors is based upon the following assumptions:

a. Pipe rupture loads are "point" loads.

b. Restraints/anchors act as "fixed" supports.

c. A plastic hinge is not formed in stainless steel pipe until the upper limit value of the ultimate load carrying capability is exceeded.

d. Loads can originate from either a pipe rupture force in the piping system under consideration or a jet impingement force resulting from a break in adjacent Class I, II or III piping.

4.6.2.3 Elimination of Large Primary Loop Pipe Rupture as the Structural Design Basis

On February 1, 1984 the NRC issued Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops" (Reference 49). This safety evaluation was based on review of Westinghouse analyses which demonstrated on a generic basis that reactor coolant system primary loop pipe breaks are highly unlikely and should not be included in the structural design basis of Westinghouse plants. These analyses are referred to as "leak-before-break" (LBB).

In order to demonstrate the applicability of the generic Westinghouse evaluations to the Prairie Island units, Westinghouse performed a fracture mechanics evaluation, a determination of leak rates from a through-wall crack, a fatigue crack growth evaluation, and an assessment of margins for both Unit 1 and 2 (References 50, 51, 68, 69). These reports provided the basis for elimination of reactor coolant system primary loop pipe breaks from the design basis. Thermal aging and degradation of cast stainless steel was considered in these evaluations. Additional consideration of thermal aging effects was completed by the utilities in the Westinghouse Owners' Group (Reference 76).

The analyses submitted by Northern States Power Company were accepted by the NRC as documented in a Safety Evaluation Report (Reference 70). In this safety evaluation the NRC found that the criteria provided in Chapter 5.0 of NUREG-1061, Volume 3, for evaluation of compliance with General Design Criterion 4, (GDC 4) of Appendix A to 10CFR50 as revised were satisfied and concluded that "the probability or likelihood of large breaks occurring in the primary coolant loops of Prairie Island Units 1 and 2 is sufficiently low such that dynamic effects associated with postulated pipe breaks in these facilities need not be a design basis. Furthermore, the staff concludes that the licensee is in compliance with the GDC 4 as revised."

It should be noted that there are limitations regarding the use of the design basis change and the LBB technology. As stated in NUREG-1061, Volume 3, the dynamic effects which may be excluded are:

1. pipe whip and other pipe break reaction forces,

2. jet impingement forces,

3. vessel cavity or sub-compartment pressurization including asymmetric transient effects, and

4. pipe break-associated transient loadings in functional systems or portions thereof whose pressure-retaining integrity remains intact.

The exemption and LBB technology do not apply to ECCS, containment or other system design.

Finally, the NRC also based their acceptance of the LBB technology on the ability of the reactor coolant system leak detection system to detect leakage from the RCS at a factor of 10 more restrictive than the reference flaw size. The leak detection system at Prairie Island is consistent with the guidelines of Regulatory Guide 1.45 for detecting leakage of 1 gpm in one hour. Reference 103 quotes sensitivities of the leakage detection system in excess of those cited to meet the guidance of the regulatory guide. The original Westinghouse evaluation uses the regulatory guide values for comparison. Operating history at Prairie Island shows these values to be conservative when compared to the leaks that have been detected.

The pipe rupture restraints installed in the reactor vessel shield wall have been removed.

4.6.2.4 Elimination of Pressurizer Surge Line Rupture as the Structural Design Basis for Prairie Island Unit 1

On December 20, 1988 the NRC issued NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification" which required a stress and fatigue analysis for pressurizer surge lines considering the effects of thermal stratification and cycling. Results of this analysis for Prairie Island Unit 1 showed that in order to keep stress levels below ASME limits, modification of pipe whip restraints was required to allow unrestrained pipe movement (Reference 98).

A leak-before-break (LBB) analysis was performed by Westinghouse consistent with the criteria in NUREG-1061, Volume 3 thereby complying with General Design Criterion-4 (GDC-4) of Appendix A to 10 CFR Part 50. The analysis concluded that the probability of large pipe breaks occurring in the surge line is sufficiently low such that dynamic effects associated with the postulated pipe breaks need not be a design basis. The LBB analysis was submitted by NSP for NRC review (Reference 99, Reference 95), and was approved as documented in a NRC Safety Evaluation Report (Reference 100). The NRC staff conclusion was conditioned on NSP's commitment to remove the shims or modify the gaps of the whip restraints to allow the surge line to satisfy NRC Bulletin 88-11.

The pipe rupture restraint shim packs for the Prairie Island Unit 1 pressurizer surge line were removed during the Fall 1992 outage. The LBB analysis now eliminates the need for pipe rupture restraints in the design basis for the Unit 1 surge line.

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TABLE 4.1-1 MATERIALS OF CONSTRUCTION OF THE REACTOR COOLANT SYSTEM COMPONENTS (Page 1 of 2)

Component	Section	Materials	
Reactor Vessel	Pressure Plate	SA-533, Grade B, Class 1	
	Shell & Nozzle Forgings	SA-508 Class 3	
	Cladding, Stainless Weld Rod	Type 304 or equivalent and Inconel	
	Core Support	Inconel	
	Thermal Shield and Internals	A-204, Type 304	
	Insulation	Reflective Type (100% type 304 SS construction)	
	Control Rod Housing	Inconel and Type 304	
	Instrumentation Nozzles	Inconel	
Steam Generator	Pressure Plate	SA-533, Grade A, Class 1	00222
	Cladding, Stainless Weld Rod	Type 304 or equivalent	
	Cladding for Tube Sheets	Inconel	
	Tubes	SB-163	
	Channel Head Castings	SA-216 Grade WCC	
Pressurizer	Shell	SA-533, Grade A, Class 1	
	Heads and Nozzles	SA-216 Grade WCC	
	External Plate (Skirt)	SA-516, Grade 70	00222
	Cladding, Stainless	Type 304 or equivalent	
	Internal Plate	SA-240 Type 304	

TABLE 4.1-1 MATERIALS OF CONSTRUCTION OF THE REACTOR COOLANT SYSTEM COMPONENTS (Page 2 of 2)

Component	Section	Materials
	Internal Piping	SA-376 Type 316
Pressurizer	Shell	A-285 GR C
Relief Tank	Heads	A-285 GR C
Piping	Pipes	Unit 1: A-376 Type 316
		Unit 2: A-351, CF8M
	Fittings	A-351, CF8M
	Nozzles	A-182 F316
Pump	Shaft	Type 347
	Impeller	A-351, CF8
	Casing	A-351, CF8
Valves	Pressure Containing Parts	A-351, CF8M and A-182 F316

00222

TABLE 4.1-2 REACTOR COOLANT SYSTEM DESIGN PARAMETERS AND PRESSURE SETTINGS

Total Primary Heat Output, MWt	1650
Total Primary Heat Output, Btu/hr	5631×10^6
Number of Loops	2
Coolant Volume (Liquid), including pressurizer volume, at full power, ft^3	6191
Total Reactor Coolant Flow, lb/hr	68.2×10^6

	Pressure, psig
Design Pressure	2485
Operating Pressure (at pressurizer)	2235
Safety Valves	2485
Power Relief Valves	2335
Pressurizer Spray Valves (open)	2260
High Pressure Trip	2385
High Pressure Alarm	2335
Low Pressure Trip	1900
Hydrostatic Test Pressure (Cold), Initial hydro only	3107

85010

TABLE 4.1-3 REACTOR VESSEL DESIGN DATA

Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure, psig	3107
Design Temperature, °F	650
Overall height of Vessel and Closure Head, ft-in. (Bottom Head O.D. to top of Control Rod Mechanism Housing)	39 - 1.4
Water Volume (with core and internals in place), ft^3	2473
Thickness of Insulation, min., in.	3
Number of Reactor Closure Head Studs	48
Diameter of Reactor Closure Head Studs, in.	6
Flange, ID, in.	123.8
Flange, OD, in.	157.3
ID at Shell, in.	132
Inlet Nozzle ID, in.	27.5
Outlet Nozzle ID, in.	29.0
Clad Thickness, min., in.	0.125
Lower Head Thickness, min., in.	4.252
Vessel Belt-Line Thickness, min., in.	6.7
Closure head Thickness, in.	5.51
Reactor Coolant Inlet Temperature, °F	535.5
Reactor Coolant Outlet Temperature, °F	599.1
Reactor Coolant Flow, lb/hr	68.2×10^6
Safety Injection Nozzle, number/size, in.	2/4

00264

TABLE 4.1-4 PRESSURIZER & PRESSURIZER RELIEF TANK DESIGN DATA
(Page 1 of 2)

Pressurizer

Design/Operating Pressure, psig	2485/2235
Hydrostatic Test Pressure (cold), psig	3107
Design/Operating Temperature, °F	680/653
Water Volume, Full Power, ft^3 (1)	600
Steam Volume, Full Power, ft^3	400
Surge Line Nozzle Diameter, in./Pipe Schedule	14/140
Shell ID, in./Minimum Shell Thickness, in.	84/4.1
Minimum Clad Thickness, in.	0.188
Electric Heaters Capacity, kw (total)	1000(2)
Heatup rate of Pressurizer using Heaters only, °F/hr	55 (Approx.)(2)
Power Relief Valves	
Number	2
Set Pressure (open), psig	2335
Capacity, lb/hr Saturated steam/valve	179,000
Safety Valves	
Number	2
Set Pressure, psig	2485
Capacity (ASME rated flow) lb/hr/valve	345,000

(1) 60% of net internal volume (maximum calculated power)

(2) These are original design values. Due to operational and maintenance considerations, pressurizer heater capacity may be reduced below the original design value; which results in a slower heat up rate.

TABLE 4.1-4 PRESSURIZER & PRESSURIZER RELIEF TANK DESIGN DATA

Pressurizer Relief Tank

Design pressure, psig	100
Rupture Disc Release Pressure at 120°F, psig	99
Design temperature, °F	340
Normal water temperature, °F	120
Total volume, ft^3	800
Rupture Disc Relief Capacity, lb/hr	8.0×10^5

00249

TABLE 4.1-6 REACTOR COOLANT PUMPS DESIGN DATA
(Page 1 of 2)

Number of Pumps	2	
Design Pressure/Operating Pressure, psig	2485/2235	
Hydrostatic Test Pressure (cold), psig	3107	
Design Temperature (casing), °F	650	
RPM at Nameplate Rating	1189	
Suction, Temperature, °F	544.5	
Net Positive Suction Head, ft.	172	
Developed Head, ft	259/277*	
Capacity, gpm/pump	89,000/88,500*	
Seal Water Injection, gpm/pump	8	
Seal Water Return, gpm/pump	3	
Pump Discharge Nozzle, ID, in.	27 - 1/2	00051
Pump Suction Nozzle ID, in.	31	
Overall Unit Height, ft.	27	00051
Water Volume, ft^3/pump	56	
Pump-Motor Moment of Inertia, $lb\text{-}ft^2$	80,000 (21 RCP is slightly higher)	00051
Motor Data:		
Type	AC Induction Single Speed, Air Cooled	
Voltage	4000	

* For Unit 2 - #21 RCP only.

TABLE 4.1-6 REACTOR COOLANT PUMPS DESIGN DATA
(Page 2 of 2)

Insulation Class	B Thermalastic Epoxy
Phase	3
Frequency, Hz	60
Starting Current, maximum, amp	4800
Input (hot reactor coolant), kw, approx.	4600
Input (cold reactor coolant), kw, approx.	6000
Power, HP (nameplate)	6000

00051

TABLE 4.1-7 REACTOR COOLANT PIPING DESIGN DATA

	Unit 1	Unit 2	
Design/Operating Pressure, psig	2485/2235	2485/2235	
Hydrostatic Test Pressure, (cold) psig	3107	3107	
Design Temperature, °F	650	650	
Design Temperature, (pressurizer surge line), °F	680	680	
Reactor Inlet Piping, ID, in.	27 - 1/2	27 - 1/2	
Reactor Inlet Piping, minimum wall thickness, in.	2.215	2.56	00248
Reactor Outlet Piping, ID, in.	29	29	
Reactor Outlet Piping, minimum wall thickness, in.	2.335	2.70	00248
Coolant Pump Suction Piping, ID, in.	31	31	
Coolant Pump Suction Piping, minimum wall thickness, in.	2.495	2.88	00248
Pressurizer Surge Line Piping, ID, in./Pipe Schedule*	10/140	10/140	
Pressurizer Surge Line Piping, nominal thickness, in.	1	1	
Water Volume, (2 loops) ft^3	565	565	

* Surge line fitted with a 14″/10″ adapter at the pressurizer

TABLE 4.1-8 REACTOR COOLANT SYSTEM OPERATING TRANSIENTS USED FOR DESIGN (40-YEAR PLANT LIFE)
(Page 1 of 2)

	Operating Cycle	Occurrences+
1.	Heatup at 100°F/hr	200
	Cooldown at 100°F/hr (Pressurizer 200°F/hr)	200
2.	Plant Loading at 5% of nominal full load/min	18,300
	Plant Unloading at 5% of nominal full load/min	18,300
3.	Step Load Increase of 10% of nominal full load	2,000
	Step Load Decrease of 10% of nominal full load	2,000
4.	Large Step Decrease in Load (with steam dump)	200
5.	Loss of Load (without immediate turbine or reactor trip)	80
6.	Loss of Offsite Power (LOOP with natural circulation in Reactor Coolant System)	40
7.	Loss of Flow (partial loss of flow one pump only)	80
8.	Reactor Trip From Full Power	400
9.	Turbine Roll Test	10
10.	Hydrostatic Test Condition	
	a. Primary Side Hydrostatic Test Before Initial Startup at 3107 psig	5
	b. Secondary Side Hydrostatic Test Before Initial Startup at 1356 psig	5
11.	Primary Side Leak Test at 2500 psia	50

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SECTION 5

CONTAINMENT SYSTEM

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5.2 PRIMARY CONTAINMENT SYSTEM

5.2.1 Design Criteria

5.2.1.1 Containment System Criteria

The Reactor Containment Vessel is designed for a maximum internal pressure of 46 psig and a temperature of 268°F. The Reactor Containment Vessel design internal pressure as defined by ASME Boiler and Pressure Vessel Code is 41.4 psig.

The vessel is 105 ft. inside diameter and contains an internal net free volume of 1,320,000 ft^3.

The vessel plate nominal thickness does not exceed 1-1/2" at the welded joints so the vessel, as an integral structure did not require field stress relieving. Reinforcing plates at penetration openings exceed 1-1/2" in thickness; however, these were fabricated as penetration weldment assemblies and were stress relieved before they were welded to adjacent vessel shell plates.

The loadings considered in the design of the Reactor Containment Vessel, in addition to the pressure and temperature conditions described above are discussed in Section 12.2.2.

The Reactor Containment Vessel, including penetrations is designed for low leakage. The initial measured leakage rate was approximately 0.02% by weight in 24 hours at a nominal internal pressure of 46 psig.

5.2.1.2 Containment Auxiliary Systems Criteria

5.2.1.2.1 Reactor Containment Vessel Isolation Systems

Criterion: Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus (GDC 53).

Isolation valves are provided as required for fluid system lines penetrating containment to assure that:

a. Leakage through all fluid line penetrations not serving accident consequence-limiting systems is minimized by a double barrier. The double barriers take the form of closed pipe systems, both inside and outside the Reactor Containment Vessel, and various types of isolation valves. The double barrier arrangement provides two reliable low leakage barriers between the Reactor Coolant System or containment atmosphere and the environment. The failure of any one barrier will not prevent suitable isolation;

b. Fluid line penetrations normally serving accident consequence limiting systems can be isolated by manual action if the system should malfunction;

c. No single credible failure or malfunction (expected fault condition) occurring in any active system component can result in loss-of-isolation or intolerable leakage.

An isolation actuation system is provided to automatically close fluid line penetrations used during normal operation but not required for engineered safety features functions. The automatic closure occurs upon a Safety Injection signal or manual initiation, EXCEPT for Instrument Air Isolation valves which requires an additional input from Loop A MSIV Auto Closure Signal or High High Containment Pressure Signal. (See Section 7.)

Actuation signals for the containment isolation system are generated by the same pressure transmitters used for safety injection and MSIV closure. The testing of this instrumentation is the same as for the safety injection system.

5.2.1.2.2 Vacuum Relief System

Vacuum relief devices or systems are provided to protect the Reactor Containment Vessel against excess differential pressures. Such differential pressure conditions (vacuum) may exist inside the Containment Vessel if the Containment Air Cooling Systems are operated with a heat removal capability in excess of the heat inputs at any time during normal or post accident operations.

The vacuum relief valves are sized to assure that the Reactor Containment Vessel will not be subjected to an internal pressure in excess of 0.8 psi below the external pressure. The design basis for sizing the relief system has been identified as the inadvertent and simultaneous operation of all Containment Air Cooling Systems during normal operation or following a plant shutdown, when heat inputs to the containment are minimal and the cooling water temperatures produce the largest heat removal rates for the respective cooling systems. The Containment Air Cooling Systems to be included in the analysis are the two full-capacity Containment Internal Spray Systems and the four containment fan-coil units.

5.2.1.3 Containment Penetrations

5.2.1.3.1 General

To maintain designed containment integrity, containment penetrations have the following design characteristics:

a. They are capable of withstanding the maximum internal pressure which could occur due to the postulated rupture of any pipe inside the Reactor Containment Vessel.

b. They are capable of withstanding the jet forces associated with the flow from a postulated rupture of the pipe in the penetration or adjacent to it, while still maintaining the integrity of containment.

c. They are capable of accommodating the thermal and mechanical stresses which may be encountered during all modes of operation and test.

d. Materials of piping penetrations furnished as a part of the Reactor Containment Vessel are either ASME SA-333 GR. 6 or ASME SA-312 TP304 material. All hot penetrations are fabricated of ASME SA-333 Gr 6. The cold penetrations are fabricated of either of the foregoing materials, but in all cases are compatible with the material of the process line which is to be welded directly to the penetration nozzle.

e. The materials for penetrations, including the personnel access air locks, the equipment access hatch, the piping and duct penetration sleeves, and the electrical penetration sleeves, conform to the requirement of the ASME Nuclear Vessel Code and USAS B 31.1 Code for Pressure Piping and applicable code cases. The process and guard pipes were designed, specified, and fabricated in accordance with the Code for Pressurized Piping, ANSI B31.1-1967, without the use of code cases. The flued heads were designed and fabricated in accordance with the ASME Code for Nuclear Vessels, Section IIIB, 1968. The materials specified for penetrations meet the necessary NDT impact values as specified in the ASME Nuclear Vessels Code.

All Containment Vessel penetrations nozzles are designed to meet the requirements for Class B vessels under Section III of ASME Boiler and Pressure Vessel Code. In compliance with the code, the operating stresses in a containment vessel penetration nozzle caused by the attached penetration assembly are limited to the allowable values given in the code. For earthquake analysis, Section III of the ASME code permits the use of 1.5 times the allowable stress value for the material being used.

The double-bellows expansion joints in the hot-pipe penetration assemblies and the Shield Building flexible seals for all pipes are designed to accommodate the maximum combination of vertical, radial, and horizontal differential movements between the Reactor Containment vessel; the Shield Building and the piping. This design considers the calculated displacements resulting from earthquake, pressure and temperature (as shown in Figure 5.2-1), and also accounts for the actual measured displacement of representative penetration nozzles made during the initial pressure testing of the Containment Vessel.

The shield building flexible seals are designed to withstand the process piping temperature and provide an adequate leak-tight seal consistent with overall allowable Shield Building leakage.

5.2.1.3.2 Hot Penetrations

Hot piping penetration assemblies are provided to:

a. Prevent unacceptable thermal and cyclic stress on Reactor Containment Vessel penetration nozzles;

b. Accommodate thermal movement;

c. Protect containment from the effects of a hot process pipe rupture in the annulus between the Shield Building and the Reactor Containment Vessel.

Where hot penetration assemblies traverse the Shield Building annulus, they are designed to provide considerable margin between code allowable stress values and maximum calculated stresses in the pipe. This was accomplished by using 1.5 times the system design pressure to calculate the pipe wall thickness for the process and guard pipe, using the formula and allowable stresses given in USAS B31.1.0-1967. Under normal B31.1.0 code practice, the system design pressure alone is adequate for calculating the pipe wall thickness. The same procedure was used to set the thickness of the guard pipe and the multiple flued head.

5.2.1.3.3 Main Steam Line Penetration

The main steam line between the anchor inside containment and the first isolation valve outside of containment has a wall thickness selected by using 1.5 times the system pressure and normal code allowable stress values. The main steam anchor inside containment is designed to sustain the full force resulting from a 360° circumferential break of the main steam line. The other requirements previously discussed for a hot-pipe penetration assembly are also met.

5.2.1.4 Containment Vessel Air Handling System

The Containment Vessel Air Handling System consists of the Containment Air Cooling system, the Containment Internal Cleanup System, the Inservice Purge and the purge ventilation system. The function of the Air Cooling system is to remove the heat lost to the Containment Vessel environment during normal operations, from equipment and piping inside the Containment Vessel and during post-accident conditions, to remove energy from the Containment Vessel as required and described in Section 6.1. The function of the Containment Internal Clean-up System is to recirculate containment air through filters to clean up containment atmosphere prior to limited personnel access. The function of the purge system is to provide fresh, tempered air for comfort during maintenance and refueling operations and to purge contaminated air through charcoal filters from either or both Reactor Containment Vessel(s) while shutdown. The function of the Inservice Purge System is similar to that of the purge system except that a smaller volume of air is handled and exhaust air is processed through HEPA filters and charcoal absorbers.

The Containment Air Cooling System is sized such that any three fan coil units will provide adequate heat removal capacity from the Reactor Containment during normal and full-power operation, to maintain interior air temperatures below the maximum temperature allowable at any component, and to obtain temperatures below 104°F in accessible areas during hot standby operation. The fan-coil units are also utilized for emergency cooling under post-accident conditions. Their use for that purpose is described in Section 6.3.

The fan-coil units of the Air Cooling System are utilized to distribute air adequately over equipment and around occupied spaces in the building for ventilation service. Unit heaters or electric heaters provide for heating within the Containment Vessel when required during shutdown.

The purge and ventilation system is sized to provide a reduction of the radioactivity in the Containment Vessel air following normal full-power operation to the level defined by 10CFR20 for a 40 hour occupational work week, within 2-6 hours after reactor shutdown. Purging of the Containment Vessel will normally be accomplished within two hours following the beginning of purge.

Provision is made in the design of the purge and ventilation system for 1½ air changes per hour during refueling and maintenance operations. The Containment System Vent provides for the discharge of air at an elevation near the top of the Shield Building within the influence of the building wake effect, to improve the dispersion of gaseous releases.

5.2.2 Description

5.2.2.1 Primary Containment Auxiliary Systems

5.2.2.1.1 Isolation System

Table 5.2-1 lists the containment penetrations and the isolation provided for each penetration. The seven classes of penetration isolation provided below are used in the Tables in the column designated PENETRATION CLASS.

a. Class 1 (Outgoing Lines, Reactor Coolant System)

   Outgoing lines connected to the Reactor Coolant System are provided with two automatically operated trip valves in series located near the Reactor Containment Vessel (one inside and one outside). A non-automatic isolation valve either locked closed or maintained under direct administrative control is equivalent to an automatic isolation valve.

b. Class 2 (Outgoing Lines)

Outgoing lines not connected to the Reactor Coolant System, not protected from missiles throughout their length and not required to be open for assumed post-accident conditions are provided with two automatically operated trip valves. At least one valve is external to the Reactor Containment Vessel, the other may be internal or external.

c. Class 3 (Incoming Lines)

Incoming lines connected to open systems outside the Reactor Containment Vessel are provided with two check valves in series, one located inside and one outside the Reactor Containment Vessel. The internal check valve will be located near the Reactor Containment Vessel shell.

Incoming lines connected to closed systems outside the containment are provided with at least one check valve located near the Containment Vessel on the inside and a manually operated (local or remote) isolation valve outside the Containment Vessel. In this instance, the closed system outside of containment serves as the secondary containment boundary. The manually operated isolation valve may be closed, if desired.

An automatically operated trip valve or a non-automatic isolation valve either locked closed or maintained under direct administrative control is considered to be the equivalent of a check valve and vice-versa.

d. Class 4 (Missile Protected)

Incoming and outgoing lines which penetrate the Reactor Containment Vessel and are connected to closed systems inside the Reactor Containment Vessel and which have a low probability of being ruptured by the assumed accident, are provided with at least one remotely operated valve located outside the Reactor Containment Vessel.

Steam Generator secondary side isolation valves receive special treatment because their function is not containment isolation for the loss-of-coolant accident but only containment isolation for main steam line rupture within containment. Leakage rate and test requirements may be adjusted accordingly.

e. Class 5 (Normally Closed Lines Open to the Containment)

Lines which penetrate the Reactor Containment Vessel and which can be opened to the Containment Vessel atmosphere but which are normally closed during reactor operation and are provided with two isolation valves in series, two blind flanges, or one isolation valve and one blind flange. One valve or flange is located inside and the second valve or flange is located outside the Reactor Containment Vessel.

Several of the flanges are provided with a double "O"-Ring seal and located in areas of containment which have a low probability of being affected by the assumed accident. Both "O"-Rings are tested separately through installed test connections. In this instance, these "O"-Rings provide the required redundant barrier isolation. Other isolation capabilities in these lines (e.g., valve, damper, flange gasket) provide additional levels of redundancy.

f. Class 6 (Systems Required to Operate in the Post-Accident Condition)

   The design and operational criteria for the isolation valves in these systems is governed by the functional requirements of the systems as outlined in the section in which the system is described.

g. Class 7 (Normally Closed Lines with Leakage Returned to Shield Building Annulus)

   Class 7 lines are the same as Class 5 lines with the addition of a feature to assure that any small leakage through the isolation system is returned to the Shield Building annulus for processing by the Shield Building Vent System.

Instrument lines associated with closed systems, such as containment pressure instrumentation, which are fabricated to withstand the maximum containment pressure, the maximum containment temperature, and are protected from missiles and dynamic effects are acceptable without containment isolation valves.

The actuation systems for automatic containment isolation are discussed in Section 7. Isolation valves inside containment are equipped with operators and actuation devices capable of operating reliably under post-accident containment conditions. Air Operated Control Valves which are designated as automatic trip isolation valves are designed to either fail closed upon loss of actuation power and/or loss of power to control logic or are provided with a reliable source of actuation power and a "fail-close mode" upon loss of power to control logic.

Motor Operated Valves (MOV) designed as automatic trip isolation valves are designed to fail in the "as is" position on a loss of power. When a MOV is used for containment isolation purposes, a redundant barrier is provided (for example, a closed system, check valve, another MOV, etc.). If two MOVs are used to isolate a containment penetration, they are powered from redundant power supplies to ensure the penetration is isolated in the event of a single active failure.

Certain valves for Class 6 usage (engineered safety features) are excepted from the "fail-close" criterion. The operation of valves in these systems is governed by the functional requirements of the system as outlined in other Sections.

5.2.2.1.2 Vacuum Relief System

Two vacuum breakers are used in each of two large vent lines which permit air to flow from the Shield Building annulus into the Reactor Containment Vessel. The vacuum breakers consist of an air to close, spring loaded to open butterfly valve and a self-actuated horizontally installed, swinging disc check valve as shown in Figure 5.2-3. The vent lines enter the containment Vessel through independent and widely separated containment penetration nozzles.

Vendor-supplied flow versus pressure-drop information was used to ensure that sufficient flow area is available in each line so that the combined pressure drop at rated flow through both valves in series will not result in the differential pressure between Containment and the Shield Building exceeding the permissible pressure. Satisfactory operation of either of the vacuum relief lines is adequate to meet the design conditions.

5.2.2.2 Containment Penetrations

5.2.2.2.1 Electrical Penetrations

The electrical penetration assemblies are designed for field installation. D G O'BRIEN assemblies are installed by welding to the inside end of the nozzle type penetration passing through the Reactor Containment Vessel wall. Conax assemblies are installed by welding to the outside end of nozzle type penetrations. 9910

Each penetration assembly is provided with a single connection to allow pressure testing for leaks. All components of the penetration assemblies are designed to withstand, without damage or interruption of operations, the forces resulting from an earthquake, in addition to the normal and accident design requirements.

All materials used in the design are selected for their resistance to environments existing under normal operation and the DBA.

Figure 5.2-4 shows the configuration of selected D G O'Brien and Conax penetration assemblies. Electrical penetrations are provided for the following purposes: 99107

1. Medium Voltage Power (MVP) - 5000 volt insulation for use on 4160 volt resistance grounded system. 4 provided per unit.

2. Low Voltage Power (LVP) - 600 volt insulation. 16 provided per unit.

3. Instrument and Control (I&C) - 600 volt insulation. 20 provided per unit. 99107

4. Control Rod Drive Power (CRDP) - 1000 volt insulation for use on 140 D-C. 5 provided per unit.

5. Nuclear Instrumentation Systems (NIS) Triaxial Cables. 4 provided per unit.

6. Radiation Monitoring Cables (RM). 1 provided per unit. 99107

7. High Radiation Monitor. 2 provided per unit.

8. Low Voltage Power (LVP) - Power supply conductors for containment hydrogen recombiners. 2 provided per unit.

9. Excore Neutron Flux and Incore Thermocouples. 2 provided per unit. 99107

The design, fabrication and installation of D G O'BRIEN penetration assemblies installed in 1981 is in accordance with the requirements of the 1977 ASME Boiler and Pressure Vessel Code, Section III, Division I, Subsection NE. The electrical modules of the penetrations are in accordance with the IEEE Standards 317-1976, 323-1974, and 344-1975.

The design, fabrication, and installation of Conax penetration assemblies installed in 1982 and 1983 is in accordance with the requirements of the Winter 1981 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE. The electrical modules of the penetration assemblies are in accordance with IEEE Standards 317-1976, 323-1974 and 344-1975.

5.2.2.2.2 Piping Penetrations

All piping penetrations listed in Table 5.2-1, except the vacuum breakers, penetrate the Shield Building as well as the Reactor Containment Vessel. All isolation valves that are needed to maintain primary and secondary containment integrity are listed in Table 5.2-1. Both the Reactor Containment Vessel and Shield Building are provided with welded capped spare penetrations for possible future requirements.

All process lines traverse the boundary between the inside of the Reactor Containment Vessel and the outside of the Shield Building by means of piping penetration assemblies made up of several elements. Two general types of piping penetration assemblies are provided; i.e., those that are not required to accommodate thermal movement (designated as cold penetrations in Figure 5.2-5) and those which accommodate thermal movement (designated as hot penetrations in Figure 5.2-6A and Figure 5.2-6B).

Both hot and cold piping penetration assemblies consist of a containment penetration nozzle, a process pipe, a Shield Building penetration sleeve and a Shield Building seal. In the case of a cold penetration, the Containment Vessel penetration nozzle is an integral part of the process pipe, or for instrument tubing and some small bore piping the tube or pipe passes through and is welded to a plate which is in turn welded to the nozzle. For hot penetrations, a multiple-flued head becomes an integral part of the process pipe, and is used to attach a guard pipe and an expansion joint bellows. The expansion joint bellows is welded to the containment vessel penetration nozzle.

At the termination of penetration assemblies on the Shield Building side, a low-pressure leakage barrier is provided in the form of a flexible seal as shown in Figures 5.2-5, 5.2-6A and 5.2-7. These devices provide a flexible closure between the Shield Building penetration sleeve, which is embedded in the Shield Building, and the process pipe. In the case of hot penetrations 8 and 11, a circular plate is used rather than a flexible seal as shown in Figure 5.2-6B. This plate serves as both an anchor and a Shield Building seal.

5.2.2.2.2.1 Hot Penetrations

A hot piping penetration assembly is used when the differential between the normal operating temperature of the fluid carried by a process line and the Reactor Containment Vessel wall temperature would create unacceptable thermal or cyclic stress at the attachment of the Vessel penetration nozzle.

In addition to the elements contained in a cold piping penetration assembly as shown in Figure 5.2-5, a hot assembly has a multiple-flued head, a guard pipe, an expansion bellows and an impingement ring. The multiple-flued head is machined from a solid forging. It is welded into and becomes an integral part of the process line. The inner flue provides support for the guard pipe and the outer flue provides support for the expansion joint bellows. The guard pipe is located concentric to the process pipe, and is cantilever supported by a weld attachment to the inner flue of the flued head. The length of the guard pipe is set so that it extends past the Reactor Containment Vessel penetration nozzle into the Vessel.

Adequate support is provided for the multiple-flued heads and the process line by the shield building which acts as a horizontal and vertical guide. As inferred above, the flued head fitting is the only part of the penetration assembly which comes into contact with the Shield Building at any time. This interaction takes place in one of two ways and is described as follows:

a. For the main steam, feedwater and residual heat removal penetrations, the multiple-flued head passes through a sleeve in the Shield Building as shown in Figures 5.2-6A and 5.2-7. The sleeve acts as a horizontal and vertical guide which allows rotational and axial movements. The piping system and hence the flued head is allowed to rotate or move axially within the Shield Building sleeve but is restrained by the sleeve from moving in any direction perpendicular to the axis of the process line for all seismic, temperature, weight and jet loads. There are no pressure loads that have any effect on the flued head - Shield Building interaction for these assemblies other than the vertical and transverse movements of the Containment Vessel due to internal Vessel pressure. The loads due to this movement are small, being a function of the transverse spring constant of the penetration expansion bellows, and have been considered in the design of the process line, the multiple flued head and the Shield Building sleeve.

   The main steam hot penetration assembly analysis for thermal, pressure, seismic and pipe rupture loads was based on an equivalent set of static loads quantitatively representing the dynamic loading conditions. Seismic loads were based on the results of a dynamic model analysis of the main steam piping system. Pressure loads were analyzed for both the process and guard pipes at maximum operating pressure conditions. Jet load on the guard pipe was considered in a worst case manner as a force, equal to the main steam pipe rupture reaction force, acting at the end of the guard pipe. Stress levels due to temperature gradients were developed based on conservatively assumed step function changes in steam temperature at maximum steam flow conditions. These conditions and loads were included in the specification given to the vendor and used in this analysis of the main steam hot penetration. All calculations for the resultant stress distribution were based on finite element computer techniques. The individual load cases were summated to produce a required combination satisfying the total maximum loading conditions. The finite element analysis showed that the resultant stresses were well below the code allowables.

b. For the steam generator blowdown and letdown line penetrations the multiple-flued head is anchored to a sleeve in the Shield Building as shown in Figure 5.2-6B. All movement of the flued head and consequently the process line is restrained at this point. The design of this anchor and the process line considered all loads due to seismic, weight, temperature, pressure and pipe rupture jet effects of both the piping system and the structures.

The spacing between the process pipe supports and the flued heads for systems with hot penetration assemblies was determined for each system by performing stress analyses considering all operating conditions including the OBE, DBE and process pipe rupture. The piping system and all pipe supports, including the flued head assembly (which is an integral part of the process pipe) was modeled in these stress analyses. The flued heads were modeled in the stress analysis as either points of lateral restraint or anchors. Based on the stress analyses results, modifications were made in the location and type of hangers and restraints until satisfactory results were obtained.

After the pipe support locations and types had been determined for thermal, deadweight and seismic conditions, the pipe rupture analysis of each system was performed in accordance with Section 12.2.2.1.9. Additional restraints were added as required to satisfy the pipe rupture criteria as stated in Section 4.6.2. In the pipe rupture analyses, the flued heads were modeled as either points of lateral restraint or anchors as was done for the operating conditions mentioned above.

The loads acting on the flued heads as determined by the above analyses were then included in the flued head assemblies stress analysis. The resultant stress levels in the hot penetration assemblies were maintained well below the allowable values. The results shown for the guard pipe on Table 5.2-2 are typical.

The relative motion between the Shield Building and the process pipe and the Shield Building and the Containment Vessel was determined from the above piping system analyses and the seismic and DBA analyses of the pertinent structures (see Figure 5.2-1). For the large piping hot penetration assemblies as shown in Figure 5.2-6, the Shield Building acts as a guide (lateral restraint) for the flued head assembly, thereby allowing for the relative axial motion between the Shield Building and the process pipe.

Relative motion between the Shield Building and process pipe is due to thermal expansion of the process pipe for the various operating conditions and/or the relative seismic displacements between the Shield Building and the reactor support structure to which the process pipe is anchored. For thermal expansion, the magnitude of these relative displacements varies from 0.25" to a maximum axial displacement of approximately 1.0." The maximum relative seismic displacements, which increase with elevation, vary from 0.032" to approximately 0.110" for the OBE condition. The relative structural displacements have been modeled in the seismic analysis as well as the relative displacements between the Reactor Building and the Auxiliary Building or Turbine Building, as applicable.

For the small piping hot penetration assemblies, the flued head is anchored at the Shield Building, as shown in Figure 5.2-6B. Because of this design, there is no relative motion between the Shield Building and the penetration assembly.

The relative motion between the Shield Building and the Containment Vessel is due to the DBA pressure and temperature growth of the Containment Vessel as well as the relative seismic displacements between the two structures. These relative displacements will affect only the bellows assembly, and are independent of any process line movements. Relative seismic displacements between the Containment Vessel and the Shield Building at the penetration elevations range from 0.032" to 0.148" for the OBE condition. The maximum DBA movement of the Containment Vessel resulting from both pressure and temperature is approximately 1.375" radially and 0.687" vertically (this can be seen in Figure 5.2-1). All of the relative displacements between both the Shield Building and process pipe and the Shield Building and the Containment Vessel have been considered in the design of the bellows assembly and the piping system.

The location of the most critical hot process pipe penetration assembly with respect to the largest relative motion which must be accommodated varies with the direction of displacement considered. Each individual bellows assembly has been designed to accommodate the largest relative motion that is possible for any individual occurrence or combination of occurrence. Table 5.2-3 shows the most critical movements for each hot penetration. These are based on a combination of normal operating, DBA and DBE movements.

The expansion joint bellows is attached at one end to the outer flue of the flued head and at the other end to the Reactor Containment Vessel penetration sleeve. The expansion joint is provided with a double layered bellows that has a connection between bellows for integrity testing. An impingement ring is mounted on the guard pipe to protect the expansion joint bellows from jet forces that might result from a pipe rupture inside containment.

ASME Section III Code Case 1330-1 permits the use of bellows-type expansion joints under Section III of the Code for Class B and Class C vessels under the rules of Section III for a Class B vessel (such as the containment vessel) with the following additional requirements [following each requirement is a discussion of how it is satisfied]:

a. The welded joint (longitudinal) in the bellows portion is to be of the type prescribed for Category A in Par. N-462.1 of Section III. This requires the joint to be full-penetration, to be 100% radiographed, and final weld to be also liquid penetrant or magnetic particle examined.

   The manufacturer's fabricating procedure utilizes a fully approved automatic gas tungsten arc weld (GTAW) stainless-to-stainless, full penetration groove weld procedure. This is followed with 100% Radiographic Examination (RT) via an approved procedure prior to forming the convolutions, following with 100% Liquid Penetrant Examination (PT) via a fully approved procedure after forming. The welding was all fully qualified per ASME Section IX, and the results of RT and PT were fully documented.

b. The bellows portion of the joint is to be attached by full butt type circumferential welds having full penetration through the thickness of the bellows to be followed by examination by either PT or MT or PT only if a "non-magnetic" weld is made.

   Examination of the manufacturer's fabricating detail drawings and their itemized fabricating procedures indicates the double-ply element is attached by a full circumferential weld. The weld was properly qualified and approved, utilizing a procedure (hand GTAW) with a joint configuration very similar to Sketch 1 of Case 1330-1 for joining stainless to carbon steel, (P8 to P1). This entire joint was subsequently inspected using an approved liquid penetrant procedure. The results of this inspection were fully documented.

From the foregoing, it can be readily seen that the design and fabrication fully complies with the requirements of Code Case 1330-1.

All bellows, including those associated with the fuel transfer tube, that are part of containment boundary are fitted with protective covers which are removable for visual inspection.

All flued head materials were given a Charpy V-notch test based on an impact load of 20 ft lbs at 0°F. Process and guard pipes of ASTM-A555 pipe have been radiographed (longitudinal seams) and hydrostatically tested at 1.5 times the process pipe system design pressure prior to fabrication.

The results of the Charpy V-notch tests performed on the materials used for the flued head fittings used in the hot penetrations are as listed on Table 5.2-4.

Since the flued head and guard pipe at the hot penetration are a part of the process pipe, the temperature of the flued head and guard pipe can be monitored by taking the temperature of the process fluid which is approximately the same as the temperature of the process pipe.

Flued heads in the as-forged condition are ultrasonically tested and either magnetic particle or liquid penetrant tested. The process piping to flued head welds are radiographed and either magnetic particle or dye penetrant tested. The guard pipe to flued head welds are also radiographed and the final weld surfaces dye penetrant tested. The bellows are given a soap bubble test while the space between the plies is pressurized with air at 60 psig.

The multiple-flued head with its associated guard pipe and expansion joint bellows provides a leak-tight seal for the extension of the containment boundary where the hot penetration assembly traverses the Shield Building annulus.

Guard pipes are designed, fabricated, and tested in accordance with ANSI B31.1-1967, ASTM A-106, Grade B, ASTM A-155, KC-70, Class I, ASTM A-358, Class I, TP-304, ASTM A-312, TP-316, as applicable. Certified test reports of chemical and physical properties, traceable to heat numbers, were obtained. All welds are 100% radiographed and either dye penetrant or magnetic particle inspected. Allowable stresses are as noted in ANSI B31.1-1967.

The design criterion applied to the hot penetration guard pipes is defined in Table 12.2-13 and is the same as that applied to all Class I piping. The allowable stress values and stress analysis results for normal, upset and faulted loading conditions for the main steam and feedwater guard pipes are shown in Table 5.2-2. The values listed represent the peak values that will occur at any given location in the guard pipe. This table shows that the calculated guard pipe stress levels are well within those allowed by the criteria given in Table 12.2-13 for all loading conditions. The main steam and feedwater penetrations were selected for this study because their failure would have the most severe consequences in the Shield Building Annulus. A review of the design of the other hot penetration guard pipes indicates that calculated stress levels for these guard pipes would be of the same order of magnitude as for the main steam and feedwater guard pipes and well within allowable values as shown in Table 5.2-2.

The expansion joint bellows were designed as part of the containment vessel. All of the bellows are of ASTM-A240, Type 304 materials, designed for a pressure range of -0.8 to 50 psig and a maximum temperature of 268°F. A discussion of the analysis performed for all bellows assemblies is covered in Section 5.2.3.2.2.

5.2.2.2.2.2 Main Steam Line Penetration

The main steam piping penetration assembly, shown in Figure 5.2-7 uses the same elements as a hot piping penetration assembly. In addition, the main steam line is anchored to the interior concrete of the Reactor Containment Vessel. A limit stop designed to control lateral movement but permits axial movement is provided around the main steam line inside containment. This limit stop serves to limit pipe movement in the event of a longitudinal pipe break thus serving to control pipe whip inside containment. The multiple- flued head is also designed to transfer lateral loads that could result in the event of a main steam line rupture exterior to the Shield Building, to a specially designed structural arrangement in the Shield Building.

5.2.2.2.2.3 Equipment and Personnel Access

The equipment hatch and air locks are supported entirely by the Reactor Containment Vessel and are not connected either directly or indirectly to any other structure.

The equipment hatch was fabricated from welded steel and furnished with a double-gasketed flange and bolted dished door. Provision is made to pressure test the space between the double gaskets of its flange.

Two personnel air locks are provided. Each personnel air lock is a double- door welded steel assembly. Quick-acting type equalizing valves are provided to equalize pressure in the air lock when entering or leaving the Reactor Containment Vessel. Provision is made to test pressurize the air locks for periodic leakage rate tests.

The two doors in each personnel air lock are interlocked to prevent both being opened simultaneously and to ensure that one door is completely closed before the opposite door can be opened. Remote indicating lights and annunciators in the control room indicate the door operational status. Each door lock hinge can be adjusted to assist proper seating. A lighting and communication system that can be operated from an external emergency supply is provided within each air lock.

5.2.2.2.2.4 Fuel Transfer Penetration

The fuel transfer penetration provided is for fuel movement between the refueling cavity in the Reactor Containment Vessel and the spent fuel pool. This penetration consists of a 20-inch stainless steel pipe installed inside a 24-inch pipe. The inner pipe acts as the transfer tube.

The outer pipe is welded to the Reactor Containment Vessel. Bellows expansion joints are provided between the two pipes to compensate for any differential movements. A double gasketed blind flange is bolted on the refueling canal end of the transfer tube to seal the reactor containment. The end of the tube outside the containment is closed by a gate valve.

5.2.2.2.2.5 Containment Supply and Exhaust Purge Duct Penetrations

The ventilation system purge duct and make-up duct penetrations are welded directly to the penetration nozzles in a manner similar to the cold piping penetration. The ducts are circular in section and designed to withstand the Reactor Containment Vessel maximum internal pressure. They are provided with isolation valves and blank flanges as described in Section 5.2.2.3.3. The blank flanges were installed to increase the assurance that the penetration will be leak tight.

5.2.2.3 Containment Vessel Air Handling System

Units 1 and 2 share, as a common facility, some portions of the purge and ventilation system, only the supply fan and ducting are shared. All other portions of the Containment

Vessel Air Handling Systems are provided as separate, identical facilities for each Unit. Therefore, the remainder of this section discusses a single Unit, equally applicable to either Unit (except where noted).

5.2.2.3.1 Containment Air Cooling System

The containment air cooling system functions (1) to remove normal heat loss from equipment and piping in the containment during plant operation and maintaining a normal ambient temperature less than 120°F, (2) to remove sufficient heat from temperature monitored equipment to meet the designed thermal gradients, and (3) to depressurize the containment atmosphere to the order of 3 psig and 150°F in the long term post accident. The design performance data and heat removal rates for containment air cooling system are tabulated in Tables 5.2-5 and 5.2-6.

5.2.2.3.1.1 Reactor Coolant Pumps Cooling

The two reactor coolant pumps are single speed centrifugal units driven by air-cooled, three phase induction motors. Each pump is designed to operate at a volute temperature of 650°F and 300°F housing temperature. Cooling air for the pumps is fed into the coolant pump vaults. 00236 The air flowing through the vault which houses the pump removes heat from a pump unit at a rate of 900,000 Btu/hr (750,000 Btu/hr from the motor and 150,000 Btu/hr from the uninsulated section) to maintain its temperature below design temperature. The heated air is then discharged into the containment atmosphere. The stator windings of each motor and the thrust and radial oil lubricated bearing, and the seal injection water are temperature monitored. Should the stator temperature exceed a pre-determined value, a high temperature alarm is actuated in the control room.

5.2.2.3.1.2 Control Rod Drive Mechanism Cooling

The chilled water system provides chilled water during normal operation for the control rod drive mechanism (CRDM) shroud cooling coils. Each control rod drive mechanism shroud cooling assembly consists of two 13,000 CFM fans, two cooling coils and a plenum. The CRDM shroud cooling removes the heat from the containment air which then passes over the drive mechanisms.

At least one CRDM shroud cooling fan is operated at all times during reactor operation. Isolation valves for the chilled water supply to the shroud cooling coils are controlled at the control panel near the chiller. These valves remain open at all times during normal reactor operation. Either cooling water or chilled water can be used to supply the shroud cooling coils. The valves will automatically close upon receipt of a safety injection signal from either unit, loss of power, or actuation of the master isolation switch in the main control room.

5.2.2.3.1.3 Reactor Vessel Support Pads Cooling

The reactor vessel is supported on six individual air-cooled support pads. The support pads are hollow box-type built-up plate structures equipped with ten 1/4-inch plate split steel cooling fins welded to the inside walls. Approximate outside dimensions of the pad structures are 15-inches wide x 18-inches high x 58 inches long. The boxes are welded to the top cap plates of six full-length support columns that are embedded in the concrete shield structure. The box structures support the reactor vessel shoes supplied with the vessel. Attachment of the shoes is by means of bolting.

The support pads provide for the vertical and lateral support of the reactor vessel. In addition, the pads provide a means to obtain a suitable temperature gradient between the reactor vessel support points and the supporting concrete and steel structures of the building.

The pads are cooled by an interconnecting forced air duct system embedded in the concrete shield structure.

The pads are structurally designed for (1) reactor vessel vertical loads, (2) radial temperature expansion friction forces of the reactor vessel, (3) lateral seismic and pipe rupture loads, and (4) temperature stresses caused by temperature gradients within the supports pads.

The pad structure was analyzed as a closed box type structure using STRUDL computer codes.

The main purpose of pad-cooling is to maintain a satisfactory temperature profile along the support coordinate, rather than heat-removal from the support system. The optimum operating conditions are such that the temperature at the bottom plate of the rectangular ventilated pad is kept sufficiently low so that heat transferred from the pads into the surrounding concrete becomes negligible. Based on a design temperature of 650°F for the reactor vessel, the design criteria of thermal gradients across the support system are as follows:

1. The minimum temperature at the integral nozzle interface is 300°F, equivalent to a maximum permissible temperature drop of 350°F in the nozzle.

2. The temperature drop across the side walls of the rectangular finned pad must not exceed 150°F.

3. The temperature at the bottom plate of the rectangular pad is 150°F or lower.

Thermocouples were installed in Unit 1 and were used to confirm that criteria (2) and (3) were satisfied.

With an air flow rate of 1500 cfm per support at 120°F available for the pad cooling, a rectangular finned pad as shown in Figure 5.2-10 was designed to satisfy all the criteria mentioned above.

The calculation results based on the dimensions given in Figure 5.2-10 are summarized in Table 5.2-12.

The predicted heat removal rate of 5580 Btu/hr from each wall of the rectangular pad implies an effective nozzle heat transfer area of approximately 4 ft^2 which is believed to be acceptable by investigating the nozzle configuration and the nozzle interface area. The study also indicated that the effective nozzle area for heat transfer is not so sensitive as to change the temperature profile significantly for the designed ventilated pad. It is therefore concluded that the pad-cooling design is adequate.

5.2.2.3.1.4 Reactor Cavity Cooling

The duct work provides air flow from the containment fan coil units to the reactor gap and the neutron detector wells to remove heat from the reactor vessel, the primary concrete, and the neutron detector housing.

A reactor vessel gap cooling fan, being in parallel with a redundant fan of same capacity, delivers 10,000 CFM of cooling air from two fan coil units (supplying its suction) connected in parallel to the reactor cavity with a 24 inch duct which is connected to a distribution ring of 16 inch pipe surrounding the reactor vessel in the bottom. Air flow is uniformly distributed into the gap by means of eight equally spaced 8 inch exits along the ring duct. The cooling air directed upward in the reactor gap is capable of removing heat from the reactor vessel surfaces below the reactor head flange at a rate of 80,000 Btu/hr and 25,000 Btu/hr from the primary concrete.

Duct work is also provided for the neutron detector cooling purpose. For Unit 1 only, a 6000 CFM booster fan is installed in the containment ductwork in order to assure necessary air circulation and cooling to the neutron detector area. Cooling air at 750 CFM from each duct exit is directed to one of the eight neutron detector wells which house the detector housing tubes. The air flow is capable of removing 10,000 Btu/hr from each neutron detector housing and its walls to maintain the housing temperature below 135°F. The heated air is then forced into the containment atmosphere.

5.2.2.3.1.5 Steam Pipe Penetration Through the Shield Building

For hot or steam pipe penetrations a pipe sleeve is embedded in the Shield Building concrete. The flued head passes thorugh this pipe sleeve. Bridge lugs of trapezoid cross section, having the top surface area of approximately 1/2 x 36 inches in the axial direction, are welded inside the pipe sleeve to allow a gap of 1/16 inch between lugs and the flued head. The hot penetration design is such that no forced air cooling is necessary. Should local contact of the flued head and a bridge lug happen, analysis indicated that natural convective heat loss to the ambient air at 120°F will result in a maximum temperature of 442°F at the contact, and then asymptotically decreasing to 167°F along the circumferential direction of the conduit.

5.2.2.3.2 Containment Internal Clean-up System

The Containment Internal Clean-up System is sized to reduce the containment airborne activity to a level which would permit two hours of access with reactor at power after the system has been in operation for a period of 32 hours. It is also intended that the units could be operated prior to a plant shutdown to reduce the inventory of fission product activities in the containment atmosphere which would otherwise be removed in the purge system.

5.2.2.3.3 Containment Vessel Purge and Inservice Purge Systems

The purge and inservice purge systems consist of a shared tempered air supply and an exhaust system which discharge through either a purge or a vent filter to the exhaust fan. The fan discharges to the monitored Containment System Vent which extends through the Shield Building annulus to a point approximately five feet above the lower edge of the Shield Building domed roof. The equipment of the purge and ventilation system is located outside the Containment Building. These two systems provide for containment venting and purging:

a. Containment purge system (33,000 cfm), utilizing 36-inch supply and exhaust line used to ventilate containment following reactor shutdown to permit access for inspection and maintenance. One isolation valve and one isolation damper are provided in each supply line and the exhaust line. Each receives an automatic closure signal on receipt of a safety injection or containment high radiation signal. The exhaust and supply ducts also have blank flanges installed inside containment, whenever the reactor is above cold shutdown, which serves the containment integrity function.

b. Containment inservice purge system (4,000 cfm), utilizing 18-inch supply and exhaust lines, which provides charcoal adsorption and particulate filtration of containment air prior to release. This system is used as a low volume normal purge and vent system during cold shutdown and refueling operations. It may also be used to assist the internal clean-up system in permitting containment access when airborne radioactivity levels preclude entry during other plant conditions. Two containment isolation valves are provided on each supply and exhaust line which receive an automatic closure signal on receipt of a safety injection or containment high radiation signal. The supply and exhaust lines will normally have blank flanges installed where the lines pass through the shield building annulus. Normally, during power operation the blind flanges serve the containment integrity function. When containment purge is required, the valves will be leak tested and the blank flanges removed and replaced with a spool piece. A debris screen is installed on each line inside of containment preventing foreign material from inhibiting the proper closing of the valve.

The containment ventilation isolation circuitry is designed so that the ventilation isolation valves will not reset until all signals have been cleared. Once reset, the circuit will immediately accept a new trip signal, and the valves will not reopen without operator action.

The containment inservice purge valves were analyzed to show they are capable of closing during the design base LOCA. The analysis was performed by Henry Pratt Company and is discussed in Reference 1. It was concluded in the analysis that the valve structure and the valve actuator are capable of withstanding combined seismic and LOCA-induced loads.

5.2.3 Performance Analysis

5.2.3.1 Primary Containment Auxiliary System

5.2.3.1.1 Isolation System

Assurance that valves of adequate leak tightness are provided for containment isolation service is obtained through the specification and testing of these valves in accordance with valve manufacturer's standard practice MSS-SP-61 "Manufacturer's Standardization Society, Standard Practice Edition 1961." The specified maximum permissible leakage rates, as manufactured, is 1/10 of a standard cubic foot of air per hour per inch of diameter of nominal valve size. Leak tightness of valves over an extended period will be tested as part of the integrated leak-rate tests for the Containment System and as part of the periodic valve operability and leakage tests.

The containment isolation valves have been examined to assure that they are capable of withstanding the maximum potential seismic loads with respect to the following:

a. The design of the overall valve assembly - body, bonnet, yoke, operator, position-indicating limit switches, and other appendages is reviewed for adequacy with respect to the accelerations and resultant loads at the valve location. Insofar as possible, valves are located in a manner to minimize the magnitude of the accelerations to which they will be subjected.

b. For those valves which must operate under seismic loading, the operator forces have been reviewed to assure that system function is preserved.

c. Control wires and piping to the valve operators have been designed and installed to assure that the flexure of the line does not endanger the control system.

Piping extending from containment to the outside isolation barrier (valve or closed system) are designed to the same seismic criteria as the Reactor Containment Vessel and are assumed to be an extension of containment.

In order to qualify as containment isolation, valves or closed systems inside the containment must be located or protected from potential internally generated missiles such that there is no loss of function following an accident. Also, manual or remotely operated valves must be of a type that can be either locked closed or otherwise be maintained under administrative control in the closed position.

All power operated (air, motor) containment isolation valves in non-essential systems are designed to close upon receipt of an automatic isolation signal. No valves change position when the containment isolation is reset. In addition to resetting the containment isolation, manual action is required before valves change position.

Penetrations such as the pressurizer relief tank sample line, the primary system vent header and the reactor coolant drain tanks gas sample line penetrations fall into the Class 2 penetration category and may have both isolation valves located outside the containment as stated in Section 5.2.2.1. The above three penetrations are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere and fall under 1971 GDC Criterion 57. As discussed elsewhere, the PINGP was licensed to the draft 1967 GDC. However, at AEC request during licensing, a review of containment isolation was performed against the criterion in the 1971 GDC. This criterion requires at least one isolation valve to be located outside the containment and as close to the containment as practical. Since small pipes and tubing are more easily damaged by missiles than larger pipe, it was considered prudent to locate these small isolation valves outside containment in an area where there is low missile probability and where testing, maintenance, and observation are more easily accomplished. The area in which the valves are located is in a Class I structure and within the Auxiliary Building Ventilation Zone.

The residual heat removal system outlet line is normally closed during power operation and for this reason no isolation signal has been provided for the isolation valve in this line. This valve is open only during the later stages of plant cooldown, i.e. less than 350°F and 425 psig.

5.2.3.1.2 Vacuum Relief System

A malfunction analysis of Vacuum Breaker System components is presented in Table 5.2-7.

Analysis is performed assuming maximum containment cooling with a single vacuum breaker assembly functioning (Reference 30). The results of the analysis of the system are presented graphically in Figures 5.2-8 and 5.2-9. These results assume that the butterfly isolation valve starts to open at a differential pressure of 0.5 psi (external to internal), and no air flow is credited through the valve assembly until it is fully open. 00057

The following assumptions were used for the analysis:

a. No heat energy being added to the containment atmosphere;

b. Two Containment Vessel Internal Spray Systems each operating at runout flow with 0 psig backpressure in containment with water at 70°F; 00057

c. Four containment fan-coil units operating in fast speed with a cooling water flow rate of 1200 gpm each with inlet water temperature of 32°F and a fouling factor for a clean heat exchanger (assuming all four CFCUs are in fast speed provides margin to account for any other active containment cooling mechanisms such as the CRDM shroud cooling coils). 00057

d. Initial ambient conditions both inside the Reactor Containment Vessel and within the Shield Building annulus at 120°F, 14.7 psia.

The analysis indicates that one vacuum relief assembly is sufficient to prevent the Reactor Containment Vessel from exceeding the maximum permissible external/internal pressure differential (0.8 psi). 00057

Concurrent with the vacuum relief of the Containment Vessel, the pressure in the Shield Building annulus is reduced to between 13.2 and 13.3 psia, resulting in an external/internal pressure differential of 1.5 psi. This is well within the design conditions for the Shield Building. 00057

The analysis has been based on conservative assumptions, for example: 00057

a. No heat sources are assumed within containment. Such a condition would not normally occur coincident with such low cooling-water temperature and 120°F initial ambient air conditions;

b. No credit is taken for heat transferred from the steel shell and surfaces of the containment vessel during the transient;

c. No air flow is assumed to pass through the vacuum breaker assembly until the butterfly isolation valve is fully open; that is, air flow is not assumed as the valve is stroking open. 00057

d. The calculation uses the specification air flow rate vs. differential pressure for the vacuum breaker assembly. As shown in Figure 5.2-12, this is conservative relative to the values from the flow testing. 00057

Under these assumptions, a relatively rapid pressure transient was calculated to occur as the result of inadvertently initiating full cooling. The analysis shows that either vacuum relief valve assembly will terminate this transient before it exceeds 0.8 psi differential pressure. 00057

For the case of a loss-of-coolant accident, it is convenient to include the effects of stored heat, which are well-defined as part of the post accident calculations relating to the effectiveness of the cooling. In this case, it is found that the rate of pressure decrease caused by sustained maximum cooling is far less when atmospheric pressure is approached than for the design basis where heat capacitance was neglected. Thus, if the relief-valves were assumed to open as the result of sustained overcooling, the peak transient pressure differential would be much less than the design value.

Detailed CONTEMPT calculations show that a rapid cooldown with two spray units and four fan coil units initiated at 60 seconds following the LOCA will depressurize the containment vessel to 5 psig in 1,000 seconds. The annulus air pressure would, meanwhile, have reached a quasi-steady state vacuum of about -3 in. W.C. (-0.108 psig) from operation of one Shield Building Vent System. Continuation of heat removal by all the cooling units beyond 1,000 seconds will cause the containment pressure to decrease further and to approach atmospheric pressure, but at a very moderate and decreasing rate.

The containment pressure is still slightly above atmospheric even at 1 hour. Operation of the containment cooling units can be controlled, and since containment pressure can be monitored it can also be controlled. By not deliberately overcooling and causing the containment pressure to go below atmospheric pressure, the vacuum relief systems will not be required to operate.

Smaller break sizes delay the occurrence of peak pressure and could result in slightly higher containment temperatures, and hence greater initial rates of cooling when the sprays and added fan cooler start, but the results with regard to time of total depressurization would be essentially unchanged.

5.2.3.2 Containment Penetrations

5.2.3.2.1 General

Seismic loads on Containment Vessel penetration nozzles were determined by performing a dynamic nodal analysis of the piping systems. Response spectra at the piping system anchor points were used in this analysis. These response spectra were developed from the results of a dynamic time-history seismic analysis of the plant structures. Differential movement between points in the various structures have been included in the analysis of the piping.

The validity of the computer program used to perform the piping system seismic analysis was proven by comparison with an independent analysis of selected systems performed by recognized consultants.

Loads on Containment Vessel penetration nozzles due to thermal expansion of the pipe, thermal and pressure movements of the Reactor Containment Vessel, and piping system weight were determined by a flexibility analysis of the piping system. This analysis was performed with the aid of a computer program using established methods documented in technical literature. The piping configuration and supports, restraints or anchors on either side of the penetrations were designed to limit the stresses in the Containment Vessel at the penetration nozzle to the criteria defined in Section 12.2.1.5.

5.2.3.2.2 Hot Penetrations

The design movements specified for the penetrations were chosen to include the maximum relative movements of the shield building and containment as well as the movement of the piping due to pressure, temperature and seismic effects. For a one-time movement beyond the design values the bellows would take a minimum of 20 percent additional movement before major deformation would occur.

Stresses resulting from the combination of loads defined in Section 12.2.1.5 were calculated for a typical process pipe in a hot piping penetration assembly using the cross sectional area of the pipe wall thickness as required to meet 1.5 times system design pressure. Comparison of calculated stress values with Code allowable stresses shows:

a. Thermal stresses are less than 50% of allowable;

b. Combined longitudinal stresses are less than 50% of allowable;

c. Hoop stresses are less than 60% of allowable.

The following analysis was performed for all bellows assemblies, including those associated with the fuel transfer tube, that are a part of the containment boundary to establish the critical stresses and deformations.

a. Pressure Stresses

Hoop stresses were calculated in both the convoluted and straight portions of the bellows assemblies and compared to code allowables.

Bellows are subject to a condition of elastic instability or squirm. The condition is equivalent to elastic column buckling where the pressure end load is the column load and the stiffness parameters of the bellows make up the EI term. Since the buckling loads indicated by classical theory are generally greater than those indicated by physical examples, the buckling or squirm pressures for bellows have been established by test. Allowable pressures for bellows were derated by a factor of 2.5 from the experimentally determined values.

b. Critical Deformations

Axial, lateral and angular movements imposed on a joint by seismic, thermal and design basis accident are converted to an equivalent axial traverse per convolution. The critical levels have been established by empirical means. The criterion for joints used as containment seal is 1,000 cycles of this maximum feasible offset condition.

5.2.3.3 Containment Vessel Air Handling System

The Air Handling, Purge and Inservice Purge Systems are shown in Figure 6.3-1, "Containment Air Handling Systems".

The Containment Cooling System consists of four fan-coil units located in the Reactor Containment Vessel. These will re-circulate and cool the Reactor Containment Vessel atmosphere. The heat sink for the fan coils is provided by the containment and auxiliary building chilled water system during normal summertime operation and by the cooling water system for normal wintertime operation. During emergency situation the heat sink for the fan coils is provided by the cooling water system. Additional circulating fans are provided as required to insure a positive flow of air to the areas around the CRDMS, the reactor cavity, and reactor primary coolant pump motors.

The Containment Internal Clean-up System is independent of the Containment Cooling System and employs two separate trains of filter units and fans. The units consist of activated charcoal filters with a capacity of approximately 4000 scfm per unit, which will provide a 10% recirculation of containment atmosphere per hour when both units are operating. The flow through the units passes through HEPA filters before entering the charcoal filters.

These units are not considered a part of the engineered safety features, are not missile protected, and are not intended to operate in the post-accident environment.

After cooldown for shutdown entry, the Reactor Containment Vessel is purged, if necessary, using the containment purge or inservice purge system to reduce the concentration of radioactive gases and airborne particulates.

The inservice purge system may be utilized when it is necessary to prepare the Reactor Containment Vessel for personnel entry in the hot shutdown condition or during power operation to accommodate emergency repair. While the emergency repair or inspection is being performed, radioactivity levels within the Reactor Containment Vessel will be monitored to assure personnel safety, and access will be limited accordingly.

Pressure buildup in containment is of slight concern in this plant. Numerous sources can contribute to a pressure rise; however, leakage from the reactor coolant system and instrument and equipment operational air are the greatest contributors. Instrument air leakage is minor since no constant bleed control valves are used, equipment air leakage will be mainly from control valves and any fitting leaks. A small flow of air will continue to discharge through the containment air monitor to the auxiliary building vent system. The discharge will negate the need for frequent purging of containment and reduce pressure buildup due to instrument air leakage.

The purge and inservice purge systems are designed on the following assumptions:

a. Reactor coolant leakage of 30 lbs/day

b. 1% fuel defects in the core

c. Containment Cleanup System Operation for 32 hours

These assumptions produce a Xe-133 concentration of 4.6×10^{-4} µCi/cc. If I-131 was dispersed in the containment atmosphere, a concentration of 1.2×10^{-7} µCi/cc, would be obtained, however, I-131 has not been reported in other containment atmospheres. The purge systems are designed to reduce the Xe-133 concentration to at least half the 4.6×10^{-4} µCi/cc value.

Prior to normal entry into containment, radiological airborne conditions will be determined and actions will be taken, if necessary, to keep personnel exposures ALARA. Engineering controls such as the containment cleanup system or inservice purge may be used to reduce containment atmosphere radioiodines. Purging using the inservice purge system through the charcoal filter (4000 cfm) would require approximately 5 hours for one air exchange.

During the purge operation, radiation monitors in the Containment System will be used to assure that the discharge rate and filtering efficiency is such that dose rates set by the ODCM are not exceeded. Purge flow is controlled from a local control station. Radiation levels are monitored in the Control Room.

Should some condition occur to cause the air activity to increase above the allowable setpoint of the Containment System Vent Monitor, the vent and purge line dampers would both be automatically closed.

With the single exception of the containment purge system, all of the Containment Air Handling systems are operated remote-manually from the main control room by control switches. The instrumentation associated with each of the systems consists of flow test facilities for initial setup; temperature sensors in the air stream and in the case of the Control Rod Drive Mechanism Cooling Booster Fans, air temperature and fan running alarms.

The Containment Fan Coil Units are also controlled remote-manually from the main control board, but are responsive to safety injection signals so as to switch to an optimum mode to handle a loss of coolant accident. The Containment Fan Coil Units provide for all heat removal, by passing containment air over water cooled coils, the cooled discharge air then feeding the other subsystems of the containment cooling system. On the Containment Fan Coil subsystems, there are temperature sensors in the air and water passages as well as flow sensors in the water passages. During normal plant operation, the chilled water or cooling water flow to the cooling coils is modulated by an orifice on the water return line from each train of Fan Coil Units. A bypass valve around the orifice opens on an "S" signal to return the system to full flow.

Fan coil units inside containment are provided with water from the plant cooling water system when they are operating in their safeguards mode. Portions of the cooling water system serving the fan coil units are designed to tolerate a single active failure, designed as Class I seismic, and are missile protected. With the exception of the initial hours after an accident, Cooling Water System pressure exceeds postulated containment accident pressure. Thus, there is minimal potential for leakage of radioactive material out of the containment via the cooling water system. Any leakage would be detected by the Radiation Monitors and the affected FCU isolated as discussed below.

The cooling water supply lines to the fan coil units are provided with a remote manual motor operated gate valve outside containment. Return lines are provided with a remote manual motor operated gate valve inside containment and a remote manual motor operated globe valve outside containment.

In the event of accident, the cooling water supply and return isolation valves position to full open to satisfy their safeguards function. In the event of a fan coil unit or associated piping rupture the containment remote manual motor operated isolation valves would be closed to prevent the entry of non-borated water into containment. Pressure against the closed isolation valves is maintained by equalizing lines. The water supply for this "seal" is provided by the cooling water system pumps (3 motor driven and 2 diesel driven) which take suction from the Mississippi River.

A lapse of integrity of the cooling water system piping or fan coil units inside containment would be indicated immediately by an increase in containment unidentified leak rate. Leakage would also be evident during monthly inspections of the containment. System pressure tests are conducted in accordance with the Prairie Island ASME Code Section XI Inservice Inspection and Testing Program at least once each inspection interval (10 years).

The containment purge and in-service purge line isolation valves are normally closed during operation. If in-service purging is being performed the valves are closed upon safety injection, manual containment spray, manual containment isolation or on detection of high radiation at the shield building vent stack monitors.

With respect to the testing of the automatic activation instrumentation, and specifically those of the radiation monitors, an actual or simulated signal is used driving it up past its trip point. Since the operation of the inservice purge valves in carrying the test to completion will not interfere with normal operation, the foregoing test is made to completion.

5.2.3.4 Containment Vessel Instrumentation

The containment vessel is provided with redundant instrumentation to provide the control room a continuous indication of the containment vessel pressure, water level and hydrogen concentration.

The redundant instruments are provided per requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements." The addition of containment pressure and water level monitoring instruments are further discussed in Section 7. The addition of the hydrogen concentration monitor is discussed in Section 5.4.2.4.

5.2.4 Inspection and Testing

5.2.4.1 Containment Vessel

5.2.4.1.1 Pre-Operational Quality Assurance and Testing

5.2.4.1.1.1 General Requirements

Test, code, cleanliness, and Quality Assurance requirements accompanied each specification or purchase order for work materials and equipment. Tests performed by the supplying manufacturers are enumerated in the specifications together with the requirements, if any, for test witnessing by an inspector. Fabrication, including final cleaning and sealing, are described together with shipping procedures. Standards and tests were specified in accordance with applicable regulations, recognized technical society codes, and current industrial practices.

5.2.4.1.1.2 Quality Assurance

Fabrication procedures, non-destructive testing, and sample coupon tests for the Containment Vessel are in accordance with ASME Code for Boilers and Pressure Vessels, Section III, Subsection B.

All materials incorporated into the Containment Vessel and airlocks were subject to inspection at mill, shop and field and all materials conformed to the testing requirements of the ASME Code. All seam welds for the containment shell were 100% radiographed. All penetrations nozzles are welded into the shell and were radiographed or inspected by dye penetrant methods where radiographic methods could be ambiguous or difficult to interpret.

The Reactor Containment Vessel design, fabrication, material and testing conformed to all the requirements of the ASME Code and the ASME Code stamp of acceptance has been issued for and applied to the vessel.

During an NRC inspection at another facility, examination by radiography of primary containment liner penetration sleeve-to-process pipe (flued head fitting) welds revealed rejectable defects not originally found by ultrasonic examination. Ultrasonic signals from the weld backing bar apparently masked signals from defects. IE Bulletin 80-08, "Examination of Containment Liner Penetration Welds", was issued to acquire information from all facilities to determine the generic nature of the problem. The initial Prairie Island responses to IE Bulletin 80-08 (References 22, 23, 24 and 25), identified two penetration butt welds with backing rings (Unit 1 penetration 7A, field welds #8 and #14) which were ultrasonically tested but could not be successfully radiographed as required by IE Bulletin 80-08. Per recommendations made in Reference 26 for the resolution of this issue, Northern States Power provided additional information for staff review in Reference 27. The NRC Staff reviewed the information provided by Reference 27 and found it acceptable (Reference 28). The Prairie Island response to IE Bulletin 80-08 was closed by the NRC Staff in Reference 28.

5.2.4.1.1.3 Testing

The Containment Vessel and Airlocks Specifications required acceptance testing was carried out on the constructed Containment Vessel prior to installation of internals and penetrations.

These tests included soap bubble tests at 5 psig and 41.4 psig, an over-pressure test at 51.8 psig, and an integrated leakage test at 41.4 psig.

After successful completion of the initial soap bubble test, the pneumatic pressure structural test was performed on the Containment Vessel and each of the personnel airlocks at 51.8 psig. Both the inner and the outer doors of the personnel airlocks were tested at this pressure.

After placement of external support concrete, removal of temporary stiffeners, T-ring girder and pipe columns, and placement of internal support concrete (stiffener for knuckle), but prior to fueling the reactor, the over-pressure test (at a pressure of 51.8 psig) was performed to provide assurance that removal of the stiffeners or other changes in the system during construction have not compromised the structural integrity of the Containment Vessel. This test was in accordance with the requirements of paragraph UG-100 of Division 1, Section VIII of the ASME B&PV Code.

The test pressure for this test was determined in accordance with the rules of paragraph N-1314 (d) of Section III and paragraph UG-100 (b) of Section VIII of the Code as stated in Section 5.2.1.

Following the successful completion of the soap bubble and initial over- pressure tests, the leakage test at 46 psig pressure was performed on the Containment Vessel with the personnel airlock inner doors closed.

Pressure was maintained for the length of time required to demonstrate full compliance with the airtightness requirements.

The leakage rate was determined by the "Reference System Method" which consists of measuring the pressure differential between the contained air and that of a hermetically closed reference system within the Containment Vessel. The "Absolute Method" which consists of measuring the temperature, pressure and humidity of the contained air, and making suitable corrections for changes in temperature and humidity was used to confirm the results. The results of both methods were in agreement.

Continuous hourly readings were taken until it was satisfactorily shown that the total leakage during a consecutive 72 hour period did not exceed 0.15 per cent of the total contained weight of air. The actual loss per 24 hours was less than 0.02%.

On September 21, 1973, a report "Containment Vessel Strength Test - June 17, 1973" (Reference 2), was submitted to the NRC. The purpose of this test (a repeat of the over-pressure acceptance test) was to provide assurance that removal of the stiffeners or other changes during construction did not compromise the structural integrity of the containment vessel. A tabulation of penetration displacements is included in Appendix A of Reference 2. A comparison of the above displacements data with Figure 5.2-1 indicates that containment displacement under pressure was somewhat greater than predicted by calculations.

5.2.4.1.2 Periodic Leak Tight Integrity Tests

The specific leakage test program to verify that potential leakage from the containment system is maintained within allowable limits is listed below. All testing is performed in accordance with the 10CFR Part 50, Appendix J, Leakage Rate Testing Program. The type of testing applied to the specific penetrations (Type A, B, or C in Appendix J 10CFR50) is identified in Table 5.2-1.

5.2.4.1.2.1 Integrated Leakage Rate Tests

a. An integrated leakage rate test (Type A test) was performed prior to initial plant operation at the Design Basis Accident peak pressure and at an intermediate test pressure to establish the respective measured leak rate. The minimum test temperature was 50°F.

b. Periodic testing is performed per ANSI/ANS 56.8-1994 and the Appendix J Leakage Rate Testing Program. The test duration is adequate for integrated leakage rate measurements, and the validity and accuracy of the testing results is verified by supplementary means.

c. Fluid systems which, under post-accident conditions, are open to the primary containment atmosphere (as defined by ANSI/ANS 56.8-1994) will be vented to the containment atmosphere prior to the test or, if not vented, provisions are included in the test procedure to account for this. Closure of containment isolation valves will be accomplished by the normal mode of operation.

d. Acceptance Criteria - The maximum allowable leakage rate shall not exceed the limit set forth in the Technical Specifications.

e. Frequency of periodic integrated leakage rate tests is specified in the Appendix J leakage rate testing program.

f. Type B and Type C testing will determine leak tightness and the leakage rate if significant leakage during the Type A test is detected.

5.2.4.1.2.2 Isolation Valves and Local Leakage Rate Tests

a. Isolation valves are tested for operability as required by the Inservice Testing Program.

b. Testable isolation valves and other penetrations of the Containment System are leak-tested at a pressure of 46 psig as required by the Appendix J Leakage Rate Testing Program.

c. Bolted double-gasketed seals are tested whenever the seal is closed after having been opened.

d. If the combined leakage rate from all local leakage tests as determined by the sum of the most recent results for each penetration exceeds the limits in the Technical Specifications, repairs and retest are performed to demonstrate reduction of the combined leakage rate to the acceptable value.

"Type B" penetrations include items such as blind flanges, air locks, double bellows seals for certain piping penetrations, flange seals for the equipment hatch, and electrical penetrations of the containment shell. Local leakage rate test procedures for these penetrations are, to the extent practical, conducted in such a manner as to determine whether the penetration seal is leak-tight and to quantify the leakage rate if significant leakage is determined. The normal procedure involves pressurization between seals with leakage rate determined by pressure decay, metering supply flow to maintain equilibrium pressure, or other appropriate methods.

"Type C" penetrations involve the valves identified in Table 5.2-1 and the containment leakage rate testing program. Leakage tests for these isolation valves utilize similar test procedures; however, the procedure for a specific set of valves takes into account the equipment arrangement, location and accessibility factors. Similar to the Type B penetrations, the procedure will determine the leakage rate by pressure decay, metering supply flow to maintain equilibrium pressure, or other appropriate methods.

5.2.4.1.2.3 Residual Heat Removal Systems

a. Those portions of the Residual Heat Removal System external to the isolation valves at the containment are hydrostatically tested at 350 psi at each refueling outage.

b. The total leakage from either train shall not exceed two gallons per hour. Visible leakage that cannot be stopped at test conditions shall be suitably measured to demonstrate compliance with this specification.

5.2.4.1.3 Containment Leak Tightness

Continued leak tightness of the containment and penetrations during the operating cycle is inherent to the design, fabrication and periodic testing requirements.

The Containment Vessel is conservatively designed in accordance with the ASME Code for nuclear vessels and was rigorously analyzed for loading conditions of a design basis accident as well as all other types of loading conditions that could be experienced.

The welds and shell plates are designed as an integral independent structural system. No leaks in weld seams are credible once the leak-tight integrity of the vessel has been established. Furthermore, it is implausible to postulate any conditions that would contribute to leakage of the vessel weld seams during normal operation.

As discussed in Section 5.2.4.1.1.2, all seam welds for the steel shell were 100% radiographed. All penetration nozzles are welded into the shell and were radiographed or inspected by dye penetrant methods where radiographic methods could be ambiguous or difficult to interpret.

The Containment Vessel structural integrity test was performed at an over-pressure in accordance with the ASME Code. The initial acceptance leakage rate test was performed at 46.0 psig and the leakage rate was less than 0.02% (by weight) in 24 hours.

In addition to these strident design and fabrication measures, the periodic testing described above in Section 5.2.4.1.2 provide reasonable assurance of continued containment leak tight capability.

5.2.4.2 Electrical Penetrations

Each prototype penetration assembly including connectors was tested to assure the integrity of design and materials. The tests included the following:

a. Leakage Rate Test

b. Pressure Test

c. Environment Test

d. Thermal Test

e. Short Circuit Tests

f. Insulation Resistance Tests

g. Voltage Tests

h. Repeat Leakage Rate Test

i. Electrical Continuity Tests

j. Seismic

Production testing of electrical penetration assemblies consisted of the following:

a. Leakage Rate Test

b. Dielectric Strength Test

c. Insulation Resistance Tests

d. Electrical Continuity Tests

Tests and sequence of test performed on both prototype and production penetrations are listed in Table 5.2-10.

For the electrical penetrations installed during original construction, leakage rate tests were performed using Helium leak detection procedures. Tests were performed at a temperature of 270°F with a helium differential pressure of 52 psig. The acceptance criteria for the maximum allowable leak rate was 1×10^{-6} cc per second. Leakage tests were performed twice during the sequence of tests on prototype model and once on each production unit.

The measured leakage rates during the testing satisfied the acceptance criterion.

Electrical penetrations installed subsequently have been tested per IEEE 317-1976.

5.2.4.3 Containment Vessel Air Handling System

The ventilation isolation valves are included as part of the containment isolation systems listed in Table 5.2-1. In-Service Purge and Containment Vent and Purge Systems normally have blank flanges installed for containment isolation.

The valves immediately outside the Reactor Containment Vessel are conventional butterfly valves, specified to be adequately leak-tight with maximum internal pressure inside the Containment Vessel. The inservice purge valves inside the Reactor Containment Vessel are butterfly valves which are leak-tight with maximum internal pressure on either side of the valves. This permits the space between the two inservice purge isolation valves to be pressurized to the maximum internal pressure at any time to ascertain continued leak tightness. The purge isolation valves will fail in the closed position upon loss of actuating power (electric or air).

The ventilation dampers and valves which perform a containment isolation function are designed for the necessary earthquake loadings. These valves have withstood tests at 0.5g horizontal simultaneous with 0.25g vertical accelerations. Each isolation valve was reviewed during final design of piping systems to determine the extent of support required. Most of the valves are supported by the piping system of which they are a part. Special supports are included for any valves which might require special consideration.

5.2.4.4 Vacuum Relief System

A prototype test on one complete vacuum breaker assembly, including the butterfly isolation valve, have been performed to verify that functional requirements (flow and pressure) are met. The test results are shown on Figure 5.2-12. 00057

The leakage test for the vacuum breaker system is accomplished by pressurizing the piping between the vacuum breaker valve and the butterfly isolation valve. The vacuum breaker assemblies are provided with a test connection in the piping between the vacuum breaker valve and the isolation valve. These valves are leak tested per Technical Specification 4.4.

The vacuum breaker is equipped with an externally mounted, air actuated, spring loaded, fail safe mechanism for testing the vacuum breaker locally during operation. Limit switches indicate full open and full closed positions of the vacuum breaker valve. A test circuit in the controls for the butterfly valve permits testing of the isolation valve.

5.2.4.5 Containment Periodic Inspection

A periodic inservice inspection of the containment vessel is performed to satisfy the requirements of ASME Code, Section XI, Subsection IWE. The inspection program complies with 10CFR50.55a.

5.3 SECONDARY CONTAINMENT SYSTEM

5.3.1 Shield Building Design

5.3.1.1 Design Basis

The Shield Building completely encloses the Containment Vessel, the access openings, the equipment hatch, and that portion of all penetrations that are associated with primary containment. The design of the Shield Building provides for (1) biological shielding, (2) controlled releases of the annulus atmosphere under accident conditions, and (3) environmental protection of the Containment Vessel.

The Shield Building is primarily a shielding structure and as such it is not subjected to the internal pressure loads of a pressure containment vessel. The structure therefore will not be subject to bi-axial tension and cracking due to pressure loads. The reinforcement arrangements are based primarily on the needs to withstand the more conventional structural loads from environmental effects.

The design criteria for the openings are:

a. To provide reinforcement around the openings to carry all loads by frame action. Because the Shield Building wall thickness is set to meet radiation shielding requirements, the thickness is generally in excess of that necessary for structural requirements; therefore, it was necessary to add additional bars around the perimeter of the opening to provide a reinforced concrete frame.

b. To provide for horizontal and vertical shearing forces acting in the plane of the opening, diagonal bars are provided forming an octagonal pattern of reinforcement around the perimeter of the opening.

5.3.1.2 Description

The Shield Building is a reinforced concrete structure of vertical cylinder configuration with a shallow dome roof. An annular space is provided between the Containment Vessel shell and the wall of the Shield Building to permit construction operations and periodic visual inspection of the Reactor Containment Vessel. The volume contained within this annulus is approximately 374,000 cubic feet.

The Shield Building concrete wall is 2′-6″ thick and the dome is 2′-0″ thick for biological shielding requirements. The design bases for shielding requirements for operational radiation protection are discussed in Section 12.3. The results of analysis with respect to assumed post-accident conditions using these design parameters are discussed in Section 14.9.

The normal ambient temperature in the annular space is set by heat loss through the Containment Vessel and Shield Building. The design assures that the Containment Vessel metal temperature can be maintained above 30°F.

A minimum containment shell temperature of 30°F is required per Technical Specifications to provide assurance that an adequate margin above NDTT exists. The containment has a NDTT of 0°F; therefore, this limit provides a margin of not less than 30°F above NDTT. The Technical Specification also specifies a maximum allowable temperature differential between the average containment and annulus air temperatures of 44°F to provide assurance that offsite doses in the event of an accident remain below those calculated in Section 14. Evaluation of data collected during the first fuel cycle of Unit No. 1 showed that the existing limiting containment shell temperature of 30°F and the limiting temperature differential of 44°F between the average containment and annulus air temperature can be approached only when the plant is in cold shutdown. Additional surveillances, to verify containment air and shell temperatures and annulus air temperature, prior to plant heatup from cold shutdown provide assurance that the above cited parameters are within acceptable limits prior to establishing conditions requiring containment integrity (Reference 3).

Following the Design Basis Accident (DBA), heat transferred to the air in the annular space could cause a slight pressure rise. This temperature-induced pressure transient is limited to less than 5" H_2O by venting the annular space. Conservative assumptions for temperature transmission to the space, and pressure drop in the Shield Building Ventilation system was used in sizing the ventilation system. Following this initial pressure transient, the Shield Building is maintained at a slight negative pressure - approximately 2" H_2O. The Shield Building seals are designed to accommodate these pressures.

The structure was analyzed to assure adequate strength to accommodate thermal stresses resulting from the above temperature-induced thermal gradients.

The following loadings are considered in the design of the Shield Building:

a. Structure dead load

b. DBA load

c. Live loads

d. Wind load

e. Tornado load

f. Uplift due to buoyant forces

g. Earthquake loads

h. External missiles

5.3.1.3 Performance Analysis

The Shield Building was designed so that its inleakage rate is not greater than the amounts indicated in Figure 5.3-1. The contribution to total leakage rate from the various sources of inleakage, with a differential pressure of 1/4" of water are shown in Table 5.3-1. These leakage rates will in most cases vary linearly with pressure, and extrapolations made on this basis are shown in Figure 5.3-1.

The design Shield Building leakage rate increased from an estimated 0.6 volume% per day value given in the Facility Description and Safety Analysis Report to the 8.4 volume% per day value given on Table 5.3-1 due to a change in the type of personnel doors. The personnel doors were changed from a bulkhead type to a weather stripped type in order to improve accessibility in view of their anticipated frequent use.

Subsequently, the as-built leakage characteristics of the Unit 1 Shield Building are higher than predicted in Figure 5.3-1. Figure 5.3-1 was based upon a predicted in-leakage rate of 10% per day at a minus 1/4 inch of water. Measured performance of the Shield Building indicates an in-leakage approaching 50% per day at a minus 1/4 inch water. Figure 5.3-2 reflects this higher in-leakage rate.

The increase in the shield building in-leakage affects the long term exhaust flow from the shield building, increasing the exhaust flow from approximately 200 cfm to approximately 1000 cfm. In turn, the larger exhaust flow has a direct effect on the iodine release rate and results in an increase in calculated thyroid dose.

The allowable leakage rate for the primary containment was changed to 0.25 weight percent per day to compensate for the larger long term exhaust flow from the shield building. Initial preoperational tests of the primary containment showed an actual leakage (measured) of 0.0234 weight percent per day at the design basis accident pressure (46 psig).

The Shield Building penetrations for piping ducts and electrical cable are designed to withstand the normal environmental conditions which may prevail during plant operation and also to retain their integrity during and following postulated accidents.

The openings into the Shield Building, including personnel access openings, equipment access openings and penetrations for piping, duct, and electrical cable, are designed for leak tightness consistent with the specified leakage rates for the Shield Building.

The Shield Building is provided with three access openings, one located adjacent to the maintenance air lock, one adjacent to the personnel air lock and one adjacent to the equipment hatch. Each access opening adjacent to an airlock is provided with double inter-locked doors. A bolted, sealed door is provided at the equipment opening.

Pipe penetrations through the Shield Building are sealed with low-pressure flexible closures. For the cold penetrations, a "flexible boot" is installed, this boot consists of a sewn bellows, fabric reinforced, made of hypalon/nylon material with nylon-backed zipper, neoprene coated seams which are attached to a 3" extension of the penetration sleeve by two stainless steel band clamps at one end. The other end of the bellows is attached to the cold process pipe by two more stainless steel band clamps. A silicone compound is used to seal each end of the bellows to its attachment and is also coated to the external surface of the zipper as a sealant. A prototype suitable for attachment to 8-5/8" O.D. piping was exhaustively tested for structural integrity at 3 psi. At 6" W.C. pressure differential, the leakage for this one penetration was found to be 0.01 cfm. This amounts to considerably less than the 500 cubic feet per 24 hours leakage rate listed for the penetrations in Table 5.3-1. The conservative allowance of 500 cubic feet amounts to only 0.1337% of the total building volume in a 24 hour period, relative to a nominal total inleakage of 8.4%/day.

The design pressure for all of these cold penetration flexible boots established per the above prototype test is 3 psi. The design temperature is the same as the piping system temperature. These flexible boots are not being used on any process penetration above 250°F. The material is capable of 300°F temperature.

For the large "hot" penetrations (those where the process piping temperature will exceed 250°F), the corresponding "flexible boot" is a stainless steel expansion joint, one end of which is attached to a 1/2" thick extension sleeve, field welded to the flued head. The material of this sleeve extension is the same as the flued head. The other end of the expansion joint is welded to a sleeve extension which, in turn is field welded to the Shield Building penetration sleeve. The design is such that the expansion joint, though protected with a light gauge, removable, metallic covering, may be fully exposed for periodic inspection. For small "hot" penetrations (Letdown and Steam Generator Blowdown) the flued head is anchored and sealed at the Shield Building penetration.

Leak testing of these joints is unnecessary. Excessive inleakage into the Shield Building annulus, from these penetrations and others, would be observed during periodic testing of the Shield Building vent system. Surveillance is possible at any time by entering the five foot annulus through double doored access ports.

Flexibility of all cables is provided between the Shield Building and the Containment Vessel so that no damage can occur to the cables or structures due to differential movements between the two structures.

5.3.1.4 Inspection and Testing

5.3.1.4.1 Pre-Operational Testing and Inspection

5.3.1.4.1.1 General Requirements

Appropriate ASTM Material Specifications are cited in the Building Specifications for all construction materials which describe the testing and basis for acceptance of materials. Standards and tests are specified in accordance with applicable regulations and current building practices. The testing of concrete and reinforcing bar welding is referred to in Section 5.3.1.

Inspections were performed as necessary to verify compliance with specifications.

It should be noted that the Shield Building has no significant pressure containing function and, therefore, except for attention to good leak tightness, standard building construction and quality control practices are satisfactory.

5.3.1.4.1.2 Leak Tightness

Provisions have been made to test the leak-tightness of the Shield Building. The leak-tightness will be determined concurrently with the testing of the Shield Building Ventilation System. The testing program of the system is discussed below. Additional discussion of Shield Building Ventilation System Testing can be found in Section 5.3.2. 00092

a. A pre-operational acceptance test was performed and supplemented by analysis that either or both of the redundant trains of the Shield Building Ventilation System are capable of accomplishing the following aspects of their design function following the occurrence of the Design Basis Accident:

    1. to limit the initial positive pressure rise in the Shield Building annulus to 0.5 psi.

    2. to produce a net vacuum everywhere within the Shield Building annulus within three minutes after actuation of the system.

b. This capability has been demonstrated in the absence of accident-related heat sources:

    1. by both systems operating together, and by each system operating independently, with the other system disabled.

    2. under calm wind conditions (5 mph), and again at wind speeds in excess of 20 mph.

c. Each test was initiated from equilibrium conditions by simulation of a safety injection or high containment building pressure signal. The prompt occurrence of net vacuum was measured directly at selected locations during these drawdown tests.

d. The results of these drawdown tests were compared with the results of the analysis, and the analysis model adjusted as necessary to adequately simulate the measured performance of the system.

e. The effects of accident-related heat sources were incorporated in the analysis and it was demonstrated, by means of conservative assumptions where necessary, that the required performance limits would also have been met during actual accident conditions.

f. Each redundant train will be activated separately during these periodic tests to demonstrate its operability.

g. Each system will be determined to be operable at the time of its periodic test if it promptly produces measurable indicated vacuum and obtains equilibrium discharge conditions that demonstrate that shield building leakage is within acceptable limits. 00092

h. The Shield Building Ventilation System is tested to ensure that it will automatically start from a safety injection or a high containment building pressure signal. 00092

5.3.1.4.2 Additional Testing Considerations

The need for an acceptance strength pressure test and possible surveillance strength testing of the shield building was carefully considered during the initial licensing. A review and discussion of the design considerations is presented to substantiate the conclusion that such testing was unnecessary.

Leak tightness provisions were made to test the leak tightness of the Shield Building. The leak tightness was determined concurrently with the testing of the Shield Building Ventilation System. The test determined that the Shield Building does meet the leak rate specified and additionally verifies the leak integrity of the flexible seals on the Shield Building penetrations. Testing requirements for the Shield Building Ventilation System are covered in the Technical Specification.

a. Tornado Considerations

The structure is designed in accordance with the design criteria to withstand the effects of tornadoes (and earthquakes, see Section 12) without loss of capability to perform its safety function. The criteria do not require design provision for strength testing under conditions simulating these natural phenomena, and such a test requirement would be most unusual, even if direct means of simulation were available.

The design calculation significant in the determination of tornado resistance considered the combined frontal pressure effects of the components of a design tornado wind condition: a 300 mph tangential velocity plus a 60 mph velocity of progression. The results of thorough investigation of the non-symmetrical loads and stresses resulting from these frontal pressure effects are described in Section 12.

Primarily as a check calculation for completeness of analysis, the structure was also analyzed for the condition corresponding to an internal pressure of 3 psi, a pressure differential greater than the maximum values reported to be associated with tornadoes. The uniform pressure differential might be construed to relate generally to conditions within the eye of a large tornado.

This 3 psi symmetrical load condition resulted in much lower maximum stresses than those determined for the non-symmetrical loads caused by the design wind condition. Thus, the internal pressure calculation is relevant only in that it demonstrates that a symmetrical load condition is not critical to determination of the tornado resistance of the building.

It is concluded on this basis that a pressure test would not provide a meaningful test of the capability of the structure to withstand a tornado. Neither the function of the structure nor the design tornado conditions relate to pressure vessels; therefore, pressure testing of the structure is not considered relevant in this regard.

b. Pressure Rise Associated with Post-Accident Leakage Collection

The role of the Shield Building in the design basis accident is to provide radiation shielding and in conjunction with the Shield Building Vent System, to collect and process leakage from the containment vessel. The structure is not intended as a pressure vessel, and the small pressure differentials incidental to the collection process do not warrant strength testing of the structure or its penetrations.

The only positive pressure that can be experienced is that due to initial thermal expansion of the containment vessel, and of the air in the annulus before its relief to the atmosphere. The design basis pressure rise, which considers no action of the Shield Building Vent System for the first 36 seconds following the accident, is only 3 inches of water column, or 0.1 psi; the maximum predicted for the case of an exaggerated heat transfer coefficient is 11 inches or 0.4 psi. Following this momentary positive pulse, net vacuum will be established and maintained throughout the annulus by either one or both trains of the shield building Vent System. Initial relief with a low initial positive pressure is assured by action of these redundant engineered safety features plus the outleakage that will occur through the low-leakage structure during the positive pressure interval.

It should be recognized, however that the Shield Building is not designed nor intended to be a pressure vessel; moreover the appropriately calculated peak accident induced pressure (see Appendix G, Section G.3) is no greater in magnitude than the negative pressure induced by test of the SBVS.

Further, although the Shield Building penetration seals have been specified and strength tested to withstand the 3 psi induced pressure differential associated with a tornado, they are all located within the confines of the Auxiliary Building, so that the possibility of subjecting them to that differential is extremely remote, if not impossible.

From a practical engineering standpoint, it appears meaningless to strength test at conditions near its low functional design loading a structure that is designed also to withstand far greater loads from postulated acts of nature.

In conclusion, pressure testing of this structure with regard to post-accident pressure transients seems unnecessary, unprecedented and without purpose.

However, to satisfy a commitment made to the AEC in the FSAR, a one time positive pressure test at approximately 10 inches water column was completed satisfactorily.

c. Accessibility

The only critical parts of the shield building function are the active components of the Shield Building Vent System and the filter units. These are located outside the Shield Building and they are always accessible. The five foot annulus itself, and its penetrations, are also accessible by two access openings having double interlocked doors.

5.3.2 Shield Building Ventilation System (SBVS)

Units 1 and 2 have separate Shield Building Ventilation Systems. The components of each unit are redundant. The remainder of this section, is presented for a single Unit, and is equally applicable to either Unit.

5.3.2.1 Design Basis

The Shield Building Ventilation System is designed to minimize the release of radioactivity from the reactor containment system following the DBA. The system is designed to reduce the release to less than 10% of the limits set forth in 10CFR100 using the ultra conservative TID-14844 assumptions.

The post-accident thermal expansion of the atmosphere within the annulus is prevented from exceeding 3.5" H_2O positive at any time. Negative pressure is established and maintained within 4.5 minutes of the accident. The rate of expansion and pressurization within the annulus is calculated utilizing the containment shell temperature curve resulting from containment pressure transient studies with only one containment pressure-reducing system operative (one train of Fan Coil Units and one train of Containment Spray).

The capacity of the recirculation fan as selected returns the annulus to a negative condition within three minutes after the re-circulation fan is started. The flow capacity of the filter as selected matches the capacity of the fan and the charcoal bed capacity has been checked to assure adequate capability for removing the long term leakage of radioiodine.

The heating elements are designed to produce a relative humidity of less than 70% at the charcoal bed with 100% relative humidity in the air entering the filter.

The exhaust fan is selected with an appropriate head-capacity characteristic to maintain the Shield Building annulus of approximately 2" water column negative pressure for calculated Shield Building leakage rates. In order to provide an operating margin, the size and type of exhaust fan has been selected so that its head-capacity curve would allow it to perform its function for in-leakage flows in excess of 300% of the calculated leakage rating of the Shield Building at the negative pressures set by the fan head (see Figure 5.3-1). Further conservatism is inherent in the exhaust fan selection because the door style selected for the personnel doors on the Shield Building, the equipment hatch and the penetration seals are of a type that are somewhat more leak-tight than the models used for the basis of calculation. The actual leakage rate of the Shield Building as determined by test is shown in Figure 5.3-2.

5.3.2.2 Description

5.3.2.2.1 Design Conditions

The Shield Building Ventilation System is a system of fans and ducts for collecting the leakage from the Reactor Containment Vessel penetrations into the annulus of the Shield Building and discharging it through filters (particulate, absolute and charcoal) to the monitored Containment System Vent.

The Shield Building Ventilation System is normally in a standby condition during normal operation of the plant. Dampers located in the system prevent the flow of air through the filters from wind-induced pressure gradients. The filters are thereby retained in a fresh, unloaded condition for maximum efficiency during post-accident usage. The Shield Building Ventilation System discharge dampers are opened and fans are started by a Safety Injection. (Section 6 describes inputs which result in a possible SI signal).

The Shield Building Ventilation system is designed to provide three functions. One is to produce a slightly negative pressure within the annulus within the initial minutes following the loss-of-coolant accident. The second is to ensure the mixing of any Containment Vessel penetration leakage into a large portion of the Shield Building annulus, thereby avoiding potential direct streaming of the radioisotopes to the exhaust duct and hence increasing holdup within the annulus. The third function is to provide long-term cleanup of fission products from the annulus air by recirculation after the loss-of-coolant accident.

The normal temperature of the air within the Shield Building annulus will be approximately the same as the temperature of the air within the Containment Vessel. In the event of a loss-of-coolant accident, the air temperature would increase as the Containment Vessel shell temperature increases. The resultant thermal expansion of the air would pressurize the annulus during the first few minutes after the postulated accident unless suitably relieved.

Drawing slight negative pressure, relieves any pressure from thermal expansion that could cause out- leakage through the Shield Building. Such out-leakage would bypass the Shield Building charcoal filters, but would be picked up by the Auxiliary Building Special Ventilation System charcoal filters.

The pressure transient in the annulus poses no structural hazard to the Containment Vessel or Shield Building. Since the pressure increase in the annulus results from the pressure and temperature transient imposed on the Containment Vessel following the LOCA, the Containment Vessel internal pressure will be positive with respect to the external (annulus) pressure. The Shield Building is capable of structurally accommodating any foreseeable pressure transient of the annulus air.

The amount and rate of thermal expansion of the air during this initial period is dependent upon the rate of rise in temperature of the Containment Vessel shell and the rate of heat transfer from the shell to the air in the annulus. The rate of venting is set by the flow resistance of the filters and the vent ducting and the characteristics of the recirculation fan, which acts as an exhaust fan until the annulus reaches a sub-atmospheric condition and the recirculation dampers opens. While thermal expansion continues, the recirculation fan exhausts the excess volume of air and maintains a negative pressure in the annulus. The negative pressure is sufficiently low so that no internal effects will cause a localized area of the annulus to return to a positive pressure.

When the annulus has been drawn to a negative pressure, the full capacity of the recirculation fan is available to recirculate air within the annulus to ensure mixing. A smaller exhaust fan is then capable of exhausting the in-leakage to the annulus and continues to maintain a negative pressure in the annulus. In-leakage to the annulus is mixed with annulus air and drawn through the filter units to the monitored Containment System Vent Stack. The recirculation fans continue to recirculate the contaminated air of the annulus through the filters for long-term clean-up during the post-accident period.

5.3.2.2.2 System Description

The Shield Building Ventilation System flow diagram is shown on Figure 6.3-1.

The Vent System consists of two full-capacity, redundant, fan and filter systems which share a common Containment System Vent Stack. The Vent exhaust pipe (stack) is located in the Shield Building annulus and extends approximately five feet above the Shield Building. The fans and filters are located in the Auxiliary Building.

Each system is made up of heater elements, particulate (roughing), absolute, and charcoal filters, all in series, and two fans. One fan is used for recirculation and mixing of the Shield Building air volume and one small fan is used to hold the annulus at a slightly negative pressure with respect to the atmosphere.

The discharge of the recirculation fan can either flow to the Containment System Vent Stack through an automatically operated damper, or can be recirculated to the annulus through another automatically operated damper.

The discharge of the small exhaust fan contains a backdraft damper to prevent wind-induced flow of air through the filter. The recirculated discharge from the larger fan in each system is returned to the annulus through ducting designed to enhance circulation within the annulus.

Back-draft dampers are provided in the discharge lines from both fans. These dampers prevent back flow through the ducts when annulus pressure decreases due to cooldown of the air after the thermal expansion transient. When fan head is insufficient to discharge air due to low annulus pressure, the dampers will seat and allow the negative pressure in the annulus to be maintained. The back draft damper in the recirculation duct prevents out-leakage while the annulus is at positive pressure in the unlikely event of a spurious opening of the automatic damper in the recirculation duct.

5.3.2.2.3 Actuation and System Operation

Following a loss-of-coolant accident the Shield Building Ventilation Systems are placed into operation by the Safety Injection Signal. The signal causes the automatically operated dampers in the discharge lines to the Containment System Vent Stack to open. The fans are started early in the emergency power loading sequence for engineered safety features. The negative pressure setting on an annulus differential pressure switch will signal the opening of the recirculation dampers. There are two pressure signals (pressure switches) one per train, which are separated; using separate penetrations for sensing lines and separated (train) wiring. Testing of this system can be carried to completion at any time without affecting plant operations. Hence the "one-out-of one" per train arrangement meets IEEE 279. An auxiliary contact on the recirculation fan in each loop will allow the recirculation damper in that loop to open only if that respective fan is operating. Following initiation of recirculation, the small exhaust fan continues to discharge in-leakage flow (caused by the negative pressure in the annulus) through the filters to the vent stack, thereby holding a negative pressure throughout the annulus as the recirculation fan continues to recirculate filtered air.

As the Containment Vessel shell is cooled by the Containment Air Cooling Systems, the annulus air will begin to cool causing a further reduction in annulus pressure. During this period, if the annulus pressure draws the system below the head capacity of the exhaust fan, a backdraft damper in the exhaust duct will close to prevent backflow.

In the period following cooldown of the annulus air, the negative pressure and discharge flow will be determined by the Shield Building in-leakage rate and the head-capacity performance characteristics of the fans.

5.3.2.2.4 Component Design

5.3.2.2.4.1 Fans

The exhaust and recirculation fans are vaneaxial, direct-connected fans of standard construction. The recirculation fan is nominally rated at 5000 cfm and the exhaust fan at 00047
200 cfm.

5.3.2.2.4.2 Filter Assemblies

The filter assemblies are composite units consisting of electric heating elements, Pre-filter section, HEPA filter section, and an impregnated charcoal bed filter section. Each section is designed as follows:

a. The heating coils are designed to be capable of increasing the temperature of the incoming air by a sufficient amount to assure a 70% relative humidity entering the charcoal bed with 100% saturated air entering the heaters. Temperature sensing devices actuate the heaters in an on-off fashion to prevent over-heating the heater elements.

b. The high-efficiency particulate filters are designed to be capable of removing 99.97 percent minimum of particulate matter 0.3 micron or larger in size. Filter design is water and fire resistant, and meet all requirements of AEC Health and Safety Bulletin 212-1965. Table 5.3-2 lists the material specifications.

   The units are tested to meet the requirements of MIL Spec 51068, which requires heated air testing at 700°F. These filter assemblies have successfully passed the testing at this temperature which is far greater than those which could be experienced by the filter assemblies as a result of overheating either from fission product decay heat or from a postulated malfunction of an electric heater.

   Radiation resistance of the materials in HEPA filters has shown that the media will lose some tensile strength after exposure, but that filter efficiency is not affected (Reference 4). Tensile strengths are not reduced to the point where filter integrity becomes questionable because of the large margins present in the basic filter design.

c. The iodine filter is an impregnated activated charcoal bed, capable of removing 99.9 percent minimum of elemental iodine (and 95% minimum of methyl iodide) when exposed to an atmosphere at 150°F, 70 percent relative humidity, based on a filter depth of 2 inches and a residence time of 0.25 seconds. The ignition temperature for the charcoal used is greater than 394°C.

   The design parameters of charcoal filters in the Shield Building Ventilation System for the Prairie Island Nuclear Power Plant are listed in Table 5.3-3. These design parameters were originally used for equipment sizing and selection, and do not necessarily reflect operating conditions.

   The Auxiliary Building Special Ventilation System filter assemblies are essentially identical in design parameters.

   Figure 5.3-3 shows the arrangement of the filter assemblies and the relative spacing of heaters, filters, etc.

   The particulate and charcoal filters have a nominal flow rating of 6000 cfm and are sized to retain the fission products released to the Shield Building following any of the postulated accidents without exceeding a loading of 10 mg/gm for elemental iodine or 3 mg/gm for organic iodine. For sizing criteria, it has been assumed that 10% of the total iodine will occur in the organic form.

The heater control scheme design includes energizing the respective electric heater at the inlet to the filter units at the same time the recirculation fan in the Shield Building Vent System is started. Alarms in the control room annunciate if the humidity in one of the vent system trains rises above 70%. Since the electric heaters, under this mode of control, are always energized when the system operates, this heat contribution to the annulus air of the Shield Building Vent System was included in the computer model analysis performed on the SBVS. However this heat contribution to the Shield Building atmosphere from the electric heater in the Shield Building filter assembly is extremely small compared to the overall heat contribution from the containment vessel itself following the loss of coolant accident.

To preclude the possibility of a heater remaining on under a no flow or low flow condition, interlocks have been provided to trip the respective heater in the event the recirculation fan (SBVS) or Auxiliary Building Special Ventilation System exhaust fan trip. A heater trip is also provided on a low flow condition through the filter. Additional low flow protection is provided to trip the heaters upon observing a high temperature near the downstream face of the heater. Failure of a heater to turn off as a consequence of electrical shorting is prevented by overcurrent trip devices provided on each heater.

Independent and diverse safety features are provided in the heater controls to trip the heaters upon indication of any condition associated with loss of flow or overheating, as was described above. These safety features have been incorporated specifically to prevent continued heater output during the postulated loss-of-flow condition. Therefore an assumed failure of the heaters to turn off upon loss of flow is not regarded as credible.

Normally the filters are cooled by the air flowing through them. Even if the air flow is terminated and the filter train isolated, the filter and filter housing will dissipate fission product decay heat without filter damage. The results of an analysis of the charcoal filter ignition hazard are reported in Section 14.9.7. It is concluded that no ignition problems are anticipated with the charcoal filter design.

To provide further protection against fission product release due to high carbon temperatures, a deluge system is installed in each filter assembly, in the Shield Building Ventilation and Auxiliary Building Special Ventilation Systems. Analysis has shown that the carbon temperatures will not become high enough to cause the release of fission products without the deluge system in operation (see Section 14.9.7). Therefore, the deluge system is unnecessary.

5.3.2.2.4.3 Charcoal Filter Water Deluge Feature

Table 5.3-4 lists the single failure analysis for the charcoal filter water deluge feature. The PAC ventilation filters in the SBVS and the Auxiliary Building Special Vent System are equipped with a water spray nozzle for the charcoal beds. Deluge water is supplied to each of these filters from either the Fire Protection Header or Cooling Water Header. A direct acting solenoid valve has been supplied for each filter just upstream of its spray nozzle. These solenoid valves are normally closed and are energized to open by U.L. approved temperature switches, one switch in the filter and one switch after the filter, either switch will actuate spray. Each temperature switch is set at approximately 250°F, which closes a contact thereby energizing the solenoid. Paddle type flow switches are provided upstream of each solenoid valve for remote indication of water flow in the line. The flow switches serve primarily to alarm inadvertent actuation of the spray system.

5.3.2.2.4.4 Instrumentation and Control

Indicating lights and annunciation are provided in the control room for flow and temperature switches.

The temperature switches and associated alarms are periodically functionally tested. Verification of actuation of the solenoid valves is not required as it is not practical and the operability of the PAC filters is not dependent on the fire protection system per this section and section 14.9.7.

5.3.2.3 Performance Analysis

In order to provide assurance that the system will perform its intended function, extensive analytical evaluations have been performed. These have included evaluations of those system variables that might have significant effects on the system performance.

A detailed study of the mathematical models used to predict the behavior of the system was reported in the preliminary FDSAR. Several minor improvements were made to the computer programs to make them more rigorous, although the added sophistication had only minor effects on predicted system behavior. The Shield Building Ventilation System

(SBVS) computer model is discussed in detail in Appendix G. Revision of the computer code to predict SBVS system performance was discussed in "Prairie Island Containment System Special Analysis Report" submitted to the NRC in a letter dated April 9, 1976 (Reference 3). The report also presented the evaluation of computer model accuracy in predicting SBVS performance during power operation in cold weather.

The basic input or forcing function for this analysis is obtained from the Containment Vessel Pressure Transient. The Computer Code used for this latter transient is the well accepted code CONTEMPT developed by Phillips Petroleum for the NRC. This Code analyzes the Containment pressure following DBA and provides a temperature - time history of the containment vessel wall. See Appendix K for additional information.

The SBVS mathematical model involves the following three transient phenomena which interact with each other:

a. Heat transfer from the steel shell to the air in the annulus and that from the air to the concrete wall of the Shield Building.

b. Pressurization and depressurization of the air in the annulus corresponding to the air temperature and air mass remaining in the annulus.

c. The flow of air through a network of ducts, fans, dampers and the charcoal filter system along with the in- or out- leakage of air through the walls of the Shield Building.

5.3.2.3.1 Time History Performance of SBVS

The Shield Building Ventilation System Analysis is described in detail in Appendix G and, therefore, it will not be repeated here. However, it is necessary to briefly describe the various periods of the time history performance of the SBVS. The SBVS is not required to operate for normal plant operation. All dampers are normally closed.

Time Period - 1 (Time 0 to Time SBVS Fans Start)

A Safety Injection signal, following the hypothetical LOCA, causes each SBVS fan to start and the associated fan discharge dampers to open. Until the fans start, the containment vessel expands due to the blowdown pressure and temperature increase. The annulus pressure increases due to the decreased volume and increased air temperature. The pressure continues to increase due to the thermal transient following the blowdown until the SBVS fans start.

Time Period - 2 (Time SBVS Fans Start Until Recirculation Set Point Is Reached)

Period 2 (a) (Time SBVS Fans Start To the Time To Reach Zero Annulus Pressure)

During this period, both the large recirculation fan and the small exhaust fan discharge the filtered annulus air to the Containment System Vent Stack. The pressure continues to drop from its positive peak until it reaches a zero value.

Thus, the annulus is at a positive pressure from time zero until this point is achieved (approximately 2.6 minutes). Hereafter, the annulus pressure is negative for the remainder of the LOCA. The exact time required to reach an average zero pressure differential between the SB annulus and the external atmosphere is not the critical consideration in estimating the dose from the SBVS, but rather it is the total SB outflow that must be examined.

Period 2 (b) (Time Annulus Pressure Reaches Zero Until Recirculation Set Point Is Reached)

The large and small fans continue exhausting the filtered annulus air until the "Recirculation Set Point", (as detected by differential pressure switches that measure the pressure differential between the annulus and the Auxiliary Building Special Ventilation Zone) is reached. Achievement of this negative pressure (-2.0" WC) causes the opening of the recirculation damper, in the discharge of the recirculation fan.

Realizing that the duration of positive pressure and the time required to achieve a nearly steady negative pressure differential influence the SB outflow, a conservative outflow envelope was selected for dose calculations. (See Table 14.9-1 and Figure 14.9-4). During the 0-20 minute period the calculated SB outflow (Figure 14.9- 4) is approximately 30,000 ft^3. The dose calculations assume an outflow of approximately 53,000 ft^3 in this time period. This margin has been conservatively chosen to accommodate arbitrary deviations from the reference case. For example if it is postulated that the wind increases from 0-30 mph during the period of positive annulus pressure, the average external SB surface pressure might drop by 0.5 in. W.C. relative to the average annulus pressure.

Calculations show that an additional 25000 ft^3 would need to be removed from the annulus to overcome the additional pressure differential change. The effect of the wind change is well within the margin of outflow, and the effect on SBVS dose at the site boundary would be an increase of less than 5% of the effect of wind velocity if plane dispersion is ignored. However, when considering the favorable effect to the X/Q values the overall effect would be to actually reduce the dose.

Time Period - 3 (Time Recirculation Begins To Time Equilibrium Recirculation is Achieved)

When the recirculation damper opens, the discharge flow from the large fan hydraulically splits, part being exhausted through the Containment System Vent Stack and part recirculated to the annulus. The annulus pressure rises slightly though still negative, until the fans reverse this trend once again. All during this time the thermal transient is leveling off and the ratio of exhaust flow to recirculation flow from the recirculation fan is continuously decreasing. This trend continues until equilibrium flow conditions are reached. This time is reached for the DBA in approximately 20 minutes.

Time Period 4 - (Remainder of the LOCA Transient)

The magnitude of this exhaust flow during equilibrium recirculation is equivalent to the Shield Building in-leakage plus primary containment leakage to the annulus. This becomes an important parameter in that it sets the relationship between the amount of filtered exhaust discharged to the environment and that retained and recirculated. Minimizing the Shield Building in-leakage increases the effectiveness of the system during long-term operation.

5.3.2.3.2 Significant Parameters

The significant parameters during the four time periods mentioned earlier are discussed in the same order.

Time Period - 1

a. Fan Starting Time

   The analysis assumes a 36 second delay following a DBA before the fans are started. This time delay allows for the Diesel Generator starting, loading sequence and a conservative margin of an additional 20 seconds.

   With no loss of offsite power the system is in operation in less than 10 seconds and the positive pressure duration is reduced considerably.

   The acceptance test on the Diesel Generator starting and loading sequence verifies the conservatism in this parameter.

b. Heat Transfer Coefficient From the Steel Shell to the Annulus Air

   The heat transfer coefficients are derived from well established experimental data. An extensive literature search was made to determine the most appropriate correlation to be used in the analysis. The results of this search indicated that the correlation used in Appendix G, equations G.3-1 to G.3-4 are appropriate. The details of this search are summarized in the Section following the discussion on Significant Parameters.

   In addition the extensive heat transfer parameter studies are reported in Appendix G.

   Therefore, tests to verify the heat transfer coefficients were not necessary.

c. Instantaneous Expansion of the Shell

   The analysis assumes an instantaneous stretching of the containment vessel due to the blowdown pressure peak thus causing instantaneous pressure rise in the annulus. In addition, for the analysis, no relief is granted to the annulus air even after the vessel has contracted as the containment pressure drops.

Time Period - 2

Period - 2(a): The significant parameters during this positive pressure period are as discussed above in Time Period 1, paragraph (b).

Period - 2(b): The only significant parameter during this period is the set point for recirculation initiation. The set point which causes the opening of the recirculation damper is sufficiently low so as to keep the annulus pressure negative following initiation of recirculation and for the remainder of the transient.

Time Periods 3 and 4

The only parameter of any influence is the heat transfer coefficient. This was discussed above in Time Period 1, paragraph (b).

Other Parameters

There are several other parameters of significance which influence the pressure and dose transient more than any of the above discussed parameters. These are discussed below:

a. Shield Building Leakage

   The SBVS performance analysis is based on an assumed leak characteristic of the Shield Building of 10%/day at 1/4" W.C. differential pressure. Measurements made during Unit 1 Plant initial tests indicated an in leakage rate of 50%/day at 1/4 inch water. See Section 5.3.1.3 for additional discussion. The dose analysis is based on an assumed leakage rates shown in Table 14.9-1.

b. Fan Characteristics and Physical Parameters of the System

   Manufacturer's shop tests and the "Pull-Down" test demonstrates the capability of the fans to perform its functions.

c. Containment Vessel Leak Rate

   The SBVS performance analysis is based on an assumed containment leak rate of 2.5%/day. The dose analysis is based on an assumed containment leakage rate of 0.25 weight % day for the first 24 hours. In reality the containment vessel is specified for a leak rate of less than 0.1% day. Therefore the analysis based on a 0.25%/day has ample margin for the offsite dose evaluations. To accommodate the as-built shield building leakage, the Technical Specification allowable leakage was reduced to 0.25 wt%/day. This effectively reduces calculated doses to original design levels.

5.3.2.3.3 Literature Search On Heat Transfer Coefficient By Natural Convection For A Huge Vertical Plate

The Buckingham's theorem (Reference 5), proved mathematically by Langhaar (Reference 6) and known as the basis of all dimensional analyses, has been extensively applied to fluid mechanics and heat transfer in extending experimental data from a model to its full scale prototype. In applying dimensional analysis, the variables significant to a given problem are formed into dimensionless groups which, (without providing any information about the mechanism of the process) aid in correlating experimental data and developing functional relationships between dimensionless groups. The effect of any dimensional factor can therefore be evaluated from such functional relations.

Lorenz's (Reference 7) analytical solution to natural convection adjacent to a heated vertical wall was the first to incorporate all of the variables significant to natural convection. Correlations involving the Grashof and Prandtl numbers had not been popularized at the time of Lorenz's work, however, his solution reduces naturally to the form of equation (1), as dimensional analysis (Reference 8) has predicted.

$$Nu = c \quad (Gr\ Pr)^n. \tag{1}$$

The correlation shown in equation (1) was further reviewed and supported by many outstanding researchers (References 9, 10, 11) whose effort in the prediction of the constants c and n for laminar and turbulent flow conditions was rather remarkable.

It is well established that the transition of flow pattern from laminar to turbulent flow occurs approximately at 2 feet (Reference 12) from the leading edge of a vertical plate. For turbulent natural convective flow along a vertical wall, a widely accepted equation originally correlated by Nusselt and King (Reference 9), and recommended by Jakob and Linke (Reference 10) is

$$Nu = 0.129 \quad (Gr\ Pr)^{1/3} \tag{2}$$

where $10^9 < (Gr\ Pr) < 10^{12}$.

King compared the behavior of the heat transfer on short and long vertical surfaces and led to a conclusion that with an increase in Grashof numbers, Gr, the mean heat transfer per unit area and therefore also the mean coefficient of heat transfer becomes independent of the height. The exponent of 1/3 in equation (2) is naturally consistent with the result of King's observation.

Theoretical studies performed on this topic have also accumulated information enough to support that equation (2) is adequately applicable to a system of higher Grashof numbers, at least up to 10^{15}. Bayley's (Reference 13) theory of applying appropriate temperature and velocity profiles to the boundary layer yielded somewhat different values for c and n, but the solutions are in fair agreement with experimental data in the range of $10^9 < (Gr\ Pr) < 10^{12}$.

$$Nu = 0.10\ (Gr\ Pr)^{1/3} \quad (3)$$

for $2 \times 10^9 < (Gr\ Pr) < 10^{12}$,

and $$Nu = 0.183\ (Gr\ Pr)^{0.31} \quad (4)$$

for $(Gr\ Pr) < 10^{15}$.

Still in a separate study, Eckert and Jackson (Reference 14). who applied Karman's integral momentum equation for the boundary layer and data on the wall shearing stress and heat transfer in forced convection flow of very low Reynolds numbers derived a semi-empirical equation (equation 5) for the turbulent natural convection along a vertical surface.

$$Nu = 0.021\ (Gr\ Pr)^{2/5}\ . \quad (5)$$

Since this equation was proved to be in good agreement with experimental data (References 11, 15) in the range of Grashof numbers from 10^{10} to 10^{12}, Eckert suggested that the equation can be used in the case of higher Grashof numbers. Equations (2) (3) and (5) together with experimental data were plotted in Figure 5.3-4 for comparison.

It is of interest to note that theoretical solutions do indicate an exponential relationship between heat transfer coefficient and the characteristic length in higher Grashof numbers. However, discrepancy of analytical solutions, presumably due to the differences in dealing with temperature and velocity profiles (Reference 16) in the boundary layer, are well covered by the empirical equation (2) in which the heat transfer coefficient is independent of the height. In the discussion of the work of Cheesewright (Reference 17) who tended to agree with Eckert's theory, Warner (Reference 18) pointed out that a definite trend toward milder temperature gradient was detected in the very vicinity of the vertical wall and therefore gave substantial credence to Bayley's theory. Incidentally, heat transfer coefficient predicted by using equation (2) is approximately 22% higher than that obtained by Bayley's equation at Grashof number in the order of 10^{14}.

The predicted Grashof numbers for the Prairie Island containment vessels are in the range of 10^{13} to 10^{14} with a vertical height of 130 feet.

Literature survey has indicated that equation (2) is most appropriate for the prediction of heat transfer coefficient of the system because of the minimum deviation from theoretical solutions and experimental data as well.

The turbulent natural convective heat transfer coefficient used in Appendix G (equations G.3-1 to G.3-4) to evaluate the performance of the Shield Building Ventilation System is shown in equation (6).

$$h = 0.196 (\Delta T)^{1/3} \qquad (6)$$

This equation (6) known in a simplified form, is accurate within 3 percent to equation (2). It is concluded that, with experimental and analytical background, equation (6) or equation (2) is sufficiently accurate for predicting the heat transfer coefficients between the vertical surface of the containment vessel and the bulk annulus air. Naturally, it is also valid for predicting heat transfer coefficient between the annulus air and the vertical wall of the Shield Building.

5.3.2.4 Inspection and Testing

5.3.2.4.1 Quality Assurance

The following inspections and tests were performed to provide assurance that the functional intent of the system is achieved during the manufacture of the components and the construction of the system:

a. All ducting and filter assemblies were given a pneumatic pressure test and leak test.

b. Each filter assembly received a filter performance test. Each HEPA and charcoal filter bank was tested in place to verify performance.

c. Dimensional tolerances on filter assemblies and frame assemblies were checked to assure that suitable gasket compression is uniformly achieved on the filter sealing faces. Periodic tests of the filter assemblies are made in accordance with Technical Specifications.

d. The manufacturer has demonstrated, by testing charcoal essentially identical in composition to that furnished with the filter assembly, that the charcoal bed is capable of removing 99.5% of molecular iodine - 131 in the presence of a gaseous concentration of 50 mg per m^3 of non-radioactive molecular iodine or 95.0% of methyl-iodide - 131 in the presence of a gaseous concentration of 5 mg per m^3 of non-radioactive methyl iodide. This performance level was maintained until the amount of non-radioactive I_2 which reached the test unit was equivalent to 100 gm in the full-scale system. Following this loading, air at 70% RH and 150°F was drawn through the test unit at its rated flow for two hours. The integrated I_2-131 removal efficiency for the test unit, including both iodine feed and elution periods, was no less than 99.0% for the molecular iodine - 131 and no less than 95.0% of the methyl iodide - 131. The I_2-131 and CH_3-I-131 activity during feed periods was between 10 and 100 millicurie/gm of non-radioactive I_2 fed.

e. Each charcoal bed filter was assembled at the manufacturer's shop and given a Flow Resistance Test and a Leak Test.

f. A sample of each lot of carbon was tested for iodine collection capability in configuration and at a gas flowrate and conditions comparable with the filter design. The iodine concentration upstream of the bed was 1000 mg/m^3 and the penetration did not exceed 0.01% for a period of 850 seconds.

g. High-efficiency particulate absolute filters were random tested to demonstrate the filter's ability to withstand a pressure differential of 10 inches of water without loss of filtering efficiency.

h. HEPA filters of identical design to those in the filter assemblies were subjected to a rough handling test (3/4 inch amplitude at 200 cycles/min.) following which the filter demonstrated no loss of filtering efficiency.

5.3.2.4.2 Surveillance Tests

Periodic tests of the filter assemblies are made in accordance with Technical Specifications Section 4.4. The time required to test a filter should not exceed 30 minutes, even if it is necessary to probe and retest in the event of excessive leakage; and normal test periods would be much shorter. For this conservative duration of test exposure and an injection rate of 1 gm/min of Dioctyl Phthalate (DOP), a total deposition of approximately 30 gm of DOP would be deposited during each test of an upstream HEPA filter bank. With annual tests over a five year period, the total DOP deposited on the upstream HEPA filter bank would be on the order of 150 gm.

Since the vapor pressure of DOP is extremely low (2×10^{-4} mm Hg at 170°F), the maximum anticipated post-accident temperature increase in the Shield Building (from 70 to 170°F) could cause negligible release less than 1% of DOP from the HEPA filters. Furthermore, no more than 30 percent of any DOP release that might occur from an upstream HEPA filter could be deposited and retained on the charcoal filter, because of its extremely poor efficiency of adsorption for DOP. If it is assumed that this DOP were all released as a result of post-accident temperature increase and were then caught and retained on the charcoal, the loading on the bed would be less than 1.65×10^{-6} grams of DOP per gram of charcoal. This is a quantity so minute as to be immeasurable - and inconsequential with regard to effectiveness of iodine removal, even if it were arbitrarily assumed in addition that deposition was confined to the outer one percent thickness of each layer of charcoal. It is apparent from this result, and from consideration of the conservatism of the estimate, that the very small amount of DOP contaminant cannot have any effect on the ignition temperature of the charcoal.

The same conservative estimate may be extended to consideration of potential production of methyl iodide by assuming that the total amount of DOP that was presumed to collect on the charcoal reacts completely with the iodine or iodide that is also present on the charcoal, either as initial impregnant or as collected radioactive containment leakage. The effect is again inconsequential because of the minute amount of DOP and the relatively large amount of impregnated iodine. DOP is 74 percent carbon by weight, so the total carbon available for reaction with iodine to form CH_3I is 1.65×10^{-6} or 1.22×10^{-6} grams of carbon per gram of charcoal. The carbon could react with the iodine in the ratio of atomic weights, so the amount of iodine that might be combined organically is:

$$1.22 \times 10^{-6} \frac{\text{gm carbon}}{\text{gm charcoal}} \times \frac{\text{127 gm iodine}}{\text{12 gm carbon}} = 1.29 \times 10^{-5} \frac{\text{gm iodine}}{\text{gm charcoal}}$$

The abundance of the initial iodine impregnant in the charcoal is five percent by weight. Thus the maximum fraction of the initial iodine that might be converted to methyl iodide is:

$$\frac{1.29 \times 10^{-5}}{.05} = .026\%$$

The fraction applies identically to the initial iodine impregnant or to the several grams of radioactive iodine that could be deposited on the charcoal during the post-accident period. Under the most adverse conditions, and for consistently conservative assumptions, no more than .026 percent of the iodine might be converted to organic form by the effects of DOP residue.

The charcoal filter material will have a retention efficiency of more than 95% for methyl iodide; thus, the fraction of radioactive iodine released by DOP effects will be correspondingly less, and indeed a negligible fraction (1.3×10^{-4}) of the 4 percent organic iodides which are specified in Regulatory Guide 1.4 and which are presumed to escape from containment.

It is concluded that the extremely small quantities of DOP that will be used in testing of the HEPA filters will have essentially no effect on the functional capabilities of the charcoal filters.

5.3.2.4.3 System Acceptance Tests

In order to prove that the Shield Building Ventilation System (SBVS) would perform in accordance with its design criteria, acceptance tests of the system were performed as described below prior to plant startup.

5.3.2.4.3.1 Pull-down Tests

The Pull-down test was conducted for the following purposes:

a. To verify that the Shield Building Ventilation System produces a measurable vacuum in the annulus within 30 seconds after actuation of the system.

b. To verify that the recirculation valve is opened and recirculation is initiated at -2.0" W.C. annulus pressure.

c. To verify that the system performance confirms the computer prediction and

d. To determine the leakage rate of the as-built Shield Building with all its penetrations, doors, etc., installed.

The SBVS was initiated with all normal penetrations and doors, etc., installed. The test was "cold" without any attempt to simulate temperature conditions. The two fans (large recirculation and small exhaust) initially discharged to the containment system vent. When the annulus reached the recirculation mode set-point pressure, the recirculation damper opened. Continuous flow (at the containment system vent) and annulus pressure measurements were made.

a. The annulus pressure measurement verified that measurable vacuum is achieved in less than 30 seconds.

b. Signals from the differential pressure switches and annulus pressure measurement verified that recirculation is initiated at -2.0" W.C.

c. The pressure and flow measurement as a function of time were compared to curves generated by the computer code for similar conditions. The appropriate curves were developed by the computer using the measured leakage rate of the Shield Building (see (d) below). The comparison verifies that the system performs according to the computer prediction.

d. The steady state flow and pressure measurement determined the Shield Building leakage rate. This measurement provided the bases for Figure 5.3-2.

5.3.2.4.3.2 Other Tests

Additional tests on the Shield Building Ventilation System (SBVS) were performed as follows:

5.3.2.4.3.2.1 For Negative Pressure at Various Locations

This test demonstrated system ability to attain a negative pressure at several representative locations in the annulus.

With Train A first, and then Train B, the SBVS was operated until equilibrium conditions existed. Then accurate differential pressure measurements between the annulus and the Category I Ventilation Zone of the Auxiliary Building were made at the following locations within the annulus.

	UNIT 1		UNIT 2	
	Elevation	Aximuth	Elevation	Aximuth
(a)	708′ -3″	330° containment vessel	Same as on Unit 1	Equivalent azimuths to reflect mirror image on Unit 1
(b)	716′ -0″	45°		
(c)	724′ -0″	270°		
(d)	734′ -0″	330°		
(e)	755′ -0″	280°		
(f)	766′ -0″	330°		

Accurate barometric readings were taken at the reference location within the Category I zone and outside the Auxiliary Building. Temperatures were obtained to correct these barometric readings.

In addition to demonstrating the occurrence of sufficient vacuum at all locations of measurement, the results of these tests indicated the magnitude of the actual pressure differences associated with air movement.

5.3.2.4.3.2.2 Tests on Additional Resistance and Inleakage

Additional tests were also performed to evaluate the system response

a. to added flow resistance at different locations in the duct work and

b. to additional inleakage through openings of known sizes.

5.3.2.4.3.2.3 Partially Blocked Discharge Vent

This type of test was performed for both Train A and Train B of the SBVS. The exit from the vent was partially blocked with temporary materials, such that part of the free discharge area is blocked off. Then the time required to draw the annulus down to a sufficiently negative pressure was determined and compared to the computer code for similar resistances.

5.3.2.4.3.2.4 Partially Closed Recirculation Damper

With Train A of the SBVS operating at equilibrium, additional flow restriction was added to the recirculation duct by partial closing of the recirculation damper. Test data was obtained until the check damper to the vent opens. A partially closed recirculation control valve has the effect of causing the annulus to become more negative than in normal operation since in that case, more air would be diverted to the exhaust vent.

5.3.2.4.3.2.5 Additional Flow Resistance

A manometer was connected across the filter section, and a resistance that simulates dirty filters and/or additional flow resistance was placed in the section. Annulus pressure vs time measurements were recorded. Resistance was added until an adequate negative pressure was no longer obtained. The information was then compared to the computer code predictions.

5.3.2.4.3.2.6 Additional Air Inleakage

As a part of the cold pull-down testing procedure, means were provided to introduce rates of air flow into the Shield Building Ventilation System through additional openings of known sizes, to simulate additional in-leakage. This portion of the test started with simulating first a low air in-flow and then was followed by a number of increments of increasing air in-flow. Vent flow was determined at each incremental increase in air in-flow. In this way, the maximum additional air flow at which the annulus can no longer be maintained at a sufficiently negative pressure was determined.

This test was conducted with both large and small fans running, first with Train A only, then Train B.

5.3.2.4.3.3 Updating Computer Code with Test Results

The results from the various tests provided the data that established the actual pressure vs leakage curve as shown in Figure 5.3-2. The expression was then used to update the computer code. When other measured system parameters differed from those initially used in the code, the code was modified to utilize the actual values.

The SBVS computer code describes an exact analytical method to predict pressure change and net air discharge between every two successive statuses in the shield building annulus. The code predicts pressure and flow transients and, therefore, exact analytical solutions.

5.5 REFERENCES

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Replace all old Section 5 Tables with the new Section 5 Tables

TABLE 5.2-1 - (PART A)
UNIT 1 CONTAINMENT VESSEL PENETRATIONS

Penet No.	Description	Penet Class (Note 2)	Penet Group (Note 3)	Nominal Line Size	Approx. Line Size @ Penetration	Valve Number	Inside/ Outside	Valve Type	Normal Op. Position (Note 1)	Post-LOCA Position	Type of Test	USAR Section
1	PRT Sample to Gas Analyzer	2	I	3/8"	3/8"	CV31318 CV31319	Outside Outside	RSV-T RSV-T	Open Open	Closed Closed	C	5.2.3.1.1
2	PRT Nitrogen Supply	3	III	3/4"	3/4"	RC-5-1 CV31221	Inside Outside	Check RSV-T	Installed Closed	Installed Closed	C	
3A 3B	Spare Inst. (Yellow Containment Press)		 II	3/8" 3/8"	3/8" 3/8"	- - - Sealed	Inside Outside	- - - - - -	Capped - - -	Capped - - -	 A (Note 5)	
4	Primary System Vent Header	2	I	1"	2"	CV31434 CV31435	Outside Outside	RSV-T RSV-T	Open Open	Closed Closed	C	5.2.3.1.1
5	RC Drain Tank Pump Discharge	2	I	3"	3"	CV31436 CV31437	Outside Outside	RSV-T RSV-T	Open Open	Closed Closed	C	5.2.3.1.1
6A	Main Steam Header	4A	V	30"	30"	CV31098 MV32045 MV32016 Safety's/ PORV's Closed System	Outside Outside Outside Outside Inside	RSV-MS RSV RSV Relief Valves	Open Closed Open Closed	Closed Closed Open Closed	A (Note 6)	
6B	Main Steam Header	4A	V	30"	30"	CV31099 MV32047 MV32017 Safety's/ PORV's Closed System	Outside Outside Outside Outside Inside	RSV-MS RSV RSV Relief Valves	Open Closed Open Closed	Closed Closed Open Closed	A (Note 6)	
7A	Main Feedwater Headers	4A	V	16"	16"	MV32023 Closed System	Outside Inside	RSV-T	Open	Closed	A (Note 6)	
7B	Main Feedwater Headers	4A	V	16"	16"	MV32024 Closed System	Outside Inside	RSV-T	Open	Closed	A (Note 6)	
8A	Steam Generator Blowdown	4A	V	2"	2"	MV32044 Closed System	Outside Inside	RSV-T	Open	Closed	A (Note 6)	

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TABLE 5.2-1 - (PART A)
UNIT 1 CONTAINMENT VESSEL PENETRATIONS

Penet No.	Description	Penet Class (Note 2)	Penet Group (Note 3)	Nominal Line Size	Approx. Line Size @ Penetration	Valve Number	Inside/ Outside	Valve Type	Normal Op Position (Note 1)	Post-LOCA Position	Type of Test	USAR Section
8B	Steam Generator Blowdown	4A	V	2"	2"	MV32058 Closed System	Outside Inside	RSV-T	Open	Closed	A (Note 6)	
9	RHR Loop Out	6	VII	10"	10"	MV32165 MV32231 RH-8-1	Inside Inside Inside	RSV RSV Relief	Closed Closed Installed	Closed Closed Installed	A (Note 7)	
10	RHR Loop In	6	VII	10"	10"	MV32066 MV32065 RH-6-1 MV32234	Inside Inside Inside Inside	RSV RSV Check RSV	VCBO Open Installed VCBO	Closed Open Installed Closed	A (Note 7)	
11	CVCS Letdown Line	1	II	2"	2"	CV31325 CV31326 CV31327 CV31339	Inside Inside Inside Outside	RSV-T RSV-T RSV-T RSV-T	Open Open Open Open	Closed Closed Closed Closed	C	
12	CVCS Charging Line	3	II	2"	2"	VC-8-1 Closed System	Inside Outside	Check - - -	Installed - - -	Installed - - -	C	10.2.3.3.4
13A	RCP Seal Water Supply	3	II	2"	2"	VC-8-5 Closed System	Inside Outside	Check - - -	Installed - - -	Installed - - -	C	10.2.3.3.4
13B	RCP Seal Water Supply	3	II	2"	2"	VC-8-4 Closed System	Inside Outside	Check - - -	Installed - - -	Installed - - -	C	10.2.3.3.4
14	RCP Seal Water Return	1	II	3"	3"	MV32199 MV32166	Inside Outside	RSV-T RSV-T	Open Open	Closed Closed	C	
15	Pressurizer Steam Sample	1	VI	3/8"	3/8"	MV32400 MV32401	Inside Outside	RSV-T RSV-T	Closed Closed	Closed Closed	C	
16	Pressurizer Liquid Sample	1	VI	3/8"	3/8"	MV32402 MV32403	Inside Outside	RSV-T RSV-T	Closed Closed	Closed Closed	C	
17	RCS Loop B Sample	1	VI	3/8"	3/8"	MV32404 MV32405	Inside Outside	RSV-T RSV-T	Closed Closed	Closed Closed	C	
18	Fuel Transfer Tube	5	V	20"	20"	Blind Flange	Inside	- - -	Installed	Installed	B	

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TABLE 5.2-1 - (PART A)
UNIT 1 CONTAINMENT VESSEL PENETRATIONS

Penet No.	Description	Penet Class (Note 2)	Penet Group (Note 3)	Nominal Line Size	Approx. Line Size @ Penetration	Valve Number	Inside/ Outside	Valve Type	Normal Op Position (Note 1)	Post-LOCA Position	Type of Test	USAR Section	
19	Service Air	7	IV	2"	2"	Blind Flange	Inside	- - -	Installed	Installed	B		
20	Instrument Air	3	III	2"	2"	CV31741 CV31740	Inside Outside	RSV-MSP RSV-MSP	Open Open	Closed Closed	C		
21	RCDT to Gas Analyzer	2	I	3/8"	3/8"	CV31545 CV31546	Outside Outside	RSV-T RSV-T	Open Open	Closed Closed	C	5.2.3.1.1	
22	Containment Air Sample In	6	VI	1"	2"	CV31092 CV31022	Inside Outside	RSV-T RSV-T	Open Open	Closed Closed	C		
23	Containment Air Sample Out	6	VI	1"	2"	CV31019 CV31750	Inside Outside	RSV-T RSV-T	Open Open	Closed Closed	C		99177
25A	Containment Vent & Purge Exhaust Duct	5	VI	36"	36"	Blind Flange	Inside	- - -	Installed	Installed	B	5.2.2.3.3	
25B	Containment Vent & Purge Supply Duct	5	VI	36"	36"	Blind Flange	Inside	- - -	Installed	Installed	B	5.2.2.3.3	
26	Containment Sump "A" Pump Discharge	2	I	3"	3"	CV31438 CV31439	Outside Outside	RSV-T RSV-T	Open Open	Closed Closed	C	5.2.3.1.1	
27A1 27A2	Steam Generator Blowdown Sample	4A 4A	V V	3/8" 3/8"	3/8" 3/8"	CV31402 CV31403 Closed System	Outside Outside Inside	RSV-T RSV-T	Open Open	Closed Closed	A (Note 6) A (Note 6)		
27B	Fire Protection	7	IV	4"	4"	Blind Flange	Inside	- - -	Installed	Installed	B		
27C1 27C2	Containment Pressure Test Panel	5	VI	1" 1"	1" 1"	Blind Flange	Outside	- - -	Installed	Installed	B		
28A	Vessel Injection Safety Injection	6	VII	3"	4"	SI-16-6 SI-16-7 Shared Isol. w/Penet 35	Inside Inside	Check Check	Installed Installed	Installed Installed	A (Note 7)		99129
28B	Cold Leg Safety Injection	6	VII	3"	4"	SI-16-5 SI-16-4 CV31442 CV31445	Inside Inside Inside Inside	Check Check RSV RSV	Installed Installed Closed Closed	Installed Installed Closed Closed	A (Note 7)		99177

TABLE 5.2-1 - (PART A)
UNIT 1 CONTAINMENT VESSEL PENETRATIONS

Penet No.	Description	Penet Class (Note 2)	Penet Group (Note 3)	Nominal Line Size	Approx. Line Size @ Penetration	Valve Number	Inside/ Outside	Valve Type	Normal Op Position (Note 1)	Post-LOCA Position	Type of Test	USAR Section
29A	Containment Spray	6	VII	6"	6"	CS-19 MV32105 CS-12	Outside Outside Outside	Check RSV Manual	Installed Closed Locked Closed	Installed Open Locked Closed	C	
29B	Containment Spray	6	VII	6"	6"	CS-18 MV32103 CS-11	Outside Outside Outside	Check RSV Manual	Installed Closed Locked Closed	Installed Open Locked Closed	C	
30A	RHR Suction From Sump B	6	VII	12"	12"	MV32075 Closed System	Inside Outside	RSV - - -	Closed - - -	Open - - -	A (Note 7)	
30B	RHR Suction From Sump B	6	VII	12"	12"	MV32076 Closed System	Inside Outside	RSV - - -	Closed - - -	Open - - -	A (Note 7)	
31	Nitrogen to Accumulator	3	III	1"	2"	CV31441 CV31444 CV31242 CV31440	Inside Inside Inside Outside	RSV RSV RSV RSV-T	Closed Closed Closed Closed	Closed Closed Closed Closed	C	
32A	Component Cooling to 11 RCP	6	V	4"	4"	MV32089 Closed System	Outside Inside	RSV - - -	VOBO - - -	Open - - -	A (Note 7)	
32B	Component Cooling to 12 RCP	6	V	4"	4"	MV32091 Closed System	Outside Inside	RSV - - -	VOBO - - -	Open - - -	A (Note 7)	
33A	Component Cooling from 11 RCP	6	V	4"	4"	MV32090 Closed System	Outside Inside	RSV - - -	VOBO - - -	Open - - -	A (Note 7)	
33B	Component Cooling from 12 RCP	6	V	4"	4"	MV32092 Closed System	Outside Inside	RSV - - -	VOBO - - -	Open - - -	A (Note 7)	
34	Electrical Penetrations		VIII								B	

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TABLE 5.2-1 - (PART A)
UNIT 1 CONTAINMENT VESSEL PENETRATIONS

Penet No.	Description	Penet Class (Note 2)	Penet Group (Note 3)	Nominal Line Size	Approx. Line Size @ Penetration	Valve Number	Inside/ Outside	Valve Type	Normal Op Position (Note 1)	Post-LOCA Position	Type of Test	USAR Section	
35	SI and Accumulator Test Line	1	VI	3/4"	3/4"	CV31447 CV31448 CV31449 CV31450 SI-20-16 Shared Isol. w/Penet 28A	Inside Inside Inside Inside Outside	RSV RSV RSV RSV Manual	Closed Closed Closed Closed Locked Closed	Closed Closed Closed Closed Locked Closed	A (Note 7)		99177 99129
36D	Inst. (Red Containment Press)		II	3/8"	3/8"	Sealed	Outside	- - -	- - -	- - -	A (Note 5)		
37A	Cooling Water to 13 CFCU	6	V	8"	8"	Closed System MV32378	Inside Outside	 RSV	 Open	 Open	A (Note 7)	5.2.3.3	
37B	Cooling Water to 11 CFCU	6	V	8"	8"	Closed System MV32377	Inside Outside	 RSV	 Open	 Open	A (Note 7)	5.2.3.3	
37C	Cooling Water to 12 CFCU	6	V	8"	8"	Closed System MV32379	Inside Outside	 RSV	 Open	 Open	A (Note 7)	5.2.3.3	
37D	Cooling Water to 14 CFCU	6	V	8"	8"	Closed System MV32380	Inside Outside	 RSV	 Open	 Open	A (Note 7)	5.2.3.3	99177
38A	Cooling Water from 13 CFCU	6	V	8"	8"	MV32138 MV32139	Inside Outside	RSV RSV	Open Open	Open Open	A (Note 7)	5.2.3.3	
38B	Cooling Water from 11 CFCU	6	V	8"	8"	MV32132 MV32133	Inside Outside	RSV RSV	Open Open	Open Open	A (Note 7)	5.2.3.3	
38C	Cooling Water from 14 CFCU	6	V	8"	8"	MV32141 MV32142	Inside Outside	RSV RSV	Open Open	Open Open	A (Note 7)	5.2.3.3	
38D	Cooling Water from 12 CFCU	6	V	8"	8"	MV32135 MV32136	Inside Outside	RSV RSV	Open Open	Open Open	A (Note 7)	5.2.3.3	
39	Comp Cooling to Excess Letdown Heat Exchanger	4	V	3"	2"	Closed System MV32095	Inside Outside	 RSV-T	 Open	 Closed	A (Note 7)		

TABLE 5.2-1 - (PART A)
UNIT 1 CONTAINMENT VESSEL PENETRATIONS

Penet No.	Description	Penet Class (Note 2)	Penet Group (Note 3)	Nominal Line Size	Approx. Line Size @ Penetration	Valve Number	Inside/ Outside	Valve Type	Normal Op Position (Note 1)	Post-LOCA Position	Type of Test	USAR Section
40	Comp Cooling from Excess Letdown Heat Exchanger	4	V	3"	2"	CV31252 Closed System	Outside Inside	RSV-T - - -	Open - - -	Closed - - -	A (Note 7)	
41A	Containment Vacuum Breaker	7B	VIII	18"	18"	CV31621 CV31624	Outside Outside	RSV-T VRV	Open Installed	Closed Installed	C	Table 5.2-7
41B	Containment Vacuum Breaker	7B	VIII	18"	18"	CV31622 CV31625	Outside Outside	RSV-T VRV	Open Installed	Closed Installed	C	Table 5.2-7
42A	Post LOCA H2 Control Air Supply and Vent	7 5	IV VI	2" 2"	2" 2"	HC-2-2 MV32276 MV32273 CV31927 CV31929	Inside Outside Inside Outside Outside	Check RSV RSV RSV RSV	Installed Closed Closed Closed Closed	Installed Closed Closed Closed Closed	C C	
42B	In-Service Purge Supply	7	VIII	18"	14"	Blind Flange CV31634 CV31633	Outside Inside Outside	- - - RSV-TT RSV-TT	Installed Closed Closed	Installed Closed Closed	B (Note 4)	5.2.2.3.3
42C	Heating Steam Supply	7	IV	4"	4"	Blind Flange	Inside	- - -	Installed	Installed	B	
42D	RVLIS Instrumentation		II	6 x 3/16"	6 x 3/16"	Sealed	Outside	- - -	- - -	- - -	A (Note 5)	
42F1	Heating Steam Condensate Return	7	IV	2"	2"	Blind Flange	Inside	- - -	Installed	Installed	B	
42F2	Heating Steam Return Vent	7	IV	2"	2"	Blind Flange	Inside	- - -	Installed	Installed	B	
43A	In-Service Purge Exhaust	7	VIII	18"	18"	Blind Flange CV31310 CV31311	Outside Outside Inside	- - - RSV-TT RSV-TT	Installed Closed Closed	Installed Closed Closed	B (Note 4)	5.2.2.3.3
44	Containment Vessel Pressurization (ILRT)	5	VI	6"	10"	Blind Flange	Inside	- - -	Installed	Installed	B	
45	Rx M/U Water to PRT	3	VI	2"	2"	RC-3-1 CV31321	Inside Outside	Check RSV-T	Installed Open	Installed Closed	C	
46A	Auxiliary Feedwater	6	VII	3"	4"	AF-16-2	Inside	Check	Installed	Installed	A (Note 6)	

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TABLE 5.2-1 - (PART A)
UNIT 1 CONTAINMENT VESSEL PENETRATIONS

Penet No.	Description	Penet Class (Note 2)	Penet Group (Note 3)	Nominal Line Size	Approx. Line Size @ Penetration	Valve Number	Inside/ Outside	Valve Type	Normal Op Position (Note 1)	Post-LOCA Position	Type of Test	USAR Section
46B	Auxiliary Feedwater	6	VII	3"	4"	AF-16-1	Inside	Check	Installed	Installed	A (Note 6)	
47	Electrical Penetrations		VIII								B	
48	Low Head Safety Injection to Reactor Vessel	6	VII	6"	6"	SI-26-1 MV32064	Inside Inside	Relief RSV	Installed Open	Installed Open	A (Note 7)	
49A	Inst. (Blue Containment Press)		II	3/8"	3/8"	Sealed	Outside	- - -	- - -	- - -	A (Note 5)	
49B	Demineralized Water	7	IV	2"	4"	Blind Flange	Inside	- - -	Installed	Installed	B	
50	Post LOCA H2 Control Air Supply and Vent Lines	7 5	IV VI	2" 2"	2" 2"	HC-2-1 MV32274 MV32271 CV31925 CV31923	Inside Outside Inside Outside Outside	Check RSV RSV RSV RSV	Installed Closed Closed Closed Closed	Installed Closed Closed Closed Closed	C C	
	Inst. (White Containment Press)		II	3/8"	3/8"	Sealed	Outside				A (Note 5)	
	Equipment Hatch		VIII								B	
	Personnel Airlock		VIII								B	
	Maintenance Airlock		VIII								B	

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TABLE 5.2-1 - (PART B)
UNIT 2 CONTAINMENT VESSEL PENETRATIONS

Penet No.	Description	Penet Class (Note 2)	Penet Group (Note 3)	Nominal Line Size	Approx. Line Size @ Penetration	Valve Number	Inside/ Outside	Valve Type	Normal Op Position (Note 1)	Post-LOCA Position	Type of Test	USAR Section
1	PRT Sample to Gas Analyzer	2	I	3/8"	3/8"	CV31344 CV31345	Outside Outside	RSV-T RSV-T	Open Open	Closed Closed	C	5.2.3.1.1
2	PRT Nitrogen Supply	3	III	3/4"	3/4"	2RC-5-1 CV31209	Inside Outside	Check RSV-T	Installed Closed	Installed Closed	C	
3A 3B	Spare Inst. (Yellow Containment Press)		 II	3/8" 3/8"	3/8" 3/8"	- - - Sealed	Inside Outside	- - - - - -	Capped - - -	Capped - - -	 A (Note 5)	
4	Primary System Vent Header	2	I	1"	2"	CV31733 CV31734	Outside Outside	RSV-T RSV-T	Open Open	Closed Closed	C	5.2.3.1.1
5	RC Drain Tank Pump Discharge	2	I	3"	3"	CV31735 CV31736	Outside Outside	RSV-T RSV-T	Open Open	Closed Closed	C	5.2.3.1.1
6C	Main Steam Header	4A	V	30"	30"	CV31116 MV32048 MV32019 Safety's/ PORV's Closed System	Outside Outside Outside Outside Inside	RSV-MS RSV RSV Relief Valves	Open Closed Open Closed	Closed Closed Open Closed	A (Note 6)	
6D	Main Steam Header	4A	V	30"	30"	CV31117 MV32050 MV32020 Safety's/ PORV's Closed System	Outside Outside Outside Outside Inside	RSV-MS RSV RSV Relief Valves	Open Closed Open Closed	Closed Closed Open Closed	A (Note 6)	
7C	Main Feedwater Headers	4A	V	16"	16"	MV32028 Closed System	Outside Inside	RSV-T	Open	Closed	A (Note 6)	
7D	Main Feedwater Headers	4A	V	16"	16"	MV32029 Closed System	Outside Inside	RSV-T	Open	Closed	A (Note 6)	
8C	Steam Generator Blowdown	4A	V	2"	2"	MV32051 Closed System	Outside Inside	RSV-T	Open	Closed	A (Note 6)	

TABLE 5.2-1 - (PART B)
UNIT 2 CONTAINMENT VESSEL PENETRATIONS

Penet No.	Description	Penet Class (Note 2)	Penet Group (Note 3)	Nominal Line Size	Approx. Line Size @ Penetration	Valve Number	Inside/ Outside	Valve Type	Normal Op Position (Note 1)	Post-LOCA Position	Type of Test	USAR Section
8D	Steam Generator Blowdown	4A	V	2"	2"	MV32059 Closed System	Outside Inside	RSV-T	Open	Closed	A (Note 6)	
9	RHR Loop Out	6	VII	10"	10"	MV32193 MV32233 2RH-8-1	Inside Inside Inside	RSV RSV Relief	Closed Closed Installed	Closed Closed Installed	A (Note 7)	
10	RHR Loop In	6	VII	10"	10"	MV32167 MV32169 2RH-6-1 MV32235	Inside Inside Inside Inside	RSV RSV Check RSV	Open VCBO Installed VCBO	Open Closed Installed Closed	A (Note 7)	
11	CVCS Letdown Line	1	II	2"	2"	CV31347 CV31348 CV31349 CV31430	Inside Inside Inside Outside	RSV-T RSV-T RSV-T RSV-T	Open Open Open Open	Closed Closed Closed Closed	C	
12	CVCS Charging Line	3	II	2"	2"	2VC-8-1 Closed System	Inside Outside	Check - - -	Installed - - -	Installed - - -	C	10.2.3.3.4
13A	RCP Seal Water Supply	3	II	2"	2"	2VC-8-5 Closed System	Inside Outside	Check - - -	Installed - - -	Installed - - -	C	10.2.3.3.4
13B	RCP Seal Water Supply	3	II	2"	2"	2VC-8-4 Closed System	Inside Outside	Check - - -	Installed - - -	Installed - - -	C	10.2.3.3.4
14	RCP Seal Water Return	1	II	3"	6"	MV32210 MV32194	Inside Outside	RSV-T RSV-T	Open Open	Closed Closed	C	
15	Pressurizer Steam Space Sample	1	VI	3/8"	3/8"	MV32406 MV32407	Inside Outside	RSV-T RSV-T	Closed Closed	Closed Closed	C	
16	Pressurizer Liquid Sample	1	VI	3/8"	3/8"	MV32408 MV32409	Inside Outside	RSV-T RSV-T	Closed Closed	Closed Closed	C	
17	RCS Loop B Sample	1	VI	3/8"	3/8"	MV32410 MV32411	Inside Outside	RSV-T RSV-T	Closed Closed	Closed Closed	C	
18	Fuel Transfer Tube	5	V	20"	20"	Blind Flange	Inside	- - -	Installed	Installed	B	

TABLE 5.2-1 - (PART B)
UNIT 2 CONTAINMENT VESSEL PENETRATIONS

Penet No.	Description	Penet Class (Note 2)	Penet Group (Note 3)	Nominal Line Size	Approx. Line Size @ Penetration	Valve Number	Inside/ Outside	Valve Type	Normal Op Position (Note 1)	Post-LOCA Position	Type of Test	USAR Section
19	Service Air	7	IV	2"	2"	Blind Flange	Inside	- - -	Installed	Installed	B	
20	Instrument Air	3	III	2"	2"	CV31743 CV31742	Inside Outside	RSV-MSP RSV-MSP	Open Open	Closed Closed	C	
21	RCDT to Gas Analyzer	2	I	3/8"	3/8"	CV31732 CV31731	Outside Outside	RSV-T RSV-T	Open Open	Closed Closed	C	5.2.3.1.1
22	Containment Air Sample In	6	VI	1"	2"	CV31129 CV31644	Inside Outside	RSV-T RSV-T	Open Open	Closed Closed	C	
23	Containment Air Sample Out	6	VI	1"	2"	CV31643 CV31642	Inside Outside	RSV-T RSV-T	Open Open	Closed Closed	C	
25A	Containment Purge Exhaust	5	VI	36"	36"	Blind Flange	Inside	- - -	Installed	Installed	B	5.2.2.3.3
25B	Containment Purge Supply	5	VI	36"	36"	Blind Flange	Inside	- - -	Installed	Installed	B	5.2.2.3.3
26	Containment Sump "A" Pump Discharge	2	I	3"	3"	CV31620 CV31619	Outside Outside	RSV-T RSV-T	Open Open	Closed Closed	C	5.2.3.1.1
27A1 27A2	Steam Generator Blowdown Sample	4A 4A	V V	3/8" 3/8"	3/8" 3/8"	CV31412 CV31413 Closed System	Outside Outside Inside	RSV-T RSV-T	Open Open	Closed Closed	A (Note 6) A (Note 6)	
27C1 27C2	Containment Pressure Test Panel	5	VI	1" 1"	1" 1"	Blind Flange	Outside	- - -	Installed	Installed	B	
28A	Vessel Injection Safety Injection	6	VII	3"	4"	2SI-16-5 2SI-16-7 Shared Isol. w/Penet 35	Inside Inside	Check Check	Installed Installed	Installed Installed	A (Note 7)	
28B	Cold Leg Safety Injection	6	VII	3"	4"	2SI-16-6 2SI-16-4 CV31517 CV31518	Inside Inside Inside Inside	Check Check RSV RSV	Installed Installed Closed Closed	Installed Installed Closed Closed	A (Note 7)	

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TABLE 5.2-1 - (PART B)
UNIT 2 CONTAINMENT VESSEL PENETRATIONS

Penet No.	Description	Penet Class (Note 2)	Penet Group (Note 3)	Nominal Line Size	Approx. Line Size @ Penetration	Valve Number	Inside/ Outside	Valve Type	Normal Op Position (Note 1)	Post-LOCA Position	Type of Test	USAR Section
29A	Containment Spray	6	VII	6"	6"	CS-49 MV32114 CS-42	Outside Outside Outside	Check RSV Manual	Installed Closed Locked Closed	Installed Open Locked Closed	C	
29B	Containment Spray	6	VII	6"	6"	CS-48 MV32116 CS-41	Outside Outside Outside	Check RSV Manual	Installed Closed Locked Closed	Installed Open Locked Closed	C	
30A	RHR Suction from Sump B	6	VII	12"	12"	MV32178 Closed System	Inside Outside	RSV - - -	Closed - - -	Open - - -	A (Note 7)	
30B	RHR Suction from Sump B	6	VII	12"	12"	MV32179 Closed System	Inside Outside	RSV - - -	Closed - - -	Open - - -	A (Note 7)	
31	Nitrogen Supply to Accumulator	3	III	1"	2"	CV31244 CV31511 CV31512 CV31554	Inside Inside Inside Outside	RSV RSV RSV RSV-T	Closed Closed Closed Closed	Closed Closed Closed Closed	C	
32A	Component Cooling to 21 RCP	6	V	4"	4"	MV32124 Closed System	Outside Inside	RSV - - -	VOBO - - -	Open - - -	A (Note 7)	
32B	Component Cooling to 22 RCP	6	V	4"	4"	MV32126 Closed System	Outside Inside	RSV - - -	VOBO - - -	Open - - -	A (Note 7)	
33A	Component Cooling Return from 21 RCP	6	V	4"	4"	MV32125 Closed System	Outside Inside	RSV - - -	VOBO - - -	Open - - -	A (Note 7)	
33B	Component Cooling Return from 22 RCP	6	V	4"	4"	MV32127 Closed System	Outside Inside	RSV - - -	VOBO - - -	Open - - -	A (Note 7)	
34	Electrical Penetrations		VIII								B	

TABLE 5.2-1 - (PART B)
UNIT 2 CONTAINMENT VESSEL PENETRATIONS

Penet No.	Description	Penet Class (Note 2)	Penet Group (Note 3)	Nominal Line Size	Approx. Line Size @ Penetration	Valve Number	Inside/ Outside	Valve Type	Normal Op Position (Note 1)	Post-LOCA Position	Type of Test	USAR Section
35	SI and Accumulator Test Line	1	VI	3/4"	3/4"	CV31462 CV31461 CV31459 CV31460 2SI-20-16 Shared Isol. w/Penet 28A	Inside Inside Inside Inside Outside	RSV RSV RSV RSV Manual	Closed Closed Closed Closed Locked Closed	Closed Closed Closed Closed Locked Closed	A (Note 7)	
37A	Cooling Water to 21 CFCU	6	V	8"	8"	MV32386 Closed System	Outside Inside	RSV - - -	Open - - -	Open - - -	A (Note 7)	5.2.3.3
37B	Cooling Water to 22 CFCU	6	V	8"	8"	MV32387 Closed System	Outside Inside	RSV - - -	Open - - -	Open - - -	A (Note 7)	5.2.3.3
37C	Cooling Water to 23 CFCU	6	V	8"	8"	MV32388 Closed System	Outside Inside	RSV - - -	Open - - -	Open - - -	A (Note 7)	5.2.3.3
37D	Cooling Water to 24 CFCU	6	V	8"	8"	MV32389 Closed System	Outside Inside	RSV - - -	Open - - -	Open - - -	A (Note 7)	5.2.3.3
38A	Cooling Water from 21 CFCU	6	V	8"	8"	MV32147 MV32148	Inside Outside	RSV RSV	Open Open	Open Open	A (Note 7)	5.2.3.3
38B	Cooling Water from 22 CFCU	6	V	8"	8"	MV32150 MV32151	Inside Outside	RSV RSV	Open Open	Open Open	A (Note 7)	5.2.3.3
38C	Cooling Water from 23 CFCU	6	V	8"	8"	MV32153 MV32154	Inside Outside	RSV RSV	Open Open	Open Open	A (Note 7)	5.2.3.3
38D	Cooling Water from 24 CFCU	6	V	8"	8"	MV32156 MV32157	Inside Outside	RSV RSV	Open Open	Open Open	A (Note 7)	5.2.3.3
39	Comp Cooling Supply to Excess Letdown Heat Exchanger	4	V	3"	3"	MV32130 Closed System	Outside Inside	RSV-T - - -	Open - - -	Closed - - -	A (Note 7)	
40	Comp Cooling Return from Excess Letdown Heat Exchanger	4	V	3"	3"	CV31253 Closed System	Outside Inside	RSV-T - - -	Open - - -	Closed - - -	A (Note 7)	

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TABLE 5.2-1 - (PART B)
UNIT 2 CONTAINMENT VESSEL PENETRATIONS

Penet No.	Description	Penet Class (Note 2)	Penet Group (Note 3)	Nominal Line Size	Approx. Line Size @ Penetration	Valve Number	Inside/ Outside	Valve Type	Normal Op Position (Note 1)	Post-LOCA Position	Type of Test	USAR Section
41A	Containment Vacuum Breakers	7B	VIII	18"	18"	CV31627 CV31630	Outside Outside	RSV-T VRV	Open Installed	Closed Installed	C	Table 5.2-7
41B	Containment Vacuum Breakers	7B	VIII	18"	18"	CV31628 CV31631	Outside Outside	RSV-T VRV	Open Installed	Closed Installed	C	Table 5.2-7
42A	Post LOCA Hydrogen Control	7 5	IV VI	2" 2"	2" 2"	2HC-2-1 MV32293 MV32290 CV31924 CV31926	Inside Outside Inside Outside Outside	Check RSV RSV RSV RSV	Installed Closed Closed Closed Closed	Installed Closed Closed Closed Closed	C C	
	Inst. (White Containment Press)		II	3/8"	3/8"	Sealed	Outside				A (Note 5)	
42B	Reactor Vessel Level Instrumentation		II	6 x 3/16"	6 x 3/16"	Sealed - - -	- - -	- - -	- - -	- - -	A (Note 5)	
42E1	Heating Steam Return Vent	7	IV	2"	2"	Blind Flange	Inside	- - -	Installed	Installed	B	
42E-2	Heating Steam Condensate Return	7	IV	2"	2"	Blind Flange	Inside	- - -	Installed	Installed	B	
44	Containment Vessel Pressurization (ILRT)	5	VI	6"	10"	Blind Flange	Inside	- - -	Installed	Installed	B	
45	Reactor Makeup Water to PRT	3	VI	2"	2"	2RC-3-1 CV31342	Inside Outside	Check RSV-T	Installed Open	Installed Closed	C	
46C	Auxiliary Feedwater to 22 Steam Generator	6	VII	3"	4"	AF-16-3	Inside	Check	Installed	Installed	A (Note 6)	
46D	Auxiliary Feedwater to 21 Steam Generator	6	VII	3"	4"	AF-16-4	Inside	Check	Installed	Installed	A (Note 6)	
47	Electrical Penetrations		VIII								B	
48	Low Head Safety Injection to Reactor Vessel	6	VII	6"	6"	2SI-26-1 MV32168	Inside Inside	Relief RSV	Installed Open	Installed Open	A (Note 7)	
49A	Inst. (Blue Containment Press)		II	3/8"	3/8"	Sealed	Outside	- - -	- - -	- - -	A (Note 5)	

TABLE 5.2-1 - (PART B)
UNIT 2 CONTAINMENT VESSEL PENETRATIONS

Penet No.	Description	Penet Class (Note 2)	Penet Group (Note 3)	Nominal Line Size	Approx. Line Size @ Penetration	Valve Number	Inside/ Outside	Valve Type	Normal Op Position (Note 1)	Post-LOCA Position	Type of Test	USAR Section
50	Post LOCA Hydrogen Control	7 5	IV VI	2" 2"	2" 2"	2HC-2-2 MV32295 MV32292 CV31930 CV31928	Inside Outside Inside Outside Outside	Check RSV RSV RSV RSV	Installed Closed Closed Closed Closed	Installed Closed Closed Closed Closed	C C	
51	Fire Protection	7	IV	4"	4"	Blind Flange	Inside	- - -	Installed	Installed	B	
52	In-Service Purge Exhaust	7	VIII	18"	18"	Blind Flange CV31314 CV31315	Outside Outside Inside	- - - RSV-TT RSV-TT	Installed Closed Closed	Installed Closed Closed	B (Note 4)	5.2.2.3.3
53	In-Service Purge Supply	7	VIII	18"	14"	Blind Flange CV31636 CV31635	Outside Inside Outside	- - - RSV-TT RSV-TT	Installed Closed Closed	Installed Closed Closed	B (Note 4)	5.2.2.3.3
54	Steam Heating Supply	7	IV	4"	4"	Blind Flange	Inside	- - -	Installed	Installed	B	
55	Demineralized Water to Hose Connections	7	IV	2"	4"	Blind Flange	Inside	- - -	Installed	Installed	B	
56	Inst. (Red Containment Press)		II	3/8"	3/8"	Sealed	Outside	- - -	- - -	- - -	A (Note 5)	
	Equipment Hatch		VIII								B	
	Personnel Airlock		VIII								B	
	Maintenance Airlock		VIII								B	

TABLE 5.2-1 PART C
REACTOR CONTAINMENT VESSEL PENETRATIONS
ABBREVIATIONS AND NOTES FOR TABLE 5.2-1

VALVES

RSV	Remotely-Operated Stop Valve
RSV+RSV	Remotely-Operated Stop Valve in Series
VRV	Vacuum Relief Valve
-T	Tripped Closed on Safety Injection Signal
-TT	Tripped Closed on Safety Injection Signal or High Containment Radiation
-MS	Tripped Closed on Main Steam Isolation Signal
-MSP	Tripped Closed on Loop A Main Steam Isolation Signal or High High Containment Pressure Signal
MANUAL	Manual Valve
VOBO	Valve Open Breaker Open
VCBO	Valve Closed Breaker Open

NOTE 1:
OPERATING FUNCTION
Denotes the position of the valve during normal reactor operation. Valve positions may change due to operating procedures, isolation, etc.

NOTE 2:
PENETRATION CLASS
Number classifications are defined in Section 5.2.2. Letter designation is defined as follows:

A: The isolation system for these penetrations are subject to special consideration on leakage and testing requirements because their principal function is related to rupture of steam generator secondary side systems and not loss of coolant; for loss of coolant accident, the barrier is the steam generator tube sheet and tubes.

B: The automatically operated relief valve actuated from containment vacuum qualifies as an isolation valve because increasing pressure causes valve to stay in the closed position.

NOTE 3: Penetration groups are explained in Appendix G, Section G.2.

NOTE 4: In-Service purge supply and exhaust penetrations normally have a blind flange installed in the annulus.

NOTE 5: Instrumentation lines. No Type B or C testing required (Reference 29).

NOTE 6: Steam, Feedwater, Blowdown and SG Sample lines. Type C testing not required since valves are not relied upon to prevent containment (Reference 29).

NOTE 7: Safety Injection, RHR, Cooling Water, and Component Cooling Water system valves are not relied upon to prevent containment leakage (Reference 29).

NOTE 8: Table 5.2-1 only includes the credited isolation barriers for each penetration. Vent and drain valves and test connections which form portions of the isolation boundary are not included. These valves are identified and controlled i.a.w. PINGP Operating Procedures.

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TABLE 5.2-2 CALCULATED GUARD PIPE STRESS LEVELS

				Upset Cond.		Faulted Condition		
	Bending Due To Thermal	Thermal ΔT Hoop	Long. Bend. WT	Long. Bend. WT + OBE	WT + DBE	WT+ DBE +Press	Thermal Δt + Press Hoop	WT+ DBE+ Rupt. Jet
Allow. Stress (PSI)	S_A (26,250)	S_h (17,500)	S_h (17,500)	1.2 S_h (21,000)	1.8 Sh or Sy (31,500)	1.8 Sh or Sy (31,500)	1.2 SH (21,000)	1.8 Sh or Sy (31,500)
Main Steam	11,655.	11,375.	221.4	226.02	230.6	2211.6	3191.	1806.9
Feedwater	7883.	11,134.	284.7	301.8	318.9	726.9	8014.	2051.3

TABLE 5.2-3 BELLOWS MOVEMENTS - NORMAL, DBA*, SEISMIC (INCHES)

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PENETRATION NUMBER	AXIAL DISPLACEMENT	TRANSVERSE DISPLACEMENT
6A (MS)	+1.5	1.6
6B (MS)	+1.4	1.0
7A (FW)	+2.5	0.8
7B (FW)	+2.0	1.0
8A (SGB)	+3.4	1.8
8B (SGB)	+3.4	1.8
9 (RHR)	+2.1	0.4
10 (RHR)	+2.1	0.4
11 (CVCS)	+2.2	0.6

NOTE: Compression of Bellows is +
Extension of Bellows is -

* Pressure movements based on field tests data were larger than the theoretical displacement.

TABLE 5.2-4 CHARPY V-NOTCH TEST DATA FOR FLUED HEAD FITTING MATERIAL

	UNIT #1	0°F.
6A	Heat No. 6066057 34-44-54 ft-lb	Full Size-V-Notch
6B	Same Heat 34-44-54 ft-lb	Full Size-V-Notch
7A	Same Heat 44-54-34 ft-lb	Full Size-V-Notch
7B	Same Heat 34-44-54 ft-lb	Full Size-V-Notch
8A	Heat #6730573 30-44-28 ft-lb	Full Size-V-Notch
8B	Same Heat # 30-44-28 ft-lb	Full Size-V-Notch

	UNIT #2	0°F.
6C	Heat #6057177 20-16-22 ft-lb	-30°F Full Size-V-Notch
6D	Heat #6066057 26-32-26 ft-lb	0°F
7C	Heat #6066057 44-54-34 ft-lb	0°F
7D	Heat #6066057 44-54-34 ft-lb	0°F
8C	Heat #6730573 30-44-28 ft-lb	0°
8D	Heat #6730573 36-43-38 ft-lb	0°F

TABLE 5.2-5 CONTAINMENT AIR COOLING SYSTEM DESIGN PERFORMANCE DATA
(PAGE 1 OF 2)

Performance Design Conditions		Normal	Post-Accident *		
			West Coils	Aero Coils	
Heat load per fan coil unit (at 0.002 fouling factor)	Btu/hr x 10^6	1.86	51.849	52.8	
Inlet air flow	CFM	62,000	30,000	34,641 (acfm)**	99044
Entering air temp	°F	120	270	270	
Entering air density	lbm/ft^3	0.068	-	-	
Entering mixture density	lbm/ft^3	-	0.1712	0.1712	
Exit air temp	°F	90	265	258	99044
Air face velocity	FPM	414	205	205	
Air pressure drop W.G. @ 0.075 lbm/ft^3	-	-	0.154	0.15	
Water flow rate	GPM	450	900	900	
Entering water temp	°F	85	85	85	
Exit water temp (at 0.002f.f)	°F	93.4	191	191	99044
(at 0.000f.f.)	°F	-	233	248	
Water velocity	FPS	-	7.05	7.44	
Water pressure drop	ft. hd.	12.5	38.5	35	
Design pressure, internal	psig	150	150	150	
Design pressure, external	psig	-	46	46	
Design temperature	°F	-	300	300	

Seismic criteria: 1.5g Horizontal
1.0g Vertical

* Peak temperature performance tabulated. For performance at other containment temperatures refer to USAR Figure 5.2-11.

** Calculated value based on 30,000 cfm leaving the coils. 99044

TABLE 5.2-5 CONTAINMENT AIR COOLING SYSTEM DESIGN PERFORMANCE DATA (PAGE 2 OF 2)

Performance Design Conditions

Spray chemistry:	3000 ppm of boron, as boric acid, with pH adjusted to 7.0 to 10.5 with sodium hydroxide. 99044
Operation mode:	Normal; indefinite, air at 120°F, at 30% R.H. 99044

TABLE 5.2-6 ESTIMATED HEAT LOSSES FROM EQUIPMENT OR HEAT REMOVAL BY THE CONTAINMENT AIR COOLING SYSTEM AT NORMAL FULL POWER OPERATION

Equipment	Heat Removal Rate, Btu/hr	Design Temp. °F	Operating Temp. °F	
Reactor coolant pumps, two	1,800,000	650	544.5	
Steam generators, two	400,000	600 (Steam)	510.8 (Steam)	99044
Pressurizer	100,000	680	653	
Control rod drive mechanisms	1,350,000	450	392	
Pressurizer relief tank	14,000	340	120	
Primary concrete shield	25,000	210	195	
Reactor vessel support pads, six	72,000	300(1)	423	99044
Reactor vessel (above seal)	20,000	650	Ave. 568.4	
Reactor vessel (below seal)	80,000	650	Ave. 568.4	
Piping	120,000			
Contingency	419,000			
Total	4,400,000			

(1) The design temperature is a maximum of 350°F temperature differential across the Reactor Vessel nozzle. The 300°F is a minimum value based on 650 - 350 = 300°F. 99044

TABLE 5.2-7 SINGLE FAILURE ANALYSIS - VACUUM BREAKER SYSTEM

Component	Malfunction	Comments
Butterfly Valve	Fails to open	Two systems provided. Each system consists of one butterfly valve and one self-actuating swing disc check valve. Evaluation based on operation of one system.
Swing Disc Check Valve	Fails to open	
Butterfly Valve	Fails open on loss of air or electrical power.	Each system consists of one air to close, spring loaded to open, remote operated butterfly valve in series with a self-actuating swing disc check valve. These two valves in series are sufficient to satisfy the single failure criteria. 99044
Swing Check Valve	Fails to close	
Instrumentation and Control	Loss of pressure switch Loss of DC Loss of isolation singal	Butterfly valve will fail to close but check valve will still provide the required isolation.

TABLE 5.2-10 ELECTRICAL PENETRATIONS TEST (INCLUDING CONNECTORS)

		PROTO-TYPE TEST						PRODUCTION TEST					
	Sequence & Test	MVP	LVP	I&C	CRDP	NIS	RM	MVP	LVP	I&C	CRDP	NIS	RM
1.	Dye Penetrant	X	X	X	X	X	X	X	X	X	X	X	X
2.	Continuity							X	X	X	X	X	X
3.	Seismic	X	X	X	X	X	X						
4.	Helium Leakage	X	X	X	X	X	X						
5.	Environ-mental	X	X	X	X	X	X						
6.	Continuous Current	X	X		X								
7.	Interrupting Fault	X	X										
8.	Insul. Resistance	X	X	X	X	X	X	X	X	X	X	X	X
9.	Diel. Strength	X	X	X	X	X	X	X	X	X	X	X	X
10.	Helium Leakage	X	X	X	X	X	X	X	X	X	X	X	X
11.	Continuity	X	X	X	X	X	X						
12.	Steam Pressure	X	X	X	X		X						
13.	Insul. Resistance	X	X	X	X	X	X						
14.	Diel. Strength	X	X	X	X	X	X						
15.	RF Test					X	X						

The above sequence of tests is applicable to the original containment electrical penetrations provided by D.G. O'BRIEN, Inc. Electrical penetrations installed subsequently (D.G. O'BRIEN and CONAX) have been tested per IEEE 317-1976. 99154

TABLE 5.2-12 REACTOR VESSEL SUPPORT STRUCTURE COOLING DATA

Design Reactor Vessel Temperature, t_w	650°F
Air Flow Rate Per Pad	1500 cfm
Bulk Air Temperature, $t_\emptyset$	120°F
Predicted Heat Transfer Coefficient	6.7 Btu/hr-ft2
Fin Efficiency, η	0.406
Predicted Nozzle Interface Temperature, t_i	423°F
Predicted Temperature at the Top of the Side Walls of the Finned Pad, t_R	252°F
Predicted Temperature at the Bottom of the Side Walls of the Finned Pad, t_L	133°F
Heat Dissipation Rate Per Each Wall of the Finned Pad, q''_t	5580 Btu/hr

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TABLE 5.3-1 SHIELD BUILDING LEAKAGE RATES

(Based upon Data presented in the Report
NAA-SR-10100, Conventional Buildings for Reactor Containment)

Source of Leakage	Leakage Rate * (Cubic Feet in 24 Hours)	Leakage Rate * (Percent of Annulus Volume in 24 Hours)
Concrete Surface of Wall & Dome	10	2.67×10^{-3}
Construction Joints	20	5.35×10^{-3}
Cracks in Concrete:		
a Temperature Cracks	50	13.37×10^{-3}
b. Shrinkage Cracks	3	0.8×10^{-3}
c. Earthquake Cracks	Negligible	Negligible
d. Stress Cracks at Springline	2000	0.535
Penetrations (All)	500	0.1337
Equipment Door	30	8.02×10^{-3}
Personnel Door -2	28,800	7.7
	-	-
Total Leakage (In leakage to Shield Building Vent System)	31,413	8.4
		= = =

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* At 1/4" W.C. Differential pressure and 374,000 cu. ft. annulus free volume.

TABLE 5.3-2 MATERIAL SPECIFICATION FOR HEPA FILTERS

Filter type	Flanders Model 7C83-L
Filter medium type	F- 700 micro-glass media
	minimum base weight 44 lbm
Filter cell material	16 Ga. Cadmium plated steel
Frame material	16 Ga. Cold rolled carbon steel
Separator material	.0015 in. thick aluminum alloy Aluminum 5052-H39 or 3003-H19
Adhesive material	Organic base, fire-retardant, meets UL-586
Gasket material	SCE-43 Neoprene (ASTM D1056 applies)

TABLE 5.3-3 DESIGN PARAMETERS OF CHARCOAL FILTERS IN THE SHIELD BUILDING VENTING SYSTEM AND THE AUXILIARY BUILDING SPECIAL VENTILATION SYSTEM

(Page 1 of 3)

<u>Adsorber Design Details & Parameters</u> 00082

Adsorber Drawers	18	
Trays per Drawer	2	
Bed Depth	2″	00082
Face Area of Tray	23-7/8″ x 7-7/8″	
Depth of Tray	26″	
Air Resistance - in. wg.	1.15	00082
Air Flow Rating - CFM		
Adsorber or tray	400	00082
Equiv. 2′ x 2′ face area	1200	
Type of Charcoal	Activated Coconut Shell	
Maximum Air Velocity	45fpm*	
Construction Material	Carbon Steel	
Gasket Material	Closed Sponge Neoprene base (grade SCE-43 per ASTM D 1056) and cured adhesive which is resilient water resistant and resistant to a minimum temperature of 250°F	

* ABSVS filters have been evaluated for 72 fpm by Safety Evaluation #473.

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TABLE 5.3-3 DESIGN PARAMETERS OF CHARCOAL FILTERS IN THE SHIELD BUILDING VENTING SYSTEM AND THE AUXILIARY BUILDING SPECIAL VENTILATION SYSTEM

Material Sizes

Casing Thickness	Double 16 gage, ribbed
Casing Face Flange	11 gage
Spacers, Caps & Dividers	20 gage
Perforated Screens	
Material	26 gage 304 stainless steel
Blank Overlay	
side edges	1/2"
side edges	1/2"

TABLE 5.3-3 DESIGN PARAMETERS OF CHARCOAL FILTERS IN THE SHIELD BUILDING VENTING SYSTEM AND THE AUXILIARY BUILDING SPECIAL VENTILATION SYSTEM

Cover gasket (thickness)	
free	3/8"
compressed	1/8"
Face Dimensions (nom)	24" x 8"
Charcoal Volume - ft^3	1.45 ± 0.05
Face Gasket	
width	7/8"
thickness	1/2"
Mfg. Tolerances	
Face Dimensions	+ 0, -1/8"
Squareness (Diag.)	+ 1/16"

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TABLE 5.3-4 SINGLE FAILURE ANALYSIS-CHARCOAL FILTER WATER DELUGE FEATURE

Component	QA Type	Malfunction	Remarks
Solenoid Valve	I	Fails to open	High temperature alarm in control room coincident w/no flow alarm. Also radiation monitor in exhaust stack.
Solenoid Valve	I	Opens Inadvertantly	Flow is annunciated in control room. Operator must take action to close disch. damper and shutdown fan.
Temperature Switch	I	Fails to function	Multiple temperature switches provided for each filter. 99045
Flow Switch	I (*)	Fails to indicate flow	High temperature alarm to water flow. 99045
Pipe Failure	I	Loss of water supply	Operator must take action to shutdown exhaust fan and close discharge damper it high temperature alarms.
Filter Heater	I (Supplied w/filter)	Overheat due to loss of air flow	Heater automatically trips if recirculation fan trips, or on high temperature downstream of the heater or on low air flow. 99045

* Pressure Boundary only 99045

TABLE 5.4-2 SOURCES AND ASSUMPTIONS FOR HYDROGEN CALCULATIONS

Coolant Absorption of Radiation from Fuel	**Most Conservative Estimate Using Safety Guide 7**
Halogens in fuel	50%
Noble Gases in fuel	0
Other fission products in fuel	99%
Gamma energy fraction absorbed in water	.10
Beta energy fraction absorbed in water	0
$G(H_2)$, molecules/100 ev	0.50
Sources In Coolant	
Halogens in coolant	50%
Noble Gases in coolant	0
Other fission products in coolant	1%
Gamma energy fraction absorbed	1.0
Beta energy fraction absorbed	1.0
$G(H_2)$ molecules/100 ev	0.50
Initial Zirconium-Water Reaction	≥5 times Appendix K value

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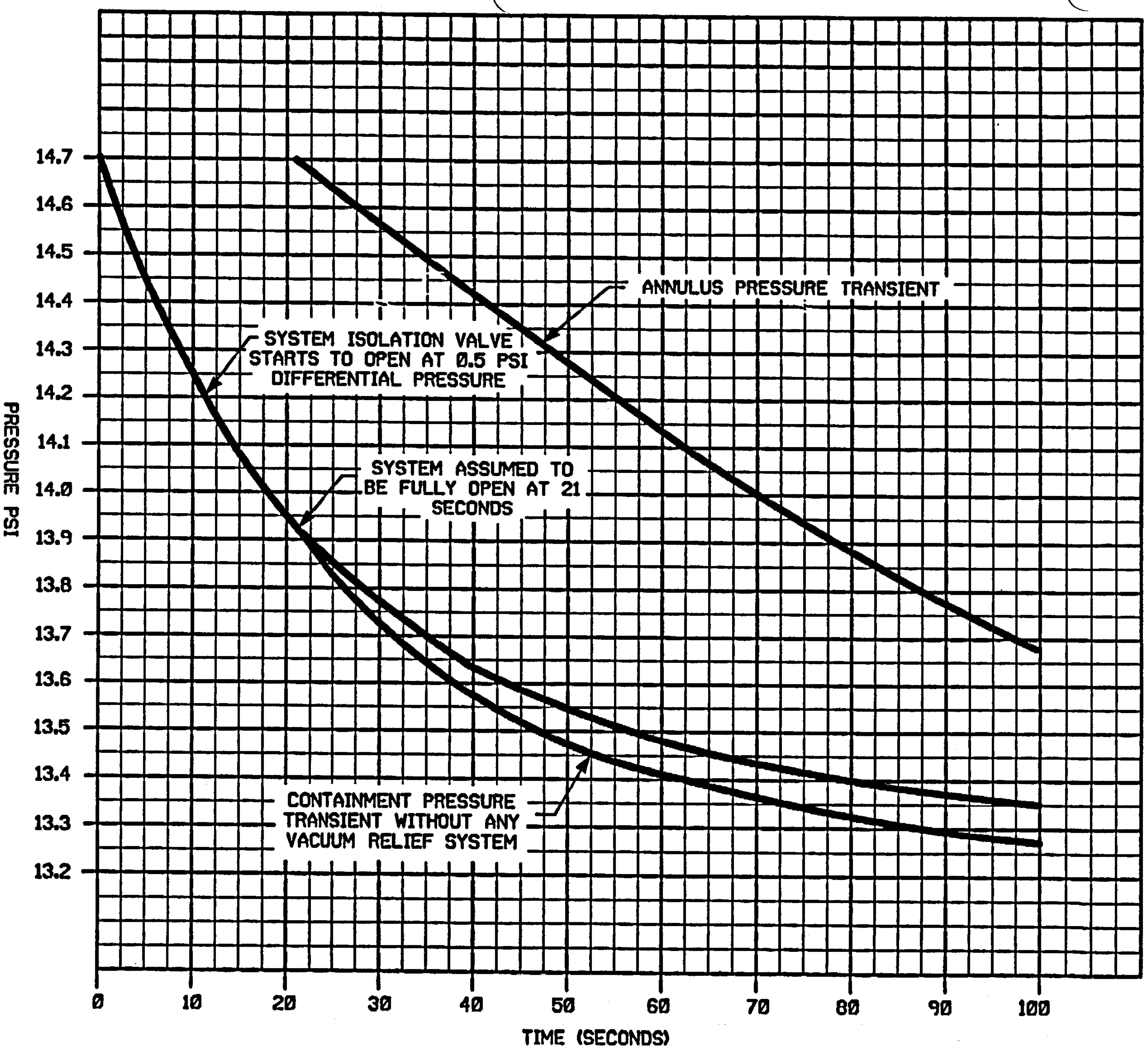

CONTAINMENT AND ANNULUS PRESSURE TRANSIENT

NORTHERN STATES POWER COMPANY
PRAIRIE ISLAND NUCLEAR GENERATING PLANT
RED WING MINNESOTA

DWN T.MILLER | DATE 6-23-99
CHECKED | CAD FILE FIG528.DGN
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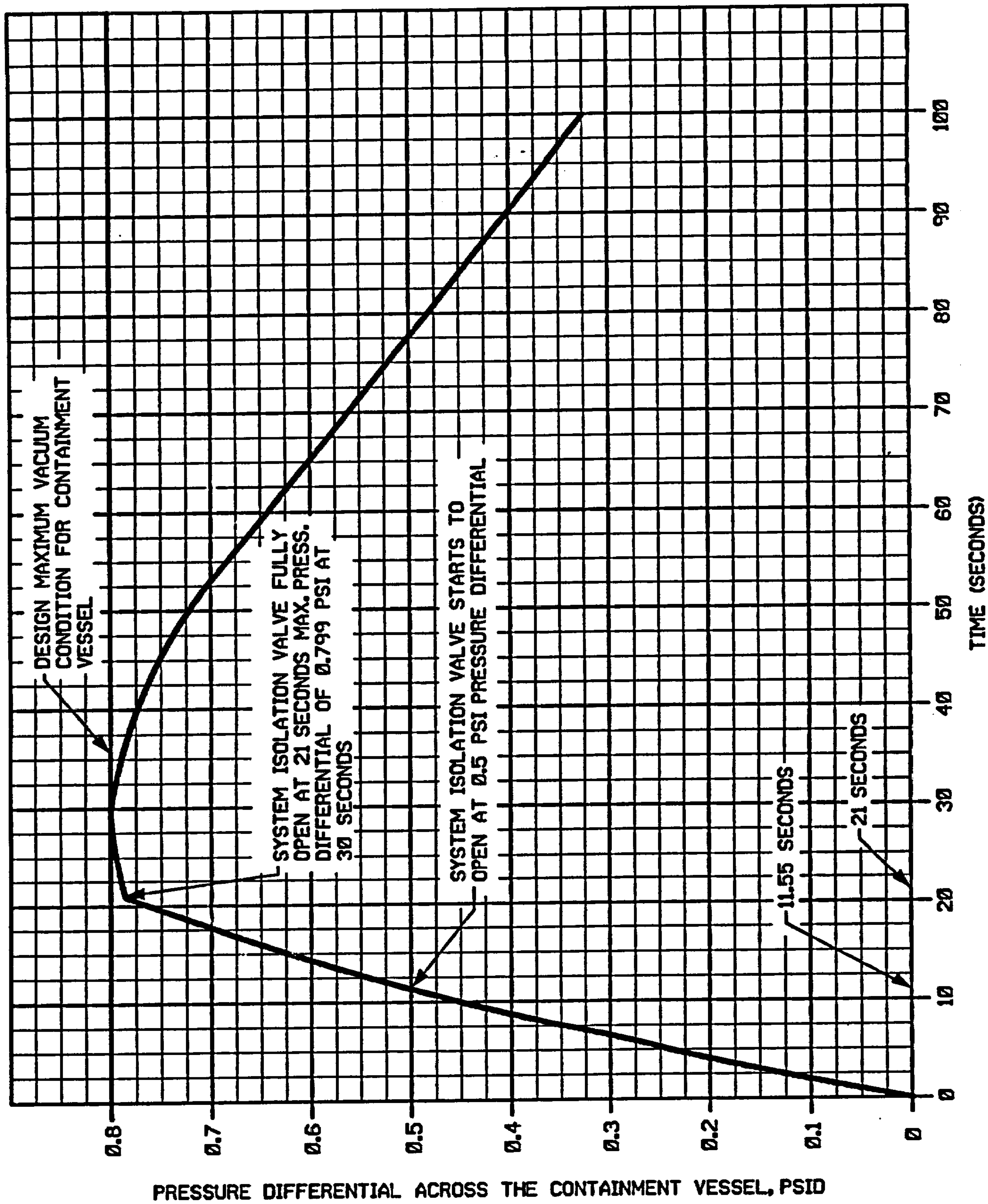
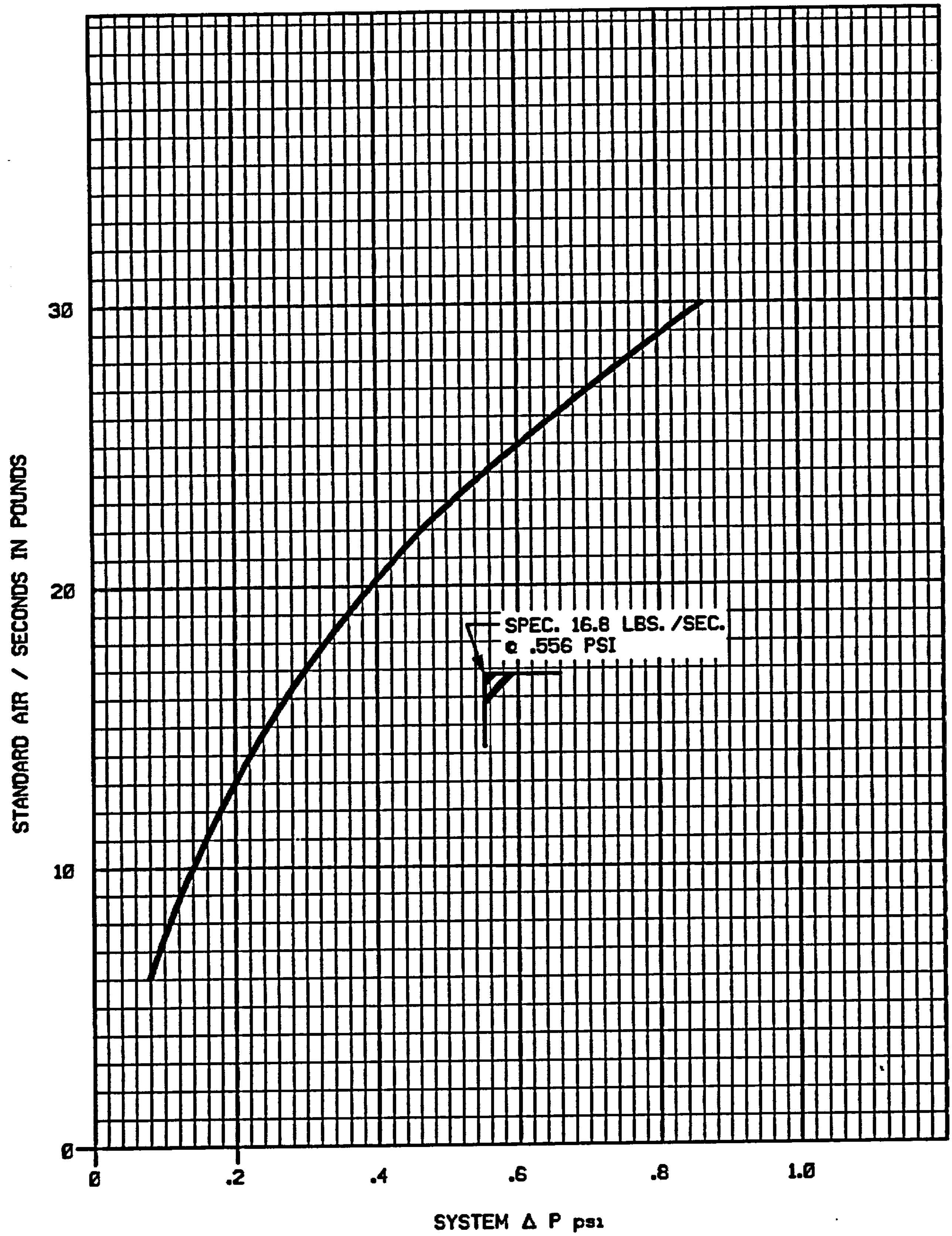
FIGURE 5.2-8 REV. 22



DIFFERENTIAL PRESSURE ACROSS REACTOR CONTAINMENT VESSEL STEEL SHELL WITH ONE VACUUM RELIEF SYSTEM

DWN T. MILLER	DATE 6-23-99	NORTHERN STATES POWER COMPANY PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U05209.DGN		FIGURE 5.2-9 REV. 22



VACUUM BREAKER ASSEMBLY FLOW TEST RESULTS

DWN T. MILLER	DATE 8-15-00	NORTHERN STATES POWER COMPANY PRAIRIE ISLAND NUCLEAR GENERATING PLANT RED WING MINNESOTA	SCALE: NONE
CHECKED	CAD FILE U05212.DGN		FIGURE 5.2-12 REV. 22