Enclosure A to
NMP1L 1782

NINE MILE POINT – UNIT 1

10CFR50.59 EVALUATION SUMMARY REPORT

2003

Docket No. 50-220
License No. DPR-63

50.59 Evaluation No.:	00-027 Rev. 1 & 2
Implementation Document No.:	Procedures GAP-POL-01, NIP-TQS-01
UFSAR Affected Pages:	XIII-6; Figure XIII-4
System:	N/A
Title of Change:	Radiation Protection Department Organizational Change

Description of Change:

The position titled "Radiation Specialist" has been added to the functional areas staffing under the Unit 2 Radiation Protection Manager. The titled position "Supervisor Radiation Protection (Equipment)," under the Unit 1 Radiation Protection Manager, has been eliminated. All duties and functions of this titled position have been transferred to the existing position titled "Supervisor Instrument Calibration." The functional area "Instrument Calibrations" has been relocated from the Unit 2 Radiation Protection Manager to the Unit 2 Maintenance Manager. A descriptive qualifier has been added to clarify the existing requirement that source handling is performed by individuals qualified in radiation protection procedures. The qualifications of non-licensed site organization staff members have been redefined to make them more consistent with applicable standards, or with equivalent existing organizational positions.

50.59 Evaluation Summary:

The changes to the Radiation Protection Department organization, and the qualifications of non-licensed department staff members, conform to the Unit 2 Improved Technical Specifications Section 5.0 and Technical Specifications Section 6.2.1, and the Unit 1 Technical Specifications Section 6.2.1. The changes do not impact initiation of accidents or a malfunction of equipment important to safety.

Based on the evaluation performed, it is concluded that these changes do not involve an unreviewed safety question.

50.59 Evaluation No.: 01-005

Implementation Document No.: DDC 1M01111

UFSAR Affected Pages: Figure XI-7

System: Feedwater

Title of Change: Extension of 'Q' Boundary for Motor-Driven Feedwater Pump 6" Recirculation Lines

Description of Change:

This evaluation addresses changing the high-pressure coolant injection (HPCI) pressure boundary in the feedwater 6" recirculation lines from manual blocking valves BV-29-55 and BV-29-57 to the recirculation flow control valves FCV-29-51 and FCV-29-52. This change allows valves BV-29-55 and BV-29-57 to be positioned to normally open, and eliminates the need for operators to manually reposition these valves during startups and shutdowns.

50.59 Evaluation Summary:

Both the original 6-inch recirculation flow control valves (FCV-29-51 and FCV-29-52) and the manual blocking valves (BV-29-55 and BV-29-57) were installed per Modification 82-69-2. All of these valves were designed to ANSI B16.34 code requirements. Modification 82-69-2 was installed as safety related and met 1980 ASME Boiler and Pressure Vessel Code Section III for design and installation. The use of ASME Boiler and Pressure Vessel Code Section III met or exceeded the original construction code requirements of ANSI B31.1-1955. The replacement flow control valves, from Control Components, Inc., (CCI), were also designed to ANSI B16.34 code requirements. Although Design Change N1-97-030 was installed as nonsafety related, it was designed and installed to the requirements of ANSI B31.1-1986. As part of the design change, CCI provided Niagara Mohawk with a design report demonstrating the flow control valve assemblies are capable of withstanding design basis earthquake seismic loads (Calculation S14-29V001). Additionally, Structural Design Engineering evaluated the replacement valves and piping to ensure compliance with the original seismic qualifications (Calculation Disposition S12-29-P00300A).

Flow control valves FCV-29-51 and FCV-29-52 are normally closed during power operation conditions, fail closed on a loss of motive force or a loss of air, and receive an additional close signal upon HPCI initiation. Thus, the flow control

50.59 Evaluation No.: 01-005 (cont'd.)

50.59 Evaluation Summary: (cont'd.)

valves will reliably provide the isolation function for the 6-inch recirculation lines from HPCI during power-operated conditions.

Based on the evaluation performed, it is concluded that these changes do not require prior NRC approval.

50.59 Evaluation No.:	02-001
Implementation Document No.:	Temp. Change Package N1-02-125
UFSAR Affected Pages:	N/A
System:	Emergency Cooling
Title of Change:	Temporary Plugging the Vent Port of Solenoid-Operated Valve SOV-39-05E

Description of Change:

In an effort to preclude an inadvertent opening of air-operated valve IV-39-05, the exhaust port of solenoid-operated valve SOV-39-05E was temporarily plugged. This compensatory action rendered SOV-39-05E inoperable and eliminated any instrument air leakage from the valve, thus reducing the possibility of inadvertently opening IV-39-05. Valve SOV-39-05E was replaced during refueling outage (RFO) 17.

50.59 Evaluation Summary:

The plant safety analysis and emergency cooling system design basis assume that the emergency condensers will initiate automatically once the plant parameters (reactor pressure and level) reach initiation setpoints. The safe shutdown analysis indicates there is the capability to manually initiate the system and control the cooldown rate from the remote shutdown panels, independent of the control room. Plugging the exhaust port of SOV-39-05E is a compensatory action that renders the SOV inoperable; however, only one of the two ac SOVs is required to operate for Appendix R purposes. Valve SOV-39-05F is still available to perform this function. Plugging the exhaust port of SOV-39-05E does not affect the passive safety function of the SOV. The Appendix R valve SOV-39-05E has no other system interactions, so this temporary change has no effect on the safety function of the emergency condensers/emergency cooling system. In addition, this temporary change does not affect the ability of operators to initiate emergency cooling from the remote shutdown panels. Therefore, this temporary change will not affect the Appendix R Safe Shutdown Analysis, the plant safety analysis, or the emergency cooling system design basis.

Based on the evaluation performed, it is concluded that this temporary change does not require prior NRC approval.

50.59 Evaluation No.:	02-002
Implementation Document No.:	Modification N1-01-149
UFSAR Affected Pages:	III-43, III-44, V-6
System:	Feedwater Zinc Injection System
Title of Change:	Zinc Injection Skid Installation

Description of Change:

This modification installed a passive zinc injection skid which adds zinc to the feedwater and thereby to the reactor coolant. Addition of zinc into the reactor coolant is a demonstrated means of reducing Co-60 buildup in primary piping corrosion films. This has the major benefit of reducing radiation dose rates in the drywell, thus reducing radiation exposure during outages.

50.59 Evaluation Summary:

The General Electric Co. zinc injection passivation (GEZIP) process does not change or affect any reactor operational condition and, therefore, no change to the pressure/temperature loadings stresses previously evaluated for any equipment. It also does not change thermal properties/energy content of the fuel or thermal/radiological properties of the reactor coolant during normal reactor operation, any transient, or any accident evaluated in the UFSAR. Therefore, it has no effect on short- or long-term drywell/containment pressure or temperature responses, or any radiological consequences from any reactor coolant loss evaluated in the UFSAR. Because the GEZIP process has no effect on the radiological/thermal properties of the fuel during normal operation or after an accident, or any leakage path previously evaluated such as through the containment, there is no change to the assumptions used and analytical results in the radiological analysis in the UFSAR. Likewise, there would be no effect on the performance and function of the reactor building emergency ventilation system. The GEZIP process has no effect on the functioning of any mechanical or structural component. Thus, it will not promote a malfunction or create the possibility for a malfunction of any structure, system, or component including those that are safety related. Since the GEZIP process does not affect any of the safety-related parameters or assumptions used in any of the safety analyses evaluated in the UFSAR, it does not affect plant/reactor safety or the health and safety of the public. Based on the evaluation performed, it is concluded that this change does not require prior NRC approval.

50.59 Evaluation No.:	03-001
Implementation Document No.:	General Electric Topical Report NEDC-32992P
UFSAR Affected Pages:	IV-14
System:	Reactor Core
Title of Change:	Use of ODYSY for Restricted Region Calculations

Description of Change:

General Electric Co. has licensed the use of the ODYSY code to replace the FABLE/BYPSSS code in order to improve the accuracy of the calculation of stability restricted regions. The ODYSY code was used for refueling outage (RFO) 17 stability analysis.

50.59 Evaluation Summary:

The ODYSY code uses methods of evaluation which have been approved by the NRC. The subject methods of evaluation are appropriate for the intended application, and the terms and conditions for their use, as specified by the NRC, have been satisfied. By definition (a)(2) of the new 10 CFR 50.59 rule, NRC-approved methods of evaluation are not considered a departure from methods described in the UFSAR. Therefore, from a criterion (viii) review, the use of ODYSY for restricted region calculations does not require prior NRC review.

Based on the evaluation performed, it is concluded that this change does not require prior NRC approval.

50.59 Evaluation No.:	03-002
Implementation Document No.:	General Electric Topical Reports NEDE-32601P, NEDC-32694P
UFSAR Affected Pages:	IV-11, IV-33
System:	Reactor Core
Title of Change:	Use of New Methods for the Safety Limit Minimum Critical Power Ratio

Description of Change:

The safety limit minimum critical power ratio (SLMCPR) has been calculated with new methods to improve MCPR operating margins for refueling outage (RFO) 17. This allowed achieving the desire cycle energy with fewer fuel bundles for significant cost savings.

50.59 Evaluation Summary:

This activity uses methods of evaluation that have been approved by the NRC. The subject methods of evaluation are appropriate for the intended application, and the terms and conditions for their use, as specified by the NRC, have been satisfied. By definition (a)(2) of the new 10 CFR 50.59 rule, NRC-approved methods of evaluation are not considered a departure from methods described in the UFSAR. Therefore, from a criterion (viii) review, the use of the methods of NEDC-32694P for SLMCPR calculations does not require prior NRC review.

Based on the evaluation performed, it is concluded that this change does not require prior NRC approval.

50.59 Evaluation No.:	03-003
Implementation Document No.:	LDCR 1-03-UFS-013
UFSAR Affected Pages:	Table VI-3b Sh 2, 3 & 4
System:	Core Spray, Containment Spray
Title of Change:	Elimination of Water Leak Rate Requirements for Core Spray (CRS) and Containment Spray (CTN-SP) Torus Suction

Description of Change:

This change eliminated water leak rate testing for core spray system suction valves and containment spray suction valves. These valves are tested in accordance with the Inservice Testing (IST) Program plan to assure that the valves will close to perform their containment isolation function.

50.59 Evaluation Summary:

Testing of these valves includes full exercise, stroke time closed, and position indication verification. These tests adequately demonstrate valve closure capability.

Based on the evaluation performed, it is concluded that this change does not require prior NRC approval.

Enclosure B to
NMP1L 1782

NINE MILE POINT - UNIT 1

TECHNICAL SPECIFICATONS BASES
CHANGE SUMMARY

2003

Docket No. 50-220
License No. DPR-63

Amendment 171	Bases Sections 3/4.4.4 (Page 176) and 3/4.4.5 (Page 180) were revised to discuss the increase in radioactive methyl iodide removal efficiency from 90 to 95 percent and update the regulatory basis documents consistent with NRC Generic Letter 99-02, "Laboratory Testing of Nuclear Grade Activated Charcoal." These changes reflect changes to the Technical Specifications approved in License Amendment No. 171. The affected Bases pages are annotated with the amendment number due to issuance by the NRC with the Technical Specifications amendment.
Amendment 172	Bases Sections 3/4.2.3 (Page 98) and 3/4.2.6 (Page 107) were revised to indicate that Niagara Mohawk Power Corporation is no longer the licensee for Nine Mile Point Units 1 and 2 (NMP1 and NMP2) due to transfer of its interests in the plants to Nine Mile Point Nuclear Station, LLC (NMPNS). These changes reflect direct transfer of the Operating Licenses for NMP1 and NMP2 to NMPNS and incorporate the conforming changes to the Technical Specifications as approved in License Amendment No. 172. The affected Bases pages are annotated with the amendment number due to issuance by the NRC with the Technical Specifications amendment.
Revision 1	Bases Section 3/4.2.9 (Page 122) was revised to clarify the setpoint requirements for the solenoid-actuated pressure relief valves and provide the limits for the Allowable Value. In addition, a List of Effective Pages (Pages LEP-1 through LEP-5) was established for the Technical Specifications and Bases as part of this Bases revision. These were Bases changes only, i.e., they did not involve a Technical Specifications amendment.
Revision 2	Bases Sections 4.0.1 (Page 27), 3/4.2.6 (Page 107), and 3/4.2.7 (Page 115) were revised to support the establishment of Administrative Controls Specification 6.17, "Inservice Testing Program." Note that Bases Page 107 was deleted since the corresponding Technical Specifications requirements were removed from the Technical Specifications. These changes reflect Technical Specifications changes approved in License Amendment No. 173 to update the requirements consistent with current NRC

	guidance and the improved Standard Technical Specifications for BWR/4 and BWR/6 (NUREG-1433 and NUREG-1434).
Revision 3	Bases Section 3/4.6.4 (Page 264) was revised to provide the basis for an alternate inspection schedule as specified in the Technical Specifications for the piping shock suppressors (snubbers). These changes reflect Technical Specifications changes approved in License Amendment No. 175 which adopt the snubber visual inspection and acceptance requirements of the model Technical Specifications included in Generic Letter 90-09, "Alternative Requirements for Snubber Visual Inspection and Corrective Actions."
Amendment 176	Bases Sections 3/4.6.2 (Page 252), 3/4.6.14 (Page 294), 3/4.6.15 (Pages 296 and 308 through 313), 3/4.6.16 (Page 316), 3/4.6.17 (Page 318), 3/4.6.18 (Page 320), 3/4.6.19 (Page 322), 3/4.6.20 (Page 333), 3/4.6.21 (Page 335), and 3/4.6.22 (Page 338) were revised to support changes to the Radiological Effluent Technical Specifications (RETS). Specifically, the Bases changes reflect changes to the RETS that: (1) established programmatic controls for RETS in the Administrative Controls section of the Technical Specifications, (2) relocated procedural details to licensee-controlled documents and new programs, and (3) updated the references to 10 CFR 20. Note that Bases Pages 294, 308 - 313, 316, 318, 320, 322, 333, 335, and 338 were deleted since the corresponding RETS requirements were relocated out of the Technical Specifications. The RETS changes were approved in Amendment No. 176. The affected Bases pages are annotated with the amendment number due to issuance by the NRC with the Technical Specifications amendment.
Revision 4	Bases Section 3/4.1.1 (Pages 39, 41, and 43) was revised to specify the new Technical Specifications thermal power limit for rod worth minimizer operability and to update the references to the control rod drop accident analyses. These changes reflect Technical Specifications changes approved in Amendment No. 178 relating to use of the rod worth minimizer.

Revision 5	Bases Section 3/4.1.1 (Pages 37, 37a, 37b, 37c, and 43) was revised to provide additional background and basis information to support the shutdown margin (SDM) changes in the Technical Specifications. Bases Section 3/4.7.1 (Page 341) was also revised to eliminate the restriction requiring SDM demonstration prior to power operation. These changes reflect Technical Specifications changes approved in Amendment No. 180 that were intended to clarify the existing requirements and make the requirements more consistent with the Standard Technical Specifications.
Revision 6	Bases Sections 3/4.2.7 (Page 115) and 3/4.3.4 (Page 150) were revised to incorporate conforming reformatting and renumbering changes to support changes to Technical Specifications Section 6.0, "Administrative Controls," requirements. The Bases changes reflect Technical Specifications changes approved in Amendment No. 181 that were intended to clarify the existing requirements and make the requirements more consistent with the NMP2 Technical Specifications and the Standard Technical Specifications.
Revision 7	Bases Sections 4.0.1 (Pages 27b and 27c), 4.0.2 (Page 27c), and 4.0.3 (Pages 27d and 27e) were revised to support adoption of Standard Technical Specifications SRs 3.0.1 (NMP1 Technical Specifications Section 4.0.1) and 3.0.3 (NMP1 Technical Specifications Section 4.0.3) consistent with NRC approved Industry/Technical Specification Task Force (TSTF) change TSTF-358 regarding missed surveillances. The Bases changes were revised consistent with the Standard Technical Specifications Bases for SRs 3.0.1 and 3.0.3, and also include conforming to the Technical Specifications. The Bases changes reflect Technical Specifications changes approved in Amendment No. 182.

Nine Mile Point Unit 1 Technical Specifications Bases

INSERTION INSTRUCTIONS

TECHNICAL SPECIFICATIONS BASES

The following instructions are for the insertion of revised Bases pages into the Nine Mile Point Unit 1 Technical Specifications Bases.

Remove pages, tables, and/or figures listed in the REMOVE column and replace them with the pages, tables, and/or figures listed in the INSERT column. Dashes (---) in either column indicate no action required.

REMOVE	INSERT
---	LEP-1 through LEP-5
---	27b
---	27c
---	27d
---	27e
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NMP1 FACILITY OPERATING LICENSE (FOL) AND
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Specifications 4.0.1 through 4.0.3 establish general requirements applicable to all specifications in Sections 4.1 through 4.7 and apply at all times, unless otherwise stated.

4.0.1 Specification 4.0.1 establishes the requirement that SRs must be met during the applicable reactor operating or other specified conditions for which the requirements of the LCO apply, unless otherwise specified in the individual SRs. This specification is to ensure that surveillances are performed to verify the operability of systems and components, and that variables are within specified limits. Failure to meet a surveillance within the specified frequency, in accordance with Specification 4.0.2, constitutes a failure to meet an LCO. Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire surveillance is performed within the specified frequency.

Systems and components are assumed to be operable when the associated SRs have been met. Nothing in this specification, however, is to be construed as implying that systems or components are operable when either:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the surveillance(s) are known to be not met between required surveillance performances.

Surveillances do not have to be performed when the unit is in a reactor operating or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a special test exception LCO are only applicable when the special test exception LCO is used as an allowable exception to the requirements of a specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given reactor operating or other specified condition.

Surveillances, including surveillances invoked by LCO actions, do not have to be performed on inoperable equipment because the applicable individual specifications define the remedial measures that apply. Surveillances have to be met and performed in accordance with Specification 4.0.2, prior to returning equipment to operable status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment operable. This includes ensuring applicable surveillances are not failed and their most recent performance is in accordance with Specification 4.0.2. Post maintenance testing may not be possible in the current reactor operating or other specified conditions in the LCO due to the necessary unit parameters not having been established. In these situations, the equipment may be considered operable provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a reactor operating or other specified condition where other necessary post maintenance tests can be completed.

4.0.2 Specification 4.0.2 establishes the limit for which the specified time interval for SRs may be extended. It permits an allowable extension of the surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with a 24 month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Specification 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the SRs. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

4.0.3 Specification 4.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a surveillance has not been completed within the specified frequency. A delay period of up to 24 hours or up to the limit of the specified frequency, whichever is greater, applies from the point in time it is discovered that the surveillance has not been performed in accordance with Specification 4.0.2, and not at the time that the specified frequency was not met. This delay period permits the completion of a surveillance before complying with LCO actions or other remedial measures that might preclude completion of the surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the surveillance, the safety significance of the delay in completing the required surveillance, and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the requirements.

When a surveillance with a frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to power operation, or in accordance with the 10 CFR 50 Appendix J Testing Program Plan, etc.) is discovered to not have been performed when specified, Specification 4.0.3 allows for the full delay period of up to the specified frequency to perform the surveillance. However, since there is not a time interval specified, the missed surveillance should be performed at the first reasonable opportunity.

Specification 4.0.3 provides a time limit for, and allowances for the performance of, surveillances that become applicable as a consequence of operating condition changes imposed by LCO actions.

Failure to comply with specified frequencies for surveillance requirements is expected to be an infrequent occurrence. Use of the delay period established by Specification 4.0.3 is a flexibility which is not intended to be used as an operational convenience to extend surveillance intervals. While up to 24 hours or the limit of the specified frequency is provided to perform the missed surveillance, it is expected that the missed surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the surveillance as well as any plant configuration changes required or shutting the plant down to perform the surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the surveillance. The risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed surveillances will be placed in the Corrective Action Program.

If a surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable then is considered outside the specified limits and entry into the applicable LCO actions begin immediately upon expiration of the delay period. If a surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and entry into the applicable LCO actions begin immediately upon failure of the surveillance.

Completion of the surveillance within the delay period allowed by this specification, or within the times allowed by LCO actions, restores compliance with Specification 4.0.1.

a. Reactivity Limitations

(1) Reactivity margin - core loading

The control rod drop accident analysis assumes the core is subcritical with the highest worth control rod withdrawn. Typically, the first control rod withdrawn has a very high reactivity worth and, should the core be critical during the withdrawal of the first control rod, the consequences of a control rod drop accident could exceed the fuel damage limits for the accident.

Prevention or mitigation of reactivity insertion events is necessary to limit energy deposition in the fuel to prevent significant fuel damage, which could result in undue release of radioactivity. Adequate SDM ensures inadvertent criticalities and potential control rod drop accidents involving high worth control rods (namely the first control rod withdrawn) will not cause significant fuel damage.

The SDM limits specified in Specification 3.1.1a(1)(a) account for the uncertainty in the demonstration of SDM by testing. Separate SDM limits are provided for testing where the highest worth control rod is determined analytically or by measurement. This is due to the reduced uncertainty in the SDM test when the highest worth control rod is determined by measurement. When SDM is demonstrated by calculations not associated with a test (e.g., to confirm SDM during the fuel loading sequence), additional margin must be added to the specified SDM limit to account for uncertainties in the calculation. To ensure adequate SDM, a design margin is included to account for uncertainties in the design calculations (Reference (8)).

The inability to meet the SDM limits during power operating conditions would most likely be due to withdrawn control rods that cannot be inserted. A reduced SDM is not considered an immediate threat to nuclear safety; therefore, time is allowed for analysis to ensure Specification 3.1.1a(1)(a) is met, and for repair before requiring the plant to undergo a transient to achieve a shutdown condition. The allowed completion times of 6 hours for analysis and an additional 6 hours for repair, if Specification 3.1.1a(1)(a) is not met, are considered reasonable while limiting the potential for further reductions in SDM or the occurrence of a transient.

If the SDM cannot be restored within the allowed time, a plant shutdown is required to minimize the potential for, and consequences of, an accident or malfunction of equipment important to safety. The allowed completion time of 10 hours is considered reasonable to achieve the shutdown condition from full power in an orderly manner and without challenging plant systems.

The inability to meet the SDM limits in the hot shutdown condition or the cold shutdown condition could be due to withdrawn control rods that cannot be inserted, discovery of errors in the SDM analysis, or discovery of errors in previous core alterations. The immediate action to fully insert all insertable control rods will result in the least reactive condition for the core and maximizes SDM. This action must continue until all insertable control rods are fully inserted. Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is operable, at least one emergency ventilation system is operable, and secondary containment isolation capability is available in each associated secondary containment penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases (i.e., at least one secondary containment isolation valve and associated instrumentation are operable, or other acceptable administrative controls to assure isolation capability. These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated). This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the surveillances needed to demonstrate the operability of the components. If, however, any required component is inoperable, then it must be restored to operable status. In this case, surveillances may need to be performed to restore the component to operable status. Actions must continue until all required components are operable.

The inability to meet the SDM limits in the refueling condition would most likely be due to fuel loading errors. The immediate action to suspend core alterations (e.g., fuel loading) prevents further reductions in SDM. Suspension of core alterations shall not preclude completion of movement of a component to a safe condition. Inserting control rods or removing fuel from the core will reduce the total reactivity and is, therefore, allowed in order to recover SDM. Action must also be immediately initiated to fully insert all insertable control rods in core cells containing one or more fuel assemblies. This action must continue until all insertable control rods in core cells containing one or more fuel assemblies have been fully inserted. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and, therefore, do not have to be inserted.

Adequate SDM must be verified to ensure that the reactor can be made subcritical from any initial reactor operating condition, except the major maintenance condition. This can be accomplished by a test, an evaluation, or a combination of the two. Adequate SDM is demonstrated by testing before or during the first startup after fuel movement, or shuffling within the reactor pressure vessel, or control rod replacement. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain SDM, the initial measured value must be increased by an adder, "R", which is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated BOC core reactivity. If the value of R is negative (that is, BOC is the most reactive point in the cycle), no correction to the BOC is required. For the SDM demonstrations that rely solely on calculation of the highest worth control rod, additional margin (0.10% $\Delta k/k$) must be added to the SDM limit of 0.28% $\Delta k/k$ to account for uncertainties in the calculation.

The SDM may be demonstrated during an in-sequence control rod withdrawal, in which the highest worth control rod is analytically determined, or during local criticals, where the highest worth control rod is determined by testing. Local critical tests require the withdrawal of out of sequence control rods.

The frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and have appropriate verification.

During the refueling condition, adequate SDM is also required to ensure the reactor does not reach criticality during control rod withdrawals. An evaluation of each in vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload or reload sequences inherently satisfy the surveillance, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

(2) Reactivity margin – stuck control rods

The specified limits provide sufficient scram capability to accommodate failure to scram of any one operable rod. This failure is in addition to any inoperable rods that exist in the core, provided that those inoperable rods met the core reactivity Specification 3.1.1a(1)(a).

Control rods which cannot be moved with control rod drive pressure are indicative of an abnormal operating condition on the affected rods and are, therefore, considered to be inoperable. Inoperable rods are valved out of service to fix their position in the core and assure predictable behavior. If the rod is fully inserted and then valved out of service, it is in a safe position of maximum contribution to shutdown reactivity. If it is valved out of service in a non-fully inserted position, that position is required to be consistent with the shutdown reactivity limitation stated in Specification 3.1.1a(1)(a), which assures the core can be shut down at all times with control rods.

(2) The rod housing support is provided to prevent control rod ejection accidents. Its design is discussed in Section VII-E*. Procedural control shall assure that the housing supports are in place for all control rods.

(3) Control rod withdrawal and insertion sequences are established to assure that the maximum in-sequence individual control rod or control rod segments which are withdrawn could not be worth enough to cause the core to be more than 0.013 Δk supercritical if they were to drop out of the core in the manner defined for the Rod Drop Accident.[3] These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow the sequences is backed up by the operation of the RWM. This 0.013 Δk limit, together with the integral rod velocity limiters and the action of the control rod drive system, limits potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy content of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in reference 1.

Improvements in analytical capability have allowed more refined analysis of the control rod drop accident [1][2][3][4][5][7]. By using the analytical models described in these references coupled with conservative or worst-case input parameters, it has been determined that for power levels less than 10% of rated power, the specified limit on in-sequence control rod or control rod segment worths will limit the peak fuel enthalpy content to less than 280 cal/gm. Above 10% power, even multiple operator errors cannot result in a peak fuel enthalpy content of 280 cal/gm should a postulated control rod drop accident occur.

The following conservative or worst-case bounding assumptions have been made in the analysis used to determine the specified 0.013 Δk limit on in-sequence control rod or control rod segment worths. The allowable boundary conditions used in the analysis are quantified in references (4) and (5). Each core reload will be analyzed to show conformance to the limiting parameters.

*FSAR

The RWM provides automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. It serves as an independent backup of the normal withdrawal procedure followed by the operator. In the event that the RWM is out of service when required, a second independent operator or engineer can manually fulfill the operator-follower control rod pattern conformance function of the RWM. In this case, procedural control is exercised by verifying all control rod positions after the withdrawal of each group, prior to proceeding to the next group. Allowing substitution of a second independent operator or engineer in case of RWM inoperability recognizes the capability to adequately monitor proper rod sequencing in an alternate manner without unduly restricting plant operations. Above 10% power, there is no requirement that the RWM be operable since the control rod drop accident with out-of-sequence rods will result in a peak fuel energy content of less than 280 cal/gm. To assure high RWM availability, the RWM is required to be operating during a startup for the withdrawal of a significant number of control rods for any startup.

(4) The source range monitor (SRM) system performs no automatic safety function. It does provide the operator with a visual indication of neutron level which is needed for knowledgeable and efficient reactor startup at low neutron levels. The results of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 cps assures that any transient begins at or above the initial value of 10^{-8} of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to critical using homogeneous patterns of scattered control rods. A minimum of three operable SRMs is required as an added conservation.

c. Scram Insertion Times

The revised scram insertion times have been established as the limiting condition for operation since the postulated rod drop analysis and associated maximum in-sequence control rod worth are based on the revised scram insertion times. The specified times are based on design requirements for control rod scram at reactor pressures above 950 psig. For reactor pressures above 800 psig and below 950 psig the measured scram times may be longer. The analysis discussed in the next paragraph is still valid since the use of the revised scram insertion times would result in greater margins to safety valves lifting.

f. Reactivity Anomalies

During each fuel cycle excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary controls is burned. The magnitude of this excess reactivity is indicated by the integrated worth of control rods inserted into the core, referred to as the control rod inventory in the core. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of actual rod inventory at any base equilibrium core state to predicted rod inventory at that state. Equilibrium xenon, samarium and power distribution are considered in establishing the steady-state base condition to minimize any source of error. During an initial period, (on the order of 1000 MWD/T core average exposure following core reloading or modification) rod inventory predictions can be normalized to actual rod patterns to eliminate calculational uncertainties. Experience with other operating BWR's indicates that the control rod inventory should be predictable to the equivalent of one percent in reactivity. Deviations beyond this magnitude would not be expected and would require thorough evaluation. One percent reactivity limit is considered safe since an insertion of this reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

(1) Paone, C. J., Stirn, R. C., and Wooley, J. A., "Rod Drop Accident Analysis for Large Boiling Water Reactors," NEDO-10527, March 1972.

(2) Stirn, R. C., Paone, C. J., and Young, R. M., "Rod Drop Accident Analysis for Large BWRs," Supplement 1 - NEDO-10527, July 1972.

(3) Stirn, R. C., Paone, C. J., and Haun, J. M., "Rod Drop Accident Analysis for Large Boiling Water Reactors Addendum No. 2 Exposed Cores," Supplement 2 - NEDO-10527, January 1973.

(4) Report entitled "Technical Basis for Changes to Allowable Rod Worth Specified in Technical Specification 3.3.B.3," transmitted by letter from L. O. Mayer (NSP) to J. F. O'Leary (USAEC), dated October 4, 1973.

(5) Letter, R. R. Schneider, Niagara Mohawk Power Corporation to A. Giambusso, USAEC, dated November 15, 1973.

(6) To include the power spike effect caused by gaps between fuel pellets.

(7) NRC Safety Evaluation, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," dated December 27, 1987.

(8) Licensing Topical Report, "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, latest approved revision.

In its May 8, 1997 letter, the NRC required that the licensee submit an application for amendment to address the differences between the current TS conductivity limits for reactor coolant chemistry and the analysis assumptions for the core shroud crack growth evaluations. The purpose of this specification is to limit intergranular stress corrosion cracking (IGSCC) crack growth rates through the control of reactor coolant chemistry. The LCO values ensure that transient conditions are acted on to restore reactor coolant chemistry values to normal in a reasonable time frame. Under transient conditions, potential crack growth rates could exceed analytical assumptions, however, the duration will be limited so that any effect on potential crack growth is minimized and the design basis assumptions are maintained. The plant is normally operated such that the average coolant chemistry for the operating cycle is maintained at the conservative values of <0.19 μmho/cm for conductivity and <5 ppb for chloride ions and <5 ppb for sulfate ions. This will ensure that the crack growth rate is bounded by the core shroud analysis assumptions. Since these are average values, there are no specific LCO actions to be taken if these values are exceeded at a specific point in time. The EPRI "BWR Water Chemistry Guidelines-1996 Revision" (EPRI TR-103515-R1, BWRVIP-29) action level 1 guidelines suggest that if conductivity is above 0.3 μS/cm, or chloride or sulfate ions exceed 5 ppb, that corrective action be initiated as soon as possible and to restore levels below level 1 within 96 hours. If the parameters are not reduced to below these levels within 96 hours, complete a review and implement a program and schedule for implementing corrective measures.

Specifications 3.2.3a, b, and c are consistent with the licensee's commitment to Table 4.4 of the BWR water chemistry guidelines. The 24 hour action time period for exceeding the coolant chemistry limits described in 3.2.3a and b ensures that prompt action is taken to restore coolant chemistry to normal operating levels. The requirement to commence a shutdown within 1 hour, and to be shutdown and reactor coolant temperature be reduced to <200 degrees F within 10 hours minimizes the potential for IGSCC crack growth. Reactor water samples are analyzed daily to ensure that reactor water quality remains within the BWR water chemistry guidelines. These samples are analyzed and compared to action level 1 values.

The conductivity of the reactor coolant is continuously monitored. The continuous conductivity monitor is visually checked shiftly in accordance with procedures. The monitor alarms at the local panel. The recorder, which is located in the Control Room, alarms in the Control Room. The samples of the coolant which are analyzed for conductivity daily will serve as a comparison with the continuous conductivity monitor. The primary sample point for the reactor water conductivity samples is the non-regenerative heat exchanger in the reactor water cleanup system. An alternate sample point is the #11 recirculation loop. The reactor coolant samples will also be used to determine the chloride and sulfate concentrations. Therefore, the sampling frequency is considered adequate to detect long-term changes in the chloride and sulfate ion content. However, if the conductivity becomes abnormal (>0.19 μmho/cm), other than short term spikes, chloride and sulfate measurements will be made within 8 hours to assure that the normal limits (<5 ppb of chloride or sulfate ions) are maintained. A short term spike is defined as a rise in conductivity (>0.19 μmho/cm) such as that which could arise from injection of additional feedwater flow for a duration of approximately 30 minutes in time. These actions will minimize the potential for IGSCC crack growth.

NMP1 will use Noble Metal Chemical Addition (NMCA) as a method to enhance the effectiveness of Hydrogen Water Chemistry (HWC) in mitigating IGSCC. NMCA will result in temporary increases in reactor coolant conductivity values during and following application. During application, the conductivity limit specified in 3.2.3a and 3.2.3c.1 is increased to 20 μmho/cm. The application period includes post-NMCA injection cleanup activities conducted prior to returning the plant to power operation. An increase in conductivity is expected principally due to residual ionic species from the NMCA. However, these species have minor effects on IGSCC and are, therefore, acceptable. During NMCA, samples will be obtained from the temporary skid which is placed in service during the NMCA injection process.

(Deleted)

BASES FOR 3.2.7 AND 4.2.7 REACTOR COOLANT SYSTEM ISOLATION VALVES

The list of reactor coolant isolation valves is contained in the procedure governing controlled lists and have been removed from the Technical Specifications per Generic Letter 91-08. Revisions will be processed in accordance with Quality Assurance Program requirements.

Double isolation valves are provided in lines which connect to the reactor coolant system to assure isolation and minimize reactor coolant loss in the event of a line rupture. The specified valve requirements assure that isolation is already accomplished with one valve shut or provide redundancy in an open line with two operative valves. Except where check valves are used as one or both of a set of double isolation valves, the isolation valves shall be capable of automatic initiation. Valve closure times are selected to minimize coolant losses in the event of the specific line rupturing and are procedurally controlled. Using the longest closure time on the main-steam-line valves following a main-steam-line break (Section XV C.1.0)[1], the core is still covered by the time the valves close. Following a specific system line break, the cleanup and shutdown cooling closing times will upon initiation from a low-low level signal limit coolant loss such that the core is not uncovered. Feedwater flow would quickly restore coolant levels to prevent clad damage. Closure times are discussed in Section VI-D.1.0[1].

The valve operability test intervals are based on periods not likely to significantly affect operations, and are consistent with testing of other systems. Results obtained during closure testing are not expected to differ appreciably from closure times under accident conditions as in most cases, flow helps to seal the valve.

The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} (Fifth Supplement, p. 115)[2] that a line will not isolate. Additional surveillances are in accordance with the Inservice Testing Program described in Specification 6.5.4.

(1) UFSAR
(2) FSAR

BASES FOR 3.2.9 AND 4.2.9 PRESSURE RELIEF SYSTEM – SOLENOID ACTUATED PRESSURE RELIEF VALVES

As discussed in 2.2.2 and 3.2.8 above, the solenoid-actuated pressure relief valves are used to avoid actuation of the safety valves. The set points of the six relief valves are staggered. Two valves are set at 1090 psig, two are set at 1095 psig, and two are set at 1100 psig. The operator will endeavor to place the set-point at these figures. However, the Allowable Value for each valve can be as much as ± 24 psig. The as found value for at least 2 relief valves must be greater than the as found high reactor pressure scram value.

Six valves are provided for the automatic depressurization function, as described in 3.1.5. However, only five valves are required to prevent actuation of the safety valves, as discussed in the Technical Supplement to Petition to Increase Power Level, Section II.XV, letter, T.J. Brosnan to Peter A. Morris dated February 28, 1972, and letter, Philip D. Raymond to A. Giambusso, dated October 15, 1973.

The basis for the surveillance requirement is given in 4.1.5.

The list of primary containment isolation valves is contained in the procedure governing controlled lists have been removed from the Technical Specifications per Generic Letter 91-08. Revisions will be processed in accordance with Quality Assurance Program requirements.

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Except where check valves are used as one or both of a set of double isolation valves, the isolation valves shall be capable of automatic initiation. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident. Details of the isolation valves are discussed in Section VI-D.[1] For allowable leakage rate specification, see Section 3.3.3/4.3.3.

For the design basis loss-of-coolant accident fuel rod perforation would not occur until the fuel temperature reached 1700°F which occurs in approximately 100 seconds.[2] The required closing times for all primary containment isolation valves are established to prevent fission product release through lines connecting to the primary containment.

For reactor coolant system temperatures less than 215°F, the containment could not become pressurized due to a loss-of-coolant accident. The 215°F limit is based on preventing pressurization of the reactor building and rupture of the blowout panels.

The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate (Fifth Supplement, p. 115).[3] More frequent testing for valve operability results in a more reliable system.

In addition to routine surveillance as outlined in Section VI-D.1.0[1] each instrument-line flow check valve will be tested for operability. All instruments on a given line will be isolated at each instrument. The line will be purged by isolating the flow check valve, opening the bypass valves, and opening the drain valve to the equipment drain tank. When purging is sufficient to clear the line of non-condensibles and crud the flow-check valve will be cut into service and the bypass valve closed. The main valve will again be opened and the flow-check valve allowed to close. The flow-check valve will be reset by closing the drain valve and opening the bypass valve depressurizing part of the system. Instruments will be cut into service after closing the bypass valve. Repressurizing of the individual instruments assures that flow-check valves have reset to the open position.

(1) UFSAR
(2) Nine Mile Point Nuclear Generation Station Unit 1 Safer/Corecool/GESTR-LOCA Loss of Coolant Accident Analysis, NEDC-31446P, Supplement 3, September, 1990.
(3) FSAR

The emergency ventilation system is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions. Both emergency ventilation system fans are designed to automatically start upon high radiation in the reactor building ventilation duct or at the refueling platform and to maintain the reactor building pressure to the design negative pressure so as to minimize in-leakage. Should one system fail to start, the redundant system is designed to start automatically. Each of the two fans has 100 percent capacity.

High efficiency particulate absolute (HEPA) filters are installed before and after the charcoal adsorbers to minimize potential release of particulates to the environment and to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 95 percent, which is derived from applying a safety factor of 2 to the charcoal filter efficiency of 90 percent assumed in analyses of design basis accidents. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10CFR100 and General Design Criterion 19 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

Only one of the two emergency ventilation systems is needed to cleanup the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance and reactor operation or refueling operation may continue while repairs are being made. If neither circuit is operable, the plant is brought to a condition where the emergency ventilation system is not required.

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

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test should allow for charcoal sampling to be conducted using an ASTM D3803-1989 approved method. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent meeting the physical property specifications of Table 5-1 of ANSI 509-1980.

BASES FOR 3.4.5 AND 4.4.5 CONTROL ROOM AIR TREATMENT SYSTEM

The control room air treatment system is designed to filter the control room atmosphere for intake air. A roughing filter is used for recirculation flow during normal control room air treatment operation. The control room air treatment system is designed to maintain the control room pressure to the design positive pressure (one-sixteenth inch water) so that all leakage should be out leakage. The control room air treatment system starts automatically upon receipt of a LOCA (high drywell pressure or low-low reactor water level) or Main Steam Line Break (MSLB) (high steam flow main-steam line or high temperature main-steam line tunnel) signal. The system can also be manually initiated.

High efficiency particulate absolute (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorber. The charcoal adsorbers are installed to reduce the potential intake of radioiodine to the control room. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 95 percent, which is derived from applying a safety factor of 2 to the charcoal filter efficiency of 90 percent assumed in analyses of design basis accidents. If the efficiencies of the HEPA filter and charcoal adsorbers are as specified, adequate radiation protection will be provided such that resulting doses will be less than the allowable levels stated in Criterion 19 of the General Design Criteria for Nuclear Power Plants, Appendix A to 10CFR Part 50. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

If the system is found to be inoperable, there is no immediate threat to the control room and reactor operation or refueling operation may continue for a limited period of time while repairs are being made. If the makeup system cannot be repaired within seven days, the reactor is shutdown and brought to cold shutdown within 36 hours or refueling operations are terminated.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 1.5 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to show system performance capability.

The frequency of tests and sample analysis are necessary to show the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test should allow for charcoal sampling to be conducted using an ASTM D3803-1989 approved method. If test results are unacceptable, all adsorbent in the system shall be replaced with an adsorbent meeting the physical property specifications of Table 5-1 of ANSI 509-1980. The replacement charcoal for the adsorber tray removed for the test should meet the same adsorbent quality. Any HEPA filters found defective shall be replaced with filters qualified pursuant to ANSI 509-1980.

High Flow-Main Steam Line, ±1 psid

High Flow-Emergency Cooling Line, ±1 psid

High Area Temperature-Main Steam Line, ±10°F

High Area Temperature-Clean-up and Shutdown, ±6°F

High Radiation-Main Steam Line, +100% and -50% of set point value

High Radiation-Reactor Building Vent, +100% and -50% of set point

High Radiation-Refueling Platform, +100% and -50% of set point

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," and MDE-77-0485, "Technical Specification Improvement Analysis for Nine Mile Point Nuclear Station, Unit 1."

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A Suppl2, "Technical Specification Improvement Analyses for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," and with NEDC-31677P-A, "Technical Specification Improvement Analyses for BWR Isolation Actuation Instrumentation." Because of local high radiation, testing instrumentation in the area of the main steam line isolation valves can only be done during periods of Station shutdown. These functions include high area temperature isolation and isolation valve position scram.

Snubbers are required to be operable to ensure that the structural integrity of the reactor coolant system and other safety related systems is maintained during and following a seismic or other event initiating dynamic loads.

The visual inspecton frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval is based on the number of unacceptable snubbers found during the previous inspection in proportion to the population of the various snubber types and categories. The inspection schedule is based on the guidance provided in Generic Letter 90-09. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

BASES FOR 3.6.15 AND 4.6.15 MAIN CONDENSER OFFGAS

Restricting the gross radioactivity rate of noble gases from the main condenser provides assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a very small fraction of the limits of 10CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50. The primary purpose of providing this specification is to limit buildup of fission product activity within the station systems which would result if high fuel leakage were to be permitted over extended periods.

BASES FOR 3.7.1 AND 4.7.1 SHUTDOWN MARGIN DEMONSTRATION

The shutdown margin demonstration may be performed prior to power operation. However, the mode switch must be placed in the startup position to allow withdrawal of more than one control rod. Specifications 3.7.1 and 4.7.1 require certain restrictions in order to ensure that an inadvertent criticality does not occur while performing the shutdown margin demonstration.

This special test exception provides the appropriate additional controls to allow the shutdown margin demonstration to be performed in the cold shutdown condition with the vessel head in place. Compliance with this special test exception is optional and applies only if the shutdown margin demonstration will be performed prior to the reactor coolant system pressure and control rod scram time tests following refueling outages when core alterations are performed. The shutdown margin demonstration is performed using the in-sequence non-critical method.

U.S. NUCLEAR REGULATORY
COMMISSION
DOCKET 50-220
LICENSE DPR-63

NINE MILE POINT NUCLEAR STATION UNIT 1

FINAL SAFETY ANALYSIS REPORT (UPDATED)

OCTOBER 2003

REVISION 18

INSERTION INSTRUCTIONS

The following instructions are for the insertion of the current revision into the Nine Mile Point Unit 1 FSAR (Updated).

Remove pages listed in the REMOVE column and replace them with the pages listed in the INSERT column. Dashes (---) in either column indicate no action required.

Vertical bars have been placed in the margins of inserted pages and tables to indicate revision locations.

INSERTION INSTRUCTIONS

LIST OF EFFECTIVE PAGES

REMOVE	INSERT
EP i	EP i
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EP 4-1	EP 4-1
EP 5-1	EP 5-1
EP 6-1	EP 6-1
EP 7-1	EP 7-1
EP 8-1 thru EP 8-2	EP 8-1 thru EP 8-2
EP 9-1	EP 9-1
EP 10-1 thru EP 10-6	EP 10-1 thru EP 10-6
EP 11-1	EP 11-1
EP 12-1	EP 12-1
EP 13-1	EP 13-1
EP 14-1	EP 14-1
EP 15-1 thru EP 15-3	EP 15-1 thru EP 15-3
EP 16-1 thru EP 16-3	EP 16-1 thru EP 16-3
EP 17-1 thru EP 17-2	EP 17-1 thru EP 17-2
EP 18-1	EP 18-1
EP A-1	EP A-1
EP B-1	EP B-1 thru EP B-2

INSERTION INSTRUCTIONS

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xvii/xviia	xvii/xviia
xviib/xviii	xviib/xviii
xix/xx	xix/xx
xxa/xxb	xxa/xxb
xxi/xxii	xxi/xxii
xxv/xxvi	xxv/xxvi
---	xxvia/xxvib
xxxiii/xxxiv	xxxiii/xxxiv
xxxiva/xxxivb	xxxiva/xxxivb
xlvii/xlviia	xlvii/xlviia
I-1/2	I-1/2
I-7/8	I-7/8
I-15/16	I-15/16
I-19/20 thru I-21/-	I-19/20 thru I-21/-
---	T I-2 Sh 1 thru 5
II-1/2	II-1/2
II-5/6 thru II-9/-	II-5/6 thru II-9/-
T II-5/T II-6	T II-5/T II-6
III-43/44	III-43/44
F III-3	F III-3
F III-13	F III-13
F III-14	F III-14
F III-16	F III-16
IV-7/8	IV-7/8
IV-11/12 thru IV-15/16	IV-11/12 thru IV-15/16
IV-19/19a	IV-19/19a
IV-31/32	IV-31/32
---	IV-33/-
V-5/6 thru V-6a/6b	V-5/6 thru V-6a/6b
V-21/-	V-21/-
T V-3/T V-4	T V-3/T V-4

INSERTION INSTRUCTIONS

VOLUME 2

REMOVE	INSERT
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xi/xii	xi/xii
xvii/xviia	xvii/xviia
xviib/xviii	xviib/xviii
xix/xx	xix/xx
xxa/xxb	xxa/xxb
xxi/xxii	xxi/xxii
xxv/xxvi	xxv/xxvi
---	xxvia/xxvib
xxxiii/xxxiv	xxxiii/xxxiv
xxxiva/xxxivb	xxxiva/xxxivb
xlvii/xlviia	xlvii/xlviia
T VI-3a Sh 1 thru Sh 3	T VI-3a Sh 1 thru Sh 4
T VI-3b Sh 2 thru Sh 4	T VI-3b Sh 2 thru Sh 4
F VI-24	F VI-24
VII-7/8	VII-7/8
VII-11/12	VII-11/12
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VII-37b/38	---
VII-39/40 thru VII-42a/42b	VII-39/40 thru VII-42a/42b
F VII-3	F VII-3
VIII-3/4	VIII-3/4
VIII-9/9a	VIII-9/9a
VIII-13/14	VIII-13/14
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---	VIII-38a/38b
VIII-39/40 thru VIII-41/42	VIII-39/40 thru VIII-41/42
---	VIII-42a/42b
VIII-43/44 thru VIII-45/46	VIII-43/44 thru VIII-45/46
VIII-49/50	VIII-49/50
VIII-55/-	VIII-55/-
T VIII-3 Sh 3	T VIII-3 Sh 3
T VIII-3 Sh 7/8 thru Sh 9/10	T VIII-3 Sh 7/8 thru Sh 9/10
---	T VIII-3 Sh 10a/-
F VIII-2	F VIII-2
F VIII-28	F VIII-28
IX-3/4	IX-3/4
IX-17/17a	IX-17/17a
IX-25/26 thru IX-27/28	IX-25/26 thru IX-27/28
F IX-1	F IX-1

INSERTION INSTRUCTIONS

VOLUME 3

REMOVE	INSERT
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xi/xii	xi/xii
xvii/xviia	xvii/xviia
xviib/xviii	xviib/xviii
xix/xx	xix/xx
xxa/xxb	xxa/xxb
xxi/xxii	xxi/xxii
xxv/xxvi	xxv/xxvi
---	xxvia/xxvib
xxxiii/xxxiv	xxxiii/xxxiv
xxxiva/xxxivb	xxxiva/xxxivb
xlvii/xlviia	xlvii/xlviia
X-5b/6 thru X-7/8	X-5b/6 thru X-7/8
X-29/30	X-29/30
X-47/48	X-47/48
10A-1/2	10A-1/2
10A-7/8	10A-7/8
---	10A-8a/8b
10A-9/10 thru 10A-11/12	10A-9/10 thru 10A-11/12
10A-17/18	10A-17/18
---	10A-18a/18b
10A-21/22	10A-21/22
10A-29/30 thru 10A-31/32	10A-29/30 thru 10A-31/32
10A-45/46	10A-45/46
10A-49/50 thru 10A-53/54	10A-49/50 thru 10A-53/54
10A-63/64	10A-63/64
10A-81/82	10A-81/82
10A-85/- thru 10A-89/-	10A-85/- thru 10A-89/-
---	10A-89a/-
10A-104/- thru 10A-108/-	10A-104/- thru 10A-108/-
---	10A-108a/- thru 10A-108b/-
10A-109/- thru 10A-114/-	10A-109/- thru 10A-114/-
---	10A-114a/- thru 10A-114b/-
10A-115/- thru 10A-123/-	10A-115/- thru 10A-123/-
---	10A-123a/-
10A-124/- thru 10A-127/-	10A-124/- thru 10A-127/-
---	10A-127a/-
F 10A-3 thru F 10A-3D	F 10A-3 thru F 10A-3D
10B-i/ii thru 10B-iii/-	10B-i/ii thru 10B-iii/-
10B-1/2	10B-1/2
10B-9/10 thru 10B-17/18	10B-9/10 thru 10B-17/18
---	10B-18a/18b
10B-37/38	10B-37/38

INSERTION INSTRUCTIONS

VOLUME 3 (Cont'd.)

REMOVE	INSERT
10B-42/- thru 10B-44/-	10B-42/- thru 10B-44/-
10B-47/- thru 10B-48/-	10B-47/- thru 10B-48/-
10B-53/54 thru 10B-57/-	10B-53/54 thru 10B-57/-
---	10B-57a/-
10B-58/-	10B-58/-
10B-63/64 thru 10B-65/-	10B-63/64 thru 10B-65/-
10B-79/-	10B-79/-
10B-111/-	10B-111/-
10B-115/116	10B-115/116
10B-194/- thru 10B-195/196	10B-194/- thru 10B-195/196
10B-215/216 thru 10B-217/218	10B-215/216 thru 10B-217/218
XI-7/8 thru XI-11/11a	XI-7/8 thru XI-11/11a
F XI-3	F XI-3
F XI-7	F XI-7
XII-1/2	XII-1/2
XII-9/10	XII-9/9a
---	XII-9b/10
XII-17/18 thru XII-19/20	XII-17/18 thru XII-19/20
XII-23/24 thru XII-27/28	XII-23/24 thru XII-27/28
XIII-1/2 thru XIII-9/10	XIII-1/2 thru XIII-9/9a
---	XIII-9b/10
XIII-11/12 thru XIII-19/20	XIII-11/12 thru XIII-19/20
T XIII-1/-	T XIII-1/-
F XIII-1 thru F XIII-4	F XIII-1 thru F XIII-4
---	F XIII-4a thru F XIII-4c
F XIII-5	F XIII-5

INSERTION INSTRUCTIONS

VOLUME 4

REMOVE	INSERT
xa/xb	xa/xb
xi/xii	xi/xii
xvii/xviia	xvii/xviia
xviib/xviii	xviib/xviii
xix/xx	xix/xx
xxa/xxb	xxa/xxb
xxi/xxii	xxi/xxii
xxv/xxvi	xxv/xxvi
---	xxvia/xxvib
xxxiii/xxxiv	xxxiii/xxxiv
xxxiva/xxxivb	xxxiva/xxxivb
xlvii/xlviia	xlvii/xlviia
XIV-13/-	XIV-13/14
XV-3/4 thru XV-7/8	XV-3/4 thru XV-7/8
XV-15/16	XV-15/16
XV-35/36	XV-35/36
XV-47/48	XV-47/48
XV-59/60 thru XV-61/62	XV-59/60 thru XV-61b/62
XV-67/68	XV-67/68
XV-79/80 thru XV-81/82	XV-79/80 thru XV-81/82
XVI-19/20	XVI-19/20
XVI-51/52	XVI-51/52
XVI-55b/56	XVI-55b/56
T XVI-2 Sh 1	T XVI-2 Sh 1
T XVI-7/-	T XVI-7/-
XVII-33/34	XVII-33/34
XVIII-1/2	XVIII-1/2
XVIII-5/6	XVIII-5/6
XVIII-9/10 thru XVIII-11/12	XVIII-9/10 thru XVIII-11/12
T XVIII-1/-	T XVIII-1/-
B-i/ii	B-i/ii
B.0-1/2	B.0-1/2
---	B.0-3/-
B.1-1/2 thru B.1-5/-	B.1-1/2 thru B.1-5/-
B.2-1/2 thru B.2-11/12	B.2-1/2 thru B.2-11/12
---	B.2-13/-

INSERTION INSTRUCTIONS

VOLUME 4 (Cont'd.)

REMOVE	INSERT
B.3-1/2	B.3-1/2
B.5-1/2 thru B.5-5/-	B.5-1/2 thru B.5-5/-
B.7-1/2 thru B.7-3/4	B.7-1/2 thru B.7-3/4
B.9-1/2 thru B.9-3/-	B.9-1/2 thru B.9-3/-
B.15-1/2	B.15-1/2
B.17-1/2 thru B.17-3/-	B.17-1/2 thru B.17-3/-
B.18-1/2	B.18-1/2
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xv	17	li	16
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xviia	18	liv	16
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NINE MILE POINT NUCLEAR STATION UNIT 1

FINAL SAFETY ANALYSIS REPORT (UPDATED)

VOLUME 1

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SECTION I

INTRODUCTION AND SUMMARY

This report is submitted in accordance with 10 CFR Part 50.71(e) entitled "Periodic Updating of Final Safety Analysis Reports" for Nine Mile Point Nuclear Station - Unit 1 (Unit 1). The Station is located on the southeast shore of Lake Ontario, in Oswego County, New York, 7 mi northeast of the city of Oswego.

The operating license (OL) was transferred to Nine Mile Point Nuclear Station, LLC (NMPNS), on November 7, 2001, under License Amendment No. 172.

A. PRINCIPAL DESIGN CRITERIA

The following paragraphs describing the principal design criteria are oriented toward the twenty-seven criteria issued by the United States Atomic Energy Commission (USAEC) on November 22, 1965.[1] The twenty-seven criteria represented proposed "General Design Criteria for Nuclear Power Plant Construction Permits." The twenty-seven criteria are presented here for historical reference and are followed by the Unit 1 principal design criteria.

Table I-1 provides historical information regarding an assessment of Unit 1 against criteria that were being used by the USAEC at the time of the Unit 1 application for a full-term OL.

Facility

Criterion 1

Those features of reactor facilities which are essential to the prevention of accidents or to the mitigation of their consequences must be designed, fabricated, and erected to:

(a) Quality standards that reflect the importance of the safety function to be performed. It should be recognized, in this respect, that design codes commonly used for nonnuclear applications may not be adequate.

(b) Performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces imposed by the most severe earthquakes, flooding conditions, winds, ice, and other natural phenomena anticipated at the proposed site.

Criterion 2

Provisions must be included to limit the extent and the consequences of credible chemical reactions that could cause or materially augment the release of significant amounts of fission products from the facility.

Criterion 3

Protection must be provided against possibilities for damage of the safeguarding features of the facility by missiles generated through equipment failures inside the containment.

Radioactivity Control

Criterion 24

All fuel storage and waste handling systems must be contained if necessary to prevent the accidental release of radioactivity in amounts which could affect the health and safety of the public.

Criterion 25

The fuel handling and storage facilities must be designed to prevent criticality and to maintain adequate shielding and cooling for spent fuel under all anticipated normal and abnormal conditions, and credible accident conditions. Variables upon which health and safety of the public depend must be monitored.

Criterion 26

Where unfavorable environmental conditions can be expected to require limitations upon the release of operational radioactive effluents to the environment, appropriate holdup capacity must be provided for retention of gaseous, liquid, or solid effluents.

Criterion 27

The plant must be provided with systems capable of monitoring the release of radioactivity under accident conditions.

1.0 General

The Station is intended as a high load factor generating facility. The recirculation flow control system described in Section VIII contributes to this objective by providing a relatively fast means for adjusting the Station output over a preselected power range. Overall reliability, routine and periodic test requirements, and other design considerations must also be compatible with this objective.

Careful attention has been given to fabrication procedures and adherence to Code requirements. The rigid requirements of specific portions of various codes have been arbitrarily applied to some safety-related systems to ensure quality construction in such cases where the complete Code does not apply.

For piping, the ASA B31.1-1955 Code was used and where exceptions were taken, safety evaluations were performed to document that an adequate margin of safety was maintained.

Periodic test programs have been developed for required engineered safeguards equipment. These tests cover component testing such as pumps and valves and full system tests, duplicating as closely as possible the accident conditions under which a given system must perform.

2.0 Buildings and Structures

The Station plot plan, design and arrangement of the various buildings and structures are described in Section III. Principal structures and equipment which may serve either to prevent accidents or to mitigate their consequences are designed, fabricated and erected in accordance with applicable codes to withstand the most severe earthquake, flooding condition, windstorm, ice condition, temperature and other deleterious natural phenomena which can be expected to occur at the site.

3.0 Reactor

1. A direct-cycle boiling water system reactor (BWR), described in Section IV, is employed to produce steam (1030 psig in reactor vessel, 950 psig turbine inlet) for use in a steam-driven turbine generator. The rated thermal output of the reactor is 1850 MWt.

2. The reactor is fueled with slightly enriched uranium dioxide contained in Zircaloy clad fuel rods described in Section IV. Selected fuel rods also incorporate small amounts of gadolinium as burnable poison.

3. To avoid fuel damage, the minimum critical power ratio (MCPR) is maintained greater than or equal to the safety limit CPR.

4. The fuel rod cladding is designed to maintain its integrity throughout the anticipated fuel life as described in Section IV. Fission gas release within the rods and other factors affecting design life are considered for the maximum expected burnup.

5. The reactor and associated systems are designed so that there is no inherent tendency for undamped oscillations. A stability analysis evaluation is given in Section IV.

6. Heat removal systems are provided which are capable of safely accommodating core decay heat under all credible circumstances, including isolation from the main condenser and loss of coolant from the reactor. Each different system so provided has appropriate redundant features.

   Independent auxiliary cooling means are provided to cool the reactor under a variety of conditions. The

B. CHARACTERISTICS

The following is a summary of design and operating characteristics.

1.0 Site

Location	Oswego County, New York State
Size of Site	900 Acres
Site and Station Ownership	Nine Mile Point Nuclear Station, LLC (NMPNS)
Net Electrical Output	615 MW (Maximum)

2.0 Reactor

Reference Rated Thermal Output	1850 MW
Dome Pressure	1030 psig
Turbine Inlet Pressure	950 psig
Total Core Coolant Flow Rate	67.5×10^6 lb/hr
Steam Flow Rate	7.32×10^6 lb/hr

3.0 Core

Circumscribed Core Diameter	167.16 in
Active Core Height + Assembly	171.125 in

4.0 Fuel Assembly

Number of Fuel Assemblies	532
Fuel Rod Array	SRLR[2]
Fuel Rod Pitch	Reference 3
Cladding Material	Reference 3
Fuel Material	UO_2 and UO_2-Gd_2O3
Active Fuel Length	Reference 3
Cladding Outside Diameter	Reference 3
Cladding Thickness	Reference 3
Fuel Channel Material	Reference 3

5.0 Control System

Number of Movable Control Rods	129
Shape of Movable Control Rods	Cruciform
Pitch of Movable Control Rods	12.0 in
Control Material in Movable Control Rods	B_4C - 70% Theoretical Density; Hafnium
Type of Control Drives	Bottom Entry, Hydraulic Actuated

Control of Reactor Output	Movement of Control Rods and Variation of Coolant Flow Rate

6.0 Core Design and Operating Conditions

Maximum Linear Heat Generation Rate	Core Operating Limits Report
Heat Transfer Surface Area	*
Average Heat Flux - Rated Power	*
Minimum Critical Power Ratio for Most Limiting Transients	Core Operating Limits Report
Core Average Void Fraction - Coolant within Assemblies	*
Core Average Exit Quality - Coolant within Assemblies	*

7.0 Design Power Peaking Factor

Total Peaking Factor	GE11	- 2.94**
		- 2.62***

8.0 Nuclear Design Data

Average Initial Volume Metric Enrichment	Reference 3
Beginning of Cycle - Core Effective Multiplication and Control System Worth - No Voids, 20C(2)	
Uncontrolled	SRLR(2)
Fully Controlled	SRLR(2)
Strongest Control Rod Out	SRLR(2)

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* These parameters are recalculated for each reload because of their dependency on core composition and exposure. These calculated values are intermediate quantities that do not represent design requirements or operating limits and thus are not separately reported in the SRLR(2).

** Maximum total peaking factor for the portion of the bundle containing part length rods.

*** Maximum total peaking factor for the region above the part length rods.

C. IDENTIFICATION OF CONTRACTORS

The General Electric Company (GE) was engaged to design, fabricate and deliver the nuclear steam supply system (NSSS), turbine generator, and other major elements and systems. GE also furnished the complete core design and nuclear fuel supply for the initial core. Global Nuclear Fuel (GNF) is currently furnishing replacement cores.

Niagara Mohawk Power Corporation (NMPC), acting as the architect-engineer, specified and procured the remaining systems and components, including the pressure suppression containment system, and coordinated the complete integrated Station. Stone and Webster Engineering Corporation (SWEC) was engaged to manage field construction. Currently, various contractors are utilized to assist in continuous Station modifications.

D. GENERAL CONCLUSIONS

The favorable site characteristics, criteria and design requirements of all the systems related to safety, the potential consequences of postulated accidents, and the technical competence of the applicant and its contractors, assure that Unit 1 can be operated without endangering the health and safety of the public.

E. REFERENCES

1. USAEC Press Release H-252, "General Design Criteria for Nuclear Power Plant Construction Permits," November 22, 1965.

2. 0000-0005-1174SRLR, Revision 0, "Supplemental Reload Licensing Report for NMP1, Reload 17, Cycle 16," January 2003.

3. GE Fuel Bundle Designs, General Electric Company Proprietary, NEDE-31152P, Revision 5, June 1996.

TABLE I-2

ABBREVIATIONS AND ACRONYMS USED IN UFSAR

ACI	American Concrete Institute
ADS	Automatic depressurization system
AISC	American Institute of Steel Construction
ALARA	As low as reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOV	Air-operated valve
APRM	Average power range monitor
ARI	Alternate rod injection
ARMS	Area radiation monitoring system
ASSS	Assistant Station Shift Supervisor
ASTM	American Society for Testing and Materials
ATWS	Anticipated transient without scram
BOC	Beginning of cycle
BOP	Balance of plant
BPWS	Banked position withdrawal sequence
BTP	Branch technical position
BWR	Boiling water reactor
BWROG	Boiling Water Reactor Owners' Group
CAD	Containment atmosphere dilution (device)
CEO	Chief Executive Officer
CFR	Code of Federal Regulations
CGCS	Combustible gas control system
CHF	Critical heat flux
CIV	Combined intermediate valve
CND	Condensate demineralizer
CO2	Carbon dioxide
COLR	Core Operating Limits Report
CPR	Critical power ratio
CRD	Control rod drive
CRDA	Control rod drop accident
CRPI	Control rod position indication
CRT	Cathode ray tube
CSO	Chief Shift Operator
CST	Condensate storage tank
DAC	Dominant area of concern
DBA	Design basis accident
DBE	Design basis earthquake
DCRDR	Detailed control room design review
DEC	Department of Environmental Conservation
DER	Deviation/Event Report
DER	Double-ended rupture
DG	Diesel generator
DOP	Dioctylphthalate
DOT	Department of Transportation

TABLE I-2 (Cont'd.)

ECCS	Emergency core cooling system
ECP	Electrochemical corrosion potential
EDG	Emergency diesel generator
EFPY	Effective full-power years
EIC	Energy Information Center
EOC	End of cycle
EOF	Emergency Operations Facility
EOL	End of life
EOP	Emergency operating procedure
EPA	Environmental Protection Agency
EPDM	Ethylene-propylene-diene-monomer
EPG	Emergency procedure guideline
EPRI	Electric Power Research Institute
ESF	Engineered safety feature
ESW	Emergency service water
FA	Fire area
FCV	Flow control valve
FHA	Fire Hazards Analysis
FMEA	Failure modes and effects analysis
FSA	Fire subarea
FSAR	Final Safety Analysis Report
FZ	Fire zone
GDC	General Design Criterion
GE	General Electric Company
GL	Generic Letter
HAZ	Heat-affected zone
HCU	Hydraulic control unit
HEM	Homogeneous equilibrium model
HEO	Human engineering observation
HEPA	High-efficiency particulate air/absolute (filter)
HPCI	High-pressure coolant injection
HVAC	Heating, ventilating, and air conditioning
HWC	Hydrogen water chemistry
HX	Heat exchanger
I&C	Instrumentation & control
ID	Inner diameter
IGSCC	Intergranular stress corrosion cracking
ILRT	Integrated leakage rate test
INPO	Institute of Nuclear Power Operations
ISEG	Independent Safety Engineering Group
ISI	Inservice inspection
IST	Inservice testing
LCO	Limiting condition of operation
LHGR	Linear heat generation rate
LLD	Lower limit of detection

TABLE I-2 (Cont'd.)

LLL	Low-low limit
LOCA	Loss-of-coolant accident
LOFW	Loss of feedwater
LOOP	Loss of offsite power
LPCS	Low-pressure core spray
LPRM	Local power range monitor
LPSP	Low power setpoint
LPZ	Low population zone
LSSS	Limiting safety system setting
M&TE	Measuring and testing equipment
MAPLHGR	Maximum average planar linear heat generation rate
MCC	Motor control center
MCPR	Minimum critical power ratio
MG	Motor generator
MLHGR	Maximum linear heat generation rate
MOV	Motor-operated valve
MSIV	Main steam isolation valve
MSL	Main steam line
MSLB	Main steam line break
NDT	Nil ductility transition
NDT	Nondestructive testing
NDTT	Nil ductility transition temperature
NFPA	National Fire Protection Association
NMPC	Niagara Mohawk Power Corporation
NPSH	Net positive suction head
NRC	Nuclear Regulatory Commission
NRV	Nonreturn valve
NSSS	Nuclear steam supply system
NVLAP	National Voluntary Laboratory Accreditation Program
NYPA	New York Power Authority
NYPP	New York Power Pool
OBE	Operating basis earthquake
OEA	Operating experience assessment
OL	Operating license
OOS	Out of service
OSC	Operational Support Center
OT	Operational transient
PA	Public address (system)
PASS	Post-accident sampling system
PCI	Pellet-cladding interaction
PCT	Peak cladding temperature
p.f.	Power factor
P&ID	Piping and instrumentation diagram
PP/PA	Page party/public address (system)
PSAR	Preliminary Safety Analysis Report

TABLE I-2 (Cont'd.)

PSTG	Plant-specific technical guideline
P-T	Pressure-temperature
PVC	Polyvinyl chloride
QA	Quality assurance
QATR	Quality Assurance Topical Report
RBCLCW	Reactor building closed loop cooling water
RBM	Rod block monitor
RCA	Radiologically-controlled area
RCPB	Reactor coolant pressure boundary
RCS	Reactor coolant system
RG	Regulatory Guide
RMS	Radiation monitoring system
RO	Reactor Operator
RPIS	Rod position information system
RPS	Reactor protection (trip) system
RPT	Recirculation pump trip
RPV	Reactor pressure vessel
RSP	Remote shutdown panel
RSS	Remote shutdown system
RTD	Resistance temperature detector
RTNDT	Reference temperature nil ductility transition
RWCU	Reactor water cleanup
RWE	Rod withdrawal error
RWM	Rod worth minimizer
RWP	Radiation work permit
SAP	Severe accident procedure
SAR	Safety analysis report
SAS	Secondary alarm system
SBO	Station blackout
SCBA	Self-contained breathing apparatus
SDM	Shutdown margin
SDV	Scram discharge volume
SER	Safety Evaluation Report
SFC	Spent fuel pool cooling and cleanup
SIL	Service Information Letter
SJAE	Steam jet air ejector
SOE	Sequence of events
SOP	Special operating procedure
SORC	Station Operations Review Committee
SOV	Solenoid-operated valve
SPDS	Safety parameter display system
SRAB	Safety Review and Audit Board
SRLR	Supplemental Reload Licensing Report
SRM	Source range monitor
SRO	Senior Reactor Operator
SRP	Standard Review Plan
SRV	Safety/relief valve

TABLE I-2 (Cont'd.)

SRVDL	Safety/relief valve discharge line
SSA	Safe Shutdown Analysis
SSS	Station Shift Supervisor
SWEC	Stone & Webster Engineering Corporation
SWP	Service water system
TAF	Top of active fuel
TBCLCW	Turbine building closed loop cooling water
TCV	Turbine control valve
TDH	Total developed head
TIP	Traversing in-core probe
TLD	Thermoluminescence dosimeter
TMI	Three Mile Island
TSC	Technical Support Center
TSVC	Turbine stop valve closure
TVD	Test, vent and drain
UBC	Uniform Building Code
UHS	Ultimate heat sink
UL	Underwriters' Laboratories Inc.
Unit 1	Nine Mile Point Nuclear Station - Unit 1
Unit 2	Nine Mile Point Nuclear Station - Unit 2
UPS	Uninterruptible power supply
URC	Ultrasonic resin cleaning
U.S.	United States
USBM	U.S. Bureau of Mines
USLS	U.S. Land Survey
UT	Ultrasonic testing
VWO	Valve wide open

SECTION II

STATION SITE AND ENVIRONMENT

A. SITE DESCRIPTION

1.0 General

The Nine Mile Point Nuclear Station - Unit 1 (Unit 1), owned by Nine Mile Point Nuclear Station, LLC (NMPNS), is located on the western portion of the Nine Mile Point promontory. Approximately 300 ft due east is Nine Mile Point Nuclear Station - Unit 2 (Unit 2). The eastern portion of the promontory is comprised of the James A. FitzPatrick Nuclear Power Plant, owned by Entergy Nuclear FitzPatrick, LLC.

The site is on Lake Ontario in Oswego County, approximately 5 mi north-northeast of the nearest boundary of the city of Oswego.

Figure II-1 shows the Station location on an outline map of the state of New York. It is 230 mi northwest of New York City, 143.5 mi east-northeast of Buffalo, and 36 mi north-northwest of Syracuse. Figure II-2 is a detailed map of the area within about 50 mi of the Station.

2.0 Physical Features

Figure II-3 is a detailed site map showing Station location; an associated plot plan is presented as Figure III-1 of the following section. Station buildings are situated in the western quadrant of a 200-acre cleared area centrally located along the lakeshore. Site property consists of partially-wooded land formerly used almost exclusively for residential and recreational purposes. For many miles west, east, and south of the site the country is characterized by rolling terrain rising gently up from the lake.

Grade elevation at the site is 10 ft above the record high lake level, while underlying rock structure is among the most structurally stable in the United States (U.S.) from the standpoint of tilting and folding. There is no record of wave activity, such as seiche or tsunami, of such a magnitude as to make inundation of the site likely. A shore protection dike composed of rock fill from the excavation separates the buildings and the lake.

All elevations in this report refer to the United States Land Survey (USLS) 1935 data.

1. To convert elevations to 1955 International Great Lakes Data (IGLD 1955), subtract 0.375m (1.23 ft).

2. To convert elevations to 1985 International Great Lakes Data (IGLD 1985), subtract 0.217m (0.71 ft).

Exclusion distances for the site are approximately 1 mi to the east, a mile to the southwest, and over a mile to the southern site boundary.

3.0 Property Use and Development

There are no residences, agricultural or industrial developments (other than the James A. FitzPatrick Nuclear Power Plant) on the site; all former summer homes and farm buildings have been removed. Site boundaries and the former country road which traverses the site are posted as private property. The area immediately around the Station buildings is fenced, with building access controlled by Station security personnel.

A visitors' Energy Information Center, manned by NMPNS and Entergy Nuclear Operations personnel, and the Nuclear Learning Center are located about 1,000 ft west of the Station, per Figure II-3. These installations may be reached by the public over private drives maintained by the company.

C. METEOROLOGY

An original 2-yr study was performed to determine the site meteorological characteristics. This study is presented in Section XVII-A.

The meteorological monitoring system measures parameters to provide data that are representative of atmospheric conditions that exist at all gaseous effluent release points. Meteorological data is compiled for quarterly periods in accordance with the Offsite Dose Calculation Manual. This data is used to provide information which may be used to develop atmospheric diffusion parameters to estimate potential radiation doses to the public resulting from actual routine or accidental releases of radioactive materials to the atmosphere.

D. LIMNOLOGY

A comprehensive research program, designed to monitor various parameters of the aquatic environment in the vicinity of Nine Mile Point, was begun in 1963. This detailed lake program was continued through 1978.

Currently, an aquatic ecology study program (closely coordinated with James A. FitzPatrick Nuclear Power Plant) is conducted in the vicinity of Nine Mile Point on Lake Ontario to monitor the effects of plant operation with respect to selected ecological parameters, and to perform impingement studies on the traveling screens in the intake screenwell. This program is carried out and results reported in accordance with the station State Pollutant Discharge Elimination System (SPDES) Discharge Permit.

E. EARTH SCIENCES

A preconstruction evaluation of the geology, hydrology, and seismology of the Nine Mile Point promontory is presented in Section XVII-C.

Subsequent inspection of rock exposed during excavations for the reactor and cooling water tunnels allowed for a more detailed study of subsurface conditions. No faults were encountered and no unusual conditions were observed. The structures rest on a firm, almost impervious rock foundation.

Station seismic design criteria were based upon a conservative evaluation of the maximum earthquake ground motion which might conceivably occur at the site. This condition was calculated by assuming that the worst shock ever observed within an effective range of the site might be located at the closest position to the site at which an earthquake of any intensity occurred. The "maximum possible" shock assumed for Station structure acceleration calculations is of magnitude 7 at a 50-mi epicentral distance. Dames and Moore estimates that this shock will probably never occur unless unusual regional geologic changes take place.

F. ENVIRONMENTAL RADIOLOGY

Controlled releases of radioactive materials in liquid and gaseous effluents to the environment is part of normal Station operation. A Radiological Environmental Monitoring Program ensures that the release rates for all effluents are within the limits specified in 10CFR20 and the release of radioactive material above background to unrestricted areas conforms with Appendix I to 10CFR50.

Comprehensive studies were originally conducted to establish the effluent emission rates which would produce the above limiting conditions in the uncontrolled environment.

Currently, a Radiological Environmental Monitoring Program[1], inclusive of Unit 1, is in operation. This program details the design objectives for control of liquid and gaseous wastes, including specifications for liquid and gaseous waste effluents, and specifications for liquid and gaseous waste sampling and monitoring. An annual Environmental Operating Report and Radioactive Effluent Release Report are prepared and submitted in accordance with the reporting requirements in the Technical Specifications.

G. REFERENCES

1. Nine Mile Point Nuclear Station "Offsite Dose Calculation Manual."

TABLE II-5

INDUSTRIAL FIRMS WITHIN 8 KM (5 MI) OF UNIT 1

Firm	Distance/ Direction from Site (km)	Products	Employment
Alcan Aluminum Corporation	4.5/SW	Aluminum sheet and plate	1,000
James A. FitzPatrick Nuclear Power Plant	<1/E	Electrical generation	500
Nine Mile Point Unit 2	Adjacent to Unit 1	Electrical generation	1,100
Sithe Energies USA Independence Generation Plant	3.5/SW	Electrical generation	75

NOTE: For complete listing of major industries in Oswego County, reference Oswego County Industrial Directory.

TABLE II-6

PUBLIC UTILITIES IN OSWEGO COUNTY

	Location	Service
Niagara Mohawk Power Corporation	Many sites	Gas
New York Telephone Company	Many sites	Communications
Penn Central Railroad	--	Shipping
Oswego County Telephone Company	Oswego	Communications
Alltel New York, Inc.	Fulton	Communications
New York Power Authority	Many sites	Gas and Electric

nonseismic, nonnuclear safety-related systems and components. Instrumentation and control systems are provided to achieve required space temperature conditions and to maintain air flow requirements to provide acceptable building and process area pressure relationships. Relative humidity is not controlled, although it is maintained at reasonable levels by the HVAC system. All operating control functions are automatic. Temperature control systems in the fresh air supply and recirculating atmospheric cleanup systems are independent. Air flow control systems in the fresh air supply system and the exhaust ventilation system include interlock provisions to maintain pressure relationships upon de-energizing an exhaust or supply fan. Air flow controls of the recirculating atmospheric cleanup system are independent of the other systems. Redundant temperature sensing and control loops are provided in the fresh air supply and recirculating atmospheric cleanup system. Local instruments and remote indication and/or annunciation are provided.

2.3 Shielding and Access Control[3]

The RSSB is designed to minimize exposure to plant personnel and the public by its location and design. The RSSB is located within the protected area and is heavily shielded by reinforced concrete.

3.0 Use

The RSSB was constructed with the specific intent of providing onsite storage of low-level radioactive waste (LLW). The need to store LLW onsite is the result of the federal Low-Level Radioactive Waste Policy Act as amended in 1985, which initiated the process by which the three existing LLW disposal sites (Barnwell, SC; Beatty, NV; and Hanford, WA) would no longer be required to receive LLW. Although originally designed to store Unit 1 LLW, the RSSB is capable of providing interim storage of LLW produced at both Unit 1 and Unit 2. From a technical standpoint, the storage of Unit 2 waste at Unit 1 is considered acceptable based on the following:

1. The isotopic distributions of the waste stored in the RSSB from the two units are similar and expected to remain similar as both units have applied noble metals, inject depleted zinc, and inject low levels of hydrogen.

2. The selective storage of the high-activity LLW from both units in the RSSB (and the low-activity LLW at

Unit 2) creates the potential for the storage of greater average activity concentration in the building, although not greater volume. However, since the RSSB was designed assuming the storage of incinerated resins which represent a bounding activity concentration, the building design is considered adequate for the combined storage from both units;

3. Total activity in the RSSB will ultimately be controlled per the Site radiation protection program to ensure that both onsite and offsite dose and dose rate limits are maintained; and

4. The transfer of by-product material between Unit 1 and Unit 2 will be conducted in accordance with approved radiation protection implementing procedures.

Radioactive piping is routed through a shielded pipe tunnel and in shielded areas to limit exposure. Major pieces of equipment that can be significant sources of radiation exposure are each provided with a separate shielded cubicle. The storage vaults are shielded with 48 in of concrete in the storage zone (below crane). The roof is 24-in thick. The tank cubicles are shielded by 36 in of concrete. The east-west truck bay is equipped with a retracting shield door in the ceiling which mitigates albedo radiation in the truck bay from the storage vaults. The low-level storage room and the process equipment cubicle are equipped with sliding shield doors.

Access is controlled administratively by the Unit 1 Radiation Protection Program. Physical control of high radiation areas is maintained in accordance with Technical Specifications.

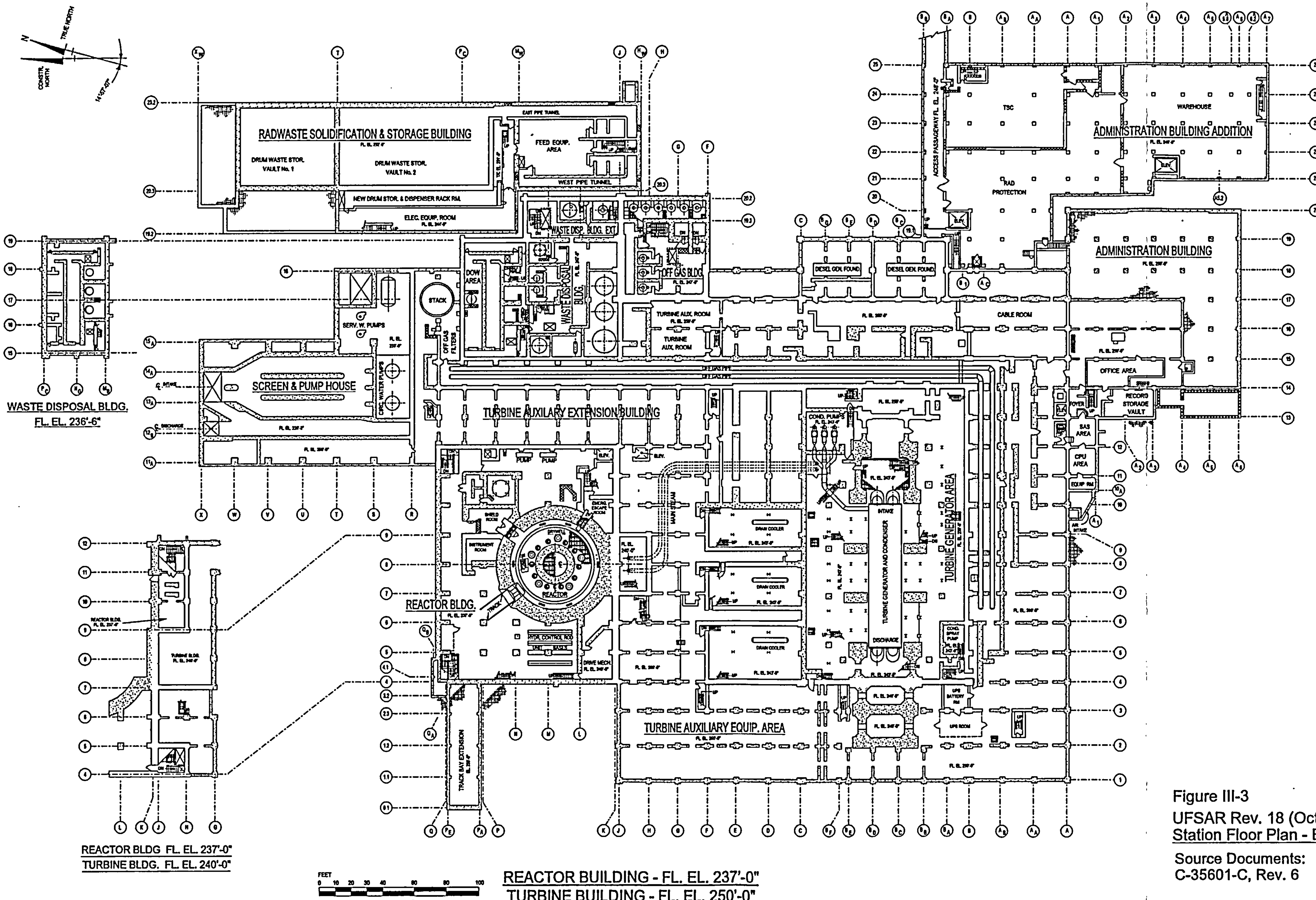


Figure III-3

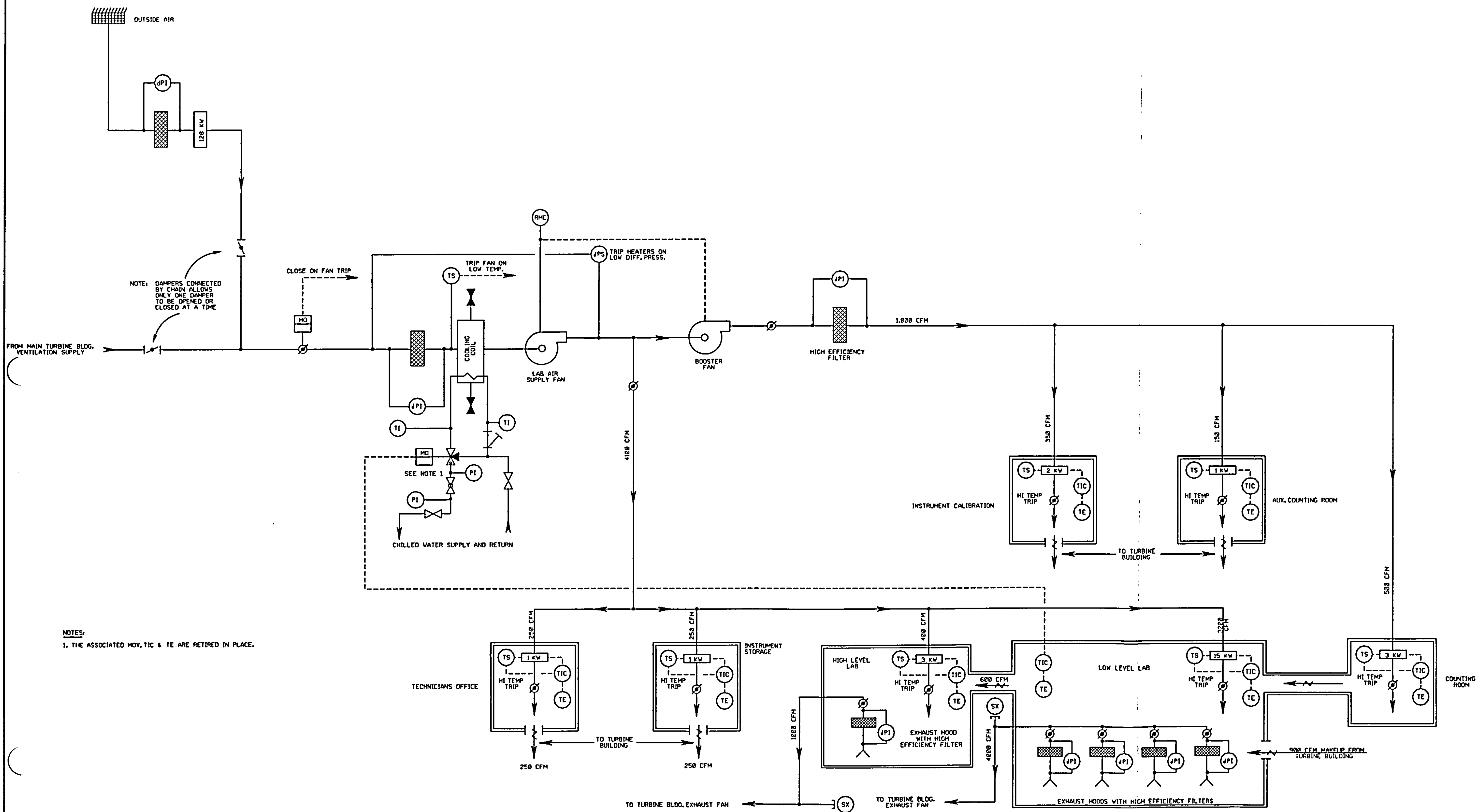
UFSAR Rev. 18 (October 2003)

Station Floor Plan - El. 237'-0" & 250'-0"

Source Documents:

C-35601-C, Rev. 6

LABORATORY AND RADIATION PROTECTION FACILITY VENTILATION SYSTEM

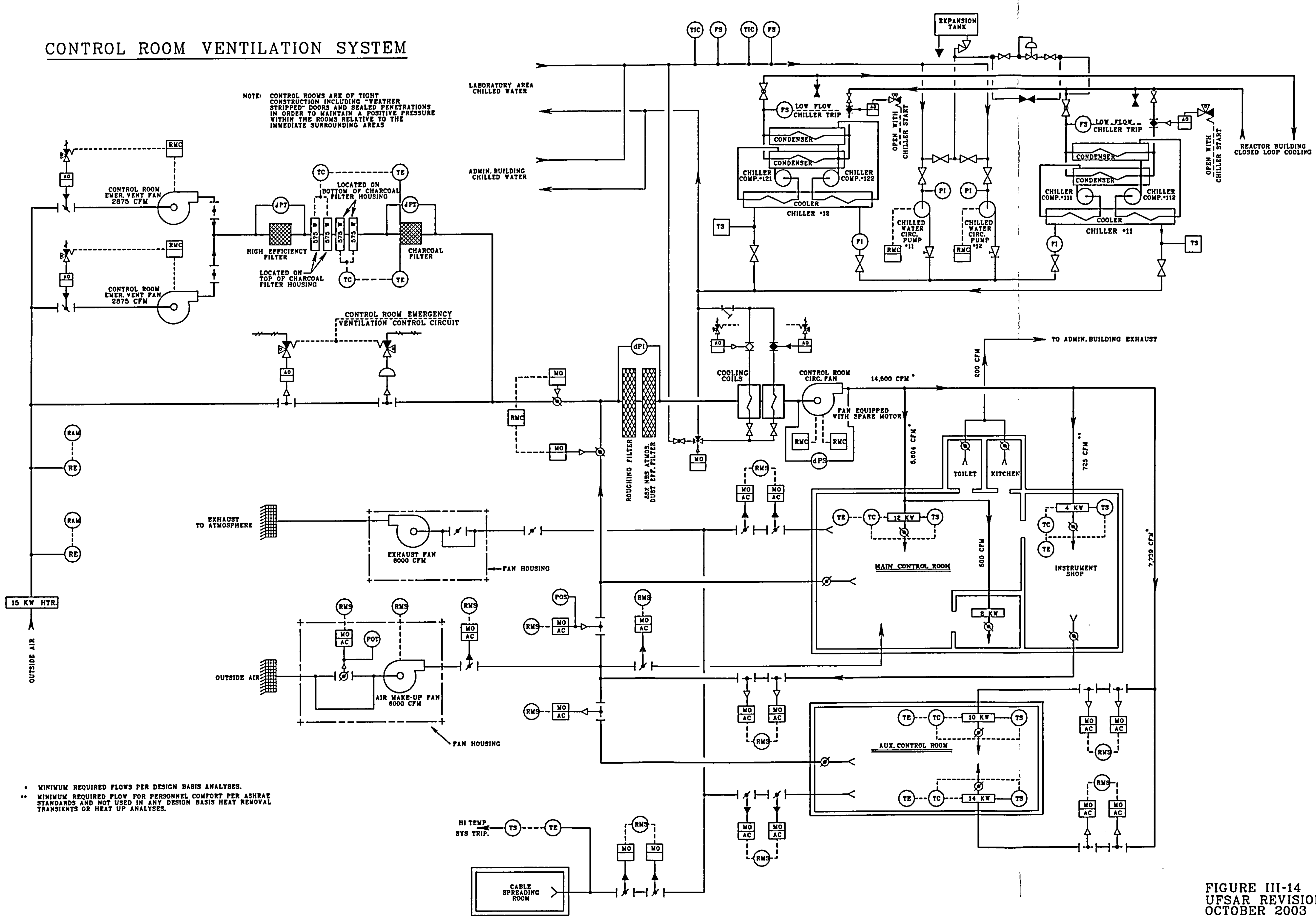


NOTES:

1. THE ASSOCIATED MOV, TIC & TE ARE RETIRED IN PLACE.

FIGURE III-13
UFSAR REVISION 18
OCTOBER 2003

CONTROL ROOM VENTILATION SYSTEM



* MINIMUM REQUIRED FLOWS PER DESIGN BASIS ANALYSES.

** MINIMUM REQUIRED FLOW FOR PERSONNEL COMFORT PER ASHRAE STANDARDS AND NOT USED IN ANY DESIGN BASIS HEAT REMOVAL TRANSIENTS OR HEAT UP ANALYSES.

FIGURE III-14
UFSAR REVISION 18
OCTOBER 2003

WASTE DISPOSAL BUILDING EXTENSION
VENTILATION SYSTEM

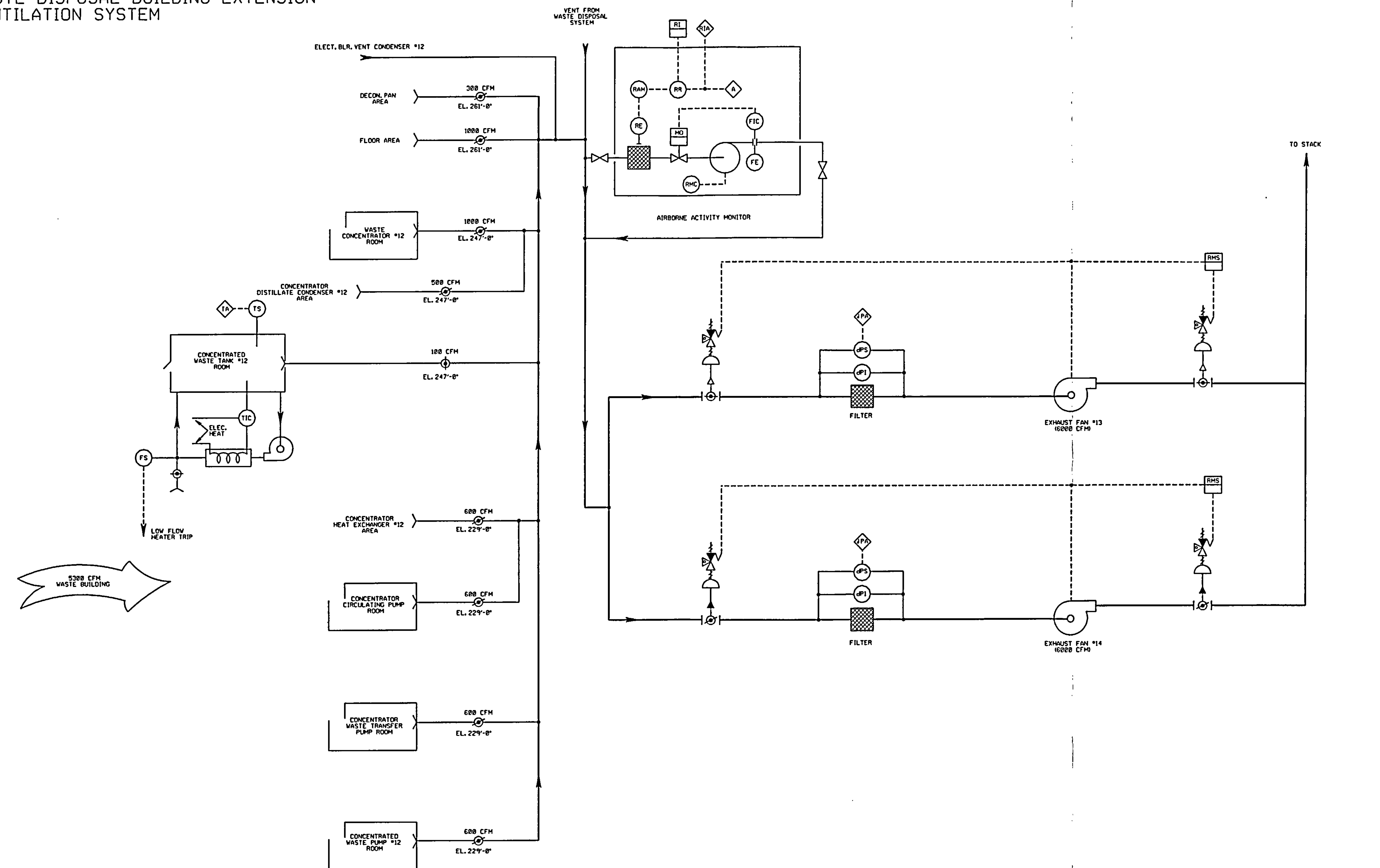


FIGURE III-16
UFSAR REVISION 18
OCTOBER 2003

2.3 Refueling Cycle Reactivity Balance

For each fuel cycle, the reactivity balance at the beginning of cycle (BOC) and the R-factor are calculated. The results are as follows:

Cold Clean Core, 20°C

K_{eff}, Uncontrolled	SRLR[1]
K_{eff}, Fully Controlled	SRLR[1]
K_{eff}, Strongest Control Rod Out	SRLR[1]
R, Maximum Increase in Cold Core Reactivity with Exposure, ΔK_{eff}	SRLR[1]

The R-factor is the difference in reactivity with the strongest control rod out at BOC, and the reactivity with the maximum calculated strongest rod out at any exposure.

3.0 Thermal and Hydraulic Characteristics

3.1 Thermal and Hydraulic Design

3.1.1 Recirculation Flow Control

Reactor power can be controlled over an approximate 50-percent power range, but no lower than about 40 percent of full power, by adjustment of the reactor recirculation flow with no control rod movement. Reactor power change is accomplished by utilizing the large negative power coefficient characteristic of BWR designs. To increase reactor power, recirculation flow is increased which reduces the void accumulation in the core by removing the steam at a faster rate. A positive reactivity input is balanced by negative reactivity effects of higher fuel temperature and new void formation. When these effects balance out, the reactor will be operating at a higher power level. The feedwater level controller increases the feedwater flow to match the increased steam generation. Conversely, when a power reduction is required, recirculation flow is reduced. A typical relationship between coolant flow rate and reactor power is shown on Figure IV-1.

The reactor recirculation pump speed control is discussed in detail in Section VIII-B.

3.1.2 Core Thermal Limits

Core thermal limits are based on two potential thermal damage modes: excessive cladding temperature and excessive cladding strain. Fuel damage is defined as a loss of cladding integrity allowing release of fission products to the coolant. The clad temperature failure mode is dependent on MCPR; the clad strain mode is dependent on the magnitude of the linear heat generation rate (LHGR). Clad failure due to high temperature following a loss-of-coolant accident (LOCA) is dependent on MAPLHGR.

3.1.2.1 Excessive Clad Temperature

During normal operation and operational transients, nucleate boiling occurs in the core. Nucleate boiling is characterized by a small clad-to-coolant temperature drop so that resultant clad temperatures are only slightly above the coolant temperature. At sufficiently higher power levels, the transition boiling mode would be initiated. Transition boiling is accompanied by cladding temperature fluctuations. The bundle power, at which some point within the assembly experiences onset of transition boiling, is termed the critical power. If power is increased sufficiently beyond this point, the film boiling mode would occur and result in potential clad perforations.

A figure of merit utilized for establishing reactor operating limits is the CPR. This is the ratio of the critical power to the operating bundle power. The critical power is determined at the same mass flux, inlet temperature, and pressure which exists at the specified reactor condition. Thermal margin is stated in terms of the MCPR which corresponds to the most limiting fuel assembly in the core. To ensure that adequate margin is maintained, the following transient design requirement was chosen:

> Moderate frequency transients caused by a single Operator error or equipment malfunction shall be limited such that, considering uncertainties in manufacturing and monitoring the core operating state, more than 99.9 percent of the fuel rods would be expected to avoid boiling transition.

Using this basic design requirement, both normal operating and transient thermal limits in terms of MCPR are derived. These limits are determined in accordance with the methods described in Reference 3. With each reload, compliance with the limits on MCPR is verified by transient analyses. For each cycle, the most limiting transient MCPR is reported in the SRLR(1) for that cycle and updated in the COLR.

The highest level of power and consequently the lowest value of MCPR is maintained assuming the maximum total peaking factors listed in Section I-B.7.0. The total peaking factor in turn is composed of the local, radial and axial peaking factors.

Clad failure due to high temperature following a postulated LOCA is a function of average heat generation rate of all rods of a fuel assembly. Average planar linear heat generation rate (APLHGR) is the parameter which describes the potential for that failure and a limit on MAPLHGR is established for core operation. The limits on APLHGR as a function of exposure are shown in the

3.2.2 Thermal Analysis

The objective for normal operation and transient events is to maintain nucleate boiling and, thus, avoid a transition to film boiling. Operating limits are specified to maintain adequate margin to the onset of the boiling transition. Both the transient (safety) and normal operating thermal limits in terms of MCPR are derived from this basis.

3.2.2.1 Fuel Cladding Integrity Safety Limit Analysis

The generation of the MCPR limit requires a statistical analysis of the core near the limiting MCPR condition. The statistical analysis is used to determine the MCPR corresponding to the transient design requirement of paragraph 3.1.2.1. This MCPR established fuel cladding integrity safety limit applies not only for core-wide transients, but is also conservatively applied to the localized RWE transient.

The statistical analysis utilizes a model of the BWR core which simulates the process computer function. This code produces a CPR map of the core based on inputs of power distribution and flow and on heat balance information.

Details of the procedure are documented in Reference 17. Power distribution uncertainties used in the cycle-specific statistical analysis are presented in Reference 18.

The minimum allowable CPR is set to correspond to the criterion that 99.9 percent of the rods are expected to avoid boiling transition by interpolation among the means of the distributions formed by all the trials.

Cycle-specific analyses have been performed, as described in Reference 3, which provide conservative safety limit MCPRs. The results of the analysis are summarized in Reference 1.

The results of the analyses show that at least 99.9 percent of the fuel rods in the core are expected to avoid boiling transition if the MCPR is greater than or equal to the fuel-specific safety limit, as specified in the SRLR (Reference 1) and the COLR. The fuel cladding integrity safety limit MCPR is contained in the COLR.

3.2.2.2 MCPR Operating Limit Analysis

A MCPR operating limit is established to ensure that the Fuel Cladding Integrity Safety Limit is not exceeded for any moderate frequency transient. This operating requirement is obtained by addition of the absolute, maximum ΔCPR value for the most limiting transient (including any imposed adjustment factors), from rated conditions postulated to occur at the plant to the fuel cladding integrity safety limit.

There are eight nuclear system parameter variations or transients which could pose potential deleterious effects to the nuclear steam supply system (NSSS). These parameter variations are:

1. Nuclear system pressure increase - threatens to rupture the reactor coolant pressure boundary (RCPB) from internal pressure. Also, a pressure increase collapses the voids in the moderator. This causes an insertion of positive reactivity which may result in exceeding the fuel cladding safety limits.

2. Reactor vessel water (moderator) temperature decreases - results in an insertion of positive reactivity as density increases. Positive reactivity insertions threaten the fuel cladding safety limits because of high power.

3. Positive reactivity insertion - is possible from causes other than nuclear system pressure or moderator temperature changes. Such reactivity insertions threaten the fuel cladding safety limits because of higher power.

4. Reactor vessel coolant inventory decrease - threatens the fuel as the coolant becomes unable to maintain nucleate boiling.

5. Reactor core coolant flow decrease - threatens the fuel cladding safety limits as the coolant becomes unable to maintain nucleate boiling.

6. Reactor core coolant flow increase - reduces the void content of the moderator, resulting in a positive reactivity insertion. The resulting high power may exceed fuel cladding safety limits.

7. Core coolant temperature increase - could exceed fuel cladding safety limits.

8. Excess of coolant inventory - could result in damage resulting from excessive carry-over.

Of these parameter variations, only a few are characteristic of operating transients which would result in a significant reduction in MCPR.

To determine the limiting transient events, the relative dependency of CPR upon various thermal-hydraulic parameters was examined. A sensitivity study was performed to determine the effect of changes in bundle power, bundle flow, subcooling, R-factor, and pressure on CPR for the 8x8 fuel design.

Results of the study indicate that CPR is most responsive to fluctuations in the R-factor and bundle power. A slight sensitivity to pressure and flow changes and relative independence to changes in inlet subcooling was also shown. The R-factor is a function of bundle geometry and local power distribution and is assumed to be constant throughout a transient. Therefore, transients which would be limiting because of MCPR would primarily involve significant changes in power. Based on this, the transients most likely to limit operation because of MCPR considerations are:

1. Turbine trip without bypass, or generator load rejection without bypass.

2. Loss of feedwater heating, or inadvertent high-pressure coolant injection (HPCI) startup.

3. Feedwater controller failure (maximum demand).

4. Control RWE.

5. Recirculation flow controller failure - increasing flow and an inadvertent startup of a cold recirculation loop as related to K_f curve.

The above transients are reevaluated for each reload core. The results of the analysis are summarized in the SRLR[1] and are used to establish the most limiting transient and the MCPR operating limit.

3.3 Reactor Transients

Core-wide rapid pressurization events (turbine trip without bypass and feedwater controller failure) are analyzed using the system model documented in Reference 5. The ODYN code contains a one-dimensional representation of the reactor core which is coupled to the recirculation and control system model. The integrated model is based on one-dimensional reactor kinetics, multinoded thermal-hydraulic and heat transfer relationships, and mechanical kinetic equations of the equipment. ODYN contains a refined reactor core description and a detailed steam line model to simulate pressure dynamics during a transient. For the slower core-wide transients, loss of feedwater heating is analyzed using either the steady-state 3-D BWR Simulator Code(4), or the REDY Transient Model(6). A more thorough description of the transients analyzed is given in Section XV-B.3.0.

4.0 Stability Analysis

4.1 Design Bases

Three types of stability are considered in the design of BWRs: 1) reactor core (reactivity) stability; 2) channel hydrodynamic stability, and 3) total system stability. A stable system is analytically demonstrated if no inherent limit cycle or divergent oscillation develops within the system as a result of calculated step disturbances of any critical variable, such as steam flow, pressure, neutron flux, or recirculation flow. The criteria for evaluating reactor dynamic performance and stability are stated in terms of two compatible parameters. First is the decay ratio, x_2/x_0, which is the ratio of the magnitude of the second overshoot to the first overshoot resulting from a step perturbation. A plot of the decay ratio is a graphic representation of the physical responsiveness of the system which is readily evaluated in a time-domain analysis. Second is the damping coefficient, ζn, the definition of which corresponds to the dominant pole pair closest to the imaginary axis in the s-plane for the system closed-loop transfer function. As ζn decreases, the closed-loop roots approach the imaginary axis and the response becomes increasingly oscillatory.

4.2 Stability Analysis Method

The stability analysis methods for the Nine Mile Point Nuclear Power Station are documented in GENE-A13-00360-02, "Application of Stability Long-Term Solution Option II to Nine Mile Point Nuclear Station Unit 1," August 30, 1995, and NEDC-32992P-A, "ODYSY Application for Stability Licensing Calculations," July 2001.

GE11 stability compares favorably to GE6/7 (P8x8R) fuel, as reported in "GE11 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR II)," NEDE-31917, Section 2.9. GE6/7 fuel design was selected as the reference fuel design for comparison, consistent with Amendment 22 criteria. The favorable comparison of GE6/7 is largely due to the effects of GE11 part-length rod design. GE8's reduced upper tie-plate pressure drop and four-water-rod design provide it with stability characteristics comparable to or better than those for GE6/7.

5.0 Mechanical Design and Evaluation

5.1 Fuel Mechanical Design

5.1.1 Design Bases

To meet the performance objectives "2" and "3", the fuel rod is designed with adequate margin to assure that excessive fuel failures will not occur during normal operation or anticipated operational transients. Fuel failure is defined as perforation of the fuel cladding which would permit release of fission products to the reactor coolant. Details of the fuel design can be found in Reference 7.

5.1.2 Fuel Rods

The reactor fuel consists of high-density ceramic uranium dioxide pellets, manufactured by compacting and sintering uranium dioxide powder into right cylindrical pellets with flat ends and chamfered edges. The pellets are enclosed in Zircaloy-2 tubes which are evacuated, backfilled with helium, and sealed by welding Zircaloy plugs into each end. Ceramic uranium dioxide is chemically inert to the cladding at operating temperatures and is resistant to attack by water. Several U-235 enrichments are used in the fuel assemblies to reduce the local peak-to-average fuel rod power ratios. Selected fuel rods within each reload bundle also incorporate small amounts of gadolinium as burnable poison. Gd_2O_3 is uniformly distributed in the UO_2 pellet and forms a solid solution. The fuel rods are described in further detail in References 3 and 7.

The fuel cladding thickness is adequate to satisfy the requirement that the clad be "freestanding" and capable of withstanding pressures well beyond operating reactor pressure without collapsing onto the contained pellets. Adequate free volume is provided within each fuel rod in the form of a pellet-to-cladding gap, and a plenum region at the top of the fuel rod to accommodate thermal and irradiation expansion of the UO_2, and the internal pressures resulting from the helium fill gas, impurities and gaseous fission products liberated over the design life of the fuel. A plenum spring or retainer is provided in the plenum space to minimize movement of the fuel column inside the fuel rod during shipping and handling.

In barrier fuel, the fuel cladding incorporates an inner lining of pure zirconium: this lining decreases the probability of pellet-clad-interaction induced fuel failures without altering the other properties of the fuel.

Two types of fuel rods are used in a fuel bundle: tie-rods and standard rods. Tie-rods are described in Section 5.1.4. The end plugs of the standard rods have shanks which fit into bosses in the tie-plates. An Inconel-X expansion spring is located over the upper end plug shank of each rod in the assembly to keep the rods seated in the lower tie-plate while allowing independent axial expansion by sliding within the holes of the upper tie-plate.

5.1.3 Water Rods

The water rods are hollow Zircaloy tubes with several holes punched around the circumference near each end to allow coolant to flow through.

5.1.4 Fuel Assemblies

The fuel assemblies are described in detail in References 3 and 8. Fuel bundle specific information is provided in Reference 7.

5.1.5 Mechanical Design Limits and Stress Analysis

The fuel mechanical design limits and stress analysis are described in detail in Reference 3.

5.1.6 Relationship Between Fuel Design Limits and Fuel Damage Limits

Fuel is designed to satisfy the conservative mechanical design limits in accordance with Reference 7.

5.1.7 Surveillance and Testing

Rigid quality control requirements are enforced at every stage of fuel manufacturing to assure that design specifications are met. Written manufacturing procedures and quality control plans define the steps in the manufacturing process as described in Reference 3.

6.0 Control Rod Mechanical Design and Evaluation

6.1 Design

6.1.1 Control Rods and Drives

The control rod drive (CRD) system consists of the control rod, CRD, hydraulic scram system and the hydraulic drive system. The mechanical design and evaluation of the control rod is discussed in this section; the other portions of the control rod system are discussed in Section X-C.

can be removed from the reactor. These procedures are established to prevent accidental separation of the control rod from the CRD.

The drives position the control rods in 6-in increments of stroke and hold them in these discrete latch positions until actuated for movement by the hydraulic system to a new position. Visible indication of the position of each drive is displayed in the control room by means of illuminated numerals which correspond with the respective latched positions. In addition, indication is provided that shows insert and withdraw travel limits of the drive and an overtravel withdraw limit on the drive have been reached. Control rod seating at the lower end of the stroke prevents the overtravel withdraw limit from being reached unless the control rod is uncoupled from the drive. This allows the coupling to be checked. These indicators and those for the in-core monitors are grouped together and displayed on the control panel and arranged on the board to correspond to relative rod and in-core monitor positions in the core.

During reactor shutdown, the SDM can be verified. The SDM demonstration is performed as described in the Technical Specifications.

6.1.2 Standby Liquid Poison System

This system is described in detail in Section VII-C. The standby liquid poison system is designed to provide the capability of bringing the reactor, at any time in a cycle, from a full power and minimum control rod inventory (defined to be at peak xenon) to a subcritical condition with the reactor in the most reactive xenon-free state. The liquid poison solution is sodium pentaborate enriched in boron-10 isotope. The liquid poison system is capable of satisfying the criteria of $<0.97\ K_{eff}$. This criteria is checked for compliance for each operating cycle. The calculated liquid poison system SDM for the cold (20°C), xenon-free core condition is provided in the SRLR[1]. This SDM corresponds to a boron (B-10 isotope) concentration of 109.8 ppm in the reactor core.

6.2 Control System Evaluation

6.2.1 Rod Withdrawal Errors Evaluation

Design features provided to minimize the possibility of inadvertent continuous control rod withdrawal, and to limit potential power transients in the event they should occur, include the following:

1. The control system is designed so that only one rod can be withdrawn at a time.

2. Normal rod operation is a step (notch) at a time. Two control switches must be operated at the same time to withdraw a rod continuously.

The structural components which guide the control rods have been examined to determine the loadings which would occur in a LOCA (including a steam line break). The core structural components are designed so that deformations produced by accident loadings will not prevent insertion of control rods.

Considerable effort was expended to eliminate possible failures or control instability due to the vibration of reactor internal components. The reactor system was analyzed as a multidegree-of-freedom system. This analysis determined the system's natural frequencies, the resultant vibration mode shapes and the relationship between the vibration amplitudes and the critical stresses in the system, to show that system integrity would be maintained.

7.3 Surveillance and Testing

Rigid quality control requirements assured that the design specifications of the vessel internal components were met. These quality control methods were utilized during the fabrication of the individual components as well as during the assembly process.

Preoperational performance tests and the startup program demonstrated the design adequacy of reactor vessel internals and operability of the core spray spargers.

Periodic testing of the control rod system, i.e., reactivity margin - core loading and stuck control rods; rod scram insertion times and reactivity anomalies, is described in the Technical Specification.

C. REFERENCES

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3. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-14-US, June 2000.

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11. Safety Evaluation of the General Electric Duralife 230 Control Rod Assembly, NEDE-22290-P-A Supplement 3, May 1988.

12. GENE-523-113-0894, Rev. 1, "BWR Core Shroud Inspection and Evaluation Guidelines," March 1995.

13. BWRVIP-01, Rev. 2, "BWR Core Shroud Inspection and Flaw Evaluation Guideline," October 1996.

14. BWRVIP-07, "Guidelines for Reinspection of BWR Core Shrouds," February 1996.

15. Design Change No. N1-97-033, "Core Shroud Vertical Weld Contingency Repair."

16. GE Marathon Control Rod Assembly, NEDE-31758P-A, October 1991.

17. NEDC-32601P-A, "Methodology and Uncertainties for Safety Limit MCPR Evaluations," August 1999.

18. NEDC-32694P-A, "Power Distribution Uncertainties for Safety Limit MCPR Evaluations," August 1999.

1.4 Primary Coolant Leakage

A double O-ring type seal is provided on the reactor vessel head closure. The area between the seals is monitored for leakage. A groove between the inner and outer O-ring communicates through the vessel flange to a line in which are installed a level switch and a pressure switch between two solenoid valves. The solenoid valves are operated from the control room. The monitoring instrumentation is shown on Figure V-1.

Other primary coolant leakage is detected by monitoring leakage into the drywell floor drain tank for unidentified drywell leakage, and the drywell equipment drain tanks for identified drywell leakage. Unidentified drywell leakage from the CRDs, valve flanges, packing, component cooling water, service water, recirculation pump suction and discharge valve packing leakoff, and any other leakage not connected to the drywell equipment drain tanks, collects in the drywell floor drain tanks. Identified drywell leakage is hard piped to the drywell equipment drain tanks and includes recirculation pump seal leakage. Abnormal leakage rates for the drywell floor and equipment drain tanks are detected and alarmed in the control room.

The excess leakage alarm function for the drywell floor and equipment drain tanks is performed by measuring volume changes in gallons that occur over a predetermined time period and calculating the resultant rate of change. Volume changes are used to determine the rate of change because of the irregular shape of the drywell floor and equipment drain tanks. By using volume change, excess leakage alarm capability is achieved across the entire instrument range with alarm checking occurring upon each recalculation.

The rate of rise alarm function for the drywell floor drain tank is performed by measuring the amount of time between precise level step changes. When a level increase is detected, the change in tank volume and elapsed time since the last change are used to determine the rate of volume change. The rate of volume change is then used to determine the rate of rise. The calculated rate of rise is output to the control room chart recorders and alarm checked.

The rate of rise for the drywell equipment drain tanks is monitored by evaluating the fill rate recorded on the equipment drain tank level chart recorder in the control room. This is performed every 4 hr.

The integrated flow pumped from the drywell floor and equipment drain tanks to the waste disposal system is another means that can be used to determine leakage into the drywell floor or the equipment drain tanks.

Automatic blowdown will not occur for any primary system leak rate below the maximum allowable total operating leak rate of approximately 25 gpm. However, for breaks below about 50 gpm (although the Technical Specification limit is 25 gpm), the triple low-level setting (6 ft 3 in below minimum normal) would not be reached and automatic blowdown of relief valves would not be initiated. If normal Station offsite power were lost, both CRD hydraulic system pumps would be automatically loaded on the diesel generators to maintain water level in the vessel above the automatic blowdown trip level. It is assumed that only one CRD system is operating. The flow rate of one CRD system pump is 50 gpm at 1000 psig reactor vessel pressure and 180 gpm at zero psig reactor vessel pressure. If both pumps were operating, the flows would be greater.

For much larger leak sizes, the time to reach the automatic blowdown trip level is shown in Table V-5. This table is conservatively based on only one diesel generator and its associated CRD system pump being available.

1.5 Coolant Chemistry

The RCS is not designed to use inhibitors. Limits are set on chlorides, solids and gross coolant radioactivity during normal Station operation.

Hydrogen water chemistry (HWC) injection and noble metal chemical addition (NMCA or NobleChem) systems are installed to reduce the potential for intergranular stress corrosion cracking (IGSCC) of the stainless steel reactor vessel components and recirculation piping. The zinc injection system is installed to reduce Cobalt 60 buildup in the primary piping corrosion films. This has the major benefit of reducing radiation dose rates in the drywell, reducing radiation exposure during outages. Hydrogen injection is provided through the feedwater/condensate systems; NobleChem is periodically added through the recirculation pump differential pressure transmitter lines, and zinc injection is provided through the feedwater system.

2.0 Reactor Vessel

An isometric drawing of the reactor vessel is shown on Figure IV-9. Vessel penetrations are shown on Figure V-2 and data for the reactor vessel in Table V-1. The reactor vessel is a vertical cylindrical pressure vessel. The base plate material is high-strength alloy carbon steel SA-302, Grade B. The vessel interior is clad with Type 308L to produce a 304 composition stainless steel following application by weld overlay.

The head closure is designed for easy removal and reassembly, being bolted to the vessel with high-strength studs. Removable stud bushings are furnished in the body flange to facilitate repair of damaged threads.

The CRD housings and the in-core instrumentation thimbles are welded to the bottom head of the reactor vessel.

Steam outlets are from the vessel body, thus eliminating the need to break flanged joints in the steam lines when removing the vessel head for refueling. Safety valves are mounted on the vessel head. Solenoid-actuated relief valves are mounted on the main steam lines (MSL).

An elevation drawing of the reactor vessel and supporting concrete structures is presented as Figure V-3. The reactor vessel is supported by a steel skirt welded to the bottom head of the vessel. The base of the skirt is continuously supported by a

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F. REFERENCES

1. NRC Standard Review Plan, Section 5.2.2, Overpressurization Protection - NUREG 75-087.

2. 0000-0005-1174SRLR, Revision 0, "Supplemental Reload Licensing Report for NMP1, Reload 17, Cycle 16," January 2003.

3. K. Shure and D. J. Dudziak, "Calculating Energy Release by Fission Products," AEC Report WAPD-T-1309, March 1961.

4. K. Shure, "Fission Product Decay Heat," AEC Report WAPD-BT-24, December 1961.

TABLE V-3

FATIGUE RESISTANCE ANALYSIS

Region of Vessel	Usage Factor
Closure Studs	0.205
Basin Seal Skirt Weld	0.782
Feedwater Nozzles	
With Repair Cavities	0.489
Without Repair Cavities	0.163
Control Rod Drive Penetrations	0.060
Lower Vessel Head, Vessel Support Skirt and Core Support Cone	0.0833
Reactor Recirculation Nozzles	0.006

TABLE V-4

CODES FOR SYSTEMS FROM REACTOR VESSEL CONNECTION TO SECOND ISOLATION VALVE

	Piping Vessel Nozzle to Second Isolation Valve	Isolation Valves
Shutdown Cooling Cleanup	ASA B31.1-1955; ASME Sec I-1962 and Articles N324 and N460 to N469 of ASME Sec III-1965; ASME Sec III, Appendix F, 1986 Edition*	ASME Sec I-1962
	ASA B31.1-1955; ASME Sec I-1962 and Articles N324 and N460 to N469 of ASME Sec III-1965; ASME Sec III, Appendix F, 1986 Edition*	ASA B31.1-1955 and certain requirements of ASME Sec IIIA-1965
Feedwater Core Spray	ASA B31.1-1955; ASME Sec I-1962 and Articles N324 and N460 to N469 of ASME Sec III-1965	ASME Sec I-1962
	ASA B31.1-1955; ASME Sec I-1962 and Articles N324 and N460 to N469 of ASME Sec III-1965; ASME Sec III, Appendix F, 1986 Edition*	ASA B31.1-1955 and certain requirements of ASME Sec IIIA-1965
Liquid Poison	ASA B31.1-1955, ASME Sec I-1962 and Articles N324 and N460 to N469 of ASME Sec III-1965	ASME Sec I-1962

* For analyzing thermally-induced overpressurization conditions between isolation valves.

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NINE MILE POINT NUCLEAR STATION UNIT 1

FINAL SAFETY ANALYSIS REPORT (UPDATED)

VOLUME 2

OCTOBER 2003

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TABLE VI-3a

REACTOR COOLANT SYSTEM ISOLATION VALVES

Line or System	No. of Valves (Each Line)	Location[9] Relative to Primary Containment	Valve No.	Normal Position	Fail Position on Loss of Motive Power or Control Signal	Motive Power*	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
Main Steam[1] (Two Lines)	1 1	Inside Outside	01-01,02 01-03,04	Open Open	As Is Closed	AC Motor Pn/DC Solenoid	10 10	Close Close	Reactor water level low-low, or low reactor pressure (with mode switch in run), or main steam line high flow, or low-low-low condenser vacuum, or high temperature in the steam tunnel
Feedwater[1] (Two Lines)	1 1	Outside Outside	31-07,08 31-01R,02R	Open Open	As Is --	AC Motor Self Act. Ck.	60 --	-- --	Remote manual --
Emergency Cooling									
Steam Leaving Reactor[1] (Two Lines)	1 1	Outside Outside	39-09R,10R 39-07R,08R	Open Open	As Is As Is	AC Motor DC Motor	38 38	Close Close	High emergency cooling system flow
Condensate Return to Reactor[1] (Two Lines)	1 1	Inside Outside	39-03,04 39-05,06	Closed Closed	-- Open	Self Act. Ck. Pn/DC Solenoid	-- 60 60	-- Close Open	-- High emergency cooling system flow Reactor water level low-low or high reactor pressure
Reactor Cleanup									
Water Leaving Reactor[1] (One Line)	1 1	Inside Outside	33-02R 33-04	Open Open	As Is As Is	AC Motor DC Motor	18 18	Close Close	Reactor water level low-low, or high area temperature or liquid poison initiation
Water Return to Reactor[1] (One Line)	1 1	Inside Outside	33-01R 33-03	Open Open	As Is --	AC Motor Self Act. Ck.	18 --	Close --	 --

TABLE VI-3a (Cont'd.)

Line or System	No. of Valves (Each Line)	Location[9] Relative to Primary Containment	Valve No.	Normal Position	Fail Position on Loss of Motive Power or Control Signal	Motive Power*	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
Shutdown Cooling									
Water Leaving Reactor[4][7] (One Line)	1 1	Inside Outside	38-01 38-02	Closed Closed	As Is As Is	AC Motor DC Motor	40 40	Close Close	Reactor water level low-low, or high area temperature
Water Leaving Reactor[8] (CIV Bypass Line)	1	Inside	38-216	Closed	--	Self Act. Ck.	--	--	--
Water Return to Reactor[4][7] (One Line)	1 1	Inside Outside	38-13 38-12	Closed Closed	As Is --	AC Motor Self Act. Ck.	40 --	Close --	 --
Water Seal Parallel Valves (See Note 7)	4	Outside Outside Outside Outside	38-169 38-170 38-171 38-172	Closed Closed Closed Closed	-- -- -- --	Self Act. Ck. Self Act. Ck. Self Act. Ck. Self Act. Ck.	-- -- -- --	-- -- -- --	-- -- -- --
Liquid Poison[1] (One Line)	1 1	Inside Outside	42.1-02 42.1-03	Closed Closed	-- --	Self Act. Ck. Self Act. Ck.	-- --	-- --	-- --
Control Rod Drive Hydraulic[1] (One Line)	1 1	Inside Outside	44.3-13 44.3-12	Open Open	-- --	Self Act. Ck. Self Act. Ck.	-- --	-- --	-- --
Core Spray									
Core Spray Injection[3] (Two Lines)	2 1	Inside Outside	40-01,09, 10,11 40-02,12	Closed Open	As Is As Is	AC Motor AC Motor	22.5 22.5	Open Open	Reactor water level low-low or high drywell pressure coincident with reactor vessel pressure less than 365 psig
Core Spray Penetration[3] (Piping Thermal Relief)	1	Inside	40-80 40-83	Closed	--	Self Act. Ck.	--	--	
Core Spray High Point Vent[4] (Two Lines)	1 1	Inside Outside	40-30,31 40-32,33	Closed Closed	As Is Closed	AC Motor Pn/DC Solenoid	27 27	Close Close	Reactor water level low-low or high drywell pressure
Core Spray Condensate Supply (Keep Fill)[5] (Two Lines)	2	Outside	40-20,21, 22,23	Open	--	Self Act. Ck.	--	--	--

TABLE VI-3a (Cont'd.)

Line or System	No. of Valves (Each Line)	Location[9] Relative to Primary Containment	Valve No.	Normal Position	Fail Position on Loss of Motive Power or Control Signal	Motive Power*	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
Core Spray System Valves[5] (Two Lines)	1	Outside	40-03,13	Closed	--	Self Act. Ck.	--	--	--
Core Spray Pump Discharge[4] (Two Test Lines to Suppression Chamber)	1	Outside	40-05,06	Closed	As Is	AC Motor	27	Close	Reactor water level low-low or high drywell pressure
Scram Discharge Volume System Vent**[1] (One Line)	2	Outside	44.2-15,16	Open	Closed	Pn/AC Solenoid	10	Close	Automatic or manual reactor scram
Scram Discharge Volume System Drain**[1] (One Line)	2	Outside	44.2-17,18	Open	Closed	Pn/AC Solenoid	10	Close	Automatic or manual reactor scram
Post-accident Reactor Sampling[1][6] (One Line)	1	Outside	44.1-07	Open	--	Self Act. Flow Fuse	--	--	--
	1	Outside	122-03	Closed	Closed	Pn/DC Solenoid	30	Close	Reactor water level low-low or low-low-low condenser vacuum or reactor low pressure (with mode switch in RUN) or high temperature in the steam tunnel or main steam line high flow
Reactor Recirculation System Sampling[1] (One Line)	1	Inside	110-127	Closed	As Is	AC Motor	20	Close	
	1	Outside	110-128	Closed	As Is	DC Motor	20	Close	

NOTES:

* Pn - Pneumatically Operated.

** Technical Specification Section 3.1.1e for LCO requirements.

(1) These valves do not have to be vented during the Type A test. However, Type C leakage from these valves is added to the Type A test results, if not vented.

(2) Deleted.

(3) The inside core spray injection isolation valves are water sealed during and after an accident. These valves are leak rate tested with water in accordance with the Appendix J Program. Under 10CFR50, Appendix J, Option B, through RG 1.163, water-sealed CIV test frequency may be set using a performance basis in a manner similar to that described in NEI 94-01, Revision 0, dated 7/26/95, for Type B and Type C test intervals. The outside core spray injection isolation valves are open with their breakers locked in the OFF position. Therefore, the outside core spray injection valves do not have to be tested under the IST or Appendix J Leakage Program.

TABLE VI-3a (Cont'd.)

NOTES:

(4) These valves are provided with a water seal. Valves shall be tested consistent with Appendix J water seal testing requirements. Under 10CFR50, Appendix J, Option B, through RG 1.163, water-sealed CIV test frequency may be set using a performance basis in a manner similar to that described in NEI 94-01, Revision 0, dated 7/26/95, for Type B and Type C test intervals. Leakage rates shall be conservatively limited to 0.5 gpm per nominal inch of valve diameter up to a maximum of 5 gpm.

(5) These valves are tested in accordance with Technical Specification Section 4.2.7.1a.

(6) The self-actuating flow fuse is tested in accordance with Technical Specification Section 4.3.4c.

(7) Two 1" globe valves (38-206 and 208) are provided outside in the seal water (core spray) flow test line and one 3/4" globe valve (38-209) is provided outside in the seal water supply line drain, which also serve as RCS isolation valves.

(8) One 3/4" check valve (38-216) is provided inside primary containment around isolation valve 38-01. This valve is provided with a water seal and tested under the Appendix J program for limited flow in the open direction, and under the IST Program, exercised closed for isolation capability.

TABLE VI-3b (Cont'd.)

Line or System	No. of Valves (Each Line)	Location Relative to Primary Containment	Valve No.	Normal Position	Fail Position on Loss of Motive Power or Control Signal	Motive Power*	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
Reactor Cleanup System Relief Valve Discharge[2] (One Line to Suppression Chamber)	2	Outside	63.1-01,02	Closed	--	Self Act. Ck.	--	--	--
H_2-O_2 #11 Sampling									
Drywell Supply (One Line)	2	Outside	201.7-01, 02	Open	Closed	Pn/DC Solenoid	60	Close	Reactor water level low-low or high drywell pressure
Suppression Chamber Supply (One Line)	2	Outside	201.2-110, 111	Open	Closed	Pn/DC Solenoid	60	Close	
Drywell Return (One Line)	2	Outside	201.7-10,11	Open	Closed	Pn/DC Solenoid	60	Close	
Suppression Chamber Return (One Line)	2	Outside	201.2-109, 112	Open	Closed	Pn/DC Solenoid	60	Close	
H_2-O_2 #12 Sampling									
Drywell Supply[1] (One Line)	2	Outside	201.2-29, 30	Open	Closed	DC Solenoid	60	Close	Reactor water level low-low or high drywell pressure
Suppression Chamber Supply[1] (One Line)	2	Outside	201.2-23,24	Open	Closed	DC Solenoid	60	Close	
Drywell Return[1] (One Line)	2	Outside	201.2-67,68	Open	--	Self Act. Ck.	--	--	--
Suppression Chamber Return[1] (One Line)	2	Outside	201.2-70,71	Open	--	Self Act. Ck.	--	--	--
Core Spray									
Pump Suction (Four Lines From Suppression Chamber)	1	Outside	81-01,02 81-21,22	Open	As Is	AC Motor	90	--	Remote manual
Pump Discharge[4] (Two Test Lines to Suppression Chamber)	1	Outside	40-05,06	Closed	As Is	AC Motor	27	Close	Reactor water level low-low or high drywell pressure

TABLE VI-3b (Cont'd.)

Line or System	No. of Valves (Each Line)	Location Relative to Primary Containment	Valve No.	Normal Position	Fail Position on Loss of Motive Power or Control Signal	Motive Power*	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
Core Spray (cont'd.)									
Pump Recirculation (Four Lines Connect to Two Test Lines to Suppression Chamber)	1	Outside	81-241,242, 243,244	Closed	--	Self Act. Relief Valve	--	--	--
Containment Spray									
Drywell & Suppression Chamber Common Supply(2) (Four Lines)	1	Outside	80-15,16, 35,36	Open	As Is(6)	Pn/DC Solenoid	60	--	Remote manual
Drywell Branch(2) (Four Lines)	1	Outside	80-17,18, 37,38	Closed	--	Self Act. Ck.	--	--	--
Suppression Chamber Branch(2) (One Branch for Each System)	2**	Outside	80-19,39, 65,66,67,68	Closed	--	Self Act. Ck.	--	--	--
Pump Suction From Suppression Chamber (Four Lines)	1	Outside	80-01,02, 21,22	Open	As Is	AC Motor	70	--	Remote manual
Containment Spray Test Line to Torus(2) (One Line)	1	Outside	80-118	Closed	As Is	AC Motor	60	--	Remote manual
Compressed Air Test Line(2) (Two Lines)	1	Outside	80-90,91	Closed	--	--	--	--	Local manual
Emergency Cooling Vent to Torus(2) (One Line)	2	Outside	05-05,07	Closed	As Is	AC Motor	--	--	Remote manual

TABLE VI-3b (Cont'd.)

Line or System	No. of Valves (Each Line)	Location Relative to Primary Containment	Valve No.	Normal Position	Fail Position on Loss of Motive Power or Control Signal	Motive Power*	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
Containment Atmosphere Monitoring Supply Line (One Line)	2	Outside	201.7-08,09	Open	Closed	Pn/DC Solenoid	60	Close	Reactor water level low-low or high drywell pressure
Containment Post-LOCA Vent (Two Lines)	2	Outside	201.1-09, 11,14,16	Closed	Closed	Pn/DC Solenoid	60	Close	Reactor water level low-low or high drywell pressure
N2 Purge - TIP Indexers(1) (One Line)	2	Outside	201.2-39,40	Open	--	Self Act. Ck.	--	--	--
Traversing In-core Probe(1) (Four Lines)	1	Outside	36-147,148, 149,150	Closed	Closed	AC Solenoid	60	Close	Reactor water level low-low or high drywell pressure
Breathing Air Connection (One Line)	1 1	Inside Outside	114-116 114-114	Locked Closed Closed	-- --	-- --	-- --	-- --	Local manual
Service Water Connection(1) (One Line)	1 1	Inside Outside	72-480 72-479	Locked Closed Closed	-- --	-- --	-- --	-- --	Local manual
Recirc. Pump Cooling Water(5) Supply Line Return Line	 1 1	 Outside Outside	 70-93 70-92	 Open Open	 -- As Is	 Self Act. Ck. DC Motor	 -- 60	 -- --	 -- Remote manual
Drywell Cooler Water(5) Supply Line Return Line	 1 1	 Outside Outside	 70-95 70-94	 Open Open	 -- As Is	 Self Act. Ck. DC Motor	 -- 60	 -- --	 -- Remote manual

NOTES:

* Pn - Pneumatically Operated.
** One valve in each separate line and one valve in each common line.

(1) These valves do not have to be vented during the Type A test. However, Type C leakage from these valves is added to the Type A test results, if not vented.
(2) These valves are provided with a water seal capability. No Appendix J or IST leakage rate testing is required.
(3) This note deleted.
(4) These valves are provided with a water seal. Valves shall be tested consistent with Appendix J water seal testing requirements. Under 10CFR50, Appendix J, Option B, through RG 1.163, water-sealed CIV test frequency may be set using a performance basis in a manner similar to that described in NEI 94-01, Revision 0, dated 7/29/95, for Type B and Type C test intervals. Leakage rates shall be conservatively limited to 0.5 gpm per nominal inch of valve diameter up to a maximum of 5 gpm.
(5) These valves do not meet the requirements of 10CFR50 Appendix J, Section II-H. No testing required.
(6) Containment spray isolation valve fails open on loss of electrical (dc) power and fails as-is on loss of air.

REACTOR BUILDING VENTILATION SYSTEM

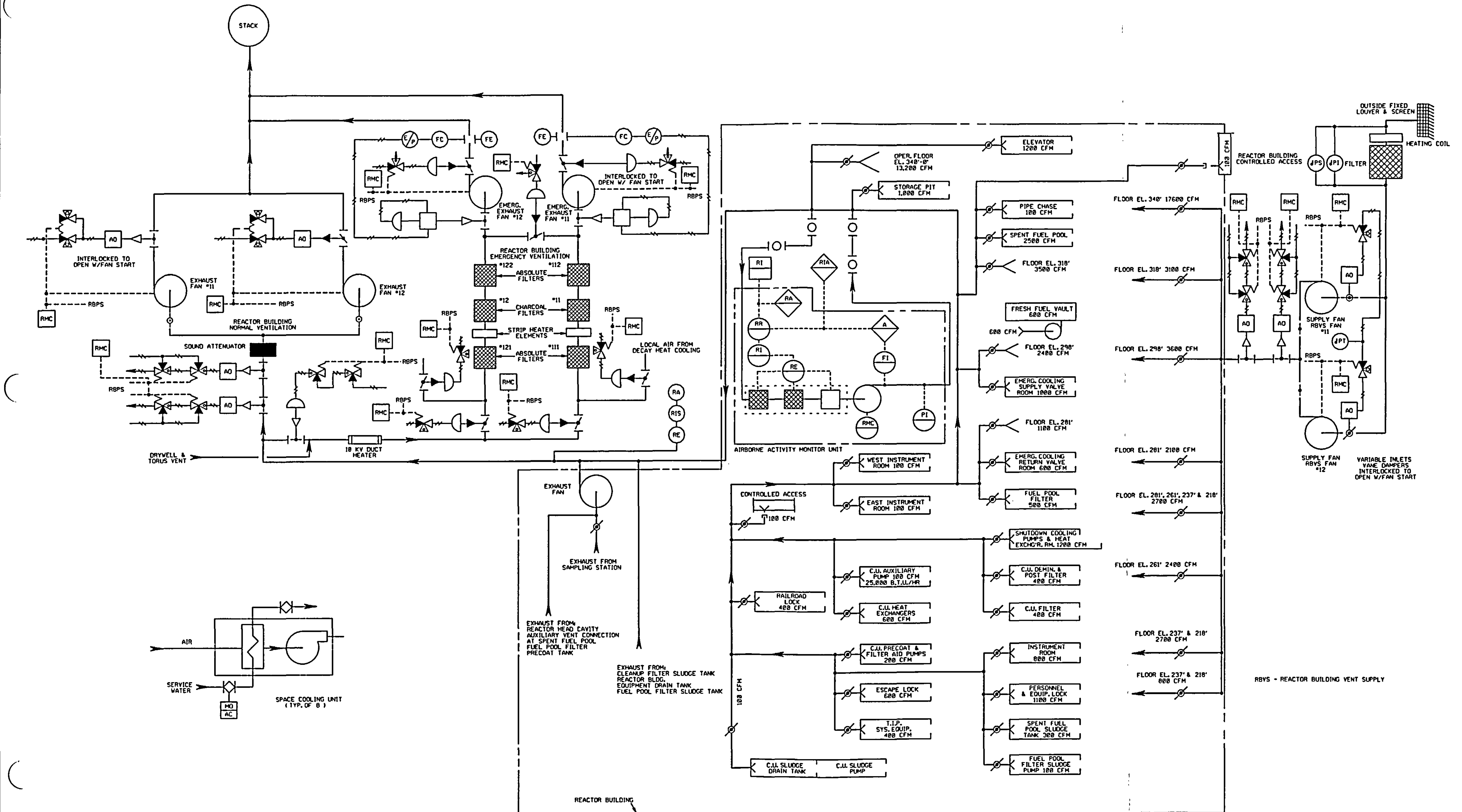

FIGURE VI-24
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water to the suppression chamber and the corresponding flow and pressure data obtained.

Subsequently, the core spray and topping pumps are periodically operated, and the water pumped from the suppression chamber through the appropriate supply lines to the outer system isolation valve, then returned to the suppression chamber. Flow into the reactor vessel is not attempted since this would introduce relatively impure water into the reactor coolant. Data on the flow rate and pressure at various points for each supply loop are obtained for comparison with the previously established normal conditions. Interlocks are provided such that the valve in the test line cannot be opened unless the motor-operated containment system isolation valves both inside and outside the drywell are closed. These valves cannot be reopened until the test valve is closed. The MOVs on the pump discharge lines to the reactor vessel are periodically opened fully and the time to open is recorded. These valves shall be fully open within 22.5 sec (valve stroke time) after the signal is given to assure that, under accident conditions, the total delay in achieving full core spray flow is less than 37 sec. The safety valves on the core spray lines outside the second system isolation valve are periodically removed and tested for setpoint, as recommended by the ASME Code, Section III-B-1965. These valves are also containment isolation valves and are subject to Appendix J Type B and C testing.

The pumps and valves are tested quarterly by recycling water to the suppression chamber.

During each refueling outage, condensate water is introduced into the pump suction and automatic initiation of the pumps and valves is tested.

At least once per month verification is made that the keep-full system piping is filled with water.

Once each quarter during the scheduled operability test, the system is visually inspected for leakage, and maintenance is performed as required.

B. CONTAINMENT SPRAY SYSTEM

1.0 Licensing Basis Requirements

The following regulatory documents are applicable to the containment spray system (CSS) and, in general terms, form the basis on which the system is designed and operated.

1.1 10CFR50.49 - Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants

An EQ program for electrical equipment has been conducted in accordance with 10CFR50.49. Consequently, electrical equipment important to safety in the CSS system has been qualified to operate in the LOCA environment.

1.2 10CFR50 Appendix A - General Design Criteria for Nuclear Power Plants

The Technical Supplement to Petition for Conversion from Power Operating License to Full Term Operating License covered the Unit 1 positions relative to the General Design Criteria (GDC). Those portions of the documentation that cover both the description of the requirements and NMPC's positions relative to these requirements, as they pertain directly to the CSS system, have been extracted and are shown below:

<u>Criterion 16</u>

<u>Containment Design</u> Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment, and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

A pressure suppression containment system consisting of a drywell, suppression chamber (torus), and interconnecting vent piping is the primary containment for the main coolant system. During normal operation, the reactor building, containing the pressure suppression system, provides a secondary containment barrier.

To ensure the integrity of the primary containment, integrated leak tests were performed prior to Station operation and periodically thereafter, as provided in the Technical Specifications. The results demonstrated that the containment met the design leak rate of 0.5 percent per day at a pressure of 35 psig and, therefore, provides an essentially leak-tight barrier. The design basis LOCA was evaluated at the primary containment maximum allowable accident leak rate of 1.9 percent per day at 35 psig. The analysis demonstrates that the offsite

2.0 Design Bases

2.1 Design Basis Functional Requirements

The CSS system shall perform the following functions important to safety in order to prevent containment pressure and temperature from exceeding its design values for reactor coolant system (RCS) leaks up to and including the DBA, double-ended break of a reactor coolant recirculation line:

1. Functional Requirement - Remove energy from the drywell and torus following vessel leaks, up to and including a LOCA, to reduce containment temperature and pressure and maintain them below containment design pressure and temperature limits.

   Basis - A means of removing energy from containment following a LOCA and of transferring energy to the UHS is required by GDC 38 and GDC 44. The CSS system provides the primary means of energy removal from containment after a LOCA.

2. Functional Requirement - Ensure the torus water temperature does not exceed that required to satisfy containment spray and core spray NPSH requirements.

   Basis - Inadequate NPSH can limit the containment spray and containment raw water pump performance and reliability. Without adequate NPSH, the ability of the system to remove energy from containment may be diminished.

3. Functional Requirement - Provide the capability to isolate CSS system piping that penetrates the containment boundary.

   Basis - Unit 1 did not commit to providing isolation valves in the CSS system as would be required to satisfy GDC 56. Containment spray was originally designed as an extension of primary containment. However, Unit 1 has committed to maintaining a water seal in lieu of leak rate testing of the isolation valves.

4. Functional Requirement - The CSS system piping must provide an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.

   Basis - The CSS system was originally designed as an extension of primary containment. As such, the containment spray piping must satisfy the intent of GDC 16 and provide an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment.

2.2 Controlling Parameters

To meet the design requirements of Section VII-B-2.1, the CSS system must be capable of meeting the following operational requirements:

- CSS pump flow through the drywell sparger nozzles must be ≥3300 gpm.
- CSS pump flow through the torus sparger nozzles must be ≥300 gpm.
- CSS drywell and torus sparger spray droplet size must be ≤1000 microns.
- CSS pump flow in the torus cooling mode must be ≥2800 gpm.
- CSS shell side heat exchanger flow must be ≥3600 gpm (during containment spray).
- CSS pump available NPSH must be ≥36.6 ft for the most restrictive case (least NPSH margin) in which two pumps are operating through one strainer assembly at a total flow rate of 7684 gpm.
- CSS raw water pump flow, through the heat exchanger tube side, must be ≥3000 gpm.
- CSS raw water pump available NPSH must be ≥31 ft.
- CSS drywell and torus sparger nozzle pressure must be ≥30 psi above containment pressure for a sufficient number of nozzles to achieve minimum required flows.
- CSS spray header pressure must be 110 percent of containment pressure or ≥38.5 psig.
- CSS heat exchangers must be capable of removing at least 120 million Btu/hr, with two containment spray pumps operating and a spray water temperature reduction from 140°F to 100°F.

3.0 System Design

3.1 System Function

The CSS system is an engineered safeguards system designed to prevent overheating and overpressurization of the containment, and to control the pressure suppression chamber water temperature the applicable sections of ASME Code, Section III-B-1965, and the ASA B31.1-1955 Piping Code with nuclear interpretations.

Nitrogen supply for a complete purge of the primary containment to less than 4-percent oxygen is stored in liquid form in a bulk nitrogen storage tank. This tank is connected to a steam vaporizer which can deliver 300,000 scfh of gaseous nitrogen at 50°F. Steam for vaporizing the nitrogen is provided by an electric boiler. Two methods of purging the primary containment are employed; the preferred "Continuous Feed and Bleed Method" or the alternative "Batch Blow Method." Approximately 600,000 cu ft of gaseous nitrogen are required to initially inert the primary containment. Operation of the drywell cooler fans provides for maximum mixing within containment during purging.

Nitrogen required for makeup during normal operation is supplied by two redundant nitrogen supplying systems (Section VII-G.3.1). Each system can supply 0-100 cfm at 50-60°F gaseous nitrogen to the drywell and torus. Makeup is initiated manually from the control room.

Normally, the nitrogen is exhausted directly to the stack by a fan rated at 10,000 cfm. The nitrogen leaves the drywell at about 6,000 cfm through the upper purge and vent line. The gas leaves the suppression chamber at approximately 4,000 cfm by the same purge and vent line through which it entered. Should the nitrogen atmosphere be significantly contaminated, it can be passed through the emergency ventilation system at a rate of not more than 1,600 cfm.

The system is also provided with a hardened vent path which bypasses the containment vent and purge fan. This path provides emergency containment venting capability under degraded accident conditions.

2.2 Design Evaluation

By purging the primary containment with nitrogen, the oxygen content of the primary containment atmosphere is reduced to 4.0 percent or less by volume. This initial inerting of the primary containment is sufficient to prevent a flammable hydrogen-oxygen mixture from accumulating if a metal-water reaction were to occur immediately following a LOCA.

3.0 Containment Atmospheric Dilution System

3.1 System Design

The CAD system is designed to limit the oxygen concentration of the primary containment atmosphere to less than 4.0 percent during a LOCA. Following a LOCA, hydrogen and oxygen may be released within the primary containment from postulated metal-water reactions and from radiolysis. The initially inerted primary containment prevents the combustion of hydrogen evolved from a metal-water reaction. However, radiolytic decomposition results in the release of both hydrogen and oxygen. The CAD system functions by adding nitrogen to the primary containment atmosphere as the radiolytic formation of oxygen occurs. Oxygen concentration is, therefore, diluted to remain below 4 percent by volume. Since the radiolysis rate decreases with time as a result of fission product decay, the required nitrogen addition rate will also decrease with time.

Nitrogen required for CAD system operation is supplied by two redundant nitrogen supply systems. Each nitrogen supply system (Figure VII-12) consists of a storage tank, vaporizer, electric heater and all required piping, valves and instrumentation. Discharge from either system is to the normal containment inerting system piping downstream of the two isolation valves.

The preferred nitrogen supply system used under accident conditions utilizes the same nitrogen storage tank that supplies the containment inerting system. This storage tank contains 11,300 gal of liquid nitrogen. The redundant nitrogen supply system utilizes a 4000-gal capacity storage tank.

Each nitrogen supply system is designed to supply 0-100 cfm at 50-60°F gaseous nitrogen to the drywell and suppression chamber.

The system also employs a containment venting capability. This capability provides for venting the primary containment through the emergency ventilation system or to the main condenser. It also provides vent paths with pressure control valves to ensure that the downstream pressure does not exceed 0.5 psig. The level of radioactivity in the atmosphere inside the primary containment is monitored, should the necessity arise for venting after a LOCA.

Two redundant hydrogen and oxygen sampling systems (Figure VII-13) are also an integral part of the CAD system. They continuously monitor the hydrogen and oxygen concentrations within the drywell and suppression chamber to minimize sampling errors. Two sampling probes check the drywell atmosphere while two sampling probes check the suppression chamber atmosphere. A continuous indication of hydrogen concentration (0-20 percent) and oxygen concentration (0-5 percent and 0-25 percent) in the primary containment atmosphere is provided in the control room.

All equipment was designed to operate in the most severe environmental conditions. The environmental qualification of the system components has been reevaluated through the equipment qualification project.

CAD system operation is controlled completely from the control room. Primary containment pressure (Section VIII-C.2.0) is also monitored and displayed in the control room.

common supply duct, a dual bank of filters for removal of particulates and halogens, a 1,000-W heater in each filter bank, a motor-driven fan in each bank, isolation valves at the supply and exhaust of each bank, and separate discharge ductwork from each fan provided with independent flow nozzles and flow control instrumentation.

The duct work and other equipment are designed to operate in a reactor building environment of 150°F, based on the worst-case accident for the secondary containment. The ductwork is also designed to withstand 0.5 psig, negative or positive. This pressure is based on the pressure head developed by the fans.

Each filter bank has a rated flow capacity of 1,600 cfm with the building at negative pressure of 0.25 in W.G. relative to the outside. One of the two filter banks is considered as a full-capacity spare, since each is capable of one complete change of air in the reactor building per day and performing the required filtering duty. However, upon a high radiation signal from the reactor building normal ventilation system or refueling platform, both loops will be activated until one loop is shut down by an Operator. The filter banks have a common supply header, but have independent exhaust ductwork from each fan, thereby satisfying the single failure criterion. A system isolation valve is provided in the common supply header (i.e., connection of normal and emergency ventilation systems). An isolation valve is also provided in the supply duct to each filter bank. Provisions have also been made to admit turbine building air to each filter bank for cooling, should a filter become overloaded or damaged, and removal from service becomes necessary. If a filter system were shut down after an accident, the other fan could draw outside air through the idle filter. A maximum of 60 cfm of turbine building air can be admitted to the inlet of the idle filter. The fan associated with the operating filter draws the turbine building air through the idle filter by means of a valved interconnection located between the filter outlets and the fans. An electric heater is provided in the common supply duct of sufficient capacity (10 kW) to reduce the relative humidity from 100 percent to 70 percent. Since both trains actuate simultaneously, the humidity can be as high as 80 percent for a time period of up to 30 min. This will allow for Operator action to turn one train off. However, filter efficiency remains at 95 percent or greater as demonstrated by testing required by Section 3.4.4.c of the Technical Specifications. This assures that filter efficiencies remain high since, in the extremely high humidity ranges (approximately 95 percent), efficiency is adversely affected.

Each filter bank includes the following, in sequence of treatment:

1. A high-efficiency particulate absolute (HEPA) filter, water resistant, capable of removing 99.97 percent minimum of particulate matter which is 0.3 micron or

larger in size. Filter design is fire resistant for temperatures up to 500°F.

2. A charcoal filter, with activated and specially impregnated carbon, which was originally specified to be capable of removing 99.0 percent of radioactive methyliodide and other iodine forms. Actual iodine removal efficiency is verified by testing in accordance with the Technical Specifications. Tests[2,3] have demonstrated that impregnated charcoal filters are capable of adsorbing organic iodines up to at least 3,500 μg CH_3I/gm charcoal, which is much greater than the design basis load of about 936 μg CH_3I/gm. The filters are cooled by the normal air flow of 1,600 cfm. Provisions have been made for additional cooling should the situation arise. A 1,000-W heater is provided for each charcoal filter to prevent condensation when the system is first placed in service. Tests have shown that high efficiencies can be maintained even under reasonably high humidity conditions (~70 percent).[4,5] The 10-kW duct heater reduces humidity within the ranges covered by these tests.

3. A second HEPA filter, following the iodine filter, is also capable of removing 99.97 percent minimum of particulates larger than 0.3 micron. This filter is provided to collect any particles which might become dislodged from the charcoal filter.

The emergency ventilation system with gas cleaning equipment is placed in operation automatically when the normal reactor building ventilation system is automatically shut down and isolated. Isolation of the reactor building ventilation system and startup of the emergency ventilation system occur upon high radiation in the discharge line to the normal system exhaust fans, or from high radiation at the refueling platform during refueling operation; both loops will be activated until one loop is shut down by an Operator. The system can be manually initiated.

This system can be used as an alternate discharge system for reactor vessel venting if flooding of the pressure suppression system and the reactor vessel is required.

2.1 Operator Assessment

For each of the two emergency ventilation fans a control room, panel-mounted control switch with indicating lights is provided. Position indicator lights for fan inlet and outlet valves are furnished. Separate flow control instrumentation monitors the discharge flow rate in each independent exhaust duct. Low flow alarms are annunciated in the control room. Differential pressure switches across each of the absolute filters and the charcoal filters monitor plugging of these filters and provide a high differential pressure alarm to the control room annunciator and the Station computer. A filter inlet high temperature alarm is also provided for each system.

All sensors are in accessible locations and are provided with suitable valving for in-place testing at any time.

3.0 Design Evaluation

The emergency ventilation fans discharge a volume equivalent to 100 percent of the building volume per 24 hr through high efficiency particulate and charcoal filters. The filter heating values are calculated from the amount of radioactive iodine which would be available to be deposited on the charcoal filters, based on a leakage rate of 1.5 percent per day for containment. The total iodine deposited in the filters would be approximately 936 µg CH_3I/gm charcoal which is well within the filter design capabilities of 3500 µg CH_3I/gm charcoal. A bypass arrangement utilizing turbine building air is provided to assist in cooling the filters, should one bank become overloaded and have to be removed from service. To maintain the filter temperature below 500°F, only about 10 cfm of turbine building air at 100°F would be required for the decay heat load generated by the total iodine deposited. Charcoal ignition temperature is approximately 650°F. At the full heat load the bypass arrangement described above can supply 60 cfm of cooling air. With this amount of cooling air, charcoal temperature would not exceed 200°F.

The capability of the system to maintain a negative pressure of 0.25 in of water with only one fan will prevent exfiltration from the reactor building.

4.0 Tests and Inspections

Particulate filters are shop tested with DOP (dioctylphthalate) for a minimum removal efficiency of 99.97 percent. Immediately prior to installation each filter is thoroughly inspected for damage, tears and pinholes by illuminating the back side with strong light. Any such damage is cause for rejection.

After installation the filters are tested to demonstrate that they are undamaged and properly sealed in place. For the particulate filters the test consists of injecting DOP upstream of the filter and surveying the downstream side of the filter for leaks. The DOP is introduced in a manner to provide good mixing, and care is taken to keep the mixture from the charcoal filters. For the charcoal filters, the test consists of injecting methyliodide or other halogenated hydrocarbon upstream of the filters and surveying the downstream side of the filters for the material injected.

The emergency ventilation system is normally a standby system which must perform only in the event of an accident. To assure that the filters have not deteriorated or lost capacity, periodic efficiency testing is performed in accordance with Technical Specifications.

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

Demonstration of the automatic initiation capability and operability of filter cooling (at least once per operating cycle) assures system performance capability.

Test connections installed upstream of each flow element are available to utilize portable test equipment to verify the accuracy of the flow elements.

I. HIGH-PRESSURE COOLANT INJECTION

1.0 Design Bases

The high-pressure coolant injection (HPCI) system is an operating mode of the feedwater system available in the event of a small reactor coolant line break which exceeds the capability of the CRD pumps (0.003 ft^2). HPCI, along with one emergency cooling system, has the capability of keeping the swollen reactor coolant level above the top of active fuel (TAF) for small reactor coolant boundary breaks up to 0.063 ft^2 for at least 1000 sec. The HPCI system, with one of the two emergency cooling systems and two core spray systems, will provide core cooling for the complete spectrum of break sizes up to the maximum design basis recirculation discharge line break (5.446 ft^2). Its primary purpose is to:

1. Provide adequate cooling of the reactor core under abnormal and accident conditions.

2. Remove the heat from radioactive decay and residual heat from the reactor core at such a rate that fuel clad melting would be prevented.

3. Provide for continuity of core cooling over the complete range of postulated break sizes in the primary system process barrier.

HPCI is not an engineered safeguards system and is not considered in any LOCA analyses. It is discussed in this section because of its capability to provide makeup water at reactor operating pressure.

2.0 System Design

The HPCI system utilizes the two condensate storage tanks (CST), the main condenser hotwell, two condensate pumps, condensate demineralizers, two feedwater booster pumps, feedwater heaters, two motor-driven feedwater pumps, an integrated control system and all associated piping and valves. The system is capable of delivering 6840 gpm into the reactor vessel at reactor pressure when using two trains of feedwater pumps. The condensate and feedwater booster pumps are capable of supplying the required 3420 gpm at approximately reactor pressures up to 332 psig*. Above 332 psig, a motor-driven feedwater pump is necessary to provide the required flow rate.

The feedwater system pumps have recirculation lines with air-operated flow control valves to prevent the pumps from operating against a closed system. In the event of loss of air pressure, these valves open, recycling part of the HPCI flow to

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* 332 psig provides for system pump degradation of 10 percent.

the hotwell. HPCI flow would be reduced to approximately 2600 gpm at a reactor pressure of 1,150 psig and 3420 gpm at a reactor pressure of 940 psig.

Condensate inventory is maintained at an available minimum volume of 180,000 gal.

3.0 Design Evaluation

During a LOCA within the drywell, high drywell pressure due to a line break will cause a reactor scram. This automatic scram will cause a turbine trip after a 5-sec delay. Feedwater flow would be available for considerable time from the shaft-driven feedwater pump. The shaft-driven feedwater pump would coast down while the electric motor-driven condensate pumps and feedwater booster pumps would continue to operate. The coastdown time to reach 3420 gpm delivery to the core is approximately 3.2 min (Figure VII-17), since both the condensate and feedwater booster pumps will continue to operate on offsite power. The curve on Figure VII-17 shows how flow from the shaft-driven feedwater pump decreases as the main turbine is coasting down following a trip. The curve is a representation of the feedwater capability of the shaft-driven pump after a turbine trip at a set of finite conditions. The margin to reach the 3.2-min coastdown time is governed by the turbine coastdown rate and the shaft-driven pump, not system resistance such as flow control valve (FCV) position.

The turbine trip will signal the motor-driven feedwater pump to start. The signal will be simultaneous with the start of the shaft pump coastdown. The motor-driven feedwater pump will be up to speed and capable of supplying 3420 gpm in about 10 sec. As a backup, low reactor water level will also signal the motor-driven pump to start. The initiation signal transfers control from the normal feedwater to the HPCI instrumentation and controller which has been continuously tracking the normal feedwater control signal. To maximize the NPSH to the motor-driven feedwater pumps when operating in HPCI mode, #11 flow control valve (FCV11) for #11 motor-driven feedwater pump (FWP11) does not open if there is sufficient total feedwater flow into the reactor. FCV11 remains closed until total feedwater flow into the reactor drops below 4.5×10^6 lbm/hr (9000 gpm). This logic is bypassed if FWP12 is not running or locked out. In addition, the level setpoint setdown controller (ID66B) limits the controller output to 70 percent of maximum following HPCI actuation. Feedwater flow will continue to be provided by the shaft-driven feedwater pump during turbine coastdown. Thus, there will be a continuous supply of feedwater to the reactor.

The HPCI single element control system will attempt to maintain reactor vessel water level at 65 in or 72 in (depending upon which pump, 11 or 12, respectively, is in service) with a maximum feedwater flow limit of 3420 gpm.

A sustained high reactor water level RPS signal coincident with an open feedwater flow control valve will selectively trip the associated feedwater pump. The clutch of the shaft-driven pump will also be disengaged immediately upon high reactor water level. Independent of the original high water level trip installed to meet NUREG-0737 commitments, a nonselective backup trip of the motor-driven feedwater pumps will be actuated if reactor water level remains high.

Should the reactor water level reach the low level scram setpoint, the motor-driven pump that tripped on high reactor water level will restart. Necessary feedwater pump recirculation is provided to allow for continued pump operation with the FCV closed.

As feedwater is pumped out of the condenser hotwell, through the selected equipment of the condensate and feedwater systems and into the reactor, the condenser hotwell level will fall. Since condensed steam from the turbine no longer replenishes the

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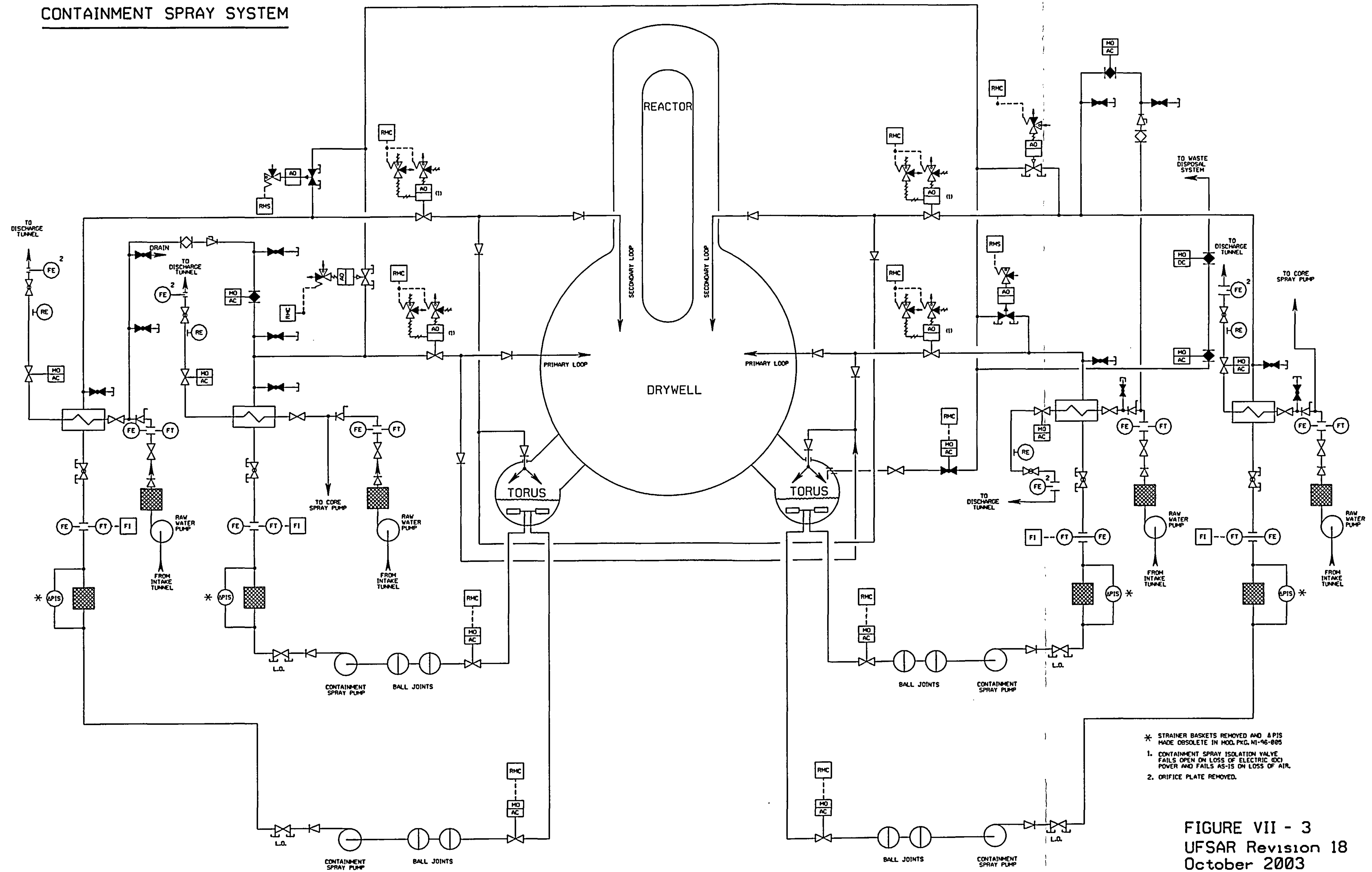
CONTAINMENT SPRAY SYSTEM



* STRAINER BASKETS REMOVED AND ΔPIS MADE OBSOLETE IN MOD. PKG. N1-96-005

1. CONTAINMENT SPRAY ISOLATION VALVE FAILS OPEN ON LOSS OF ELECTRIC (DC) POWER AND FAILS AS-IS ON LOSS OF AIR.

2. ORIFICE PLATE REMOVED.

FIGURE VII - 3
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3. Liquid poison injection system - manually.

4. Emergency ventilation system is initiated by high radiation level in reactor building ventilation system exhaust, or high radiation at the refueling platform if the bypass switch is in the refuel mode.

The emergency cooling system is initiated manually or automatically by high reactor pressure if the high-pressure condition persists for 12 sec, or by low-low reactor water level after a 12-sec time delay (to assist in depressurization for small breaks). High steam flow on the condenser tube side, indicative of an emergency cooling system line break, automatically isolates the affected set emergency condensers (EC). High radiation in the vent line or high area temperature provide alarm function only. Operator action is required to isolate one set of emergency cooling condensers.

Control rod withdrawal is prohibited by the following conditions:

1. Fuel hoist loaded with fuel and over the reactor.

2. Rod worth minimizer (RWM) below a preset power level if established withdrawal sequence is not followed.

3. High neutron flux (setpoint varied with recirculation flow).

4. Neutron monitoring instrumentation off-normal.

5. Mode switch in shutdown.

6. Withdrawal of more than one rod is prohibited with mode switch in the refuel mode.

7. Bypass of high water level scram in scram dump volume.

Offgas and vacuum pump isolation is initiated by the following:

1. Offgas

   a. High radiation from offgas line.

   b. Manually.

2. Vacuum Pump

   a. High radiation from MSL.

   b. Manually.

Diesel generators are initiated by the loss of ac voltage to power boards (PB) 102 and 103, or by a persistent degraded voltage condition to these power boards.

High-pressure coolant injection (HPCI) is initiated by any of the following:

1. Low reactor water level.
2. Automatic or manual turbine trip.
3. Manually.

The control room emergency ventilation system is initiated manually or automatically by any one of the following conditions:

1. Low-low reactor water level.
2. High drywell pressure.
3. High steam flow in MSL.
4. High temperature in MSL tunnel.
5. High radiation in the air intake.

The protective system components and their associated electrical cables located within the primary containment were designed to operate in an environment of 150°F and 100-percent relative humidity. The components and electrical cables which are required to function during and following loss-of-coolant accident (LOCA) are expected to withstand the accident conditions. The environmental qualification (EQ) of system components is controlled through the equipment qualification program. The seismic criterion observed in the design of the protective system components was that equipment would successfully withstand forces resulting from acceleration factors of 0.20g horizontal and 0.10g vertical.

1.2 Anticipated Transients Without Scram Mitigation System

A redundant anticipated transient without scram (ATWS) mitigation system is designed to mitigate the effects of an ATWS event. This supplementary protection system utilizes alternate rod injection (ARI) and reactor RPT which will increase the reliability of the present scram system, thereby decreasing the probability of an ATWS event.

ATWS is initiated automatically by the following conditions:

1. Low-low reactor water level, or
2. High reactor pressure

Primary Containment Pressure

An abnormal rise in primary containment pressure could indicate a rupture of, or excessive leakage from, the primary system within the drywell. In addition to a scram, isolation of the drywell as described in Section VI-C is initiated. The scram setting is ≤3.5 psig. Following a line severance, the containment pressure rises so rapidly (Section VI-A) that trip point errors are not critical. A high primary containment pressure signal combined with low-low-low reactor water level initiates opening of solenoid-actuated relief valves to depressurize the reactor on a small line break. Primary containment pressure greater than the high-pressure setpoint, combined with a low-low reactor water level, will initiate the containment spray system.

Reactor Water Level

A reactor low water level signal (at 1 ft below the normal water level control range) scrams the reactor when there is about 12 ft of water above the core. HPCI is also initiated on a low water level signal. Continued decrease in level to low-low (5 ft below the normal water level control range) will trip the recirculation pumps, initiate the core spray pumps, close reactor vessel isolation valves, close all containment isolation valves, isolate cleanup and shutdown cooling system, and initiate the containment spray system if drywell pressure is equal to or greater than the high-pressure setpoint. Continued decrease to low-low-low level (-10 in indicator scale) will open solenoid-actuated relief valves if primary containment pressure is equal to or greater than 3.5 psig. The maximum instrument error will not affect the results of the transient analyses for feedwater pump trip nor the LOCA, both of which are discussed in Section XV.

Scram Dump Volume Level

High level in the scram dump volume scrams the reactor while there is still sufficient free volume in the scram discharge system to receive the control rod drives (CRD) discharge from a scram. The trip point is set with at least one scram volume above the water level. Adequate allowance is made for the closed gas volume above the water.

Condenser Vacuum

The low condenser vacuum anticipates loss of the main heat sink. A reduction in vacuum will initiate closure of turbine stop valves at ~20 in, turbine bypass valves at ~10 in, and MSIVs at ~7 in Hg vacuum.

Main Steam Line Radiation

The radiation monitors on the steam line in the pipe tunnel near the isolation valves will isolate the mechanical vacuum pumps on a high radiation signal and initiate an annunciator in the control room.

Position of Main Steam Isolation Valves

Partial closure of MSIVs in both steam lines produces a scram so that the reactor is not operated without its main heat sink. The trip switches are set at about 10 percent from full open.

Turbine Trip

1. Generator Load Rejection

   The generator load rejection scram is initiated by the signal for turbine control valve fast closure due to a loss of oil pressure to the acceleration relay any time the turbine first-stage steam pressure is above a value corresponding to 833 MWt, i.e., 45 percent of 1850 MWt.

2. Turbine Stop Valve Closure

   The turbine stop valve closure scram is initiated at ≤10 percent of valve closure setting (stem position) from full open whenever the turbine first-stage steam pressure is above a value corresponding to 833 MWt, i.e., 45 percent of 1850 MWt.

a high flux scram would occur. Consequently, a backup electrical interlock system (described in Section VIII-C.1.1.4) is available to prevent rod withdrawal to outside the acceptable power and flow range.

2.3 Pressure and Turbine Control

Control and supervisory equipment for the turbine generator is arranged for remote operation from the control room. The turbine control includes the customary speed governor, overspeed governor, control valves, turbine stop valves and combined reheat intercept valves. Normally, the electrical pressure regulator controls the turbine control valve position to maintain constant turbine throttle pressure. The ability of the Station to follow system load requirements is accomplished by adjusting the reactor power level, either by regulating the reactor recirculating flow or by moving control rods. However, the turbine overspeed governor can override the initial pressure regulator, and the turbine control valves will close when an increase in system frequency or a loss of generator load causes the speed of the turbine to increase. In the event that the reactor is delivering more steam than the control valves will pass, excess steam up to approximately 2,500,000 lb/hr flow will be bypassed directly to the main condenser automatically by the turbine bypass valves.

A block diagram of the turbine control is shown on Figure VIII-4. A single pressure regulator with a backup regulator is used to control both the turbine control valves and the turbine bypass valves. The two sets of valves are coupled together by a linkage system.

Normally, the bypass valves are held closed and the pressure regulator controls the turbine control valves, utilizing all the steam production to make electrical power. If the governor or load limit reduces the steam flow to the turbine, the regulator controls the turbine throttle pressure by opening the bypass valves. If the capacity of the bypass valves is exceeded when the governor or load limit reduces the steam flow to the turbine, the reactor pressure will rise and ultimately scram the reactor.

A second, or backup, mechanical pressure regulator is provided to limit reactor pressure automatically in the event that the operating regulator should fail. The setpoint of the backup pressure regulator is normally a few psig above the setpoint of the operating pressure regulator. Revised requirements for MCPR and LHGR for operation without a backup pressure regulator are contained in the COLR. These limits are based on a TCV closure speed ≥5 sec (Reference 32).

The linkage system also contains a mechanical stop arrangement (lift limit) to limit the total steam flow. This mechanical stop is adjustable with a maximum position at about 110 percent of turbine control valves-wide-open (VWO) steam flow.

2.4 Reactor Feedwater Control

Water flow to the reactor vessel is controlled by a three-element level control system. This system uses measurements of steam flow to turbine, feedwater flow to reactor, and reactor water level to modulate feedwater control valves, in order to maintain water supply to the reactor in direct proportion to the reactor steam output and to hold the specified reactor water level.

In addition to the three-element mode, there are also single-element, low-flow and HPCI modes of feedwater control. The HPCI mode bypasses all other modes of operation (see Section VII-I.3).

Each reactor feedwater pump has conventional throttling recirculation controls, which pass feedwater back to the condenser when individual feed pump flow is below minimum flow required to cool the pumps.

Reactor water level, feedwater flow, and steam flow are recorded in the control room. High and low reactor water level is annunciated.

3.0 System Evaluation

3.1 Control Rod Adjustment Control

The core nuclear characteristics, as described in Section IV, have established restrictions pertaining to the maximum amount and rate of reactivity addition. Such design restrictions are imposed in order to limit potential consequences which might arise from reactivity insertion accidents, as described in that section. In order to provide reasonable assurance that those reactivity addition restrictions can be achieved without relying solely on operating procedures, a hydraulic system has been provided for the CRDs. The hydraulic system is evaluated in Section X-C.

3.2 Recirculation Flow Control

Malfunction of the recirculation flow control system is discussed in Section XV.

3.3 Pressure and Turbine Control

Malfunction of the pressure and turbine control system is discussed in Section XV.

3.4 Reactor Feedwater Control

Malfunction of the reactor feedwater control system is discussed in Section XV.

Air Ejector Offgas Monitor

The air ejector offgas monitoring system, shown on Figure VIII-24, continuously monitors the radioactivity level of the effluent gases removed from the main condenser by the steam jet air ejector (SJAE) system.

Two channels of instrumentation are provided. A channel consists of an ionization chamber, a six-decade logarithmic amplifier, and a shared two-pen recorder. The logarithmic amplifier is equipped with upscale trip and downscale alarm.

In normal operation, a sample of gas is drawn from the offgas line into a special section of pipe where it is seen by the ionization chambers. The sample is returned to the condenser, the low pressure point in the system. The holdup time in the sample line allows for approximately 2 min delay of the N-16 and O-19 so that the activity of the isotopes signaling the presence of a ruptured fuel element is not masked. The output of each channel is recorded continuously on one pen of a two-pen recorder. The other pen is used by one channel of the stack gas monitor. Two such recorders are provided. A continuous recording of offgas flow and sample flow is also provided in the control room. Low sample flow is annunciated.

When the radiation level of the offgas exceeds the maximum offgas vent release rate, control action is initiated to close the offgas isolation valve immediately. A holdup volume in the offgas line after the sample point provides a 30-min delay after the high radiation signal before the radioactivity passes the downstream isolation valve. Therefore, automatic isolation occurring up to 30 min after the high radiation signal prevents highly radioactive materials from being discharged. The system includes three trips; one downscale, one high and one high-high. The downscale and high-high trips are initiated by the radiation monitor itself while the high alarm is initiated by the recorder. A downscale trip gives warning of instrument malfunction. The two channels are so arranged that they operate independently of each other. The logic is so arranged that a closure of the offgas line is initiated by two high-high level signals, or an upscale in combination with a downscale.

Means are provided to take routine grab samples of the offgas so that the offgas monitors can be calibrated.

Emergency Condenser Vent Radiation Monitor (Figure VIII-25)

Monitoring of gross radiation is provided for each EC vent line with two detectors. The amplifiers associated with the detectors are logarithmic with a range of 0.1 to 1000 mr/hr. The detectors are identical to those used for the ARMS. The output of each monitor is indicated in the control room. When the gross activity in a condenser vent line reaches a preset level during system operation, indicating tube leaks in the EC, an alarm is sounded in the control room. Operator action is required to isolate one set of emergency cooling condensers from the rest of the primary system. Isolation of the EC loop is initiated manually.

3.1.2 Other Radiation Monitors (Figures VIII-26 and VIII-26a)

Stack Effluent Monitors

1. OGESMS

The offgas effluent stack monitoring system (OGESMS) stack monitor continuously monitors the activity of the gas released through the stack. A sample is collected by an isokinetic probe located about ten stack diameters above the highest point at which gases enter the stack. The sample is passed through a particulate filter and a halogen filter before being introduced to four scintillation detectors monitoring the stack gas sample. The sample of monitored gas is pumped back into the stack.

Two of the detectors are connected to a seven-decade log count rate meter, and are calibrated to monitor radiation in the 10^{-7} uCi/cc to 1 uCi/cc range. The remaining two detectors are connected to a five-decade meter and are calibrated to monitor radiation in the 10^{-7} uCi/cc to 10^{-2} uCi/cc range. The lower range detectors meet the lower limit of detection (LLD) requirement in the Offsite Dose Calculation Manual. The filters are removed periodically and analyzed for particulate and halogen activity. The flow of gas through the sampler is indicated and alarmed on low flow to indicate a failure of the pump or a stoppage in the filters. An installed spare pump is provided for reliability of the system.

If high radiation is sensed when this monitor is selected, the monitor sends a signal to isolate the drywell and suppression chamber vent and purge valves.

If the stack effluent monitor is inoperable, effluent sampling is performed in accordance with the Offsite Dose Calculation Manual using auxiliary sampling equipment and approved procedures.

2. RAGEMS

When selected, the radioactive gaseous effluent monitoring system (RAGEMS) continuously monitors the activity of noble gas released through the stack. A sample is collected by an isokinetic probe located in the stack. The sample is passed through a particulate and iodine filter and then into the noble gas unit for sampling. Particulate and iodine filters are counted manually in the onsite laboratory in accordance with plant procedures. The sample of monitored gas is pumped back into the stack.

The range of the noble gas monitor is 5.0E-8 to 2.0E5 µCi/cc.

If high radiation is sensed and RAGEMS is selected, the monitor sends a signal to isolate the drywell and suppression chamber vent and purge valves.

Radwaste System Liquid Effluent Monitor

The radwaste system liquid effluent monitor provides a radiation level indication of the radwaste system liquid discharges. The monitor consists of a gamma-sensitive scintillation detector mounted in a shield which surrounds the pipe containing the liquid being monitored. The sensitivity of the monitor can be adjusted up to a factor of 100 from 10^{-5} to 10^{-3} uc/ml for 5 cps.

The scintillation detector is connected to a seven-decade log count rate meter which is equipped with an upscale trip for high-level alarm. The log count rate meter also provides a level signal to the Station computer.

Reactor Building Cooling Water Monitor

The reactor building cooling water system return line is continuously monitored for radioactivity concentration levels to

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detect any in-leakage of primary system contaminants through any of the system heat exchangers. The monitor is identical to the radwaste system effluent monitor described in "Radwaste System Liquid Effluent Monitor" above.

Service Water Monitor

An off-line liquid monitor is used to take alternating 15-min samples from the turbine building service water return line and reactor building service water return line. Samples are pumped past a shielded gamma-sensitive scintillation detector. Samples are returned to service water discharge line downstream from the sampling locations. Detector sensitivity ranges from 1E-07 to 1E-01 uCi/ml for Cs-137.

Radioactivity concentrations in excess of Technical Specification limits result in control room annunciation and provide a signal to the Station computer. Control room annunciation also occurs if the monitor fails or becomes inoperable.

Grab samples are taken as required by the Offsite Dose Calculation Manual using approved procedures.

Surveillances are performed in accordance with the Offsite Dose Calculation Manual and approved procedures.

Containment Spray Heat Exchanger Raw Water Effluent Monitor (Figure VIII-27)

The raw water effluent from each of the four containment spray heat exchangers is monitored for gross gamma activity to detect any leakage of activity through the heat exchangers when the containment spray system is operated.

The amplifier associated with each detector is logarithmic with a minimum range of .01 to 100 mr/hr. The detectors are identical to those used for ARMS. The output of each monitor is indicated in the control room. The amplifier is equipped with an upscale trip for high-level alarm and provides a level signal for the Station data logger.

Refueling Bridge High-Radiation Monitor

The refueling bridge high-radiation monitor is attached to the refueling bridge and monitors radiation levels during refueling operations. It gives an alarm both in the control room and in the refueling area to warn of abnormal radiation levels.

A radiation monitor is used with a special five-decade logarithmic readout of radiation level. The monitor has an upscale trip point to indicate an increase in radiation level and a downscale trip point to indicate instrument malfunction.

The upscale trip point will trip out the normal reactor building ventilation system and place the emergency ventilation system in service.

Containment Atmospheric Monitoring System

Containment airborne radioactivity is continuously monitored by the containment atmospheric monitoring system (Figure VIII-28). This system consists of a pump, filter with radiation monitors, electric heater and associated valves, instrumentation and piping. Samples are taken directly from the drywell atmosphere, passed through a filter (where the sample is monitored for gross beta-gamma radioactivity), then returned to the drywell by way of a hydrogen-oxygen sample return line.

A continuous indication of the containment airborne activity is provided in the control room.

Drywell High-Range Radiation Monitors

Two high-range radiation detection and indication units are provided to measure the drywell radiation in the range of 10^0-10^8 R/hr. The detectors are located in separate drywell mechanical penetrations and are powered from emergency power sources. Each detector has an indicator and alarm trip unit located in the control room. Radiation levels are continuously recorded on a two-pen recorder. Both monitors have inputs to the computer in the control room and Technical Support Center (TSC).

Both radiation monitors meet the requirements of Regulatory Guide (RG) 1.97.

Area Monitors

Area monitoring systems are described in Section XII-B.2.0.

3.2 Evaluation

Radiation monitors, which provide signals to the protective system for isolation valve closure, equipment isolation, or unit shutdown, are designed so that a single component failure does not prevent the required automatic action. All monitors are capable of self-supervision, i.e., give an alarm when downscale or de-energized. Alarms are also provided to give warning if the monitor's sampling system flow is low. All monitors are capable of convenient, operational verification by means of test signals of radioactive check sources.

All monitors are provided with continuous indication in the control room. As a general requirement, the critical process monitors are capable of initiating appropriate alarms and/or actuating control equipment in order to assure containment of radioactive materials, if preestablished limits are exceeded.

4.0 Other Instrumentation

4.1 Rod Worth Minimizer

4.1.1 Design Bases

The RWM system prevents the selection of a control rod which could allow individual control rods to have greater than acceptable reactivity worth and result in excess reactivity within the core (>0.013Δk). The RWM accomplishes this by forcing adherence to established control rod sequences during reactor startup, shutdown and periods of reduced power operation. The RWM also serves to limit control rod worths so that in the event of a control rod drop from the reactor core, the reactivity addition would neither lead to damage to the primary coolant system nor produce significant fuel damage. The RWM is intended to monitor and reinforce prescribed procedures and, therefore, causes minimum interference with desired Station operation.

The RWM system consists of the following components:

1. Rod Position Information System (RPIS)
2. Output Buffer
3. Process Computer
    a. I/O Typer
    b. RWM Program
4. Display Panel
5. Keylock Bypass Switch

A discussion of the RWM development program is included in GE Report No. APED 5449.

The block diagram shown on Figure VIII-29 illustrates the central role of a process computer in the RWM system. The communication channels involve the RPIS and output buffer. The RPIS serves to couple Station input data (control rod position, feedwater, indication of power level and system control signals) to the input/output (I/O) control. The output buffer couples the computer output commands to the manual control system to produce the control rod insert and withdrawal blocks.

Three man-machine communication channels are incorporated into the system. The I/O Typer channel is used to load the RWM program and provide a typed record of most system errors. The display control panel provides the Station Operator with a status display in the control room, and allows the Operator to perform a limited number of control actions with the RWM system. The video displays allow a graphic display of the control rods.

A keylock switch, located in the control room, permits the Operator to bypass the RWM system in the event of equipment failure. When bypassed, the control rod outputs are switched to the permissive state and remain in that configuration until the RWM system is subsequently restored to operation. Control rod dropout accidents analyzed in Section XV indicate that less fuel damage occurs as higher initial power levels are considered for the accident. Based on analysis in Reference 33, it was concluded that the RWM is not required above 10 percent power. Therefore, an adjustable setpoint is provided to automatically remove RWM constraints when above the setpoint.

Both steam flow and feedwater flow provide inputs to the RWM as indirect measurements of reactor power. On decreasing power, either the steam flow input or the feedwater flow input will trip the low power setpoint (LPSP) above 10 percent reactor power to enable the RWM. On increasing power, both steam flow and feedwater flow inputs are required to disable the RWM above the LPSP. After the LPSP has been exceeded, the RWM does not inhibit rod selection or movement.

RWM logic is incorporated into the plant process computer via the control rod scan and control rod alarm programs. Control rod select-and-position data are applied simultaneously to the control room display panel and the plant process computer data input lines from the RWM RPIS. Thus, the process computer performs rod position scanning.

The desired control rod movements are stored in the process computer memory together with the actual rod positions. The preestablished control rod pattern is entered into the computer by means of the I/O Typer; the actual rod position data is received from the control rod position indicating system. Rod selection and rod drive motion are evaluated by the computer with reference to permissible and existing control rod patterns. As long as rod operation is in accordance with the selected withdrawal or insert sequence, the RWM output is permissive. If the Operator attempts a rod selection or movement that deviates by one notch from the selected program, the RWM either alarms or blocks such action.

4.1.2 Evaluation

Detailed on-demand system diagnostic routines are provided which test the RWM program and the control interlock function. The test routine will be exercised periodically to verify proper system operation.

The RWM system is designed to be a highly reliable monitor of Operator control rod actions during Station operation. Emphasis has been directed toward two primary goals; namely, a high degree of credibility in the control rod permissive/block outputs and, simultaneously, minimum interference with the Operator during normal operation.

4.2 Offgas System Explosive Gas Monitoring

4.2.1 Design Bases

Offgas system explosive gas monitoring is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas treatment system is maintained below the flammability limits of hydrogen. Automatic control features are included in the system to prevent the hydrogen concentration from reaching these flammability limits. Maintaining the concentration of hydrogen below flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion (GDC) 60 of Appendix A to 10CFR50.

The explosive gas monitoring program requirements are described in Technical Specifications. The system is designed to withstand the effects of a hydrogen explosion. The following surveillance requirements and actions will be taken when deficiencies are identified.

Actions

1. A minimum of one hydrogen monitor on each recombiner shall be operable during offgas system operation. With the number of channels operable less than the number required, operation of the main condenser offgas treatment system may continue provided gas samples are collected and analyzed once per 8 hr. Restore the hydrogen monitoring channel to operable status within 30 days or outline in the next Radiological Effluent Release Report the cause of the inoperability and how the monitoring channel was or will be restored to operable status.

2. The concentration of hydrogen in the main condenser offgas treatment system shall be limited to 4 percent by volume. If the concentration of hydrogen in the main condenser offgas treatment system exceeds this limit, restore the concentrations to within the limit within 48 hr.

4.2.2 Surveillance Requirements

4.2.2.1 Hydrogen Monitor Operability Demonstration

Each hydrogen monitor shall be demonstrated operable by:

1. Performance of a sensor check at least once per day during main condenser offgas treatment system operation.

2. Performance of a channel test at least once per month.

3. Performance of a channel calibration at least once per 3 months. The channel calibration shall include the use of standard gas samples containing a nominal:

   a. One volume percent of hydrogen, balance nitrogen, and

   b. Four volume percent hydrogen, balance nitrogen.

4.2.2.2 Hydrogen Concentration Requirement

The concentration of hydrogen in the main condenser offgas treatment system shall be determined to be within 4 percent hydrogen by volume by continuously monitoring the waste gases in the main condenser offgas treatment system in accordance with Section 4.2.1, Item 1.

5.0 Regulatory Guide 1.97 (Revision 2) Instrumentation

Requirements for implementation of RG 1.97[3] are specified in Section 6 of NUREG-0737 Supplement 1. (NUREG-0737 Supplement 1 was issued to licensees under Nuclear Regulatory Commission (NRC) Generic Letter No. 82-33, dated December 17, 1982.)

Key features defining the basis and general approach for implementation of RG 1.97 at Nine Mile Point Nuclear Station - Unit 1 (Unit 1) are detailed in this section.

Detailed information regarding plant-specific RG 1.97 instrumentation is provided in various design criteria documents that are issued and maintained by the Nuclear Engineering and Licensing Departments.

NOTE: References cited in the subsequent portions of this section are listed and identified/described, by the reference number, in subsection D.

5.1 Licensing Activities - Background

Unit 1 was designed, constructed, and licensed to operate well before NUREG-0737 Supplement 1 and RG 1.97 were issued. As such, absolute compliance with all of the instrument design criteria specified in RG 1.97 was not (and is not) part of Unit 1's plant design basis or licensing basis.

Implementation of RG 1.97 at Unit 1 was pursued on a basis which included the performance of plant-unique reviews and evaluations of specific design criteria for selected instrumentation, as documented in various Unit 1 letters to the NRC[4-13] and in associated NRC letters and inspection/evaluation reports[14-23]. The information presented herein reflects, and is consistent with, the results of these activities. Specific actions and attributes regarding implementation of RG 1.97 at Unit 1 that were explicitly associated with obtaining NRC permission to restart in 1990 are listed and summarized in subsection 5.6.

Unit 1 commitments for assuring continuing compliance with currently established plant-specific RG 1.97 implementation criteria are detailed in Reference 10.

5.2 Definition of RG 1.97 Variable Types and Instrument Categories

Five types of RG 1.97 variables are defined for the purpose of aiding the design, specification, selection, and evaluation of accident-monitoring instrumentation. These types, and the associated definition for each, are as follows:

Type A — Those plant-specific variables that provide primary information needed to permit control room operating personnel to take the specified manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accidents (DBA).

In the context of the above definition, primary information is information that is essential for the direct accomplishment of the specified safety function; it does not include those variables that are associated with contingency actions that may also be identified in written procedures.

Type B — Those variables that provide information to indicate whether plant safety functions are being accomplished.

Type C — Those variables that provide information to indicate the potential for being breached or the actual breach of the barriers to fission product release (i.e., fuel cladding, primary coolant pressure boundary, and containment).

Type D — Those variables that provide information to indicate the operation of individual safety systems and other systems important to safety.

Type E — Those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and for continuously assessing such releases.

The recommended minimum set of Type B, C, D, and E variables for boiling water reactors (BWR) is listed in Table 1 of RG 1.97. However, no recommended list of any Type A variables is provided in the regulatory guide because these variables are acknowledged as being plant-specific and depend on the operations that the plant designer chooses for planned manual action.

The five "Type" classifications for RG 1.97 variables are not mutually exclusive. A particular variable (or instrument) may be designated as applying to more than one type, as well as to monitoring normal plant operations and/or the operation of automatically initiated safety actions.

Design, evaluation and qualification criteria for instrumentation used to monitor RG 1.97 variables are divided into three separate categories as follows:

Category 1 — Designates that instrumentation subject to design, evaluation, and qualification relative to the most stringent guidelines; applies to principal instrumentation for key variables.

Category 2	Designates that instrumentation subject to design, evaluation, and qualification relative to less stringent guidelines; generally applies to principal instrumentation for indicating system operating status.
Category 3	Designates that instrumentation subject to the least stringent design, evaluation, and qualification guidelines yet sufficient to ensure that high-quality off-the-shelf instrumentation is used; applies to backup and diagnostic instrumentation, and may also apply when the state of the art will not support the recommended use of higher qualified instrumentation.

The scheme of Category 1, Category 2, and Category 3 classifications provides a method of implementing a graded approach for RG 1.97 instrument design, evaluation, and qualification based on, and consistent with, the relative importance to safety of the measurement and display of the status of each particular variable.

The category specified for each RG 1.97 variable generally reflects whether the variable is considered to be a key variable, or a variable for monitoring system status, or a variable for backup indication or diagnosis. For example:

- For variable Types B and C, the key variables are generally designated Category 1; backup variables are generally designated Category 3.

- For variable Types D and E, the key variables are generally designated Category 2; backup variables are generally designated Category 3.

Instrumentation for monitoring key variables should be designed, evaluated, and qualified to more stringent criteria than that for monitoring system status and/or backup variables.

5.3 Determination of RG 1.97 Type A Variables for Unit 1

For the purpose of determining the Type A variables for Unit 1 using an approach consistent with the definition of Type A variables provided in RG 1.97, the accomplishment of primary safety functions for DBAs is defined as follows:

- Fuel cladding integrity is maintained (i.e., the core remains adequately cooled), and

- Reactor coolant system (RCS) integrity is maintained, and

- Primary containment integrity is maintained.

The automatic and manual actions required to assure that these principal safety functions are accomplished at Unit 1 for DBA events were investigated and evaluated. Relevant information documented in the Unit 1 Final Safety Analysis Report (FSAR) (Section XV, "Safety Analyses") and in the Unit 1 Technical Specifications[24] (principally, Technical Specifications Section 2.1.1, Fuel Cladding Safety Limit and Associated Limiting Safety System Setpoints, and Technical Specification Section 2.2.1, Reactor Coolant System Safety Limit and Associated Limiting Safety System Setpoints) was considered. Based on the results of these evaluations, it was concluded that automatic actions, initiated as specified by Limiting Safety System Settings and other reactor and containment protection system setpoint devices (e.g., reactor scram for MSIV closure, electromatic relief valve operation for high reactor pressure vessel (RPV) pressure conditions, etc.), provide adequate assurance that the identified principal safety functions are accomplished for design basis events without any need for additional manual Operator actions. Accordingly, it is therefore determined that there are no RG 1.97 Type A variables for Unit 1.

NOTE: Complete documentation of the detailed analysis supporting the determination that, for Unit 1, there are no RG 1.97 Type A variables, is provided in Enclosure 1 of Reference 8.

5.4 Determination of EOP Key Parameters for Unit 1

5.4.1 Determination Basis/Approach

Despite the plant-unique determination that for Unit 1 there are no RG 1.97 Type A variables, certain key plant parameters are monitored and controlled by manual Operator actions during accident and post-accident conditions in accordance with instructions specified in the Unit 1 Emergency Operating Procedures (EOP). Nine Mile Point Nuclear Station, LLC (NMPNS), considers monitoring the status of these "EOP Key Parameters" to be important for assuring plant safety, and thus has generally designated them as subject to having associated RG 1.97 Category 1 display instrumentation.

The process employed for determining the Unit 1 EOP Key Parameters proceeds, in sequence, as follows:

1. Define primary safety functions (the same as was done for the Type A variable review/determination).

2. Identify the association of EOPs to each of the specified primary safety functions.

3. Review the series of Operator actions specified in the identified EOPs to determine the appropriate key parameters - those principal parameters which, in

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combination and when controlled within limits specified in the EOPs, are key to assuring safe plant conditions.

5.4.2 Definition of Primary Safety Functions

Unit 1 has defined the accomplishment of primary safety functions as follows:

- Fuel cladding integrity is maintained (i.e., the core remains adequately cooled);
- RCS integrity is maintained;
- Primary containment integrity is maintained, and the conditions within the primary containment are maintained within limits associated with operability of equipment that is located in the primary containment and is important to safety.

The above definition is consistent with that presented in various documents issued by the NRC and other nuclear industry organizations (e.g., EPRI, INPO) intended for use in developing safety parameter display systems (SPDS) based on an analysis of primary safety functions. Thus, through application of this definition, close continuity is demonstrated and maintained (at the plant-specific level of implementation) among three specific emergency response activities as recommended in Supplement 1 to NUREG-0737:

- EOPs
- SPDS
- RG 1.97

Control of radioactivity releases to the environment and control of secondary containment conditions are considered contingency actions (i.e., actions that would apply given a failure to achieve one or more of the principal safety functions). Per RG 1.97, such contingency actions are exempt from consideration in the determination of Type A variables and, therefore, are also excluded from consideration in the determination of the Unit 1 EOP Key Parameters.

5.4.3 Association of EOPs to Primary Safety Functions

The stated purpose of several of the Unit 1 EOPs (as described in the supporting technical basis documents) relates directly to the accomplishment of the identified primary safety functions. In some cases, one EOP may apply to the accomplishment of more than one primary safety function; this occurs as the result of the close interrelationships that exist between the various reactor vessel and primary containment parameters - a characteristic inherent in the design of BWRs.

The association between the identified primary safety functions and the Unit 1 EOPs is shown in Table VIII-1.

5.4.4 Identification of EOP Key Parameters

The Unit 1 EOPs are based on, and developed from, the generic Boiling Water Reactor Owners' Group (BWROG) Emergency Procedure Guidelines and Severe Accident Guidelines (EPG/SAG). The EPGs/SAGs were purposely developed and constructed to be symptomatic in the approach for directing emergency response actions, and this same symptomatic approach is clearly evidenced in the Unit 1 EOPs/severe accident procedures (SAP). As a result, the EOPs/SAPs are easily analyzed with regard to plant conditions (status of parameters) and associated Operator actions.

The EOPs/SAPs Key Parameters are defined to be those reactor, RPV, and primary containment parameters that are explicitly directed to be "monitored and controlled" and/or to be "maintained above/below" a specified value (or a particular limit) in accordance with the Operator actions detailed in the EOPs/SAPs. Typically, these actions constitute a principal path within an EOP/SAP (e.g., control of RPV water level per the instructions specified in N1-EOP-2), or are an EOP/SAP unto themselves (e.g., control of drywell water level per the instructions specified in N1-SAP-1). Parameters associated exclusively with monitoring and controlling secondary containment conditions and/or releases of radioactivity outside of containment are excluded from the determination of EOP/SAP Key Parameters since they are considered contingency actions (i.e., actions that would apply given a failure to accomplish one or more of the principal safety functions). Also, by definition, system-level parameters (e.g., core spray pump flow) are not considered to be EOP/SAP Key Parameters.

The list of EOP/SAP Key Parameters (per the definition and discussion presented above), resulting from the performance of a detailed review of the Unit 1 EOPs/SAPs listed in Table VIII-1, is shown in Table VIII-2.

NOTE: Complete documentation of the detailed step-by-step analysis of the EOPs, that was conducted to produce the list shown in Table VIII-2, is provided in Unit 1 Specification Document No. NMP1-RG197-01[25].

5.5 Unit 1 RG 1.97 Variables, Variable Type, and Associated Instrument Category Designations

A composite list of the RG 1.97 variables, designated variable type (EOP, B, C, D, and/or E), and the associated instrument category (1, 2, or 3) assigned for each listed variable, is shown in Table VIII-3. Specific notes that apply to individual entries in the table (applicability as indicated immediately under the variable name) are provided at the end of the table. General comments that apply to the entire table are as follows:

1. The list of variables includes all EOP Key Parameters and all of the variables listed in RG 1.97 Table 1 ("BWR Variables"). For each variable included in RG 1.97 Table 1 that is not applicable to Unit 1, an appropriate explanation/justification is provided in the Note identified as applicable to the variable.

2. By definition, Category 1 designation for associated display instrumentation is specified to apply to variables which are EOP Key Parameters. This specification acknowledges the relative similarity between the RG 1.97 Type A variables (for those plants determined to have such variables) and the Unit 1 EOP Key Parameters, as regards the importance to safety of associated accident-monitoring instrumentation.

5.6 Summary of the RG 1.97 Instrument Design and Implementation Criteria that were Established for Unit 1 as Part of the Unit 1 1990 Restart Activities

As previously noted (in Subsection 5.1), Unit 1 was designed, constructed and licensed to operate prior to the issuance of either NUREG-0737 Supplement 1 or RG 1.97. As such, full and absolute compliance with all of the individual instrument design and implementation criteria specified in RG 1.97 was not (and is not) part of Unit 1's plant design or licensing basis. Nonetheless, implementation of recommendations presented in RG 1.97 was pursued at Unit 1 on a basis consistent with the following three standards:

1. Conformance to Unit 1 design and licensing bases;

2. Compliance with commitments made to the NRC that pertain to particular RG 1.97 issues; and

3. Safe plant operation as determined by conformance to the bases and assumptions applicable to the plant-specific analysis of DBAs (as documented in FSAR Chapter XV, "Safety Analyses") and the development and execution of EOPs.

Summarized below are the RG 1.97 instrument design and implementation criteria that were established for Unit 1 as part of the Unit 1 1990 restart activities. This summary reflects, and is consistent with, the results of plant-unique reviews and evaluations of specific design criteria for selected instrumentation, as documented in various Unit 1 letters previously submitted to the NRC[4-13] and in associated NRC inspection/evaluation reports[14-23].

Unit 1 commitments for assuring continuing compliance with currently established plant-specific RG 1.97 instrument design and implementation criteria are fully detailed in Reference 8.

5.6.1 No Type A Variables

It has been determined that for Unit 1 there are no RG 1.97 Type A variables.

For further discussion of this item, refer to the detailed information presented in Subsection 5.3.

5.6.2 EOP Key Parameters

A unique group of RG 1.97 variables has been defined for Unit 1. These variables are termed "EOP Key Parameters," and consist of the following:

- Neutron Flux - APRM
- RPV Water Level
- RPV Pressure
- Drywell Pressure
- Torus Airspace Pressure
- Drywell Ambient (Atmospheric) Bulk Average Temperature
- Torus (Suppression Pool) Water Level
- Torus (Suppression Pool) Bulk Average Water Temperature
- Primary Containment (Drywell and Suppression Chamber) Oxygen Concentration
- Primary Containment (Drywell and Suppression Chamber) Hydrogen Concentration
- Drywell Water Level

By definition, Category 1 designation is specified to apply to monitoring instrumentation for the variables which are EOP Key Parameters. This specification acknowledges the relative similarity between the RG 1.97 Type A variables (for those plants determined to have such variables) and the Unit 1 EOP Key Parameters as regards the importance to safety of associated accident-monitoring instrumentation.

For further discussion of the topic of Unit 1 EOP Key Parameters, refer to the detailed information presented in Subsection 5.4.

electrical isolation devices, and power distribution system/circuit devices, which were not previously classified as safety related have been newly classified as safety related to assure that all future planned maintenance activities, testing activities, and the procurement and installation of all replacement parts for these instrument loops is performed in accordance with the highest standards of quality. The procedure for upgrading the classification of these RG 1.97 Category 1 instrument loop components to safety related was effected as an administrative activity at the time that the new safety-related classification was initially applied to existing (i.e., currently installed) components. This philosophy and approach for upgrading EOP Key Parameter Category 1 instrument loop components to safety related was presented to the NRC during a meeting with the NRC staff in February 1989. Implementation of changes in the designated safety classification of components was performed in conjunction with the completion of associated Unit 1 1990 restart activities.

Safety-related classification for components of RG 1.97 Category 1 instrument loops for the EOP Key Parameters is documented in Unit 1 "Determination of (10CFR50) Appendix B Quality Requirements" No. 87-015. Additional relevant information on this subject is contained in Unit 1 letter NMP1L 0507(9).

5.6.6 Safety-Related Classification of Instrumentation for RG 1.97 Variable Types other than the EOP Key Parameters

Safety-related classification is also applied to instrumentation associated with RG 1.97 variable types other than the EOP Key Parameters. This classification is applied on a case-by-case basis to monitoring instrumentation for individual parameters of RG 1.97 Type B, C, D, and E variables as appropriate, considering the parameter's relative importance to safety and consistent with the existing criteria established by the Unit 1 10CFR50 Appendix B program. Such classification does not establish or define any new/upgraded Unit 1 design basis or licensing basis criteria for the monitoring instrumentation associated with those RG 1.97 variable types that are other than EOP Key Parameters.

The safety-related classification of individual instrument loops and/or components of Category 1 and Category 2 instrumentation for the listed parameters of RG 1.97 Type B, C, D, and E variables may be updated/revised consistent with the case-by-case approach and associated considerations described above.

5.6.7 Routing and Separation of Channelized Category 1 Instrument Loop Cables

Some cables of RG 1.97 Category 1 instrument loops are not routed consistent with their respective channelized power supply (RPS bus 11, RPS bus 12), and/or some cables of functionally redundant RG 1.97 Category 1 instrument loops lack the physical separation recommended by current/upgraded design guidelines for safety-related cables. For instruments that were originally intended and designed (prior to RG 1.97) solely for monitoring and indication, such routing was not (and is not) in violation of any established Unit 1 design basis or licensing basis criteria.

A detailed in-plant walkdown and evaluation of the areas through which RG 1.97 Category 1 instrument cables are routed was performed prior to the Unit 1 1990 restart, and it was confirmed that no hazard sources existed that could, through a single event, render any two functionally redundant RG 1.97 Category 1 instrument loops inoperable. On this basis, it was concluded that no immediate (short term) changes or modifications to the as-built routing of the RG 1.97 Category 1 instrument loop cables were necessary.

For further information on cable routing and cable separation for RG 1.97 Category 1 instrument loops, refer to Design Basis Reconstitution (DBR) Program criteria documents.

5.6.8 Electrical Isolation of Category 1 Instrument Loops from Associated Components that are not Safety Related

Some of the RG 1.97 Category 1 instrument loops do not have an approved Class 1E isolation device installed at the interface with the plant process computer. Also, the RG 1.97 Category 1 instrument loops for RCS pressure do not have an approved Class 1E isolation device at interfaces with the nonsafety-related feedwater control system circuitry. For instruments that were originally intended and designed (prior to RG 1.97) solely for monitoring and indication, the lack of a Class 1E isolation device at such interfaces was not (and is not) in violation of any established Unit 1 design basis or licensing basis criteria.

A Failure Modes and Effects Analysis (FMEA) was performed to evaluate the possible adverse consequences of fault conditions occurring on the nonsafety-related side of RG 1.97 Category 1 instrument loop circuit configurations which do not have an approved Class 1E isolation device at their interface with the plant process computer. The FMEA study was supplemented by maximum credible fault (MCF) testing of the same types of computer input cards actually in use. The results of the FMEA and associated MCF testing successfully demonstrated that the existing configuration of the subject instrument loops, without any modification, provided adequate isolation of the safety-related circuitry (i.e., that the safety-related circuitry continued to function at an acceptable level) for the spectrum of credible faults that could possibly occur (both direct and induced) to the associated nonsafety-related circuitry/components.

An additional study was performed to specifically evaluate the unique configuration of the RG 1.97 Category 1 RCS pressure instrument loop interfaces with the nonsafety-related feedwater control system circuitry. For this particular analysis,

29. NMPC letter to the NRC, NMP1L 0851, dated August 23, 1994, documenting commitment change regarding drywell water level recorder.

30. NRC (Office of Nuclear Reactor Regulation) letter to NMPC, dated October 26, 1994, "Proposed Deletion of Commitment to Install Drywell Level Strip-Chart Recorder for Nine Mile Point Nuclear Station Unit 1."

31. General Electric Company Nuclear Energy Report, GENE B2400005-01-01, "Nine Mile Point 1 Relief Valve Setpoint Tolerance Relaxation Evaluation," March 1999.

32. General Electric Company, GENE J11-03433-16-01-00, "Pressure Regulator Out-of-Service Calculations for Nine Mile Point Unit 1 Cycle 14," March 2001.

33. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A (latest approved revision).

34. NRC (Office of Nuclear Reactor Regulation) letter to Nine Mile Point Nuclear Station, LLC, dated September 11, 2002, "Nine Mile Point Nuclear Station, Unit No. 1 - Use of the Offgas Effluent Stack Monitoring System to Meet Regulatory Guide 1.97, Revision 2, and NUREG-0737 Guidance (TAC No. MB2443)."

TABLE VIII-3 (Cont'd.)

Type →	EOP	B			C			D			E		
VARIABLE Category →	1	1	2	3	1	2	3	1	2	3	1	2	3
37. Standby Liquid Control (Liquid Poison) System Flow Rate (Injection to RPV) (Note 14)													
38. Standby Liquid Control (Liquid Poison) System Storage Tank Liquid Level									X				
39. RHR System Flow Rate (Injection to RPV) (Note 15)													
40. RHR Heat Exchanger Tube Side (Reactor Coolant) Inlet and Outlet Temperature									X				
41. Cooling Water Flow to Engineering Safeguards Features System (ECCS) Components (Note 16)													
42. Cooling Water Temperature to Engineering Safeguards Features System (ECCS) Components (Note 16)													
43. High Radioactivity Liquid Tank Level										X			
44. Emergency Ventilation Damper Position									X				
45. Status of Standby Power and Other Energy Sources Important to Safety									X				
46. Radiation Level (equipment areas outside containment) (Note 17)													X
47. Airborne Radioactivity Releases of Noble Gases and Ventilation Flow Rate (Note 24)						X	X					X	X
48. Airborne Radioactivity Releases of Particulates and Halogens and Ventilation Flow Rate (Note 18)													
49. Environs Radiation Exposure (meters, for continuous indication at fixed locations) (Note 2)													
50. Environs Airborne Radiohalogens and Particulates													X
51. Plant and Environs Radiation (portable instrumentation)													X

TABLE VIII-3 (Cont'd.)

NOTES (Cont'd.)

The Category 3 instrument designation for this variable at Unit 1 has been reviewed and accepted by the NRC in the SER addressing Unit 1 conformance to RG 1.97 (Revision 2) dated November 19, 1986.

5. DELETED.

6. Direct monitoring of reactor coolant radioactivity concentration is not implemented at Unit 1 for the following reason:

    - The purpose stated RG 1.97 (Revision 2) for this variable is "detection of breach." Timely detection of a breach in fuel cladding integrity is able to be fully accomplished through monitoring of other variables. Specifically, these include containment radiation level (Item 21 in the Table), and analysis results from grab samples of reactor coolant obtained using the post-accident sample system (Item 56 in the Table).

    The Unit 1 decision not to directly monitor this variable is consistent with the associated BWROG evaluation, conclusion, and recommendation (Ref: BWROG submittal to the NRC regarding RG 1.97, dated April 6, 1983), and has been reviewed and accepted by the NRC in the SER addressing Unit 1 conformance to RG 1.97 (Revision 2), dated November 19, 1986.

7. Included under Item 57 in the Table.

8. Included under Item 47 in the Table.

9. "Indication of primary containment breach" is the purpose stated in RG 1.97 for monitoring this variable, and RG 1.97 recommends designating Category 2 instrumentation for this variable. The Unit 1 position is that secondary containment area radiation level is not the most appropriate parameter to use for assessing primary containment leakage or detecting significant releases, and therefore designates Category 3 instrumentation for this variable at Unit 1.

    The change from Category 2 to Category 3 instrumentation for this variable is consistent with the associated BWROG evaluation, conclusion, and recommendation (Ref: BWROG submittal to the NRC regarding RG 1.97, dated April 6, 1983).

TABLE VIII-3 (Cont'd.)

NOTES (Cont'd.)

The Category 3 instrument designation for this variable at Unit 1 has been reviewed and accepted by the NRC in the SER addressing Unit 1 conformance to RG 1.97 (Revision 2), dated November 19, 1986.

10. The design of the containment spray system at Unit 1 is such that, upon initiation, system flow is directed simultaneously to both the drywell and the suppression chamber (torus), with a fixed proportion of the pump flow distributed to each header. Containment spray pump discharge flow rate is therefore monitored rather than flow rate in the separate (drywell and torus) spray headers.

    In the SER addressing Unit 1 conformance to RG 1.97 (Revision 2), dated November 19, 1986, the NRC stated that, based on the identified plant-specific system design features, the currently installed flow monitoring instrumentation is acceptable (i.e., separate monitoring of flow rate in each [drywell and torus] spray header is not necessary).

11. Unit 1 does not have this system and, therefore, monitoring of this variable is not applicable.

12. Included under Item 18 in the Table.

13. At Unit 1 the HPCI function is performed by the feedwater pumps. Refer to Item 25 in the Table.

14. Liquid poison system flow rate is not directly monitored at Unit 1. Proper functioning of the liquid poison system can be verified by monitoring pump discharge pressure (Item 66 in the Table), storage tank liquid level (Item 38 in the Table), neutron flux level (Items 1, 12, and 13 in the Table), and squib valve status (Item 67 in the Table). Therefore, monitoring system flow rate is not considered to be necessary.

    In the SER addressing Unit 1 conformance to RG 1.97 (Revision 2), dated November 19, 1986, the NRC stated that the identified instrumentation is valid as an acceptable alternative indication of liquid poison system flow rate.

15. At Unit 1 the shutdown cooling system is the functional equivalent of the residual heat removal (RHR) system. However, shutdown cooling system flow rate is not directly monitored. Shutdown cooling system flow rate is adjusted as required to control reactor coolant cooldown rate (heat removal) within applicable limits. The following parameters

TABLE VIII-3 (Cont'd.)

NOTES (Cont'd.)

are monitored to verify proper shutdown cooling system operation:

- Reactor vessel water level (Item 2 in the Table).
- Shutdown cooling system pump discharge pressure (Item 68 in the Table).
- Shutdown cooling system heat exchanger tube side (reactor coolant) inlet and outlet temperatures (Item 40 in the Table).
- Shutdown cooling system heat exchanger shell side (cooling water) inlet and outlet temperatures (Item 69 in the Table).
- Shutdown cooling system valve position - flow path from and to the reactor vessel (Item 70 in the Table).

Additionally, the shutdown cooling system is not expected to be operated during accident or immediate post-accident conditions. It would be operated only in the long term after the unit is in a normal stable shutdown condition.

In the Safety Evaluation Report (SER) addressing Unit 1 conformance to RG 1.97 (Revision 2), dated November 19, 1986, the NRC stated that, based on the identified alternate instrumentation and the design function of the shutdown cooling system, the deviation from the recommended flow monitoring instrumentation is acceptable.

16. Cooling water flow and cooling water temperature for the core spray and containment spray pumps are not directly monitored. The cooling water is recirculated pump discharge flow. Pump suction is normally from the suppression pool, thus torus water temperature (Item 4 in the Table) provides indication of the temperature of the cooling water supplied to the pumps.

    In the SER addressing Unit 1 conformance to RG 1.97 (Revision 2), dated November 19, 1986, the NRC stated that, based on the identified plant-specific system design features, the deviation from the recommended cooling water flow and temperature monitoring instrumentation is acceptable.

TABLE VIII-3 (Cont'd.)

NOTES (Cont'd.)

17. In the SER addressing Unit 1 conformance to RG 1.97 (Revision 2), dated November 19, 1986, the NRC determined that, because Revision 3 to RG 1.97 recommended a Category 3 classification for this variable, no deviation in Category exists. The NRC concluded that the use of Category 3 instrumentation for this variable is acceptable.

18. Included under Item 47 in the Table.

19. Included under Item 51 in the Table.

20. The ability to determine/monitor bulk average temperature is necessary for this EOP Key Parameter.

21. Criteria specified in NEDO-31558-A[26] apply in lieu of those specified in RG 1.97. See NMPC letters NMP1L 0765[13] and NMP1L 0813[27], and NRC letter dated February 10, 1994[28], for additional information.

22. Neutron flux level below the APRM range is not a key variable for accomplishing mitigative actions for any DBA or transient (including those anticipated operational occurrences required to be considered in the implementation of the ATWS Rule [10CFR50.62]); required Operator actions specified in the plant EOPs for such events can be accomplished without reliance on reactor power information below the APRM range. On this basis, the designation of Category 3 instrumentation (in lieu of Category 1 instrumentation as recommended by RG 1.97) is appropriate for monitoring intermediate range and source range neutron flux.

23. Operator actions based on drywell water level would be a contingency action and, therefore, do not meet the definition of a Type A variable. Since drywell water level is not a RG 1.97 Revision 2 recommended variable, the drywell water level recorder does not need to meet the Category 1 criteria. Therefore, a drywell water level recorder is not needed.[29,30]

24. RG 1.97 recommends that noble gas effluent monitoring instrumentation be designed with a range of 1E-06 μCi/cc to 1E+03 μCi/cc. The range of the offgas effluent stack monitoring system (OGESMS) is 1E-07 μCi/cc to 1 μCi/cc (Xe-133). The OGESMS lower limit of detection of 1E-05 μCi/cc meets the NUREG-0737, Item II.F.1, Attachment 1, Position (2) criterion of the instrumentation range beginning at normal conditions (as low as reasonably achievable (ALARA)). The OGESMS upper range limit of

TABLE VIII-3 (Cont'd.)

NOTES (Cont'd.)

1 μCi/cc (Xe-133) provides a safety margin greater than a factor of two for the site-specific design basis effluent release which occurs at NMP1 from a LOCA.

RG 1.97 recommends particulates and halogens instrumentation be designed with a range of 1E-03 μCi/cc to 1E+02 μCi/cc, with a 30-min sampling time for detection of significant releases, release assessment, and long-term surveillance. With the use of OGESMS, the particulate samples would be collected by OGESMS and taken to an onsite facility. The onsite analysis facility has a range of 1E-03 μCi/cc to 0.1 μCi/cc with a 30-min sampling time. The onsite analysis facility's upper range of 0.1 μCi/cc provides a safety margin of two for a design basis effluent release from a LOCA. Using NMP1's design basis effluent release from a LOCA, in lieu of 1E+02 μCi/cc as specified in NUREG-0737 and RG 1.97, to determine doses to personnel working with the sampling media during an accident, the results in estimated exposures would be less than the GDC 19 limits.

In summary, OGESMS meets the objective and purpose of the NUREG-0737 and RG 1.97 guidance. The deviations from NUREG-0737 and RG 1.97 are acceptable.[34]

REACTOR PROTECTION SYSTEM ELEMENTARY DIAGRAM

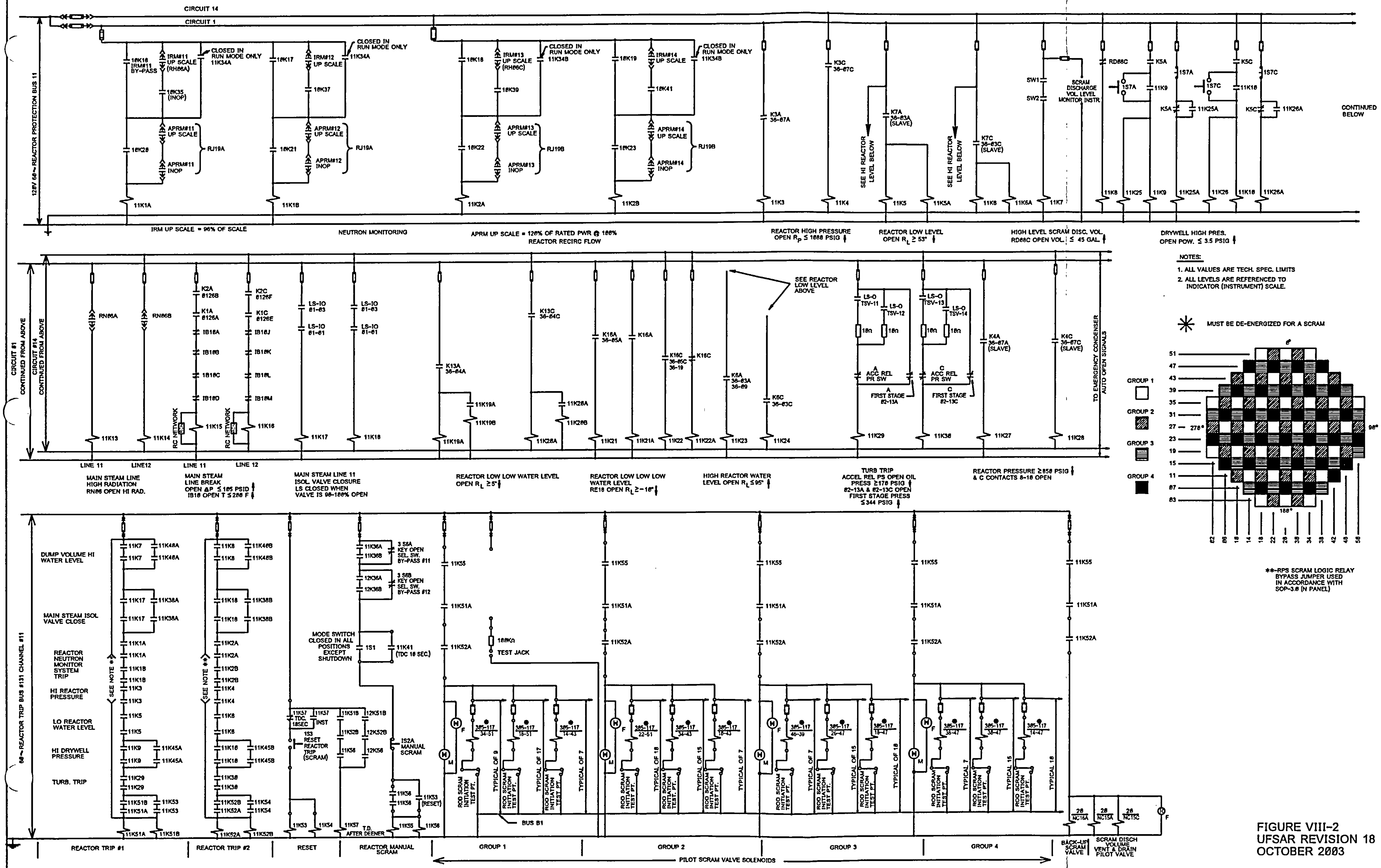

FIGURE VIII-2
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CONTAINMENT ATMOSPHERIC MONITORING SYSTEM

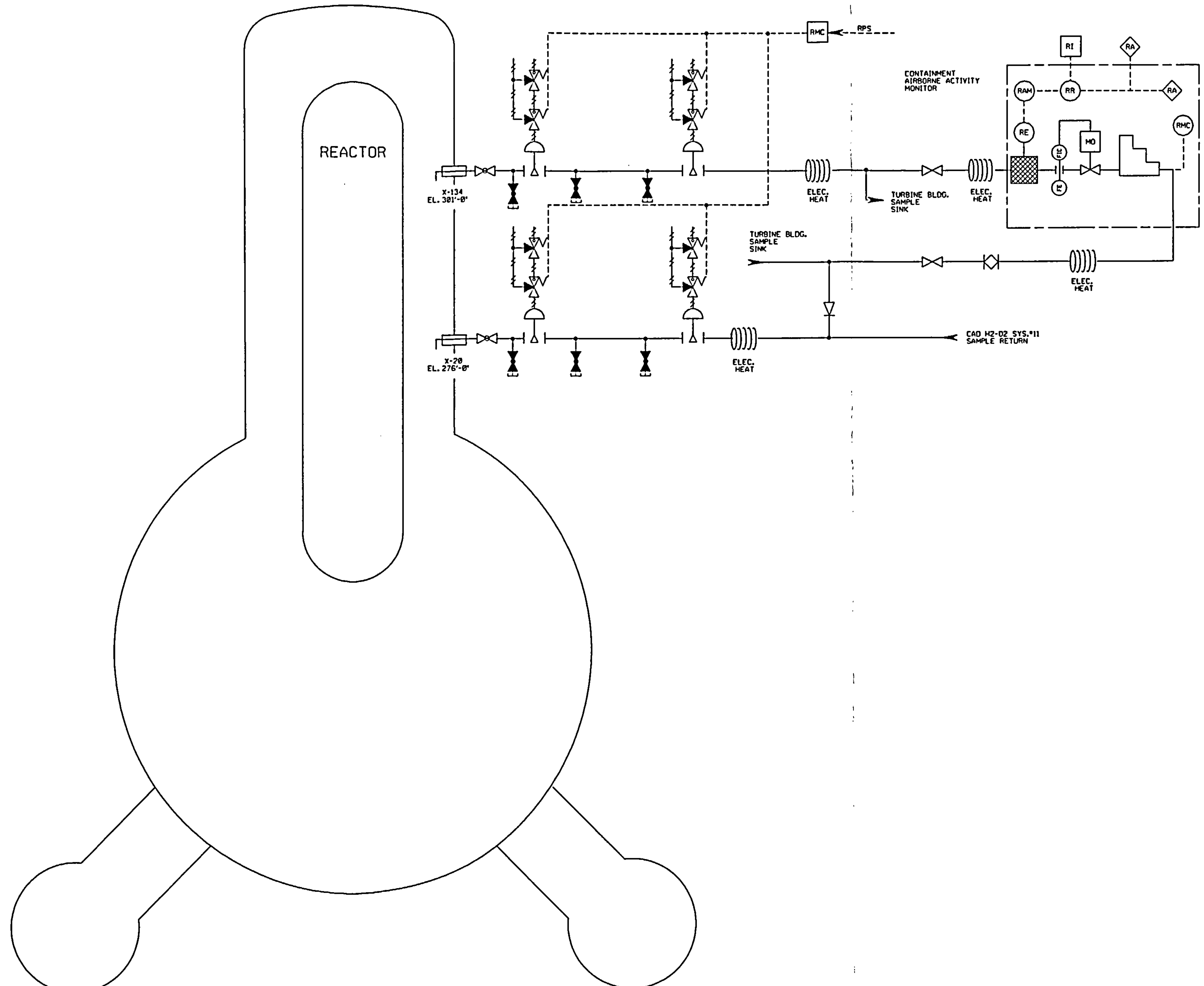

FIGURE VIII - 28
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11 and 12. The use of backfeed through the T1 or T2 transformers provides an additional source of offsite power in addition to the two 115-kV reserve feeds.

1.2 115-kV System

Power for Station startup, the reserve supply to the auxiliaries, and the normal supply to selected auxiliaries is obtained from the 115-kV bus. This bus is fed by two 115-kV transmission lines from remote generating stations. One line is from the South Oswego Steam Station (Line #1), approximately 12 mi away. The other line is from the Lighthouse Hill Station, approximately 26 mi away, through the J. A. FitzPatrick switchyard (Line #4). Both stations have other tie line connections into the Company statewide transmission system. Lighthouse Hill includes hydroelectric generators which have the capability of startup without power input from outside sources (Black Start).

The lines are designed to meet or exceed the requirements of the National Electric Safety Code for heavy loading districts, Grade B.

Each line is protected by a 115-kV, 1200-amp, three-phase, 5,000-MVa oil circuit breaker. Two redundant sets of protective relays are provided on each line for automatic tripping of the circuit breakers under fault conditions. Recognizing that most line faults are transient in nature, automatic reclosing equipment and circuitry is provided to reenergize the lines after the extremely short interruption required to clear a temporary fault.

The following failure mode and effects analysis (FMEA) includes the effects of all failure modes of the 115-kV reserve bus and 4.16-kV power boards. The unit is operating at or near rated load, and all auxiliaries are being supplied from their normal sources.

System or Equipment	Malfunction	Effect
115-kV transmission lines	Both lines lost due to an accident common to the transmission lines.	115-kV bus will be de-energized. PB 11 and 12 will not be affected. PB 101, 102 and 103 will be de-energized. PB 102 and 103 will be reenergized automatically via diesel generators 102 and 103. PB 101 will remain de-energized.
115-kV transmission lines	Major system power disturbance resulting in temporary loss of	115-kV bus will be de-energized. PB 11 and 12 will not be affected.

System or Equipment	Malfunction	Effect
	input power to the Nine Mile Point 115-kV reserve bus. Lines intact and unit continues to run at or near rated load.	PB 101, 102 and 103 will be de-energized, but PB 102 and 103 will be reenergized by the diesel generators. Undervoltage relays will trip the line breakers (R40, R10) in 30 sec. Protective relay schemes at Lighthouse Hill will automatically clear all necessary buses at this Station and actuate an alarm at the EMS Central Regional Control Center located at Henry Clay Boulevard in Liverpool. The protective relay scheme will also initiate an automated control scheme to switch one of two generators at Bennetts Bridge to the line supplying the Lighthouse Hill Station. With the line energized to Lighthouse Hill, the Lighthouse Hill line to the James A. FitzPatrick Nuclear Power Plant bus and to Nine Mile Point Nuclear Station will be energized. The line breaker at Nine Mile Point will automatically close with a live line and dead bus, energizing one-half of the 115-kV reserve bus. This will allow the Operator to switch 102 or 103 back to the offsite generation at his discretion. The other Nine Mile Point line breaker will close only if the Oswego Steam

3. Thermocouple extension cable wire generally is PVC insulated and jacketed or cross-linked polyolefin insulated with a Hypalon jacket. Thermocouple extension cables in the drywell are magnesium-oxide insulated with a 304 stainless steel sheath for high temperature.

4. Substitution for the described special cables may be necessary if the specified cable is not available at the time of replacement or addition.

The insulation associated with safety-related cables purchased and installed since the middle of 1974 meets the requirements of IEEE-383 flame test. The insulation associated with nonsafety-related cables purchased and installed since the middle of 1974 also generally meets the requirements of IEEE-383 flame test, except those routed totally in conduit.

3.5 Design and Spacing of Cable Trays

3.5.1 Tray Design Specifications

1. Ladder Tray - Medium-steel cable ladder, 6-, 12- or 24-in wide, typically having 3-in or 4 1/2-in side members of 12-gauge steel and 1-in O.D. rungs spaced on 9-in centers, cold-swaged or welded into side members. System is furnished hot-dip-galvanized after fabrication and used for power cable up to 5 kV and control cable throughout the Station.

2. Solid Tray - Radio-Frequency Communications Tray - Solid 12-gauge steel all around, 24 in wide by 3 in deep, with solid cover and special barriers designed to provide a magnetic path through the cover, dividing the tray into three sections (signal, control, and control power).

3.5.2 Tray Spacing

Figures IX-3, IX-4, and IX-5 show the cable routing in the turbine building and typical spacing of all Station cable trays. The vertical spacing shown, e.g., 12 in, is a general spacing only and is not a design criteria.

4.0 Emergency Power

4.1 Diesel Generator System

Two sources of electrical power, which are completely separate and are self-contained within the Station and therefore not dependent on any outside source, have been provided by installing two diesel generators. These standby generators each have adequate capacity to start and carry all of the loads required during a maximum emergency power requirement period. Each is designed to be capable of starting and picking up load in 10 sec or less, when lube oil and jacket water temperatures are at or above 85°F and ambient temperature at least 50°F.

Questions/Answers, December 27, 1989, and NUMARC 87-00, Major Assumptions, December 27, 1989, except where RG 1.155 takes precedence. The results of this evaluation were submitted to the NRC in References 1, 2, 3, 5, 6 and 8 and are summarized below. NRC evaluations and acceptance of the Unit 1 response to the SBO rule were documented in References 4, 7 and 9.

6.1 Station Blackout Duration

A SBO duration of 4 hr was determined based on the following plant factors:

1. Ac Power Design Characteristic Group is "P2" based on:

   a. Expected frequency of grid-related LOOP events does not exceed once per 20 yr.

   b. Estimated frequency of LOOP due to extremely severe weather places the plant in extremely severe weather group 1.

   c. Estimated frequency of LOOP due to severe weather places the plant in severe weather group 3.

   d. The offsite power site is in the I3 group.

   e. Plant-specific prehurricane shutdown requirements and procedures are not required for Unit 1, nor are such procedures credited in the determination of the Ac Power Design Characteristic Group.

2. The Emergency Ac Power Configuration Group is "C" based on:

   a. There are two emergency ac power supplies not credited as alternate ac power sources.

   b. One emergency ac power supply is necessary to operate safe shutdown equipment following a LOOP.

3. The target emergency diesel generator reliability is 0.975. A target emergency diesel generator reliability of 0.975 was selected based on having a nuclear unit average emergency diesel generator reliability for the last 20 demands greater than 0.90.

   An analysis showing the emergency diesel generator reliability statistics for the last 20, 50, and 100 demands which supports this target reliability has also been performed.

6.2 Station Blackout Coping Capability

The characteristics of the following plant systems were reviewed to assure that the systems have the availability, adequacy and capability to achieve and maintain a safe plant shutdown and to recover from a SBO for the 4-hr coping duration.

Condensate Inventory for Decay Heat Removal

It has been determined that 58,700 gal of water are required for decay heat removal and cooldown for 4 hr. The minimum permissible emergency condenser (EC) gravity feed, EC makeup tank and EC levels, per Technical Specifications, provide 114,720 gal of water, which is adequate to provide for decay heat removal for at least 4 hr even if both EC level control valves fail open on loss of air at the start of the SBO, and no other Operator actions are taken. Therefore, no plant modifications or Operator actions are required to ensure adequate condensate capacity exists for decay heat removal during a 4-hr SBO.

Station Battery Capacity

Battery capacity calculations performed pursuant to NUMARC 87-00, Section 7.2.2, and IEEE-485-1978, verified that the Station batteries have sufficient capacity to meet SBO loads for 4 hr. Operator action is required to shed nonessential loads from Class 1E batteries to cope with a SBO duration of 4 hr. The shedding of the nonessential loads from Class 1E batteries is identified in plant procedures.

Compressed Air

Air-operated valves (AOVs) relied upon to cope with a SBO for 4 hr can either be operated manually or have sufficient backup sources independent of the preferred and blacked out Unit's Class 1E power supply. Valves requiring manual operation or that need backup sources for operation are identified in plant procedures.

Effects of Loss of Ventilation

The key areas in which the loss of ventilation cooling causes a concern for equipment operability were identified based on the equipment used to respond to the SBO event. Heatup calculations were performed for the:

1. EC condensate return isolation valve room (el 281')
2. EC steam supply isolation valve room (el 298')
3. Reactor building, el 318'
4. Reactor building, el 340'
5. Primary containment

6. Control room

The control room at Unit 1 does not exceed 120°F during a SBO and, therefore, is not a dominant area of concern (DAC).

Reasonable assurance of the operability of SBO response equipment in the dominant areas of concern has been assessed using Appendix F to NUMARC 87-00 and the Topical Report. No hardware modifications are required to provide reasonable assurance for equipment operability.

Procedures direct the Operators to open the control room and auxiliary control room instrument cabinet doors which will increase the cooling of the control room equipment by natural convection.

Containment Isolation

The plant list of containment isolation valves has been reviewed to verify that valves which must be capable of being closed or that must be operated (cycled) under SBO conditions can be positioned (with indication) independent of the preferred and blacked-out Class 1E power supplies. Plant procedures identify valves which must be operated to isolate containment during a SBO.

Reactor Coolant Inventory

An analysis of reactor coolant system (RCS) inventory was performed assuming a leak rate of 18 gpm per recirculation pump (5 pumps) and the maximum allowable (25 gpm) Technical Specification leak rate. The results indicate that reactor water level would reach top of active fuel (TAF) in approximately 1.8 hr.

With a constant leak rate of 115 gpm, plant procedures direct the Operator to actuate the automatic depressurization system (ADS) at or before the time the water level reaches the minimum steam cooling RPV water level (MSCRWL). After the vessel is depressurized, plant procedures direct the Operator to initiate reactor vessel makeup using the diesel-driven fire pump.

6.3 Procedures and Training

Plant procedures, SBO response guidelines, ac power restoration procedures, and SW procedures have been reviewed, and changes necessary to meet NUMARC 87-00, Section 4, guidelines have been implemented to ensure an appropriate response to a SBO event.

Personnel training to ensure an effective response to a SBO event has been incorporated into the training program.

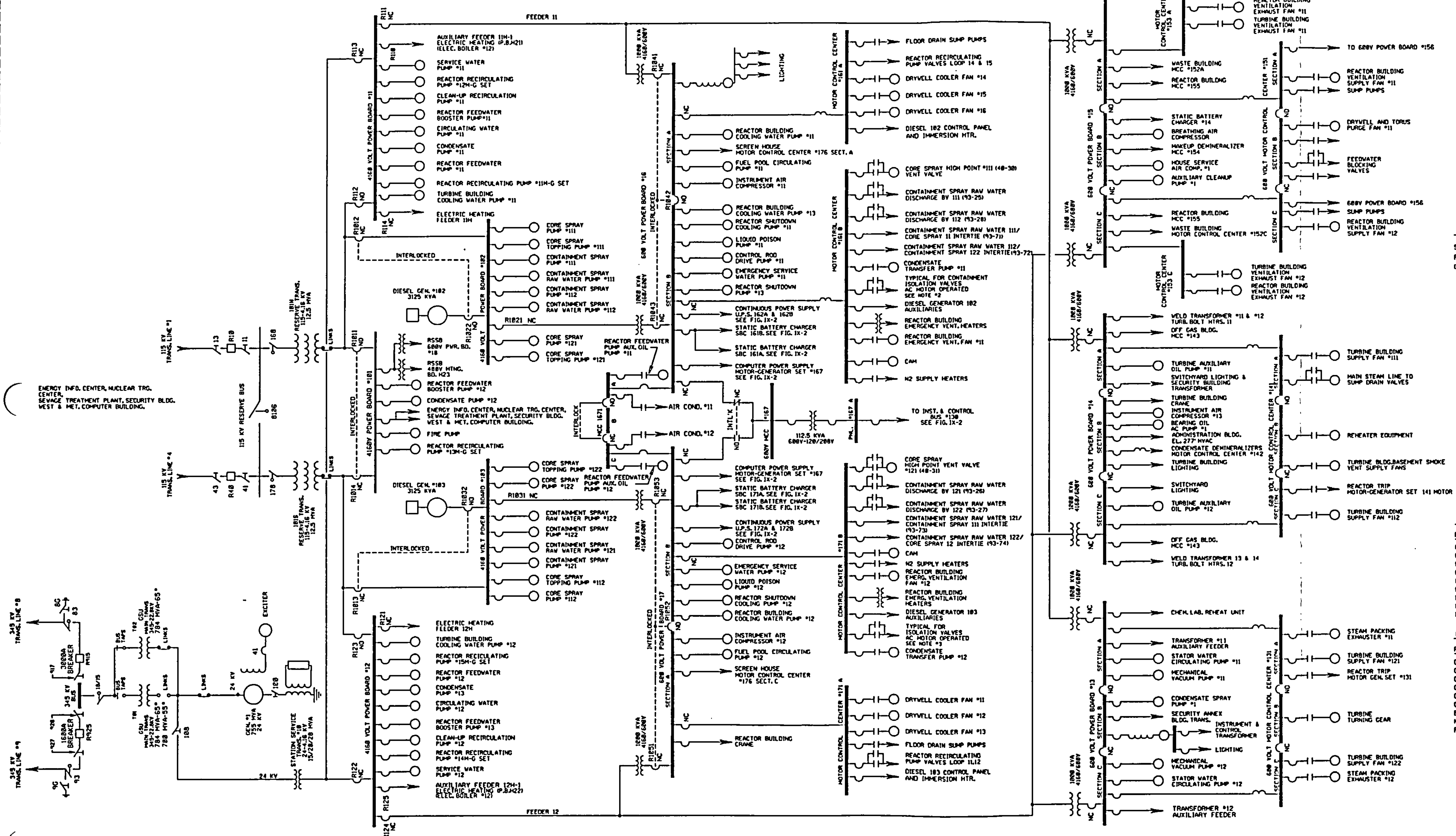
6.4 Quality Assurance

Based on a review of the equipment relied upon to carry out the SBO response, all nonsafety-related components have been upgraded to a "Q" classification and are covered under the Quality Related Program for Nine Mile Point Nuclear Station Operations, which is consistent with the guidance of RG 1.155, Appendix A. The remaining SBO equipment is safety related and is covered by existing quality assurance requirements in 10CFR50, Appendix B.

6.5 Emergency Diesel Generator Reliability Program

An Emergency Diesel Generator Reliability Program has been developed for Unit 1 which conforms to the guidance of RG 1.155, Position C.1.2. The program includes a 0.975 emergency diesel generator target reliability based on emergency diesel generator reliability data for the last 20, 50 and 100 demands.

A.C. STATION POWER DISTRIBUTION



NOTE *1

SELECTED LOADS ARE SHOWN ON THIS DRAWING. FOR ADDITIONAL LOADS SEE THE FOLLOWING ONE-LINE DIAGRAMS C-19409-C SHEETS 2-11.

PB*101	SH.2
PB*102	SH.3
PB*103	SH.3
PB*11	SH.2
PB*12	SH.2
PB*13	SH.4
MCC*131	SH.4,4A
PB*14	SH.5
MCC*141	SH.5
PB*15	SH.6
MCC*151	SH.7
MCC*151	SH.7
MCC*152	SH.7A
PB*153	SH.6
MCC*154	SH.7
MCC*155	SH.6
MCC*156	SH.12
PB*16	SH.8
MCC*161	SH.8
PB*167	SH.10
PB*1671	SH.11
PB*17	SH.9
MCC*171	SH.9
MCC*176	SH.10

NOTE *2

POWER BOARD 161B ISOLATION VALVES

CORE SPRAY SUCTION ISOLATION VALVE / 111 (81-21)
CORE SPRAY DISCHARGE ISOLATION VALVE / 111 (40-11)
CORE SPRAY SUCTION ISOLATION VALVE / 121 (81-01)
CORE SPRAY DISCHARGE ISOLATION VALVE / 121 (40-01)
CLEAN UP SUPPLY ISOLATION VALVE / 11 (33-02R)
CONTAINMENT SPRAY SUCTION ISOLATION VALVE / 111 (80-01)
EMER. CONDENSERS STEAM ISOLATION VALVE / 121 (39-10R)
MAIN STEAM ISOLATION VALVE / 111 (01-01)
FEEDWATER ISOLATION VALVE / 11 (31-07)

NOTE *3

POWER BOARD 171B ISOLATION VALVES

CORE SPRAY SUCTION ISOLATION VALVE / 112 (81-22)
CORE SPRAY DISCHARGE ISOLATION VALVE /112 (40-10)
CORE SPRAY SUCTION ISOLATION VALVE / 122 (81-02)
CORE SPRAY DISCHARGE ISOLATION VALVE / 122 (40-09)
CLEAN UP RETURN ISOLATION VALVE /1 (33-01R)
CONTAINMENT SPRAY SUCTION ISOLATION VALVE /122 (80-22)
EMER. CONDENSER STEAM ISOLATION VALVE / 111 (39-09R)
MAIN STEAM ISOLATION VALVE /121 (01-02)
FEEDWATER ISOLATION VALVE / 12 (31-08)

FIGURE IX-1
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FILE NAME: C19409C_001B