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Constellation Energy
Nine Mile Point Nuclear Station

October 26, 2009

U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ATTENTION: Document Control Desk

SUBJECT: Nine Mile Point Nuclear Station
Unit No. 1; Docket No. 50-220

Submittal of Revision 21 to the Final Safety Analysis Report (Updated), 10 CFR 50.59 Evaluation Summary Report, Technical Specifications Bases Changes, and Report Consistent with 10 CFR 54.37(b)

Pursuant to the requirements of 10 CFR 50.71(e), 10 CFR 50.59(d)(2), and the Nine Mile Point Unit 1 (NMP1) Technical Specifications (TS) Bases Control Program (TS 6.5.6), Nine Mile Point Nuclear Station, LLC (NMPNS) hereby submits the following:

- Revision 21 to the NMP1 Final Safety Analysis Report (Updated) (UFSAR).

- NMP1 Technical Specifications Bases Changes.

The UFSAR Revision 21 pages are contained in Attachment 1. The UFSAR revision contains changes made since the submittal of Revision 20 in October 2007. The revision reflects all changes up to April 2009. No 10 CFR 50.59 evaluations were completed for NMP1 between April 2007 and April 2009.

Attachment 2 contains the revised Technical Specifications Bases pages, which incorporate changes made since April 2007.

Consistent with 10 CFR 54.37(b), Attachment 3 contains a report describing how the effects of aging of newly-identified structures, systems, or components will be managed, such that the intended functions described in 10 CFR 54.4 will be effectively maintained during the license renewal period of extended operation.

A053
NRR

Should you have any questions regarding the information in this submittal, please contact T. F. Syrell, Licensing Director, at (315) 349-5219.

Very truly yours,

CERTIFICATION:

I, Sam Belcher, certify that I am Vice President-Nine Mile Point and that the information contained in this submittal accurately presents changes made since the previous submittal necessary to reflect information and analysis submitted to the Commission or prepared pursuant to Commission requirement.

SLB/RJC

Attachments:
1. Final Safety Analysis Report (Updated) Pages
2. Revised Technical Specifications Bases Pages
3. Report Consistent with 10 CFR 54.37(b) on How Effects of Aging of Newly-Identified Structures, Systems, or Components are Managed

cc: S. J. Collins, NRC
R. V. Guzman, NRC
Resident Inspector, NRC

bcc: (w/o Attachments 1 and 2)

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NMP1L 2395

COMMITMENTS IDENTIFIED IN THIS CORRESPONDENCE:
• None

Posting Requirements for Responses -- NOV/Order **No**

ATTACHMENT 1

FINAL SAFETY ANALYSIS REPORT (UPDATED) PAGES

ATTACHMENT 2

REVISED TECHNICAL SPECIFICATIONS BASES PAGES

NINE MILE POINT UNIT 1 TECHNICAL SPECIFICATIONS BASES
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If there is an additional page being added to the Technical Specification Bases, dashes (-----) will be shown in the Remove column. Likewise, if a page is being removed with no replacement dashes (-----) will be shown in the Insert column.

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The allowable inoperable rod patterns will be determined using information obtained in the startup test program supplemented by calculations. During initial startup, the reactivity condition of the as-built core will be determined. Also, sub-critical patterns of widely separated withdrawn control rods will be observed in the control rod sequences being used. The observations, together with calculated strengths of the strongest control rods in these patterns will comprise a set of allowable separations of malfunctioning rods. During the fuel cycle, similar observations made during any cold shutdown can be used to update and/or increase the allowable patterns.

The number of rods permitted to be valved out of service could be many more than the six allowed by the specification, particularly late in the operating cycle; however, the occurrence of more than six could be indicative of a generic problem and the reactor will be shut down. Placing the reactor in the shutdown condition inserts the control rods and accomplishes the objective of the specifications on control rod operability. This operation is normally expected to be accomplished within ten hours.

Control rod insertion capability is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures that the control rod is not stuck and is free to insert on a scram signal. This surveillance is not required when thermal power is less than or equal to the low power set point (LPSP) of the RWM, since notch insertion may not be compatible with the requirements of the RWM. The 31 day surveillance test frequency takes into account operating experience related to changes in CRD performance. This surveillance requirement allows 31 days after withdrawal of the control rod concurrent with thermal power greater than the LPSP of the RWM to perform the surveillance.

The requirement to exercise control rods at least once per 24 hours in the event power operation is continuing with two or more control rods which are valved out of service or one fully or partially withdrawn control rod which can not be moved, provides a reasonable time to test the control rods and provide assurance of the reliability of the remaining control rods.

b. Control Rod Withdrawal

(1) Control rod dropout accidents as discussed in Appendix E* can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides an indirect verification that the rod is coupled to its drive. Details of the control rod drive coupling are given in Section IV.B.6.1*.

*FSAR

a. A startup inter-assembly local power peaking factor of 1.30 or less.[6]

b. An end of cycle delayed neutron fraction of 0.005.

c. A beginning of life Doppler reactivity feedback.

d. The Technical Specification rod scram insertion rate.

e. The maximum possible rod drop velocity (3.11 ft/sec).

f. The design accident and scram reactivity shape function.

g. The moderator temperature at which criticality occurs.

It is recognized that these bounds are conservative with respect to expected operating conditions. If any one of the above conditions is not satisfied, a more detailed calculation will be done to show compliance with the 280 cal/gm design limit.

In most cases the worth of in-sequence rods or rod segments will be substantially less than 0.013 Δk. Further, the addition of 0.013 Δk worth of reactivity as a result of a rod drop in conjunction with the actual values of the other important accident analysis parameters described above would most likely result in a peak fuel enthalpy substantially less than the 280 cal/gm design limit. However, the 0.013 Δk limit is applied in order to allow room for future reload changes and ease of verification without repetitive Technical Specification changes.

Should a control rod drop accident (CRDA) result in a peak fuel energy content of 280 cal/gm, less than 660 (7 x 7) fuel rods were conservatively estimated to perforate. For 8 x 8 fuel, less than 850 rods were conservatively estimated to perforate, which is bounding for GE11 9 x 9 fuel. As noted in UFSAR Section XV-C.4.2, CRDA results for banked position withdrawal sequence (BPWS) plants have been statistically analyzed and show that, in all cases, the peak fuel enthalpy in a CRDA would be much less than the 280 cal/gm design limit. Thus, the CRDA has been deleted from the standard GE BWR reload package for BPWS plants. The radiological consequences of a CRDA have been shown to remain well within the regulatory limits.

The RWM provides automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted during startup or shutdown, such that only specified control rod sequences and relative positions are allowed over the operating range from all rods inserted to 10% RTP. It serves as an independent backup of the normal withdrawal procedure followed by the operator. With the RWM inoperable during a reactor startup, the operator is still capable of enforcing the prescribed control rod sequence, however, the overall reliability is reduced because a single operator error can result in violating the control rod sequence.

If the RWM becomes inoperable after at least 12 control rods have been withdrawn, startup may continue if the RWM function is performed manually and a required verification of control rod movement by a second licensed operator (Reactor Operator or Senior Reactor Operator) or by a qualified member of the technical staff (a qualified shift technical advisor or reactor engineer) is performed. Also, if the RWM is inoperable prior to commencement of startup, or becomes inoperable during a startup, prior to complete withdrawal of the first 12 control rods, startup may continue if the RWM function is performed manually and a required verification of control rod movement by a second licensed operator (Reactor Operator or Senior Reactor Operator) or by a qualified member of the technical staff (a qualified shift technical advisor or reactor engineer) is performed, and provided that a startup with the RWM inoperable was not performed in the last 12 months. In both cases, 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 licensed operator or other qualified member of the technical staff 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. The allowed frequency requirements of performing a reactor startup with the RWM inoperable (i.e., if not performed in the last 12 months) minimizes the number of reactor startups performed with the RWM inoperable.

(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 conservatism.

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.

The insertion times previously selected were based on the large number of actual scrams of prototype control rod drive mechanisms as discussed in Section IV-B.6.3*. Rapid control rod insertion following a demand to scram will terminate Station transients before any possibility of damage to the core is approached. The primary consideration in setting scram time is to permit rapid termination of steam generation following an isolation transient (i.e., main-steam-line closure or turbine trip without bypass) such that operation of solenoid-actuated relief valves will prevent the safety valves from lifting. Analyses presented in Appendix E-1*, the Second Supplement and the Technical Supplement to Petition to Increase Power Level were based on times which are slower than the proposed revised times.

The scram times generated at each refueling outage when compared to previous scram times demonstrate that the control rod drive scram function has not deteriorated.

*FSAR

d. Control Rod Accumulators

The basis for this specification was not described in the FSAR and, therefore, is presented in its entirety. Requiring no more than one malfunctioned accumulator in any nine-rod square array is based on a series of XY PDQ-4 quarter core problems of a cold, clean core. The worst one in a nine-rod withdrawal sequence resulted in a k_{eff} <1.0--other repeating rod sequence with more rods withdrawn resulted in k_{eff} >1.0. At reactor pressures in excess of 800 psig, even those control rods with malfunctioned accumulators will be able to meet required scram insertion times due to the action of reactor pressure. In addition, they may be normally inserted using the control-rod-drive hydraulic system. Procedural control will assure that control rods with malfunctioned accumulators will be spaced in a one-in-nine array rather than grouped together.

e. Scram Discharge Volume

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram, isolate the reactor coolant system from the containment when required, and to comply with the requirements of the NRC Confirmatory letter of June 24, 1983. The fill/drain test was determined to be an acceptable alternative to a reactor scram test at approximately 50% ROD DENSITY. Performance of a water fill/drain test during cold shutdown will verify that the Scram Discharge Volume is OPERABLE and instrument lines are not plugged. The volume comparison test of water drained equal water used to fill will demonstrate that there is no blockage in the system. By comparing the response of the individual instrument lines during the drain test, partial or complete blockage in one line can be detected.

The SDV Instrumentation/valve response surveillance test will be satisfied anytime a scram occurs (less than or equal to 50% rod density) or by the fill/drain test not to exceed an operating cycle.

The liquid poison system also has a post-LOCA safety function to buffer the suppression pool pH in order to maintain the bulk pH above 7.0. This function is necessary to prevent iodine re-evolution consistent with the Alternative Source Term analysis methodology. Manual system initiation is used, and the minimum amount of sodium pentaborate solution required to be injected for suppression pool pH buffering is 1114 gallons at a minimum concentration of 9.423 weight percent. This volume consists of the minimum required volume of 1325 gallons minus the 197 gallons that are contained below the point where the pump takes suction from the storage tank and minus 14 gallons that are assumed to remain in the pump suction and discharge piping after injection stops. Operation of a single liquid poison pump can satisfy this post-LOCA function. This function applies to the power operating condition, and also whenever the reactor coolant system temperature is greater than 212°F except for reactor vessel hydrostatic or leakage testing with the reactor not critical.

Nearly all maintenance can be completed within a few days. Infrequently, however, major maintenance might be required. Replacement of principal system components could necessitate outages of more than 7 days. In spite of the best efforts of the operator to return equipment to service, some maintenance could require up to 6 months.

The system test specified demonstrates component response such as pump starting upon manual system initiation and is similar to the operating requirement under accident conditions. The only difference is that demineralized water rather than the boron solution will be pumped to the reactor vessel. The test interval between operating cycles results in a system failure probability of 1.1×10^{-6} (Fifth Supplement, p. 115)* and is consistent with practical considerations.

Pump operability will be demonstrated on a more frequent basis. A continuity check of the firing circuit on the explosive valves is provided by pilot lights in the control room. Tank level and temperature alarms are provided to alert the operator of off-normal conditions.

The functional test and other surveillance on components, along with the monitoring instrumentation, gives a high reliability for liquid poison system operability.

*FSAR

BASES FOR 3.2.4 AND 4.2.4 REACTOR COOLANT SPECIFIC ACTIVITY

The specific activity in the reactor coolant is an initial condition for evaluation of the radiological consequences of a main steam line break (MSLB) outside of primary containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely. The specific iodine activity is limited to $\leq$ 0.2 μCi/gm Dose Equivalent I-131. This limit ensures that the source term assumed in the radiological consequences analysis for the MSLB accident is not exceeded, so that any release of radioactivity to the environment during a MSLB results in offsite and control room radiation doses that satisfy the acceptance criteria of 10 CFR 50.67 and Regulatory Guide 1.183. It is also conservative with respect to the value used in the radiological consequences analyses for other postulated small break loss of coolant accidents outside of primary containment and for postulated instrument line breaks.

The limits on reactor coolant specific activity are applicable in the power operating and hot shutdown conditions, since there is an escape path for release of radioactive material from the reactor coolant system to the environment in the event of a MSLB outside of primary containment. In the cold shutdown, refueling, and major maintenance conditions, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

When the reactor coolant specific activity exceeds the limit of 0.2 μCi/gm Dose Equivalent I-131, but is $\leq$ 4.0 μCi/gm, samples must be analyzed for Dose Equivalent I-131 at least once every 4 hours. In addition, the specific activity must be restored to the limit within 48 hours. The completion time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour completion time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

The isotopic analyses of reactor water samples will be used to assure that the limit of Specification 3.2.4 is not exceeded during normal operation. The 7 day frequency is adequate to trend changes in the iodine activity level. The surveillance requirement need only be performed during the power operating condition because the level of fission products generated in other operating conditions is much less. In addition, the trend of the stack offgas release rate, which is continuously monitored, is a good indicator of the trend of the iodine activity in the reactor coolant.

Since the concentration of radioactivity in the reactor coolant is not continuously measured, coolant sampling would be ineffective as a means to rapidly detect gross fuel element failures. However, as discussed in the bases for Specification 3.6.2, some capability to detect gross fuel element failures is inherent in the radiation monitors in the offgas system and on the main steam lines.

BASES FOR 3.2.4 AND 4.2.4 REACTOR COOLANT SPECIFIC ACTIVITY

In the event of a large primary system break under reactor vessel hydrostatic or leakage test conditions with the reactor coolant temperature > 215°F, the reactor not critical, and primary containment integrity not established, calculations show the resultant radiological dose at the exclusion area boundary to be conservatively bounded by the dose calculated for a main steam line break outside primary containment. This dose was calculated on the basis of the reactor coolant specific activity limit of 0.2 μCi/gm Dose Equivalent I-131. The reactor coolant sample required by Specification 4.2.4.b will be used to assure that the limit of Specification 3.2.4.d is not exceeded. The sample shall be taken during steady state conditions to ensure the results are representative of the steady state radioactive concentration for reactor vessel hydrostatic or leakage test conditions.

The list of reactor coolant isolation valves is contained in the procedure governing controlled lists and has 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. Closure of the isolation valves also minimizes potential leakage paths from the primary containment in the event of a loss-of-coolant accident. In addition, whenever fuel is in the reactor vessel and the reactor coolant temperature is less than or equal to 212°F (encompassing the cold shutdown and refueling operating conditions), closure of the shutdown cooling system isolation valves ensures that the reactor vessel water level does not drop below the top of the active fuel during a vessel draindown event caused by a leak or line break in the shutdown cooling system. 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].

It is not intended that compliance with Technical Specification actions would prevent changes in modes or other specified conditions that are part of a shutdown of the unit. Accordingly, if during a plant shutdown any shutdown cooling system isolation valve becomes inoperable for closing while placing shutdown cooling in operation, it is recommended <u>not</u> to take the action specified in 3.2.7.b to isolate one valve in the line having the inoperable valve within 4 hours. This is because, once the line is isolated, the Technical Specifications preclude unisolating the line unless it is for the purpose of demonstrating operability of the inoperable valve. It is, therefore, recommended to take the action specified in 3.2.7.c within 4 hours (instead of the action specified in 3.2.7.b) and proceed with the shutdown actions using shutdown cooling as necessary to reduce reactor coolant temperature to less than 212°F within the following 10 hours. Thereafter, the actions specified in 3.2.7.e and 3.2.7.f would need to be met. An inoperable shutdown cooling isolation valve may be opened with the shutdown cooling permissives met (reactor pressure ≤ 120 psig and temperature ≤ 350°F) in order to comply with the shutdown actions specified in 3.2.7.c.

During plant operation, the isolation valves in the shutdown cooling system are normally closed. In lieu of performing Type C leak rate testing on these isolation valves, a water seal is provided to prevent containment atmosphere leakage through these valves in the event of an accident requiring primary containment isolation. The seal water, supplied from the core spray system, would pressurize the piping between the inboard and outboard isolation valves. To prevent a spurious or inadvertent valve opening from defeating the water seal, the motor-operated shutdown cooling system isolation valves are required to be de-activated (power is removed) during normal plant operation. Thus, the motor-operated shutdown cooling system isolation valves are considered operable when the valves are closed and de-activated and the water seal is capable of performing its function.

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(1) UFSAR

When the shutdown cooling system is placed in service for plant cooldown (with reactor pressure ≤ 120 psig and temperature ≤ 350°F), power for the motor-operated isolation valves must be restored and the valves opened. Should a loss of coolant accident occur at this time, failure of an isolation valve to close upon receipt of an isolation signal could cause a loss of the water seal. The risk associated with this potential single failure has been determined to be acceptable based on the low probability of a core damage event occurring during shutdown cooling system operation(2).

Specification 3.2.7.d requires operability of the shutdown cooling system isolation valves whenever fuel is in the reactor vessel and the reactor coolant temperature is less than or equal to 212°F. If any isolation valve becomes inoperable, Specification 3.2.7.e requires that, within 4 hours, at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition. However, if the shutdown cooling function is needed to provide core cooling, isolating the shutdown cooling line is not desirable. Specification 3.2.7.f allows the shutdown cooling line to remain unisolated provided action is immediately initiated to suspend operations with a potential for draining the reactor vessel (OPDRVs). If suspending the OPDRVs would result in closing the shutdown cooling system isolation valves, an alternative action is provided to immediately initiate action to restore the valve(s) to operable status. This allows the shutdown cooling system to remain in service while actions are being taken to restore the valve(s). The term "immediately" means that the action should be pursued without delay and in a controlled manner. Either of the actions identified in Specification 3.2.7.f must continue until OPDRVs are suspended or the valves are restored to operable status. Operation with the shutdown cooling system in service is not considered an OPDRV so long as system integrity is maintained. System integrity is maintained provided the piping is intact and no maintenance is being performed that has the potential for draining the reactor vessel through the system. In addition, with the reactor coolant temperature less than or equal to 212°F, the water seal function is not required to consider the shutdown cooling system isolation valves operable since primary containment integrity is not required with reactor coolant temperature less than or equal to 215°F.

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)(3) that a line will not isolate. Additional surveillances are in accordance with the Inservice Testing Program described in Specification 6.5.4.

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(2) Letter from G. E. Edison (NRC) to B. R. Sylvia (NMPC) dated March 20, 1995, Issuance of Amendment for Nine Mile Point Nuclear Station Unit No. 1 (License Amendment No. 154).

(3) FSAR

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The peak drywell pressure would be 35 psig which would rapidly reduce to 22 psig within 100 seconds following the pipe break. The total time the drywell pressure would be above 22 psig is calculated to be about 10 seconds. Following the pipe break, the suppression chamber pressure rises to 22 psig within 10 seconds, equalizes with drywell pressure and thereafter rapidly decays with the drywell pressure decay.(1)

The design pressures of the drywell and suppression chamber are 62 psig and 35 psig, respectively.(2) As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the calculated primary containment pressure response discussed above and the suppression chamber design pressure; primary containment preoperational test pressures were chosen. Also, based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than testing the individual components separately.

The function of the primary containment is to isolate and contain fission products released from the reactor coolant system following design basis accidents (DBA). The primary containment provides an essentially leak tight barrier against an uncontrolled release of radioactive material to the environment. The DBA that postulates the maximum release of radioactive material within the primary containment is a loss of coolant accident (LOCA). In the analysis of this accident, it is assumed that primary containment integrity is maintained such that release of fission products to the environment is controlled by the rate of primary containment leakage.

The LOCA radiological consequences analysis is based on an alternative source term (AST) methodology (10 CFR 50.67 and Regulatory Guide 1.183). This analysis concluded that the calculated total effective dose equivalent (TEDE) values to the control room occupants, the exclusion area boundary, and the low population zone are within the TEDE criteria established in 10 CFR 50.67. Primary containment leakage at the rate of 1.5% by weight of the containment air per 24 hours is assumed in the accident analysis. Margin is achieved by establishing the allowable operational leak rate. The operational limit is derived by multiplying the allowable test leak rate by 0.75 thereby providing a 25% margin to allow for leakage deterioration which may occur during the periods between leak rate tests.

Closure of the containment isolation valves for the purpose of the test is accomplished by the means provided for normal operation of the valves. The reactor is vented to the containment atmosphere during ILRT testing.

The primary containment leak rate test frequency is based on maintaining adequate assurance that the leak rate remains within the specification. The leak rate test frequency is based on Option B of 10 CFR 50 Appendix J.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a double-gasketed penetration (primary containment head equipment hatches and the suppression chamber access hatch) is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. The test pressure of 35 psig is consistent with the accident analyses and the maximum preoperational leak rate test pressure. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized.

Leakage from airlocks is measured under accident pressures in accordance with Option B of 10 CFR 50 Appendix J.

The Type A test follows the guidelines stated in ANSI/ANS-56.8[8] and/or the Bechtel Topical Report.[4] This program provides adequate assurance that the test results realistically estimates the degree of containment leakage following a loss-of-coolant accident. The containment leakage rate is calculated using the Absolute Methodology.[8]

The specific treatment of selective valve arrangements including the acceptability of the interpretations of 10 CFR 50 Appendix J requirements are given in References 5, 6, and 7. Core Spray and Containment Spray suction valves will be tested in accordance with the IST Program.

References:

(1) FSAR, Volume II, Appendix E

(2) UFSAR, Section VI B.2.1

(3) (Deleted)

(4) BN-TOP-1 "Testing Criteria for Integrated Leakage Rate Testing of Primary Containment Structures for Nuclear Power Plants," Revision 1, Bechtel Corporation, November 1, 1972

(5) NRC Safety Evaluation Report dated May 6, 1988, "Regarding Proposed Technical Specifications and Exemption Requests Related to Appendix J."

(6) Niagara Mohawk Letter dated July 28, 1988, "Clarifications, Justifications & Conformance with 10 CFR 50 Appendix J SER."

(7) NRC Letter dated November 9, 1988, "Review of the July 28, 1988 Letter on Appendix J Containment Leakage Rate Testing at Nine Mile Point Unit 1."

(8) ANSI/ANS - 56.8 - 1994, "Containment System Leakage Testing Requirements."

The list of primary containment isolation valves is contained in the procedure governing controlled lists and has 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. Alternatively, operability testing of excess flow check valves may be performed prior to installation using a test set-up that simulates an instrument line break condition.

(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 secondary containment is designed to minimize any ground level release of radioactive materials that might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service. The reactor building provides primary containment during periods when the reactor is shutdown, the drywell is open, and activities are ongoing that require secondary containment to be in effect.

There are two principal accidents for which credit is taken for reactor building (secondary containment) integrity. These are a loss of coolant accident (LOCA) and a refueling accident involving "recently irradiated" fuel. The reactor building performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that the release of radioactive materials is restricted to those leakage paths and associated leakage rates assumed in the accident analysis and that fission products entrapped within the reactor building structure will be treated by the Reactor Building Emergency Ventilation System (RBEVS) prior to discharge to the environment.

In addition to these limiting events, events occurring during handling of an irradiated fuel cask and operations with a potential for draining the reactor vessel (OPDRVs) can be postulated to cause a fission product release. During these events, the reactor building would be the only barrier to a release to the environment. Thus, reactor building integrity is required during handling of an irradiated fuel cask and during OPDRVs.

The Refueling Accident analysis is based on an alternative source term (AST) methodology (10 CFR 50.67 and Regulatory Guide 1.183). This analysis concluded that the calculated total effective dose equivalent (TEDE) values to the control room occupants, the exclusion area boundary, and the low population zone are well below the TEDE criteria established in 10 CFR 50.67 without crediting reactor building integrity, operation of the RBEVS, or operation of the Control Room Air Treatment System (CRATS), as long as the fuel is allowed to decay for at least 24 hours following reactor shutdown. As a result, "recently irradiated" fuel is defined as fuel that has occupied part of a critical reactor core within 24 hours; i.e., reactor fuel that has decayed less than 24 hours following reactor shutdown. Therefore, reactor building integrity is not required, and RBEVS and CRATS are not required to be operable, during movement of decayed irradiated fuel that is no longer considered "recently irradiated." Conversely, reactor building integrity is required, and RBEVS and CRATS are required to be operable, during movement of recently irradiated fuel assemblies.

In the answers to Questions II-3 and IV-5 of the Second Supplement and also in the Fifth Supplement*, the relationships among wind speed, direction, pressure distribution outside the building, building internal pressure, and reactor building leakage are discussed. The curve of pressure in Figure 3.4.1 represents the wind direction which results in the least building leakage. It is assumed that when the test is performed, the wind direction is that which gives the least leakage.

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*FSAR

If the wind direction was not from the direction which gave the least reactor building leakage, building internal pressure would not be as negative as Figure 3.4.1 indicates. Therefore, to reduce pressure, the fan flow rate would have to be increased. This erroneously indicates that reactor building leakage is greater than if wind direction were accounted for. If wind direction were accounted for, another pressure curve could be used which was less negative. This would mean that less fan flow (or measured leakage) would be required to establish building pressure. However, for simplicity it is assumed that the test is conducted during conditions leading to the least leakage while the accident is assumed to occur during conditions leading to the greatest reactor building leakage.

As discussed in the Second Supplement and Fifth Supplement, the pressure for Figure 3.4.1 is independent of the reactor building leakage rate referenced to zero mph wind speed at a negative differential pressure of 0.25 inch of water. Regardless of the leakage rate at these design conditions, the pressure versus wind speed relationship remains unchanged for any given wind direction.

By requiring the reactor building pressure to remain within the limits presented in Figure 3.4.1 and a reactor building leakage rate of less than 1600 cfm, exfiltration would be prevented. This would assure that the leakage from the primary containment is directed through the filter system and discharged from the 350-foot stack.

Isolation of the reactor building occurs automatically upon high radiation of the normal building exhaust ducts or from high radiation at the refueling platform (See 3.6.2). Isolation will assure that any fission products entering the reactor building will be routed to the emergency ventilation system prior to discharge to the environment (Section VII-H.3.0 of the FSAR).

The two principal accidents for which the reactor building isolation valves must be operable are a loss of coolant accident (LOCA) and a refueling accident involving "recently irradiated" fuel. In addition to these limiting events, events occurring during handling of an irradiated fuel cask and operations with a potential for draining the reactor vessel (OPDRVs) can be postulated to cause a fission product release. During these events, the reactor building would be the only barrier to a release to the environment. Thus, the reactor building isolation valves are required to be operable during handling of an irradiated fuel cask and during OPDRVs.

The Refueling Accident analysis is based on an alternative source term (AST) methodology (10 CFR 50.67 and Regulatory Guide 1.183). This analysis concluded that the calculated total effective dose equivalent (TEDE) values to the control room occupants, the exclusion area boundary, and the low population zone are well below the TEDE criteria established in 10 CFR 50.67 without crediting reactor building integrity or operation of the reactor building emergency ventilation system (RBEVS), as long as the fuel is allowed to decay for at least 24 hours following reactor shutdown. As a result, "recently irradiated" fuel is defined as fuel that has occupied part of a critical reactor core within 24 hours; i.e., reactor fuel that has decayed less than 24 hours following reactor shutdown. Therefore, reactor building integrity is not required and the reactor building isolation valves are not required to be operable during movement of decayed irradiated fuel that is no longer considered "recently irradiated." Conversely, reactor building integrity is required and the reactor building isolation valves are required to be operable during movement of recently irradiated fuel assemblies.

The secondary containment is designed to minimize any ground level release of radioactive materials that might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service. The reactor building provides primary containment during periods when the reactor is shutdown, the drywell is open, and activities are ongoing that require secondary containment to be in effect.

There are two principal accidents for which credit is taken for reactor building (secondary containment) integrity. These are a loss of coolant accident (LOCA) and a refueling accident involving "recently irradiated" fuel. The reactor building performs no active function in response to each of these limiting events; however, its leak tightness is required to ensure that the release of radioactive materials is restricted to those leakage paths and associated leakage rates assumed in the accident analysis and that fission products entrapped within the reactor building structure will be treated by the Reactor Building Emergency Ventilation System (RBEVS) prior to discharge to the environment.

In addition to these limiting events, events occurring during handling of an irradiated fuel cask and operations with a potential for draining the reactor vessel (OPDRVs) can be postulated to cause a fission product release. During these events, the reactor building would be the only barrier to a release to the environment. Thus, reactor building integrity is required during handling of an irradiated fuel cask and during OPDRVs.

The Refueling Accident analysis is based on an alternative source term (AST) methodology (10 CFR 50.67 and Regulatory Guide 1.183). This analysis concluded that the calculated total effective dose equivalent (TEDE) values to the control room occupants, the exclusion area boundary, and the low population zone are well below the TEDE criteria established in 10 CFR 50.67 without crediting reactor building integrity, operation of the RBEVS, or operation of the Control Room Air Treatment System (CRATS), as long as the fuel is allowed to decay for at least 24 hours following reactor shutdown. As a result, "recently irradiated" fuel is defined as fuel that has occupied part of a critical reactor core within 24 hours; i.e., reactor fuel that has decayed less than 24 hours following reactor shutdown. Therefore, reactor building integrity is not required during movement of decayed irradiated fuel that is no longer considered "recently irradiated." Conversely, reactor building integrity is required during movement of recently irradiated fuel assemblies.

As discussed in Section VI-F* all access openings of the reactor building have as a minimum two doors in series. Appropriate local alarms and control room indicators are provided to always insure that reactor building integrity is maintained. Surveillance of the reactor building access doors provides additional assurance that reactor building integrity is maintained.

Maintaining closed doors on the pump compartments ensures that suction to the core and containment spray pumps is not lost in case of a gross leak from the suppression chamber.

*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 analysis 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 10 CFR 50.67 acceptance criteria 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 activities 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. 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.

The two principal accidents for which the Reactor Building Emergency Ventilation System (RBEVS) must be operable are a loss of coolant accident (LOCA) and a refueling accident involving "recently irradiated" fuel. In addition to these limiting events, events occurring during handling of an irradiated fuel cask and operations with a potential for draining the reactor vessel (OPDRVs) can be postulated to cause a fission product release. During these events, the reactor building would be the only barrier to a release to the environment. Thus, the RBEVS is required to be operable during handling of an irradiated fuel cask and during OPDRVs.

The Refueling Accident analysis is based on an alternative source term (AST) methodology (10 CFR 50.67 and Regulatory Guide 1.183). This analysis concluded that the calculated total effective dose equivalent (TEDE) values to the control room occupants, the exclusion area boundary, and the low population zone are well below the TEDE criteria established in 10 CFR 50.67 without crediting reactor building integrity or operation of the RBEVS, as long as the fuel is allowed to decay for at least 24 hours following reactor shutdown. As a result, "recently irradiated" fuel is defined as fuel that has occupied part of a critical reactor core within 24 hours; i.e., reactor fuel that has decayed less than 24 hours following reactor shutdown. Therefore, reactor building integrity is not required and the RBEVS is not required to be operable during movement of decayed irradiated fuel that is no longer considered "recently irradiated." Conversely, reactor building integrity is required and the RBEVS is required to be operable during movement of recently irradiated fuel assemblies.

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.

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

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

Demonstration of the automatic initiation capability and operability of filter cooling is necessary to assure system performance capability. If one emergency ventilation system is inoperable, the other system must be verified to be operable daily. This substantiates the availability of the operable system and thus reactor operation or refueling operation may continue during this period.

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) to minimize inleakage of unfiltered air. 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.

The Control Room Envelope (CRE) is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. The CRE is protected for normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The operability of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

The control room air treatment system provides protection from smoke and hazardous chemicals to the CRE occupants. The analysis of hazardous chemical releases demonstrates that the toxicity limits are not exceeded in the CRE following a hazardous chemical release (Ref. 1). The evaluation of a smoke challenge demonstrates that it will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown panels (Ref. 1).

A periodic offsite chemical survey and procedures for controlling onsite chemicals are essential elements of CRE protection against hazardous chemicals. Changes in offsite, mobile, and onsite hazardous chemical types or quantities are assessed in accordance with the Control Room Envelope Habitability Program. The assessments provide the necessary justification for not installing a toxic gas monitoring automatic isolation system.

In order for the control room air treatment system to be considered operable, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke.

The CRE boundary may be opened intermittently under administrative controls. This only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

If the control room air treatment system is found to be inoperable for any reason other than an inoperable CRE during the power operating condition, there is no immediate threat to the CRE occupants and reactor operation may continue for a limited period of time while repairs are being made. If the system cannot be repaired within seven days, the reactor is shutdown and brought to a cold shutdown within 36 hours.

If the control room air treatment system is found to be inoperable for any reason whenever recently irradiated fuel or an irradiated fuel cask is being handled in the reactor building, or during operations with a potential for draining the reactor vessel (OPDRVs), there is no immediate threat to the CRE occupants and these activities may continue for a limited period of time while repairs are being made. If the system cannot be repaired within seven days, these activities must be immediately suspended.

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem total effective dose equivalent TEDE)), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. If in the power operating condition, actions must be taken to restore an operable CRE boundary within 90 days. During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the condition, regardless of whether entry is intentional or unintentional. The 24 hour period allowed is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day period is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day period is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

The testing performed for TS 4.4.5.g verifies the operability of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program. The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem TEDE and the CRE occupants are protected from hazardous chemicals and smoke. This surveillance requirement verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate during the power operating condition, TS 3.4.5.f must be entered. The actions allow time to restore the CRE boundary to operable status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, (Ref. 2) which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 3). These compensatory measures may also be used as mitigating actions as required by TS 3.4.5.f. Temporary analytical methods may also be used as compensatory measures to restore operability

(Ref. 4). Options for restoring the CRE boundary to operable status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to operable status.

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 10 CFR 50.67. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

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.

The two principal accidents for which the Control Room Air Treatment System (CRATS) must be operable are a loss of coolant accident (LOCA) and a refueling accident involving "recently irradiated" fuel. In addition to these limiting events, events occurring during handling of an irradiated fuel cask and operations with a potential for draining the reactor vessel (OPDRVs) can be postulated to cause a fission product release. Thus, the CRATS is required to be operable during handling of an irradiated fuel cask and during OPDRVs.

The Refueling Accident analysis is based on an alternative source term (AST) methodology (10 CFR 50.67 and Regulatory Guide 1.183). This analysis concluded that the calculated total effective dose equivalent (TEDE) values to the control room occupants, the exclusion area boundary, and the low population zone are well below the TEDE criteria established in 10 CFR 50.67 without crediting operation of the CRATS, as long as the fuel is allowed to decay for at least 24 hours following reactor shutdown. As a result, "recently irradiated" fuel is defined as fuel that has occupied part of a critical reactor core within 24 hours; i.e., reactor fuel that has decayed less than 24 hours following reactor shutdown. Therefore, the CRATS is not required to be operable during movement of decayed irradiated fuel that is no longer considered "recently irradiated." Conversely, the CRATS is required to be operable during movement of recently irradiated fuel assemblies.

Operation of the system for 15 minutes every month will demonstrate operability of the filters and adsorber system.

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

References:

1. UFSAR, Section III.B.

2. Regulatory Guide 1.196, Revision 0, May 2001.

3. NEI 99-03, "Control Room Habitability Assessment." June 2001.

4. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, "NEI Draft White Paper, Use of the Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability. " (ADAMS Accession No. ML040160868).

Each reactor operating condition has a related reactor mode switch position for the safety system. The instrumentation system operability for each mode switch position is based on the requirements of the related safety system. For example, the specific high drywell pressure trip systems must be tripped or operable any time core spray, containment spray, automatic depressurization or containment isolation functions are required.

In instrumentation systems where two trip systems are required to initiate action, either both trip systems are operable or one is tripped. Having one trip system already tripped does not decrease the reliability in terms of initiating the desired action. However, the probability of spurious actuation is increased. Certain instrument channels or sensor inputs to instrument channels may be bypassed without affecting safe operation. The basis for allowing bypassing of the specified SRM's, IRM's, LPRM's and APRM's is discussed in Volume I (Section VII-C.1.2)*. The high area temperature isolation function for the cleanup system has one trip system. There are three instrument channels; each has four sensor inputs. Only two instrument channels are required since the area covered by any one sensor is also covered by a sensor in one of the other two instrument channels. The shutdown cooling system also has one trip system for high area temperature isolation. However, since the area of concern is much smaller, only one instrument channel is provided. Four sensors provide input to the channel. Since the area covered is relatively small only three of the four sensors are required to be operable in order to assure isolation when needed.

Table 3.6.2b requires that the low-low reactor vessel water level instrumentation that initiates isolation of the shutdown cooling system be operable with the reactor mode switch in the Shutdown and Refuel positions. Two trip systems must be operable or in the tripped condition in the hot shutdown condition. However, in the cold shutdown and refueling conditions, only one trip system (with two instrument channels) must be operable so long as shutdown cooling system integrity is maintained. System integrity is maintained provided the piping is intact and no maintenance is being performed that has the potential for draining the reactor vessel through the system. If one low-low water level instrument channel in a required Trip System becomes inoperable and cannot be restored or placed in the tripped condition within the allowed time, the associated shutdown cooling line should be isolated. However, if the shutdown cooling function is needed to provide core cooling, isolating the shutdown cooling line is not desirable. Table 3.6.2b, Note (j), allows the shutdown cooling line to remain unisolated and the system to remain in service provided action is immediately initiated to restore the channel to operable status. The alternative action is to immediately initiate action to isolate the shutdown cooling system, which may require that alternate decay heat removal capabilities be provided. The term “immediately” means that the action should be pursued without delay and in a controlled manner. Either of these actions must continue until the channel is restored to operable status or the shutdown cooling system is isolated.

Manual initiation is available for scram, reactor isolation and containment isolation. In order to manually initiate other systems, each pump and each valve is independently initiated from the control room. Containment spray raw water cooling is not automatically initiated. Manual initiation of each pump is required as discussed in 3.3.7 above.

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*FSAR; Letter, R.R. Schneider to A. Giambusso, dated November 15, 1973

a. The set points included in the tables are those used in the transient analysis and the accident analysis. The high flow set point for the main steam line is 105 psi differential. This represents a flow of approximately 4.4×10^6 lb/hr. The high flow set point for the emergency cooling system supply line is ≤ 11.5 psi differential. This represents a flow of approximately 9.8×10^5 lb/hr at rated conditions.

Emergency Cooling Initiation

The emergency cooling initiation logic is separated into two trip systems which use a one-out-of-two taken twice logic configuration. The actuation of a single trip system will cause a half emergency cooling system initiation. A trip system for the emergency cooling initiation parameter provides the protective action of de-energizing one of the two DC solenoid valves for each of the two air-operated condensate return isolation valves. A high reactor pressure or low-low reactor water level signal from an instrument channel will de-energize its corresponding time delay relay after 12 seconds. If either of the two time delay relays in a trip system times out, the two control circuits associated with that trip system will change state causing one of the two DC solenoid valves for each of the two condensate return isolation valves to de-energize. This results in the insertion of a half emergency cooling system initiation signal where the condensate return isolation valves do not open. A full initiation will occur when at least one time delay relay in each of the two trip systems times out, and all four control circuits change state to de-energize both DC solenoid valves for each of the two condensate return isolation valves, thereby opening both valves. It is important to recognize that pulling the fuses for (or otherwise de-energizing) the DC solenoid valves for the condensate return isolation valves will affect the isolation capability on a high steam flow isolation signal.

Emergency Cooling Isolation

Automatic isolation of the emergency cooling systems (loops) occurs on a high steam flow isolation signal from the four ΔP transmitters connected to the steam supply lines (two transmitters per steam line). Each ΔP transmitter provides the sensor inputs to its respective instrument channel. Automatic isolation of an emergency cooling system involves closure of both motor-operated steam supply isolation valves and the condensate return isolation valve in the affected system. [Note that the requirements of Table 3.6.2c do not apply to the drain and common loop vent valves since the isolation of these valves is to prevent bypass leakage. Requirements also do not apply to the individual loop vent valves which allow for vent isolation of one (1) Emergency Cooling loop while maintaining the other loop operable.] For the high steam flow isolation parameter, each emergency cooling system is required to have two tripped or operable trip systems, with two operable instrument channels per operable trip system. Both instrument channels for a given emergency cooling system provide isolation trip signals to both of the system's trip systems in a one-out-of-two logic configuration for each trip system. The trip of either trip system will initiate an isolation of the affected system. A trip system for the high steam flow isolation parameter provides the protective action of closing one of the two steam supply isolation valves and energizing one of the two DC solenoid valves to close the condensate return isolation valve.

The high level in the scram discharge volume is provided to assure that there is still sufficient free volume in the discharge system to receive the control rod drives discharge. Following a scram, bypassing is permitted to allow draining of the discharge volume and resetting of the reactor protection system relays. Since all control rods are completely inserted following a scram and since the bypass of this particular scram initiates a control rod block, it is permissible to bypass this scram function. The scram trip associated with the shutdown position of the mode switch can be reset after 10 seconds.

The condenser low-low-low vacuum and the main steam line isolation valve position signals are bypassed in the startup and refuel positions of the reactor mode switch when the reactor pressure is less than 600 psig. These are bypassed to allow warmup of the main steam lines and to provide a heat sink during startup.

Other than the Station turbine generator, the Station is supplied by four independent sources of ac power; two 115 kv transmission lines, and two diesel-generators. Any one of the required power sources will provide the power required for a LOCA. Engineering calculations show that a LOCA concurrent with a loss of offsite power and the single failure of one of the diesel-generators (DG) results in a loading for the remaining diesel-generator that is below the unit's 2000 hour/year rating. This loading is greater than that required during a Station shutdown condition. The monthly test run paralleled with the system is based on the manufacturer's recommendation for these units in this type of service. The testing during operating cycle will simulate the accident conditions under which operation of the diesel-generators is required. The major equipment comprising the maximum diesel-generator loading is given in Figure IX-6*.

As mentioned above, a single diesel-generator is capable of providing the required power to equipment following a LOCA. Two fuel oil storage tanks are provided with piping interties to permit supplying either diesel-generator. A two-day supply will provide adequate time to arrange for fuel makeup if needed. The full capacity of both tanks will hold a four-day supply.

It has been demonstrated in Section XV.B.3.23* that even with complete dc loss the reactor can be safely isolated and the emergency cooling system will be operative with makeup water to the emergency cooling system shells maintained manually. Having at least one dc battery system available will permit: automatic makeup to the shells rather than manual, closing of the d-c actuated isolation valve on all lines from the primary system and the suppression chamber, maintenance of electrical switching functions in the Station and providing emergency lighting and communications power.

There are two physically separate and electrically independent, safety-related battery systems (11 and 12). Each system includes one 125-volt station battery, two 100% capacity static battery chargers connected in parallel, and one dc power distribution (battery) board.

During normal operation, the 125-volt dc loads are powered from the battery chargers with the batteries floating on the system. Each battery charger has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery fully charged. In case of loss of normal power to the battery charger, the dc loads are automatically powered from the battery.

Both battery systems, each consisting of one battery, at least one battery charger, and the associated dc power distribution (battery) board, are required to be operable for all reactor operating conditions except cold shutdown. In addition, a battery system must be operable whenever its associated DG is required to be operable since the battery system is a support system for the DG. A battery system shall have a minimum of 106 volts at the battery terminals to be considered operable.

*FSAR

If a battery system becomes inoperable, that battery system must be returned to an operable status within 24 hours. If the 24-hour allowed outage time cannot be met, then Specification 3.0.1 must be entered immediately. The second paragraph of Specification 3.0.1 provides two options:

1. Place the unit in a condition consistent with the individual specification; however, in this case, the individual specification (i.e., 3.6.3) does not provide any action to take when a battery system has been inoperable for more than 24 hours. To determine required actions and action completion times, the individual specifications for the systems supported by the battery system should be entered and reviewed to determine applicable actions. If no actions are applicable for the given reactor operating condition, then no actions are required.

or

2. Place the unit in an operational condition in which the specification is not applicable (i.e., cold shutdown).

Accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980, NUREG-0661, "Safety Evaluation Report Mark I Containment Long Term Program," and the NRC Final Rule, "Combustible Gas Control in Containment," made effective October 16, 2003 (68 FR 54123).

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications," as approved by the NRC and documented in the SER (letter to R. D. Binz IV from C. E. Rossi dated July 21, 1992).

Action 3a and Action 4a of Table 3.6.11-2 require that with the number of OPERABLE channels less than the total Number of Channels shown in Table 3.6.11-1, a Special Report must be prepared and submitted to the NRC within 14 days following the event. The term "event" refers to the reason that an instrument channel is inoperable. For the purpose of applying Action 3a and Action 4a of Table 3.6.11-2, removal of a single accident monitoring instrumentation channel from service for the sole purpose of performing routine TS required surveillances is not considered an event requiring preparation and submittal of a 14-day Special Report. If a single accident monitoring instrumentation channel is removed from service for other activities (e.g., to perform preventive maintenance), or if a channel fails, these events require preparation and submittal of a Special Report in accordance with Actions 3a and 4a.

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 effective dose equivalent to an individual at the exclusion area boundary will not exceed a very small fraction of the limits of 10 CFR 50.67 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 10 CFR 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.

ATTACHMENT 3

REPORT CONSISTENT WITH 10 CFR 54.37(b) ON HOW EFFECTS OF AGING OF NEWLY-IDENTIFIED STRUCTURES, SYSTEMS, OR COMPONENTS ARE MANAGED

ATTACHMENT 3

REPORT CONSISTENT WITH 10 CFR 54.37(b) ON HOW EFFECTS OF AGING OF NEWLY-IDENTIFIED STRUCTURES, SYSTEMS, OR COMPONENTS ARE MANAGED

This report is in lieu of adding a level of detail to the Nine Mile Point Unit 1 Updated Final Safety Analysis Report (UFSAR) that is greater than in the remainder of the UFSAR, including the License Renewal Supplement. An entry on the NRC website, "Frequently Asked Questions (FAQs) About License Renewal Inspection Procedure (IP) 71003, 'Post-Approval Site Inspection for License Renewal,'" relates to the amount of detail required per 10 CFR 54.37(b). It states, "The NRC staff will consider it acceptable if the summary information included in the FSAR update is consistent with the requirements of 10 CFR 54.21(d), and the guidance provided in Revision 1 of NUREG-1800, 'Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants' (SRP-LR), provided that the licensee has supplied the technical details (as described in RIS 2007-16) in another documented submittal to the NRC." The information in this report is consistent with the technical information previously submitted to the NRC with the Amended License Renewal Application (ALRA).

On July 14, 2005, Nine Mile Point Nuclear Station, LLC (NMPNS) submitted an ALRA to the NRC to renew the operating licenses for Nine Mile Point Nuclear Station Unit 1 (NMP1) and Unit 2 (NMP2) for an additional 20 years beyond the original expiration dates of August 22, 2009 (NMP1) and October 31, 2026 (NMP2). Within the ALRA, system tables were provided to define the component types, functions and the Aging Management Programs that applied. Lists of individual components within scope of license renewal were not required to be provided.

Subsequent to the completion of the necessary reviews, audits, responses to Requests for Additional Information (RAIs), and resolutions of other questions, the NRC published NUREG-1900, Safety Evaluation Report Related to the License Renewal of Nine Mile Point Nuclear Station, Units 1 and 2, in September of 2006, which documented the NRC staff's review of the information submitted to them through April 21, 2006. The renewed operating licenses for NMP1 and NMP2 were issued on October 31, 2006, extending the license for NMP1 to August 22, 2029, and NMP2 to October 31, 2046.

For holders of a renewed operating license, 10 CFR 54.37(b) requires that newly-identified Structures, Systems, or Components (SSCs) be included in the Final Safety Analysis Report (FSAR) update required by 10 CFR 50.71(e) describing how the effects of aging will be managed. Newly-identified SSCs are those SSCs that were installed in the plant at the time of the License Renewal of NMP1 and NMP2, but were not evaluated as part of the ALRA (as discussed in RIS 2007-16).

During the period of January 2008 to May 2009, a review of updated drawings and the site component database revealed approximately 5600 components installed before October 31, 2006 that had not previously been screened for license renewal applicability.

Of the components that were identified as in-scope and subject to aging management review, 222 components were found to be subject to aging management requirements and are, therefore, "newly identified" and subject to 10 CFR 54.37(b) reporting requirements. The 222 components can be broken down into two groups. The first group of 179 components is already addressed within the tables submitted with the ALRA. The second group consists of 43 components that would not have been addressed under any of the existing tables in the ALRA.

The 222 "newly-identified" components have been assigned to existing Aging Management Programs and appropriate aging management strategies have been invoked to adequately detect and manage the applicable aging effects throughout the period of extended operation and can be verified by NRC inspection.

ATTACHMENT 3

REPORT CONSISTENT WITH 10 CFR 54.37(b) ON HOW EFFECTS OF AGING OF NEWLY-IDENTIFIED STRUCTURES, SYSTEMS, OR COMPONENTS ARE MANAGED

Although the tables provided in the ALRA are not actually being revised, the attached tables show the changes that would have been made had the 43 "newly-identified" components been included in the ALRA. The table numbers shown herein correlate with those provided in the ALRA. A list of the 222 components is not provided in this document consistent with the detail provided in the ALRA. The changes from the existing ALRA are shown in italics in the attached tables.

Table 3.3.2.A-1 Auxiliary Systems
NMP1 Circulating Water System - Summary of Aging Management Evaluation
(12 Components)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
External Surfaces	LBS	*Copper Alloys (Zinc <15%)*	Air	None	None			None
Valves	LBS	*Copper Alloys (Zinc <15%)*	Raw Water	Loss of Material	*Open Cycle Cooling Water Program*	VII.C.1.2.1	3.3.1.A-17	A

Table 3.3.2.A-7 Auxiliary Systems
NMP1 Emergency Diesel Generator System - Summary of Aging Management Evaluation
(4 Components)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Valves	PB	Copper Alloys (Zinc >15%) And Aluminum Bronze	*Treated Water Temperature <140°F*	None	None			None

Table 3.3.2.A-19 Auxiliary Systems
NMP1 Service Water System - Summary of Aging Management Evaluation
(14 Components)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
External Surfaces	LBS	*Cast Austenitic Stainless Steel*	Air	None	None			None
Valves	LBS	*Cast Austenitic Stainless Steel*	Raw Water	Loss of Material	*Open Cycle Cooling Water System Program*	VII.C.1.2-a	3.3.1.A-17	A

Table 3.3.2.A-23 Auxiliary Systems
NMP1 Turbine Building HVAC System - Summary of Aging Management Evaluation
(9 Components)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
Heat Exchanger	*LBS*	Carbon or Low Alloy Steel (Yield Strength <100 Ksi)	Air	Loss of Material	*Preventive Maintenance Program*	VII.F.2.1.2	3.3.1.A-05	A

REPORT PURSUANT TO 10 CFR 54.37(b) ON HOW EFFECTS OF AGING OF NEWLY-IDENTIFIED STRUCTURES, SYSTEMS, OR COMPONENTS ARE MANAGED

Table 3.4.2.A-5 Steam and Power Conversion System
NMP1 Condenser Air Removal and Off-Gas System - Summary of Aging Management Evaluation
(3 Components)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
External Surfaces	LBS	*Cast Austenitic Stainless Steel*	Air	None	None			None
Valves	LBS	*Cast Austenitic Stainless Steel*	*Treated Water or Steam, temperature ≥ 482°F, Low Flow*	Loss of Material, Cracking	*One Time Inspection Program* *Water Chemistry Control Program*		3.4.1.A-02	F

Table 3.5.2.A-4 Structures and Component Supports
NMP1 Fuel Handling System - Summary of Aging Management Evaluation
(1 Component)

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG-1801 Volume 2 Item	Table 1 Item	Notes
AuxBridge	NSS	Aluminum Alloy	Air	None	None			None

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

NINE MILE POINT NUCLEAR STATION UNIT 1

FINAL SAFETY ANALYSIS REPORT (UPDATED)

OCTOBER 2009

REVISION 21

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
EP 1-1	EP 1-1
EP 2-1	EP 2-1
EP 3-1	EP 3-1
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 10-7	EP 10-1 thru EP 10-7
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
EP C-1	EP C-1

INSERTION INSTRUCTIONS

VOLUME 1

REMOVE	INSERT
i/ii	i/ii
iia/iib	---
iii/iv thru xa/xb	iii/iv thru xa/xb
---	xc/xd
xvii/xviia thru xxi/xxii	xvii/xviia thru xxi/xxii
xxv/xxvi thru xxvia/xxvib	xxv/xxvi thru xxvia/xxvib
xxxiii/xxxiv	xxxiii/xxxiv
xxxiva/xxxivb	---
xxxv/xxxvi	xxxv/xxxvi
xxxvii/xxxviii	xxxvii/xxxviia
---	xxxviib/xxxviii
xlvii/xlviia	xlvii/xlviia
xlix/l	xlix/l
---	la/lb
I-9/10	I-9/9a
---	I-9b/10
I-11/12 thru I-13/14	I-11/12 thru I-13/14
---	I-14a/14b
I-17/18	I-17/18
---	I-18a/18b
I-21/-	I-21/-
T I-2 Sh 1 thru 7	T I-2 Sh 1 thru 7
II-3/4	II-3/4
---	II-4a/4b thru II-4c/4d
II-9/-	II-9/-
T II-5/T II-6	T II-5/T II-6
---	T II-9 Sh 1/2
---	T II-10 Sh 1 & 2
III-9/10 thru III-12a/12b	III-9/10 thru III-12a/12b
III-39/40	III-39/40
F III-1	F III-1
IV-19/19a	IV-19/19a
IV-31/32	IV-31/32
V-3/4	V-3/3a
---	V-3b/4
V-21/-	V-21/-
T V-1 Sh 3/T V-2	T V-1 Sh 3/-
---	T V-2 Sh 1/2
T V-3/T V-4	T V-3/T V-4

INSERTION INSTRUCTIONS

VOLUME 2

REMOVE	INSERT
i/ii	i/ii
iia/iib	---
iii/iv thru xa/xb	iii/iv thru xa/xb
---	xc/xd
xvii/xviia thru xxi/xxii	xvii/xviia thru xxi/xxii
xxv/xxvi thru xxvia/xxvib	xxv/xxvi thru xxvia/xxvib
xxxiii/xxxiv	xxxiii/xxxiv
xxxiva/xxxivb	---
xxxv/xxxvi	xxxv/xxxvi
xxxvii/xxxviii	xxxvii/xxxviia
---	xxxviib/xxxviii
xlvii/xlviia	xlvii/xlviia
xlix/l	xlix/l
---	la/lb
VI-19/20 thru VI-23b/24	VI-19/20 thru VI-23b/24
VI-29/30	VI-29/30
T VI-3a Sh 4	T VI-3a Sh 4
VII-7b/8 thru VII-12a/12b	VII-7b/8 thru VII-12a/12b
---	VII-12c/12d
VII-14e/14f thru VII-14g/14h	VII-14e/14f thru VII-14g/14h
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T VIII-3 Sh 2	T VIII-3 Sh 2
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T VIII-3 Sh 10a/10b	T VIII-3 Sh 10a/10b
T VIII-4 Sh 2	T VIII-4 Sh 2
F IX-6	F IX-6

INSERTION INSTRUCTIONS

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i/ii	i/ii
iia/iib	---
iii/iv thru xa/xb	iii/iv thru xa/xb
---	xc/xd
xvii/xviia thru xxi/xxii	xvii/xviia thru xxi/xxii
xxv/xxvi thru xxvia/xxvib	xxv/xxvi thru xxvia/xxvib
xxxiii/xxxiv	xxxiii/xxxiv
xxxiva/xxxivb	---
xxxv/xxxvi	xxxv/xxxvi
xxxvii/xxxviii	xxxvii/xxxviia
---	xxxviib/xxxviii
xlvii/xlviia	xlvii/xlviia
xlix/l	xlix/l
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X-9/10	X-9/10
---	X-10a/10b
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---	X-43b/43c thru X-43f/44
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---	10A-14a/14b
10A-17/18	10A-17/18
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10A-91/92	10A-91/92
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10B-205/-	10B-205/-
---	10B-205a/-
10B-219/220 thru 10B-223/-	10B-219/220 thru 10B-223/224
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---	XII-16a/16b
XII-21/22	XII-21/21a
---	XII-21b/22
XIII-1/2 thru XIII-25/26	XIII-1/2 thru XIII-25/-
T XIII-1	T XIII-1 Sh 1/2
T XIII-2	T XIII-2
F XIII-1 thru F XIII-3	F XIII-1 thru F XIII-3
---	F XIII-3a
F XIII-4 thru F XIII-5	F XIII-4 thru F XIII-5

INSERTION INSTRUCTIONS

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iia/iib	---
iii/iv thru xa/xb	iii/iv thru xa/xb
---	xc/xd
xvii/xviia thru xxi/xxii	xvii/xviia thru xxi/xxii
xxv/xxvi thru xxvia/xxvib	xxv/xxvi thru xxvia/xxvib
xxxiii/xxxiv	xxxiii/xxxiv
xxxiva/xxxivb	---
xxxv/xxxvi	xxxv/xxxvi
xxxvii/xxxviii	xxxvii/xxxviia
---	xxxviib/xxxviii
xlvii/xlviia	xlvii/xlviia
xlix/l	xlix/l
---	la/lb
XV-31/32 thru XV-33/34	XV-31/32 thru XV-33/34
XV-43/44 thru XV-49/50	XV-43/44 thru XV-49/50
XV-55/56 thru XV-61b/62	XV-55/56 thru XV-61b/62
XV-67/68 thru XV-69/70	XV-67/68 thru XV-69/70
---	XV-70a/70b
XV-75/76 thru XV-77/78	XV-75/76 thru XV-77/78
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---	T XV-26 Sh 1/2
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F XV-56h	---
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---	F XV-74
XVI-23/24	XVI-23/24

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T XVI-9a Sh 1/2	T XVI-9a Sh 1/2
C.1-7/8	C.1-7/7a
---	C.1-7b/8
C.1-9/10 thru C.1-11/12	C.1-9/10 thru C.1-11/12
T C-1 Sh 1 thru 10	T C-1 Sh 1 thru 10

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

NINE MILE POINT NUCLEAR STATION UNIT 1

FINAL SAFETY ANALYSIS REPORT (UPDATED)

OCTOBER 2009

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DOCKET 50-220
LICENSE DPR-63

NINE MILE POINT NUCLEAR STATION UNIT 1

FINAL SAFETY ANALYSIS REPORT (UPDATED)

VOLUME 1

OCTOBER 2009

REVISION 21

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normal auxiliary cooling means during shutdown and refueling is the shutdown cooling system described in Section X-A. A redundant emergency cooling system, described in Section V-E, is provided to remove decay heat in the event the reactor is isolated from the main condenser while still under pressure. Additional cooling capability is also available from the high-pressure coolant injection (HPCI) system and the fire protection system.

Redundant and independent core spray systems are provided to cool the core in the event of a loss-of-coolant accident (LOCA). Automatic depressurization is included to rapidly reduce pressure to assist with core spray operation (see Section VII-A).

Operation of the core spray system assures that any metal-water reaction following a postulated LOCA will be limited to less than 1 percent of the Zircaloy clad.

7. Reactivity shutdown capability is provided to make and hold the core adequately subcritical, by control rod action, from any point in the operating cycle and at any temperature down to room temperature, assuming that any one control rod is fully withdrawn and unavailable for use.

   This capability is demonstrated in Section IV-B. A physical description of the movable control rods is given in Section IV-B. The control rod drive (CRD) hydraulic system is described in Section X-C.

   The force available to scram a control rod is approximately 3000 lb at the beginning of a scram stroke. This is well in excess of the 440-lb force required in the event of fuel channel pinching of the control rod blade during a LOCA, as discussed in Section XV. Even with scram accumulator failure, a force of at least 1100 lb from reactor pressure acting alone is available with reactor pressures in excess of 800 psig.

8. Redundant reactivity shutdown capability is provided independent of normal reactivity control provisions. This system has the capability, as shown in Section

VII-C, to bring the reactor to a cold shutdown condition at any time in the core life, independent of the control rod system capabilities. Cycle-specific results are contained in the SRLR[2].

9. A flow restrictor in the main steam line (MSL) limits coolant loss from the reactor vessel in the event of a MSL break (Section VII-F).

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4.0 Reactor Vessel

1. The reactor core and vessel are designed to accommodate tripping of the turbine generator, loss of power to the reactor recirculation system and other transients, and maneuvers which can be expected without compromising safety and without fuel damage.

   A bypass system having a design capacity of approximately 40 percent of turbine steam flow for the throttle valves wide open (VWO) condition partially mitigates the effects of sudden load rejection. An actual bypass system test was performed and the results indicated a system bypass capacity of about 2,500,000 lb/hr. This and other transients and maneuvers which have been analyzed are detailed in Section XV.

2. Separate systems to prevent serious reactor coolant system (RCS) overpressure are incorporated in the design. These include an overpressure scram, solenoid-actuated relief valves, safety valves and the turbine bypass system. An analysis of the adequacy of RCS pressure relief devices is included in Section V-C.

3. Power excursions which could result from any credible reactivity addition accident will not cause damage, either by motion or rupture, to the pressure vessel, or impair operation of required safeguards systems.

   The magnitude of credible reactivity addition accidents is curtailed by control rod velocity limiters (Section VII-D), by a control rod housing support structure (Section VII-E), and by procedural controls supplemented by a rod worth minimizer (RWM) (Section VIII-C). Power excursion analyses for control rod dropout accidents are included in Section XV.

4. The reactor vessel will not be substantially pressurized until the vessel wall temperature is in excess of the nil ductility reference temperature (RT_{NDT}) + 60°F. The initial RT_{NDT} of the reactor vessel material is no greater than 40°F. The change of RT_{NDT} with radiation exposure has been evaluated in accordance with Regulatory Guide (RG) 1.99 Revision 2 to determine an adjusted reference temperature (ART) for the most limiting vessel material. Vessel material surveillance samples are located within the reactor vessel to permit periodic verification of material properties with exposure.

5.0 Containment

1. The primary containment, including the drywell, pressure suppression chamber, and associated access

openings and penetrations, is designed, fabricated and erected to accommodate, without failure, the pressures and temperatures resulting from or subsequent to the double-ended rupture (DER) or equivalent failure of any coolant pipe within the drywell.

The primary containment is designed to accommodate the pressures following a LOCA, including the generation of hydrogen from a metal-water reaction. Pressure transients, including hydrogen effects, are presented in Section XV.

The initial NDTT for the primary containment system is about -20°F and is not expected to increase during the lifetime of the Station.

These structures are described in Sections VI-A, B and C. Additional details, particularly those related to design and fabrication, are included in Section XVI.

2. Provisions are made for the removal of heat from within the primary containment, for reasonable protection of the containment from fluid jets or missiles, and such other measures as may be necessary to maintain the integrity of the containment system as long as necessary following a LOCA.

   Redundant containment spray systems, described in Section VII, pump water from the suppression chamber through independent heat exchangers to spray nozzles, which discharge into the drywell and suppression chamber. Water sprayed into the drywell is returned by gravity to the suppression chamber to complete the cooling cycle. Studies performed to verify the capability of the containment system to withstand potential fluid jets and missiles are summarized in Section XVI.

3. Provision is made for periodic integrated leakage rate tests (ILRT) to be performed in accordance with 10CFR50 Appendix J. Provision is also made for leak testing penetrations and access openings and for periodically demonstrating the integrity of the reactor building. These provisions are all described in Section VI-F.

4. The containment system and all other necessary engineered safeguards were originally designed and maintained such that offsite doses resulting from postulated accidents are below the values stated in 10CFR100. The offsite doses have been re-analyzed in accordance with RG 1.183 and 10CFR50.67. The analysis results are detailed in Section XV.

5. Double isolation valves are provided on most lines directly entering the primary containment freespace, or penetrating the primary containment and connected to the RCS. Lines which are not equipped with double isolation valves have been determined to be acceptable based upon the fact that the system reliability is not compromised, the system is closed outside containment, and a single active failure can be accommodated with only one isolation valve in the line. Periodic testing of these valves will assure their capability to isolate at all times. The isolation valve system is discussed in detail in Section VI-D.

6. The reactor building provides secondary containment when the pressure suppression system is in service and serves as the primary containment barrier during refueling and other periods when the pressure suppression system is open or not required. This structure is described in Section VI-C. An emergency ventilation system (Section VII-H) provides a means for controlled release of halogens and particulates via filters from the reactor building to the stack under accident conditions.

6.0 Control and Instrumentation

1. The Station is provided with a control room (Section III-B) which has adequate shielding and other emergency features to permit occupancy during all credible accident situations.

2. Interlocks or other protective features are provided to augment the reliability of procedural controls in preventing serious accidents.

   Interlock systems are provided which block or prevent rod withdrawal from a multitude of abnormal conditions. The control rod block logic is shown on Figures VIII-6 and VIII-8, respectively, for the source range monitor (SRM) and intermediate range monitor (IRM) neutron instrumentation. In the power range, average power range monitor (APRM) instrumentation provides both control rod and recirculation flow control blocks, as shown on Figure VIII-14.

Reactivity excursions involving the control rods are either prevented or their consequences substantially mitigated by a control RWM (Section VIII-C.4.0) which supplements procedural controls in avoiding patterns of high rod worths, a local power range monitor (LPRM) neutron monitoring and alarm system (Section VIII-C.1.1.3), and a control rod position indicating system (Section IV-B.6.0), both of which enable the Operator to observe rod movement, thus verifying his actions. A control rod overtravel position light verifies that the blade is coupled to a withdrawn CRD.

A refueling platform operation interlock is discussed in Section XV, Refueling Accident, which, along with other procedures and supplemented by automatic interlocks, serves to prevent criticality accidents in the refueling mode.

A cold water addition reactivity excursion is prevented by the procedures and interlocks described in Section XV, Startup of Cold Recirculation Loop (Transient Analysis).

Containment integrity is maintained through the use of strict procedural controls and is enforced by interlocking mechanisms at the airlock doors to the drywell and a local alarm system at the access openings of the reactor building.

3. A reliable, dual-logic channel reactor protection system (RPS), described in Section VIII-A, is provided to automatically initiate appropriate action whenever various parameters exceed preset limits. Each logic channel contains two subchannels with completely independent sensors, each capable of tripping the logic channel. A trip of one-of-two subchannels in each logic channel results in a reactor scram. The trip in each logic channel may occur from unrelated parameters, i.e., high neutron flux in one logic channel coupled with high pressure in the other logic

channel will result in a scram. The RPS circuitry fails in a direction to cause a reactor scram in the event of loss of power or loss of air supply to the scram solenoid valves. Periodic testing and calibration of individual subchannels is performed to assure system reliability. The ability of the RPS to safely terminate a variety of Station malfunctions is demonstrated in Section XV.

4. Redundant sensors and circuitry are provided for the actuation of equipment required to function under post-accident conditions. This redundancy is described in the various sections of the text discussing system design.

7.0 Electrical Power

Sufficient normal and standby auxiliary sources of electrical power are provided to assure a capability for prompt shutdown and continued maintenance of the Station in a safe condition under all credible circumstances. These features are discussed in Section IX.

8.0 Radioactive Waste Disposal

1. Gaseous, liquid and solid waste disposal facilities are designed so that discharge of effluents is in accordance with 10CFR20 and 10CFR50 Appendix I. The facility descriptions are given in Section XII-A while the development of appropriate limits is covered in Section II.

2. Gaseous discharge from the Station is appropriately monitored, as discussed in Section VIII, and automatic isolation features are incorporated to maintain releases below the limits of 10CFR20 and 10CFR50 Appendix I.

9.0 Shielding and Access Control

Radiation shielding and access control patterns are such that doses will be less than those specified in 10CFR20. These features are described in Section XII-B.

10.0 Fuel Handling and Storage

Appropriate fuel handling and storage facilities which preclude accidental criticality and provide adequate cooling for spent fuel are described in Section X.

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Standby Liquid Control System Capability:

ppm	Shutdown Margin (Δk) (20C, Xenon Free)
SRLR(2)	SRLR(2)

9.0 Reactor Vessel

Inside Diameter	17 ft - 9 in
Internal Height	63 ft - 10 in
Design Pressure	1250 psig at 575°F

10.0 Coolant Recirculation Loops

Location of Recirculation Loops	Containment Drywell
Number of Recirculation Loops and Pumps	5
Pipe Size	28 in

11.0 Primary Containment

Type	Pressure Suppression
Design Pressure of Drywell Vessel	62 psig
Design Pressure of Suppression Chamber Vessel	35 psig
Design Leakage Rate	0.5 weight percent per day at 35 psig

12.0 Secondary Containment

Type	Reinforced concrete and steel superstructure with metal siding
Internal Design Pressure	40 lb/ft^2
Design Leakage Rate	100% free volume per day discharged via stack while maintaining 0.25-in water negative pressure in the reactor building relative to atmosphere

13.0 Structural Design

Seismic Ground Acceleration	0.11g
Sustained Wind Loading	125 mph, 30 ft above ground level
Control Room Shielding	Normal Operation - Dose not to exceed hourly equivalent (based on 40-hr week) of maximum permissible quarterly dose specified in 10CFR20. Accident Conditions - Meets the design total effective dose equivalent (TEDE) dose for personnel in the control room such that the exposure limits of 10CFR50.67 will not be exceeded in the course of the LOCA. In addition, the cumulative dose from any design basis accident (DBA) would also meet 10CFR50.67 limits.

14.0 Station Electrical System

Incoming Power Sources	Two 115-kV transmission lines
Outgoing Power Lines	Two 345-kV transmission lines
Onsite Power Sources Provided	Two diesel generators Two safety-related Station batteries One Q-related 125-V dc battery system

15.0 Reactor Instrumentation System

Location of Neutron Monitor Sensors	In-core

Ranges of Nuclear Instrumentation:

Four Startup Range Monitors	Source to 0.01% rated power and to 8.3% with chamber retraction
Eight Intermediate Range Monitors	0.0003% to 40% rated power
120 Power Range Monitors	5% to 125% rated power

16.0 Reactor Protection System

Number of Channels in Reactor Protection System	2
Number of Channels Required to Scram or Effect Other Protective Functions	2
Number of Sensors per Monitored Variable in each Channel (Minimum for scram function)	2

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E. REFERENCES

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

2. 0000-0084-3226-SRLR, Revision 0, "Supplemental Reload Licensing Report for NMP1, Reload 20, Cycle 19," December 2008.

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
ALRA	Amended license renewal application
AMP	Aging Management Program
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
ART	Adjusted reference temperature
AST	Alternative source term
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
BWRT	Backwash receiving tank
BWRVIP	Boiling Water Reactor Vessel and Internals Program
CAD	Containment atmosphere dilution (device)
CCCWS	Closed-cycle cooling water system
CEO	Chief Executive Officer
CFR	Code of Federal Regulations
CFS	Condensate filtration system
CGCS	Combustible gas control system
CHF	Critical heat flux
CIV	Combined intermediate valve
CND	Condensate demineralizer
CO_2	Carbon dioxide
COLR	Core Operating Limits Report
CPR	Critical power ratio
CRD	Control rod drive
CRDA	Control rod drop accident
CRDRL	Control rod drive return line

TABLE I-2 (Cont'd.)

CRPI	Control rod position indication
CRS	Control Room Supervisor
CRT	Cathode ray tube
CSO	Chief Shift Operator
CST	Condensate storage tank
CUF	Cumulative usage factor
CWT	Concentrated waste tank
DAC	Dominant area of concern
DBA	Design basis accident
DBE	Design basis earthquake
DCRDR	Detailed control room design review
DE	Dose equivalent
DEC	Department of Environmental Conservation
DER	Deviation/Event Report
DER	Double-ended rupture
DG	Diesel generator
DOP	Dioctylphthalate
DOT	Department of Transportation
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
EQ	Environmental qualification
ESF	Engineered safety feature
ESW	Emergency service water
FA	Fire area
FAC	Flow-accelerated corrosion
FCV	Flow control valve
FHA	Fire Hazards Analysis
FMEA	Failure modes and effects analysis
FMP	Fatigue Monitoring Program
FRC	Franklin Research Center

TABLE I-2 (Cont'd.)

FSA	Fire subarea
FSAR	Final Safety Analysis Report
FZ	Fire zone
GALL	Generic aging lessons learned
GDC	General Design Criterion
GE	General Electric Company
GL	Generic Letter
GSI	Generic Safety Issue
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
ISP	Integrated Surveillance Program
IST	Inservice testing
LCO	Limiting condition of operation
LHGR	Linear heat generation rate
LLD	Lower limit of detection
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
LRA	License renewal application
LSSS	Limiting safety system setting

TABLE I-2 (Cont'd.)

LTC	Load tap changer
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
NEI	Nuclear Energy Institute
NEIL	Nuclear Electric Insurance Limited
NFPA	National Fire Protection Association
NMPC	Niagara Mohawk Power Corporation
NPSH	Net positive suction head
NRC	Nuclear Regulatory Commission
NRV	Nonreturn valve
NSRB	Nuclear Safety Review Board
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
OCCWS	Open-cycle cooling water system
OEA	Operating experience assessment
OL	Operating license
OLNC	On-Line NobleChem
OOS	Out of service
OSC	Operational Support Center
OT	Operational transient
PA	Public address (system)
PASS	Post-accident sampling system
PCI	Pellet-cladding interaction

TABLE I-2 (Cont'd.)

PCT	Peak cladding temperature
p.f.	Power factor
P&ID	Piping and instrumentation diagram
PM	Preventive maintenance
PORC	Plant Operations Review Committee
PP/PA	Page party/public address (system)
PSAR	Preliminary Safety Analysis Report
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
RIP	Reactor internals protection
RIS	Regulatory Issue Summary
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
RPVH	Reactor pressure vessel head
RSP	Remote shutdown panel
RSS	Remote shutdown system
RTD	Resistance temperature detector
RT_{NDT}	Reference temperature nil ductility transition
RWCU	Reactor water cleanup
RWE	Rod withdrawal error
RWM	Rod worth minimizer
RWP	Radiation work permit
SAG	Severe accident guideline
SAP	Severe accident procedure
SAR	Safety analysis report
SAS	Secondary alarm system
SBO	Station blackout
SCBA	Self-contained breathing apparatus

TABLE I-2 (Cont'd.)

SCC	Stress corrosion cracking
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
SM	Shift Manager
SOE	Sequence of events
SOP	Special operating procedure
SORC	Station Operations Review Committee
SOV	Solenoid-operated valve
SPDS	Safety parameter display system
SR	Surveillance requirement
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
SRVDL	Safety/relief valve discharge line
SSA	Safe Shutdown Analysis
SSC	Structures, systems and components
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
TEDE	Total effective dose equivalent
TER	Technical Evaluation Report
TIP	Traversing in-core probe
TLAA	Time-Limited Aging Analyses
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
UFSAR	Updated Final Safety Analysis Report
UHS	Ultimate heat sink

TABLE I-2 (Cont'd.)

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
USE	Upper-shelf energy
USLS	U.S. Land Survey
UT	Ultrasonic testing
VWO	Valve wide open
WNT	Waste neutralizer tank
WSLR	Within scope of license renewal

B. DESCRIPTION OF AREA ADJACENT TO THE SITE

1.0 General

The Station is located on the Lake Ontario coast in the town of Scriba in the north-central portion of Oswego County, approximately 5 mi north-northeast of the nearest boundary of the city of Oswego.

1.1 Population

Population growth in the vicinity of the Station has been very slow, with the city of Oswego showing a decrease in population. The 1960 census enumerated 22,155 residents compared to approximately 19,793 people in 1980. However, county population increased from 86,118 in 1960 to 113,901 in 1980. The total 1980 population within 12 mi of the Station is estimated to be 46,349 (see Figure II-4). This area contains all or portions of one city and ten towns. Population and population density for the ten towns and one city within this area are shown in Table II-1. Counties and towns within this area are shown on Figure II-5.

Transient population within 12 mi of the Station is limited due to the rural, undeveloped character of the area. There are, however, a number of school, industrial, and recreational facilities in the area that create small daily and seasonal changes in area populations.

The population within a 50-mi area surrounding the Station was approximately 914,193 in 1980 (see Figure II-6). The city of Syracuse is the largest population center within this area, with a population of 170,105 in 1980. Table II-2 lists cities within this 50-mi radius with populations over 10,000.

The 50-mi radius contains portions of three Canadian Census Divisions located in the province of Ontario: Prince Edward, Frontenac, and Addington/Lennox. The 1976 population counts totaled 22,559, 108,052, and 32,633, respectively.

2.0 Agriculture, Industrial and Recreational Use

2.1 Agricultural Use

The area within a 50-mi radius of the site encompasses all or portions of ten New York counties: Cayuga, Jefferson, Lewis, Madison, Oneida, Onondaga, Ontario, Oswego, Seneca, and Wayne.

Approximately 37 percent of the land within this ten-county region is used for agricultural production. Tables II-3 and II-4 present agricultural statistics for this ten-county region.

2.2 Industrial Use

Several industrial establishments are located in Oswego County, with the Novelis Corporation and the Independence Generation Plant operated by Sithe Energies USA being located nearest to the Station. The lakeshore east of Oswego is the most industrially developed area near the site. The cities of Fulton and Mexico are the only other industrial sites within 15 mi of the site. Two natural gas pipelines lie within 8 km of the plant; one pipeline supplies the Independence Plant and the other supplies Indeck Energy. Both pipelines are located on the north-south and east-west transmission line corridors. The major industrial establishments in Oswego County, their locations, and their principal products are listed in Tables II-5 and II-6.

The nearest public water supply intake in Lake Ontario is located approximately 8 mi southwest of the Station location. This intake supplies the city of Oswego and Onondaga County. Data on these and other vicinity public water supplies are listed in Table II-7. Figure II-2 shows the locations of the communities listed.

2.2.1 Toxic Chemicals

Potential Sources of Toxic Chemicals

According to Regulatory Guide (RG) 1.78, both onsite and offsite potential toxic gas hazards must be considered. Any toxic substance stored onsite in a quantity greater than 45 kg (100 lb) must be evaluated. Offsite sources to be evaluated include stationary facilities and frequent transportation of toxic substances (truck, rail, and barge) within 8 km (5 mi) of the site. Frequent shipments are defined as exceeding 10/yr for truck shipments, 30/yr for rail shipments, and 50/yr for barge shipments.

For the NMPNS site, sources of potential toxic chemical hazards include chemicals stored onsite, as well as stationary and transportation sources within 8 km of the site. Table II-9 lists the chemicals associated with each source along with their quantities and distances from the Unit 1 control room air intake. The stationary sources include the James A. FitzPatrick plant, Novelis Corporation, Oswego Wire Inc., Sithe Independence Station, and Unit 2. One transportation source of possible hazardous materials is truck traffic along Route 104, which passes within 6.2 km (3.9 mi) of the site. Another transportation source is the railroad line between Oswego and Mexico, NY. Discussions with Conrail indicate that on average, only one hazardous chemical shipment during an 18-mo period passes through the Oswego terminal. Traffic on a spur to the site is not frequent enough (<30/yr) to warrant consideration.

Only those chemicals that have the potential to form a toxic vapor cloud or plume after release to the environment need to be evaluated. This criterion is met by all chemicals listed in Table II-9.

Control Room Habitability Determination

The effect of an accidental release of each of the chemicals described in the previous section on control room habitability is evaluated by calculating vapor concentrations inside the control room as a function of time following the accident. This calculation is performed using the conservative methodology outlined in NUREG-0570 and utilizing the assumptions described in RG 1.78.

In a postulated accident, the entire content of the largest single storage container is released, resulting in a toxic vapor cloud and/or plume that is conservatively assumed to be transported by the wind directly toward the control room intake. The formation of the toxic cloud and/or plume is dependent on the characteristics of the chemical and the environment. The entire amount of a chemical stored as a gas is treated as a puff or cloud that has a finite volume determined from the quantity and density of the stored chemical. A substance stored as a liquid with a boiling point below the ambient temperature forms an instantaneous puff due to flashing (rapid gas formation) of some fraction of the stored quantity. The remaining liquid forms a puddle that quickly spreads into a thin layer on the ground, subsequently vaporizing and forming a ground-level vapor plume. A high boiling point liquid (above ambient temperature) forms a puddle that evaporates by forced convection with no flashing involved.

The calculations are done by a computer program (VAPOR) or a spreadsheet, both based on NUREG-0570 methodology that requires the following input information: chemical physical properties, control room parameters, meteorology, distance from the spill to the control room intake, and the quantity of chemical released. The following Unit 1 control parameters are used: ventilation rate of 2530 ft^3/min, and net free volume of 130,600 ft^3. The most conservative meteorological conditions are assumed for the calculations, consisting of Pasquill Class A stability, 0.5 m/sec wind speed and an ambient temperature of 33°C for sodium bisulfite solution stored onsite, and Class F stability, a wind speed of 1.0 m/sec, and an ambient temperature of 90°F for all other chemical releases.

The criteria for determining chemical toxicity and setting limits for habitability determinations are taken from regulatory guidance documents. According to RG 1.78, the toxicity limit of a chemical is the maximum concentration that can be tolerated by an average human for 2 min without physical incapacitation (severe coughing, eye burn, severe skin irritation). Standard Review Plan (SRP) Section 6.4 states that acute effects should be reversible within a short period of time (several minutes) without the benefit of medication other than the use of self-contained breathing apparatus (SCBA). The acute toxicity limits listed in RG 1.78 are used in this study except that, where more appropriate, documented sources are available[2-5].

Nonguideline toxicity limits are based on concentrations that produce no effects or minor irritation affecting mental alertness and physical coordination, assuming a 15-min exposure time. In cases where appropriate human data are not available, data are used by applying a conservative factor of 10 to lower the acute exposure limit.

The effect of the continuous outside venting of the onsite sodium bisulfite storage tank on control room habitability is evaluated by calculating the maximum sulfur dioxide vapor concentration at the control room intake. The evaluation is performed using the guidance described in RG 1.78. The toxicity limit is set at the TLV-TWA limit established in NUREG/CR-5669[5] for sulfur dioxide.

Results and Conclusions

The results of the analysis are summarized in Table II-10, which indicates that none of the toxic chemicals evaluated have the potential to incapacitate the control room operators.

2.3 Recreational Use

Seventeen state parks and one national wildlife refuge are located within a 50-mi radius of the Station. Table II-8 identifies the state parks and their facilities, capacities, and visitor counts. The Montezuma National Wildlife Refuge is located north of Cayuga Lake in Seneca County, approximately 44 mi southwest of the Station.

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G. REFERENCES

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

2. Sax, N. E. Dangerous Properties of Industrial Materials, 3rd Edition, Van Nostrand Reinhold, New York, NY, 1968.

3. NUREG/CR-5669, "Evaluation of Exposure Limits to Toxic Gases for Nuclear Reactor Control Room Operators," July 1991.

4. CRC Handbook of Chemistry and Physics, 76th Edition, David R. Lide, Editor-in-Chief.

5. Air Contaminants - Permissible Exposure Limits, Title 29 Code of Federal Regulations Part 1910-1000, OSHA 3112, 1989.

TABLE II-5

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

Firm	Distance/ Direction from Site (km)	Products	Employment
Novelis 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
Oswego Wire Incorporated	7.0/SW	Copper wire	40

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

TABLE II-9

SOURCES OF TOXIC CHEMICALS WITHIN
8 KM (5 MI) OF UNIT 1 SITE

Chemical Location	Chemical	Quantity Stored	Distance to Intake
James A. FitzPatrick Plant	N_2	40,432 lb	0.497 mi
	CO_2	26,000 lb	
	Propane	<10,000 lb	
	Halon 1301	6,000 lb	
	NaOCI	4,537 lb	
	NaOH	1,694 lb	
	Gasoline	6,005 lb	
	H_2	1,150 lb	
	Freon R-12	1,695 lb	
	Freon R-22	6,010 lb	
Novelis Corporation	CI_2	1,500 lb	3.0 mi
	HCl	5,000 lb	
	CO_2	118,000 lb	
	Propane	80,000 lb	
	N_2	50,000 lb	
	NaOH	4,200 gal	
	H_2SO_4	5,000 gal	
Route 104	HCl	12,000 lb	3.4 mi
	CO_2	6,000 lb	
	N_2	40,000 lb	
Nine Mile Point Unit 1	H_2SO_4	165 gal	
	H_2	12,000 ft^3	112 m
	CO_2	20,000 lb	100 m
	N_2	15,300 gal	140 m
	Halon 1301	500 lb	
	NaOH	165 gal	
	NaOCI	1,200 gal	

TABLE II-9 (Cont'd.)

Chemical Location	Chemical	Quantity Stored	Distance to Intake
Nine Mile Point Unit 2	H_2SO_4	11,925 gal	
	$NaHSO_3$	25,548 gal	100 m
	H_2	353.1 lb	103 m
	CO_2	52,000 lb	167 m
	Propane	250 gal	640 m
	N_2	2,052,000 scf	229 m
	Halon 1301	250 lb	
	NaOH	10,160 gal	
	NaOCI	3,930 gal	
	Ethylene Glycol	2,400 gal	
Oswego Wire Incorporated	H_2SO_4	70 gal	4.39 mi
	HCl	15 gal	
	Isopropyl Alcohol	65 gal	
	Propane	500 gal	
	N_2	2,500 gal	
	NaOH	67 gal	
Sithe Energies, Inc.	Ammonia	60,280 lb	2.17 mi
	H_2	531 lb	
	CO_2	64,000 lb	
	N_2	2,691,731 lb	
	NaOH	6,747 gal	
	NaOCI	9,112 lb	
	H_2SO_4	11,997 gal	

TABLE II-10

PREDICTED VAPOR CONCENTRATION IN
THE UNIT 1 CONTROL ROOM

Exclusion Criteria Comparisons					
Chemical Location	Chemical	Toxic Limit (mg/m^3)	Control Room Weight Limit (lb)	Max. Single Container Storage (lb)	Distance to Unit 1 Control Room (mi)
J. A. FitzPatrick	CO_2	17,998.15	37,200	28,000	0.497
	H_2	7,495.04	15,500	2,084	
	N_2	104,147.59	215,500	35,983	
	Propane	163,991.17	339,300	4,880	
Novelis Corporation	CO_2	17,998.15	4,840,900	114,000	3.0
	CI_2	43.5	11,700	4,000	
	HCl	52.19	14,000	11,891	
	N_2	104,147.59	28,012,100	50,000	
	Propane	163,991.17	44,108,000	390,400	
Oswego Wire Incorporated	HCl	52.19	64,800	149	4.39
	N_2	104,147.59	129,286,700	17,992	
	Propane	163,991.17	203,575,200	2,440	
	Isopropyl Alcohol	1,229.12	1,525,800	435	
Sithe Energy	CO_2	17,998.15	4,840,900	64,000	2.17
	H_2	7,495.04	2,015,900	531	
	N_2	104,147.59	28,012,100	2,691,731	

TABLE II-10 (Cont'd.)

Quantitative Results Comparisons				
Chemical Location	Chemical	Toxic Limit (ppm)	Control Room Max. Conc. (ppm)	Distance to Unit 1 Control Room (m)
Nine Mile Point Unit 1	CO_2	10,000	6,160	100
	H_2	90,909	2,760	112
	N_2	90,909	11,760	140
Nine Mile Point Unit 2	$NaHSO_3$asSO_2	0.262 g/m^3 5.24E-03 g/m^3	2.67E-2 g/m^3* 1.69E-3 g/m^3**	100
	H_2	90,909	4,090	103
	CO_2	10,000	6,400	167
	Propane	1,000	142	640
	N_2	90,909	14,720	229
Sithe Energy	Ammonia	300	207	3,500

Notes:

* For a release of sodium bisulfite solution to the containment berm.

** For continuous outside venting of the sodium bisulfite storage tank.

B. CONTROL ROOM

The control room is located in the southeast corner of the turbine building at el 277. It is bounded by the administration building offices on the south and east, the turbine room on the west, and the control room break area, instrumentation and control (I&C) office area, and diesel building on the north.

1.0 Design Bases

1.1 Wind and Snow Loadings

The wind and snow loadings for the control room are the same as for the turbine building.

1.2 Pressure Relief Design

There are no special pressure relief requirements for the control room.

1.3 Seismic Design and Internal Loadings

The structural design for the control room, as well as the auxiliary control room below at el 261, is Class I seismic based on the maximum credible earthquake motion outlined in the introduction to Section III. Components whose functional failure could cause significant release of radioactivity, or which are vital to safe shutdown and isolation of the reactor, are also designed as Class I. The seismic analysis resulted in the application of acceleration factors of 20.0 percent gravity horizontal and 10.0 percent gravity vertical. These acceleration factors were calculated from the dynamic analysis of the turbine building.

Although the control room is structurally a part of the turbine building, functional load stresses when combined with stresses due to earthquake loading are maintained within the established working stresses* for the structural material involved.

1.4 Heating and Ventilation

Heating and air conditioning are provided for personnel comfort and instrument protection. The ventilating system also provides clean air to the control room following an accident.

1.5 Shielding and Access Control

Normal access to the control room is provided from the administration building through security-controlled doors.

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* Also see Section XVI, Subsection G.

Shielding is supplied to allow continuous occupancy during any reactor accident. The most limiting accidents are the control rod drop accident (CRDA) and the loss-of-coolant accident (LOCA) without core spray, which are described in Section XV.

The shielding also meets the design TEDE dose rate for personnel in the control room such that the exposure limits of 10CFR50.67 will not be exceeded in the course of the LOCA. In addition, the cumulative dose from any design basis accident (DBA) would also meet 10CFR50.67 limits. Credit is taken for automatic initiation of the control room air treatment system for the LOCA. If air outside the building is contaminated, the ventilating system will be controlled to assure that contamination within the control room is minimized and kept within the above limits, as shown in Section 3.0, following.

2.0 Structure Design

Plans showing location and principal dimensions are shown on Figures III-4, III-5, and III-6.

2.1 General Structural Features

The structural steel enclosing the control room and the auxiliary control room below is supported on concrete walls and concrete foundations bearing on and keyed into sound rock. Actual rock bearing pressures are less than one-third of the allowable working bearing pressure. Lateral earthquake forces or wind loads are transmitted to the concrete foundations by the combination of structural steel bracing and concrete walls.

The control room walls, roof and floors are framed with structural steel. The west and north interior walls are 12-in solid reinforced concrete. The east wall is enclosed with insulated metal wall panels made up of FK-16 x 16 metallic-coated interior liner elements, 1 1/2-in insulation and 16 B & S gage F-2 porcelainized aluminum exterior face sheets, as manufactured by H. H. Robertson Company. The wall panel joints are sealed with a synthetic elastomer caulking material. This wall is separated from the administration building extension by a 3-in rattle space. The south interior wall consists of 8-in concrete blocks laid with steel-reinforced mortar joints. An interior metal partition wall parallel to the south wall forms a 6'-6" corridor and is provided with windows for observing the control room operations from the corridor.

The slab immediately above the control room at el 300 is pinned to the walls and provides radiation shielding, and consists of 8 1/2-in thick poured-in-place reinforced concrete supported on structural steel beam framing. Two-thirds of this slab area has a roof above at el 333 which is made up of 3-in deep metal decking, 2 in of insulation and a 5-ply roof with slag surface. The remaining third of the slab area provides a shielding roof over the control room and consists of the 8 1/2-in thick poured-in-place reinforced concrete slab to which is applied 1 1/2 in of rigid insulation and a 5-ply roof with slag surface.

The control room floor is poured-in-place reinforced concrete on 14-gauge metal decking. The gross depth of the floor slab is 8 in and the average depth of concrete is 5 3/4 in.

2.2 Heating, Ventilation and Air Conditioning System

The ventilation system shown on Figure III-14 is designed to provide outside and recirculated air to the control room and auxiliary control room areas during normal and emergency conditions.

In the normal ventilation mode, outside air enters the system through a louvered intake after which it passes through a 15-kW duct heater and normal supply isolation dampers, which are interlocked with the emergency ventilation inlet dampers. Outside air is needed to recoup air from leakage and losses and to maintain a habitable environment for personnel. The outside air then flows through an outside air mix damper and is then mixed with recirculated control room return air from the recirculation damper, which is set to maintain a positive pressure in the control room. The total amount of air (14,500 cfm minimum) then passes through a two-element dust filter and redundant cooling coils where it will be cooled, if necessary, to ensure the control room temperature does not exceed the maximum calculated temperature of 80.5°F. The cooled air enters the control room circulation fan for distribution to various areas through ducts. Air will circulate through the control room to the return ductwork for recirculation and mixing with additional outside air. In order to prevent infiltration of potentially contaminated air, doors are weather-stripped and penetrations are sealed to maintain a positive pressure to the turbine building of 1/16 in of water.

The emergency ventilation system is automatically initiated on high radiation signal from the intake radiation monitors, LOCA and/or MSLB signal from the reactor protection system (RPS), or manually initiated when required by procedures. The normal supply isolation dampers will be automatically closed, and the emergency ventilation inlet dampers will be opened. The outside air will then flow through a 15-kW duct heater and one of the full capacity control room emergency fans. The design flow rate for the control room emergency ventilation system is 2250 cfm ±10%. Air then passes through a manual throttling damper, a high efficiency particulate filter, and an activated charcoal filter unit. This filtered air will then join the normal supply ductwork and mix with control room return air to be circulated by the normal control room circulation fan. The design flow rate for the emergency ventilation system outside air is determined as that necessary to maintain a positive pressure of 1/16 in of water to the turbine building, administration building, and outside atmosphere, and is a function of control room boundary leakage. The design flow rate of 2250 cfm ±10% is within the required range of 1000 to 3750 cfm which is based on minimum required fresh air for personnel and maximum filter capability.

The emergency ventilation fans may be manually started for periodic testing.

Heating is provided by thermostatically-controlled ventilation duct heaters. Cooling is provided by two chiller units. Both the temperature control valve and/or the bypass valve for the chilled water system may be open without overcooling the control room.

Tests and inspections on the control room emergency ventilation filters are done in accordance with Technical Specifications.

2.3 Smoke and Heat Removal

To assist in maintaining a habitable atmosphere in the control room and auxiliary control room, a smoke purge capability is provided from two independent fans, one 6000-cfm makeup fan and one 8000-cfm exhaust fan (Figure III-14).

Qualitative smoke evaluations have been performed for NMP1. The evaluations assessed the effects of both external and internal fire/smoke events on the capability to maintain reactor control from either the control room or remote shutdown panels. The evaluations considered various plant design and procedural criteria in accordance with RG 1.196, "Control Room Habitability at Light Water Nuclear Power Plants," and NEI 99-03, "Control

Room Habitability Guidance," Revision 1. The evaluations confirmed that egress pathways to and including the remote shutdown panels are served by ventilation systems independent of the control room and that no single smoke/fire event could preclude use of both the control room and remote shutdown panels.

2.4 Shielding and Access Control

Normal personnel access to the control room is provided by three controlled access doors all located on el 277. The north door opens into the control room break area, the south door opens into the administration building, and the west door opens into a corridor, giving access to the administration building at el 277 and also making available the stairway to el 261 of the administration building.

In addition to the above, a stair is provided within the control room (northwest corner) down to the auxiliary control room on the ground floor, shown on Figure III-4. In case of a reactor accident, personnel access to or from the control room would be from the southerly extreme of all buildings and approximately 400 ft from the center of the reactor.

The walls, roof and floors are designed to have concrete thicknesses which provide shielding during the design basis accident (DBA).

3.0 Safety Analysis

The control room is designed for continuous occupancy by operating personnel during normal operating or accident conditions. Concrete shielding provided in the roof and floors above and in the walls facing the reactor building is more than sufficient to ensure the exposure limits of 10CFR50.67 will not be exceeded in the course of a LOCA. Maintaining positive pressure inside the control room and regulating the filtered outside air supply prevents the concentration of radioactive materials and ensures that the cumulative dose from the LOCA accident will be within the exposure limits of 10CFR50.67.

In addition, supplied air respirators are available in the control room for use if necessary.

Tracer gas testing is performed periodically using the constant injection method of ASTM E741-00, "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas

Dilution." For the constant injection method, a constant flow of tracer gas is injected into the control room envelope (CRE) until the resulting concentration reaches a steady state value. This occurs when the amount of tracer gas entering the CRE is the same as the amount leaving the CRE. By injecting the tracer gas in the outside airflow used for pressurization of the envelope, an estimate of the filtered and unfiltered airflow that provides this pressurization can be made by measuring the concentration of tracer gas in the outside airflow while at the same time measuring the steady state concentration in the CRE.

A CRE habitability program has been established to ensure that CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge.

Both normal and emergency lighting are provided in the control room together with communications, air conditioning, ventilation, heating and sanitary plumbing facilities. If normal electric power service is not available, provision has been made to power the cooling, ventilating and heating units from the emergency diesel generators.

Building components and finish materials are noncombustible and combustible materials are not stored in the control room.

The minimum distance of the control room to the centerline of the reactor is 330 ft and there are no direct connections from passageways, ventilating ducts or tube connections between the reactor building and the control room.

The floor of the control room is 16 ft above yard grade and 28 ft above maximum lake level (el 249). Therefore, the possibility of flooding or inundation is incredible.

H. SECURITY BUILDING WEST AND SECURITY BUILDING ANNEX

The security building west and security building annex are located on the southwest corner of the Station security perimeter. See Figure III-1.

Administrative offices are contained within these buildings for support of the duties associated with Station security.

Because of the nature of this subject, a detailed description of these buildings will not be discussed in this document. For additional information regarding this subject, refer to the Station security plan.

I. RADWASTE SOLIDIFICATION AND STORAGE BUILDING

1.0 Design Bases

1.1 Wind and Snow Loadings

Wind and snow loadings for the radwaste solidification and storage building (RSSB) are designed to meet or exceed those of the waste disposal building.

1.2 Pressure Relief Design

There are no special pressure relief requirements for this building.

1.3 Seismic Design and Internal Loadings(1)

The foundation mat, structural walls, columns, floors and roof of the RSSB are classified as primary structural elements. All primary structural elements are seismically designed to withstand the effects of an operating basis earthquake (OBE) in accordance with Regulatory Guide (RG) 1.143.

Secondary structure elements, including platforms, catwalks, pipe supports, equipment and vessel supports, and internal masonry walls, are classified as nonseismic-resistant items and are designed by conventional method.

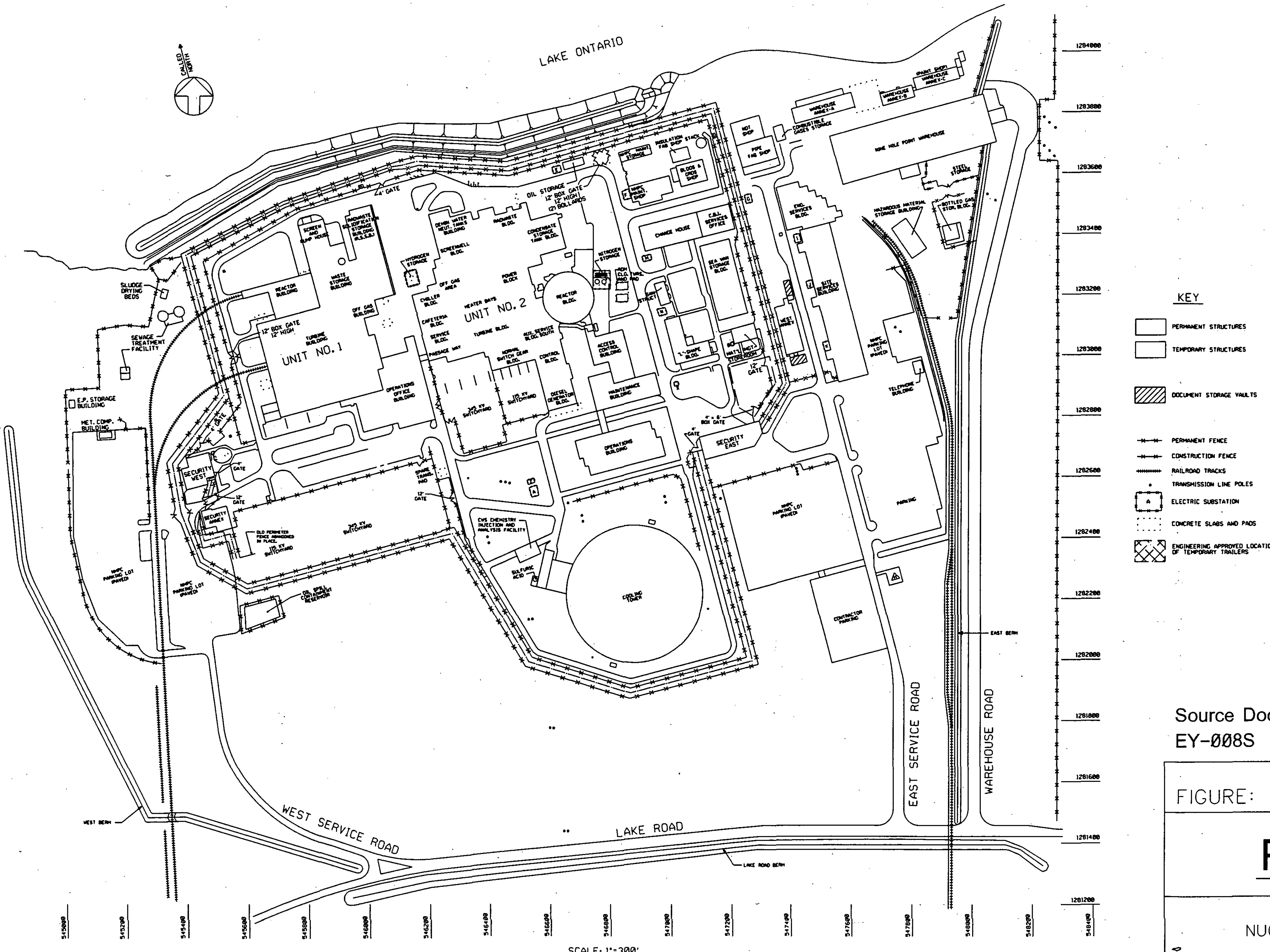
1.4 Heating, Ventilation and Air Conditioning(2)

The heating, ventilation and air conditioning (HVAC) and chilled water systems are designed for the following primary functional requirements: heat, ventilate and air condition the RSSB; remove airborne particulates from the RSSB atmosphere; prevent unfiltered exfiltration of airborne radioactivity from the building; prevent infiltration of airborne radioactivity into the RSSB control room and electrical room; control and provide a means for monitoring (via the main stack) the release of airborne radioactivity via the ventilation exhaust system; minimize the effects on the facility and its occupants from releases of radioactivity into the RSSB atmosphere; collect and filter air displaced via the vents from all RSSB tanks containing radioactive fluids; continuously purge the RSSB of truck exhaust fumes and other hazardous gases to ensure safe occupancy at all times.

1.5 Shielding and Access Control(3)

Shielding is designed to limit radiation levels on the building exterior, in the control room, in the electrical room, stairwells, and the passageway to the truck bays.

Access to the exterior of the RSSB is controlled by access to the protected area, which is controlled by Nuclear Security. Normal



Source Document:
EY-008S

FIGURE: III-1

Plot Plan

NINE MILE POINT
NUCLEAR STATION - UNIT 1
SCRIBA, N.Y.
UPDATED SAFETY ANALYSIS REPORT

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 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

1. 0000-0084-3226-SRLR, Revision 0, "Supplemental Reload Licensing Report for NMP1, Reload 20, Cycle 19," December 2008.

2. Randall and St. John, Nucleonics 16(3), 82-86, 129 (1958).

3. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-16-US, October 2007.

4. J. A. Wooley, "Three-Dimensional BWR Core Simulator," NEDO-20953A, January 1977.

5. "Qualification of the One-Dimensional Core Transient Model for BWR's," NEDO-24154, Vol. 1 and 2, and NEDE-24154-P-A, Vol. 3, February 1, 1986.

6. R. B. Linford, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, December 1986, and Amendments.

7. GE Fuel Bundle Designs, General Electric Company Proprietary, NEDE-31152P, Revision 8, April 2001.

8. GENE-770-31-1292, Rev. 2, "Engineering Report for Application of GE11 to NMP1 Reload 12," General Electric Co. Proprietary Document, April 1993.

9. Safety Evaluation of the General Electric Advanced Long Life Control Rod Assembly, NEDE-22290, January 1985.

10. Letter from C. O. Thomas (NRC) to J. F. Klapproth (GE), July 1, 1985.

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.

4.0 Cyclic Loads (Mechanical and Thermal)

Fatigue resistance of the reactor vessel was originally analyzed based on the expected number of operating cycles over the 40-yr life of the vessel. Table V-2 lists the operating cycles evaluated and their expected number of cycles. Using the operating cycles in Table V-2, stress analyses were performed on the feedwater nozzles, control rod drive (CRD) penetrations, lower vessel head, vessel support skirt, core support cone, vessel wall, other nozzles in the vessel, closure studs and the basin seal skirt weld.

Fatigue usage factors, utilizing the expected number of operating cycles in Table V-2, were calculated as follows:

$$u = \frac{n_i}{N_i}$$

Where:

u	=	fatigue usage factor
n_i	=	expected number of cycles of a given stress amplitude
N_i	=	maximum allowable number of cycles at the same stress amplitude

The calculated usage factors (Table V-3) were all within the allowable design limits (General Electric Company (GE) - 0.8, ASME Section III-1965 - 1.0).

Except for the reactor recirculation nozzles, stress analysis on other nozzles in the reactor vessel concluded that they were subjected to significantly less severe transients than the feedwater nozzles and, therefore, their fatigue usage factors were all negligible. Stress analyses of recirculation nozzle thermal transients conclude that the nozzle fatigue usages are negligible. The vessel wall was also concluded to have a negligible fatigue usage factor.

The above analyses were based on the expected number of operating cycles using assumed parameters for each transient. In addition, NMP has implemented a Fatigue Monitoring Program (FMP) that is used to manage the fatigue life of analyzed components. This program utilizes the FatiguePro software to maintain all actual calculated cumulative usage factors (CUF) below their corresponding allowables. The calculations are based on the actual, rather than expected, number of cycles experienced by the plant for each transient and, in some cases, the actual rather than assumed parameters experienced by each cycle.

5.0 Codes

Applicable codes for the RCS are included in Table V-4. Discussion of calculations demonstrating Code adherence are given in Section XV. Further summaries are provided in Section XVI.

Codes applicable up to the outside of the second isolation valve on all auxiliary and emergency systems are also given in Table V-4.

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B. SYSTEM DESIGN AND OPERATION

1.0 General

1.1 Drawings

A flow diagram of the RCS is shown on Figure V-1. This system is defined as encompassing the reactor primary system, the solenoid-actuated relief valves, primary system safety valves and the emergency cooling system. The reactor primary system, in turn, includes the reactor vessel and its internals, recirculation piping, valves, pumps and all connected piping to the external containment isolation valve.

Many other systems connect directly to the RCS besides those shown on Figure V-1. These are included in separate flow diagrams as given below:

System	Figure No.
Feedwater	XI-7
Shutdown Cooling	X-1
Cleanup	X-2
Core Spray	VII-1
Liquid Poison	VII-6

1.2 Materials of Construction

Insulation throughout the RCS within the drywell consists of metal reflective insulation and blanket insulation. In the event of small coolant leakage leading to wetting of the insulation in contact with the outer surface material of the loop, no adverse electrochemical or chemical reaction leading to excessive corrosion is anticipated.

1.3 Thermal Stresses

Heatup and cooldown rates for water in the reactor system during normal operation will be limited to 100°F/hr by procedural control. Holding this limit will assure that stresses are well within Code limits as discussed in Section V-A.4.0 above.

In the event of a short-term, more rapid blowdown greater than 100°F/hr, the vessel would be held for an equivalent amount of time at constant temperature and pressure before heating or cooling is resumed at 100°F/hr. For example, the design calculations specifically considered inadvertent operation of a single bypass or solenoid-actuated relief valve leading to a 17.5°F/min blowdown for a period of 10 min to 370°F. Following this, the vessel would be held at a constant temperature of 370°F for 1 3/4 hr, then cooldown or heatup would be resumed at 100°F/hr.

F. REFERENCES

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

2. 0000-0084-3226-SRLR, Revision 0, "Supplemental Reload Licensing Report for NMP1, Reload 20, Cycle 19," December 2008.

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.

5. NRC Letter to NMPNS dated November 8, 2004, "Nine Mile Point Nuclear Station Unit Nos. 1 and 2 - Issuance of Amendments RE: Implementation of the Reactor Pressure Vessel Integrated Surveillance Program (TAC Nos. MC1758 and MC1759)."

6. NRC Letter to NMPNS dated October 27, 2003, "Nine Mile Point Nuclear Station, Unit No. 1 - Issuance of Amendment Re: Pressure-Temperature Limit Curves and Tables (TAC No. MB6687)."

TABLE V-1 (Cont'd.)

Safety Valves		
Number		9
Capacity (each)		633,000 to 651,000 lb/hr
Pressure setting (nominal)		1218 to 1254 psig
Capacity (minimum each, certified)		644,543 lb/hr at 1278 psig
Design code		ASME Sec I-1962
Emergency Cooling System		
Condensers:		
Design pressure	- shell	15 psig at 300°F
	- tubing	1250 psig at 575°F
Design code	- shell	ASME Sec VIII (nuclear cases) and ASME Sec III, Subsection ND, 1986 edition
	- tubing	ASME Sec III, Subsection NC, Class 2, 1986 edition
Number of tube bundles		4
Capacity (rated capacity of four units)		38×10^7 Btu/hr at 1135 psig and 562°F on tube side; 5 psig and 228°F on shell side
Operating time with gravity makeup		8 hr
Isolation valves in inlet line		2 normally open motor operated
Isolation valves in outlet line		1 normally closed air operated and 1 check valve
System Pressures		
Design		1250 psig at 575°F
Initial vessel hydrostatic test pressure		1875 psig
Maximum safety valve setting		1254 psig
Minimum safety valve setting		1218 psig
Solenoid-actuated relief valve settings		2 @ 1090 psig 2 @ 1095 psig 2 @ 1100 psig
Emergency cooling system pressure actuation		≥1080 psig for 12 sec
Reactor scram		≤1080 psig
Normal operating pressure		1030 psig at 550°F

* This value represents the original design estimate of lifetime neutron fluence for the reactor vessel.

TABLE V-2

OPERATING CYCLES AND TRANSIENT
ANALYSIS RESULTS**

OPERATING CYCLE	EXPECTED NO. OF CYCLES
Vessel Head Removal	50
Vessel Head Reinstallation	50
100°F/hr Heatup	240
100°F/hr Cooldown*	229
300°F/hr Emergency Cooldown	10
Blowdown	1
Scram Cycles	280
Emergency Condenser Initiation into Isolated Loop	30
Unisolation of an Isolated Loop	30
Emergency Condenser Initiation into Idle Loop	30
Shutdown Cooling Initiation into Isolated Loop	240
Inadvertent Start of Cold Loop	20
Emergency Condenser into Pumped Loop	500
Recirculation Pump Hot Loop Startup	300

* The number of 100°F/hr cooldowns was determined by subtracting the emergency cooldowns and blowdown from the number of 100°F/hr heatups.

TABLE V-2 (Cont'd.)

** This table was used in the original fatigue evaluations of the reactor vessel as detailed in Section V-A.4.0. The NMP Fatigue Monitoring Program is now used to manage the fatigue evaluations at NMP. This program uses the FatiguePro software to maintain all actual calculated cumulative usage factors below their corresponding allowables. The calculations are based on the actual number of cycles experienced by the plant for each transient and, in some cases, the actual parameters experienced by each cycle rather than the number of cycles listed in this table.

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

* Listed usage factor values are based on original fatigue evaluations of the reactor vessel as detailed in Section V-A.4.0. Values are managed and maintained below their corresponding allowables through the NMP Fatigue Monitoring Program.

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, certain requirements of ASME Sec IIIA-1965, and ASME Sec III-1986 (IV-38-13)
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

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A computer analysis was made to determine the maximum induced seismic accelerations, displacements, shears, moments and reactions acting on the RPV and its support and on the reactor building. The analysis includes the response of the RPV and its support to the design earthquake and jet reaction forces. Also included is the effect on the RPV of the displacement of the reactor building and containment vessel due to the postulated earthquake. Results of this analysis are contained on Figures VI-6 through VI-17.

Personnel access into the reactor building is controlled from the track bay extension and from the turbine building. The track bay extension has a railroad entrance and a personnel access air lock passageway from the outside.

The track bay extension consists of a 20-ft by 20-ft by 80-ft long air lock, connected to the track bay compartment by a vertical lift inner door and an airtight seal. The track bay extension is equipped with a motor-operated double swing outer door 16 ft wide by 17 ft 6 in high. The door can also be operated manually and is designed to resist an internal or external load of 40 psf. The outer door closes against a closed cell sponge neoprene closure to provide an airtight seal. The inner vertical lift door bears against a one-piece inflatable seal of reinforced ethylene propylene diene monomer around its perimeter. The entire contact area of the inflatable seal will expand approximately 3/4 in under pressure. The seal material will remain pliable and seal at temperatures of -20°F to 210°F.

Containment integrity for the track bay compartment and extension is provided by an outside double swing door, an inside vertical lift door and personnel doors connected by an airtight access passageway. The track bay compartment (with extension) and its access openings are shown on Figure III-4. Typical door seals for the personnel and equipment doors are shown on Figure VI-18.

Interior doors with air locks are provided in the south wall of the reactor building leading into the turbine room at el 261, as shown on Figure III-4, and at el 340, as shown on Figure III-8. The doors of the air lock have neoprene seals with sealing requirements equivalent to those of the railroad door. Details are shown on Figure VI-19.

Procedures and alarms are used to control access and maintain building integrity. Primary and secondary shielding is discussed in Section XII.

D. CONTAINMENT ISOLATION SYSTEM

1.0 Design Bases

Isolation valves are provided on lines penetrating the drywell and pressure suppression chamber to assure integrity of the containment when required during emergency and post-accident periods. Isolation valves which must be closed to assure containment integrity immediately after a major accident are automatically controlled by the reactor protection system (RPS) described in Section VIII.

The drywell and suppression chamber penetrations are dedicated to specific purposes as shown in Tables VI-1 and VI-2, respectively. The tables list the number, size, and type of penetration associated with each purpose.

Containment isolation valves (also called isolation valves) are defined as any valves which are relied upon to perform a containment isolation function on lines penetrating the primary reactor containment and include all reactor coolant isolation valves and all primary containment isolation valves. Test, vent and drain (TVD) valves located on the containment pressure boundary are containment isolation valves but are not included in the tables of reactor coolant isolation valves or primary containment isolation valves.

Reactor coolant isolation valves are containment isolation valves which are on lines penetrating the primary reactor containment and are connected to the RCS (or a system containing reactor coolant) and function as reactor coolant pressure boundary (RCPB) components. Reactor coolant isolation valves function as primary containment isolation valves in the event of a LOCA.

Primary containment isolation valves are containment isolation valves on lines penetrating the primary reactor containment connecting directly to the free space enclosed by the containment.

Table VI-3a is a listing of all reactor coolant isolation valves, and Table VI-3b lists primary containment isolation valves.

All lines which are part of the RCPB and penetrate the primary reactor containment are provided with redundant isolation valves. As a general rule, one of each pair of isolation valves in series is located inside the containment. The other valve is outside the containment. On the emergency cooling system supply and on the feedwater system where it was necessary to install both valves outside the containment, a guard pipe is installed between the line and the containment vessel penetration sleeve. This sleeve is welded to the body of the first isolation valve outside the containment. This, in effect, extends the containment to include the body of the first isolation valve. For the emergency cooling system supply, the two valve bodies are welded end to end for greater integrity. For the feedwater system, the two valves are separated by a 10-in extension.

Lines which are part of the reactor coolant boundary and may be required to have flow after an accident are provided with check valves. The CRD and liquid poison systems have two check valves in series. One valve is inside the containment. The feedwater system, as described above, has two valves outside the containment, one of which is a check valve.

The cleanup and shutdown cooling systems each have redundant isolation valves with one valve inside the containment. The outer valve on the return to the reactor line is a check valve. Post-accident thermal overpressurization protection is provided for the penetration piping between the isolation valves in the shutdown cooling system.

Instrument lines are provided with redundant valving outside the containment. Automatic flow check valves minimize loss of reactor coolant in the event of an instrument line break.

All external isolation valves are located as close to the containment as possible. Where guard pipes are used between the containment penetration and the line, the outer valve is welded to the guard pipe. For reactor coolant isolation valves on low-temperature lines where no guard pipe is required, the outer valve is welded directly to the penetrations sleeve.

Most lines which connect directly to the containment atmosphere and penetrate the primary reactor containment are provided with redundant isolation valves. Two normally-closed valves outside the containment are provided for systems which are not required to function under accident conditions. Lines which are not equipped with double isolation valves have been determined to be acceptable based upon the fact that the system reliability is not compromised, the system is closed outside containment, and a single active failure can be accommodated with only one isolation valve in the line.

Instrument lines connected to containment atmosphere which penetrate primary containment are provided with two isolation barriers, such as manual valves, caps, or diaphragm assemblies.

Each containment spray line which is required to be open under accident conditions contains a check valve outside the containment. These check valves are installed to minimize bypassing of pressure suppression during the initial pressure transient of the LOCA.

The oxygen sample return line and the nitrogen purge line for the traveling in-core probes use two check valves in series outside the containment. The traveling in-core probe guide tubes use a ball valve and manually-actuated explosive shear valve in series outside containment.

Each line that penetrates primary reactor containment and is neither part of the RCPB nor connected directly to the containment atmosphere, in the case of the drywell cooling and recirculation pump cooling systems, has one isolation valve. These systems circulate cooling water in a closed system into and out of the containment. Each line carrying incoming cooling water is provided with a self-actuating check valve outside the containment. Each line which carries water out of the containment has a MOV which is actuated by remote manual control.

The isolation system for each line is designed to accommodate loss of power to an isolation valve. MOVs (ac or dc) are designed to fail in the mode in which they are when loss of power occurs. Air-operated valves (AOV) fail closed upon loss of power. Different power sources for each valve in series ensure that the isolation function will not be defeated by single failure. Failure of a single power source does not prevent isolation even where a normally open MOV fails open. Isolation is effected either by having a closed piping system which does not communicate with containment atmosphere or by having a redundant separately powered valve in series with the failed valve. In the case of systems which are required to be open following an accident, valves are normally open and fail open, are normally closed but fail open, or are normally closed but fail closed (as is) but have a redundant valve path in parallel that is open and/or fails open.

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Systems which connect to the nuclear steam supply system (NSSS) and may be required to have flow after an accident are provided either with two check valves or a check and a remote manually controlled valve in series. These are the feedwater, the CRD hydraulic, and the liquid poison systems.

Instrument lines that run from the reactor primary system through the drywell are equipped with shutoff valves and a flow check valve located outside containment as indicated on Figure VI-20.

The flow check valves meet or exceed the following design requirements:

<u>Design Conditions</u>

Operating Pressure	1250 psig
Operating Temperature	575°F
Specified Flow to Close Valve	25 gpm
Horizontal Acceleration	0.20 g
Vertical Acceleration	0.10 g

A cross section of a typical 3/4-in check valve is shown on Figure VI-21. The valve poppet is held open by the spring. The force generated by the pressure differential over the seat area acts against the spring. Flow creates a pressure differential which overcomes the spring and closes the poppet. The differential pressure then acts on the poppet seating area to keep the poppet closed.

A bypass arrangement is used on these instrument lines as a means of equalizing line pressure to open the flow check valve in the event it should close and for blowdown purposes.

Instrument line leaks can be detected by one or a combination of the following:

1. Operator comparing readings with several instruments monitoring the same process variable such as reactor level, recirculation pump flow, steam flow, and steam pressure.

2. By annunciation of the control function, either high or low in the control room.

3. By a general increase in the area radiation monitor readings throughout the reactor building.

4. By audible noise either inside the turbine building or outside the reactor building.

5. By alarms on the reactor building floor drain tank.

6. By probable increase in area temperature monitor readings in the reactor building.

Routine surveillance as indicated in items 1 through 6 is felt to be a sufficient program for the periodic testing and examination of the valves in these small-diameter instrument lines. At each major refueling outage, each instrument line flow check valve will be tested for operability.

The engineered safeguards systems which may be required to operate following an accident originally had no specific isolation requirements. These systems, which consist of core and containment spray and the emergency cooling system, were designed as containment extensions and diligent efforts were made to meet the intent of Section III-1965 of the ASME Code. Valves were provided in the lines from the suppression chamber and in those into the drywell to provide system isolation for maintenance or testing. Isolation valves for these systems are shown in Tables VI-3a and VI-3b. The opening times, failure modes, and normal position of the valves in the core spray, containment spray and emergency cooling systems are based on the individual system operational requirements as discussed in Sections V and VII.

In general, the closure time of all isolation valves is such that the release of fission products to the environment is minimized. As described in Section XV, no large-scale fission product release occurs before 1 min has elapsed. The valve closure times are thus set for a 1-min maximum unless operational restrictions are more severe.

The closure times of all valves on lines in systems connecting to the NSSS are based on preventing fuel damage from overheating with no feedwater makeup following a line break in the particular system. The valve closure time for the main steam line (MSL) is based on the MSL break accident discussed in

Section XV. By keeping the valve closure time less than about 10 sec, sufficient coolant will remain in the reactor vessel to provide adequate core cooling. The valves are designed to close and to be leak-tight during the worst conditions of pressure, temperature and steam flow following a break in the MSL outside the pressure suppression system.

The codes used in the design of Class I system containment isolation valves at the time of construction included ASME Section I-1962 or ANSI B31.1-1955 and ANSI B16.5-1955, with requirements of ASME Section III-1965 for nondestructive testing (NDT). For subsequent modifications, Regulatory Guide (RG) 1.29 recommendations are followed. Piping system segments penetrating containment and considered susceptible to thermal overpressurization are analyzed in accordance with the criteria of the ASME Boiler & Pressure Vessel Code, Section III, Appendix F (1986 Edition).

The design criteria for containment isolation valves consist of normal and special loadings, load combinations, and load combination limits. Seismic design criteria are listed in Table VI-4.

1.1 Containment Spray Appendix J Water Seal Requirements

Table VI-3b lists primary containment isolation valves of the containment spray system which enter the free space of the containment. These lines have an Appendix J water seal by virtue of system operation following the design basis LOCA. The system design basis is continuous operation following the DBA as

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documented in Section XV-5.3. The Boiling Water Reactor Owners' Group (BWROG) Emergency Procedure and Severe Accident Guidelines (EPG/SAG) restrict drywell and suppression chamber spray operation. The emergency operating procedures (EOP), based on the BWROG EPG/SAG spray limitations, are intended to provide Operator guidance to prevent beyond design basis evaporative cooling conditions from developing. The evaluation, which determined the impact of the EOP assumed actions upon the licensing basis, concluded that the radiological impact of the potential leakage from the primary containment for the conditions where the water seal is secured would result in less than 20 percent of the 10CFR100 regulatory limits, and less than 65 percent of the control room regulatory limits per 10CFR50 Appendix A, General Design Criterion (GDC) 19.

The drywell spray limitations were developed to address evaporative cooling conditions which are beyond the Unit 1 design basis. Therefore, the conditions which interrupt the 10CFR50 Appendix J water seal are evaluated as beyond design basis conditions. In this respect, the maximum potential leakage assumed in this evaluation is not included as part of the design basis primary containment leakage. The leakage is only used to compare the maximum potential leakage relative to 10CFR100 and 10CFR50 Appendix A, GDC 19. In order to ensure that assumptions used in this evaluation remain valid, surveillance tests are required to monitor packing degradation and ensure minimal system cross-tie leakage (see Section VI-F.1.2 and VII-B.4.0).

Post-accident secondary containment conditions are defined based on the integrity of the containment spray system pressure boundary and the containment isolation check valves. The secondary containment conditions are defined based on total leakage of 1.5 percent per day as defined in Section VI-F. This is based on the integrated leak rate testing (ILRT) discussed in Section VI-F.1.2. Therefore, post-accident equipment qualification conditions or post-accident vital area access is not affected by the potential leakage used to evaluate the beyond design basis EOP conditions which terminate the containment sprays.

2.0 System Design

A list of all isolation valves on lines penetrating the containment vessels and their pertinent modes and characteristics is given in Table VI-3. Figure VI-22 shows all valves, except those on instrumentation lines.

service water-cooled heat exchanger units are designed to maintain temperatures at 85°F maximum and 70°F minimum in accessible areas and 100°F maximum and 50°F minimum in inaccessible areas.

Both the main supply and exhaust ducts are equipped with two leak-tight isolation valves in series, which close automatically upon detection of high radiation levels within the building. The supply and exhaust fans trip immediately. The closure sequence of the normal supply and exhaust isolation valves ensures that reactor building negative pressure is maintained during the transition from normal to emergency ventilation for events which are not accompanied by a loss of offsite power (LOOP) (see Section XV for LOCA/LOOP discussion). They also may be controlled manually from the main control room. The inlet and outlet duct penetrations through the building walls are sealed against leakage. A steel pipe sleeve is integrally cast in the concrete, and the outer end of the sleeve has a gasketed flange which connects to the first isolation valve.

F. TEST AND INSPECTIONS

A program of testing the primary containment system has been developed based on Appendix J of 10CFR50, "Reactor Containment Leakage Testing for Water Cooled Power Reactors." The program includes overall ILRTs, local leakage rate tests and isolation valve leakage tests.

1.0 Drywell and Suppression Chamber

1.1 Preoperational Testing

Following construction of the drywell and suppression chamber, each was pressure tested at 1.15 times its design pressure. Penetrations were sealed with welded end caps as were the downcomers from the drywell to the suppression chamber. The relief lines from the suppression chamber to the drywell were also blanked. Following the strength test, the drywell and suppression chamber were tested for leakage rate at design pressure; each met the criterion for leakage at this stage of construction of less than 0.1 percent per day at design pressure. The suppression chamber was also tested while half filled with water to simulate operating conditions.

After complete installation of all penetrations, ILRTs of the drywell, suppression chamber, and associated penetrations were conducted. The tests were conducted at several test pressures up to and including 35 psig to establish a leakage rate curve. The necessary instrumentation was installed in the containment systems to provide the data to calculate the leakage rate. Table VI-5 summarizes the initial preoperational tests conducted.

1.2 Postoperational Testing

An integrated leakage rate Type A test is performed to demonstrate that leakage through the primary containment and systems and components penetrating primary containment does not exceed the allowable leakage rate specified in the Technical Specifications. The integrated leakage Type A test shall include the containment spray system piping in its operating mode Technical Specification configuration. This is based on the analysis discussed in Section VI-D.1.1 which takes credit for the ILRT as confirmation of containment spray system integrity (i.e., minor components are leak-tight, cross-tie leakage is minimal).

The integrated leakage test is conducted at the analyzed maximum accident pressure, Pa. The test pressure, as required by 10CFR50 Appendix J, is based on the design basis LOCA conditions. The peak primary containment pressure following a LOCA would be 35 psig.

The Appendix J to 10CFR50 acceptance criteria states that the maximum allowable leakage rate shall not exceed 1.5 weight percent of the contained air in 24 hr at 35 psig. The allowable

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.

(9) Reactor coolant isolation valves function as primary containment isolation valves in the event of a LOCA.

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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:

Criterion 16

Containment Design 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 1.5 percent per day at 35 psig. The analysis demonstrates that the offsite doses from this accident would be well within the limits of 10CFR50.67.

Criterion 38

Containment Heat Removal A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that, for onsite electric power system operation (assuming offsite power is not available), and for offsite electric power system operation (assuming onsite power is not available), the system safety function can be accomplished.

Two CSS system loops are provided to remove heat, reduce pressure, and restore the pressure suppression system temperature following a LOCA. Each loop is capable of removing all the decay heat and, in addition, the energy from any credible metal-water reaction at a rate that will prevent containment pressures and temperatures from exceeding their design values.

The power for the pumps is provided from redundant Station reserve power supply systems or from one of two emergency diesel generators. One of the two spray loops is automatically actuated on the combined condition of high drywell pressure and low-low reactor water level. The other loop can be manually controlled from the main control room.

Criterion 39

Inspection of Containment Heat Removal System The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

Essential CSS system components are inspected periodically to ensure the integrity and capability of the system. The system tests and inspections are described in Section VII-B-6.0 and in the Technical Specifications.

Criterion 40

Testing of Containment Heat Removal System The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure: 1) the structural and leak-tight integrity of its components, 2) the operability and performance of the active components of the system, and 3) the operability of the system as a whole, and under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

The CSS system is designed to permit appropriate periodic pressure and functional testing. Pumps are periodically tested for flow, developed pressure and automatic initiation. Containment spray injection valves are normally open and are not required to operate. The testing program demonstrates, under simulated conditions, that pump sets can be relied upon to function as they are designed to operate under accident conditions.

Periodic spraying of water into the containment is not practical. Therefore, water is recycled back to the suppression pool during tests. Air tests are used to ensure flow through the header and nozzles.

Testing of emergency power sources for containment cooling is periodically performed. The power systems are tested for automatic pickup of load required for the LOCA.

Criterion 44

Cooling Water A system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink (UHS) shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available), and for offsite electric power system operation (assuming onsite power is not available), the system safety function can be accomplished assuming a single failure.

Heat removal from containment following a LOCA, and transferring that energy to the UHS, is achieved and assured through the use of redundant pump trains drawing suction on the suppression pool and removing heat through a heat exchanger supplied by the raw water pumps. The system is designed and suitably sized to maintain the torus below the NPSH temperature limits of core spray and containment spray.

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.

5. Functional Requirement - Remove airborne fission products from the drywell atmosphere following a LOCA which results in significant fuel damage, to limit fission product releases from containment leakage paths.

   Basis - The LOCA radiological analysis implementing the alternative source term (AST) methodology described in Regulatory Guide (RG) 1.183 credits airborne fission product removal by the CSS system. The AST analysis is described in Section XV.

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 ≥34.2 ft for the most restrictive case (least NPSH margin) in which two pumps are operating through separate strainer assemblies at a flow rate of 3759 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, reduce drywell airborne fission product concentrations, and control the pressure suppression chamber water temperature following a design basis LOCA. The system is designed to provide heat removal capabilities for vessel leaks up to and including the DBA, the double-ended break of a reactor recirculation line, without core spray system operation.

3.2 System Design Description

As shown on Figure VII-3, the CSS system is designed with two redundant loops. The primary loop (Loop 11) provides water to the primary or inner drywell sparger and to the torus sparger. The secondary loop (Loop 12) provides water to the secondary or outer drywell sparger and to the torus sparger. The torus sparger is common to both loops. Each of the two loops are cross-connected through the test return lines such that each of the loops can provide flow to both the primary and secondary spargers. Each loop includes two redundant trains and consists of two suction headers, two containment spray pumps, two heat exchangers and the associated containment spray raw water pumps, a common test return line, and associated piping and control valves. All pumps in a loop are powered from the same emergency power bus. Each loop is electrically independent from the other loop.

The CSS system is normally in standby. Containment spray pump operation is automatically initiated by two RPS signals--high drywell pressure and low-low reactor water level. Automatic initiation of the containment spray pumps occurs following the core spray pumps and core spray topping pumps initiation. Upon receipt of an actuating signal, the four containment spray pumps are sequentially started when powered from either the reserve Station service or the diesel generators. Upon containment spray pump initiation, self-actuating check valves open to allow containment spray water to flow through the system. The containment spray raw water pumps must be manually initiated following automatic initiation of the containment spray pumps. A 15-min delay can be tolerated in starting a raw water pump since it provides lake water to a containment spray heat exchanger for the purpose of long-term cooling of the torus water.

Each pump takes suction from the torus through individual suction lines. The water in each suction line flows from the torus through a suction strainer assembly. Two strainers comprise each of two suction strainer assemblies. When two pumps, either 112 and 122, or 111 and 121, are operated, they will take suction from the same suction strainer assembly. The discharge from each pump passes through the shell side of a heat exchanger where it is cooled prior to being distributed to the drywell and torus spray headers. The spraying of the water in the containment increases the heat removal rate, thereby decreasing containment temperature and pressure. The spray headers inside the drywell and torus are arranged to distribute water as uniformly as possible throughout the free volume. The direction of spray from the nozzles is arranged to minimize impact on equipment and allow as much free-fall as possible to maximize steam condensation. In addition, flow from the containment spray pump discharge can be directed to the torus via a 6-in test return line that provides suppression pool cooling.

Each of the containment spray heat exchangers is supplied cooling water from a dedicated containment spray raw water pump. Each containment spray raw water pump takes suction from the condenser circulating water intake tunnel. The pump discharge passes through a duplex strainer prior to entering the tube side of the containment spray heat exchanger. After passing through the heat exchanger and cooling the suppression pool water, the raw water is released to the discharge manifold.

In the event of a total loss of the containment spray primary water source (suppression chamber water below the containment spray pump suction level), raw water pumps 112 and 121 can be aligned to supply the containment spray spargers to provide an alternate source of containment cooling. Likewise, raw water pumps 111 and 122 can be aligned to supply the core spray system.

3.3 System Design

The CSS system was originally designed to operate with Loop 11 and Loop 12 flow paths in the drywell as totally independent redundant systems. However, in order to satisfy 10CFR50 Appendix J, paragraph III.C.3(b) requirements, the current standby configuration of the system provides flow to both primary and secondary spargers, with two pumps including either train 111 or 122 in operation, to form a water seal. This is accomplished by cross-connecting the two trains via the test

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decay heat and chemical energy from a 70-percent metal-water reaction. With a maximum possible reaction of 27 percent, the analysis shows that more than sufficient heat removal capacity exists in the system. This analysis requires the CSS system to satisfy the analysis input assumptions discussed in Section XV-C-5.3.2.

To determine proper distribution of containment spray through the nozzles, testing was performed on a sample spray nozzle of the size and type used in containment spray. Water was run through the nozzle at various pressures from 10 psig to 100 psig, and spray pattern and spray particle fineness was observed. Pressure drops of 80 psig and 30 psig represent the original system configuration pressure conditions for two-pump operation and one-pump operation, respectively. The particle sizes for the two-pump operation are in the range of 10 to 400 microns. For one-pump operation, particle sizes range from 500 to 1000 microns.

The CSS system design flow, spray distribution/droplet size, and fall heights were used to determine the airborne fission product removal rate for implementation of the AST methodology described in Section XV.

4.2 System Response

After an initiation signal is received, there is a time delay of 20 sec to allow the core spray and core spray topping pumps to start. At the 25-sec mark, containment spray pumps 111 and 121 will receive a start signal, and at 30 sec, containment spray pumps 112 and 122 will receive their start signal. If the core spray and core spray topping pumps do not start, a set of backup timer contacts will start the containment spray start sequence in 50 sec to allow the core spray starting logic to be initiated a second time. This will cause pumps 111 and 121 to start at 55 sec, and pumps 112 and 122 to start in 60 sec. This interlock, delaying the starting of the containment spray pumps, is provided to avoid overloading of the diesel generators.

4.3 Interdependency With Other Engineered Safeguards Systems

The CSS system is used in conjunction with the core spray system described in Section VII-A. The core spray system removes heat from the core in the event of a LOCA. In the heat removal process, the core spray water is converted to steam, which is then released to the containment. The containment sprays condense the steam in the drywell and remove heat from the containment vessels through heat exchangers.

The raw water pumps are interconnected with the core spray system and the containment spray loops to provide an emergency source of water. Raw water pump 112 can supply water to containment spray train 122, and raw water pump 121 can supply water to containment spray train 111. The motor-operated valves between raw water and containment spray water are interlocked with the heat exchanger raw water discharge valves. If one valve is open, the other must be closed. In addition, raw water pump 111 is connected to core spray pump train 11 and raw water pump 122 is connected to core spray pump train 12. The air-operated valves located on the connection between the two systems are also interlocked with the raw water discharge valves.

The following systems must be in operation to support the CSS system:

- Instrument air must be operational to permit operation of the containment spray inlet isolation valves and bypass blocking valves.

- 4.16-kV and 600-V ac power distribution systems are required to provide power to the containment spray pumps, raw water pumps, and isolation valves.

- The RPS system is required to provide automatic initiation signals to the containment spray pumps and waste disposal isolation valves.

- The process radiation monitoring system must be operational to alert Operators of leakage of contamination into the raw water system due to heat exchanger leaks.

5.0 System Operation

5.1 Limiting Conditions for Operation

The limiting conditions for operation (LCO) pertaining to the CSS system are listed in Section 3.3.7 of the Unit 1 Technical Specifications. Other LCOs associated with generic equipment and programs are also applicable and are listed in other sections. The intent of the LCOs is to ensure that both loops of the system are operable when fuel is in the vessel and the reactor coolant temperature is greater than 215°F. One containment spray loop will provide the required containment cooling, airborne fission product removal, and pressure reduction for the DBA. However, to provide sufficient redundancy to satisfy the single failure criterion, both loops of the CSS system are required to be operable.

If a redundant component in one loop of containment spray or its associated raw water loop becomes inoperable, operation may continue provided the component is returned to an operable condition within 15 days. If a redundant component in both containment spray loops or their associated raw water loops becomes inoperable, operation may continue provided the component is returned to service within 7 days. In both cases, additional surveillance requirements are imposed. If a containment spray loop or its associated raw water loop becomes inoperable and all the components of the other loop are operable, the reactor may remain in operation for a period not to exceed 7 days.

If the LCOs are not met, then a normal orderly shutdown shall be initiated within 1 hr and the reactor shall be placed in cold shutdown within 10 hr.

6.0 Tests and Inspection

To ensure that the performance of the CSS system continues to meet the design requirements, the following surveillance tests and inservice inspections requirements must be satisfied.

- ASME Section XI inservice examination of components
- ASME OM Code inservice testing of pumps and valves
- ASME Section XI system pressure tests
- Appendix J leak rate testing
- System operability surveillance tests

Several programs have been established to meet the requirements of the ASME Code and Appendix J. These include: 1) NMP1 ISI Program Plan, 2) Inservice Pressure Testing Program Plan, 3) Pump and Valve Inservice Testing Program Plan, and 4) Appendix J Testing Program Plan.

The following CSS system tests, inspections, and surveillances are conducted to meet the requirements.

- Containment Spray System Quarterly Operability Test - verifies valve, pump and total system operability and verifies operation of valve limit switches and solenoid-operated valves

- Containment Spray Header and Nozzle Air Flow Test - verifies header, header check valve, and nozzle operability

- Containment Spray System Suction Valve Operability Test - verifies valve operability

- Containment Spray Valve Remote Position Indicator Verification - verifies operability of indicators

- Containment Spray Pressure Test - verifies integrity of the system by VT-2 visual examination

- Containment Spray Raw Water Pressure Test - verifies integrity of the system by VT-2 visual examination

- Containment Spray Raw Water System Intertie Valve Operability Test - verifies the operability of the containment spray/core spray intertie check valves

Testing of the initiating instrumentation and controls portion of the system is discussed in Section VIII. The emergency power system, which supplies electrical power to containment spray in the event that offsite power is unavailable, is tested as described in Section IX. Visual inspections of all system components located outside the drywell can be made at any time during power operation. Components inside the drywell can be visually inspected only during periods of access to the drywell.

C. LIQUID POISON INJECTION SYSTEM

1.0 Design Bases

The liquid poison injection system is provided to bring the reactor to a cold shutdown condition at any time in core life independent of the control rod system capabilities. Cycle-specific analysis results are contained in the SRLR[7]. The primary requirement imposed on the liquid poison injection system is to shut down the reactor from a full-power operating condition, assuming complete failure of the withdrawn control rods to respond to an insertion signal. Injection of liquid poison is also required following a large break LOCA to maintain the suppression pool water pH $\geq$ 7.0 in support of the AST methodology.

For the design rating of 1850 MWt, a concentration of 109.8 ppm of boron-10 isotope (equivalent to 600 ppm of natural boron) is required in the reactor to meet the reactor shutdown requirement. However, an additional 25-percent margin is included in the calculation of required liquid poison tank concentrations to allow for nonuniform mixing of the liquid poison as it is injected into the reactor. The same tank concentration level has been determined to adequately satisfy the AST support function for controlling pH above 7.0.

The rate of reactivity compensation provided by the liquid poison injection system is designed to exceed the rate of reactivity gain associated with reactor cooldown from the full-power condition. The liquid poison system is not intended to be capable of producing as rapid a shutdown as is produced by scramming the control rods, and should not be construed as a scram backup. Following a large break LOCA, initiation of the liquid poison system within 1.5 hr after the potential for significant fuel failure has been identified will ensure that the suppression pool pH is controlled for at least 30 days. The liquid poison injection system is actuated only by remote manual action from the control room, hence a deliberate action.

The liquid poison injection system can be powered from the diesel generators and, therefore, will be operable in the event of a loss of normal and reserve ac power. The liquid poison system is required to function for a maximum of 3 hr following pipe break events (accidents) that produce harsh environmental conditions.

Accordingly, EQ in accordance with 10CFR50.49 for the 3-hr post-LOCA mission time has been demonstrated for the electrical components important to safety that comprise the liquid poison system. EQ in accordance with 10CFR50.49 is not required for anticipated transients without scram (ATWS) that may produce harsh environmental conditions inside containment, but not in the reactor building where the electrical components are located.

All portions of the system are designed for earthquake loads of 0.3g horizontal and 0.1g vertical.

2.0 System Design

The liquid poison injection system, shown on Figure VII-6, consists of an ambient pressure tank with immersion heater for low-temperature sodium pentaborate solution storage, two high-pressure positive displacement pumps for injecting the solution into the reactor core, two explosive-actuated shear plug valves for isolating the liquid poison from the reactor until required, an in-vessel sparger ring, a test tank, two isolation check valves, a buffer system and additional valves, piping and associated instrumentation.

The liquid poison is stored in a 4080-gal tank which is designed for atmospheric pressure. This tank is complete with top cover, hatch with lid for adding chemicals, immersion-type electric heater, instrument connections, and nozzles for outlet, recirculation, overflow, air sparger and drain. The tank outlet nozzle is outfitted with a strainer, which extends above the tank bottom, to prevent solid particles from being discharged to the pump suction. The air sparger, which is used for mixing the solution for each initial batch, has air holes directed toward the bottom of the tank for sweeping the deposit there. The top cover and hatch lid are designed so that the solution, when agitated by the air sparger, will not spill over.

The neutron absorber in the sodium pentaborate liquid poison solution is the boron-10 isotope. The relationship between liquid poison solution concentration and boron-10 enrichment is contained in the equivalency equation. The equation is:

$$\frac{C}{13\%\,wt} \; X \; \frac{628300}{M} \; X \; \frac{Q}{86\,gpm} \; X \; \frac{E}{19.8\%\,Atom} \geq 1$$

Where:

C = Sodium pentaborate solution concentration (wt %)

M = Mass of water in reactor vessel and recirculation piping at hot rated conditions (501500 lb)

Q = Liquid poison pump flow rate (30 gpm nominal)

E = Boron-10 enrichment (Atom %)

The saturation temperature varies with solution concentrations of sodium pentaborate as shown on Figure VII-7. This saturation curve has 5°F margin above the actual saturation temperature to prevent precipitation of sodium pentaborate while in storage. The liquid poison tank contains a minimum volume of 1325 gal of sodium pentaborate solution whose (boron-10) enrichment and concentration conform to the equivalency equation.

To compensate for evaporation which could lead to precipitation, the storage tank was oversized. The nominal tank capacity of 4080 gal allows additional water to be added to the solution as a safety margin against evaporation losses.

Temperature and liquid level alarms for the storage tank are annunciated in the control room.

The 50-kW, 550-V three-phase immersion heater is automatically controlled by a temperature indicator controller. High- and low-temperature annunciators are provided to assure that the solution is above saturation temperature. Pump test results indicate adequate NPSH is available at solution temperatures up to and including 105°F. Solution temperatures up to 130°F have been analyzed and also provide adequate NPSH. To increase the rate of sodium pentaborate solution in water, a manual override on the temperature controller permits heater operation for 150°F solution temperature. This manual override may render the system inoperable. An indicator lamp is provided to denote when the heater element is shorted to the solution. Should the immersion heater fail during Station operation, no action need be taken. Normally, the building heating system will maintain the required tank temperature. The immersion heater is used to supply the endothermic heat required during solution mixing and only incidentally to maintain solution temperature. If a failure of the building heating system occurs simultaneously with a failure of the immersion heater, the ambient temperature in the liquid poison system area will decrease very slowly due to its large mass and its interior building location. Therefore, there will be ample time to provide temporary heating.

The sodium pentaborate solution is delivered to the reactor by one of two 30-gpm, positive displacement pumps, with a design discharge pressure of 1670 psig. The pumps and piping are protected from overpressure by two relief valves which discharge back to the poison storage tank. The relief valves are set to open at a pressure between 1455 and 1545 psig.

The injection pumps produce a flow rate sufficient to meet the injection requirements for all conditions of reactor operation up to the primary system design pressure of 1250 psig. Two pumps are provided to give complete redundancy. The pumps are specifically designed for standby service to be operated infrequently, only during emergencies and testing. Each operation is for 3 hr maximum.

Since the liquid poison injection system is to be operable in the event of loss of normal and reserve ac power, one pump is connected to PB 102 and the other to PB 103. These boards are powered from the diesel generators in the event of failure of their normal supply as described in Section IX, Electrical Systems.

A radiant heat shield is installed between the two liquid poison pumps to prevent fire damage to the redundant pump in the event of a fire in either pump.

The explosive valves are double squib-actuated shear plug valves. A low-current electrical monitoring system gives visible (pilot light) and analog (ammeter) indication of circuit continuity through both firing squibs in each valve. Operation of one valve provides sufficient flow passage to meet the required flow rate. Two valves are provided to give complete redundancy.

The firing reliability of these explosive valves is in excess of 99.99 percent. The approximate firing current is 2 amps and the operating time at 2 amps is a nominal 0.002 sec. The products of the explosion are completely contained.

The buffer system is composed of gas-charged diaphragm accumulators of the capacity required to absorb fluid pulsation initiated by the positive displacement pumps. Each is located as close as possible to its respective pump discharge. Each pump loop has an accumulator.

Containment isolation is provided by two check valves in the liquid poison pipe, one check valve just outside the drywell penetration and the other check valve inside the drywell.

An additional check valve is installed downstream of each relief valve connection. The purpose of each check valve is to prevent flow through an assumed defective relief valve of the idle pump loop while the second loop is in operation. This ensures that the capacity of the second pump remains unaffected.

The liquid poison sparger in the reactor pressure vessel (RPV) is a 1-in stainless steel pipe which is fastened to the inside of the vessel shroud below the core support plate. This 360-deg sparger has ten 1/4-in drilled holes which are distributed equally around the sparger and which spray toward the bottom of the vessel. The liquid poison is thereby mixed with the reactor recirculating water as it enters the reactor fuel assemblies, assuring as uniform a mixture of poison as practical. During injection following a LOCA, the solution is mixed in the vessel bottom head with core spray water flowing through the reactor core and out the break.

A test tank and demineralized water supply are an integral part of the system to facilitate system testing and flushing. All piping in the system has been designed in accordance with ASA B31.1-1955 Piping Code. Tanks are constructed in accordance with API 650. The pressure-bearing parts of the pumps are built in accordance with ASME Code Section III, Class C-1965.

Actuation of the liquid poison system is manually initiated from the control room, assuring that poison injection is caused by a deliberate act.

2.1 Operator Assessment

The Operator can assess operation of the liquid poison system by means of pressure indication and pump motor ammeters on the main control room panels. Each explosive valve has a low-current electrical monitoring system with an ammeter and lights on the main control room panel. The ammeter and lights provide indication of circuit continuity through both firing squibs in each valve; the ammeter and lights ensure firing readiness. When fired, the circuit is broken, the ammeter reads 0, the indicating lights go out, and the control room annunciator alarms.

The pressure transmitter is in an accessible location and is provided with suitable valving so that it can be tested at any time.

The liquid poison tank is provided with control room level indication in addition to alarms for high- and low-level and high- and low-solution temperature.

3.0 Design Evaluation

The liquid poison system is designed to provide the capability to bring the reactor from full design rating (1850 MWt) to a cold, xenon-free shutdown condition assuming none of the control rods can be inserted, and to buffer the suppression pool water following a large break LOCA. To meet the shutdown objective, the system is designed to inject a quantity of boron which produces a concentration of at least 109.8 ppm of boron-10 isotope in the reactor core. This concentration will bring the reactor from full design rating (1850 MWt) to a subcritical condition considering the combined effects of the control rods, coolant voids, temperature change, fuel doppler, xenon, and samarium. The same quantity of boron will maintain the suppression pool pH $\geq$ 7.0 for 30 days following a LOCA that results in significant fuel damage. Cycle-specific analysis results are contained in the SRLR[7].

The minimum required tank storage volume and conformance of the liquid poison solution concentration and boron-10 isotope enrichment assures that the expected liquid poison solution will provide the required 109.8 ppm of boron-10 isotope to the reactor core or the necessary buffering solution to the suppression pool.

The liquid poison storage tank volume concentration requirements assure that the above requirements for boron solution insertion are met with one 30-gpm liquid poison pump. Normal level is maintained between 1400 and 1700 gal. The quantity of boron-10 isotope required to be stored in solution includes an additional 25 percent margin beyond the amount needed to shut down the reactor to allow for any unexpected non-uniform mixing. The minimum tank volume requirements include consideration for 197 gal of solution which is contained below the point where the pump takes suction from the tank and, therefore, cannot be inserted into the reactor.

The solution saturation temperature varies with the concentration of sodium pentaborate. Solution temperature is maintained by Technical Specification at least 5°F above saturation temperature to guard against precipitation. Temperature and liquid level alarms for the system are annunciated in the control room.

Equipment and piping are designed to withstand the most severe conditions of loads including the design earthquake. Nozzles leading into the reactor vessel have been designed taking into account possible vessel movement due to an earthquake.

Availability of emergency diesel generator power to both of the injection pumps assures operability of the system if required during a loss of normal and reserve ac power.

4.0 Tests and Inspections

The system has been designed to permit periodic testing, maintenance, and operation of the injection pumps and appropriate valves. The pumps and valves will be tested periodically to ensure operability. Monthly pump tests are performed during Station operation either with demineralized water recirculated to the test tank, or with the solution recirculated to the poison tank. The isolation valves may be tested only during shutdown. For explosive valve tests, the valves are dismantled and inspected. The charges are removed and replaced with new charges periodically and the old charges are test fired to establish a rational charge replacement frequency.

A demineralized water purge system is provided so that the remaining portion of the system may be tested by pumping demineralized water through the distribution system and into the reactor vessel once each operating cycle.

Boron concentration and boron-10 enrichment of the solution will be periodically determined by analysis. The temperature of the solution will be monitored and annunciated in the control room to assure that the solution is above its saturation temperature. A continuity check of the firing circuit on the explosive valves is provided by pilot lights in the control room.

The functional test and other surveillance of components, along with the monitoring instrumentation, gives a high reliability for liquid poison injection system operability.

5.0 Alternate Boron Injection

In the event the liquid poison system is not available, the EOPs list alternate methods for injecting boron into the reactor. One method, referred to as the alternate boron injection system, provides for a portable pneumatic hydro pump connection between the liquid poison tank overflow/drain line and the liquid poison injection line drain valves, as shown on Figure VII-6. Borated water is then suctioned from the liquid poison tank through the hydro pump and discharged into the existing liquid poison injection line.

The air supply required for the pneumatic hydro pump can be provided from a 1-in connection to the house service air, or from the instrument air system if house air is not available.

The portable hydro pump has a design flow rate of 7.5 gpm at 1460 psig. The design pressure and flow rate of the hydro pump are sufficient to provide flow to the vessel under a worst-case vessel pressure of 1339 psig using an enriched boron solution.

The hoses (suction, discharge and air hose) and the portable hydro pump for the alternate boron injection system are stored in the vicinity of the 55-gal drum in the reactor building. The hoses are in a locked compartment to assure their availability.

The alternate boron injection system is nonsafety related and nonseismic, as the additional hoses, pump, valves, and hose connections do not perform a safety-related function and are downstream of the safety-related portions of the liquid poison system.

The other alternate boron injection method, using the reactor cleanup system, provides for filling the cleanup filter with a boron solution and injecting the solution into the vessel by placing the filter in service.

Neither alternate system is expected to be available for injection following a LOCA, since harsh environments in the reactor building will prevent the required operator access.

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D. CONTROL ROD VELOCITY LIMITER

1.0 Design Bases

The control rod velocity limiter, in conjunction with the rod worth minimizer (RWM), is provided to limit any accidental reactivity addition to rates for which the resulting excursion would not rupture the pressure vessel or impair operation of any safeguards equipment. The worst reactivity addition occurs during the control rod drop accident (CRDA) (Section XV), the consequences of which are reactivity rate dependent.

The control rod velocity limiter is an engineered safeguard that was originally designed to limit the free-fall drop velocity of the control rod to 5 ft/sec or less and, thus, limit the rate of reactivity addition. Subsequent testing and analysis demonstrated a maximum rod drop velocity of 3.11 ft/sec for use in CRDA analyses[6]. The CRDA can only happen in the event of simultaneous procedural violations and equipment malfunctions, a separation or mechanical failure in the drive line, sticking or binding of the control rod, the withdrawal of the detached control rod drive (CRD) mechanism, and then the release of the control rod by some unspecified means. The rod velocity limiter is designed to limit the consequences of the drop of the maximum worth control rod without significantly hindering the normal function of the system.

The most probable threshold for potential mechanical damage to the reactor core or other primary cooling system components is a peak fuel enthalpy in excess of 425 cal/g. By reducing the velocity of a free-falling rod, and assuring that excessive rod worth patterns are not established, the CRDA will result in peak fuel enthalpy values below the design limit of 280 cal/g.[1]

2.0 System Design

The control rod velocity limiter is an integral part of the bottom of each control rod, as shown on Figure VII-9 (typical). It is designed as a large clearance piston which travels in the control rod guide tube over the entire stroke.

The original velocity limiter assembly consists of two conical elements machined from a single 304 stainless steel casting. The lower conical element is at a 15-deg angle relative to the upper conical element, and the two elements are separated with four spacers 90 deg apart. There are no moving parts in the velocity limiter.

The rod velocity limiter provides a streamlined profile in the scram (upward) direction and a nonstreamlined profile in the dropout (downward) direction. It may be regarded as a nozzle-type limiter since, during its downward motion, a high percentage of the total water directly below the limiter flows up through the center of the limiter body and is ejected radially condenser hotwell, condensate will be transferred from the CSTs to the hotwell for makeup.

The FWS system pumps operate on 4160 V. When the plant is in operation, the power is supplied from the main generator through the Station service transformer when the generator is on-line and connected to the grid. When the main generator is off-line, the feedwater pumps are supplied with normal offsite power from the 115 kV system through the reserve transformers. If a HPCI initiation signal should occur, all HPCI/FWS system pumps would start immediately with two feedwater pump trains available for HPCI injection using the single-element feedwater control system for reactor vessel level control. If a major power disturbance were to occur that resulted in loss of the 115-kV power supply to the Nine Mile Point 115-kV bus, power would be restored from a generator located at the Bennetts Bridge Hydro Station. This generator would have the capacity of supplying approximately 6,000 kVA which is sufficient to operate one train of HPCI/FWS system pumps. If HPCI initiation were to occur, the preferred feedwater train pumps (feedwater pump 12, feedwater booster pump 13, condensate pump 13) would start. The nonpreferred train pumps would be electrically locked out on a LOOP and not start until the Operator manually reset the lockout by placing the backup pump control switch in the trip or close position. If a preferred pump train pump control switch had been manually locked out prior to the LOOP, it would remain locked out and the nonpreferred train backup pump would automatically start on HPCI initiation. If both the preferred and backup pumps are running, the preferred pump would remain in service and the backup pump will trip. The use of a Bennetts Bridge hydro generator, while not equivalent to an onsite emergency power source, provides a highly reliable alternate offsite power supply for the HPCI function of the FWS system.

4.0 Tests and Inspections

Tests and inspections of the various components are described in Section XI - Steam-to-Power Conversion.

J. REFERENCES

1. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-16, October 2007.

2. R. D. Ackley, R. E. Adams, and W. E. Browning, Jr., "Removal of Radioactive Methyl Iodide from Steam Air Systems," ORNL-4040, January 1967.

3. "Connecticut Yankee Charcoal Filter Tests," CYAP-101, December 1966.

4. R. D. Ackley et al, op cit.

5. Refer to (4), p VII-49.

6. General Electric Report NEDO-10527, "Rod Drop Accident Analysis for Large Boiling Water Reactors," March 1972.

7. 0000-0084-3226-SRLR, Revision 0, "Supplemental Reload Licensing Report for NMP1, Reload 20, Cycle 19," December 2008.

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 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.

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
19. Reactor Coolant System Radioactivity Concentration (Note 6)													
20. Analysis of Primary Coolant (Gamma Spectrum) (Note 7)													
21. Primary Containment Area High Range Radiation Level							X				X		
22. Containment Effluent Radioactivity; Noble Gases (Note 8)													
23. Radiation Exposure Rate (areas adjacent to primary containment) (Note 9)							X						X
24. Effluent Radioactivity; Noble Gases (from areas adjacent to primary containment) (Note 8)													
25. Feedwater Flow Rate									X				
26. Condensate Storage Tank Water Level										X			
27. Suppression Chamber (Torus) Spray Flow Rate, and Valve Position (Note 10)									X				
28. Drywell Spray Flow Rate, and Valve Position (Note 10)													
29. Main Steam Line Isolation Valve Leakage Control System Pressure (Note 11)													
30. Primary System Safety/Relief Valve Position (Note 27)										X			
31. Isolation Condenser Shell Side Water Level									X				
32. Isolation Condenser System Valve Position (Principal Flow Path) (Note 12)													
33. Reactor Core Isolation Cooling System Flow (Injection to RPV) (Note 11)													

TABLE VIII-3 (Cont'd.)

NOTES (Cont'd.)

in the Table), storage tank liquid level (Item 38 in the Table), neutron flux level (non-LOCA events) (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 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 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.

TABLE VIII-3 (Cont'd.)

NOTES (Cont'd.)

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.

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.

TABLE VIII-3 (Cont'd.)

NOTES (Cont'd.)

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 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

TABLE VIII-3 (Cont'd.)

NOTES (Cont'd.)

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]

25. A hydrogen monitoring system capable of diagnosing beyond-design-basis accidents will be maintained in accordance with License Amendment No. 191 (issued by NRC letter dated October 2, 2006[38]).

26. An oxygen monitoring system capable of verifying the status of the inerted containment (post-accident monitoring function) will be maintained in accordance with License Amendment No. 191 (issued by NRC letter dated October 2, 2006[38]).

27. The acoustic position indication system for the primary system safety valves and relief valves has been downgraded from RG 1.97 Category 2 to Category 3 in accordance with BWROG LTR NEDO-33160-A, Rev. 1, "Regulatory Relaxation for the Post Accident SRV Position Indication System."

		Scram	Trip Recirculation Pumps Out	Trip Core Spray On	Open Isolation Condenser Valves	Open Steam Relief Valves	Open Bypass Valves	Start Containment Spray	Inject Liquid Poison Solution	Close Bypass Valves	De-energize Fuel Hoists	Prevent Rod Withdrawal	Isolate Offgas System	Close Steam Line Isolation Valves	Close Turbine Stop Valves	Close Emergency Cooling Valves	Isolate Cleanup System	Isolate Shutdown System (b)	Isolate Drywell Pressure Maint Sys & Sump & Vent to EGTS	Isolate Building and Start Emergency Vent System	Start Diesel	Isolate Mechanical Vacuum Pump	High-Pressure Coolant Injection	Isolate Vent and Purge Valves	Initiate Control Room Emergency Ventilation
26.	Main Steam Line Isolation Valves Partially Closed	(f)													(f)										
27.	Loss Of Power To Auxiliary Bus Or Startup Transformer																				(e)				
28.	Loss Of Power To Protection System Motor Generator Set	(g)													X										
29.	Rod Worth Minimizer Prohibitive											*													
30.	High Flux - Varied With Recirculation Flow											*													
31.	Turbine Trips	(u)													X								X		
32.	Neutron Monitors Off Normal											*													
33.	Liquid Poison Initiation																X								
34.	Scram (Automatic Only)														X										
35.	High Steam Flow In Main Steam Line													X											X
36.	High Temperature In Main Steam Line Tunnel													X											X
37.	Anticipated Transients Without Scram	X	(bb)																						
38.	High Radiation At Refueling Platform																			X					
39.	High Steam Flow on Cond. Tube Side															X									
40.	High Temp. Heat Exchanger Effluent - Cleanup System																X								
41.	High Pressure At Cleanup Sys. Filters																(a)								
42.	Low Flow Cleanup Pump Suction																(a)								
43.	High Radiation At Stack Monitor																							X	
44.	High Radiation Control Room Ventilation																								(a)

KEY:

(a) After Time Delay.
(b) A Backup To The Procedures Which Require These Valves To Be Closed At All Times During Plant Operation.
(c) Bypassed On "Refuel" And On "Startup" When < 600 psi.
(d) Permits Withdrawal Of One Rod.
(e) Program Loading On Loss Of Both Auxiliary Buses.
(f) May Be Bypassed On "Startup".
(g) Eight- To Ten-Second Time Delay.
(h) At Higher Drywell Pressure Than Scram Value In Combination With Low-Low Water Level 34-Second Time Delay.
(k) Either IRM Or APRM In Startup And APRM In Run.
(l) Bypass In Refuel Or Shutdown.
(m) SRM, IRM, APRM.
(p) With Reactor Pressure ≤ 365 psig.
(r) Manually Retractable After Short Time Delay.

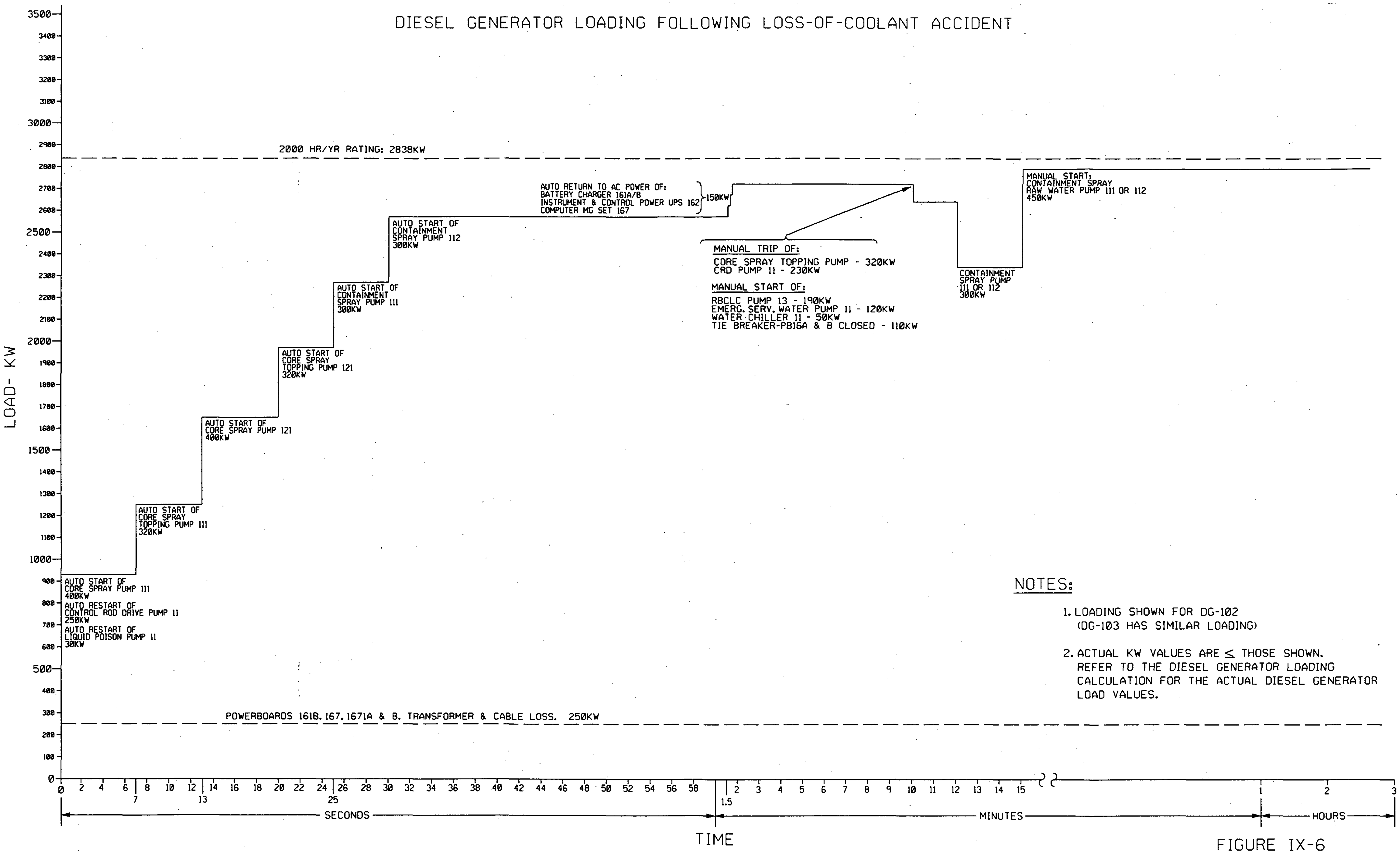

FIGURE IX-6
UFSAR Revision 21
October 2009

REFERENCE CALCULATION 4.16KVACDGES, REVISION 07

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

NINE MILE POINT NUCLEAR STATION UNIT 1

FINAL SAFETY ANALYSIS REPORT (UPDATED)

VOLUME 3

OCTOBER 2009

REVISION 21

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pressure drop which maintains the pressure for this stage is developed by a motor-operated pressure control valve. This valve is manually adjusted from the main control room, and is provided with isolation valves and a manual bypass valve for maintenance. The flow through this valve and the second-stage pressure control valve is substantially constant and the valves, therefore, act to maintain a constant differential above reactor pressure. Changes in the setting of these valves are required only to adjust for changes in the cooling requirements of the drive mechanism as the seal characteristics change with time, and for changes in pump flow characteristics.

The cooling water is monitored by a flow indicator. A differential pressure indicator indicates the difference between reactor pressure and cooling water pressure.

2.6 Exhaust Header

The exhaust header takes water discharged by the drives during operation and by the third-stage pressure controller and conducts this water to the reactor. The piping is sized to maintain a low differential (approximately 5 psi) above reactor pressure in this header. A check valve permits isolating this line from the reactor vessel and automatically prevents reactor water from flowing into this line should the supply pressure fail. A flow element and an indicator permit measuring the exhaust line flow during Station operation. A bypass line from the pump output to a point upstream of this flow meter allows checking of pump flows.

2.7 Accumulator

The accumulator on each drive is an independent source of stored energy to scram that drive. The top of the accumulator contains water; the bottom is initially precharged to approximately 600 psi with nitrogen.

To assure that it is always capable of producing a scram, the accumulator is continuously monitored for water leakage and for nitrogen pressure. A float-type level switch will actuate an alarm if water leaks past the nitrogen-water barrier and collects in the bottom of the accumulator. A pressure indicator and a pressure switch are connected to the accumulator to monitor nitrogen pressure. During normal operation the accumulator barrier has virtually zero pressure drop across it. If there should be any loss of nitrogen, the barrier will move onto a stop and further loss will cause a decrease in the nitrogen pressure. The accumulator barrier will not move down beyond the stop and, therefore, will not compress the reduced amount of gas back up to pressure. A decrease in nitrogen pressure will actuate the pressure switch and sound an alarm. An isolation valve allows each of the accumulator instruments to be isolated and serviced. A connection on the accumulator provides for precharging and bleeding.

The charging line allows isolation of the accumulator for maintenance and prevents backflow from the accumulator to the charging header. It assures that the accumulator will retain its charge even if the supply subsystem fails.

2.8 Scram Pilot Valves

During normal operation, each of the two parallel branches of the RPS energize one of the two three-way solenoid scram pilot valves associated with each drive mechanism. During normal operation, these pilot valves are energized and supply instrument air to the operators of both the inlet scram valve and the outlet scram valve, holding both scram valves closed. During a full scram, both of the RPS branches are de-energized and both pilot valves open, venting the scram valves' operators and allowing the scram valves to open. To protect against spurious scrams, the pilot valves are interconnected so that both pilot valves must be de-energized to vent the scram valves' operators. On the other hand, failure of either electric power to both solenoids or instrument air will produce a scram. The pilot valves are selected based on simplicity of design, a minimum of moving parts, fast opening time, and satisfactory statistical operating history on similar units.

For added protection, the instrument air header to all the pilot valves has a pair of backup scram pilot valves. Upon a scram signal these three-way solenoid valves close off the air supply and vent the instrument air header. This will scram any drive should either of its scram pilot valves fail to vent.

A diverse reactor trip system, alternate rod injection (ARI), has been added to provide an alternate and diverse method of venting the instrument air header. An ARI initiation signal, high reactor pressure, or low-low water level will actuate the ARI system.

2.9 Scram Valves

The inlet scram valve is a globe valve which is opened by the force of an internal spring and closes when air pressure is applied on top of the diaphragm operator. The opening force of the spring is approximately 700 lb. Each valve has a position indicator switch which energizes a light in the control room as soon as the valve starts to open. The scram valve is selected based on high operating force, fast opening time (approximately 0.1 sec) and satisfactory operating history on similar units.

Both the inlet and outlet scram valves are similar, except that the inlet scram valve is an angular pattern while the outlet scram valve is a globe pattern. The internal spring preload in the outlet scram valve is slightly greater than the inlet scram valve to provide a faster opening characteristic.

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Demineralized water is normally provided directly to the following:

Liquid Poison System

Laboratories and Sample Sinks

Stator Winding Liquid Cooling System

Condensate Filtration System Air Compressor

3.0 System Evaluation

Operation of the portable makeup system is on demand at routine infrequent intervals to replenish demineralized water in storage tanks. With the system inoperable or when the portable demineralizer skid is not available, the Station can continue operation with makeup water from the CSTs which have a combined capacity of 400,000 gal. Additional makeup water is available from the demineralized makeup water storage tank which has a 40,000-gal capacity.

As an option, Operators may take a supply of water from city water for processing, depending on the plant operating conditions.

City water is an equivalent or better source for makeup than lake water in terms of contaminants, and delivery capacity is within or exceeds the requirements for supply to the demineralized water system.

4.0 Tests and Inspections

The demineralizer effluent is controlled by effluent conductivity, but periodic samples are taken of conductivity, TOC, silica, chlorides, and sulfates.

H. SPENT FUEL STORAGE POOL FILTERING AND COOLING SYSTEM

1.0 Design Bases

This system is designed to remove the spent fuel assemblies' decay heat and the impurities from the pool water so as to maintain the temperature and purity of the spent fuel pool water at acceptable levels, assuring clarity under all anticipated conditions. The pool water temperature is maintained at or below 140°F during maximum anticipated storage conditions and 110°F during reactor power operation to maintain the secondary containment licensing basis. Normal refueling conditions are based on refueling the reactor every 24 months. During certain instances, it may be necessary to offload the entire core into the spent fuel pool. The maximum heat generation rate was determined by assuming a full core discharge (532 bundles) after 24 months, with the maximum number of previously discharged fuel bundles (3550) being present in the pool. The greatest portion of the decay heat would be produced by the bundles being discharged from the core, rather than those bundles which have been stored in the spent fuel pool from previous discharges. The long-term decay heat rate for GE11 fuel is essentially the same as for previous fuel designs. Therefore, the decay heat rate used as the basis for the spent fuel storage pool filtering and cooling system design remains unchanged.

Prior to Technical Specification Amendment No. 167, the spent fuel pool was licensed for 2776 storage cells. The north half of the pool contained 1066 nonpoison flux trap storage locations, and the south half provided 1710 locations using Boraflex as a neutron absorber. Currently, the spent fuel pool is licensed, per Technical Specification Amendment No. 167, for 4086 spent fuel storage locations using the neutron absorber material Boral, with 1840 storage locations in the north half of the pool and 2246 locations in the south half. The nonpoison racks in the north half of the pool were replaced with new poisoned racks after the 1999 refuel outage. The reracking of the south half of the pool has been partially completed. Six of the eight existing Boraflex racks have been replaced with new Boral racks, increasing the capacity from 1296 to 1656 storage locations. Two Boraflex racks remain in the south half, providing 414 storage locations. The rerack of the remaining two racks has been deferred until further capacity increase is warranted.

Unit 1 committed to the Nuclear Regulatory Commission (NRC) that refueling and core offloading operations would not begin until it was determined that the spent fuel pool cooling systems were operable, to ensure that the bulk pool temperature limits would not be exceeded.

For a normal (full core offload or core shuffle) refueling, the offload time to the spent fuel pool and the RBCLC temperatures shall be verified to be consistent with a bulk pool temperature not to exceed 140°F with one cooling train operating.

For the case of an abnormal maximum heat load (such as a full core offload shortly after a normal refueling), this would require verifying that offload time and RBCLC temperatures were consistent with a pool temperature <140°F with both cooling trains operating.

Based on past experience, sufficient clarity of the pool water can be achieved by a filter capable of removing particles as small as 25 microns in size.

2.0 System Design

The system is shown on Figure X-8. Two full-capacity (600 gpm) pumps take suction from the pool surge tanks and circulate the pool water through two parallel loops consisting of one filter and one heat exchanger. The water is returned to the pool on the side opposite the surge tank skimmers.

The spent fuel pool cooling (SFC) system is designed as seismic Category 1.

The SFC system bounding design conditions are that, under full core discharge conditions with RBCLC coolant water temperature at its maximum of 95°F, and assuming the SFC heat exchangers are fouled to their design maximum and 5 percent of the tubes are plugged, a pool water temperature of 140°F would be reached if a full core offload began 1008 hr after reactor shutdown, and was completed 1129 hr after reactor shutdown with one of the two redundant cooling trains operating.

A more expedited offload may be performed if the plant conditions exist to maintain the pool water temperature at or below 140°F with one SFC train operating.

Flow control valves regulate the flow in each loop at 600 gpm by use of a controller that may be operated in the auto or manual mode.

Cooling water is supplied to the heat exchangers from the RBCLCW system at temperatures not exceeding 95°F. A sample point is incorporated to determine any tube leakage.

Initial filling and level maintenance in the spent fuel pool and surge tanks was from the condensate transfer system. The total volume of the surge tanks is approximately 2000 cu ft. They will normally run at a level of approximately 1000 cu ft. The difference in surge tank volume allows for the displacement of water from the spent fuel storage pool when a shipping cask (or any other object) is placed in the pool.

Makeup water is provided by the condensate transfer system. Normally, makeup is directly to the spent fuel storage pool. Makeup to the spent fuel storage pool is automatically initiated when the surge tank volume decreases to 800 cu ft and stops when the volume reaches 1000 cu ft. If the makeup to the spent fuel storage pool is not sufficient to maintain surge tank volume, makeup water can be provided directly to the surge tanks. The condensate transfer system can provide a makeup rate of 75 gpm or more to either the spent fuel storage pool or the surge tanks. Makeup water can also be supplied directly to the spent fuel pool through fire water hoses.

Any particles that enter the pool either sink to the bottom to be removed by a portable vacuum cleaner or float about in the pool and eventually enter the skimmers, surge tanks and filtering loop. Provision is made for transferring water to the liquid waste disposal system for processing if the pool water becomes highly contaminated.

The precoat-type filters use porous carbon elements. Precoat material is powdered/crushed resins. One precoat mix tank and pump serves both filters. The slurry is circulated through the filter vessel and back to the tank until a uniform coating of precoat material covers all the elements. The filter is then placed in service until differential pressure signals the need for backwashing. The backwashing process consists mainly of first valving off and draining the filter, then filling the filter with condensate from the condensate transfer system. All vents are closed during this filling and air is trapped in the filter dome above the elements. When the pressure in the filter dome reaches approximately 80-100 psig, the drain valve is quickly opened and the filter cake, together with trapped impurities, washes into the fuel pool filter sludge tank. From the sludge tank the suspension of impurities and water is pumped to the waste disposal system.

Aside from its normal function of cooling and purifying the spent fuel pool water, the system is also used after reactor refueling to drain the reactor internals storage pit and head cavity. Alternate lines allow transport of the water to either the main condenser or to the waste disposal system for processing. In either case the water is filtered, demineralized and returned to the CSTs. Each major piece of equipment is designed to withstand seismic forces of 0.25g horizontally and 0.125g vertically. The ASME Boiler and Pressure Vessel Code, Section VIII-1965, is specified for pump casings, heat exchanger, filter vessels, and the sludge tank, as well as for the fuel pool surge tanks.

The fuel pool filters and the surge tanks are shielded with concrete to give a design radiation level of 5 mr/hr outside the shielded area.

3.0 Design Evaluation

Precoat-type filters capable of removing particles as small as 1 micron are provided, although experience indicates that 25-micron particle size filtration should be sufficient to maintain pool clarity.

Each pump filter heat exchanger loop is adequately sized to handle the normal heat load of the spent fuel storage facility, providing a complete standby loop. The two loops are adequate to handle the full core discharge storage heat load.

Various precautions are taken to assure minimum loss of water from the system. All penetrations into the pool are located at a minimum height from the bottom such that there will always be at least 1 ft of water above the fuel. Siphon breakers are used where necessary and the pumps are sealed externally. For flexibility, either pump may be used with a given filter heat exchanger loop.

Makeup water to the spent fuel storage pool is provided by the condensate transfer system. The condensate transfer system can be supplied emergency power from the diesel generators, ensuring the supply of makeup water in the event of loss of both normal and reserve ac power.

Makeup water is also available to the spent fuel storage pool through the fire protection system by the use of a water hose.

The fuel pool cooling system is controlled from a local panel. The Operator is provided with indications of system flow, pool water level, water temperature (on both sides of the heat exchangers), sludge tank level, and valve positions.

Alarms are provided on the annunciator and the computer for high- and low-pressure flow and temperature where critical.

The spent fuel pool system may be secured for maintenance for limited periods as long as: 1) the time available for the maintenance activity has been predicted by an approved calculation, which ensures the pool temperature will remain below 110°F; 2) the pool temperature is closely monitored during the maintenance activity to ensure the temperature does not exceed 110°F (the maintenance time available may be increased based on this empirical data); and 3) the condensate transfer system is available for makeup.

4.0 Tests and Inspections

All equipment in this system will be normally operated, as spent fuel and other components are stored in the pool. However, if equipment such as the spare pump filter heat exchanger loop should stand idle for some time, it will be exercised to assure that it operates properly.

with water to a height of 48 ft 9 in above the top of active fuel (TAF). This water is filtered and cooled by the spent fuel storage pool filtering and cooling system. Additional cooling can be provided by the reactor shutdown cooling and cleanup systems. These systems are described in Sections X-H, X-A and X-B, respectively.

A second stainless steel-lined transfer canal connects the reactor head cavity to the reactor internals storage pit. During refueling, this canal is filled with water to a depth of 19 ft 6 in. Steel-sheathed concrete plugs, 5 ft 5 in thick, shield the refuel floor from the reactor head cavity during power operation. Both canals are filled with similar plugs during power operations. The concrete plugs provide approximately 4 ft of shielding between the equipment storage pit and the reactor head cavity, and approximately 4.5 ft of shielding between the spent fuel pool and the reactor head cavity.

The reactor internals storage pit is a reinforced concrete pit, completely lined with stainless steel. The pit is flooded with water to a depth of 24 ft 0 in during refueling. This water is circulated through the spent fuel storage pool filtering and cooling system. The pit is large enough to accommodate the reactor steam separator and the reactor steam dryer assemblies side by side.

The refueling platform is equipped with a 1200-lb capacity main hoist, and two 1,000-lb capacity auxiliary hoists. Each of these hoists can be positioned over any point in the reactor head cavity or the spent fuel storage pool.

Protective interlocks, discussed in the Refueling Accident, Section XV, are installed in the power supplies to the refueling platform to prevent inadvertent reactivity additions to the core during refueling.

The operating floor is serviced by the reactor building crane, which is equipped with a 125-ton main hoist and a 25-ton auxiliary hoist. These hoists can reach all areas of the operating floor. The 125-ton main hoist is also equipped with a redundant hoisting system, which will prevent the dropping of heavy loads in the event that a cable or other critical part of the main hoist equipment should fail. Three 1/2-ton capacity portable jib cranes are provided for operations in the fresh fuel storage vault and the spent fuel storage pool. Mountings (five in all) for these cranes are provided around the periphery of the pool.

A variety of tools for remote handling of fuel and reactor internals and flow channel exchange are provided.

Fuel sipping may be required to identify fuel assemblies that contain failed fuel rods. Additionally, the fuel assemblies identified to contain failed rods may be inspected and repaired in the fuel prep machine. In order to perform this work, it is required to store fuel sipping equipment in empty control blade rack cells until such time that the sipping operation is complete.

2.1.1 Cask Drop Protection System

The cask drop protection system has been designed to 1) prevent loss of spent fuel pool integrity as a result of certain types of cask drop accidents which may occur over the spent fuel pool, and 2) minimize damage to spent fuel and other components stored in the pool. Specifically, the system has been designed to meet the following functional requirements:

1. Prevent the cask from tipping into the spent fuel pool.

2. Guide the falling cask into the hydraulic dashpot section of the structure.

3. Control the attitude of the cask as it falls through the guide structure and dashpot assembly.

4. Decelerate the cask to a low impact velocity.

5. Absorb the energy of the cask upon impact.

6. Limit loads transmitted to the floor of the spent fuel pool to acceptable values.

This system consists of a circular base plate attached to the bottom of the shipping cask and a combination guide structure--dashpot assembly which is permanently installed in the spent fuel pool (Figure X-11). The structural design of the cask drop protection system is based on the worst-case hydraulic, vertical and lateral loadings associated with a wide range of postulated cask drop accidents.[4,5,6] This design provides protection against a wide range of different size and weight shipping casks.

A summary description of the basis for conducting safe heavy load movements is provided in Section X-J.2.3. Sufficient protection from the risk associated with potential heavy load drops is also provided by satisfying the guidelines of NUREG-0612, Sections 5.1.1 and 5.3[7,8].

2.2 Operation of the Facility

Fresh fuel is brought into the reactor building through the reactor building track bay extension shown on Figure III-4, and hoisted to the operating floor through the equipment hatch utilizing the reactor building crane. (See Figures III-5 to III-9.) The fresh fuel is removed from its shipping containers, inspected, flow channels attached, and stored in the fresh fuel storage vault.

Normally prior to refueling, the fresh fuel is transferred to the spent fuel storage pool using the 25-ton auxiliary overhead hoist.

In preparation for refueling, the concrete shield plugs in the reactor head cavity and the transfer canals are removed by the reactor building crane. The drywell head and reactor vessel head are removed using the same crane.

The steam dryer and the steam separator assemblies are transferred to the reactor internals storage pit. Water levels are controlled such that the steam separator is transferred submerged.

During the disassembly process, demineralized condensate is pumped into the reactor until the head cavity and the reactor internals storage pit are flooded to the normal level of the spent fuel storage pool. The spent fuel storage pool gates are removed after the water level has reached the normal level of the spent fuel storage pool.

Spent fuel is removed from the reactor using a grapple attached to the refueling platform and placed in racks in the spent fuel storage pool. The same equipment is used to transfer the fuel from the spent fuel storage pool to the reactor.

At the completion of reactor refueling, the moisture separator, steam dryer and reactor head are put back into place following the proper maintenance procedures. The drywell head and concrete shield blocks are then restored.

After refueling, the spent fuel bundles are stored in spent fuel storage pool racks. They will remain there until NRC resolution of disposal problems is finalized.

2.3 Control of Heavy Loads Program

2.3.1 Introduction/Licensing Background

NUREG-0612 provides regulatory guidelines for the control of heavy loads to assure the safe handling of heavy loads in areas where a load drop could impact stored spent fuel, fuel in the reactor core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. In a letter dated December 22, 1980[11] (later identified as GL 80-113), as supplemented by GL 81-07[12] and GL 83-42[20], the NRC requested that licensees describe how these guidelines were satisfied at their facility. This request was divided into two phases (Phase I and Phase II). The Niagara Mohawk Power Corporation (NMPC) response to the Phase I portion of the request for Unit 1, addressing the guidelines of Section 5.1.1 of NUREG-0612, was initially provided in letters dated May 22, July 28, and September 22, 1981[13-15]. Supplemental information was subsequently provided in NMPC letters dated August 1, 1982; September 30, November 15, and December 15, 1983; July 26, 1984; and January 18, August 5, and November 25, 1985[16-19,21,22,24,25].

By letter dated March 5, 1985[7], the NRC issued their safety evaluation which concluded that the guidelines in NUREG-0612, Sections 5.1.1 and 5.3, had been satisfied for Unit 1, and that Phase I of the NMPC response for Unit 1 was acceptable. In GL 85-11[8], the NRC documented their determination that a detailed Phase II review of heavy loads was not necessary and that Phase II was considered completed.

By letter dated May 13, 1996[27], NMPC provided the required response to Bulletin 96-02[26] for Unit 1. The response reiterated that the movement of heavy loads over critical areas of the refuel floor and safety-related equipment is performed in accordance with controlled site procedures developed in accordance with NUREG-0612. Additionally, the response reaffirmed that the reactor building 125-ton crane is single-failure-proof (i.e., a dual load path, redundant hoisting system). The NRC's April 23, 1998, letter[28] accepted the NMPC response and indicated completion of tasks associated with Bulletin 96-02.

On July 28, 2008[29], the Nuclear Energy Institute (NEI) transmitted NEI 08-05, Revision 0, Industry Initiative on Control of Heavy Loads to the NRC. This document was issued to provide an industry agreed-upon approach to providing additional assurance of compliance to existing regulatory guidelines regarding control of heavy loads at nuclear power plants. This NEI document includes guidance associated with updating UFSARs to reflect a summary description of the basis for conducting safe heavy load movements. The NRC safety evaluation of the guidelines contained in NEI 08-05, transmitted to NEI by letter dated September 5, 2008[30], determined that the guidelines may be used by licensees to establish a revised licensing basis for handling of reactor vessel heads and other heavy loads, subject to the clarifications and conditions noted in the NRC's safety evaluation.

2.3.2 Safety Basis

Heavy load handling activities pose a safety risk in the areas of nuclear power plants where load drops could impact irradiated fuel or equipment necessary for safe shutdown. Implementing the guidelines of NUREG-0612, Section 5.1.1, reduces the potential for heavy load drops and provides a measure of defense-in-depth against such an occurrence.

The risk associated with load handling failures is acceptably low based on meeting the Phase I requirements of NUREG-0612, Section 5.1.1, and the use of the reactor building 125-ton single-failure-proof crane for lifting the reactor vessel head and spent fuel casks. The 125-ton reactor building crane is a single-failure-proof crane as defined in NUREG-0612, Appendix C, and has a redundant hoisting system which is independently capable of supporting the crane's rated load[23].

2.3.3 Scope of Heavy Load Handling Systems

In NUREG-0612, the scope of cranes includes:

> "Overhead handling systems that are used to handle heavy loads in the area of the reactor vessel or spent fuel in the spent fuel pool. Additionally, loads may be handled in other areas where their accidental drop may damage safe shutdown systems..."

Based on the NMPC Phase I responses in References 13 through 19, 21, 22, 24, and 25, the reactor building 125-ton crane is within the scope of Section 5.1.1 of NUREG-0612. Heavy load movements in areas of safe shutdown equipment that are handled by other load-handling systems are also performed in accordance with the requirements of NUREG-0612 as defined in controlled site procedures. These other systems include, but are not limited to, the reactor building 25-ton auxiliary crane, the turbine building 150-ton crane, and the screen and pump house 25-ton crane.

2.3.4 Control of Heavy Loads Program

The Control of Heavy Loads Program consists of the following:

1. NMPNS commitments in response to NUREG-0612, Phase I elements, as described in References 13 through 19, 21, 22, 24, and 25.

2. For reactor pressure vessel head (RPVH) and spent fuel cask lifts, the single-failure-proof reactor building 125-ton crane described in Section X-J.2.3.4.2 is used.

2.3.4.1 NMPNS Commitments in Response to NUREG-0612, Phase I Elements

NMPNS has committed to controlling the movement of heavy loads in accordance with the seven elements of Section 5.1.1 of NUREG-0612, as defined below:

1. Safe load paths for movement of heavy loads are defined in controlled plant procedures[14].

2. Controlled plant procedures are developed and implemented that control movement of heavy loads[14].

3. Crane operators are trained and qualified in accordance with controlled plant procedures[14].

4. Special lifting devices follow the guidelines of ANSI N14.6-1978[15].

5. Lifting devices not specifically designed follow the guidelines of ANSI B30.9-1971[15].

6. The reactor building 125-ton crane is inspected, tested and maintained consistent with ANSI B30.2-1976[14].

7. The reactor building 125-ton crane is designed to CMAA-70 and meets the applicable criteria and guidelines of ANSI B30.2-1976[14,15].

2.3.4.2 Reactor Pressure Vessel Head and Spent Fuel Cask Lifts

The reactor building 125-ton crane is single-failure-proof, has a redundant load path, and is designed to CMAA-70. The following attributes were defined in the design of the crane[23]:

1. Allowable stress limits are defined and conservative enough to prevent permanent deformation of individual structural members when exposed to maximum load lifts.

2. The crane is capable of stopping and holding the load during a design basis earthquake.

3. Automatic controls and limiting devices are designed so that they fail-safe and do not prevent the crane from stopping and holding the load safely.

4. The design of the wire rope reeving system includes dual wire ropes.

5. Limit switches are included to limit such items as overspeed, overload and overtravel and cause the hoisting action to stop when limits are exceeded.

6. The reeving system is designed against the destructive effects of "two-blocking."

7. Safety devices such as limit switches are provided to reduce the likelihood of a malfunction.

2.3.5 Safety Evaluation

Controls implemented by NUREG-0612, Phase I elements, together with the use of a single-failure-proof crane for RPVH and spent fuel cask lifts, make the risk of a load drop extremely unlikely and acceptably low. The risk associated with the movement of heavy loads is evaluated and controlled by station procedures.

3.0 Design Evaluation

The spacing of fuel bundles in the fresh fuel storage vault maintains keff <0.95 even if flooded with water. The vault floor drain prevents flooding. The spacing of fuel bundles in the spent fuel storage pool maintains keff <0.95. A criticality monitor in the fresh fuel storage vault provides warning in the unlikely event of a criticality incident.

Protective interlocks prevent handling of fuel over the reactor when a control rod is withdrawn. Another set of interlocks prevents control rod withdrawal when fuel is being handled over the reactor. Limit switches on the refueling platform hoists interrupt power to the hoists when the TAF is 8 ft below the surface of the water. Brakes on all equipment lock upon loss of power. Spent fuel will not be inadvertently handled with an inadequate depth of water shielding.

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The above interlocks can be bypassed to permit the unloading of a significant portion of the reactor core (full core offload, spiral offload) for such purposes as removal of temporary control curtains, CRD maintenance, inservice inspection (ISI) requirements, examination of the core support plate, etc. (Technical Specification 3.5.3).

Fuel stored in the spent fuel storage pool is covered by a minimum of 24 ft of water. Irradiated fuel being moved is at all times covered by a minimum depth of 8 ft of water over TAF, except that the fuel preparation machine is provided with mechanical stops to ensure that active fuel remains under 7 ft of water. Spent fuel pool water level is automatically controlled to ensure that during normal operation, spent fuel will be covered by a sufficient depth of water to permit unrestricted access to the operating floor.

The spent fuel storage pool cannot be completely drained. If draining should be initiated due to Operator error, level alarms will notify operating personnel and makeup water will be supplied automatically. If no action were taken, the fuel would still be covered by approximately 1 ft of water after the pool had drained down to the lowest penetration.

All reactor servicing operations are carried out within the secondary containment, which is described in Section VI-C. A bypass around the refueling platform radiation monitor will allow the monitor to be connected into the RPS during refueling operations or when recently irradiated fuel or a fuel-loaded shipping cask is being handled. This monitor provides a fast automatic isolation of the reactor building ventilation system and initiation of the reactor building emergency ventilation system.

4.0 Tests and Inspections

During testing prior to initial reactor fueling, the spent fuel storage pool, reactor head cavity, and reactor internals storage pit were filled with water and checked for leakage. Dummy fuel assemblies were run through a complete cycle from the fresh fuel storage vault to the spent fuel storage pool.

During normal operation, telltales are examined for evidence of potential leakage from the spent fuel pool. Prior to fuel handling, all hoists, cranes and tools are inspected and tested to assure safe operation.

4.0 Tests and Inspections

The preoperational test is performed to confirm the operability of the installed system components and piping configurations. The startup test is performed to verify the function of all system components and the capability of the system to control the process.

N. REFERENCES

1. Application for Amendment to Operating License, Docket No. 50-220, from D. P. Dise (NMPC), dated March 21, 1978.

2. Letter from C. V. Mangan (NMPC) to Domenic B. Vassallo (NRC), dated June 24, 1983, Spent Fuel Pool Modification.

3. Letter from C. V. Mangan (NMPC) to Domenic B. Vassallo (NRC), dated October 5, 1983, Spent Fuel Storage Capacity Expansion.

4. Letter from D. P. Dise (NMPC) to V. Stello (NRC), July 17, 1978. (Attached information: Movement of Heavy Loads Near the Spent Fuel Storage Pool or Reactor Core).

5. Letter from T. J. Brosnan (NMPC) to J. F. O'Leary (AEC), September 29, 1972. (Attached report: NMP-1 Cask Drop Protection System).

6. Letter from P. D. Raymond to D. L. Zieman (AEC), May 31, 1973. (Attached report: NMP-1 Cask Drop Protection System).

7. Letter from D. B. Vassallo (NRC) to B. G. Hooten (NMPC), March 5, 1985, Control of Heavy Loads (Phase 1).

8. NRC Generic Letter 85-11, June 28, 1985, Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612.

9. DRFC 11-61, "Calculation of Free Volume Available for Scram with Leakage for SDV Long-Term," dated June 1, 1981.

10. Technical Specification Amendment No. 167 for NMP1 to Reflect a Planned Modification to Increase the Storage Capacity of the Spent Fuel Pool, dated June 17, 1999.

11. Letter from D. G. Eisenhut (NRC) to All Licensees, December 22, 1980, Control of Heavy Loads (Generic Letter 80-113).

12. Letter from D. G. Eisenhut (NRC) to All Licensees, February 3, 1983, Control of Heavy Loads (Generic Letter 81-07).

13. Letter from D. P. Dise (NMPC) to D. G. Eisenhut (NRC), May 22, 1981.

14. Letter from D. P. Dise (NMPC) to D. G. Eisenhut (NRC), July 28, 1981.

15. Letter from D. P. Dise (NMPC) to D. G. Eisenhut (NRC), September 22, 1981.

16. Letter from T. E. Lempges (NMPC) to D. G. Eisenhut (NRC), August 1, 1982.

17. Letter from T. E. Lempges (NMPC) to D. G. Eisenhut (NRC), September 30, 1983.

18. Letter from T. E. Lempges (NMPC) to D. G. Eisenhut (NRC), November 15, 1983.

19. Letter from T. E. Lempges (NMPC) to D. G. Eisenhut (NRC), December 15, 1983.

20. Letter from D. G. Eisenhut (NRC) to All Licensees, December 19, 1983, Clarification to Generic Letter 81-07 Regarding Response to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" (Generic Letter 83-42).

21. Letter from C. V. Mangan (NMPC) to D. B. Vassallo (NRC), July 26, 1984.

22. Letter from T. E. Lempges (NMPC) to D. B. Vassallo (NRC), January 18, 1985.

23. Letter from P. D. Raymond (NMPC) to D. J. Skovholt (NRC), July 26, 1973.

24. Letter from T. E. Lempges (NMPC) to D. B. Vassallo (NRC), August 5, 1985.

25. Letter from C. V. Mangan (NMPC) to J. A. Zwolinski (NRC), November 25, 1985.

26. NRC Bulletin 96-02, Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core or Over Safety Related Equipment, April 11, 1996.

27. Letter from C. D. Terry (NMPC) to U. S. Nuclear Regulatory Commission, May 13, 1996, NRC Bulletin 96-02, Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core or Over Safety Related Equipment.

28. Letter from D. S. Hood (NRC) to J. H. Mueller (NMPC), April 23, 1998, Completion of Licensing Action for NRC Bulletin 96-02 for Nine Mile Point Nuclear Station, Unit Numbers 1 and 2.

29. Letter from A. R. Pietrangelo (NEI) to E. J. Leeds (NRC), July 28, 2008, NEI 08-05, Revision 0, Industry Initiative on Control of Heavy Loads.

30. Letter from W. H. Ruland (NRC) to T. C. Houghton (NEI), September 5, 2008, Industry Initiative on Control of Heavy Loads.

CONTROL ROD DRIVE HYDRAULIC SYSTEM

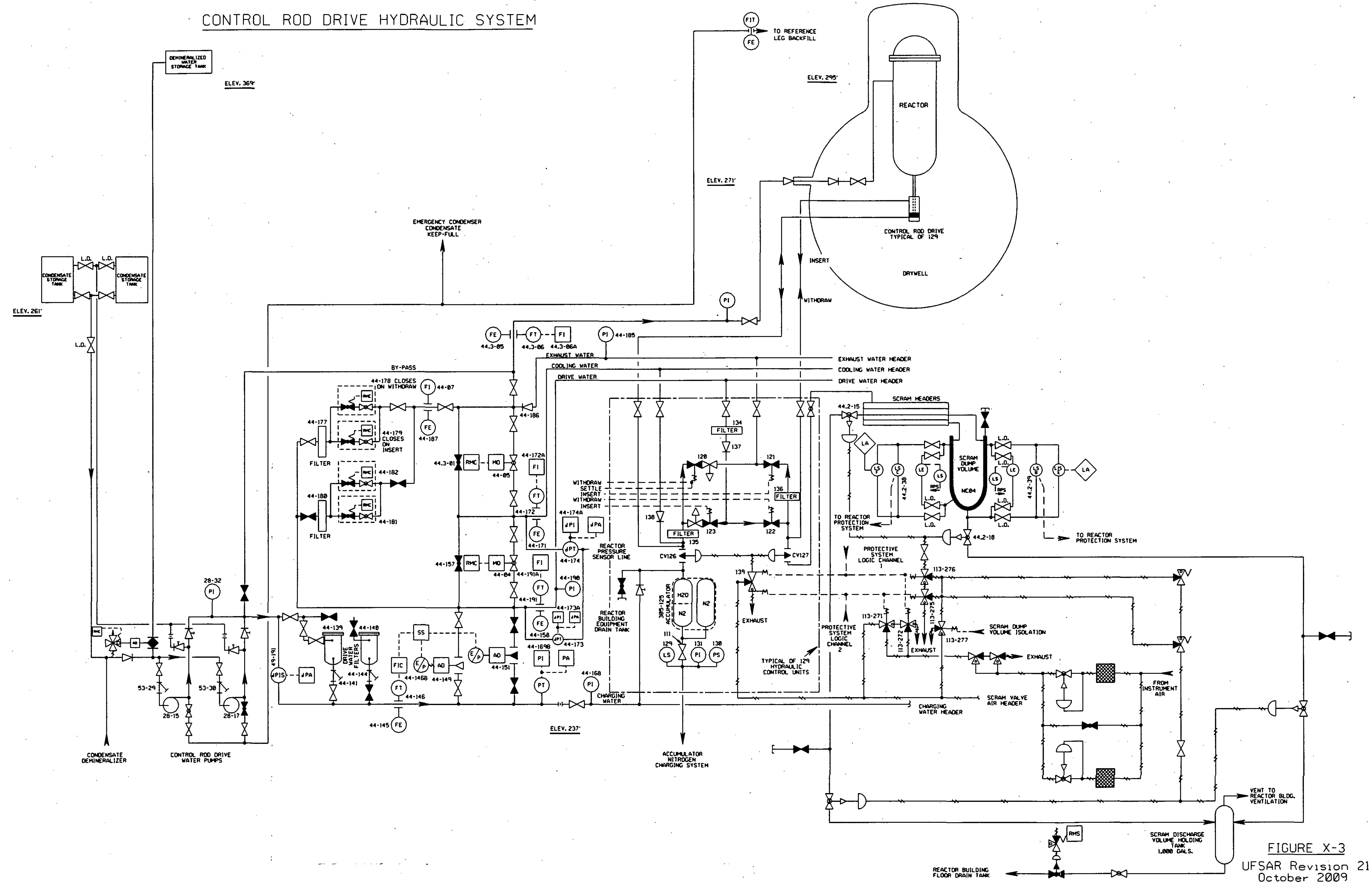

FIGURE X-3

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FIRE HAZARDS ANALYSIS

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and maintenance, training, quality assurance, testing, control of combustibles and housekeeping.

Fire Resistance Rating The time that materials or assemblies have withstood a fire exposure as established in accordance with the test procedures of NFPA Standard 251, Standard Methods of Fire Tests of Building Construction and Materials.

Fire Stop Construction that prevents fire propagation along the length of cables or prevents spreading of fire to nearby combustibles within a given fire area or fire zone.

Fire Suppression Control and extinguishing of fires (fire fighting). Manual fire suppression is the use of hoses, portable extinguishers, or manually-actuated fixed systems by plant personnel. Automatic fire suppression is the use of automatically-actuated fixed systems such as water, halon, or carbon dioxide (CO_2) systems.

Fire Watch Patrol A compensatory action option for fire areas with inoperable fire protection equipment. When chosen as the compensatory measure, a fire watch shall inspect the affected fire area for abnormal conditions (fires or fire hazards) at least once per hour. A person standing a continuous or fire watch patrol who is a member of the Fire Brigade may leave his watch/patrol to respond to a fire alarm. Upon determination that the alarm is a false alarm, the individual will return to his watch/patrol duties in a timely manner, not to exceed 1 hr. In the event the fire alarm is valid, the individual will perform his normal fire-fighting duties as required. This policy places the highest priority on responding to an actual fire; the second highest priority is to attend to fire watch/patrol duties. This position is consistent with 10CFR50 Appendix R which does not consider two simultaneous plant fires.

Fire Zone (FZ) A plant area whose boundaries need not consist of rated or approved fire barriers, but are chosen based on the physical plant design, convenience, and/or layout of the fire detection and suppression system.

Noncombustible Material

1. Material, in the form in which it is used and under anticipated conditions, which will not ignite, burn, support combustion, or release flammable vapors when subjected to fire or heat.

2. Material having a structural base of noncombustible material, as defined in item 1 above, with a surfacing not over 1/8-in thick that has a flame spread rating not higher than 50 when measured using American Society of Testing and Materials (ASTM) E-84 Test, Surface Burning Characteristics of Building Materials.

Raceway Refer to Regulatory Guide (RG) 1.75.

Restricted Area Any area to which access is controlled by the licensee for purposes of protecting individuals from exposure to radiation and radioactive materials.

Safety-Related Activities, structures, systems, components, or parts thereof which are required to assure:

1. Integrity of the reactor coolant pressure boundary (RCPB).

2. Capability to achieve and maintain safe shutdown.

3. Capability to prevent or mitigate the consequences of postulated accidents which could result in offsite exposures comparable to the acceptance criteria of 10CFR50.67.

Safe Shutdown Hot or cold shutdown (reactor subcritical) with control and coolant inventory and decay heat removal.

Hot Shutdown - The reactor is shut down, the reactor coolant inventory is being controlled while the reactor is being depressurized, and the reactor temperature is $>$ 212°F.

Cold Shutdown - The reactor is shut down, the reactor coolant inventory is being maintained with the reactor depressurized so that decay heat is being removed from the reactor vessel and transferred to the ultimate heat sink (UHS), and reactor temperature is $\leq$ 212°F.

Sprinkler System A network of piping connected to a reliable water supply that will distribute the water throughout the area protected, and will discharge the water through sprinklers in sufficient quantity either to extinguish the fire entirely or to prevent its spread. This system, usually activated by heat, includes a controlling valve and a device for actuating an alarm when the system is in operation. The following categories of sprinkler systems are defined in NFPA Standard 13, Standard for the Installation of Sprinkler Systems:

1. Wet-Pipe System
2. Dry-Pipe System
3. Preaction System
4. Deluge System (open sprinklers)

<u>Standpipe and Hose Systems</u> A fixed piping system with hose outlets, hose and nozzles connected to a reliable water supply to provide effective fire hose streams to specific areas inside the building.

<u>Water Spray System</u> A network of piping similar to a sprinkler system except that it utilizes water spray nozzles and provides protection of a specific hazard. NFPA Standard 15, Water Spray Fixed Systems, provides guidance on these systems.

2.0 BTP APCSB 9.5-1 APPENDIX A COMPARISON

2.1 OVERALL NUCLEAR PLANT FIRE PROTECTION PROGRAM REQUIREMENTS

The Fire Protection Program is a program to implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report (Updated), and as approved in the Fire Protection Safety Evaluation Report dated July 26, 1979, and in the fire protection Exemption issued March 21, 1983. Noncompliances with the above-described Fire Protection Program that adversely affect the ability to achieve and maintain safe shutdown in the event of a fire shall be reported in accordance with the requirements of 10CFR50.72 and 10CFR50.73.

2.1.1 Personnel

The Senior Constellation Nuclear Officer Responsible for Nine Mile Point has the overall management responsibility for the nuclear fire protection program.

The Vice President Nine Mile Point has the overall responsibility for the fire protection program at the Nine Mile Point Nuclear Station.

The Manager Operations reports to the Plant General Manager and is responsible for managing, overseeing, and coordinating the site's fire protection functional and technical activities.

The Fire Protection Program Manager reports to the General Supervisor Engineering Programs, who reports to the Manager Engineering Services, and is responsible for managing, overseeing, and coordinating the Fire Protection Engineering group.

2.1.1.1 Organizational Responsibilities

The Senior Constellation Nuclear Officer Responsible for Nine Mile Point has management responsibility for the formulation, implementation, and assessment of the effectiveness of the nuclear plant fire protection program.

The Vice President Nine Mile Point shall have the overall responsibility for the fire protection program at the Nine Mile Point Nuclear Station.

The Manager Operations is responsible for implementation of the site fire protection program at Unit 1 and Unit 2.

The Manager Maintenance shall ensure fire-fighting equipment, systems, and fire barrier integrity for each unit is maintained by performance of maintenance and modifications, as required.

The Manager Training ensures that Fire Brigade personnel are scheduled to attend the required fire training sessions and ensure that the required fire drills are performed. In addition, they are responsible for any necessary fire protection training of operating Station or contractor personnel.

The Fire Marshal shall provide technical support to the Fire Brigade for routine daily matters, required surveillance activities, and to support special investigations and projects:

a. Act as the site contact for fire protection matters such as American Nuclear Insurance (ANI) audits, NRC inspections, QA and Constellation Risk Management audits.

b. Act as liaison between Constellation Risk Management, Site Fire Protection Engineer, and Fire Protection personnel.

c. Consult with the Fire Protection Program Manager for interpretations of adequacy on issues concerning compliance with regulatory and/or fire protection program requirements.

d. Coordinate scheduling of required training and fire protection training drills for Stations operating and offsite fire support personnel.

e. Conduct periodic inspection to assess compliance with the Station fire protection program, and ensure unit fire protection system/equipment operability.

f. Ensure fire protection-related surveillances are performed.

g. Evaluate proposed work activities as required, and maintain an awareness of the status of repairs, modifications, or other work affecting fire protection systems and equipment.

h. Ensure notification of fire insurance carrier (ANI) and Constellation Risk Management is performed when unit fire protection system impairments occur, as required.

The General Supervisor System Engineering shall:

a. Ensure fire protection system support is maintained.

b. Evaluate fire protection system trend data and provide recommendations for corrective actions as necessary.

c. Ensure input into the Operating Experience Assessment (OEA) Program for matters concerning fire protection.

The Manager Operations shall direct the Fire Brigade's day-to-day activities associated with the implementation of the fire protection program.

a. Consult with the Fire Protection Program Manager for interpretations of adequacy on issues concerning compliance with regulatory and/or fire protection program requirements.

b. Direct investigations into fires at the unit, review the determination of cause, and recommend corrective action, as appropriate.

c. Ensure that training is established and scheduled in accordance with program requirements.

d. Evaluate the effectiveness of fire protection training.

e. Direct the inspection and testing of fire protection systems and equipment in accordance with applicable procedures.

f. Ensure a review of inspection and test results is conducted to maintain compliance with license requirements.

g. Ensure that negative performance trends on fire protection systems and equipment are reported to the Fire Marshal.

The Manager Engineering Services has overall responsibility for the design and evaluation of fire protection components and systems, and also periodically assessing, through the Nuclear Safety Review Board (NSRB), the effectiveness of the fire protection program for Nine Mile Point Nuclear Station. Specific actions include:

a. Review of modifications to plant systems by selected personnel under the direction and guidance of a Fire Protection Engineer (qualified) for impacts to the site fire protection program.

b. Designing modifications to fire protection systems in accordance with nationally recognized standards. In addition, assessment of impacts to Final Safety Analysis Report (FSAR) commitments, Fire Hazards Analyses (FHA) and NRC Safety Evaluation Reports (SER). Evaluation of deviation impact is performed by selected personnel under the direction and guidance of a Fire Protection Engineer (qualified).

c. Identification and evaluation of proposed changes to the program or systems which impact licensing prior to the change being made.

d. Identification and resolution of deviations from the program in a timely manner.

e. Development of inspection attributes for fire protection equipment.

f. Provision of support in resolution of deficiencies identified within the fire protection program.

g. Development of testing requirements meeting applicable codes for system changes.

h. Definition and maintenance of design records required to support the fire protection program.

A Fire Protection Engineer (qualified) shall provide direction and guidance to selected personnel assigned to evaluate activities and identified deficiencies to the fire protection program for impacts to program documentation, determine path for resolution of open items or questions, review inspection attributes for fire protection system design changes, recommend program improvements as required, and participate in program evaluation on a periodic basis (audit).

The Manager Engineering Services assigns a Fire Protection Program Manager, and ensures audits are conducted per the fire protection Nuclear Division Directive.

The Fire Protection Program Manager provides organization, direction, and guidance concerning the implementation of the fire protection program, and the approach to be taken regarding fire protection issues as they relate to the overall performance and adequacy of the nuclear fire protection program.

Specific duties include:

1. Maintain cognizance of regulatory positions and trends, determine adequacy/sufficiency of programs to satisfy regulatory and program requirements and commitments, and develop programs to resolve deficiencies, insure auditability and implement corrective actions to maintain acceptable levels of fire protection within the nuclear facilities.

2. Coordinate regulatory response and provide program interpretation for fire protection issues and identify potential fire protection modification requirements.

3. Coordinate the prioritization of identified fire protection work items.

4. Coordinate with appropriate training departments to ensure that required training levels are established for personnel performing fire protection engineering activities.

5. Interface and coordinate with the Fire Marshal regarding matters that impact site fire protection.

6. Provide an overview function to:

    a. Periodically review fire protection engineering evaluations, Appendix R reviews and fire protection reviews.

    b. Ensure that Unit 1 interpretations of regulatory documents and corporate policy are being applied properly and consistently.

    c. Ensure Corporate fire protection philosophy/requirements are transmitted to Nuclear Engineering and that they are implemented in design changes.

The Director Quality and Performance Assessment has overall responsibility for formulating, administering and verifying the effectiveness of the quality assurance program for fire protection (see Section 2.3). The Director Quality and Performance Assessment ensures that the quality assurance program is implemented by planned inspections and scheduled audits, assuring that the results are promptly reported to cognizant management personnel.

Constellation Risk Management has responsibility for assuring adequate fire protection for company facilities and fire personnel training; to verify appropriate measures are taken to prevent or limit losses from any perils resulting from nuclear operations; and for all matters relating to insurance of our facilities.

The Director Materials Services has overall responsibility for preparation, issuing and commercial administration of purchase orders for materials and services in support of fire protection program requirements, and for procedures used for receipt, storage, and handling of materials employed in implementing the fire protection program.

The Manager Maintenance is responsible for supervising and coordinating measuring and test equipment (M&TE) calibration activities.

Supervisors shall ensure that their department(s) observe good safety practices in the use and control of combustible materials and processes which may serve as an ignition source; in addition to good housekeeping practices, each supervisor shall ensure that activities are carried out in a manner that does not endanger essential Station equipment, cabling, piping or instrumentation necessary for safe operation of the Station.

2.1.1.2 Personnel Qualifications

Appendix R Engineer - an engineer assigned by the Manager Engineering Services, or designee, who is knowledgeable in the SSA attributes and design implications, and is capable of determining the impacts of modifications on the aforementioned analysis and the fire protection program.

Fire Protection Engineer (FPE) - an engineer assigned by the Manager Engineering Services, or designee, who is a graduate of a curriculum of accepted standing and who has completed not less than 2 yr of fire protection engineering experience indicative of growth and achievement. In instances where an individual is assigned as a FPE and does not meet the qualifications listed above, his/her work will be reviewed by an individual who does.

Fire Protection Engineer Qualified (FPEQ) - an engineer assigned by the Manager Engineering Services, or designee, who is a graduate of an engineering curriculum of accepted standing and who has completed not less than 6 yr of engineering experience indicative of growth in engineering competency and achievement, 3 yr of which shall have been in charge of fire protection engineering work, or have full Member status in the Society of Fire Protection Engineers.

As a minimum, a site Fire Brigade consisting of five Brigade members is assigned rotating shifts with fire and rescue responsibilities. Site administrative controls require that at least two first aid responders are available at all times for medical response.

2.1.1.2.1 Action

At all times, a Fire Brigade of five members shall be maintained on the Nine Mile Point site (excludes the James A. FitzPatrick Nuclear Power Plant). Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hr, in order to accommodate unexpected absence, provided immediate action is taken to fill the required position.

The members of the Fire Brigade are subjected to an initial and annual physical exam to determine their capability to perform strenuous fire-fighting activities. This is in addition to any required training to maintain the position in the Fire Brigade.

The Fire Training Specialist is in charge of classroom and hands-on training. This individual has practical experience in fireground tactics and is knowledgeable in fire protection system design as it applies to Unit 1.

2.1.2 Design Basis

Consistent with NRC guidance, the fire protection program utilizes the concept of defense-in-depth to fire protection in safety-related areas, with the following objectives:

a. Prevent fire from starting;

b. Detect rapidly, control and extinguish promptly those fires that do occur;

c. Provide protection for safety-related structures, systems, and components so that a fire that is not promptly extinguished by the fire suppression activities will not prevent the safe shutdown of the plant.

To demonstrate the review of the facility to accomplish these objectives, Section 3.0 of this document analyzes facility structures for discrete fire "hazards" along with the compensatory measures provided, which include active and passive fire protection features. In addition, the Unit 1 SSA verifies the capability of the plant to initiate and maintain safe shutdown following a complete loss of equipment in critical fire areas in the plant.

2.1.3 Backup

Where automatic suppression systems exist at Unit 1, total reliance on this system to provide suppression capabilities is not made. Suitable backup suppression capability is provided for all such areas through the use of a hose standpipe system and portable fire extinguishers.

2.1.4 Single Failure Criterion

A single failure of a fire pump or controls for the fire pumps will not affect the ability to supply water to the water distribution system. Each fire pump independently can supply 100 percent of the flow requirements for Unit 1.

Adequate separation exists in the water distribution system such that a single failure in the supply piping will not impair the primary and backup suppression systems for given areas of the plant. An exception exists in the screenhouse, where a single break in the water supply header from the fire pumps to the water distribution system will impair the availability of the Unit 1 fire pumps. In the event of this occurrence, the Nine Mile Point Nuclear Station - Unit 2 (Unit 2) water supply system can be cross-connected with the Unit 1 system south of the administration building to provide Unit 1 water supply requirements while repairs are made. (Reference also Section 2.5.2.3.)

A single supply water sump provides water for both fire pumps in the screenhouse. Isolation of, or failure of, this sump will impair the availability of the Unit 1 fire pumps. In the event of this occurrence, the Unit 2 water supply system can be cross-connected with the Unit 1 system to provide Unit 1 water supply requirements.

For emergency conditions, provision has been made to connect the municipal potable water supply through a hose to a fire department pumper, then to the fire main system in the event both fire pumps are out of service.

Lightning protection is provided for Unit 1 power block structures.

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2.2.3.2 Leak Testing

Open flame or combustion-generated smoke is not used for leak testing or similar procedures such as air flow determination for leak testing.

2.2.3.3 Combustible Material Storage

The storage of combustible supplies in safety-related areas is controlled by administrative procedures. The use of wood in safety-related areas is minimized. In general, only fire-retardant treated wood is permitted in safety-related structures. Isolated instances where untreated wood is utilized is evaluated on a case-by-case basis for program impacts and adequacy of installed fire protection systems.

2.2.4 Local Fire Department Support

The plant Fire Brigade provides primary response to fire emergencies. Upon determination by the Chief Nuclear Fire Fighter that additional fire fighting assistance is required, municipal response from local fire companies will be provided to support Station operations. Support by local fire companies is addressed by the Site Emergency Plan (SEP) and Mutual Aid Agreements.

2.2.5 Fire Brigade

The Fire Brigade is organized, trained and equipped to address fire emergencies at the plant. A variety of protective clothing and equipment, breathing equipment, salvage covers and/or forcible entry and rescue tools are provided in various locations on site in order to effectively respond to expected emergencies.

Guidance in responding to and responsibility of individuals during fire emergencies is defined in SEP procedures.

2.2.5.1 Surveillance and Maintenance

Procedures have been developed which identify required notifications, methodology and inspection attributes to support surveillance testing and periodic maintenance to fire protection equipment. Compensatory measures during system impairments are also addressed in department procedures.

Each surveillance requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval. This permits an allowable extension of the normal 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 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 this allowance 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 surveillance requirements. 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.

2.2.5.2 Fire Drills

Fire drills are conducted on a quarterly basis to insure that members work together as a team to address a simulated fire event. These drills are evaluated by supervisory personnel to assess leadership effectiveness, knowledge of responsibilities and of equipment. The drills are critiqued following completion to determine any necessary corrective action which may be warranted.

2.2.6 Fire Brigade Training

The training program for the Fire Brigade is maintained under the direction of the Manager Training and Fire Marshal, and meets or exceeds the requirements of Appendix R to 10CFR50.

Fire Brigade members are trained in accordance with approved training procedures to familiarize the individuals with fire protection systems and equipment, plant fire hazards and emergency response. This training program is also intended to ensure that the Brigade leader and at least two members have sufficient training and a knowledge of plant safety-related systems to understand the effects of fire and fire suppression on safe shutdown capability.

a. Main Transformer No. 1

b. Main Transformer No. 2

c. Station Service Transformer 10

d. Reserve Transformer 101N

e. Reserve Transformer 101S

Unit 1 utilizes an engineered wastewater treatment facility for runoff from oil spill areas including transformers. The containment of potential oil spills at the sources is accomplished with curbs and basins around the oil spill areas. A system of drainage sewers transports potential runoff from these areas to a retention basin, where the oil is separated from the runoff prior to its release. This design accounts for runoff from rainfall, as well as automatic and manual fire suppression systems. The objective of this system is to treat oily water runoff from the Unit 1 areas that have the potential for oil pollution.

2.4.1.9 Floor Drains

Unit 1 has performed an analysis of standing water damage to safety-related equipment or supporting systems necessary for the safe shutdown of the plant, resultant from automatic fire system operation with manual suppression activities. The conclusion indicated that the combination of floor drains, floor sumps and ponding capability is sufficient to prevent damage to this safety-related equipment resulting from expected fire-fighting water. In certain areas, curbs have been provided or equipment has been installed on pedestals to isolate the equipment from an oil or water spill.

2.4.1.10 Fire Barriers/Penetrations

Unit 1 utilizes primarily 2-hr and 3-hr rated fire barriers to separate fire areas or protect safety-related equipment from exposure fire hazards. These barriers are identified on Figures 10A-2 through 10A-9. The rating of these barriers is determined based on the hazard present and the evaluation of significance of equipment in the area. The barriers identified are all-inclusive of those currently maintained at the Station.

Thermal shield walls are also utilized in specific applications where a hazard cannot be sufficiently bounded by rated construction to be termed as a distinct fire area due to configuration or original plant design. Certain portions of these walls are sealed with configurations which would meet a rated fire barrier to protect important equipment outside the area from a fire inside the area. These walls are also identified on the subject figures and have been evaluated with respect to the protection required.

Penetrations made in fire-rated barriers are sealed with configurations which will maintain the integrity of the barrier. In the main steam isolation valve (MSIV) room, ten penetrations between primary containment and the turbine building are sealed in such a manner to ensure primary containment integrity. Although these penetrations are not a classic rated penetration seal, the configuration has been found to be acceptable in the location utilized.

The fire barriers identified on drawing C-39591-C (M31.1), sheets 1 through 6, separate redundant safety-related areas or provide exposure protection for safety-related areas. The barriers, including cabling, cable penetrations, pipe penetrations, fire doors and fire dampers, shall be intact. The following surveillance requirements are applicable to these barriers.

Action

With one or more of the above required fire barrier penetrations nonfunctional, within 1 hr, implement one of the following actions:

a. Establish a continuous fire watch on one side of the affected penetration, or

b. Verify the operability of fire detectors on both sides of the nonfunctional barrier and establish a daily inspection of the nonfunctional barrier to verify no increase in fire hazards within the vicinity, or

c. Verify the operability of fire detectors on one side of the nonfunctional fire barrier and establish a fire watch patrol, or

d. Implement a preplanned provision(s) in accordance with the assessment of a qualified Fire Protection Engineer.

2.4.1.10.1 Surveillance

The fire barriers, excluding penetration seals and fire dampers, shall be verified to be functional by:

a. A visual inspection at least once per two operating cycles.

b. A visual inspection of a fire barrier penetration after repair or maintenance, prior to restoring the fire barrier penetration to functional status.

Penetration seals shall be verified functional by:

a. A visual inspection at least once per operating cycle of at least 10 percent of each type of sealed penetration. If significant changes in appearance or abnormal degradation are found, a visual inspection of an additional 10 percent of that type of sealed penetration shall be made for each unsatisfactory finding. This inspection process shall continue until a 10 percent sample with no significant changes in appearance or abnormal degradation is found. Samples shall be selected so that each penetration seal will be inspected at least once every 10 cycles.

b. A visual inspection of a fire barrier penetration after repair or maintenance, prior to restoring barrier penetration to functional status.

Fire dampers shall be verified functional by:

a. A visual inspection of a sample of one-third of the fire dampers once per operating cycle. If any failures are identified in this sample, an additional one-third sample shall be inspected during that operating cycle. If any failures are identified in the second sample, then all of the fire dampers shall be inspected during that operating cycle.

b. A visual inspection of the fire damper after repair or maintenance, prior to restoring the fire damper to functional status.

2.4.2 Control of Combustibles

2.4.2.1 In Situ Combustibles

Safety-related systems at Unit 1 are protected from in situ combustibles by any one or a combination of the following methods:

a. Fire rated barriers.

b. Automatic fire suppression and detection systems.

c. Spatial separation between the combustible material and the identified equipment.

d. Engineered design provisions to limit potential exposure.

Allowance for transient combustibles, which may increase to the total combustible loading of the area, is included in the FHA Summary Tables (reference Tables 3.1.1-1 to 3.1.1-9).

2.4.2.2 Bulk Gas Storage

Bulk gas storage is not permitted within structures housing safety-related equipment. Bulk hydrogen and nitrogen storage tanks are located outside with their long axes parallel to the turbine building. However, the hydrogen and nitrogen storage tanks are perpendicular to the west wall of the reactor building (reference Section 3.11.1).

The use of compressed gasses inside site structure is controlled.

2.4.2.3 Plastic Materials

Originally-installed cables are largely polyvinyl chloride (PVC) jacketed. 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. Other requirements of cables and cable trays are discussed in Section 2.4.3. The use of plastic materials in construction for permanent plant facilities is minimized.

2.4.2.4 Flammable Liquids

Flammable liquids are stored in accordance with NFPA 30, Flammable and Combustible Liquids Code. Fire suppression and/or detection systems are provided for identified storage areas.

Generation administrative procedures control the use and storage of flammable and combustible liquids outside the bulk storage areas.

2.4.3 Electric Cable Construction, Cable Trays and Penetrations

2.4.3.1 Cable Trays

Noncombustible materials are used in the construction of cable trays.

2.4.3.2 Cable Spreading Rooms

This room is protected by total-flooding CO_2, preaction sprinkler and smoke detection systems. Manual fire hose stations and portable extinguishers have been provided for this area. See Section 2.6.3 for detailed discussion.

2.4.3.3 Sprinkler Protection

Automatic preaction sprinkler systems are installed to protect open, safety-related cable trays which are stacked more than two trays deep. Early-warning smoke detection is provided to facilitate system operation. Manually-operated hose stations are provided in the vicinity of the protected cable trays. Where identified, safety-related equipment in the vicinity of such cable trays has been protected if damage may occur from sprinkler operation. Specific design requirements of RG 1.75 are not all satisfied. The application of fire-retardant coatings to safety-related cable trays has been limited to those occurrences where sprinkler protection may not be the most desirable means of protection due to the equipment location in the area (i.e., over safety-related power boards). This coating is used primarily to prevent ignition and limit propagation of fire in the application areas. New cables installed in these trays shall be protected by engineering design in lieu of the application of fire-retardant coatings.

Based on the identified design provisions, the level of protection provided should prevent significant fire propagation and assist in cable tray suppression activities.

2.4.3.4 Cable Penetrations

Fire barrier penetrations use approved penetration seal details. The subject configurations have been tested to establish a 3-hr fire rating.

2.4.3.5 Fire Breaks

Fire breaks are provided in vertical cable trays which pass through nonrated floor/ceiling assemblies to limit the vertical propagation of fire along the tray through the building. If required by evaluation of modification activities, fire stops may also be placed in horizontal cable trays to limit horizontal propagation of fire in lieu of coating/recoating cable with a fire-retardant coating.

2.4.3.6 Cable Construction

Originally-installed cable construction does not comply with the requirements of the IEEE-383 flame test. 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. Protection for existing cable trays which contain nonqualified cable is discussed in Section 2.4.3.3 above.

2.4.3.7 Cable Decomposition

To the extent possible, new cable installations meet the requirements of IEEE-383. Selection of cable in this manner should minimize the installation of cable which may generate corrosive gasses during combustion.

2.4.3.8 Cable Run Exclusions

Only cable is permitted in cable trays or conduits. Cables are not installed in floor trenches or culverts. Miscellaneous storage is prohibited in cable trays, in addition to piping for combustible or flammable liquids or gasses.

2.4.3.9 Cable Tunnel Design

Unit 1 does not utilize cable tunnels and culverts. The cable spreading room is provided with venting capability. This is discussed in Section 2.4.4.

2.4.3.10 Control Room Cables

Cables in the control room are kept to the minimum necessary for operation. Cables entering the control room terminate there.

There is not a concealed floor in the control room or the auxiliary control room.

2.4.4 Ventilation

2.4.4.1 Products of Combustion Removal

All safety-related areas use the installed once-through ventilation to remove products of combustion.

Return air is monitored by the stack monitor prior to release by the stack to determine if the release is within the permissible limits of radioactivity to prevent an unacceptable release to the atmosphere during smoke removal operations.

Dedicated smoke removal systems have been designed for six areas of the turbine building. These areas include:

a. Smoke Zone 1 - TB 250 East

b. Smoke Zone 2 - TB 250 South/West

c. Smoke Zone 3 - TB 250 North

d. Smoke Zone 4 - CC 250 Cable Spread Room

e. Smoke Zone 5 - CC 261 Aux. Control Room

f. Smoke Zone 6 - CC 277 Control Room

These systems have sufficient smoke exhaust fans, isolation dampers, and controls dedicated to smoke removal. Portions of the normal ventilation system components are used in certain areas for smoke removal. Fire dampers are provided where ventilation ductwork penetrates fire barriers. Normally, when the temperature of the fusible element is reached, the fire damper(s) would close. These dampers can be manually reopened by plant personnel at the main or local fire alarm panel for smoke removal, provided ductwork high temperature conditions do not exist (setpoint higher than fusible elements). Details of operation of these systems are described in FSAR Sections III.A.2.3 and III.B.2.3.

Additional heat removal capability is supplied by roof-mounted heat vents installed in the turbine building. Portable fans would be used to further aid in the removal of smoke.

2.4.4.2 Ventilation System Design

The inadvertent operation or single failures of ventilation systems designed to exhaust smoke and/or corrosive gasses will not violate controlled areas of the plant. Direct readings obtained from stack gas monitors, area radiation monitors and/or radiation protection personnel during fire development will be available to control room personnel. These readings will aid control room personnel in the proper implementation of plant operating and emergency planning procedures for protection of the public and maintaining habitability for operations personnel.

2.4.4.3 Power Supplies/Controls

The power supply and controls for the dedicated smoke removal systems for the turbine building area have been removed from the affected area to the maximum extent possible. The control and power supplies are not expected to be damaged by fires within the areas they serve.

<u>Action</u>

With one or more of the above-required Halon 1301 systems inoperable, within 1 hr implement one of the following actions:

a. Verify the operability of fire detectors within the area protected by the system and establish a daily inspection of the area to verify no increase in fire hazards, or

b. Establish a continuous fire watch with backup suppression equipment, or

c. Implement a preplanned provision(s) in accordance with the assessment of a qualified FPE.

2.5.4.1.1 Surveillance

Each of the required Halon systems shall be demonstrated operable:

a. At least once per 12 months by verifying that each valve (manual, power operated or automatic) in the flow path is in its correct position.

b. At least once per 6 months by verifying Halon storage tank weight (level) and pressure.

c. At least once per 18 months by verifying the system and associated ventilation dampers and fire door release mechanisms actuate manually and automatically.

2.5.4.2 System Maintenance

The systems are periodically inspected and tested in accordance with NFPA 12A.

2.5.4.3 System Design Considerations

During the system pre-discharge period, prior to agent release, local audible and visual alarms are provided in the protected area for personnel notification purposes. In addition, the auxiliary control room and the ECIV room are provided with a glass flask of wintergreen concentrate attached to the discharge piping to add a distinctly identifiable scent to the discharge gas. This flask ruptures upon operation of the system and must be replaced after each operation.

2.5.5 Carbon Dioxide (CO_2) Suppression System

Fire extinguishment by CO_2 is either by the total-flooding or local application method. In total-flooding, sufficient CO_2 is injected into a closed room or space to inert the atmosphere and suppress combustion. Local application is employed for unenclosed hazards and involves application of CO_2 on the equipment protected to extinguish the fire, with additional discharge to permit cooling and inhibit reflash.

Unit 1 automatic CO_2 fire suppression systems have been temporarily placed in alarm-only mode due to life safety concerns until modifications to improve personnel safety are completed.

2.5.5.1 Carbon Dioxide System Design*

Total-flooding and local application CO_2 systems are installed to protect several different hazards in the plant. Automatic protection is provided for the following hazards:

a. Turbine Oil Tank Room - total-flooding; automatic actuation by rate-compensated thermal detectors.

b. Motor Generator Sets - local application to all five units simultaneously; actuated by rate-compensated thermal detectors located over each unit.

c. Power Boards 102 and 103 - total-flooding; actuation by cross-zoned smoke detectors.

d. Diesel Generator 102 and 103 - total-flooding; actuation by cross-zoned smoke, flame and thermal detectors.

e. Hydrogen Seal Oil Enclosure - total-flooding; actuation by rate-compensated thermal detectors.

f. Turbine Oil Reservoir Room - total-flooding; actuation by rate-compensated thermal detectors.

g. Cable Spreading Room - total-flooding; detection by cross-zoned smoke detectors.

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* Automatic CO_2 fire suppression systems are temporarily in alarm-only mode. See Section 2.5.5.

shows that the fire protection system will provide adequate ability to detect, prevent, and suppress postulated fire outbreaks in and around the plant. The attached FHA figures further define the extent of fire protection features (e.g., fire detection system design which has been incorporated into the Unit 1 design).

3.2 REACTOR BUILDING

3.2.1 Introduction

The reactor building is considered three fire areas (FA1, FA2, and FA3) due to the presence of an unprotected floor hoistway and stairway. It is subdivided into three fire areas for the purpose of maintaining separation of redundant safe shutdown equipment. Therefore, the east and west halves of the reactor building and the primary containment are each considered a separate fire area (see Table 3.2-1).

Fire areas FA1 and FA2 are spatially separated by an identified 20-ft fire break zone with installed smoke detection and automatic suppression systems, except as noted in the Unit 1 SSA.

The building has essentially six main floor levels, with two additional equipment levels below el 237'-0". The exterior walls below grade and up to the refueling floor are poured concrete. Those exterior walls above the refueling floor are metal panel construction.

To reduce the potential for the vertical spread of fire along cable tray risers, cable tray openings are firestopped at the floor.

Rated barriers provide for protection of equipment from exposure hazards or enclose identified hazards in the reactor building as follows:

a. Reactor building track bay

b. ECIV room

Walls of the reactor building that are common to other buildings are 3-hr rated, with the exception of the wall common to the turbine building above the operating floor. In this application, a 2-hr rated assembly protects the reactor building from a turbine building exposure fire.

Rated barriers are also provided to enclose the southeast stairwell from el 237'-0" to 340'-0".

The primary containment is also considered a fire area. However, due to the low fire loading, primary containment inerting, and lack of continuity of combustibles, a sustained fire capable of spreading within this area cannot be supported during normal operation.

3.2.2 Safety-Related Systems

The reactor building contains numerous redundant shutdown components and cabling within the identified fire area. However, these redundant components have been spatially separated and located within the identified fire areas. Loss of equipment and cabling in one fire area will not impact the ability to safely shut down the plant in accordance with the provisions of 10CFR50, Appendix R.

3.2.3 Post-Fire Analysis

A fire in a reactor building fire area will not result in loss of capability to achieve safe shutdown. If the installed fire protection systems for protection of equipment and hazards were in service, the fire should be contained within the general area of origin and be extinguished by automatic and/or manual means.

3.2.4 Radioactive Release Analysis

The reactor building has a ventilation system utilizing 100 percent outside supply air for normal operation. This system will remove smoke in the event of fire, as long as airborne radiation leaks remain below acceptable levels. The exhaust air from this system is monitored to determine if radiation levels exceed preset limits. In the event of fire, the reactor building emergency ventilation system may be used, or the entire system could be shut down should these limits be exceeded.

A fire in the reactor building is not expected to result in excessive leakage of airborne radioactivity. The acceptance criteria of 10CFR50.67 would not be exceeded.

3.2.5 Fire Detection/Suppression Systems

Early warning general area and spot smoke detection systems are provided for the reactor building to initiate alarm conditions primarily for protection of safety-related equipment and identified hazards within the structure. These zoned detection systems provide alarms locally (LFCP) and in the control room.

Preaction sprinkler systems primarily provide protection of the fire break zones and select cable tray stacks in this area (reference Section 2.4.3.3). In addition, preaction sprinkler systems have been provided in select material storage areas located within the reactor building.

A dry-pipe sprinkler system is provided for the reactor building track bay.

An automatic total-flooding Halon 1301 extinguishing system has been provided for the ECIV room.

Manual water hose stations and portable extinguishers are provided for manual suppression activities for this building.

3.3 TURBINE BUILDING

3.3.1 Introduction

The turbine building is divided primarily into 8 fire areas (see Table 3.3-1). These areas are:

FA 5 - TB 261 and above
FA 6 - TB 250 North
FA 7 - TB 250 West
FA 9 - TB 250 East
FA 16A - Battery Board Room 12 - TB 261
FA 16B - Battery Board Room 11 - TB 261
FA 17A - Battery Room 12 - TB 277
FA 17B - Battery Room 11 - TB 277

The building has essentially four levels, with a number of partial floors at various locations within this structure. In general, unprotected floor openings, open hoistways and stairwells necessitate consideration of the area above el 261'-0" as one fire area. It is not feasible to seal or enclose these openings. The exterior walls below grade are concrete, and those above grade are metal panel or precast concrete panel construction.

Rated barriers have been provided for protection of equipment from exposure hazards or to enclose identified hazards in the following identified primary fire areas.

UPS Security Battery Room	(FA7)
Battery Board Room 12	(FA16A)
Battery Board Room 11	(FA16B)
Chemistry Laboratory	(FA5)
Equipment Decon Area	(FA5)

Oil Storage Room	(FA5)
Turbine Oil Reservoir Room	(FA5)
Hydrogen Seal Oil Unit Room	(FA5)
Battery Room 11	(FA17A)
Battery Room 12	(FA17B)
Mechanical Storage Area	(FA5)
Turbine Oil Storage Area	(FA5)
Battery Room 14	(FA5)

Walls of the turbine building that are common to other buildings are 3-hr fire rated.

The turbine generator and condenser area are considered part of FA5. Walls separating this area from the remainder of the

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turbine building are of substantial poured concrete construction below el 300'-0". All penetrations below el 261'-0" have been evaluated and sealed as required to protect safety-related equipment from a turbine generator area exposure fire. In general, penetrations in the west, south and east walls are sealed with 3-hr rated configurations.

Two stairwells have been designed as egress paths and have been provided with 2-hr rated enclosures.

Due to the contents of the turbine oil storage and turbine oil reservoir room, the doorways are elevated to contain the oil in the event of tank rupture.

Fusible link-actuated heat vents are provided in the turbine building roof. Fusible link settings are specified at a temperature which should preclude operation due to a steam leak. These vents are provided to reduce the possibility of roof collapse and structural fatigue under high temperature conditions for a fire in the structure. In addition, remote manual releases are provided for these vents to aid in the removal of smoke from this area should the conditions warrant this operation.

An engineered smoke removal system has been provided for el 250'-0" of the turbine building to assist in the removal of smoke from this area should conditions warrant (reference also Section 2.4.4.1).

3.3.2 Safety-Related Systems

The turbine building contains numerous shutdown components and cabling within the identified fire areas; however, loss of any one of the primary fire areas will not impact the ability to safely shut down the plant in accordance with provisions of 10CFR50, Appendix R.

3.3.3 Post-Fire Analysis

A fire in the turbine building will not result in loss of capability for safe shutdown. If the installed fire protection systems for protection of equipment and hazards were in service, the fire should be contained within the general area of origin and be extinguished by automatic and/or manual means.

3.3.4 Radioactive Release Analysis

The turbine building ventilation system consists of a supply and exhaust air system. During normal conditions, the opening of a small number of heat vents will not result in a release of radioactivity due to the negative pressure imparted by the subject ventilation system. Moderate smoke generation would also be handled by the normal building ventilation system. In the event of a fire which would necessitate multiple operation of

3.5.2 Safety-Related Systems

The diesel generator building contains Division 11 and 12 diesel generators and support power boards. These diesel generators provide power to essential equipment should normal Station service power be lost.

3.5.3 Post-Fire Analysis

A fire in one of the divisional diesel generator or power board rooms will not result in loss of capability for safe shutdown. If the installed fire protection systems for the protection of equipment were in service, the fire should be contained within the general area of origin and be extinguished by automatic and/or manual means.

3.5.4 Radioactive Release Analysis

There is no source of radioactivity within this building; however, this structure is located in a Radiologically-controlled area (RCA). Should conditions exist where evacuation of gaseous products from these areas would be required, the acceptance criteria of 10CFR50.67 would not be exceeded.

3.5.5 Fire Detection/Suppression Systems

Early warning general area smoke detection systems are provided for each area of the diesel generator buildings. These zoned detection systems provide alarms locally (LFCP) and in the control room.

Automatic total-flooding CO_2 systems are provided for the diesel generator general areas and power board rooms.*

An automatic preaction sprinkler system provides area suppression capability for the DG 250 areas.

Manual water hose reels and portable fire extinguishers provide backup fire suppression capability for this structure.

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* Automatic CO_2 fire suppression systems are temporarily in alarm-only mode. See Section 2.5.5.

3.6 SCREENHOUSE

3.6.1 Introduction

The screenhouse is primarily divided into two fire areas and is located adjacent to and north of the turbine building extension (see Table 3.6-1).

3.7.3 Post-Fire Analysis

There is no equipment in this building required for safe shutdown of the plant. However, if the installed fire protection systems located in this area were in service, the fire should be contained within the general area of origin and be extinguished by automatic and/or manual means, as applicable.

3.7.4 Radioactive Release Analysis

The waste building ventilation system consists of one supply air system and two exhaust air systems.

Smoke generation within this structure would be handled by the normal building ventilation system. The exhaust from this system is monitored to determine if radiation levels exceed preset limits. In the event of fire, the ventilation system may be subsequently shut down should these limits be exceeded.

A fire in the waste building would not result in excessive leakage of airborne radioactivity. The acceptance criteria of 10CFR50.67 would not be exceeded.

3.7.5 Fire Detection/Suppression Systems

Early-warning general area and spot smoke detection systems are provided for the waste building. These zoned detection systems provide alarms locally (LFCP) and in the control room.

A wet-pipe sprinkler system has been provided for the waste compactor area and baler room.

A timed preaction sprinkler system provides protection of the Dow solidification process room. Installed heat detectors initiate the sprinkler valve actuation. The system is equipped with a 10-min timer. If this is not adequate, the time can be overridden and the deluge system initiated again.

A dry-pipe sprinkler system is provided for the waste building truck bay.

Manual water hose stations and portable extinguishers are provided for manual suppression activities for this area.

3.8 OFFGAS BUILDING

3.8.1 Introduction

The offgas building is primarily considered one fire zone and is part of FA5, located adjacent to the waste building, turbine building and diesel generator building (see Table 3.8-1).

The building has essentially three levels. In general, unprotected floor openings and an open stairwell necessitate consideration of this building as one fire zone. The exterior walls below grade are concrete, and those above grade are metal panel construction.

Walls of the offgas building that are common to other buildings are 3-hr fire rated.

Access to and egress from the lower levels of the offgas building is made through a single, open stairwell located in the center of the building. These lower levels are normally unoccupied.

3.8.2 Safety-Related Equipment

In general, the offgas building does not contain any safety-related equipment. However, a safety-related cable supplying normal power to power board 103 is routed through a cable tray in this building. The potential of this cable on Appendix R has been determined. As a result, DRP actions are provided to isolate this cable from the safety-related power board 103.

3.8.3 Post-Fire Analysis

A fire in the offgas building will not result in loss of capability to achieve safe shutdown. If the installed fire protection systems located in this area were in service, the fire should be contained within the general area of origin and be extinguished by automatic and/or manual means, as applicable.

3.8.4 Radioactive Release Analysis

The offgas building ventilation system consists of a single exhaust air system. Supply air is provided from the turbine building supply air system.

Smoke generation within this building would be handled by the normal building ventilation system. The exhaust air from this system is monitored to determine if radiation levels exceed preset limits. In addition, the turbine building ventilation system would also assist in the removal of smoke from this area.

In the event of fire, the ventilation system may be subsequently shut down should these limits be exceeded.

A fire in the offgas building would not result in excessive leakage of airborne radioactivity. The acceptance criteria of 10CFR50.67 would not be exceeded.

3.8.5 Fire Detection/Suppression Systems

Early-warning general area and spot smoke detection systems are provided for the offgas building. These zoned detection systems provide alarms locally (LFCP) and in the control room.

A wet-pipe sprinkler system has been provided for the shower facility and mechanical equipment storage area.

Manual water hose stations and portable fire extinguishers are provided for manual suppression activities for this area.

3.9 RADWASTE SOLIDIFICATION AND STORAGE BUILDING

3.9.1 Introduction

The RSSB is primarily considered one fire area (FA15) and is located adjacent to the waste building (see Table 3.9-1). For the Appendix R analysis, the waste building is also considered as part of FA15, although a rated barrier separates the structures.

The east portion of this structure is primarily utilized for waste storage. The remainder of the structure is utilized for support or radwaste solidification activities. The building has essentially three main elevations outside the waste storage vaults. Due to the design and use of this structure with partial-height walls and unprotected floor openings, it is considered one fire area. The exterior walls of this entire structure are poured concrete.

Rated barriers or thermal shield walls have been provided for protection of the following identified areas in this building:

a. Electrical Equipment Room

b. RSSB Control Room

c. North Truck Loading Room

d. West Truck Loading Room

e. Waste Storage Vaults

Walls of the RSSB that are common to other buildings are 3-hr fire rated.

Two remote stairwells provide egress from the access to the main elevations of this structure. These stairwells are provided with 2-hr rated enclosures.

3.9.2 Safety-Related Systems

The RSSB does not contain any safety-related equipment.

3.9.3 Post-Fire Analysis

There is no equipment in this building required for safe shutdown of the plant. If the installed fire protection systems located in this building were in service, the fire should be contained within the general area of origin and extinguished by automatic and/or manual means, if applicable.

3.9.4 Radioactive Release Analysis

Two separate ventilation systems are provided for the RSSB. One system is a supply and exhaust system provided for areas which would normally be occupied by personnel. Smoke generation within these areas would be handled by the normal building ventilation system. The exhaust from this system is monitored to determine if radiation levels exceed preset limits. In the event of fire, the ventilation system may be subsequently shut down should these limits be exceeded.

Ventilation for normally-unoccupied areas of the building is provided with a recirculating atmosphere cleanup system. This system would assist in the cleanup of smoke conditions in the RSSB. A portion of the supply air to certain select areas is exhausted through the ventilation exhaust system described for normally-occupied areas, which is monitored to determine if radiation levels exceed preset limits.

A fire in the RSSB would not result in excessive leakage of airborne radioactivity. The acceptance criteria of 10CFR50.67 would not be exceeded.

3.9.5 Fire Detection/Suppression Systems

Early-warning general area smoke detection systems are provided for select areas of this structure. The HVAC exhaust system charcoal filter is equipped with thermal fixed-rate compensated detectors. These zoned detection systems provide alarms locally (LFCP) and in the control room.

A dry-pipe sprinkler system provides protection of the access corridor to the truck loading docks, west truck loading dock and north truck loading dock.

A manually-operated water spray system provides protection of the HVAC exhaust system charcoal filter.

Protection of the waste storage vaults is provided by manual means. A fire in the north vault will be fought by the Fire Brigade through the crane access area on el 281'-0". A fire in the south vault will be fought by the placement of manually-controlled water monitor nozzles directed over the shield wall of the vault.

Separate automatic Halon 1301 fire suppression systems provide protection for the electrical equipment room and RSSB control room.

Manual water hose stations and portable extinguishers are provided for manual suppression activities for this area.

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3.10 ADMINISTRATION BUILDING

3.10.1 Introduction

The administration building is divided primarily into six areas. To satisfy the Appendix R analysis, these areas are grouped into two fire areas (see Table 3.10-1). These fire areas are:

FA12 - New AB 248
FA12 - Old AB 250/261
FA12 - New AB 261
FA12 - New AB 277
FA12 - Old AB 277
FA4 - Foam Room

The building has essentially three levels. In general, floor openings between adjacent elevations are sealed with comparably-rated sealing configurations. The exterior walls below grade are concrete, and those above grade are metal panel or precast concrete panel construction.

Rated fire barriers have been provided for protection of equipment from exposure hazards or enclosed identified hazards in the following primary fire areas.

Technical Support Center	(FA12)
Radiation Records Processing Area	(FA12)
NMP1/NMP2 Access Tunnel	(FA12)
Secondary Alarm Sys (SAS) Cont Area	(FA12)
SAS Computer Room	(FA12)
Foam Room	(FA4)
Storeroom Truck Bay	(FA12)
Storeroom Oil Storage Area	(FA12)
Warehouse	(FA12)
Telephone Equipment Room 1	(FA12)
Telephone Equipment Room 2	(FA12)

Walls of the administration building that are common to other buildings are 3-hr fire rated, with the exception of stairtowers, elevator shafts, and the wall (El 261') separating the men's locker room and the electrical shop, which are provided with at least a 2-hr rated enclosure. The foam room is separated from the rest of the administration building and other areas of the plant with 3-hr rated fire barriers.

Remotely-located stairtowers provide adequate egress and Fire Brigade access to the different areas of this structure. The stairtowers located in the eastern portion of the administration building are provided with 2-hr rated enclosures.

3.10.2 Safety-Related Systems

In general, the administration building does not contain any safety-related equipment. However, a safety-related dc power board is located in the foam room. Equipment redundant to this power board exists in other areas of the plant. Therefore, loss of equipment and/or cabling in this area will not impact the ability to safely shut down the plant in accordance with the provisions of 10CFR50, Appendix R.

3.10.3 Post-Fire Analysis

A fire in the foam room or other areas of the administration building will not result in loss of capability to achieve safe shutdown. If the installed fire protection systems located within this building were in service, the fire should be contained within the general area of origin and be extinguished by automatic and/or manual means, as applicable.

3.10.4 Radioactive Release Analysis

There is no source of radioactivity in this building.

3.10.5 Fire Detection/Suppression Systems

Early-warning smoke detection systems have been provided for select areas of this structure. The new TSC emergency ventilation system is equipped with duct-type smoke detectors (see FSAR Figure III-18). These zoned detection systems provide alarms locally (LFCPs) and in the control room.

The administration building is protected by wet-pipe sprinkler systems, with the exception of certain select areas as follows.

The offices located on the north side of el 277'-0" are protected by a preaction sprinkler system.

A dry-pipe sprinkler system protects the storeroom truck dock.

A manually-operated water spray system provides protection of the old TSC emergency ventilation system charcoal filter, located on the roof of the administration building.

Automatic total-flooding Halon 1301 fire suppression systems are provided for the two telephone switch rooms, CPU/electrical area and SAS computer area.

Manual hose stations and portable extinguishers are provided for manual suppression activities in this building.

TABLE 3.3-1

FIRE AREA/ZONE SUMMARY

Turbine Building

FIRE AREA	FIRE ZONE	DESCRIPTION	ELEV
FA5	T1	Turbine Condenser/Heater Bay Area	250
FA5	T3A	General Floor Area East of MSIV Room and Fire Zone T1	261
FA5	T3B	General Floor Area West of MSIV Room; also South and West of Fire Zone T1	261/237
FA16A	B1A(1)	Battery Board Room 12	261
FA16B	B1B(1)	Battery Board Room 11	261
FA5	T4A	General Floor Area East of Fire Zone T1	277
FA5	T4B	General Floor Area West of Fire Zone T1	277
FA5	T4C(2)	Hydrogen Seal Oil Unit Room	277
FA5	T4D	Battery Room 14	277
FA17A	B2A(1)	Battery Room 12	277
FA17B	B2B(1)	Battery Room 11	277
FA5	T5A	General Floor Area, North	291
FA5	T6A	General Floor Area, North	305
FA5	T6B	Turbine Laydown Area, East	300
FA5	T6C	General Floor Area, South	300
FA5	T6D(2)	Mechanical Storage Area	320

Table 3.3-1 (Cont'd.)

FIRE AREA	FIRE ZONE	DESCRIPTION	ELEV
FA5	T7A	General Floor Area, South	320
FA5	T8A	General Floor Area, North General Floor Area, North General Floor Area, East	333 351 369
FA5	T8B	General Floor Area, West	369
FA5	T1A[2]	MSIV Room and Steam Tunnel	240
FA6	T2A	General Floor Area, North	250
FA9	T2C	Offgas Tunnel	250
FA9	T2D	General Floor Area, East	250
FA7	T2B	General Floor Area, West	250
FA7	T2E[2]	UPS Battery Room	250

(1) Separation of the battery and battery board rooms 11 and 12 is supported by an NRC exemption and an engineering evaluation.

(2) Fire zone partially or fully separated from adjacent fire zones by rated fire barriers within this fire area.

TABLE 3.1.1-2 (Cont'd.)

ZONE		FIRE HAZARD								FIRE PROTECTION	
FIRE AREA/ ZONE	NAME	COMBUSTIBLE MATERIAL	QUANTITY		CALORIC VALUE (BTU/lbm)	TOTAL BTUs	SQUARE FEET	FIRE LOAD		DETECTION H OR S	EXTINGUISHING SYSTEMS
			GALLONS	POUNDS				BTU/FT^2	TIME (HRS)		
FA5/ T4D	TB 277 Battery Room 14	Styrene Plastic Transient Load		4,500 25	18,000 20,000 24,000 BTU/ft^2	81,000,000 500,000 17,760,000 99,260,000	740	134,135	1.68	Smoke	
FA 17A/ B2A	TB 277 Battery Room 12	Styrene Motor Insulation Wire Insulation Plastic Transient Load		3,064 1 7 50	18,000 10,000 20,000 20,000 24,000 BTU/ft^2	55,152,000 10,000 144,000 1,000,000 15,360,000 71,666,000	640	111,978	1.40	Smoke	
FA 17B/ B2B	TB 277 Battery Room 11	Styrene Wire Insulation Motor Insulation Plastic Transient Load		3,064 6 1 50	18,000 20,000 10,000 20,000 24,000 BTU/ft^2	55,152,000 120,000 10,000 1,000,000 15,360,000 71,642,000	640	111,941	1.40	Smoke	
FA5/ T5A	TB 291 Gen. Floor Area - North	Cable Insulation Class A Motor Insulation Rubber FL Liquids Grease Fiberglass Wire Insulation Plastic Storage Area Transient Load	 210 60	20,725 4,324 1 700 1,956 528 170 494 1,144	13,500 8,000 10,000 19,000 20,000 18,000 18,000 20,000 20,000 100,000 BTU/ft^2 24,000 BTU/ft^2	279,787,500 34,592,000 10,000 13,300,000 39,120,000 9,504,000 3,060,000 9,880,000 22,880,000 319,200,000 471,216,000 1,202,549,500	19,634	61,248	0.77	Smoke	Sprinklers

APPENDIX A (Cont'd.)

Fire Area	Fire Zone	Location	Fire Detection Zone(s)
12	AB1A, AB1B, AB1C, AB1D, AB1E, AB2A, AB2B, AB2C, AB2D, AB3A, AB3B, AB3C, AB3D, AB3E, AB3F, AB4A, AB4B, AB4C, AB4D, AB5	Admin. Bldg., Ships/Stores Admin. Bldg., Addition	-- --
13	S1	Screenhouse	D-5013
14	S2	Diesel Fire Pump Room	D-5023
15	WD1	Waste Disposal Building and Radwaste Solidification and Storage Building	--
16A	B1A	Battery Board Room 12	DA-2161E
16B	B1B	Battery Board Room 11	DA-2161E
17A	B2A	Battery Room 12	D-2224
17B	B2B	Battery Room 11	D-2224
18	D3	DG 102 Missile Shield	D-2151
19	D1A D2A	DG 103 Foundation DG 103 Room	DA-2041S DX-2151A, DX-2151B, DA-2151
20	D1C	DG 103 Cableway	DA-2041N
21	D1D	Area Under PB 102/PB 103	DA-2041N

APPENDIX A (Cont'd.)

Fire Area	Fire Zone	Location	Fire Detection Zone(s)
22	D1B D2B	DG 102 Foundation DG 102 Room	DA-2041N DX-2141A, DX-2141, DA-2141
23	D2C	PB 102 Room	DX-2123A, DX-2123B
24	D2D	PB 103 Room	DX-2113A, DX-2113B

hazards. In addition, smoke detectors and an automatic fire suppression system are installed in the zone unless exempted. Exposed cables in the FBZs have been coated with a flame-retardant material. Cables in conduits or cable trays covered by suppression systems are also considered as nonintervening combustibles. Penetrations in the FBZs are sealed with a minimum 1-hr fire-rated assembly to provide for overlap with the FBZs located above and below. The following is a list of the FBZs and their location:

Fire Break Zone	Location
R237N	El 237'-0" between columns N to Q and rows 8 to 9
R261N	El 261'-0" between columns N to Q and rows 8 to 9
R281N	El 281'-0" between columns M to Q and rows 6 and 7
R281S	El 281'-0" between columns K to L and rows 7 to 8
R298N	El 298'-0" between columns N to Q and rows 7.5 to 8.5
R298S	El 298'-0" between columns K to L and rows 7 to 8
R318N	El 318'-0" between columns M to Q and rows 6 to 7
R318S	El 318'-0" between columns K to M and rows 6 to 7
R340N	El 340'-0" between columns M to Q and rows 6 to 7
R340S	El 340'-0" between columns L to N and rows 7 to 8 and also includes between columns K to N and rows 7.5 to 8.5

b. <u>Rerouting of DG 102 Cooling Water Pump Power Cable (Safety Evaluation 83-05)</u>

The power cable for DG 102 cooling water pump was rerouted out of the east side of the reactor building (FSA 1) to assure the availability of DG 102 or 103 following a fire in a reactor building fire area.

c. <u>Diesel Fire Pump Room Upgrade and Emergency Spool Piece (Safety Evaluations 83-01 and 83-08)</u>

The diesel fire pump room has been upgraded to meet the requirements of Appendix R, Section III.G.2, for separation of the diesel fire pump room from the screenhouse. Detectors and automatic suppression have been installed above the pump room roof. The 4-in curb above the west wall has been extended to connect with the curb above the north wall, thus preventing runoff. Structural steel inside the pump room has been protected by application of a 3-hr rated fire barrier material.

Provisions have also been made for installation of emergency spool pieces, which make it possible for the diesel fire pump to supply water to the ESW system and/or either of the diesel generator cooling water systems. These modifications allow for use of a Station diesel generator and the ESW system following a fire in the screenhouse (FA 13) which could disable all other screenhouse pumps.

d. <u>Diesel Generator Alternate Dc Power Supply (Safety Evaluation 83-07)</u>

A normally open, nonautomatic, circuit breaker has been installed on each alternate 125-V dc power cable between the two diesel generators. As a result, the alternate dc power supply to each diesel generator is normally de-energized. This modification represents an improvement in the cold shutdown capability by preventing the possible loss of dc power to both diesel generators due to a fire in one diesel generator room.

e. <u>Diesel Generator 102 Power Cable Protection Modification (Safety Evaluation 83-02)</u>

The DG 102 output cable to PB 102 was installed in exposed conduit next to the DG 103 output cable to PB 103 in the DG enclosed cableway (FA 20). In addition, the normal supply cable from PB 101 to PB 102 was installed in exposed conduit in this same area. This could have resulted in the loss of both diesel generators following a fire in this area. The modification provided an equivalent 1-hr rated fire barrier wrap around the DG 102 output cable conduit and the PB 102 normal supply conduit, and addition of detection and automatic suppression in the area. This separation of the 102 feeder cable is considered equivalent to rerouting the cable, which was originally required by the NRC Fire Protection Safety Evaluation Report dated July 26, 1979. Therefore, the modification brings the area into compliance with the separation requirements of Appendix R, Section III.G.2. One of these cables is required following a fire for cold shutdown capability.

f. Diesel Generator 103 Control Cable Protection Modification (Safety Evaluations 83-02 and 83-13)

Control cables for DG 103 were installed in exposed conduit next to control cables for DG 102 in the DG 102 missile enclosure (FA 18). This could have resulted in the loss of both diesel generators and automatic cold shutdown capability following a fire in this area. This modification provided a 3-hr rated fire barrier around the conduits associated with DG 103. The modification, therefore, brings the area into compliance with the separation requirements of Appendix R, Section III.G.2.

2. Appendix R, Section III.G.3 Modifications

a. Automatic Depressurization System Logic Modification (Safety Evaluation 84-18)

The preferred method for achieving hot shutdown is via the ECs. This modification was required to prevent possible spurious actuation of the ADS, which would have resulted in a reactor coolant inventory loss and the loss of reactor vessel pressure necessary to sustain natural circulation, essentially negating the effectiveness of the ECs as a hot shutdown system. The ADS logic was initially de-energized to activate on the ac portion of the circuit. Therefore, upon the loss of the two RPS MG Sets 162 and 172 (existing configuration prior to installation of the UPSs by Modification N1-88-091), the system would activate provided power was available on either PB 102 or 103 and dc power was also available. Also, hot shorts to the valve logic itself could actuate the various valves. This modification provided additional energizing to activate logic located in the reactor building. This new logic is operated by the same auxiliary relays as the principle logic and is referred to as "confirmatory" logic because it takes both the original and this additional confirmatory logic to activate the system. This modification also resolved the problem of possible hot shorts to an individual valve's control logic, using coil to contact isolation relays and shunted coils.

b. Core Spray Inboard Discharge Valves Logic Modification (Safety Evaluation 84-24)

The four inboard discharge valves were electrically interlocked with both the core spray outboard discharge isolation valves and the core spray isolation test valves to torus. These electrical interlocks were all associated with PB 167, whose cables are located in one of the newly-designated FBZs in the reactor building. This could have resulted in the loss of all four of the inboard discharge valves from a fire in either the east or west side of the reactor building (FSA 1 or FSA 2) since the FBZ must be considered lost for a fire on either the east or the west side of the reactor building. This modification isolated these interlocks and provided redundant initiating logic, assuring a minimum operation of one inboard discharge valve following a fire anywhere in the reactor building.

c. Head Vent Valve Logic Modification (Safety Evaluation 83-33)

Hot shorts to the valve control logic could have spuriously operated the valve, resulting in an inventory loss path. Electrical lock-out of the motor control breaker was not an acceptable resolution for this vent path, since Operations wanted use of this valve for events where venting of the reactor vessel is required. Therefore, the modification used coil to contact isolation relays and shunted coils for circuits not in use as a solution to the spurious activation problem.

d. Emergency Condenser High Radiation Isolation Logic Removal (Safety Evaluation 84-57)

The ECs would initially isolate on a high steam flow signal or a high radiation signal. Hot shorts to the radiation monitors or the isolation logic itself could have resulted in the spurious isolation of the ECs. This modification resulted in the removal of the high radiation portion of the logic, making the high radiation signal provide annunciation only. This modification required a change to the Technical Specifications. Isolation of the ECs due to high radiation is now manual only.

e. Remote Shutdown Panel Isolation Modification (Safety Evaluation 84-26)

The RSPs were installed to provide alternate shutdown stations following an evacuation of the control room. However, the control room and RSPs were electrically connected until the transfer switch on the RSP was engaged. A fire in the control room could have potentially negated the RSPs and vice versa. This modification provided electrical separation between the control room portion of the circuits and the RSPs portion of the circuits. Presently, control of the EC system is still capable from the control room once the transfer switch is engaged, but all controls associated with the steam inlet isolation valves can be overridden from the RSPs. The modification also extended EC line break protection so it would remain in place after the transfer switch is engaged.

f. Emergency Condenser High Steam Flow Confirmatory Logic (Safety Evaluation 84-35)

The ECs initially could have been spuriously isolated due to hot shorts or grounds to the control logic. To resolve this, additional de-energize to activate logic, using the same auxiliary relays which initiate isolation in the principle logic, was added to the control circuit. Both the principle logic and this additional "confirmatory" logic are needed to isolate an EC loop. This confirmatory logic is located in the reactor building.

g. Emergency Condenser System Redundant Automatic Initiation Logic Modification (Safety Evaluation 83-29)

Hot shorts to the principle control logic could have prevented the initiation of the ECs. Additional logic, using the same auxiliary relays which initiate the principle logic, was added in the reactor building. Either the principle logic or this new "redundant" logic is capable of initiating the ECs independent of each other. Only in the condenser valve rooms can both condensers be compromised, and even that is considered highly unlikely.

h. Diesel-Driven Fire Pump Modification (Safety Evaluation 85-04)

The diesel-driven fire pump was modified for local manual start capability. Two solenoid valves, the governor inlet solenoid valve and the pump bearing oil solenoid valve, were replaced. The governor discharge solenoid valve had another manual valve placed in series with it. With manual operation of these new valves, the diesel-driven fire pump may be locally started without the need for dc control power.

The hotwell contains a baffle system in the form of a rectangular labyrinth that enables the condensate to gradually work its way toward the outside of the hotwell. This assures a retention time of approximately 5 min, allowing time for radioactive decay of short-lived isotopes from the time condensate enters the hotwell until it is removed by the condensate pumps.

The condenser shell and turbine exhaust hoods are protected by relief diaphragms in the event of a failure of the turbine bypass valves to close or on loss of condenser vacuum. The diaphragms are designed to relieve at a backpressure of 5 psig.

Deaeration is provided in the condenser for removal of any normal in-leakage of air, plus the hydrogen and oxygen gases contained in the turbine steam due to disassociation of water in the reactor. It is recommended that the oxygen content in the condensate feedwater system be maintained between 30-200 ppb at Station operating design rating per the fuels contract and approved plant procedures. However, the upper ceiling for oxygen for long-term plant operation must also consider the impact of electrochemical potential on corrosion, as described in the EPRI BWR Water Chemistry Guidelines.

3.0 Condenser Air Removal and Offgas System

Noncondensable radioactive process offgas is continuously removed from the main condenser by the condenser air removal and offgas (OFG) system (Figure XI-3). The condenser offgas normally contains activation gases (N-16, O-19 and N-13) and the radioactive noble gas parents of the biologically significant Sr-89, Sr-90, Ba-140 and Cs-137.

The condenser air removal and OFG system was designed to handle the following volume flow rate:

Dry Air	22 scfm
Hydrogen	79 scfm
Oxygen	39 scfm
Water Vapor	Saturated
Noble Gases	Negligible
Total	140 scfm

The condenser air removal and OFG system (Figure XI-3) consists of the following major equipment:

Condenser
Precooler
SJAEs 1st Stage
Intercondenser

Vent Cooler
SJAEs 2nd Stage
After Condenser
Mixing Jet
Offgas Preheater
Recombiners
Recombiner Condensers
Vent Coolers
Hydrogen Analyzers
30-Minute Holdup Pipe
Chillers
Refrigeration Equipment, associated with Chillers
Preadsorbers
Charcoal Columns
Offgas Vacuum Pumps
Offgas Moisture Separator
Offgas Vacuum Pump Coolers
Stack
Sump Tank
Mechanical Vacuum Pumps
1.75-Minute Holdup Pipe (for Steam Packing Exhaust Discharge)
Deicing Water Buffer Tank
Drain Tank (Recombiner)
Associated Valves, Piping and Instrumentation
Hydrogen Water Chemistry Oxygen Injection
Hydrogen Water Chemistry Offgas Sample

The gases to be evacuated by the OFG system are mainly concentrated in the condenser, but steam, air, and other gases evacuated by the steam packing exhauster are also discharged to the OFG system. The first-stage SJAEs extract the gases from the condenser. The gases are diluted with steam in the second-stage air ejector and in the mixing jet. This mixture enters the preheater. The preheater is used during startup to heat the steam/gas mixture to approximately 350°F. Once the system is in operation, the steam heating is secured to the preheater.

Condensate formed in the preheater is returned to the condenser via the drain tank. Leaving the preheater, the gases enter the recombiner(s). The mixture enters through the inlet nozzle and hits a baffle plate and is guided upwards. Then it flows downwards to the recombiner catalyst and outlet nozzle positioned at the bottom of the vessel. At the maximum concentration of hydrogen (4 percent by volume), the temperature inside the vessel can rise to approximately 750°F. The purpose of the recombiner(s) is to catalytically combust the hydrogen and oxygen in a controlled manner to form water. Leaving the recombiner(s), the remaining gas mixture enters the recombiner condenser(s). The condenser cools the superheated gas steam mixture and condenses the steam. The condensate is returned to the main condenser via a drain tank.

A high-point vent is provided at the inlet piping to each recombiner to allow venting of accumulated hydrogen from the inactive recombiner train to the active recombiner train when operating with a single recombiner train.

The noncondensable offgas then enters the vent cooler(s). The vent cooler lowers the moisture content of the concentrated inert gases which contain air, fission gases and traces of moisture by cooling from approximately 200°F to approximately 90°F.

After leaving the vent cooler(s), the gas mixture is directed to the 30-min holdup pipe. Prior to entering the pipe, a sample of the gas mixture is continuously drawn and checked for hydrogen concentration. The hydrogen content should normally be zero.

A sample is drawn from the 30-min holdup pipe where the activity of the gas is measured. If a high reading is detected, the system will be isolated.

Leaving the 30-min holdup pipe, the gases enter the chiller. There are three chillers in the OFG system. One will normally be placed in service while another will be on standby. The remaining unit will be in deice or precool modes. Each of the three chillers has precool, cool and deice cycles. Thus, it can be seen that a number of operations can take place at any one time. The chillers are provided to remove moisture from the gas, through cooling, prior to the gas entering the charcoal columns. After 2 hr of cool cycle, if a chiller outlet temperature exceeds 20°F for longer than a preset time, another chiller will start on a precool cycle. If after the precool cycle the running chiller still exceeds 20°F, the second chiller will begin its cool cycle and first chiller will deice automatically. The deiced water is drained to a buffer tank. From the buffer tank the water is sent to the radwaste system or to barrels, depending on the freon concentration in the water. Each chiller has its own complete refrigeration unit.

A bypass is provided around the three chillers in the event of an emergency when none of the chillers would be operative but the gas flow would have to be maintained to avoid tripping the generating unit.

After the gas leaves the chillers, it enters the preadsorbers. Between the chiller outlet and the preadsorber inlet, the offgas is sampled for freon contamination. The preadsorber is a small charcoal column. Its function is to trap particles and prevent any moisture from entering the main charcoal columns. Only one of the two preadsorbers is normally in service while the second is maintained as a standby unit.

After leaving the preadsorber, the offgas enters the charcoal columns. All six columns will normally be operated in series. However, they can be valved such that the first three or last three can be bypassed.

The charcoal columns delay the noble gases; Xenon for a minimum of 20 days and Krypton for a minimum of 33 hr by adsorption. This is accomplished by means of selected adsorption.

The offgas enters each charcoal column from the bottom and flows upward through the charcoal which rests on diagonal trays. Each set of three columns has a differential pressure indicator across it. In addition, each column has three temperature indicators located at different heights. It should be noted that the efficiency of the charcoal columns decreases (adsorption/delay) with an increase in charcoal temperature.

After leaving the charcoal columns, the offgas enters one of two offgas vacuum pumps. In order to prevent leakage of radioactive gases from the offgas system into the Station and to evacuate the gas to the stack, the system is operated under slight vacuum conditions between the mixing jet and the offgas vacuum pumps.

The two vacuum pumps establish vacuum for operation of the OFG system during startup and hold the vacuum during operation. Normally, one pump is in operation while the other pump serves as a standby unit. If necessary, both pumps can operate in parallel. The vacuum pumps are liquid ring pumps (horizontal type) and are made of noncorrosive material with top suction and discharge. The shaft is sealed through twin stuffing boxes with a liquid seal. The vacuum pumps are designed for a greater gas capacity than expected from the OFG system.

To supply the necessary gas quantities, a bypass from the discharge side of the pump is fed back into the suction line. A vacuum of approximately 12.0 psia is maintained by the vacuum pump between the discharge of the chillers and the inlet of the preabsorbers. A vacuum of approximately 11.4 psia is maintained at the inlet of the vacuum pump(s).

The gases then pass into the offgas moisture separator where the water is separated from the offgas. The water flows through the offgas vacuum pump cooler and returns to the ring water pump. The cooling water for the offgas vacuum pump cooler is reactor building closed loop cooling water (RBCLCW). Unlike the mechanical vacuum pump, there is no pump involved in transferring the water from the moisture separator back to the vacuum pump.

At the outlet of the offgas moisture separator, the offgas is passed first through a wire mesh matting, where droplets of water are separated before the offgas finally enters the stack.

The main offgas blocking valve is located after the moisture separator. This valve will isolate the offgas system if activity in the system reaches the high activity setpoint.

A mechanical vacuum pump system is provided for hogging air from the condenser prior to starting the turbine when steam is not available to operate the SJAEs. Once the SJAEs are placed in service, the suction of the mechanical vacuum pump may be diverted to the condenser water boxes. The condenser water boxes are normally primed using the circulating water priming pumps.

The system consists of two mechanical vacuum pumps, two moisture separators, two seal pumps and two mechanical vacuum pump coolers. This system is capable of evacuating the condenser and associated system from atmospheric pressure to 5-in mercury absolute in approximately 1 hr, with both pumps operating. Operation of one pump extends time to 2 hr.

The mechanical vacuum pump line is capable of automatic isolation initiated from high radioactivity (five times normal) in the main steam line (MSL).

The offgas equipment, piping, valves and filter housings are designed to withstand the high pressure generated by a possible hydrogen-oxygen explosion.

To detect the source of air in-leakage in the OFG system, use of tracer gas monitoring and analyzing equipment temporarily connected to the offgas sampling station has been evaluated. The same technique has been evaluated for condenser tube leaks.

The HWC includes an oxygen injection system to offgas, upstream of the offgas recombiner to maintain stoichiometric mixture of hydrogen and oxygen in the recombiner. The system is provided due to an excess ratio of hydrogen to oxygen at the entrance to the OFG system because of hydrogen injection through the feedwater system.

The HWC includes an additional OFG sample system for monitoring of the offgas percent oxygen concentration from the recombiners to assure that the oxygen addition flows are properly balanced. The HWC OFG sample system draws gas from downstream of the offgas vent coolers.

4.0 Circulating Water System

Two 125,000-gpm vertical, mixed flow, circulating water pumps located in the screenhouse deliver water from Lake Ontario to the condenser water box as shown on Figure XI-4. Each pump discharges in a separate line to one side of the condenser divided water box. Fish screens are installed in each circulating water inlet pipe at the entrance to the water box. These fish screens are in the open position during operation. They are closed just before the circulating water pumps are removed from service to prevent debris from backwashing from the condenser water boxes into the inlet tunnel. This debris collects on the closed fish screen and will be sluiced into the circulating water discharge tunnel.

Each pump suction pit is sectionalized to permit draining of one pit for maintenance while the other pump is in operation. After leaving the condenser, the circulating water is discharged back into the lake. The screenhouse, intake and discharge tunnels are further described in Section III-F.

5.0 Condensate Pumps

Three one-half capacity, centrifugal, motor-driven vertical condensate pumps, each rated at 4,000,000 lb/hr, take suction from the condenser hotwell and discharge it through the full-flow condensate demineralizer (CND) system, the SJAE intercondenser, and the recombiner condensers into the three feedwater booster pumps. Operation of two pumps is sufficient to handle the full operating load (100-percent power) requirements.

Alarms for low condensate discharge header pressure, low and high hotwell level, high condensate temperature leaving the hotwell, and low condenser vacuum are provided to alert the Operator of abnormal conditions.

6.0 Condensate Filtration System

The full-flow CFS is located upstream of the condensate demineralizers, as shown on Figure XI-5, and is designed to remove 99 percent of the insoluble iron and copper from the condensate water. There are four filters sized for 100 percent condensate flow. There is a 25 percent bypass line available for use during filter backwash and a 100 percent bypass line which can be used to bypass all four filters if it is necessary to take the system out of service. The purpose of the filters is to extend the lifetime of the condensate demineralizer resin by reducing the need for ultrasonic cleaning of condensate resin beds. The removal of insoluble iron and copper also results in a reduced possibility of fuel failures and in reduced radiological dose. The filters are cylindrical vessels mounted vertically. Each vessel has a fully removable top head to allow unrestricted insertion and removal of filter element bundles or modules.

The entire CFS is designed and built to the same codes and standards as the condensate and feedwater systems. In addition to the filter tanks, the CFS consists of a backwash receiving tank (BWRT), vent system, air receiver tank, air compressor, control panel, and other miscellaneous components.

The filters are backwashed and reused. Any material removed from the filters by backwashing goes to the BWRT. The material from the BWRT is then sent to radwaste for processing as required. The CFS, backwashing, and associated transfer equipment are manually controlled from a local control panel. The local control panels control all of the main flow valves, initiate and control filter backwash sequences, and contain all controls, indications and alarms for the operation of the CFS.

Replacement of filter media is accomplished from the installed work platform around the filters. The filter head is removed and the filter media is lifted, the filter media replaced, and filter head reinstalled by using the installed monorail hoists. The filter media is processed by radwaste.

The filter vessels each have a cylindrical radiation shield around them. This shield extends above and below the filter portion of the tank. The BWRTs are also shielded. Shielding is designed to ensure that area dose rates are maintained <5 mrem/hr general area.

An alarm is provided for CFS trouble in the main control room.

6.0A Condensate Demineralizer System

The full-flow CND system, as shown on Figure XI-5, assures water of the required purity to the reactor. The full flow of condensate is passed through the CNDs as required for load conditions. There are six mixed-bed demineralizers, sized for rated load condensate flow, piped in parallel. They can be used in any combination as required to remove corrosion products gathered from the turbine, condenser, and the shell side of the

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feedwater heaters; protect the reactor against condenser tube leaks; and remove condensate impurities which might enter the system in the makeup water. Three of the demineralizer tanks are rubber lined. The other three are lined with a ceramic coating. All six tanks are the carbon steel type, sized for a nominal flow rate of 50 gpm per square foot of bed surface area when six demineralizers are in service at full power (1850 MWt). When it is necessary (due to ultrasonic resin cleaning (URC) or bed replacement) to take one unit out of service (OOS), the flow is approximately 58 gpm per square foot. The maximum nominal design flow is 64 gpm per square foot of bed surface area with five demineralizers on-line.

Strainers located on the discharge side of the demineralizers prevent accidental carryover of resins to the reactor.

Demineralizer resins are normally mechanically cleaned by air scrubbing, backwashing, and sound energy, and reused. Any radioactive material removed from the exhausted resins by the cleaning and rinse solutions is transferred to the waste disposal system described in Section XII-A for processing as required.

The CND and associated transfer and cleaning system are manually controlled from two adjacent local panels, the resin transfer and cleaning panel and the CND control panel. Integrated flow, conductivity, instantaneous flow, differential pressure, and effluent strainer differential pressure monitors are provided at the CND control panel for each demineralizer to indicate when cleaning or resin bed replacement is required.

Main flow valves are remotely operated from the CND control panel. Resin transfers from the demineralizers to the cation tank, and from the resin storage tank to the empty demineralizer tank, are manually initiated. Backwash, mechanical cleaning, and rinsing of the resins, the transportation of ultrasonically cleaned resins to the resin storage tank, and resin mixing are manually initiated at the local resin transfer and cleaning panel.

The demineralizer vessels and resin cleaning tanks are located in concrete shielded areas and are arranged for remote operation. Shielding around the demineralizers is designed to give 1.5 mr/hr in the corridor and 100 mr/hr at the south wall facing turbine operating floor and in the demineralizer piping area. The piping area is shielded to give 30 mr/hr in the demineralizer valve

MAIN CONDENSER AIR REMOVAL AND OFF GAS SYSTEM

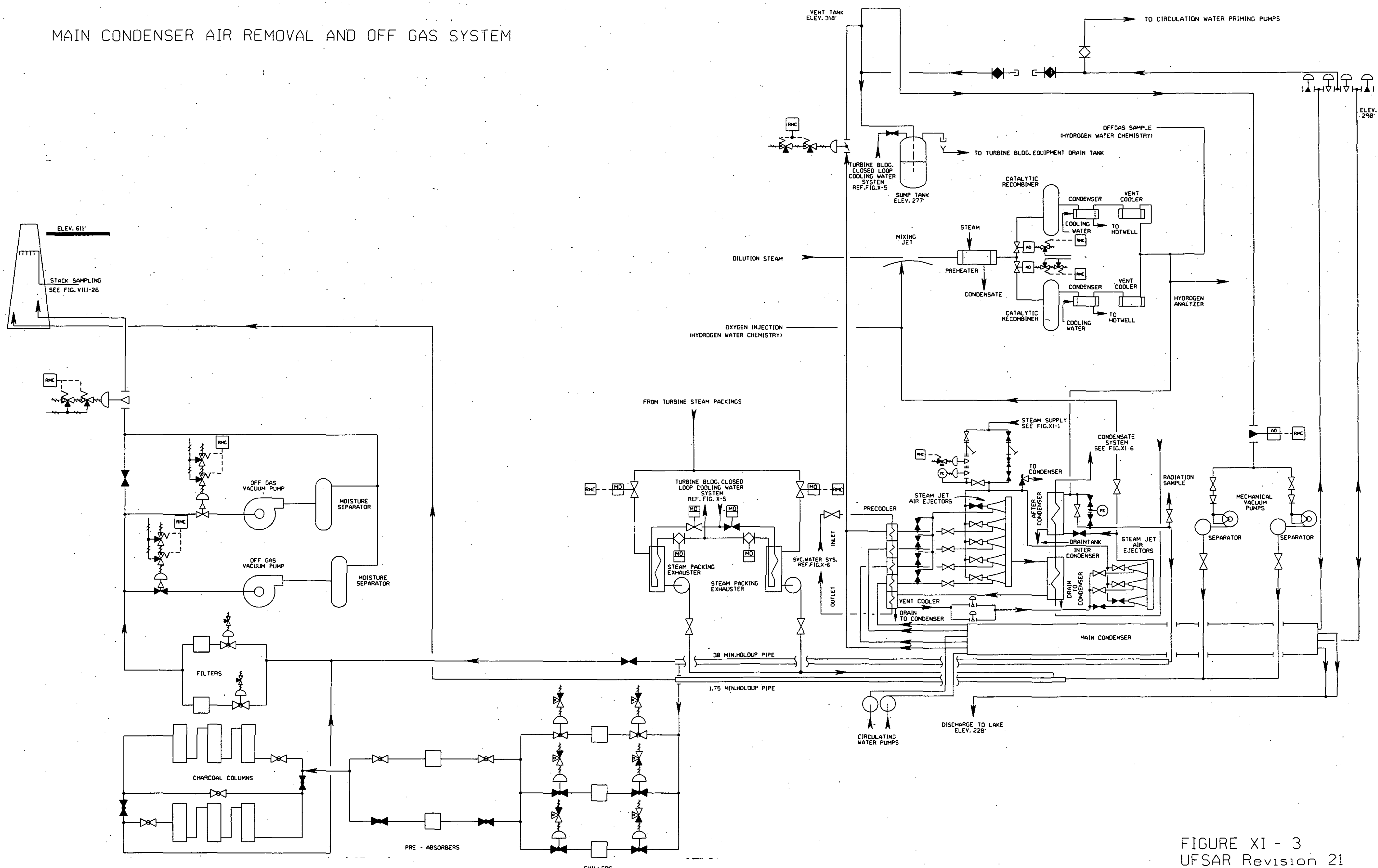

FIGURE XI - 3
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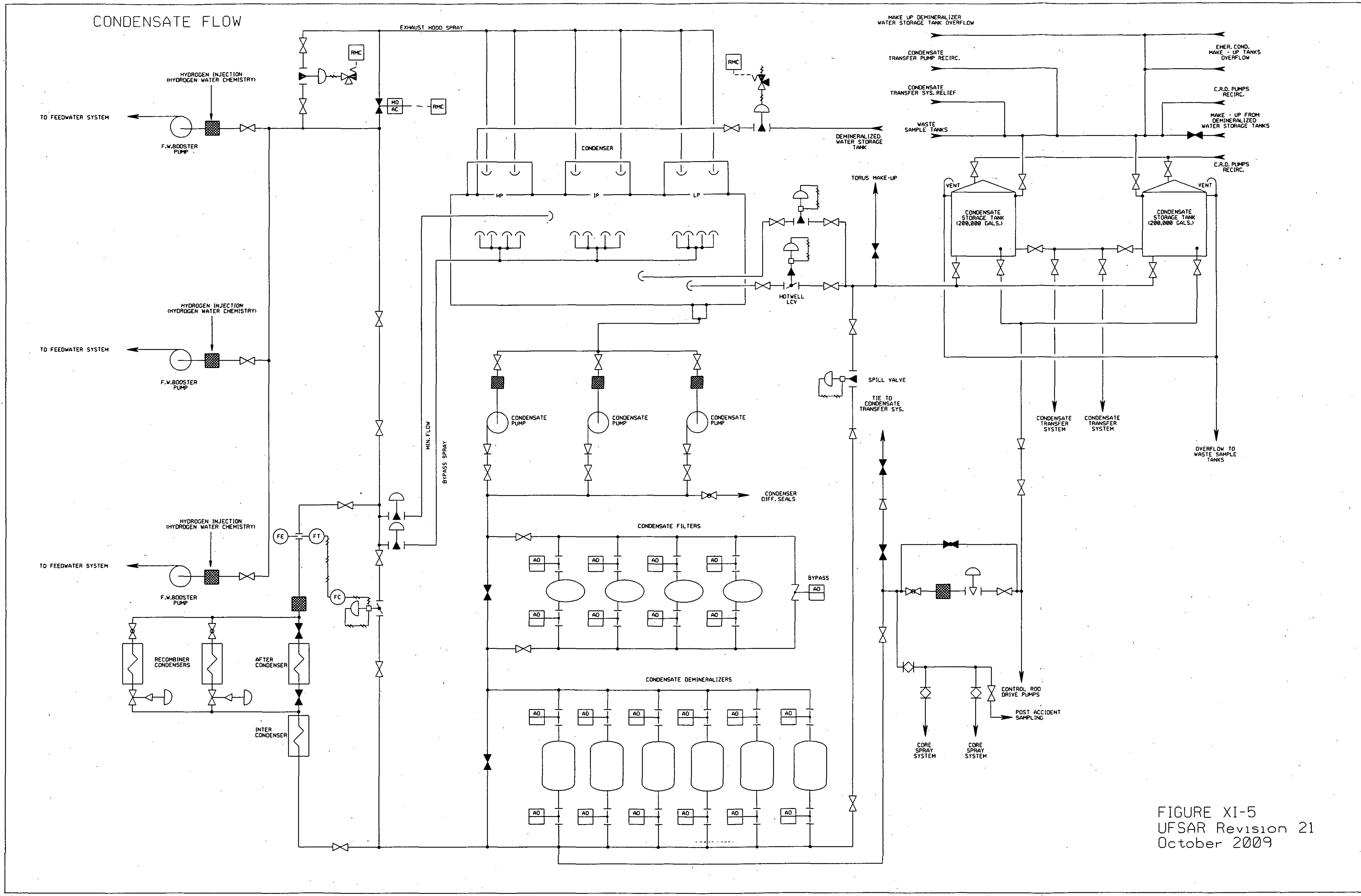

FIGURE XI-5
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Accident Analysis

Most of the monitors would be of little or no use in the analysis of conditions after an accident since they are designed to monitor the relatively low radiation levels expected during Station operation. However, two of the monitors are designed for use during or following an accident:

1. The monitor in the control room is designed primarily to keep personnel advised of the radiation levels in this area following an accident.

2. The higher-ranged monitor on the fuel pool refueling platform is designed to monitor higher radiation levels that could set the lower-ranged instrument off scale during work in the spent fuel pool. This high-range instrument could also be used in evaluating radiation levels in the reactor building following an accident.

2.1.3 Evaluation

General Station Operation

Even though the area radiation monitors are in areas of relatively low radiation levels, they do not replace portable instrument surveys but rather supplement them. In addition, the monitors are well scattered throughout the Station, so that one monitor does not serve as a "backup" for another monitor. Therefore, there are no requirements for any definite number of monitors to be in service for Station operation. Any malfunctioning monitors will be repaired as soon as is reasonably achievable.

Spent Fuel Storage Pool Operation

During work in the fuel storage pool in which recently irradiated fuel is handled, the high-range monitor on the refueling platform must be operating to ensure initiation of emergency ventilation system in the event of high radiation.

Movement of Monitors

If any of the locations chosen prove to be ineffective after a period of Station operation, monitors can be moved to more effective locations.

Monitor Range Changes

Accidental or unauthorized range changes are precluded because several internal circuit modifications are required to make such changes.

Alarm Points

Alarm points are set as specified by the General Supervisor Radiation Protection.

2.2 Area Air Contamination Monitoring System

2.2.1 Design Bases

1. Personnel occupying or entering one of the three major buildings (reactor, turbine or waste disposal) should be warned of significant airborne contamination levels.

2. Monitors actuate an alarm in the control room and at the monitor. The Station intercommunication system will also be used to advise any personnel in the area affected.

2.2.2 Design

Monitor Location

Constant air monitors are installed in low background areas sampling each of the main exhaust ducts of the three major buildings. Monitor readout and the alarm signal are transmitted to the control room. Additional portable constant air monitors with local readout are provided.

Monitor Design

Commercially available beta-gamma removable filter constant air monitors are used for the air contamination monitoring system. The monitors have a high-level trip point to indicate an abnormal airborne contamination level and a low-level trip point to indicate instrument malfunction.

2.2.3 Evaluation

General Station Operation

Since the constant air monitoring system is supplemented by constant air samples (samples taken continuously, with the activities analyzed at the end of the sampling period) and by short-term "grab" samples, it is not essential for Station operation. However, because the constant air monitoring system does provide useful information in a convenient-to-use form, any malfunctioning monitors will be repaired as soon as is reasonable.

Movement of Portable Monitors

Monitors will be moved in response to operational or maintenance activities to more effective locations, as required.

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Radiation Area. Personnel exposures in radiation areas are kept to a minimum by use of administrative procedures based on accumulated doses and by keeping time spent in radiation areas as short as possible. Radiation areas may be isolated with yellow and magenta rope and are posted with signs:

CAUTION

RADIATION AREA

2. High Radiation Area

Any area in which the radiation level is greater than 100 mrem/hr is designated as a high radiation area. Entrances to high radiation areas (100 to 1000 mrem/hr) to which personnel require frequent access are barricaded and normally kept locked and may be equipped with an alarm system which will warn the person entering. Entrances to high radiation areas above 1000 mrem/hr shall be provided with locked doors to prevent unauthorized entry, and the hard keys or access provided by magnetic keycard shall be maintained under the administrative control of the Shift Manager (SM) or designate on duty and/or the General Supervisor Radiation Protection or designate, and issued to personnel with the appropriate radiation work permit (RWP). Access to high radiation areas that may be reached only by ladders or climbing structures, such as loft spaces above false ceilings or the upper volumes of high rooms, is not controlled by automatic alarms or special barricades.

All high radiation areas will be posted with signs:

CAUTION OR DANGER

HIGH RADIATION AREA

Radiation protection personnel make routine surveys of all the accessible areas in the Station to keep abreast of any changes in the radiation levels in these areas.

3.3 Contamination Control

Contamination control is achieved in general by physical separation of the contaminated area.

3.3.1 Facility Contamination Control

Contamination of the general Station areas is prevented by using the "step-off-pad" technique when leaving areas that are contaminated. Monitoring devices are placed near the step-off-pad so that personnel can check to assure they are not

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inadvertently carrying some contamination with them. Unless specifically exempted by radiation protection supervision, personnel will monitor themselves prior to each exit from the RCA to assure that no contamination is being carried from the RCA. For maintenance jobs involving high levels of contamination, the installation of plastic or paper on the floor around the equipment to be maintained will permit quick and easy cleanup after the work is completed. Thus, spread of contamination to other equipment or other floor areas is prevented.

Radiation protection personnel make routine surveys of the contamination levels in all the accessible areas of the Station to keep abreast of any changes in contamination status. Any areas found contaminated to undesirable levels will be roped off and posted. These areas are decontaminated as soon as is reasonable.

3.3.2 Personnel Contamination Control

Contamination of personnel is controlled in two ways. First, contamination is prevented from getting into areas where personnel can unknowingly come in contact with it by using the methods described in Section 3.3.1.

Second, personnel who enter contaminated areas are protected with special protective clothing. The following types of protective clothing are used:

1. Coveralls - Worn for most work in contaminated areas.

2. Plastic suits - Worn in areas where potential exists for liquid contamination of personnel.

3. Gloves - Cotton gloves are worn for protection against dry contamination and rubber gloves for protection against dry or wet forms of contamination.

4. Shoecovers - Cloth covers are worn for protection against dry contamination; plastic shoecovers for dry or moist contamination; rubber overshoes for dry, moist or wet contamination.

5. Head protection - Caps are worn for protection against low-level contamination; cloth hoods for protection against high-level contamination; plastic hoods for protection against very high or moist contamination.

If contamination levels are moderate to high, the various pieces of clothing worn are taped together to prevent contamination from entering the joints. In some cases, double layers of clothing are worn to give additional protection.

Normally, most of the Station is accessible to personnel in street clothes or nonradioactive work clothes. To minimize the

SECTION XIII

CONDUCT OF OPERATIONS

A. ORGANIZATION AND RESPONSIBILITY

The following sections describe the organizational structure of NMPNS and delineate the lines of responsibility for the operation of Unit 1 in accordance with established administrative and quality standards. The organizational structure associated with the Quality Assurance (QA) Program for plant operation is described in the Quality Assurance Topical Report (QATR).

1.0 Management and Technical Support Organization

1.1 Station Organization

The senior level Station management organization is depicted on Figure XIII-1.

1.1.1 Vice President Nine Mile Point

The Vice President Nine Mile Point reports to the Senior Vice President and Chief Operations Officer of Constellation Energy Nuclear Group (CENG) and has overall responsibility for the administration and operation of the Nine Mile Point Nuclear Station. The Director Quality & Performance Assessment, Director Human Resources, Director Finance & Business Operations, and the Director Information Technology (IT) report directly to CENG senior management and have matrixed reporting to the Vice President Nine Mile Point for functional and priority setting direction. The Plant General Manager, Manager Training, Manager Engineering Services, Manager Nuclear Safety and Security, and Director of Project Management report directly to the Vice President Nine Mile Point.

1.1.2 Matrixed Reporting

1. The Director Quality & Performance Assessment (NMPNS) reports directly to the CGG Manager Fleet Quality & Performance Assessment for program and policy direction, with matrixed reporting to the Vice President Nine Mile Point for functional and priority setting direction. The Director Quality & Performance Assessment, in performing duties as the manager

quality assurance, has the authority and responsibility to report directly to the Vice President Nine Mile Point regarding implementation of the Nine Mile Point QA Program.

The Quality Assurance organization is depicted on Figure XIII-3.

2. The Director Human Resources reports to the Vice President Human Resources-Generation for program and policy direction, with matrixed reporting to the Vice President Nine Mile Point for functional and priority setting direction. This director is responsible for Employee/Labor Relations and Leadership/Career Development.

3. The Director Finance & Business Operations reports to the Vice President Business Operations (CENG). The Director Finance & Business Operations also reports to the Vice President Nine Mile Point for business planning functions with matrixed reporting for all other functions. This director is responsible for the functions of Business Planning, Site Accounting, Budgets, Cost Control, and PSC interface for the nuclear station.

4. The Director Information Technology (IT) reports to the Director Nuclear Fleet IT for program and policy direction, with matrixed reporting to the Vice President Nine Mile Point for functional and priority setting direction.

1.1.3 Qualifications of Support Personnel

General responsibilities and activities of management and technical support personnel are described in appropriate documents including administrative procedures and engineering procedures. Contract support for Unit 1 is utilized in the same general manner as contract support at Unit 2.

2.0 Nine Mile Point Nuclear Station, LLC, Organization

This section describes the structure, function, and responsibilities of the onsite organizations established to operate and maintain the plant. The onsite and offsite independent review committees are described in Section XIII-G. Unit 1 and Unit 2 operations are independent of each other, including backshift operation. Only licensed individuals may direct licensed activities.

An organization chart showing the title of each position is shown on Figures XIII-4 through XIII-4c. The lines of authority are described in administrative procedures.

2.1 Plant General Manager

The Plant General Manager reports to the Vice President Nine Mile Point, is responsible for overall unit operation, shall have control over those resources necessary for safe operation of the plant, and assumes the duties and responsibilities of the Vice President Nine Mile Point, in his absence, for matters affecting the Station. The Plant General Manager has overall responsibility for safe and efficient Station operation, in accordance with applicable licensing, regulatory and Quality Assurance Program requirements, and controlling the preparation, review, and approval of Station procedures.

The Plant General Manager maintains an organization comprised of the following direct reports with associated responsibilities:

1. The Manager Operations performs the following functions:

    a. Ensures safe operation of the Station in accordance with approved procedures and regulatory requirements.

    b. Advises Shift Manager (SM) (formerly the Station Shift Supervisor) during emergency conditions.

    c. Performs the duties associated with PORC membership.

    d. Assists in the development of training programs.

    e. Administers implementation of the Fire Protection Program for the Nine Mile Point site.

    f. Maintains an organization comprised of the following functional sections:

        · Station Operations
        · Operations Support
        · Fire Protection

2. The General Supervisor Radiation Protection manages radiation protection monitoring and control programs in support of Station operation. This general supervisor meets the radiation protection manager qualifications in Technical Specifications Section 6.3.1. The General Supervisor Radiation Protection has:

   - Direct access to appropriate levels of corporate management, including the Chief Nuclear Officer, to resolve radiation protection concerns.

   - Authority to require plant shutdown if unsafe radiological conditions exist.

   The General Supervisor Radiation Protection manages Radiation Protection and ALARA personnel and ensures procedures/qualifications comply with Federal and Technical Specification requirements related to monitoring, control and minimization of radiation exposure to plant personnel. This general supervisor:

   a. Performs the duties associated with PORC membership.

   b. Controls preparation, review, and approval of Radiation Protection and Waste Handling procedures, and assists in the development of training programs.

   c. Maintains an organization comprised of the following functional sections:

      - Radiological Engineering
      - Radiological Support
      - Radiation Protection Operations
      - Radiation Material Processing

3. The General Supervisor Chemistry monitors and controls programs, including personnel, procedures and qualifications, to ensure compliance with Federal and Technical Specification requirements related to primary and secondary system chemistry and radiochemistry, radioactive effluent, chemistry control, post-accident assessment, and solid radioactive waste measurements. This position:

a. Manages operation of, and waste disposal aspects of, the Sewage Treatment Facility.

b. Performs the duties associated with PORC membership.

c. Assists in development of training programs.

d. Maintains an organization comprised of the following functional sections:

- Chemistry Operations
- Chemistry Support

4. The Director Performance Improvement establishes and maintains the program documents and procedures for implementing the Corrective Action Program (CAP).

5. The Manager Maintenance ensures modifications, surveillance, maintenance, preventative maintenance, radiation instrument calibration, and housekeeping and decontamination are properly performed in accordance with applicable rules, regulations, approved procedures, codes and standards. This position:

a. Manages relay and control testing activities, measuring and test equipment calibration, and maintenance planning functions.

b. Performs the duties associated with PORC membership.

c. Assists in the development of training programs.

d. Ensures necessary maintenance personnel are available to maintain the Station in a safe and efficient manner.

e. Ensures radiologically-controlled area (RCA) housekeeping and decontamination are maintained.

f. Maintains an organization comprised of the following functional sections:

- Mechanical Maintenance
- Electrical Maintenance
- I&C Maintenance

- Maintenance Support
- FIN

6. The Manager Integrated Work Management ensures the safe and efficient planning and implementation of forced, planned and refuel outages at NMPNS, as well as planning and implementation of weekly work schedules. This position:

   a. Manages the scheduling function.

   b. Ensures integrity of the Work Control Center and scheduling databases.

   c. Maintains interfaces among Nuclear Generation departments for maintenance, modification and testing activities.

   d. Maintains an organization comprised of the following functional sections:

      - Outage Management
      - On-line Scheduling
      - Work Management Programs
      - Planning
      - Records and Document Services

7. The Director Safety & Health interprets Occupational Safety and Health Administration (OSHA) requirements and advises, assists, and coordinates efforts in the implementation of those requirements.

2.2 Other Functions Reporting to the Vice President Nine Mile Point

1. The Manager Engineering Services reports directly to the Vice President Nine Mile Point. This position has full authority to provide nuclear engineering services that comply with applicable safety, regulatory, and quality requirements within defined cost and scheduling parameters. In addition, this position has single-point accountability for technical concerns and responses.

   The Engineering Services organization chart is provided on Figure XIII-2. The following positions report to this manager:

a. The General Supervisor Design Engineering supervises design engineering services to assure safe, reliable, and economic operation of Nine Mile Point Nuclear Station. Specific responsibilities are to ensure:

- Engineering is performed in accordance with applicable regulatory and code requirements (e.g., the UFSAR, Technical Specifications, etc.).

- Detailed design/engineering is completed based upon conceptual design information including specifications and drawings necessary to implement these designs.

- As-installed conditions are reflected on drawings.

- Implementation of the NMPNS Configuration Management Program.

- Implementation of conceptual engineering.

- Plant evaluations are performed to monitor and detect internal and external factors that would indicate an actual or potential degradation of design bases or margin in design bases for initial plant systems and components.

- Development, verification and maintenance of special programs for:

  - Seismic Qualification
  - Metallurgy
  - Environmental Qualification
  - Reactor Vessel and Internals (VIP)

b. The General Supervisor Engineering Programs is responsible for ASME Programs, AOV/MOV/Check Valve Program, Fire Protection, and Maintenance Rule/EPIX.

c. The General Supervisor System Engineering reports to the Manager Engineering Services. The Manager Engineering Services has final design authority for technical issues.

d. The General Supervisor Equipment Reliability reports to the Manager Engineering Services. The General Supervisor Equipment Reliability is responsible for directing activities associated with activities, initiatives, and programs associated with Maintenance Rule activities, EPIX program, thermal performance, equipment performance monitoring, and system health reporting.

e. The Senior Engineer Nuclear Fuels has a matrix reporting relationship to the Manager Engineering Services and reports directly to the CGG General Supervisor Nuclear Fuels.

2. The Manager Training reports directly to the Vice President Nine Mile Point and manages the activities of the Training organization, including the development, administration, and coordination of training and retraining programs for NMPNS personnel. This manager ensures activities within the Training organization are properly conducted per applicable regulations, codes, standards, and procedures.

3. The Manager Nuclear Safety & Security reports directly to the Vice President Nine Mile Point and is responsible for managing the nuclear safety assurance function consistent with maintaining the NRC Operating Licenses, Technical Specifications, and UFSAR/FSAR. This position oversees Emergency Preparedness, Security, Licensing, and Environmental activities ensuring full compliance with regulatory requirements and company policies and procedures.

2.3 Supervisor Engineering-Nuclear Fuels

The Supervisor Engineering-Nuclear Fuels reports to the General Supervisor-Nuclear Fuels and is responsible for proper implementation of the Reactivity Management Program. This position:

1. Provides direction and engineering expertise to Operations and other groups for the control of reactivity.

2. Evaluates site and industry reactivity related events for applicability and lessons learned.

3. Supports review of plant procedures, maintenance activities, and modifications for potential reactivity effects.

4. Monitors the effectiveness of the Reactivity Management Program.

5. Ensures that training is provided to Operations personnel prior to implementation of new core design or new core operating strategies.

6. Controls and verifies proper implementation of the Fuel Handling Procedures.

7. Performs duties associated with PORC membership.

Acts as the Special Nuclear Material Custodian and is responsible to ensure:

1. Applicable procedures are developed and implemented to control receipt, storage, movement, and shipment of special nuclear material (SNM).

2. The possession and use of SNM is confined to the locations and purposes authorized by the Station's Operating License.

3.0 Quality Assurance

The operations phase QA Program is described in the Quality Assurance Topical Report (QATR). The QATR identifies the organizations responsible for activities affecting the operation, maintenance or modification of safety-related structures, systems, or components, and describes the assigned authorities and duties for quality-attaining functions and for quality verification functions.

4.0 Operating Shift Crews

Table XIII-2 shows the position titles, applicable Operator licensing requirements, and minimum numbers of personnel planned for each shift for the various reactor operating conditions. Unique requirements for additional personnel for the refueling condition are also noted in Table XIII-2. The following additional requirements apply:

1. At least one licensed Operator shall be in the control room when fuel is in the reactor. During reactor operation, this licensed Operator shall be present at the controls of the facility.

2. A licensed Senior Reactor Operator or licensed Senior Reactor Operator Limited to Fuel Handling shall be responsible for all movement of new and irradiated fuel within the site boundary.

5.0 Qualifications of Staff Personnel

Each member of the unit staff, with the exception of the Operator license applicants and the Radiation Protection Manager, shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions. The Radiation Protection Manager shall meet or exceed the qualifications of RG 1.8, September 1975. The education and experience eligibility requirements for Operator license applicants, and changes thereto, shall be those previously reviewed and approved by the NRC; specifically, those referenced in letter NMP1L 2184, dated December 20, 2007, and described in applicable Station training procedures.

A retraining and replacement training program for the facility staff shall be maintained under the direction of the Manager Training, and shall meet or exceed the recommendations and requirements of Section 5.5 of ANSI N18.1-1971 and of 10CFR55, and shall include familiarization with relevant industry operational experience.

B. QUALIFICATIONS AND TRAINING OF PERSONNEL

1.0 (This section deleted)

2.0 (This section deleted)

3.0 (This section deleted)

4.0 Training of Personnel

4.1 General Responsibility

The Manager Training is responsible for all training at the Nine Mile Point Nuclear Station.

4.2 Implementation

1. The Manager Training reports directly to the Vice President Nine Mile Point and manages the activities of the Training organization, including the development, administration, and coordination of training and retraining programs for site personnel.

2. The Manager Training develops and ensures implementation of the Training organization portion of the business plan.

3. The Manager Training ensures activities within the Training organization are properly conducted per applicable regulations, codes, standards, and procedures.

4. The Manager Training maintains appropriate safety and budget control programs, and ensures adequate resources are assigned within the Training organization.

4.3 Quality

Responsibility for the general quality of training in each area shall be distributed as follows:

4.3.1 For Operator Training

The Plant General Manager with the assistance of the Manager Operations.

4.3.2 For Maintenance

The Manager Maintenance.

4.3.3 For Technicians

The Manager Maintenance, General Supervisor Chemistry, General Supervisor Radiation Protection, Manager Operations.

4.3.4 For General Employee Training/Radiation Protection and Emergency Plan

The General Supervisor Chemistry, General Supervisor Radiation Protection, Director Security, Director Quality and Performance Assessment, and Director Emergency Preparedness.

4.3.5 For Industrial Safety

The Director Safety & Health.

4.3.6 For Nuclear Quality Assurance

The Director Quality and Performance Assessment.

4.3.7 For Fire Brigade

The Manager Operations.

4.3.8 For Manager Operations and General Supervisor Operations

As a minimum, either the Manager Operations or the General Supervisor Operations shall hold a SRO license. The Manager Operations, who in lieu of meeting the SRO license requirements of ANSI N18.1-1971, shall: 1) hold a SRO license at the time of appointment, or 2) have held a SRO license at Unit 1 or at a similar unit, or 3) have been certified for equivalent SRO knowledge.

4.4 Training of Licensed Operator Candidates/Licensed NRC Operator Retraining

Detailed training programs for Unit 1 Operations are designed to provide initial training, requalification training, and continuing training at all levels of the Operations organization. These programs fulfill the requirements included in the following documents:

10CFR50, Licensing of Production and Utilization Facilities

10CFR55, Operators Licenses

NUREG-0737, Clarification of TMI Action Plan Requirements

NUREG-1021, Operator Licensing Examiner Standards

ANSI N18.1-1971, Selection and Training of Nuclear Power Plant Personnel

ANSI/ANS 3.4-1983, Medical Certification and Monitoring of Personnel Requiring Operator Licenses for Nuclear Power Plants

The training program is designed in accordance with accreditation programs described in the latest approved ACAD recommendations, and uses a simulation facility acceptable to the NRC under 10CFR55.

Nuclear Division procedures contain the requirements, policies, and practices necessary to implement Operator training programs. Each program is described in dedicated nuclear training procedures (NTPs). These procedures contain the scope and purpose of the training program, an outline of the course curriculum, instructions for scheduling the program, and reporting requirements. Copies of procedures NTP-10 and NTP-11 detailing the training program were initially submitted to the NRC in a letter dated October 28, 1986, in accordance with Generic Letter 84-014. The current training procedures are:

1. NIP-TQS-01, Qualification and Certification (formerly AP-1.3.1 and AP-9.1)

2. NTP-TQS-101, Training of Licensed Operator Candidates (formerly NTP-10)

3. NTP-TQS-102, Licensed Operator Requalification Training (formerly NTP-11)

4. NTP-TQS-103, Training of Nuclear Auxiliary Operators (formerly NTP-12)

Entry into these training programs is controlled by the Operations organization. Eligibility criteria for license candidates is contained in instructions and procedures maintained by the Operations organization.

5.0 Cooperative Training With Local, State and Federal Officials

A detailed Site Emergency Plan and Procedures for Nine Mile Point Nuclear Station has been submitted to the NRC. Included in this document are the training procedures involving local, state and federal officials.

C. OPERATING PROCEDURES

The Station operating staff has prepared written operating procedures to be used for all normal operating conditions.

Changes may be initiated by the Station operating staff subject to approval by the Operations supervisory staff, and by the department manager for the functional area of the procedure, or higher levels of management as governed by administrative procedures.

Procedures cover operation of major systems such as starting the entire Station from "cold" conditions. Other procedures cover less extensive systems in detail.

Still another type of procedure instructs the Operator in the methods of operating individual pieces of equipment, such as regeneration of resin in a demineralizer.

The format for all operating procedures is essentially the same. Each procedure is prefaced with the technical limitations of the system or equipment, as set forth in the Technical Specifications, OL, or 10CFR20. Other data helpful to operation, such as a system description and plant operating requirements, are included in a separate section of the procedure. Details of operation are then set down in a stepwise procedure. Prior to startup, prepared lists are checked off by the Operators. These vary in degree with the extent of the period preceding the startup.

The emergency operating procedures (EOP) and the severe accident procedures (SAP) have been developed, validated and implemented in accordance with the requirements of NUREG-0737 Supplement 1. The EOPs/SAPs are prepared using the guidance provided by the BWR Owners' Group Emergency Procedure and Severe Accident Guidelines (BWROG EPG/SAG). The EOPs/SAPs are symptom oriented rather than event based. They address conditions beyond the design basis. They provide guidance for the entire range of available systems. This "defense-in-depth" approach provides for safe shutdown of the plant in all postulated events, including anticipated transients without scram (ATWS), thus preventing or mitigating the consequences of any accident or malfunction.

In the event of an unlikely, yet credible, accident situation which might involve radioactivity release to the public domain, the SM, in accordance with written procedures, is responsible for notifying Station supervision and outside authorities. It is recognized that a program of this type must be rehearsed; therefore, planned nuclear incident drills are held periodically to review established procedures, personnel assignments and relations with outside authorities.

D. EMERGENCY PLAN AND PROCEDURES

The Nine Mile Point Nuclear Station Site Emergency Plan describes the total preparedness program established, implemented and coordinated by the Station to assure the capability and readiness for coping with and mitigating both onsite and offsite consequences of radiological emergencies. The Site Emergency Plan covers the spectrum of emergencies from minor localized incidents to major emergencies involving protective measures by offsite response organizations. Included are guidelines for immediate response, assessment of emergency situations, defined action criteria and delineation of support functions. Site Emergency Plan Implementing Procedures provide detailed information for individuals who may be involved with specific emergency response functions. The Site Emergency Plan provides for a graded scale of response for distinct classifications of emergency conditions, action within those classifications, and criteria for escalation to a more severe classification. This classification system is the same as that used by the State of New York and the Oswego County Emergency Management Office. The plans have four emergency categories: unusual event, alert, site area emergency, and general emergency. In addition to notifying the offsite agencies of the existing emergency classification, provisions are made in the emergency procedures for the Station to advise the State and County of appropriate protective actions.

The organization for control of emergencies begins with the shift organization, and contains provisions for augmentation and extension to include other Station personnel and outside emergency response organizations (EROs).

The following emergency response facilities (ERFs) are provided to ensure the capabilities for the prompt, efficient assessment and control of situations over the entire spectrum of probable and postulated emergency conditions.

1. Technical Support Center (TSC)

2. Operations Support Center (OSC)

3. Emergency Operations Facility (EOF)

4. Joint News Center (JNC)

5. Oswego County Emergency Operations Center

6. State Emergency Operations Center

Formal training along with drills and exercises are essential in maintaining an in-depth emergency preparedness program.

The Site Emergency Plan and Implementing Procedures have been submitted to the NRC under separate cover. Changes to the Site Emergency Plan are made in accordance with the requirements of 10CFR50.54(q).

E. SECURITY

A detailed Nine Mile Point Nuclear Station Physical Security, Safeguards Contingency, and Security Training and Qualification Plan, identified as safeguards information and withheld from public disclosure in accordance with 10CFR73.21, has been submitted to the NRC.

The security plan described above details the measures taken to provide adequate Site and Station security and conforms to 10CFR73.55. Changes to the security plan are made in accordance with the requirements of 10CFR50.54(p) or 10CFR50.90, as applicable.

F. RECORDS

1.0 Operations

The following logs will be maintained by the operating staff as a part of the Station records. When electronic logs are used, the control room log and the SM log may be combined.

1.1 Control Room Log

Shall contain all information pertaining to changing core reactivity during all modes of reactor operation, including rod manipulation, orifice modifications, control rod testing, etc. Also, entries affecting Station outputs, changes in auxiliary equipment, unusual condition, line trips, annunciator signals not recorded on data logger, etc., will be entered in this log. The log shall contain the date and time of all entries and the name of the Chief Shift Operator (CSO) or other authorized personnel only. The control room log is to be treated as a legal document subject to being entered in a court record. All entries in this log shall be by the Operator on duty or his Supervisor. No other entries are authorized. Included with the control room log is a fuel log in which specific detailed fuel moves, channel changes, and in-core instrumentation changes are recorded.

1.2 Shift Manager's Log

Shall contain an overall summary of Station operation including the name of the SM on duty, the Operators and Auxiliary Operators on duty, major equipment not in service or inoperable, and the date and time of all entries. Also note any Operator surveillance tests run and deviations from acceptance criteria. The log may be written by the Control Room Supervisor (CRS) or a CRS/SM in training, or other designee, but must be signed and acknowledged by the SM.

1.3 Radwaste Log

The log shall contain pertinent information associated with the radwaste facility operation.

1.4 Waste Quantity Level Shipped

Solid waste and resins removed from site.

2.0 Maintenance

The Manager Maintenance will be responsible for maintaining a record of maintenance performed on all pertinent equipment in a maintenance log.

3.0 Radiation Protection

The General Supervisor Radiation Protection will be responsible for the following records.

3.1 Personnel Exposure

1. Dosimeter readings, daily

2. Thermoluminescence dosimeter (TLD) record, quarterly

3. Continuous exposure record conforming to 10CFR20

4. Appropriate records and forms required in 10CFR20

3.2 By-Product Material as Required by 10CFR30

3.3 Meter Calibrations of all survey meters, environmental monitors and monitors affecting radioactive discharge.

3.4 Station Radiological Conditions in Accessible Areas

1. Radiation levels

2. Contamination levels

3. Airborne activity

3.5 Administration of the Radiation Protection Program and Procedures

4.0 Chemistry and Radiochemistry

The General Supervisor Chemistry is responsible for primary and secondary system Chemistry and Radiochemistry including monitoring and control of liquid and gaseous radiological effluents.

5.0 Special Nuclear Materials

The special nuclear materials records will be maintained and reported in conformity with 10CFR70.

6.0 Calibration of Instruments

The calibration of instruments and controls, both nuclear and conventional, will be recorded, as well as maintenance performed on them.

7.0 Administrative Records and Reports

1. Investigations of abnormal operation will be prepared in report form and distributed to interested parties.

2. Records will be kept of all changes to equipment or procedures.

3. Reports of production and pertinent operating data with a summary of items of interest will be produced at regular intervals and distributed to interested parties and to those who audit Station operations.

4. Reports of exposure to individuals, loss or theft of licensed material, etc., as outlined in 10CFR20 will be reported in the time and manner specified.

G. REVIEW AND AUDIT OF OPERATIONS

A means is provided for processing changes and assuring safe operation and compliance by periodic audit through the establishment of two review bodies, as illustrated on Figure XIII-5.

1.0 Plant Operations Review Committee

The Plant General Manager shall appoint PORC members in writing, including the PORC Chairperson and Vice Chairpersons, drawn from the committee members. The PORC maintains written minutes of each meeting, and copies are provided to the Site Vice President, Chairperson of the Nuclear Safety Review Board (NSRB), and the Plant General Manager. Open items shall be assigned, tracked and resolved.

Specific PORC requirements associated with the committee composition and member qualifications, including alternates, and meeting frequency, quorums, and record requirements are contained in the QATR.

1.1 Function

The PORC functions to advise the Plant General Manager on all matters related to nuclear safety and plant operations. PORC meetings include a review of in-house and industry operating experience at the discretion of the Plant General Manager.

2.0 Nuclear Safety Review Board

The NSRB ensures that periodic independent reviews and audits of activities are conducted by qualified individuals free from the pressures of plant operations. The NSRB serves in an advisory capacity to the Chief Nuclear Officer. The NSRB ensures periodic independent reviews and audits of activities, as stated in the facility Technical Specifications and the QATR, are performed. Review of events shall include the results of any investigations made and the recommendations resulting from such investigations to prevent or reduce the probability of recurrence of the event. Additional review activities by the NSRB should be performed to verify adequate organizational response to adverse performance trends. The NSRB should monitor the results of audits, evaluations, and assessment activities to ensure that items which could affect plant safety are reviewed. The NSRB may delegate review functions to subcommittees that may include NSRB members, provided that the subcommittees report the results of their reviews to the NSRB.

Specific NSRB requirements associated with the committee composition and member qualifications, including alternates, and meeting frequency, quorums, and record requirements are contained in the QATR.

2.1 Function

The NSRB shall function to provide independent review and audit of designated activities in the areas of:

1. nuclear power plant operations
2. nuclear engineering
3. chemistry and radiochemistry
4. metallurgy
5. instrumentation and control
6. radiological safety
7. mechanical and electrical engineering
8. quality assurance practices
9. other appropriate fields associated with the unique characteristics of the nuclear power plant

3.0 Review of Operating Experience

Internal and external operating experience is reviewed and assessed via corrective action procedures to ensure that information pertinent to plant safety is supplied to Operators and other appropriate personnel, and is used for effecting design and procedural changes to correct generic or specific deficiencies and to enhance plant safety when warranted.

An initial applicability review of externally-generated operating experience shall be performed primarily by individuals in the Assessment and Corrective Action group. These reviews include, but are not limited to, NRC issuances such as Generic Letters (GL), Information Notices (IN), Bulletins, and

Administrative Letters; INPO issuances such as Significant Operating Experience Reports (SOER), Significant Event Reports, Significant Event Notifications (SEN), Significant by Others (SO), and Operations and Maintenance Reminders (O&MR); Vendor issuances such as General Electric (GE) Service Information Letters (SIL), Rapid Information Communication Service Information Letters (RICSIL), Technical Information Letters (TIL), Service Advisory Letters (SAL), and potential 10CFR21 notifications.

External operating experiences that require further evaluation are assigned to responsible Station organizations, via the Condition Report (CR) process, as appropriate, for evaluation and corrective and preventive action. The evaluations and dispositions are reviewed by the applicable Department Manager and the Plant General Manager when PORC review is required. Hardware and software modifications, procedure revisions, design changes, etc., resulting from the reviews are then implemented by the responsible groups. The evaluations and dispositions are reviewed by PORC as required by the Plant General Manager.

In-house operating experience, such as significant equipment malfunction, adverse trends developed from testing and operations surveillance, reactor core operating trends, operability problems, and/or organizational and programmatic problems that may impact plant safety and reliability, will be treated as an event/deviation and processed accordingly. Processing shall be accomplished by the appropriate Department Manager allowing the Plant General Manager to designate PORC review as appropriate.

TABLE XIII-1

ANSI STANDARD CROSS-REFERENCE UNIT 1

ANSI N18.1-1971 TITLE	SECTION	NMPNS TITLE (UNIT 1)
Plant Manager	4.2.1	•Plant General Manager
Operations Manager**	4.2.2	•Manager Operations** (Also 4.3.1, See NOTE 2)
Maintenance Manager	4.2.3	•Manager Maintenance
Technical Manager (See NOTE 1)*	4.2.4 N/A*	•General Supervisor System Engineering (Also 4.6.1) •General Supervisor Chemistry •Manager Engineering Services •General Supervisor Radiation Protection*
Supervisors Requiring NRC License (See NOTE 4)	4.3.1	•Shift Manager •Control Room Supervisor •General Supervisor Operations** (Also 4.3.2, See NOTE 2) •Manager Operations** (Also 4.2.2, See NOTE 2)
Supervisor Not Requiring NRC License	4.3.2	•General Supervisor Operations** (Also 4.3.1, See NOTE 2) •General Supervisor Maintenance Support •General Supervisor Mechanical Maintenance •General Supervisor Equipment Reliability •General Supervisor Electrical Maintenance (Also 4.4.2) •General Supervisor I&C Maintenance (Also 4.4.2) •General Supervisor FIN •General Supervisor Engineering Programs •General Supervisor Design Engineering •Supervisor Maintenance Programs •Supervisor Mechanical Maintenance •Supervisor Electrical Maintenance •Supervisor I&C Maintenance (Also 4.4.2) •Supervisor Component Specialist •Supervisor Engineering-Nuclear Fuels (Also 4.4.1) •Supervisor Radiological Materials Processing •Supervisor Engineering •Fire Marshal
Reactor Engineering and Physics	4.4.1	•Supervisor Engineering-Nuclear Fuels (Also 4.3.2)
Instrumentation and Control	4.4.2	•General Supervisor I&C Maintenance (Also 4.3.2) •Supervisor I&C Maintenance (Also 4.3.2)
Radiochemistry	4.4.3	•Supervisor Chemistry Operations •Supervisor Chemistry Support
Radiation Protection	4.4.4	•Supervisor RP Operations •Supervisor Radiological Engineering •Supervisor RP Support
Operators	4.5.1	•Chief Reactor Operator •Reactor Operator •Plant Operator •Associate Plant Operator
Technicians	4.5.2	•Chemistry Analyst •Chemistry Technician •Prin/Chief Chemistry Technician •Senior Chemist •HVAC Technician •I&C Technician-Nuclear •RP Technician •Utility Technician

TABLE XIII-1 (Cont'd.)

ANSI N18.1-1971 TITLE	SECTION	NMPNS TITLE (UNIT 1)
Repairmen	4.5.3	•Electrician •Machinist Nuclear •Mechanic Nuclear
Engineer in Charge	4.6.1	•General Supervisor Design Engineering (also 4.2.4)
Staff Specialists (See NOTE 3)***	4.6.2	•Senior Plant Health Physicist •Principal Plant Health Physicist •Senior Engineer •Engineer •Associate Engineer •Senior Engineer Analyst •Engineer Analyst •Principal Engineer •Senior Component Analyst •Component Analyst •Fire Protection Analyst***

NOTES:

(1) Manager Radiation Protection meets or exceeds the qualifications of Regulatory Guide 1.8, September 1975, per Technical Specifications Section 6.3.

(2) For Unit 1, as a minimum either the Manager Operations or the General Supervisor Operations shall hold a Senior Reactor Operator License.

(3) For Unit 1 Fire Protectin Analyst (Appendix R Engineer, Fire Protection Engineer, and Fire Protection Engineer qualified, as defined in UFSAR Appendix 10A, Fire Hazards Analysis), personnel qualifications shall meet those defined in Appendix 10A, Section 2.1.1.2.

(4) The education and experience eligibility requirements for Licensed Operator applicants, and changes thereto, shall be those previously reviewed and approved by the NRC; specifically, those referenced in letter NMP1L 2184, dated December 20, 2007, and described in applicable Station training procedures.

TABLE XIII-2

MINIMUM SHIFT CREW COMPOSITION

Position	Operating Mode				
	License Requirements	Normal Operation	Reactor Startups	Shutdown Condition	Operation Without Process Computer[1]
SM	Senior Operator	1	1	1[3]	1
CRS/STA[4]	Senior Operator	1	1	1[2]	1
Licensed Operator	Operator	2	3	2[2]	2
Non-Licensed Operator	--	2	2	2	3

NOTES:

1. For operation longer than 8 hr without the process computer.
2. Hot shutdown condition only. For cold shutdown and refueling conditions, only one Senior Operator and one Operator are required to be on shift.
3. An additional Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities shall directly supervise all core alterations.
4. Normally the Control Room Supervisor is a combined CRS/STA; however, there may be instances when a shift may be staffed by two Senior Reactor Operators plus a dedicated STA.

SENIOR LEVEL STATION MANAGEMENT ORGANIZATION CHART

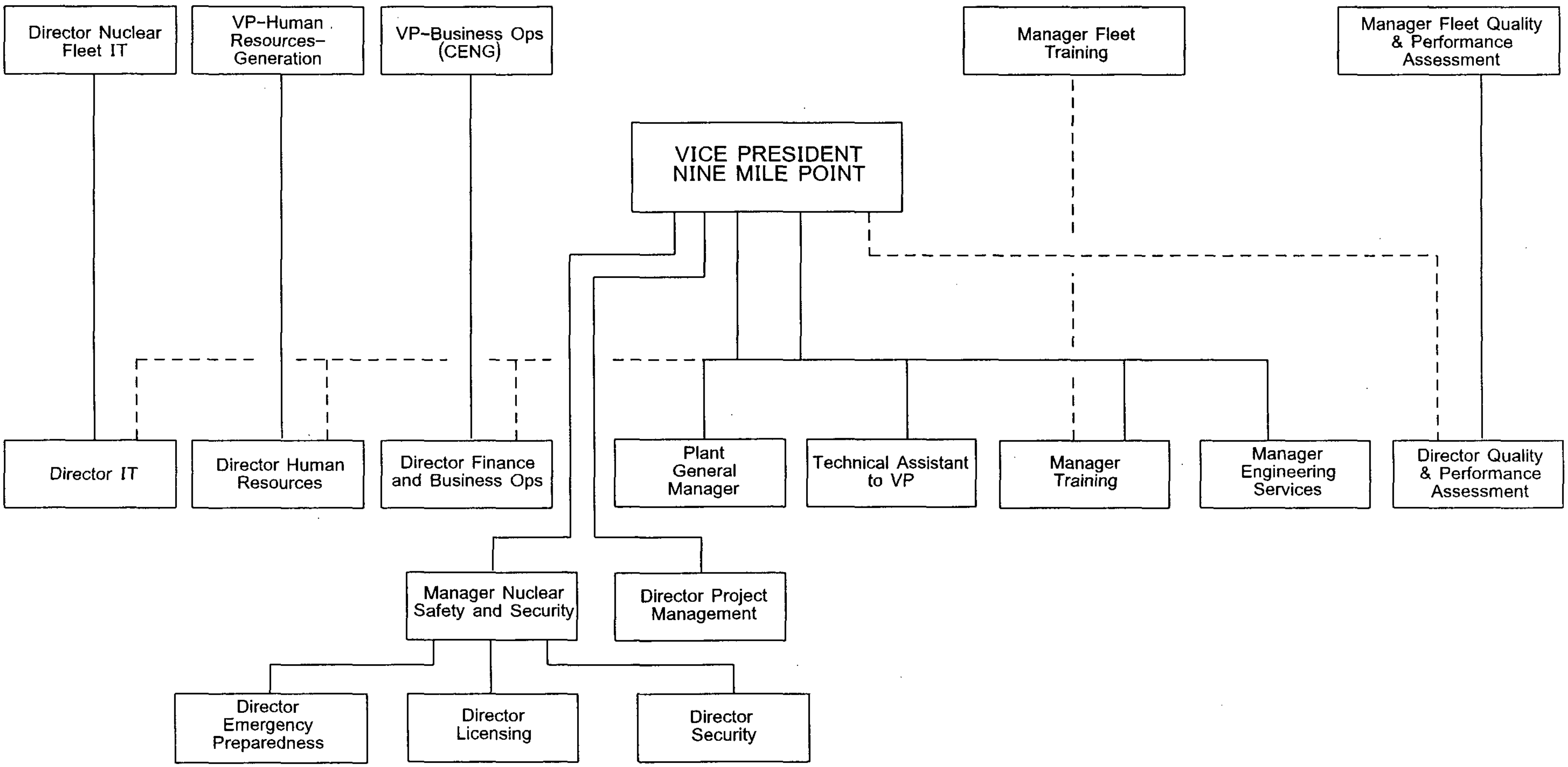

Figure XIII-1
UFSAR Revision 21
October 2009

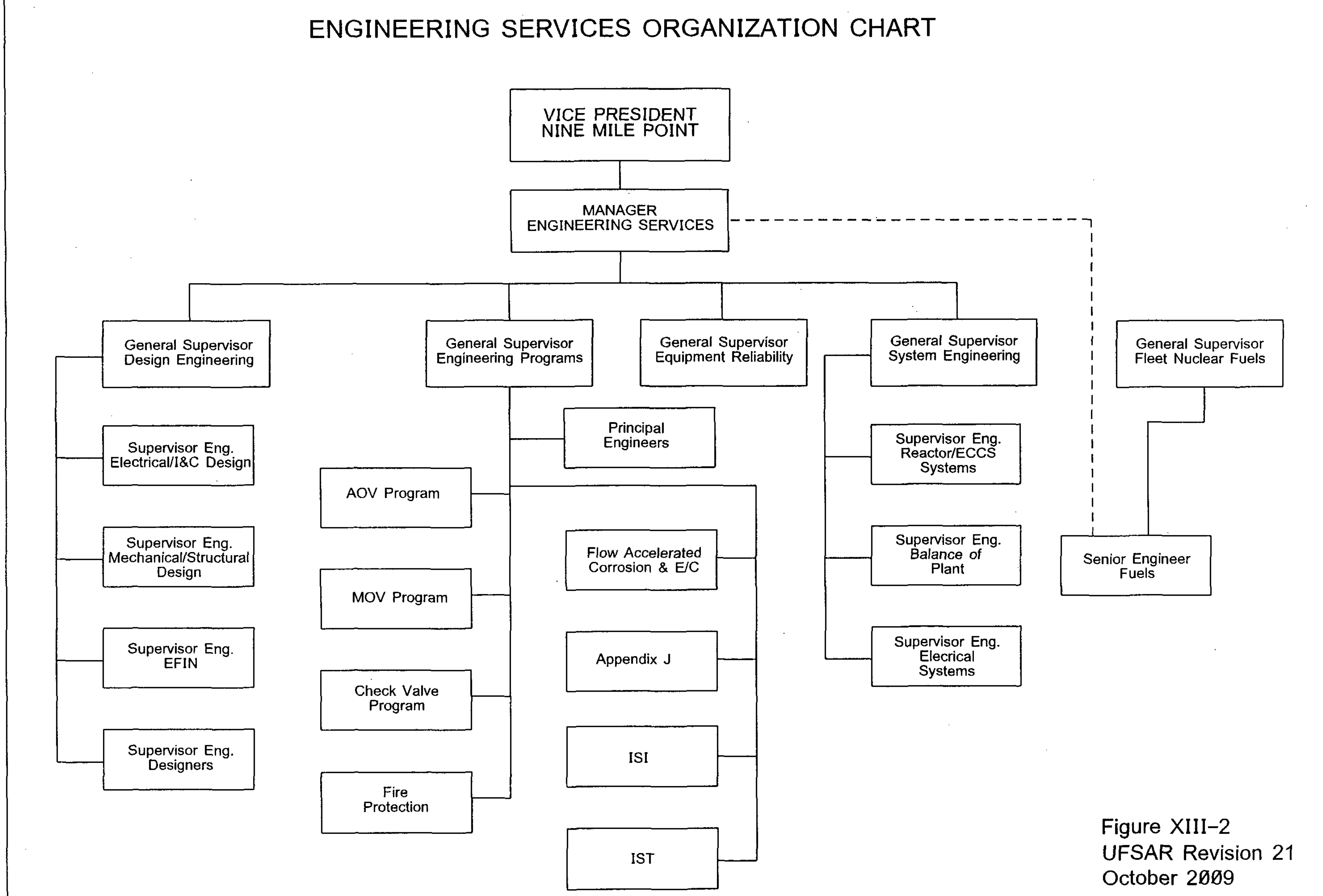

Figure XIII-2
UFSAR Revision 21
October 2009

QUALITY ASSURANCE ORGANIZATION

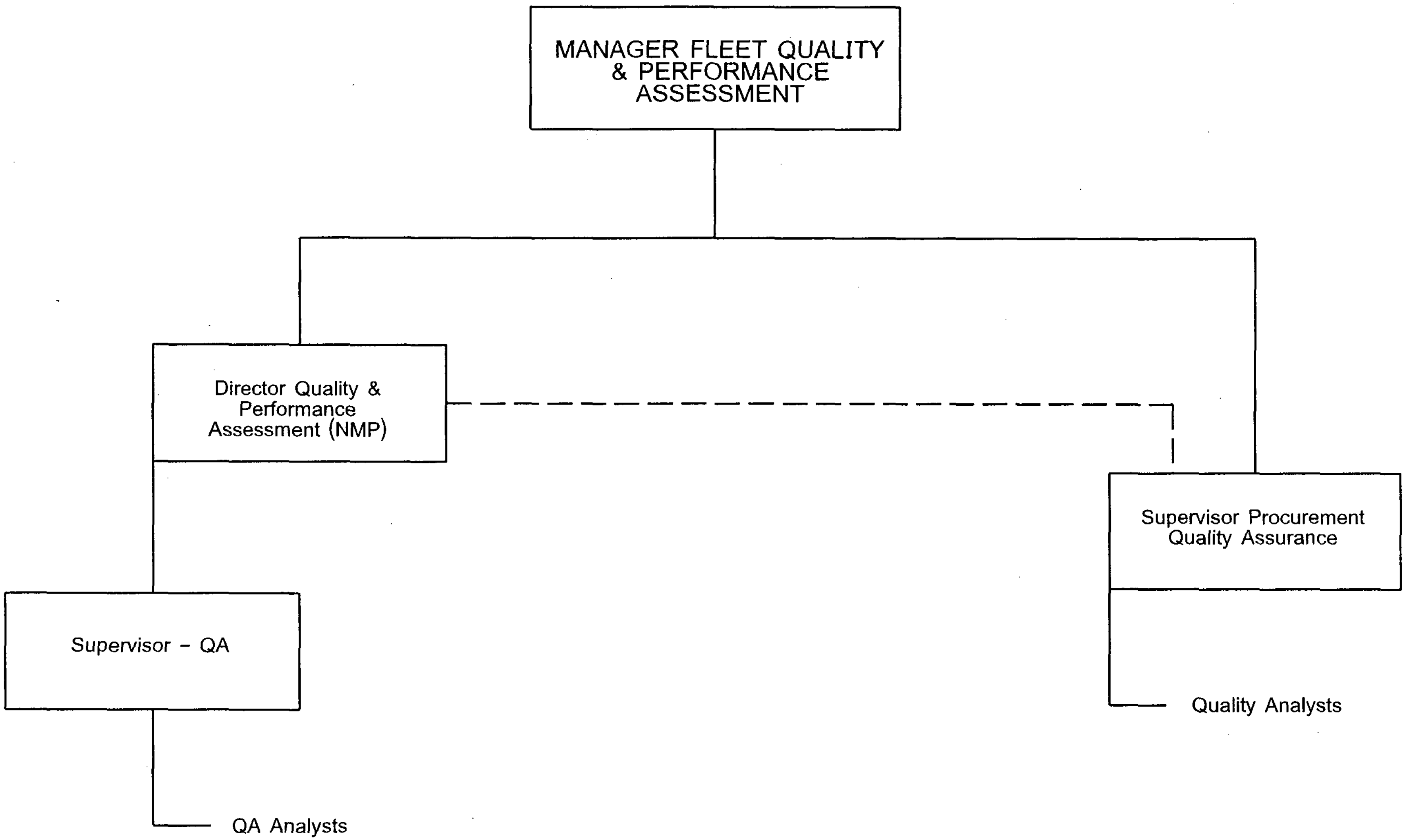

Figure XIII-3
UFSAR Revision 21
October 2009

NUCLEAR SAFETY & SECURITY ORGANIZATION CHART

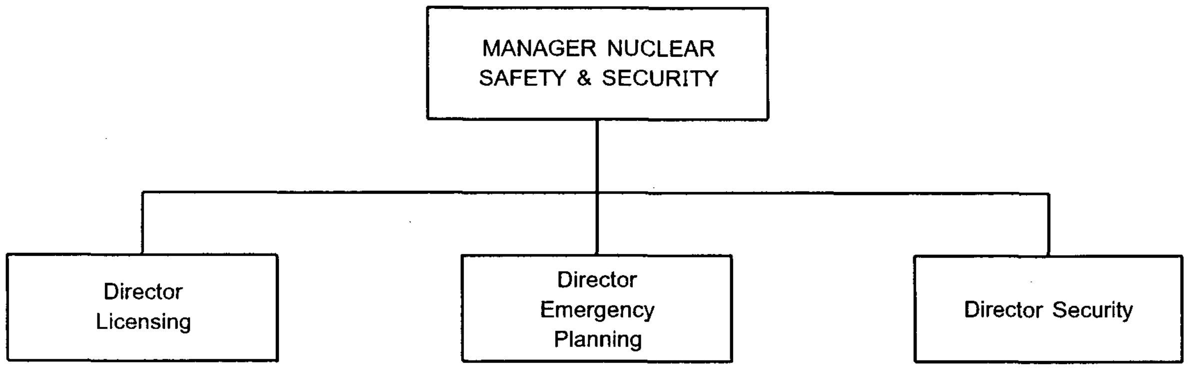

Figure XIII-3a
UFSAR Revision 21
October 2009

NINE MILE POINT NUCLEAR STATION ORGANIZATION CHART

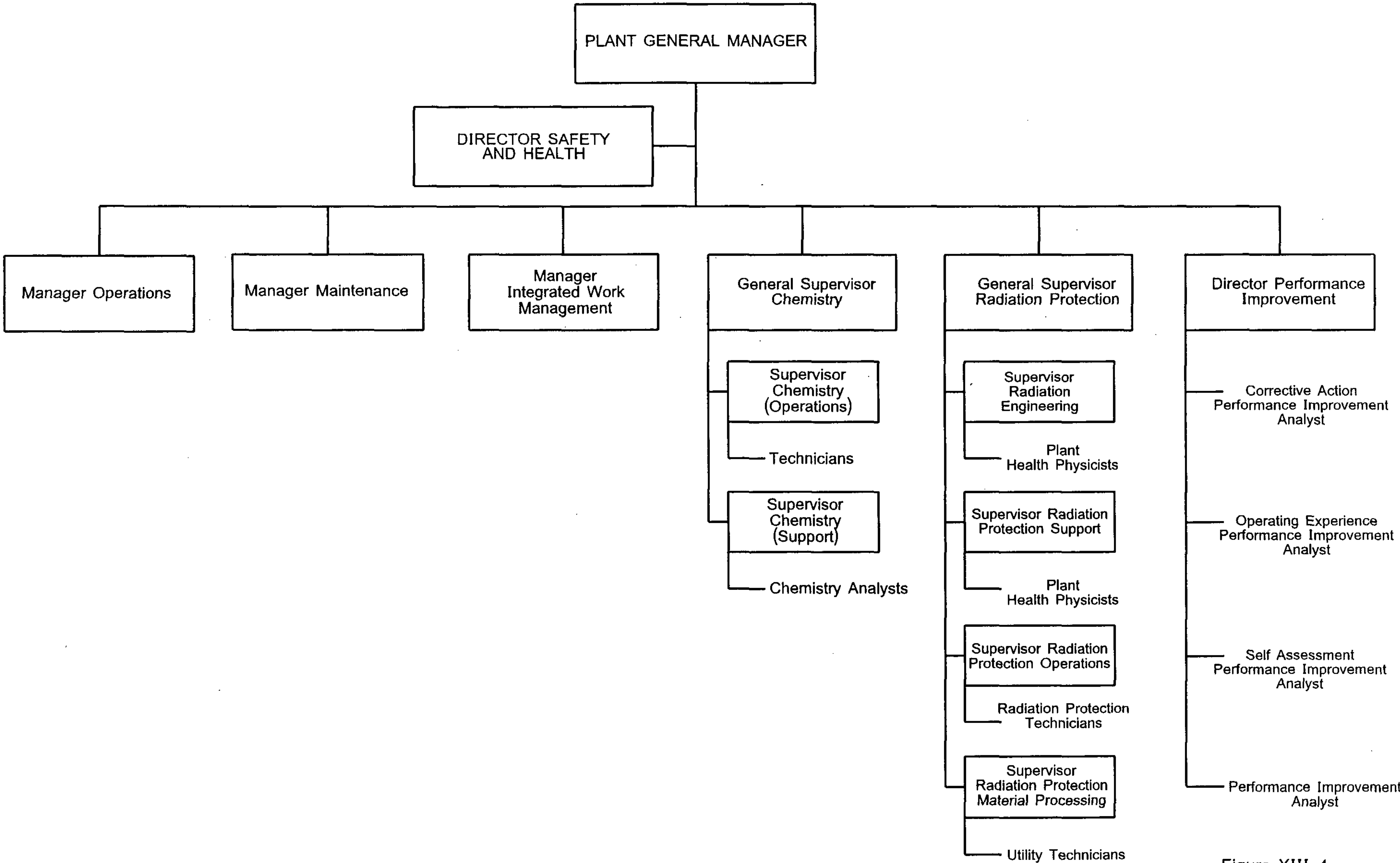

Figure XIII-4
UFSAR Revision 21
October 2009

NINE MILE POINT NUCLEAR STATION ORGANIZATION CHART

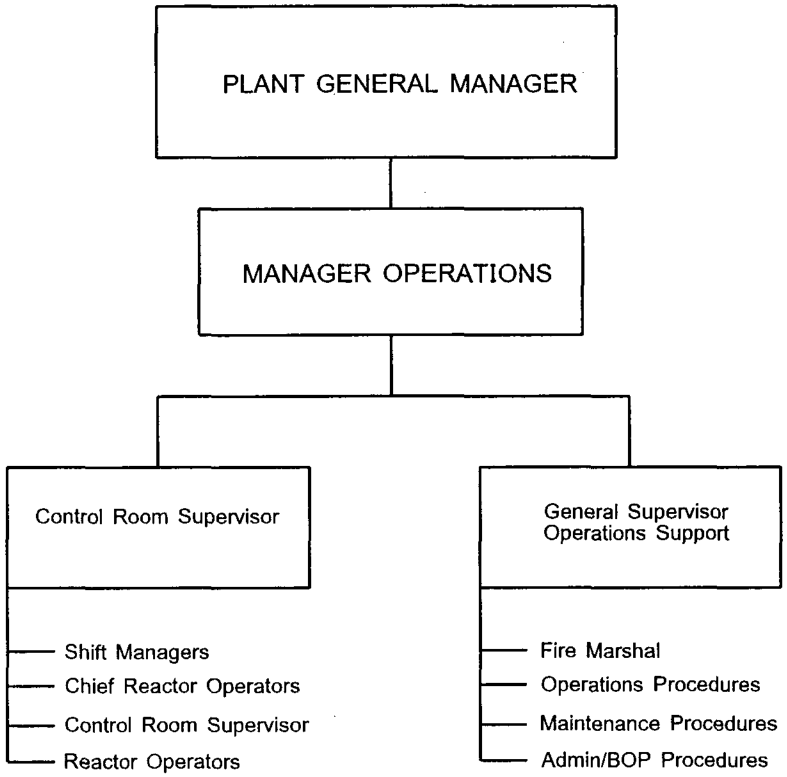

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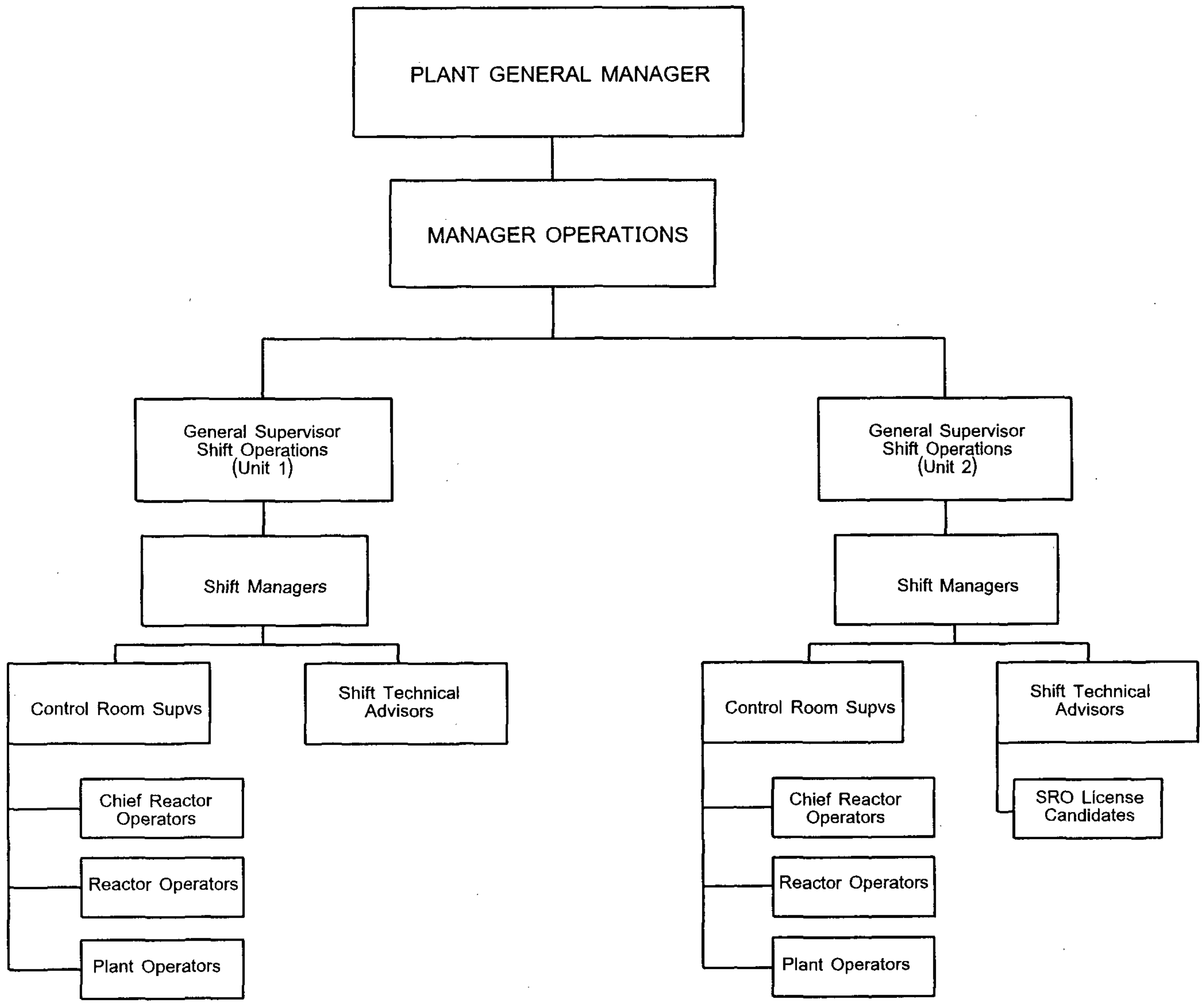
Figure XIII-4a
UFSAR Revision 21
October 2009

NINE MILE POINT NUCLEAR STATION ORGANIZATION CHART



Sh 2 of 2

Figure XIII-4a
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October 2009

NINE MILE POINT NUCLEAR STATION ORGANIZATION CHART

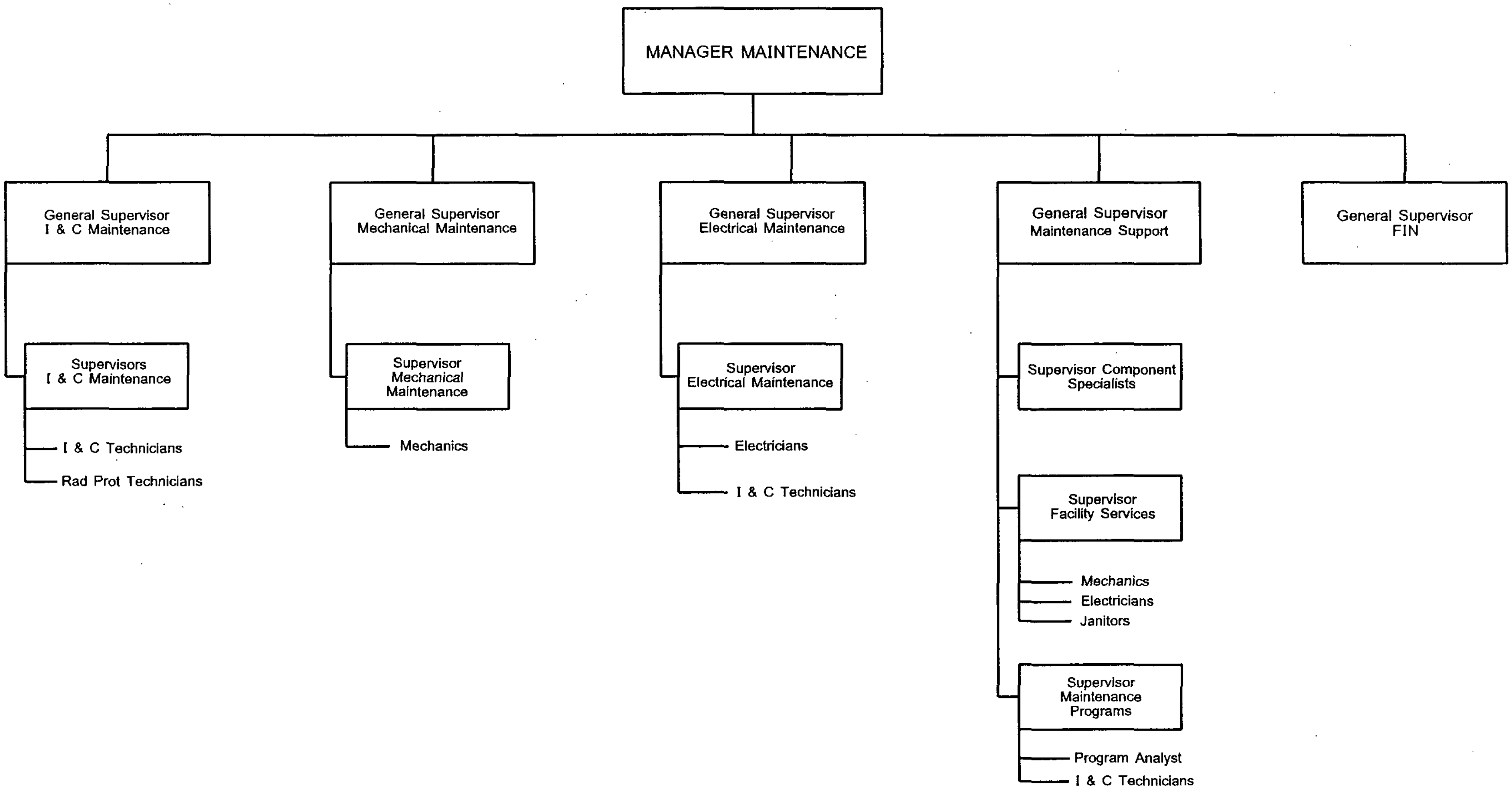

Figure XIII-4b
UFSAR Revision 21
October 2009

NINE MILE POINT NUCLEAR STATION ORGANIZATION CHART

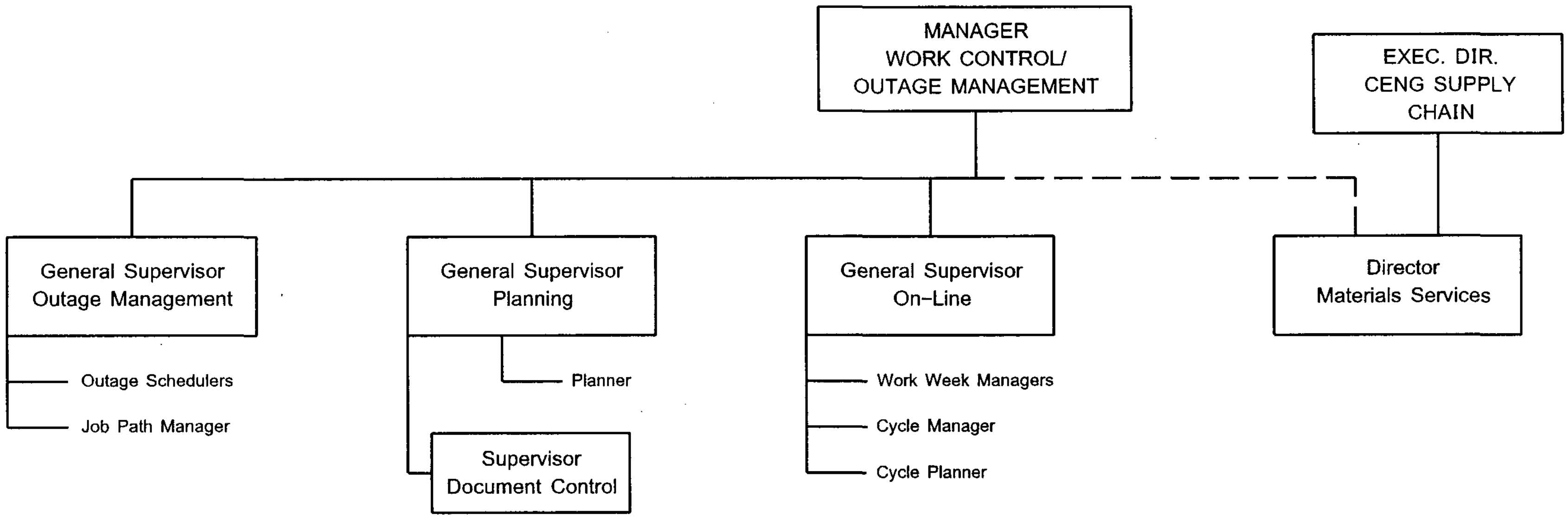

Figure XIII-4c
UFSAR Revision 21
October 2009

SAFETY ORGANIZATION

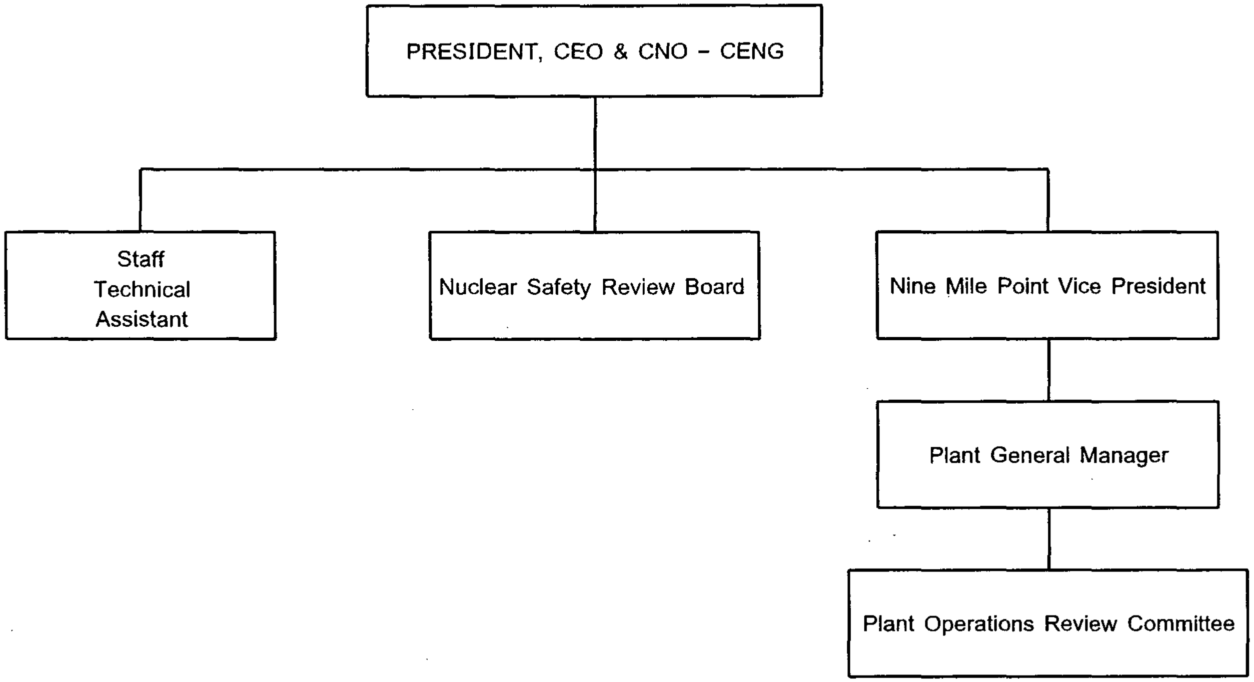

Figure XIII-5
UFSAR Revision 21
October 2009

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

NINE MILE POINT NUCLEAR STATION UNIT 1

FINAL SAFETY ANALYSIS REPORT (UPDATED)

VOLUME 4

OCTOBER 2009

REVISION 21

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subcooling is assumed for this analysis. When 880 psia saturation pressure is reached at about 9.5 sec, the subcooled water begins to flash and noticeably reduces the decompression rate. The coolant blowdown rate is not strongly affected.

1.2.6 System Pressure and Steam-Water Mass

System pressure and associated blowdown rates are determined from mass and energy balances and saturation state relationships. The effects of core heat and feedwater are included. The blowdown characteristics change abruptly when the swelling mixture reaches the steam lines. The lower curve on Figure XV-25 shows a much faster mass loss from the system when mixture blowdown begins.

The blowdown rates upon which Figure XV-25 are based are shown in Table XV-5.

During the blowdown period, until high reactor water level is reached, feedwater flow is at the maximum of 2510 lb/sec. When the mixture reaches the high water level (~0.2 sec), the feedwater control valve starts to close. The valve maximum closure rate is 8 to 10 sec with feedwater flow varying linearly with time. After the valve closes, there is no feedwater makeup to the reactor. When the MSIVs close, reactor water level rapidly drops and feedwater flow is again admitted. The mass of coolant discharged to the turbine building and the net core mass loss are shown on Figure XV-25.

1.2.7 Mixture Impact Forces

Mixture flow in the steam lines causes impact forces (pressures) on the isolation valves during closure. Maximum impact pressure rise for low-quality mixtures is about 200 psi, based on rigid pipe water-hammer analysis. The isolation valves are designed to close against this force.

1.2.8 Core Internal Forces

System decompression and the associated expansion of steam-water causes time-dependent internal forces on various components in the vessel. Internal forces are calculated by an interconnected five-compartment model of the system. Mass, energy, flow rate, and state relationships are resolved to give continuous pressure traces for each compartment. None of the pressure differentials during blowdown impair the ability to scram or to operate the core spray system.

1.3 Radiological Effects

The following analysis is based on the use of the alternative source term (AST) and RG 1.183. The operating experience at Nine Mile Point Nuclear Station - Unit 1 (Unit 1) has consistently shown iodine concentrations in the range 10^{-2} to 10^{-4}. These concentrations are much lower than those used in the AST accident analysis.

1.3.1 Radioactivity Releases

The predominate activity in the discharged coolant would normally be N-16 which would be substantially reduced by decay before the cloud reaches the site boundary. However, this analysis is based on the assumptions described in RG 1.183 for AST, which correspond to the reactor operating with elevated coolant activity due to the presence of fuel defects.

The average concentrations of iodine isotopes determined for the reactor coolant at Dresden Unit I during 1964 are given in Table XV-6.

Based on the experience of similar reactors, the maximum fission product concentrations in the reactor water would occur without the cleanup system in operation when the stack offgas emission is at about 1 curie/sec after 30 min decay. The more realistic concentrations would occur with the cleanup system in operation when the stack offgas emission was at about 0.1 curie/sec after 30 min decay.

For the analysis with AST, the reactor coolant activity is specified in dose equivalent (DE) I-131, which is controlled by the Technical Specifications. The AST analysis assumes the reactor coolant activity is a factor of 20 times the maximum activity concentration allowed during full power operation, which corresponds to 4.0 µCi/gm DE I-131. Table XV-6 shows the AST design basis reactor coolant activity corresponding to 4.0 µCi/gm DE I-131 used in the analysis.

Assuming an 11-sec isolation valve closure time (includes circuit delays and actual closing time), a total of 2,900 curies are carried out of the break including 34 curies of I-131 and 473 curies of I-133. The noble gas activity discharged from the break is ~33 curies (no decay).

Coolant loss is estimated to be 107,150 lb, of which 24.5 percent is reactor steam. Measurements of halogen concentrations in the Dresden Unit I reactor water and condensate show that the steam to water halogen concentration ratio is in the range of 3 x 10^{-5} to 10^{-5}. Therefore, the halogens carried out through the break are essentially those absorbed in the water. For this analysis, it is assumed that all halogens contained in the water which is vaporized on expansion to atmospheric pressure remain with the vapor.

The key inputs and assumptions used in the AST dose analysis are summarized in Table XV-7a. The accident is modeled as an instantaneous release to the environment. Realistically, the activity is released to the environment over a period of time. Calculated release rates, assuming no filtration, are shown in Table XV-7b for 11-sec and 2-hr release durations.

1.3.2 Meteorology and Dose Rates

Meteorology assumed for the AST MSLB accident is discussed in Section XV-C.7. The MSLB accident χ/Q values are shown in Table XV-7a.

The accident is modeled as an instantaneous release to the environment. Activity is transported into the control room assuming an infinite exchange rate with the environment and no filtration. Doses are calculated assuming the dose conversion factors specified in Federal Guidance Reports 11 and 12. Key inputs and assumptions used in the analysis are provided in Table XV-7a.

The resultant doses from the AST MSLB accident are provided in Table XV-8. The accident-specific dose acceptance criteria for the MSLB accident, assuming a pre-accident iodine activity spike and no fuel failure, are a TEDE of 25 rem at the EAB for any 2 hr, 25 rem at the outer boundary of the LPZ, and 5 rem for occupancy of the control room for the duration of the accident as specified in 10CFR50.67 and RG 1.183. The results demonstrate compliance with these acceptance criteria.

2.0 Loss-of-Coolant Accident

2.1 Introduction

Loss-of-coolant accident (LOCA) analyses have been performed for fuel types currently in the plant Core Operating Limits Report (COLR).

In 1985, an Improved LOCA Model Program was jointly undertaken between General Public Utilities (GPU), GE and Niagara Mohawk Power Corporation (NMPC) to reduce conservatism in the BWR/2 LOCA modeling.

The reason for undertaking this program was to improve plant understanding and increase operating flexibility by improving maximum average planar linear heat generation rate (MAPLHGR) limits. The methodology used in the SAFER/CORECOOL[4,5] computer code and methodology previously developed for jet pump plants was modified for use in nonjet pump plants (BWR/2s). The principal areas where SAFER/CORECOOL[4,5] is an improvement over past methods are:

Nodal Representation of the Reactor Vessel The detailed noding of the pressure vessel is improved over the present method.

Hydraulic Calculation Numerous improvements have been made, the most significant being the detailed calculations of countercurrent flow limiting (CCFL) phenomenon.

Core Heat Transfer Realistic heat transfer coefficients are employed for the flow regimens.

The SAFER/CORECOOL[4,5] modeling is used for both best estimate and licensing LOCA modeling. To get licensing values for peak cladding temperature (PCT) and cladding oxidation, Appendix K values for inputs and correlations are used. To ensure that the Appendix K calculations are adequately conservative, it was demonstrated that these calculations bound 95 percent of all

The first impact dissipates 0.80 x 17,000 or 13,600 ft-lb of energy. It is assumed that 50 percent of this energy is absorbed by the dropped fuel assembly and that the remaining 50 percent is absorbed by the struck fuel assemblies in the core. Because the fuel rods of the dropped fuel assembly are susceptible to the bending mode of failure and because 1 ft-lb of energy is sufficient to cause cladding failure as a result of bending, all 63 rods of the dropped 8x8 fuel assembly and all 62 rods of the 8x8R and P8x8R assemblies are assumed to fail. Since the tie-rods of the struck fuel assemblies are more susceptible to bending failure than the other 55 or 54 fuel rods, it is assumed that they fail on the first impact. Thus, 4 x 8 = 32 tie-rods (total in four assemblies) are assumed to fail.

Because the remaining fuel rods of the struck assemblies are held rigidly in place in the core, they are susceptible only to the compression mode of failure. To cause cladding failure of one fuel rod as a result of compression, 250 ft-lb of energy is required. To cause failure of all the remaining rods of the four struck assemblies, 250 x 56 x 4 or 56,000 ft-lb of energy would have to be absorbed in cladding alone. Thus, it is clear that not all the remaining fuel rods of the struck assemblies can fail on the first impact. The number of fuel rod failures caused by compression is computed as follows:

$$\frac{0.5 \times 13{,}600 \times \frac{11}{11 + 17}}{250} = 11 \text{ (8x8, 8x8R, P8x8R)}$$

Thus, during the first impact, fuel rod failures are as follows:

	8x8	8x8R/P8x8R
Dropped assembly	63 rods (bending)	62 rods (bending)
Struck assemblies	32 tie-rods (bending)	32 tie-rods (bending)
Struck assemblies	11 rods (compression)	11 rods (compression)
	106 failed rods	105 failed rods

Because of the less severe nature of the second impact and the distorted shape of the dropped fuel assembly, it is assumed that in only 2 of the 24 struck assemblies are the tie-rods subjected to bending failure. Thus 2 x 8 = 16 tie-rods are assumed to fail. The number of fuel rod failures caused by compression on the second impact is computed as follows:

$$\frac{\frac{0.19}{2} \times 17{,}000 \times \frac{11}{11 + 17}}{250} = 3 \text{ (8x8, 8x8R, P8x8R)}$$

Thus, during the second impact the fuel rod failures are as follows:

Struck assemblies	16 tie-rods (bending)
Struck assemblies	3 rods (compression)

The total number of failed rods resulting from the accident is as follows:

	8x8	8x8R/P8x8R	GE11[8]
First impact	106 rods	105 rods	125 rods
Second impact	19 rods	19 rods	15 rods
Third impact	0 rods	0 rods	0 rods
	125 failed rods	124 failed rods	140 failed rods

3.3 Radiological Effects

3.3.1 Fission Product Releases

Fission Product Release from Fuel

Fission product release estimates have been performed in accordance with the alternative source term (AST) methodology outlined in Regulatory Guide (RG) 1.183. The fission product source term is based on recently irradiated fuel. The source term in the fuel is increased to account for power uncertainties in accordance with the AST methodology. During the accident a number of pins are assumed to be damaged and to release the entire activity in the fuel pellet-to-clad gap region. The non-LOCA gap fractions specified in RG 1.183 are used in the analysis.

Fission Product Inventory in the Reactor Building

All of the noble gas fission products are assumed to be released from the reactor water to the reactor building. The halogens released are absorbed in the pool and evolve from the pool into the air to establish an overall pool decontamination effect.

For the dose consequence analysis based on AST, all of the activity is conservatively assumed to be released instantaneously to the environment without filtration.

Realistically, the activity will migrate from the pool water into the reactor building. As fission products are released to the reactor building, high radiation signals initiate alarms and start the emergency ventilation system. This system maintains the reactor building below atmospheric pressure and discharges a volume equivalent to 100 percent of the building volume per 24 hr through high-efficiency and charcoal filters to the stack. These safety functions are not analyzed or required for the design basis AST FHA analysis.

The airborne fission product inventory in the reactor building is shown in Table XV-22.

<u>Discharge of Fission Products to Atmosphere</u>

As described above, the dose consequence analysis based on AST conservatively assumes all activity is instantaneously released directly to the environment without filtration. The corresponding design basis release rates are shown in Table XV-23. The following description relates to the realistic release of fission products to the environment.

Realistically, the noble gases and halogens are exhausted over a period of time from the reactor building through a dryer, a high-efficiency filter, a charcoal filter and another high-efficiency filter. Because of the relatively small heat and vapor input the building remains at relatively low temperature and humidity. The building exhaust is treated so that humidity is reduced and filter efficiency is maintained. Calculated release rates via the stack, but assuming no filtration and a 2-hr duration, are shown in Table XV-23.

<u>AST Fission Product Release</u>

Key fission product release inputs and assumptions for the AST based FHA analysis are shown in Table XV-25. The AST analysis is based upon a conservative instantaneous release model directly to the environment. The AST analysis does not credit secondary containment or the filtration provided by the reactor building emergency ventilation system (RBEVS).

3.3.2 Meteorology and Dose Rates

Meteorology assumed for the AST FHA accident is discussed in Section XV-C.7. The FHA accident X/Q values are shown in Table XV-25.

The accident is modeled as an instantaneous release to the environment. Activity is transported into the control room assuming an infinite exchange rate with the environment and no filtration. Doses are calculated assuming the dose conversion factors specified in Federal Guidance Reports 11 and 12. Key inputs and assumptions used in the analysis are provided in Table XV-25.

The resultant doses from the AST FHA accident are provided in Table XV-24. The accident-specific dose acceptance criteria for the FHA accident are a total effective dose equivalent (TEDE) of 6.3 rem at the EAB for any 2 hr, 6.3 rem at the outer boundary of the LPZ, and 5 rem for occupancy of the control room for the duration of the accident as specified in 10CFR50.67 and RG 1.183. The results demonstrate compliance with these acceptance criteria.

4.0 Control Rod Drop Accident

4.1 Identification of Causes

The accidental removal of a control rod from the core at a more rapid rate than that which can be achieved by the CRD system results in a power excursion. A fully-inserted control rod is assumed to become disconnected from its drive. The drive is then fully withdrawn and, subsequently, the control rod falls out of the core.

The severity of the resulting excursion is reduced by strict procedural controls, supplemented by use of a rod worth minimizer (RWM). 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 a rod drop accident were to occur. The severity is further reduced by limiting the maximum "dropout velocity" of any control rod with the rod velocity limiter.

4.2 Accident Analysis

CRDA results from banked position withdrawal sequence (BPWS) plants have been statistically analyzed and documented in Reference 9. The results show that, in all cases, the peak fuel enthalpy in a rod drop accident would be much less than the 280 cal/gm design limit even with a maximum incremental rod worth corresponding to 95-percent probability at the 95-percent confidence level. Based on these results, it was proposed to the NRC, and subsequently found acceptable, to delete the CRDA

from the standard GE boiling water reactor (BWR) reload package for the BPWS plants.

Because of the large margin available to CRDA design limits for BPWS plants, implementation of the advanced physics methods[10] does not result in challenging the 280 cal/gm limit. Therefore, the impact of using the advanced physics methods of Reference 10, as compared to the physics methods described in Reference 11, on the generic BPWS analysis is considered negligible.

4.3 Designed Safeguards

The control rod system is designed to minimize the probability of blades sticking in the core. The blades of the control rods travel in gaps between the fuel channels with approximately 1/2-in clearance and are equipped with rollers which make contact with the channel walls. Since a control blade weighs approximately 220 lb, even if it separates from its drive, gravity forces would tend to make the blade follow its drive movement as if it were connected.

The control rod coupling to the drive index tube significantly reduces the probability of an accidental separation of a control rod from its drive. Couplings of this design have undergone extensive tests under simulated reactor conditions and also at conditions more extreme than those expected to be encountered in reactor service. They have been operated through thousands of cycles of scram operation and a separation has never occurred. Tests have shown that the coupling will not separate when subjected to pull forces up to 20 times greater than can be applied with a CRD.

Movements of the control rods, when the reactor is critical or near critical, cause changes in the neutron flux. Control rod coupling can be verified by observing the neutron flux changes during rod movement.

A velocity limiter which adds substantial hydraulic drag against downward control rod movement is incorporated in the design. Testing and analysis of the velocity limiter has demonstrated a maximum rod drop velocity of 3.11 fps[53].

4.4 Procedural Safeguards

Operating procedures require that control rod movements follow preplanned patterns to flatten the power distribution. A rod withdrawal procedure, incorporating the BPWS and a reduced notch worth procedure[12] (which minimizes the chance of short period scrams to an even greater extent than required for fulfilling CRDA safety requirements), forces adherence to certain constraints applied to all control rod withdrawals (and insertions) between 100-percent control rod density (all control rods inserted) and 10 percent of design rated power, in order to limit incremental control rod worths. A description of the BPWS and reduced notch worth procedure is given in References 13 and 12, respectively.

Operating procedures require rod following verification checks during startup and during major rod movements, and frequent verification checks on all rods not fully inserted, to assure that any rod-from-drive separation is detected. Procedures require the full insertion of rods when following is not verified.

After full withdrawal from the core, a control rod sits on a seal. Procedurally, the Operator attempts to withdraw each rod to a further overtravel position. If the drive is coupled to the control rod blade, the overtravel position cannot be attained. If the drive is uncoupled, the overtravel position is reached and an indicator light warns the Operator. The drive would then be immediately reinserted to prevent possible fallout of the stuck control rod blade. This method is used on fully withdrawn control rods during reactor startup when control rod following is not verified by observing the response of the neutron flux instrumentation.

4.5 Radiological Effects

The following radiological consequences are based on two release pathways, one through the mechanical vacuum pumps and the other through the main condenser based on the assumption of manual isolation of the MSIVs. Dose calculations for the CRDA do not indicate a need for mechanical vacuum pump line isolation. However, the capability was provided to automatically isolate the mechanical vacuum pump line on high radioactivity in the MSLs. As a result, dose calculations were not reevaluated based on mechanical vacuum pump line isolation; releases would be considerably less since the major pathway for radioactivity has been removed. The dose consequence analysis for the CRDA is based on the AST and RG 1.183.

4.5.1 Fission Product Releases

Fission Product Release from Fuel

A maximum of 10 percent of the noble gas activity and iodine activity in a fuel rod are released from the rods experiencing cladding perforation. Except for cesium and rubidium, release of solids is negligible. These conservative gap release fractions are based on AST and RG 1.183 and are the design basis for the AST dose analysis. All release fractions used in the AST dose analysis are shown in Table XV-26. Realistic estimates of maximum plenum activity based on measured activity releases from fuel with failed cladding in operating reactors[14,15] are a maximum of one percent noble gas, 0.5 percent halogen and negligible solids. The fission products generated by the excursion are negligible compared to those already in the fuel due to the long-term reactor operation.

Fission Product Transport

Fission product transport assumptions are in accordance with AST and RG 1.183. Two cases are analyzed. The first case corresponds to a ground-level release directly from the turbine/main condenser. The second case assumes mechanical vacuum pumps are in operation resulting in an elevated release from the main stack.

At hot standby, the pressure regulator maintains reactor pressure constant by bypassing steam to the condenser. A little over two full-power seconds of energy are produced in the excursion, of which less than 3 percent are released promptly and the rest released according to the relatively slow conduction heat transfer time constant of 8 to 9 sec, characteristic of UO_2 fuel rods. Therefore, the increase in steam flow to the main condenser is handled by the turbine bypass system without a significant pressure transient in the reactor or in the condenser.

A fraction of the fission products released from the perforated fuel rods are carried through the MSLs to the condenser. The activity is monitored in the MSLs, and alarmed in the control room upon a high activity signal. Position switches on the steam line isolation valves also actuate reactor scram when the valves are partly closed.

Normally, the air ejector offgas system maintains condenser vacuum and the airborne and noncondensible fission products are carried into the offgas piping. High radiation signals isolate this piping from the stack. If the radiation is not intensive enough to cause isolation of the offgas piping and the Operator fails to isolate manually, the fission products are released to the stack, after a 30-min delay, at rates below those permitted by 10CFR20. During hot standby, the mechanical vacuum pumps are used instead of air ejectors. With the reactor isolated, the mechanical vacuum pump is not automatically isolated and continues to operate. The mechanical vacuum pump flow rate (2000 cfm) is higher than the offgas flow rate. The delay in the offgas holdup piping of about 30 min is considerably longer than the 1.75-min delay that occurs in the piping from the mechanical vacuum pumps. For this reason, the accident is analyzed assuming that the high flow rate mechanical vacuum pumps are operating.

Fission products released from the fuel are spread from the reactor vessel, steam line piping, turbine and condenser to the vacuum pump system. Even though some of the released noble gases are absorbed in the reactor water, all are assumed to pass to the turbine condenser system before closure of the MSIVs.

The AST analysis assumes that all of the activity released from the failed fuel can become airborne. There is no assumed flashing fraction or partition fraction. All of the iodine that reaches the condenser is assumed to be either organic or elemental iodine. The release fractions are listed in Table XV-26.

The partition factor (concentration in water/concentration in steam) for halogens has been measured from 3×10^4 to 10^5. These measurements were made at Dresden Unit I at operating power with full steam flow and voids. The measurements demonstrate that even at high flow saturated steam conditions with a high steam void content, the halogens are absorbed in the water and remain there. The halogen concentrations in the reactor coolant water and in the condenser hotwell water were measured. The ratio of halogen concentration in the reactor water to that in the condensate is the decontamination ratio, 3×10^4 to 10^5.

The halogens are assumed to be dispersed in that amount of water which passes through the reactor during the 11 sec required for isolation valve closure.

The water carry-over fraction to the turbine at rated power is normally less than 10^{-3}. During and following the excursion, the steaming rate (including that due to decay heat and excursion energy), remains well below rated, such that the carry-over fraction is less than 10^{-3}. However, in the AST analysis the release fractions assumed are in accordance with RG 1.183 and are listed in Table XV-26. Of the activity released to the water, 10 percent of the iodine and one percent of the other fission products are assumed to reach the condenser.

Stack Release

The noble gases mix with any gases in the condenser vapor space and are removed by the vacuum pump; associated stack releases are given in Tables XV-27 and XV-28.

The halogens reaching the condenser are absorbed in the hotwell condensate. An equilibrium is established between the halogens in this water and the halogens in the condenser vapor space. In the AST analysis, 10 percent of the iodine and one percent of the other fission products in the condenser are assumed to be available for release. The condenser vapor space is about 8.5 x 10^4 ft^3 and the water volume is about 1.0 x 10^4 ft^3. The condenser volume containing the source term is 5.0 x 10^4 ft^3. The vacuum pump is operating at rated flow (2000 cfm). There are no filters in this line and holdup in the piping is only 1.75 min.

4.5.2 Meteorology and Dose Rates

Activity is transported into the control room assuming an infinite exchange rate with the environment and no filtration. Doses are calculated assuming the dose conversion factors specified in Federal Guidance Reports 11 and 12. Key inputs and assumptions used in the analysis are provided in Table XV-26.

The resultant doses from the AST CRDA are provided in Table XV-29. The accident-specific dose acceptance criteria for the CRDA are a TEDE of 6.3 rem at the EAB for any 2 hr, 6.3 rem at the outer boundary of the LPZ, and 5 rem for occupancy of the CR for the duration of the accident as specified in 10CFR50.67 and RG 1.183. The results demonstrate compliance with these acceptance criteria.

5.0 Containment Design Basis Accident

Three containment analyses are presented in this section:

1. The original recirculation line rupture with core spray analysis is discussed in Section XV-C.5.1. This analysis evaluates the chronological events occurring and the response of Station systems, and serves as the basis for the environmental qualification of equipment located inside the drywell.

2. The original containment DBA analysis is discussed in Section XV-C.5.2. As stated in Section XV-C.5.2, its purpose is to provide the basis for the containment leakage rate limits, assuming failure of all core spray systems.

3. The design basis reconstitution (DBR) analysis of the long-term post-LOCA suppression chamber temperature response is discussed in Section XV-C.5.3. The DBR analysis verifies that the containment design basis heat removal requirements are satisfied at the maximum containment spray raw water temperature. The containment leakage rate design basis established in Section XV-C.5.2 is not altered by the DBR analysis.

A new structural analysis was performed as a result of the Mark I Containment Program, which included the effects of loads not previously accounted for. Modifications were performed that restored the original margin of safety. The Mark I Containment Program is further discussed in Section VI-A.

5.1 Original Recirculation Line Rupture Analysis - With Core Spray

5.1.1 Purpose

The full range of coolant loss accidents has been analyzed, from a small rupture where the makeup flow is greater than the coolant loss rate, to the largest, a highly improbable circumferential recirculation line break. The analysis shows that the circumferential recirculation line break (26-in diameter) results in the maximum fuel temperature and containment pressure. Because the small breaks result in longer times for blowdown and subsequent core heatup, the potential for termination of the is caused by the slow rate of power decay. Core heat release is predicted in the core heatup calculation with the core spray system functioning. After 1 hr, all of the fuel rods are wetted and the core is quenched to saturation temperature at containment pressure. Decay heat is then the only heat source. The temperatures of the drywell and suppression chamber are shown on Figure XV-56G for this case.

5.1.7 Blowdown Effects on Core Components

Pressure differences across structures and members in the reactor vessel during the blowdown are determined to assure that control rod insertion can be accomplished. Of primary concern are the forces on the control rod guide tubes below the core and on the fuel channels which guide the blades into the core. The guide tubes are designed for about 100 psi pressure differential compared to the transient peak pressure difference of about 35 psi developed during the blowdown. The transient forces last but a few seconds and are not of sufficient magnitude to interfere with rod insertion, since the large scram forces developed by the drive assure insertion should any interference develop.

The most likely place at which interference might occur is in the blade space between channels. The maximum transient pressure difference across the channels varies from under 20 psi at the bottom to essentially zero at the top. This force is in a direction which could cause pinching of the blade. In that portion of the channel below the tip of a partially-inserted blade, the channel can only move until it comes into contact with the blade. This deflection is not sufficient to cause permanent distortion so that the channel springs back when the transient force decreases. Hence, no binding exists in that region of the channel except for a second or two during the transient.

However, for the portion of the channels above the control blades, some yielding of the channel walls occurs. The blade must then force the walls apart as it moves upward. Calculations are performed conservatively assuming that the transient peak pressure difference, which occurs across the channel at the bottom, is a steady force on the entire channel. The net normal force acting on each of the rollers is then calculated. Assuming only sliding could take place and using a coefficient of friction of unity, the total upward force required to force the walls apart is only 440 lb per blade. The CRD mechanism is characterized by high forces when scrammed. At zero reactor pressure, a drive develops a force of 6000 lb to insert the rod using the energy stored in the accumulator. The effect of the accumulator decreases as reactor pressure increases, but at a reactor pressure of 1000 psi there is still approximately 3000 lb at the beginning of the scram stroke, which is well in excess of the 400 lb calculated above. The drive can also be scrammed by reactor pressure alone. When the vessel is above 800 psig, the force exerted from this energy source is approximately 1100 lb throughout the scram.

5.1.8 Radiological Effects

The radiological consequences due to a LOCA are analyzed in accordance with the AST methodology as per RG 1.183. The acceptance criteria are defined by 10CFR50.67. Although the scenario presented in this section considers core spray operation, the AST analysis performed considers core damage in excess of cladding perforation.

5.1.8.1 Fission Product Releases

Release of Fission Products from the Fuel

Prior to the accident the reactor is assumed to be operating at full power. The core source inventory from which the AST LOCA analysis is based is shown in Table XV-30. The release from the fuel occurs over two phases. At 2 min after initiation of the event, the gap activity is released from the fuel over a period of 30 min. The gap release is followed by the early in-vessel phase which releases a significantly larger amount of activity over a period of 90 min. The fractions associated with the release from the fuel are shown in Table XV-31. The activity released from the fuel is directly released to the drywell in accordance with the AST methodology.

Fission Product Release from the Drywell Directly to the Environment

There are two pathways for release from the drywell directly to the environment during the LOCA using the AST methodology. The first pathway is permanent bypass leakage through several piping lines containing containment isolation valves. Leakage through these lines could bypass the reactor building and RBEVS filters resulting in a ground-level release. These lines include MSIV leakage and combined leakage from feedwater, torus vent, drywell vent, and emergency condenser vent and drain line isolation valves.

The second pathway for release from the drywell directly to the environment occurs at the beginning of the event prior to establishing a sustained negative pressure in the reactor building. During this drawdown period, the release is assumed to be directly to the environment as a ground-level release. The release during drawdown is assumed to occur due to the Technical Specifications primary containment leak rate and ESF leakage.

Key parameters related to bypass leakage are shown in Table XV-31.

Fission Product Release from Drywell to the Reactor Building

The activity released to the primary containment is subsequently assumed to be released to the reactor building at the maximum rate allowed by the Technical Specifications. Additionally, activity that is assumed to be released to the suppression pool is assumed to leak to the reactor building through ESF system leakage. Key parameters related to the leakage from the drywell to the reactor building are shown in Table XV-31.

Discharge of Fission Products from Reactor Building to Atmosphere

After a sustained negative pressure in the reactor building is established, activity released to the reactor building is transported to the environment by way of the RBEVS and the plant stack. It is assumed that RBEVS initiates automatically and provides particulate and halogen filtration. Key parameters related to the leakage from the reactor building to the environment are shown in Table XV-31.

Fission Product Transport and Removal

The release fractions, leakage rates and timing are summarized in Table XV-31. The AST analysis assumes five key fission product removal mechanisms.

1. Spray removal in the drywell

2. Natural deposition in the drywell

3. Main steam line sedimentation

4. Suppression pool iodine retention

5. Filtration

Prior to the onset of the gap release phase, the drywell sprays are assumed to automatically initiate. Aerosol removal due to the drywell sprays is evaluated using the proprietary STARNAUA computer code and four system-related parameters as code inputs. These four parameters are droplet size, spray flow rate, spray fall height, and sprayed volume. The maximum elemental iodine removal rate is limited to 20 hr^{-1} in accordance with SRP 6.5.2.

The proprietary STARNAUA computer code is also used to model the natural deposition in primary containment and the gravitational settling or sedimentation credited in the main steam line bypass leakage.

By crediting the liquid poison system capability to introduce sodium pentaborate into the reactor coolant within 1.5 hr to act as a buffer, the post-accident pH of the suppresion pool will remain above 7 for the duration of the accident. Therefore, AST assumptions regarding iodine retention in the suppression pool are valid and iodine re-evolution is not considered.

Filtration provided by the RBEVS and CRATS is assumed to reduce the organic, elemental and particulate activity at the dose receptor locations. The filtration efficiencies assumed in the analysis are shown in Table XV-31.

The fission product transport is analyzed using the RADTRAD computer code and the AST methodology. The calculated activity in the drywell following the LOCA is shown in Table XV-29b. The calculated activity in the reactor building following the LOCA is shown in Table XV-29c. The activity released to the environment that results in dose to the control room personnel and offsite is shown in Table XV-29d.

5.1.8.2 Meteorology and Dose Rates

Activity is transported offsite and into the control room assuming the parameters identified in Table XV-31 and using the RADTRAD computer code. Doses are calculated assuming the dose conversion factors specified in Federal Guidance Reports 11 and 12.

The resultant doses from the AST LOCA are provided in Table XV-32. The accident-specific dose acceptance criteria for the LOCA are a TEDE of 25 rem at the EAB for any 2 hr, 25 rem at the outer boundary of the LPZ, and 5 rem for occupancy of the CR for the duration of the accident as specified in 10CFR50.67 and RG 1.183. The results demonstrate compliance with these acceptance criteria.

5.2 Original Containment Design Basis Accident Analysis - Without Core Spray

5.2.1 Purpose

The purpose of this analysis is to provide the basis for the containment leakage rate limits, assuming a recirculation line break and failure of all core spray systems. Failure of the core spray system results in a metal-water reaction with generation of hydrogen. One of the two containment spray loops (primary or secondary) is assumed to be operative.

5.2.2 Core Heatup

After the blowdown, no coolant is assumed to flow into the core except sufficient water to support the metal-water reaction. Cooling of the core is, therefore, limited to this water flow and the resulting hydrogen flow. The core cladding temperature rises (see Figure XV-57) due to decay heat to about 2000°F. At this temperature the metal-water reaction rate, as predicted by Baker[33], begins to add appreciable energy to the heatup, and as temperatures rise higher, the metal-water energy controls the heatup. The reaction terminates because the zircaloy melts, runs down the hot fuel surfaces, falls through the end plate of the fuel bundle, and into the water below the core where it is quenched. Water is expected to be present at the bottom of the vessel, entering either through the core spray system, the feedwater system or CRD system. The same variables are shown on Figure XV-57 as are shown when a core spray system functions. The response of the core in both cases is the same for the first 30 sec, until the core spray system is actuated. If neither core spray system functions, clad temperatures continue to rise as shown. After about 50 sec, the temperature gradient across the fuel approaches a quasi-equilibrium and further temperature rise is due to decay heat and metal-water reaction only. As the cladding temperature continues to rise, the rate of metal-water reaction accelerates. Having conservatively assumed that sufficient water is available throughout the core to support the predicted rate of metal-water reaction, clad temperature rises at an increasing rate until the melting temperature of the cladding is reached. The molten cladding falls from the core and the heatup of that portion of the cladding is eliminated.

The cumulative metal-water reaction taking place during this transient is 24.5 percent. This is the total reaction including fuel cladding and fuel channels but not including the additional reaction of the molten clad material that takes place upon quenching of the molten drops.

From Figure XV-57, approximately 90 percent clad melting and 90 percent of the total metal-water reaction takes place in the first 1800 sec. The figures are used to estimate the energy and hydrogen release rates to the containment system.

Calculations of the droplet diameter as the molten metal falls from the core, based on surface tension and the fuel end plate dimensions, give droplet sizes in the range of 3/8 in in diameter. Molten drop reaction rates in water temperatures of interest indicate that a reaction depth of 60 microns correlates with observed droplet reaction test data.[34] Application of the 60-micron reaction depth to the calculated mean droplet diameter results in a 4-percent reaction of the molten zirconium leaving the core. Thus, a total estimated 24.5 percent in core (Figure XV-58) and 3 percent, i.e., 0.04 x (100 - 24.5), postmelt reaction results in the total of 27.5-percent reaction in a minimum time of 30 min.

Zirconium rod meltdown tests have been conducted to determine the range of droplet sizes leaving the molten core. These tests were done in a test assembly with simulated fuel end plates and four induction-heated zirconium rods. One test with nine rods and an actual end plate was also conducted giving similar results. All these experiments show that a normal statistical distribution of droplet sizes ranges from a minimum of 0.137-in diameter, with a mean diameter of 0.269 in. Application of the 60-micron reaction depth to the various-sized drops shows an overall reaction of 5 percent, not appreciably different from that of 4 percent calculated initially. In addition, the tests clearly show that the molten drops are cooled by the water, thus terminating any further reaction.

A description of these tests is included in APED-5454. The percent of fuel rods perforated and the percent of the fuel which is above the recrystallization temperature as a function of time are shown on Figure XV-59.

5.2.3 Containment Response

One of two containment spray loops, either the primary or secondary, is assumed to function. The core heatup results are used to determine the amount of hydrogen and energy generated and released to the containment. Uniform release rates of hydrogen and energy are assumed to occur over 1800 sec. This corresponds to the time required for 90 percent of the fuel cladding to reach melting temperature (see Figure XV-57). All the hydrogen resulting from a 27.5-percent reaction of cladding and channels is assumed released. All of the resulting chemical energy as well as the decay energy and the original sensible energy in the core are also released during this time. As a result, the containment pressure rises rapidly to 25 psig at 1800 sec. After 1800 sec the hydrogen release stops and the energy release falls to decay power level. Consequently, the containment spray loop is able to quickly cool the gases in the system, sharply reducing pressure. After 2000 sec the containment pressure response is similar to the case for which the core spray system functions, except that the pressure is approximately 11 psi higher. The 11 psi difference is the result of the hydrogen generated.

The temperature variations with time of the drywell and suppression chamber are shown on Figure XV-60.

5.3 Design Basis Reconstitution Suppression Chamber Heatup Analysis

This DBR analysis considers containment spray system operation at up to a maximum containment spray raw water temperature of 84°F.

5.3.1 Introduction

The DBR program analyzed the long-term containment suppression chamber response following the containment DBA. The containment DBA, described in Section VI-B.1.2, is identified as the instantaneous rupture of the reactor coolant system (RCS) corresponding to a double-ended break of the largest pipe in the containment (coolant recirculation line).

The DBR long-term containment suppression chamber response analysis(35) was performed consistent with the LOCA, described in Section XV-C.2.0, which assures that 10CFR50.46 limits are not exceeded. The Section XV-C.2.0 LOCA analysis is based on the loss of offsite power (LOOP), the single failure of one of the emergency diesel generators, and the dynamic effects of the postulated pipe break, which result in one core spray pump set available to provide core cooling. Therefore, the DBR analysis of the suppression chamber response considers core spray available and assumes less than 1 percent metal-water reaction consistent with the LOCA analysis and 10CFR50.46 limits.

The design basis requirement for the containment spray system is to assure that the primary containment design pressure and temperature limits are not exceeded. In addition, the containment spray heat removal system must maintain the torus water temperature such that adequate net positive suction head (NPSH) is provided to the core spray pumps and containment spray pumps, assuming no increase in containment pressure from that present prior to the postulated LOCA.

The DBR analysis of the containment heat removal design basis for the containment spray system provides a working model to assess system performance and operability since the original calculations were not available. The DBR analysis[35] methodology produces conservative results as compared with the original design basis analysis (Sections XV-C.2.0 and XVI-C.2.0). The DBR analysis results require that the heat removal requirements be increased, as compared with those described in Section VII-B, to assure the design basis requirements are satisfied. The increased heat removal requirements are necessary to maintain the DBR analysis conservative, as compared to the calculations described in Sections XV-C.2.0 and XVI-C.2.0.

The DBR analysis evaluates the containment suppression chamber response assuming the containment spray system is operated in the drywell and wetwell spray mode. Additional analyses verify that operating the containment spray system in accordance with the emergency operating procedures (EOPs) creates conditions which are bounded by the spray mode of operating the containment spray system.

5.3.2 Input to Analysis

A list of significant input parameters to the DBR suppression chamber heatup analysis is presented in Table XV-32a. The method-specific inputs are discussed in Section 5.3.3.2.

5.3.3 DBR Suppression Chamber Heatup Analysis

The DBR suppression chamber heatup analysis[35,60] determines the maximum torus water temperature which is expected to occur following the containment DBA. This analysis is intended to reconstitute the design basis for the containment spray system, such that the performance requirements for operation up to a maximum containment spray raw (lake) water temperature of 84°F can be assessed. This analysis does not supersede the design basis analysis discussed in Section VI-B.1.2.

The Reference 35 analysis has been reanalyzed with new containment spray heat exchanger heat removal rates (K-value) in Reference 60. The revised analysis also includes different modeling assumptions on vessel pressure used for the post-blowdown break flow calculations and a different modeling of the vessel liquid and metal sensible energy. The Reference 60 analysis also includes ANS 5.1-1979 (nominal) decay heat data consistent with the Reference 35 analysis but with additional actinides and activation products included per GE Service Information Letter (SIL) 636.

5.3.3.1 Computer Codes

The original calculations and/or computer analyses used to determine the design basis heat removal requirements for the containment spray system were not described in the FSAR and are not available. The DBR program chose to perform a new analysis using GE's proprietary computer code, SHEX-04. SHEX is designed to model long-term containment pressure and temperature responses to a variety of normal and abnormal operating transients, including LOCAs. SHEX-04 has been applied by GE-Nuclear Energy in this type of analysis and has been reviewed and accepted by the NRC.[35]

SHEX-04 evaluates the containment response by performing mass and energy balances on four main nodes: reactor pressure vessel (RPV), drywell, suppression pool and wetwell airspace. These nodes are interconnected via one or more of the auxiliary systems; e.g., the drywell and the suppression pool are connected by the downcomers; the suppression pool and the RPV are connected by the core spray system; the drywell and wetwell airspace are connected by the wetwell to drywell vacuum breakers, etc. External mass and energy sources such as decay heat and feedwater are added to the system.

The results predicted by this computer code are conservative when compared with the results of the original analysis performance assumptions based on the results of cases 1 and 2 of the Reference 35 analysis.

The SHEX code has been revised for the Reference 60 analysis to allow the vessel pressure modeling described in Section 5.3.3.2. However, the methods applied for this analysis are consistent with the basic GE methodology used in long-term LOCA containment analyses. The changes to the Reference 35 analysis are for inputs and modeling assumptions and do not represent any change in the methodology.

5.3.3.2 Analysis Methods

The model used in this analysis includes the RPV, drywell, wetwell (including the suppression pool), core spray system, containment spray system, feedwater, safety relief valves (SRV), main turbine, torus vents and downcomers, the drywell to wetwell vacuum breakers, and the wetwell to reactor building vacuum breakers.

6.2 Accident Analysis

Analysis of the mislocated bundle accident is performed for reload cores where the resultant CPR response may establish the operating limit MCPR.

Analysis methods for the misoriented fuel assembly are discussed in detail in Reference 38. Approval of these methods is given in Reference 39 under the stipulation that a ΔCPR penalty of 0.02 be added for the tilted misoriented bundle. This 0.02 is added onto the calculated ΔCPR used in determining the operating limit when utilizing this method. The fuel cladding integrity safety limit is applied to the accident results reported in the SRLR[2].

The mislocated bundle analysis employs a statistically corrected Haling procedure and analyzes every bundle in the core. A statistical comparison of actual process computer CPR data, with Haling power distribution fuel bundle CPR predictions, is performed. Using the operating data, it is possible to perform a Haling power distribution calculation for determining the fuel bundle CPRs, and then to correct them to achieve an improved prediction of the actual CPR in each fuel bundle. A detailed description of this procedure is presented in Reference 38.

Fuel loading errors could result in fuel failures during Station operation. The most severe fuel failure mechanism would result from rods in a fuel assembly experiencing transition boiling resulting in clad overheating and subsequent accelerated oxidation of the cladding. The consequence of accelerated clad oxidation ultimately is cladding perforation and release of stored and generated fission products. The level of release would be in the same order of magnitude as other known fuel perforation mechanisms (PCI and hydride), and thus would not be distinguished from other fuel perforations possibly in the core.

Fuel failures are detected by increased amounts of noble gas measured in the offgas system. However, Technical Specifications limit the offgas release rate (continuously monitored) and coolant activity concentrations (periodic sampling and analysis) to insure that guidelines on accidents and normal operation radiological consequences are met. Should rods in a misloaded fuel assembly fail in a more severe manner than rods which fail from normal operation, the Technical Specification limits would effectively limit operation of the plant.

6.3 Safety Requirements

Proper location and orientation of fuel assemblies can be readily verified by visual means and verification procedures to greatly reduce the possibility of a fuel bundle loading error. The fuel assembly loading error is classified as an accident, not a transient, so application of LHGR limits is not appropriate.

The fuel bundle loading error analysis results presented in the SRLR[2] show that the MCPR will be greater than the safety limit MCPR for all exposures throughout the cycle.

7.0 Meteorological Models Used in Accident Analyses

7.1 Introduction

Radiological consequences of the Unit 1 DBAs are based on atmospheric dispersion factors (χ/Q values). Using site meteorological data, calculations were performed to obtain the associated χ/Q values. These calculations used data collected by the NMPNS onsite meteorological measurements program for the 5-yr period from 1997 through 2001.

7.2 Atmospheric Dispersion Factor Calculations

Meteorological data utilized for calculation of χ/Q values were selected from the historical record of the NMPNS meteorological monitoring program. The period from 1997 through 2001 was selected because it represents a complete and accurate data set that is representative of the site meteorological data. The data was reviewed to ensure instrumentation problems and missing or anomalous observations did not affect the validity of the data. This is consistent with the guidance in RG 1.194 that considers 5 yr of hourly observations to be representative of long-term trends.

Recorded meteorological hourly average data were used to generate joint frequency distributions of wind direction, wind speed, and atmospheric stability class, in accordance with RG 1.23 and 1.145.

Three possible locations where accident radionuclide releases are assumed to occur are the reactor building blowout panel, the turbine building blowout panel, and the main stack. Information regarding these release points and their proximity to receptor locations is provided in Tables XV-34a and XV-34b.

7.2.1 Offsite - EAB and LPZ

The computer program PAVAN is used to determine χ/Q values for the assessment of dose consequences of design basis accidents. The program implements the NRC guidance provided in RG 1.145.

Utilizing joint frequency of occurrence distributions of wind direction, wind speed, and Pasquill atmospheric stability class, χ/Q values were obtained as a function of direction for various time-averaging periods at the EAB and the outer boundary of the LPZ. Analyses were made from assumed ground-level (i.e., non-elevated) releases (such as vents and building penetrations), which are less than 2.5 times the height of adjacent solid structures, and from elevated releases (i.e., stacks). Three procedures were utilized for determining χ/Q values for a direction-dependent approach, a direction-independent approach, and an overall site approach.

The reactor building blowout panel, the turbine building blowout panel, and the main stack are the assumed accident release points. The reactor and turbine building blowout panel locations do not qualify as elevated releases as per RG 1.145. Therefore, these release points were modeled as ground-type releases. The main stack was executed as an elevated release. Source-to-receptor horizontal distances are 830 m (2,722 ft) for the EAB and 6,116 m (20,060 ft) for the LPZ. Due to the close proximity of the three release points, identical distances to the EAB and LPZ were used.

NMPNS meteorological data from the 5-yr period from 1997 through 2001 was used in the analysis. Since the NMPNS meteorological data fails to provide a maximum wind speed for category 12 winds, a conservative value of 60.5 m/s was selected. The coastal sectors were not considered in determining the χ/Q values for the EAB and LPZ.

7.2.2 Control Room and Technical Support Center (Excluding MSLB)

Control Room and TSC χ/Q values were calculated using ARCON96 for various source/receptor scenarios using the procedures contained in RG 1.194. The scenarios were analyzed using the hourly-averaged meteorological joint wind and stability database for the 5-yr period from 1997 through 2001. All three of the assumed release points (the reactor building blowout panel, the turbine building blowout panel, and the main stack) were modeled as ground-level (vent) releases in accordance with RG 1.145

because their height is less than 2.5 times the highest adjacent structure. Geometry of the Unit 1 structures was used in the models to account for wake effects.

7.2.3 Control Room - MSLB Puff Release

The MSLB accident evaluation utilizes an instantaneous "puff" release χ/Q. The puff release is modeled in accordance with RG 1.194, Section C.5, with the following assumed site meteorological conditions:

- Wind speed: 1 m/s toward the receptor; and
- Stability class: F

The distance from the turbine building blowout panel (the assumed MSLB release point) to the control room intake is 71.9 m (236 ft). There is one air intake location. It takes approximately 136 sec for the puff to pass completely over the Unit 1 control room air intake. The control room air intake flow rate is the same during normal control room ventilation operation and emergency control room ventilation operation. As such, the control room air intake flow rate is modeled as a constant flow rate during the entire time that the MSLB puff release passes over the intake.

7.3 Summary of Results

The χ/Q values resulting from the ARCON96 modeling analysis of each release point and meteorological database scenario for the required time intervals are shown in Tables XV-35a and XV-35b for the control room and TSC dose assessments, respectively.

The χ/Q values for the EAB and LPZ calculated by the PAVAN modeling analysis of each release scenario are presented in Tables XV-35c and XV-35d for each of the time intervals required by RG 1.145.

For the MSLB instantaneous puff release, the integrated χ/Q value calculated for the control room air intake is 9.979E-04 sec/m^3.

7.4 Exfiltration

Pressure differential from the outside to the inside of the reactor building results in exfiltration from the building. If this occurs, radioactivity release bypasses the particulate and halogen removal equipment in the emergency ventilation system, and enters the atmosphere at essentially ground elevation instead of through the stack.

The pressure distribution about the reactor building due to wind velocity is estimated, based on the model studies of Irminger and Nøkkentved(40) and later work by Jensen and Franck(41). Based on these pressure distributions and the design rate of leakage from the building without the ventilation system in operation, it is estimated that an exfiltration rate of 50 percent per day of the reactor building volume occurs with winds of 35 mph, increasing to 100 percent per day at 50 mph. However, due to dispersion characteristics, the worst dose at the site boundary occurs for winds of approximately 11 mph. See PHSR, Volume II, Appendix A, subsection 3.52 for building wake dilution. Based on that, the ground concentration at the site boundary in units/cc for each unit/sec emitted is 1.655×10^{-11} at 11 mph.

Halogens and particulates are emitted in greater quantity than during stack release due to bypassing of the removal equipment for these fission products in the emergency ventilation system.

Analyses that evaluate the effect of wind speed and the shape of the reactor building on leakage from the reactor building are presented in the following paragraphs.

In any nonzero wind, some nonuniform pressure distribution is established around the building. The distribution and magnitude of the pressure depends on wind speed and direction, turbulence, and building configuration.

Using the design building configuration and dimensions, an attempt was made to relate the design building to model studies conducted by others. After a suitable model was found, the worst conditions of wind turbulence and direction were determined. In

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differential between internal static pressure and atmospheric static pressure is more negative than Figure XV-63 for a stack leakage of 100 percent per day, the building will leak less than the design point.

Preferential leakage analysis of building exfiltration was also performed, using the following as possible leakage paths:

1. Panel-to-panel joints

2. Panel-to-roof joints

3. Panel-to-concrete joints

Figure XV-65 shows the reactor building plan indicating these leakage paths. Details of the sections are shown on Figures XV-66 through XV-69. The only difference between this and the aforementioned analysis is the inclusion of leakage through the roof perimeter, which is at a more negative pressure distribution than any other part of the building. Building leakage is considered to be proportional to crack length. The most probable leakage would be through the panel-to-panel joints rather than through the other two paths.

Figures XV-70 and XV-71 show the results of this analysis and the relationship to the previous analysis, which tends to show less leakage than this revised analysis. Table XV-36 shows the length and area quantities used in the analysis.

The model used for the analysis has been previously referenced in the above paragraph.[42] Another model study by Pagon[46] revealed less severe pressure distributions than the study by Jensen and Franck.

For reactor building leakage tests, the assumption is that on the day of the leakage test the wind is northerly at 12 mph. The Technical Specifications require that the building internal differential pressure be at least as negative as shown on Figure XV-72, which is based on a southerly wind. This results in the most severe pressure in the reactor building, but is the case of least leakage. Since there is a northerly wind, the actual pressure curve for the design building (100 percent per day at 0 mph and 0.25 in negative pressure) exhibits less negative pressure at the same wind speeds and stack exhaust rates. Figure XV-72 shows, at the pressure curves for northerly and southerly winds, point D; that the design leakage L_D = 100 percent per day. In a northerly wind the building pressure would follow curve "a" to point A where L_A = 100 percent per day. However, to meet the pressure of Figure XV-72, flow must be increased to about 108 percent per day, which equals leakage at B (L_B = 108 percent per day). The Technical Specifications require extrapolation back to 0 mph along curve "b" indicating L_D = 108 percent per day.

However, since there is a northerly wind, extrapolation back to 0 mph should be along curve "c" (parallel to curve "a"), which gives L_C = 108 percent per day at 0.27 in negative pressure. But the design point is 0.25 in negative pressure. Therefore, L_D = (-0.25/-0.27) L_C = 100 percent per day. Thus, by extrapolating along curve "b" instead of curve "c", an error of 8 percent would be introduced if the wind were from the north. If the wind were from the south, no error would be introduced. Since wind direction probably would not remain the same throughout the test, other pressure curves between curves "a" and "b" could introduce an error ranging from 0 to 8 percent. This is conservative since the indicating leakage, as extrapolated back to the 0 mph design point, would always exceed or equal the actual leakage if the actual pressure curve for the test wind direction were used.

7.5 Secondary Containment Drawdown

7.5.1 Introduction

The AST LOCA analysis considers the reactor building positive pressure period. This is defined as the period when a loss of offsite power (LOOP) causes a loss of reactor building negative pressure relative to the external atmospheric static pressure. The start of the emergency diesel generators followed by the start of the RBEVS returns the reactor building to a negative pressure. The time of positive pressure relative to the atmospheric status pressure is called the drawdown time. The post-LOCA primary containment leakage into the reactor building is assumed to be released directly to the environment during the drawdown period.

7.5.2 Analysis

The drawdown calculations were performed using the GOTHIC 7.2a(QA) containment analysis software. In the calculations, each building's elevation was considered as well as buoyancy effects, natural circulation flow paths, and building heat sinks.

The following conservative conditions were included in the analysis:

- LOOP and failure of one of the two 100-percent capacity RBEVS trains to operate (i.e., only a single RBEVS train operates).

- Maximum reactor building in-leakage allowed by Technical Specification 3.4.1 of 1,600 cfm.

- Design basis post-LOCA reactor building heat loads, including maximum post-LOCA suppression pool heatup, operation of two core spray pump sets and one containment spray pump set, heat loads from the emergency condensers on the refuel floor elevation and from the spent fuel pool (assumed to be at a constant 90°F based on manual restart of a spent fuel pool cooling pump), electrical heat loads from equipment required to operate to mitigate the LOCA, and solar heat loads.

- Winter atmospheric conditions based on onsite meteorological data collected for the 5-yr period of 1997 through 2001 (consistent with the guidance provided in RG 1.183, Appendix A, Section 4.3). The use of summer conditions results in a drawdown time that is approximately one half that of the winter case and thus is less limiting.

7.5.3 Results

The results of the analysis (illustrated on Figures XV-73 and XV-74) show an initial rapid rise in reactor building pressure. The reactor building pressure in the area above the refuel floor elevation (el. 340 ft) remains positive for approximately 26 min, decreases to -0.15 in WG at approximately 67 min, and reaches -0.25 in WG at approximatey 5 hr. At elevations below the refuel floor, the positive pressure times and the times to achieve -0.25 in WG are considerably shorter. For example, at the 318 ft elevation (upper), the reactor building pressure remains positive for approximately 18 min and decreases to -0.25 in WG at approximately 52 min.

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TABLE XV-5

BLOWDOWN RATES

Time	Flash	Average Blowdown Rate (lb/sec)	Average Enthalpy (Btu/lb)
0 - 1.8 sec	---	3,360	1190
1.8 - 9.5	---	11,000	675
9.5 - 11.0	Subcooled Water	11,000	670

TABLE XV-6

REACTOR COOLANT CONCENTRATIONS (μCi/gm)

Isotopes	Dresden I 1964	NMP Unit 1 Cleanup System in Operation	AST MSLB Analysis
I-131	.025	.03	0.7
I-133	.100	.20	9.7
I-132	.25 (I-132, I-134, I-135)	1.33 (I-132, I-134, I-135)	13.0
I-134			23.6
I-135			9.7
Other Fission Products	.25	.14	.25 (Alkali Metals) 13.5 (Noble Gases)

TABLE XV-7

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TABLE XV-7a

MSLB ACCIDENT ANALYSIS INPUTS AND ASSUMPTIONS

Parameter	Value
Reactor Coolant Activity, DE I-131	
Equilibrium	0.2 µCi/gm
Pre-accident Spike	4.0 µCi/gm
Failed Fuel	None
Break Isolation Time	11 sec
Mass Released	107,150 lbm
Steam Fraction	24.5%
Holdup Credit	None
Radioactive Decay	None
Fission Product Removal	None
Dose Conversion Factors	FGR 11 & 12
Control Room Intake Flow	Infinite Exchange Rate
Control Room Volume	135,000 ft^3
Filtration	None
Atmospheric Dispersion, χ/Q	
EAB	1.90×10^{-4} sec/m^3
LPZ	1.63×10^{-5} sec/m^3
CR	9.98×10^{-4} sec/m^3

TABLE XV-7b

MSLB ACCIDENT RELEASE RATES

Release Duration	Iodines (Ci/sec)	Noble Gases (Ci/sec)	Alkali Metals (Ci/sec)
11 sec	250	2.99	11.1
2 hr	0.383	0.00457	0.0170

TABLE XV-8

MAIN STEAM LINE BREAK ACCIDENT DOSES

Receptor	Total Dose (rem TEDE)	Acceptance Criteria (rem TEDE)
Control Room	1.76	5
EAB*	0.530	25
LPZ	0.0450	25

* Worst 2-hr period of the accident duration

TABLE XV-21a

ANALYSIS ASSUMPTIONS FOR NINE MILE POINT 1 CALCULATIONS

		Nominal	Appendix K
1.	Decay Heat	1979 ANS	1971 ANS + 20% Reference 4
2.	Transition Boiling Temperature	Iloeje Correlation	300°F
3.	Break Flow	1.25 HEM (SUB) HEM (SAT)	Moody Slip
4.	Metal-Water Reaction	Cathcart	Baker-Just
5.	Core Power	100%	102%
6.	MAPLHGR (kW/ft) Low Exposure High Exposure	Reference 4	Reference 4
7.	ECCS Water Temperature	120°F	120°F
8.	ECCS Flow	Reference 4	Reference 4
9.	ECCS Flow to Hot Bundle	Reference 4	Reference 4
10.	Fuel Stored Energy	Best-Estimate GESTR	Best-Estimate GESTR
11.	Rod Internal Pressure	Best-Estimate GESTR	Best-Estimate GESTR
12.	Cladding Rupture Stress	BWR Design Values	BWR Design Values

TABLE XV-22

ACTIVITY RELEASED TO THE REACTOR BUILDING
FOLLOWING THE FHA
(curies)

Noble Gases		Halogens	
Isotope	Activity	Isotope	Activity
Kr-85m	1.06E+02	I-131 (organic)	3.83E+01
Kr-85	5.02E+02	I-132 (organic)	3.07E+01
Kr-87	1.79E-02	I-133 (organic)	2.37E+01
Kr-88	3.34E+01	I-135 (organic)	4.00E+00
Xe-131m	1.93E+02	I-131 (elemental)	9.52E+01
Xe-133m	9.45E+02	I-132 (elemental)	7.63E+01
Xe-133	3.25E+04	I-133 (elemental)	5.89E+01
Xe-135	7.83E+03	I-135 (elemental)	9.93E+00
Total	4.21E+04	Total	3.37E+02

TABLE XV-23

UNIFORM UNFILTERED STACK DISCHARGE RATES
FROM 0 TO 2 HR AFTER THE FHA
(curies/sec)

Noble Gases		Halogens	
Isotope	Activity	Isotope	Activity
Kr-85m	1.47E-02	I-131 (organic)	5.33E-03
Kr-85	6.97E-02	I-132 (organic)	4.27E-03
Kr-87	2.49E-06	I-133 (organic)	3.29E-03
Kr-88	4.64E-03	I-135 (organic)	5.56E-04
Xe-131m	2.69E-02	I-131 (elemental)	1.32E-02
Xe-133m	1.31E-01	I-132 (elemental)	1.06E-02
Xe-133	4.51E+00	I-133 (elemental)	8.18E-03
Xe-135	1.09E+00	I-135 (elemental)	1.38E-03
Total	5.85E+00	Total	4.68E-02

TABLE XV-24

FUEL HANDLING ACCIDENT DOSES

Receptor	Total Dose (rem TEDE)	Acceptance Criteria (rem TEDE)
Control Room	0.847	5
EAB*	0.447	6.3
LPZ	0.0384	6.3

* Worst 2-hr period of the accident duration

TABLE XV-25

FHA ANALYSIS INPUTS AND ASSUMPTIONS

Parameter	Value
Fuel Failure	2 assemblies out of 532
Fuel Decay (Recently Irradiated Fuel)	24 hr
Reactor Power Level	
Analyzed Thermal Power	1887 MWt
Peaking Factor	1.8
Gap Fractions	
I-131	8%
Kr-85	10%
Noble Gases	5%
Halogens	5%
Alkali Metals	12%
Iodine Speciation	
Organic	0.15%
Elemental	99.85%
Cesium Iodide (Particulate)	0%
Pool Decontamination Factors	
Minimum Water Depth	22.75 ft
Overall Iodine	191
Elemental Iodine	268
Alkali Metals	Infinite
Noble Gases	1
Release Duration	Instantaneous
Control Room Intake Flow	Infinite Exchange Rate
Control Room Volume	135,000 ft^3
Filtration	None

TABLE XV-25 (Cont'd.)

Parameter	Value
Atmospheric Dispersion, χ/Q	
Turbine/Condenser Release (Ground)	
EAB	1.90×10^{-4} sec/m^3
LPZ	1.63×10^{-5} sec/m^3
CR	4.82×10^{-4} sec/m^3
Dose Conversion Factors	FGR 11 & 12

TABLE XV-26

CRD ACCIDENT ANALYSIS INPUTS AND ASSUMPTIONS

Parameter	Value	
Fuel Failure		
Fuel Melt Fraction	0	
Cladding Failure Fraction	2.577% of core	
Reactor Power Level		
Analyzed Thermal Power	1887 MWt	
Peaking Factor	1.8	
Fuel Release Fraction to Coolant	Gap	
Iodine	10%	
Noble Gas	10%	
Alkali Metals	12%	
Halogens	5%	
Tellurium Group	0	
Iodine Speciation	Reactor Coolant	Turbine/ Condenser
Organic	0.15%	3%
Elemental	4.85%	97%
Cesium Iodide (Particulate)	95%	0
Condenser Release Fractions	Reaches the Condenser	Available for Release
Iodine	10%	10%
Noble Gas	100%	100%
Alkali Metals	1%	1%
Halogens	1%	1%
Tellurium Group	1%	1%
Leakage Parameters		
Main Condenser Volume	50,000 ft^3	
Main Condenser Leak Rate	1%/day	
Mechanical Vacuum Pump Exhaust Rate	280,000 lbm/hr	
Release Duration	24 hr	

TABLE XV-26 (Cont'd.)

Parameter	Value
Control Room Intake Flow	Infinite Exchange Rate
Control Room Volume	135,000 ft^3
Filtration	None
Atmospheric Dispersion, χ/Q Turbine/Condenser Release (Ground)	
EAB	1.90×10^{-4} sec/m^3
LPZ	1.63×10^{-5} sec/m^3
CR	1.03×10^{-3} sec/m^3
Atmospheric Dispersion, χ/Q Mechanical Vacuum Pump Release (Elevated)	
EAB	5.98×10^{-5} sec/m^3
LPZ	2.12×10^{-5} sec/m^3
CR	2.27×10^{-4} sec/m^3

TABLE XV-27

CRDA NOBLE GAS RELEASE

Time After Accident	Stack Discharge Rate* (Curies/Sec)
3.6 sec	6.60×10^{-3}
30 min	4.28
1 hr	6.70
1.5 hr	6.98
2 hr	6.85
4 hr	5.97
8 hr	4.55
12 hr	3.42
1 day	2.38

* Average rate over preceding time interval.

TABLE XV-28

CRDA HALOGEN RELEASE

Time After Accident	Stack Discharge Rate* (Curies/Sec)
3.6 sec	6.39×10^{-5}
30 min	6.43×10^{-2}
1 hr	1.13×10^{-1}
1.5 hr	1.19×10^{-1}
2 hr	1.15×10^{-1}
4 hr	9.45×10^{-2}
8 hr	6.62×10^{-2}
12 hr	4.61×10^{-2}
1 day	2.79×10^{-2}

* Average rate over preceding time interval.

TABLE XV-29

CONTROL ROD DROP ACCIDENT DOSES

Receptor	Case 1 Total Dose (rem TEDE)	Case 2 Total Dose (rem TEDE)	Acceptance Criteria (rem TEDE)
Control Room	0.610	1.60	5
EAB*	0.630	0.340	6.3
LPZ	0.0540	0.210	6.3

* Worst 2-hr period of the accident duration.

TABLE XV-29a

WETTING OF FUEL CLADDING BY CORE SPRAY

Time	Condition
0-1800 sec	Heat transfer by radiation.
1800	Wetting of fuel cladding begins.
3600	Wetting complete.

TABLE XV-29b

POST-LOCA AIRBORNE DRYWELL FISSION PRODUCT INVENTORY
(curies)

Time (hr)	Noble Gases	Halogens	Cesium and Rubidium	Other Solids*
0.417	9.46E+06	2.08E+06	1.38E+05	1.97E+04
0.917	6.05E+07	2.95E+06	1.73E+05	4.73E+05
2.917	9.96E+07	2.12E+05	1.14E+04	2.96E+04
8.033	8.10E+07	4.80E+04	1.10E+01	2.55E+01
24	6.17E+07	2.70E+04	2.49E-03	4.92E-03
96	3.65E+07	1.04E+04	2.36E-03	3.79E-03
240	1.57E+07	5.26E+03	2.13E-03	2.60E-03
720	1.26E+06	7.69E+02	1.61E-03	1.30E-03

* Except particulate iodine

TABLE XV-29c

POST-LOCA REACTOR BUILDING FISSION PRODUCT INVENTORY
(curies)

Time (hr)	Noble Gases	Halogens	Cesium and Rubidium	Other Solids*
0.417	1.21E-01	1.71E+00	0.00E+00	0.00E+00
0.917	4.03E-01	2.62E+00	0.00E+00	0.00E+00
2.917	1.52E+00	1.86E+00	0.00E+00	0.00E+00
8.033	1.81E+05	1.98E+04	8.92E-02	2.06E-01
24	6.66E+05	5.29E+04	2.33E-02	4.61E-02
96	2.68E+05	2.29E+04	4.70E-05	7.57E-05
240	1.12E+05	9.16E+03	1.23E-05	1.50E-05
720	7.81E+03	6.23E+02	9.28E-06	7.51E-06

* Except particulate iodine.

TABLE XV-29d

POST-LOCA DISCHARGE RATES
(curies/sec)

Time (hr)	Noble Gases	Halogens	Cesium and Rubidium	Other Solids*
		Filtered Stack Release		
0-0.417	0.00E+00	0.00E+00	0.00E+00	0.00E+00
0.417-0.917	4.81E-02	3.89E-02	0.00E+00	0.00E+00
0.917-2.917	8.51E-01	1.94E-01	0.00E+00	0.00E+00
2.917-8.033	2.62E+00	1.20E-01	3.22E-08	7.56E-08
8.033-24	1.29E+01	5.96E-02	6.54E-08	1.40E-07
24-96	9.34E+00	4.58E-02	4.49E-09	8.53E-09
96-240	4.50E+00	1.89E-02	2.14E-11	1.26E-11
240-720	9.41E-01	4.09E-03	1.27E-11	1.15E-11
		Unfiltered Ground-Level Release		
0-0.417	9.83E-01	7.14E-01	2.01E-02	2.62E-03
0.417-0.917	6.32E+00	1.23E+00	2.83E-02	5.70E-02
0.917-2.917	2.97E+01	6.70E-01	2.64E-02	6.83E-02
2.917-8.033	2.25E+01	6.69E-02	1.53E-03	3.42E-03
8.033-24	6.00E+00	4.13E-03	9.96E-05	2.00E-04
24-96	2.09E+00	7.06E-04	4.23E-07	1.13E-09
96-240	1.12E+00	3.26E-04	0.00E+00	0.00E+00
240-720	2.41E-01	1.04E-04	0.00E+00	0.00E+00

* Except particulate iodine

TABLE XV-30

CORE FISSION PRODUCT INVENTORY

Nuclide	Ci/MWt	Nuclide	Ci/MWt	Nuclide	Ci/MWt
Kr-83m	3.27E+03	Ru-106	1.76E+04	Cs-134	7.29E+03
Kr-85	3.93E+02	Rh-105	2.84E+04	Cs-136	2.28E+03
Kr-85m	6.82E+03	Sb-127	3.01E+03	Cs-137	4.35E+03
Kr-87	1.30E+04	Sb-129	8.91E+03	Ba-137m	4.12E+03
Kr-88	1.83E+04	Te-127	3.00E+03	Ba-139	4.89E+04
Kr-89	2.22E+04	Te-127m	4.05E+02	Ba-140	4.71E+04
Rb-86	7.29E+01	Te-129	8.76E+03	La-140	5.12E+04
Sr-89	2.45E+04	Te-129m	1.30E+03	La-141	4.45E+04
Sr-90	3.14E+03	Te-131m	3.97E+03	La-142	4.29E+04
Sr-91	3.10E+04	Te-132	3.85E+04	Ce-141	4.47E+04
Sr-92	3.38E+04	I-131	2.71E+04	Ce-143	4.11E+04
Y-90	3.24E+03	I-132	3.92E+04	Ce-144	3.70E+04
Y-91	3.18E+04	I-133	5.51E+04	Pr-143	3.97E+04
Y-92	3.40E+04	I-134	6.03E+04	Nd-147	1.80E+04
Y-93	3.96E+04	I-135	5.16E+04	Np-239	5.78E+05
Zr-95	4.46E+04	Xe-131m	3.04E+02	Pu-238	1.45E+02
Zr-97	4.51E+04	Xe-133	5.27E+04	Pu-239	1.34E+01
Nb-95	4.48E+04	Xe-133m	1.63E+03	Pu-240	1.89E+01
Mo-99	5.13E+04	Xe-135	1.91E+04	Pu-241	5.49E+03
Tc-99m	4.49E+04	Xe-135m	1.09E+04	Am-241	7.48E+00
Ru-103	4.29E+04	Xe-137	4.80E+04	Cm-242	1.85E+03
Ru-105	3.01E+04	Xe-138	4.50E+04	Cm-244	1.23E+02

TABLE XV-31

LOCA ANALYSIS INPUTS AND ASSUMPTIONS

Parameter	Value	
Reactor Power Level	1887 MWt	
Fission Product Release Fractions	Gap Phase	Early In-Vessel
Halogens	5%	25%
Noble Gases	5%	95%
Alkali Metals	5%	20%
Tellurium Group	0	5%
Ba, Sr	0	2%
Noble Metals	0	0.25%
Cerium Group	0	0.05%
Lanthanides	0	0.02%
Release Timing	Onset	Duration
Gap Phase	2 min	0.5 hr
Early In-Vessel	0.5 hr	1.5 hr
Iodine Speciation	Drywell	Suppression Pool
Organic	1.5%	3%
Elemental	4.85%	97%
Cesium Iodide (Particulate)	95%	0
Release of Activity to Suppression Pool	30% of core iodine inventory	
ESF Leakage		
Leak Rate to Reactor Building	1,200 gph	
Flashing Fraction	10%	
Containment Leakage (0 to 24 hr)		
Primary Containment Leak Rate	1.5 weight % per day	
Non-MSIV Reactor Building Bypass	91 scfh	
MSIV Reactor Building Bypass	100 scfh total; 50 scfh max per line	
Containment Leakage (24 to 720 hours)	50% reduction	
Reactor Building Drawdown	6 hr	

TABLE XV-31 (Cont'd.)

Parameter	Value	
Volumes		
Drywell Airspace (Min)	180,000 ft^3	
Wetwell Vapor Space (Min)	120,000 ft^3	
Suppression Pool (Min)	79,700 ft^3	
Reactor Building Free Volume	2,100,000 ft^3	
Reactor Building Holdup Volume	3.01E+10 cc	
Control Room	135,000 ft^3	
Flow Rates		
RBEVS	1,600 cfm	
Control Room Intake	2,025 cfm	
Assumed CR Unfiltered In-Leakage	100 cfm	
Filtration	CR	RBEVS
Organic	90%	90%
Elemental	95%	95%
Particulates	95%	95%
Isolation Time	<2 min	<2 min
Fission Product Removal Inputs		
Drywell Spray Flow Rate	6,383 gpm	
Drywell Accident Conditions	35 psig, 281°F	
Steam Line T/H Conditions	1,050 psia saturated conditions	
Main Steam Line Volume	82.4 ft^3 (inboard to outboard MSIV; each line)	
Breathing Rates	CR	Offsite
0 to 8 hr	3.5E-4 m^3/s	3.5E-4 m^3/s
8 to 24 hr	3.5E-4 m^3/s	1.8E-4 m^3/s
24 to 720 hr	3.5E-4 m^3/s	2.3E-4 m^3/s
CR Occupancy Factors		
0 to 24 hr	1.0	
24 to 96 hr	0.6	
96 to 720 hr	0.4	

TABLE XV-31 (Cont'd.)

Parameter	Value
Dose Conversion Factors	FGR 11 & 12
Atmospheric Dispersion, χ/Q	Tables XV-35a, XV-35b, XV-35c and XV-35d

TABLE XV-32

LOSS OF COOLANT ACCIDENT DOSES

Receptor	Total Dose (rem TEDE)	Acceptance Criteria (rem TEDE)
Control Room	4.81	5
EAB*	9.02	25
LPZ	1.60	25

* Worst 2-hr period of the accident duration

TABLE XV-33

THIS TABLE HAS BEEN DELETED

TABLE XV-34

THIS TABLE HAS BEEN DELETED

TABLE XV-34a

RELEASE/INTAKE ELEVATIONS

Point of Interest	Elevation (ft)	Elevation (m)
Main Stack	350	106.7
Reactor Building Blowout Panel (relative to bottom of panel)	78.9	24
Turbine Building Blowout Panel (relative to bottom of panel)	72.4	22.1
Control Room Intake (height equal to roof elevation)	72	29.95
Technical Support Center	21	6.4

TABLE XV-34b

RELEASE/INTAKE DISTANCE AND DIRECTIONS

Release/Intake	Horizontal Distance (ft)	Horizontal Distance (m)	Sector Bearing Relative to True North
Unit 1 Reactor Building Blowout Panel (from midpoint of panel)/Unit 1 Control Room Intake	340	103.6	149° SSE
Unit 1 Turbine Building Blowout Panel (from midpoint of panel)/Unit 1 Control Room Intake	236	71.9	117° ESE
Unit 1 Main Stack/Unit 1 Control Room Intake	400	121.9	166° SSE
Unit 1 Reactor Building Blowout Panel (from midpoint of panel)/Unit 1 Technical Support Center	343	104.5	86° ESE
Unit 1 Turbine Building Blowout Panel (from midpoint of panel)/Unit 1 Technical Suport Center	328	100.0	86° E
Unit 1 Main Stack/Unit 1 Technical Support Center	330	100.6	140° SE

TABLE XV-35

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TABLE XV-35a

χ/Q VALUES FOR THE CONTROL ROOM

Release Point	χ/Q Dispersion Coefficients (s/m^3)				
	0-2 hr	2-8 hr	8-24 hr	1-4 days	4-30 days
Unit 1 Reactor Building Blowout Panel	4.82E-04	2.61E-04	9.25E-05	6.70E-05	4.93E-05
Unit 1 Turbine Building Blowout Panel	1.03E-03	5.85E-04	2.07E-04	1.75E-04	1.52E-04
Unit 1 Main Stack	2.27E-04	1.26E-04	4.30E-05	3.58E-05	2.59E-05

TABLE XV-35b

χ/Q VALUES FOR THE TECHNICAL SUPPORT CENTER

Release Point	χ/Q Dispersion Coefficients (s/m^3)				
	0-2 hr	2-8 hr	8-24 hr	1-4 days	4-30 days
Unit 1 Reactor Building Blowout Panel	7.09E-04	5.60E-04	2.345E-04	1.71E-04	1.41E-04
Unit 1 Turbine Building Blowout Panel	5.91E-04	4.26E-04	1.63E-04	1.35E-04	1.16E-04
Unit 1 Main Stack	3.47E-04	2.42E-04	8.22E-05	6.06E-05	5.00E-05

TABLE XV-35c

OFFSITE χ/Q VALUES FOR GROUND-LEVEL RELEASES

Boundary	χ/Q Dispersion Coefficients (s/m^3)				
	0-2 hr	0-8 hr	8-24 hr	1-4 days	4-30 days
EAB	1.90E-04	---	---	---	---
LPZ	---	1.63E-05	1.10E-05	4.67E-06	1.67E-06

TABLE XV-35d

OFFSITE χ/Q VALUES FOR ELEVATED RELEASES

Boundary	χ/Q Dispersion Coefficients (s/m^3)				
	0-2 hr	0-8 hr	8-24 hr	1-4 days	4-30 days
EAB	5.98E-05	---	---	---	---
LPZ	---	2.12E-05	8.40E-07	3.45E-07	1.11E-07

TABLE XV-36

REACTOR BUILDING LEAKAGE PATHS

		Joint Length	Area
Walls		2@ 5,147 ft	2@ 9,370 ft^2
Walls		2@ 3,778 ft	2@ 6,850 ft^2
Roof		550 ft	---
	Total	18,400 ft	32,440 ft^2

Reactor Building Pressure vs. Time by Reactor Building Elevation

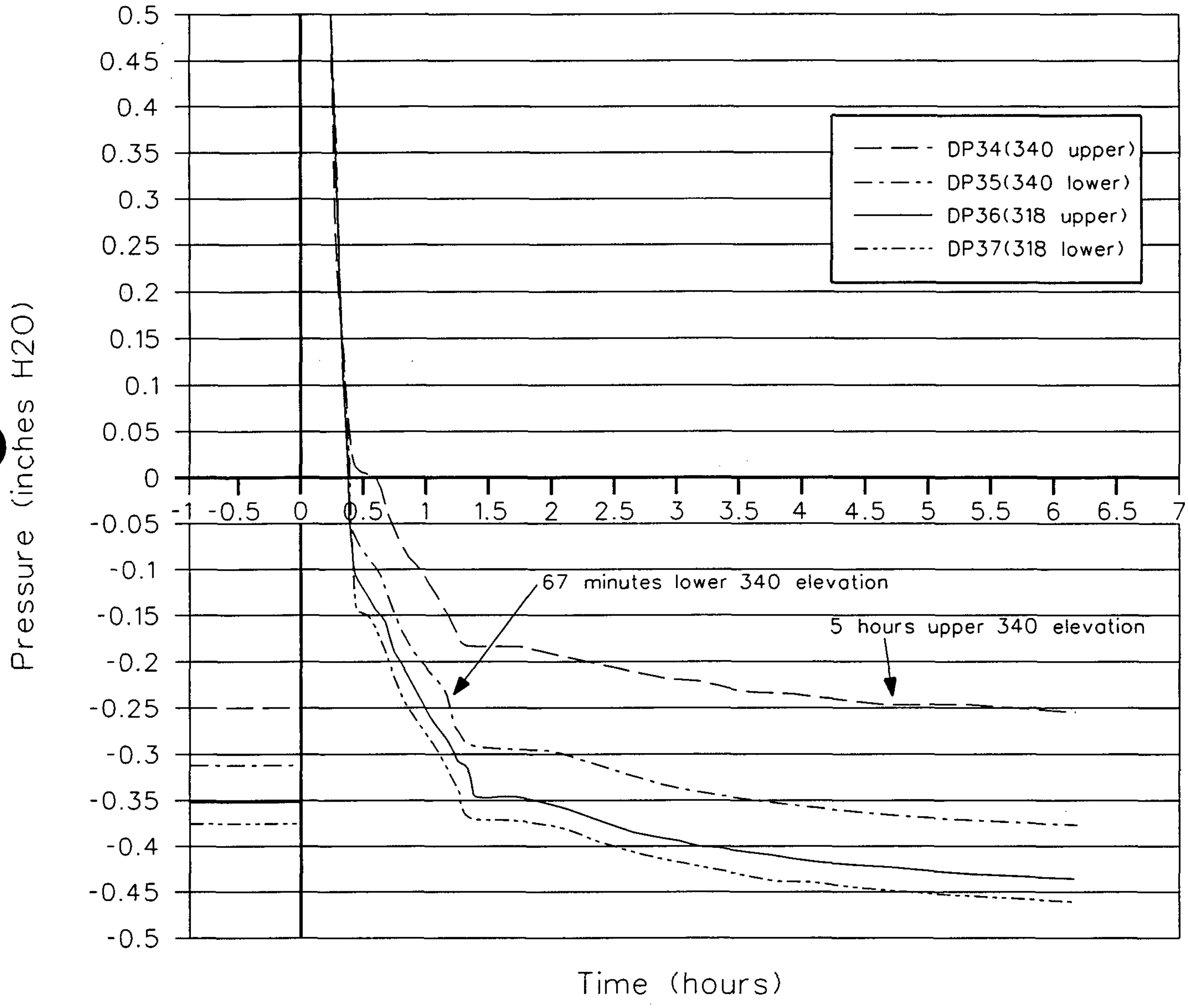

FIGURE XV-73
UFSAR Revision 21
October 2009

Reactor Building Pressure vs. Time by Reactor Building Elevation
(Focused on the Initial 2.5 hours)

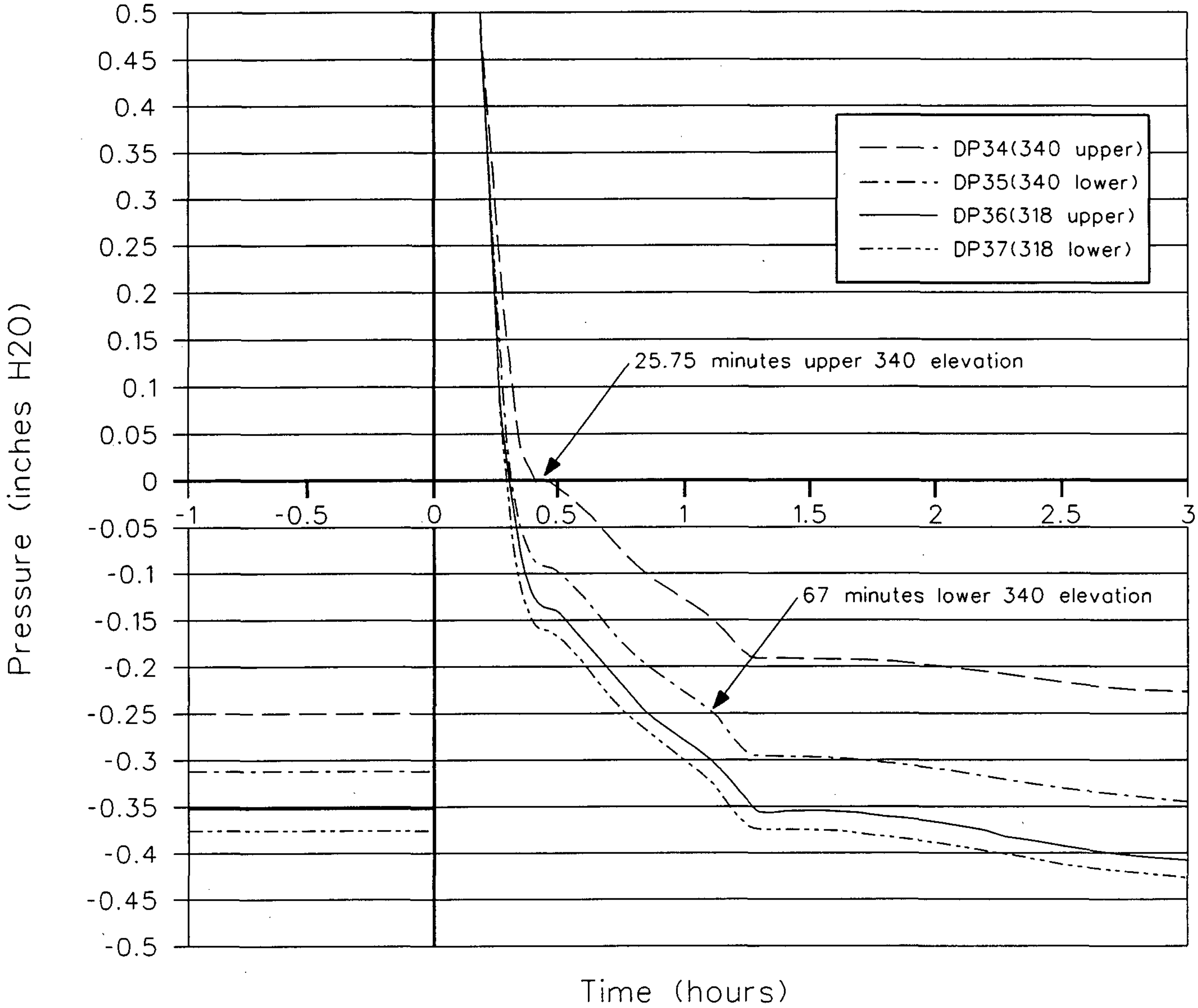

FIGURE XV-74
UFSAR Revision 21
October 2009

2.0 Design Analysis

2.1 Code Approval Calculations Under Rated Conditions

The results of the calculations used to determine if the containment vessels and penetrations meet Code requirements are discussed in Section XVI-F.

2.2 Ultimate Capability Under Accident Conditions

To determine the ultimate capability of the containment, it was assumed that a circumferential recirculation line break occurred and all core and containment spray systems failed. The pressure transient was calculated and is shown on Figure XVI-13.

The containment is expected to maintain its integrity up to the original test pressure. Suppression chamber integrity could not be ensured after approximately 1300 sec, the time to reach 40.25 psig. The drywell integrity could not be ensured after approximately 55,000 sec, the time to reach 71.4 psig.

2.3 Capability to Withstand Internal Missiles and Jet Forces

Several potential missile and jet hazards within the containment were examined. Table XVI-10 lists the potential hazards and data on the forces acting on the shell. Table XVI-11 gives the results of the analysis and demonstrates that none of the jets or missiles considered would cause rupture of the containment.

The method used to determine the final deformation of the steel wall and the possibility of penetration is as follows.

A load is assumed; the deformation resulting from this load is calculated according to the method of Roark(9) for missiles striking cylindrical portions of the drywell, and according to the method of Bijlaard(10) for spherical portions of the drywell. With the load and deformation known, the strain energy absorbed in the steel can be calculated. For Cases A, B, C, and F the steel wall is backed up by a concrete wall approximately 2 in away. For these cases, the energy required to crush the concrete is added to the strain energy of the steel to obtain the total energy absorbed. By assuming various loads, a plot can be constructed of the force and energy absorbed in steel and concrete as a function of displacement. The intersection of this curve with a plot of total energy to be absorbed (energy of missile plus energy of jet) yields the value of displacement at which the energy of the missile and jet is absorbed. But this displacement corresponds to a force; if no rupture is to occur, this force cannot exceed the force required to shear the plate. In all cases the energy is absorbed before the displacement force causes the plate to be sheared.

2.4 Flooding Capabilities of the Containment

For long-term post-accident recovery, provision is made to flood the containment to above core level. Drywell pressure and water level indication and alarms are provided in the main control room. The stresses on the containment structure resulting from flooding up to el 333' (about 43 ft above the core or 7 ft below the operating floor) have been analyzed.

Review of Chicago Bridge & Iron Company computations shows that the containment integrity will be maintained when the containment is flooded to el 333' and subjected to the seismic forces from the design earthquake. Buckling of the shell material will not occur since critical unit loading for buckling is not approached. Table XVI-12 compares the maximum stresses with the critical stresses at which buckling could occur.

If containment venting to the atmosphere is not feasible because of high fission product inventory, flooding to above core level cannot be achieved. The containment can be partially flooded, but flooding above el 233' (7.6 ft above the concrete base) could result in a pressure exceeding the original suppression chamber test pressure of 40.25 psig. The pressure suppression system was subjected to a rigorous analysis to determine the maximum plate stresses for a combined temperature and flooding loading condition.

The analysis consisted of two parts, the model and the shell. To determine reactions, displacements, and rotations necessary for the shell analysis, the pressure suppression system was modeled as a space frame consisting of 240 joints and 480 members. STRUDEL, the computer program used to solve the model, was developed by the Massachusetts Institute of Technology. The shell analysis was based on the paper, "Numerical Analysis of Unsymmetrical Bending of Shells of Revolution," by Budianski and Radkowski, using the program, "Unsymmetrical Bending of Shells of Revolution," developed by AVCO Corporation.

The stress analysis for combined loads included the following: dead load of the pressure suppression chamber steel, weight of water in flooded pressure suppression chamber, pressure due to the water in the drywell to el 301'-0", thermal (minimum operating = 32°F, maximum post-incident = 205°F), and 0.15g horizontal and 0.055g vertical earthquake accelerations acting simultaneously. The resulting maximum circumferential bending stress at the top of the pressure suppression chamber is 4,870 psi. This stress is then combined with the net circumferential membrane stress of 5,190 psi to give a total maximum stress of 10,060 psi, which is well within normal Code values.

2.5 Drywell Air Gap

Thermal and pressure requirements of the drywell determined the size of the air gap. Based on an incident temperature of 310°F and a pressure of 62 psig, the magnitudes of expansion of the drywell shell plate in the critical areas are as indicated on Figure XVI-15. The calculated maximum seismic deflection of the drywell with respect to the reactor building is 0.05 in at the equator of the spherical portion of the drywell. The minimum as-built gap between the exterior of the drywell shell plate and the forms was established at 2 in and was well maintained. The maximum as-built gap from field dimensions is 3 in; the average as-built gap is 2 1/2 in.

Shop-fabricated fiberglass forms (1/2 in minimum thickness) with flanges for bolting were used to retain the poured-in-place concrete surrounding the drywell. Detailed form drawings were made based on the containment vessel shop-detailed drawings extrapolated for the 2-in air gap.

Individual form panels were from 6 to 8 ft in width and 14 to 16 ft in height. Spacer blocks were wired at the panel corners during erection to ensure proper clearance to the vessel shell. A nominal clearance of 1/2 in between top, bottom, and sides of panels was allowed for adjusting and shimming of the panels. The shimming material consisted of 1/2 in by 3/4 in compressible polyurethane strips. Each panel was bolted in its relaxed state with shims in place and the joints taped with fiberglass mat strip and polyester resin sealant for leak-tightness.

After a horizontal course of forms was completed with spacer blocks secure and the gap dimension checked, concrete was poured in lifts of 3 to 4 ft to about 3 ft below the top of the form to provide free board in protecting the air gap above. The air gap above was closed as forms were erected and before concrete was poured, by stuffing burlap rope into the gap and wiring it to the panels. A plastic sheet was flashed to the vessel above the formwork and draped over the top of the forms to prevent concrete spatter on the vessel shell and the top of forms.

The intersections of penetration sleeves and formwork were also sealed with fiberglass strip and polyester resin, as shown on Figure XVI-16. This method produced a tight joint with no leakage problems.

Field adjustment and, in some instances, field alteration of the forms was required to meet the specified clearances to the shell plate and penetrations. Field personnel that supervised the air gap dimensioning at the shell plate and penetrations also guarded against foreign objects getting into the air space. All supervisors and craftsmen in this phase of the work were

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carefully instructed about the necessity of maintaining the air gap in an exact and clear condition.

The air gap is ventilated at the top and bottom as follows: at the top of the air gap are two 12-in diameter emergency condenser (EC) pipes just below the head cover of the containment. These pipes, with a 2-in minimum clearance to the penetration sleeves, vent the air space to the outside of the biological shield concrete. At the bottom of the air gap are ten 6 ft 9 in vent pipes that lead to the suppression chamber. These pipes, with a 2-in minimum clearance to the penetration sleeves, vent the air space into that part of the building where the suppression chamber is located.

In addition, the annulus between many of the penetrations from the drywell and their surrounding concrete sleeves provides available ventilation area to the air space. The air gap is drained with ten 4-in diameter pipes at the top of the sand cushion as shown on Figure XVI-15. No condensation or water has been found in the drains.

2.5.1 Tests and Inspections

Those portions of the air gap that are in the vicinity of the majority of the piping penetrations are available for visual inspection.

2.6 Reactor Shield Wall

The reactor shield wall has been analyzed as a double-walled cylinder with a 21-ft 2 1/2-in ID supported at its base (el 258-1 1/2) and its top (el 303-3). Figure XVI-17 shows the overall plan and structural details.

The reactor shield wall is assumed to be a cylinder which is simply supported at its ends and pressurized internally.

Both inner and outer 5/8-in thick steel liner plates are stressed in tension due to internal pressure. Since the loading distribution between the inner and outer plates was within 5 percent, they were assumed equal. The plates are connected to the 27WF177 columns by continuous 3/8-in fillet welds (E70 electrodes) on the inner plate and intermittent 5/16-in fillet welds, 6 in long on 12-in centers on the outer plate. These welds are the weakest part of the structure for the internal pressure loading. Allowable stresses as shown in the 1969 AISC Handbook* were increased 50 percent for this loading (see Table XVI-13), which is assumed to be 0.9 of the shear yield stress.

The results of this analysis show that the reactor shield wall is capable of withstanding more than 96 psi.

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* Also see Section XVI, Subsection G.

The allowable strain in the containment vessel demonstrates the adequacy of the structure against impacts resulting from pipe break and associated whip. The ultimate strain of the containment vessel is 10 percent. The calculated accumulative strains for the conditions analyzed are given below:

Condition	Vessel Thickness (in)	Strain (%)
Recirculation Loop Elbow	1.5	6.1
Recirculation Loop Elbow	0.768	9.2
Structural Beam (12WF40)	0.768	4.0
Structural Beam (12WF40)	1.5	1.1

With respect to the structural beams, the highest velocity (115 fps) occurs for the 10WF33 beam. For the sake of conservatism, the analyses used a heavier 12WF40 beam with a velocity of 115 fps rather than the actual velocity of 60 fps.

2.1.2 Systems Affected by Line Break

Main Steam

The MSLs discharge from the vessel at el 310 and at the 90-deg and 270-deg radial locations and descend. In this area there are two 10-in EC lines at the 67.5-deg and 292.5-deg radial locations. These lines are separated by great enough distances so as not to be affected by any break of the MSLs.

The MSLs proceed downward through el 295 where they pass by some 1-in instrument piping at the 90-deg radial location and a CRD exhaust line at the 270-deg radial location.

Continuing on down from el 295 to el 264, the steam lines pass by containment spray spargers, feedwater lines on each side of the steam lines, relief valve discharge lines, shutdown cooling, reactor recirculation, core spray and rod drive exhaust. All could be ruptured except for the recirculation lines because of their larger size. From this point, the steam lines head toward 180 deg and exit the drywell adjacent to the two 18-in incoming feedwater lines.

The required systems for core cooling and safe shutdown in the case of a MSL break are the containment spray, core spray, and feedwater. All of these systems have the required redundancy or backup as discussed in Section 2.1.3 below.

Feedwater

The two 18-in feedwater lines enter the primary containment at el 263, adjacent to the two 24-in MSLs. At this point the lines curve around to the 90-deg and 270-deg radial directions. At ±45 deg on each side of these lines, two 10-in lines proceed inward and then ascend vertically. In this run the feedwater lines pass by some 12-in core spray lines, 14-in shutdown cooling lines, 6-in containment spray lines, and 6-in cleanup system lines.

Proceeding upward to el 295, the feedwater lines enter the reactor vessel at the 45-deg, 135-deg, 225-deg, and 315-deg radial locations. During this ascension they pass by the 12-in core spray lines and containment spray spargers, and a 1 1/2-in liquid poison line.

The only damage which could occur is to one of the redundant containment sprays and to the liquid poison lines because of their size. As discussed in Section 2.1.3 below, this system has adequate redundancy even in the event that one system is incapacitated.

The liquid poison system is a post-LOCA suppression pool pH control and backup reactivity control system only. The liquid poison system is not required to function following a feedwater line break, since the primary reactivity control system (CRD) is not impacted by a feedwater line break.

Reactor Recirculation System

There are five recirculation pumps, each of which has a suction and discharge line. These lines are at the 0-deg, 42-deg, 73-deg, 114-deg, 144-deg, 186-deg, 216-deg, 258-deg, 288-deg, and 330-deg radial locations between el 225 and 275. These are 28-in and 26-in diameter lines for the suction and discharge, respectively. Other lines in the area of these recirculation lines are:

1. 10- and 12-in ECs (330 deg, 0 deg).

2. 4- and 6-in containment spray (0 deg to 360 deg).

3. 10- and 18-in feedwater (288 deg, 258 deg, 216 deg, 144 deg, 114 deg).

4. 12-in core spray (258 deg, 115 deg).

5. 14-in relief valve discharge (288 deg, 216 deg).

6. 6-in cleanup (42 deg).

7. 24-in main steam (258 deg, 216 deg, 186 deg, 144 deg, 114 deg).

8. 14-in shutdown cooling (330 deg).

9. 4-in, 2-in, 1 1/2-in, and 3/4-in reactor building closed loop cooling (RBCLC) piping supplying recirculation pump seal and motor coolers.*

The first four systems (1, 2, 3 and 4) may be required following a break in the recirculation system. All have the required redundancy or backup.

Containment Spray

Due to the small size of the containment spray lines and the fact that they are not pressurized during normal operation, rupture would not cause damage to any other larger lines.

Liquid Poison

The liquid poison line, due to its small size, would not impart damage on any other system.

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* The original pipe whip analysis considered only the impact on engineered safety features (ESF) required to maintain containment integrity and adequate core cooling. The conclusion reached was that ESF systems have adequate redundancy and separation of piping inside the drywell to ensure the ESF function is maintained. The RBCLC system was not considered an ESF system and was not evaluated. Since the RBCLC system provides a safety-related post-LOCA function, the consequences of a loss of RBCLC on the ability to maintain post-LOCA safe shutdown has been evaluated on a coping basis. The conclusion reached is that adequate provisions exist to either isolate the RBCLC drywell piping and restore the system post-LOCA, or that alternate methods could be implemented to maintain post-LOCA safe shutdown conditions assuming RBCLC is not restored.

Emergency Condensers

The 10-in EC supply lines are located in the area between 270 deg and 90 deg radially at el 306'. In this area there are only some small instrument lines and some 1 1/2-in containment spray headers. The supply lines leave the drywell and the 10-in return lines enter at el 269'. In this there are four lines:

1. 6-in cleanup.

2. 6-in, 8-in, and 12-in containment spray.

The only required lines in the event of a break of an EC line are the containment spray lines. These are supply lines to spargers, and damaging one would not render the system inoperable since there are four spargers. Only one sparger could be damaged by the break of any one EC line.

Control Rod Drive Discharge

Due to the small size of the discharge piping in relation to the other lines, no other damage would be imparted due to a rupture of this line.

Core Spray

There are two 12-in core spray lines which enter the containment at el 240' at about 45 deg on each side of the 180-deg direction. The one line in the quadrant from 180 deg to 270 deg passes by the following lines and then rises vertically:

1. 10- and 18-in feedwater.

2. 24-in main steam.

3. 14-in relief valve discharge.

4. 6-in containment spray.

The other core spray line, in the quadrant from 90 deg to 180 deg, enters the drywell and then runs north to the 0 deg-90 deg quadrant where it rises vertically. This line passes by the following lines:

1. 10- and 18-in feedwater.

2. 14-in relief valve discharge.

3. 6- and 8-in containment spray.

Both lines rise to el 295 where they enter the reactor vessel 180 deg apart. In their rise they pass by 10-in feedwater lines and some small containment spray headers. The only lines which the core spray could damage are the four smaller feedwater lines and the four containment spray lines to spargers.

Relief Valve Discharge

In each 180-deg radial segment of the drywell that is from 0 deg-180 deg and 180 deg-360 deg, there are three 14-in discharge lines.

In the 0 deg-180 deg sector there are the following lines:

1. 6-in cleanup.

2. 10- and 18-in feedwater.

3. 12-in core spray.

4. 6-, 8- and 12-in containment spray.

5. 24-in main steam.

In the 180 deg-360 deg segment there are the following lines:

1. 14-in shutdown cooling.

2. 10- and 18-in feedwater.

3. 12-in core spray.

4. 6-, 8-, and 12-in containment spray.

5. 24-in main steam.

The lines which could be damaged by the relief valve discharge that are required for core cooling are the core spray, containment spray and feedwater lines. No single relief valve discharge line failure could eliminate redundancy to the point where the safeguards function is inadequate.

Shutdown Cooling

In the 270-deg to 0-deg radial location there are 14-in supply and return lines to the shutdown cooling system at el 270. In this area there are the following systems:

1. 10-in feedwater line.

2. 6- and 12-in containment spray.

3. 14-in relief valve discharge.

4. 3-in exhaust from the CRD.

The first three systems may be required following a shutdown cooling system line break. However, no line break in this system could result in loss of redundancy in those required systems to the point where the safeguards function is inadequate.

Cleanup System

In the 0-deg to 90-deg radial location, a 6-in line comes out of one of the recirculation lines at el 263, leaves the drywell, then reenters at el 263 and returns into a feedwater line. The only lines in that area are:

1. 10-in feedwater.

2. 12-in core spray.

3. 10-in EC return line.

4. 14-in safety valve discharge.

5. 6- and 12-in containment spray.

Due to the large sizes of these lines, there would be no effect on them because of a rupture of the cleanup system.

2.1.3 Engineered Safeguards Protection

The preceding analysis shows that engineered safeguard systems could be damaged as a result of pipe whip. However, in no case is the damage extensive enough to result in loss of core cooling, a safe shutdown capability.

As described in Section 2.1.2 above, the feedwater system (high-pressure coolant injection [HPCI]) could be damaged as a result of a rupture in the main steam, recirculating, core spray, relief valve discharge or the shutdown cooling system lines. In any event, since there are two feedwater lines which are physically separated in the areas of concern, only one could be damaged. In addition, core spray and autodepressurization are the prime sources of core cooling. HPCI is only a backup system. There is no single pipe rupture which could result in loss of feedwater and both core spray systems.

There are two independent core spray lines 180 deg apart. These could be damaged by rupture of either the recirculation or relief valve discharge lines. However, because of redundancy and physical separation, only one line could be damaged. HPCI serves as a backup.

NOTE: The ability of the redundant core spray line to provide adequacy of core cooling is evaluated consistent with the pipe whip analysis design basis and is not based on 10CFR50 Appendix K LOCA methods.

The containment spray system could be damaged as a result of a rupture in the following systems:

- reactor recirculation
- feedwater
- main steam
- emergency condensers
- core spray
- relief valve discharge
- shutdown cooling

There are two containment spray systems, each one consisting of a supply and a set of spargers inside the containment. Both sets of spargers in each containment spray system could be damaged as a result of a single line break due to close proximity of the spargers. This would not result in a loss of containment cooling since the suppression chamber water would still be circulated through the containment spray heat exchangers.

Degradation of spray efficiency could occur and would depend on the extent of sparger damage. In any event, some spray efficiency would remain.

The EC supply and return lines on both systems could be damaged by a rupture of the main steam or reactor recirculation system lines. However, this system is not required to maintain core cooling. Feedwater, core spray, and autodepressurization provide the core cooling function in the event of a line rupture within the drywell.

The CRD hydraulic system could be damaged by a rupture in the main steam, relief valve discharge or reactor recirculation system. However, should these lines be damaged, the rods would scram on reactor pressure. The liquid poison system, which serves as a backup to the control rod system, is not subject to damage by the same lines.

The only line whose rupture could damage the liquid poison system is a line in the feedwater system.

However, the liquid poison system is not normally used and is not required to function following a feedwater line break for post-LOCA suppression pool pH control. It is also a backup to the CRD system which is not subject to damage by a ruptured feedwater line.

2.2 Outside Primary Containment

All high-energy lines were analyzed to determine the effects of postulated pipe breaks. In all cases, safe shutdown of the reactor can be accomplished and the Station can be maintained in the shutdown condition[30]. Table XVI-29 lists the high-energy systems which were analyzed. It was assumed that a line break could occur at any point outside of the primary containment.

A pipe break could cause failures of other systems because of pipe whip, jet forces or environmental effects on equipment. Systems which could be affected by various line breaks are listed in Table XVI-30.

The design of the Station incorporates a number of features which mitigate the effects of pipe rupture. There are redundant systems and components for each shutdown or accident protection function. The locations of equipment, power supplies, cables and instrumentation for redundant systems are physically separated to preclude common-mode failures. Cables, motors, power boards and other equipment are designed to be operated in the environments expected after a line rupture.

3.0 Building Separation Analysis

The building separation was determined in design so that horizontal deflection will not result in the striking of adjacent structures. Figures XVI-46 through XVI-55 indicate building separation and computed maximum horizontal deflections (Δ) of various structures. Bedrock at the site is sound and competent. No permanent relative displacements would occur during any possible earthquake. Class I piping between buildings (e.g., condensate supply to emergency core cooling system [ECCS]) is sufficiently flexible to withstand the relative displacements indicated on Figures XVI-46 through XVI-61.

4.0 Tornado Protection

The probability of a tornado occurrence at the Nine Mile Point site is close to zero. The map of tornado probability published by Fawbush shows less than one occurrence in this general area during a 30-yr period[31]. The more complete data assembled by Spohn indicate, for the 1916-1961 period, a total of one to three tornadoes in the 1-deg latitude-longitude square surrounding the area[32]. Flora provides details on New York State and shows only a single tornado occurrence in the immediate area of the site itself[33]. Using work by Thom[34], one could calculate a 1-in-19,000 yr tornado occurrence probability for the Nine Mile Point site area.

In the event of a tornado, it would be necessary to maintain the integrity of certain areas of the Station to conduct a safe shutdown. Those areas and the structures surrounding those areas were investigated by determining the pressure and/or wind velocity at which functional failure would occur. The results of this investigation are given in Table XVI-31.

Functional failure of a structural member is reached when that member no longer performs in a reliable manner. Working stress design was used for evaluating Station structures. The allowable or working stress was derived from the AISC or ACI applicable Codes*. The stress at functional failure is the working stress times a safety factor of:

Structural Steel	$1.5 \times F_a \leq 0.9\ F_{cr} \leq 0.9\ F_y$
Concrete	$1.6 \times 0.45\ F'_c = 0.75\ F'_c = F_c$
Reinforcing Steel	$1.80 \times 0.5\ F_y = 0.9\ F_y$

Definitions (all units in psi)

F_a = normal design stress (AISC handbook)

F_{cr} = minimum buckling stress of the material

F_y = yield stress of the material

F_c = normal allowable stress (ACI Code 318)

F'_c = ultimate concrete strength

Internal pressure relief panels in the reactor and turbine buildings are designed to release at 65 psf and 62 psf, respectively. The ratio of relief area to building volume is 1.6 $ft^2/1000\ ft^3$ for the reactor building, and 0.21 $ft^2/1000\ ft^3$ for the turbine building.

The capability of the metal siding on the buildings to resist tornado-induced missiles is not known. The H. H. Robertson Company, in conjunction with the Gulf General Atomic Corporation, conducted a series of impact tests to determine the capability of their metal panel siding to withstand certain missiles. The results of these tests and the conclusions drawn were not available.

* Also see Section XVI, Subsection G.

TABLE XVI-9a

CORE SHROUD REPAIR DESIGN SUPPORTING DOCUMENTATION

Document Number	Description
GENE-B13-01739-04 (NMPC Calculation #SQ-Vessel-M028)	NMP1 Shroud and Shroud Repair Hardware Analysis
GENE-B13-01739-05 (NMPC 50.59 Evaluation 94-080)	Safety Evaluation for Installation of Stabilizers on the NMP1 Core Shroud
GENE-B13-01739-03 (NMPC Calculation #SQ-Vessel-M027)	Seismic Design Report of the Shroud Repair for NMP1 Power Plant
24A56426 (NMPC Calculation #SQ-Vessel-M026)	Stress Report, "Shroud & Stabilizers Code Design Specification - Shroud Stabilizers"
25A5583	Design Specification, "Shroud Repair Hardware"
107E5679, Sheets 1-4	Modifications & Installation Drawings
25A5584	Fabrication Specification, "Fabrication of Shroud Stabilizer"
FDI 0245-90800	Field Disposition Instruction
25A5585	Installation Specification, "Stabilizer Installation"
21A2040	Cleaning and Cleanliness Control
24A5586	Shroud Stabilizer Code, Design Specification
GENE-771-44-0894	Justification of Allowable Displacements of the Core Plate and Top Guide - Shroud Repair
GENE-B13-01739-5.1 (NMPC 50.59 Evaluation 96-018)	Modification to GE Core Shroud Repair Design
NMPC 50.59 Evaluation 98-103	Core Shroud Vertical Weld Repair Clamp

TABLE XVI-9a (Cont'd.)

Document Number	Description
NER-IM-059	NMP-1 Core Shroud Vertical Weld Repair Design Report, MPR Report No. MPR-1966
NRC Safety Evaluation	NMP1 Core Shroud Repair, dated 3/31/95
NRC Safety Evaluation	Modifications to Correct Core Shroud Repair Deviations, dated 3/3/97
NRC Safety Evaluation	Modifications to Core Shroud Stabilizer Lower Wedge Retaining Clip and Evaluation of Shroud Vertical Weld Cracking, dated 5/8/97
NRC Safety Evaluation	Modification of Core Shroud Tie Rod Upper Spring Assemblies, dated 6/7/99
NRC Safety Evaluation	Modification of Core Shroud Tie Rod Upper Support and Tie Rod Nut Assemblies, dated 10/3/07

4. The baseline inspections recommended in BWRVIP-47 for the BWR lower plenum components will be incorporated into the program.

5. If the October 19, 2005, draft of Code Case N-730 is approved by ASME, Unit 1 will implement the final Code case as conditioned by the NRC. If the Code case is approved by ASME but not yet listed in RG 1.147, Unit 1 will seek NRC approval of the Code case on a plant-specific basis as conditioned by the NRC.

   It will be programmatically required that during the period of extended operation, should a control rod drive (CRD) stub tube rolled in accordance with the provisions of the Code case resume leaking, Unit 1 will implement one of the following zero leakage permanent repair strategies prior to startup from the outage in which the leakage was detected:

   a. A welded repair consistent with BWRVIP-58-A, "BWRVIP Internal Access Weld Repair" and Code Case N-606-1, as endorsed by the NRC in RG 1.147.

   b. A variation of the welded repair geometry specified in BWRVIP-58-A subject to the approval of the NRC using Code Case N-606-1.

   c. A future developed mechanical/welded repair method subject to the approval of the NRC.

6. Unit 1 will evaluate component susceptibility to loss of fracture toughness due to neutron fluence and thermal embrittlement. Assessments and inspections will be performed, as necessary, to ensure that intended functions are not impacted by the aging effect.

7. An EVT-1 examination of the Unit 1 feedwater sparger end bracket welds will be added to the BWRVIP. The inspection extent and frequency of the end bracket weld inspection will be the same as the ASME Section XI inspection of the feedwater sparger bracket vessel attachment welds.

8. Unit 1 will perform an EVT-1 inspection of the thermal shield to flow shield weld starting in 2007, and

proceeding at a 10-yr frequency thereafter consistent with the ISI inspection interval.

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Enhancements will be completed prior to the period of extended operation.

C.1.13 Closed-Cycle Cooling Water System Program

The Closed-Cycle Cooling Water System (CCCWS) Program manages loss of material and fouling of components exposed to CCCW environments. The applicable piping systems include the reactor building closed loop cooling (RBCLC) system, control room heating, ventilation and air conditioning (HVAC) system, and the heat exchanger jacket water cooling portions of the emergency diesel generator (EDG) system. Also included are portions of non-safety related systems credited in the aging management review. Program activities include chemistry monitoring, surveillance testing, data trending, and component inspections. The CCCWS Program implements the guidelines for controlling system performance and aging effects described in Electric Power Research Institute (EPRI) Report TR-107396. However, specific exception is taken to maintaining chemical corrosion inhibitor concentrations. NMPNS utilizes corrosion control without chemicals (i.e., pure water) for the RBCLC and control room HVAC systems, and the EDG system chromate concentrations are maintained higher per EDG vendor recommendation. This is an exception to the program described in NUREG-1801.

Enhancements to the CCCWS Program include the following revisions to existing activities that are credited for license renewal:

1. Direct periodic inspections to monitor for loss of material in the piping of the CCCW systems.

2. Implement a Corrosion Monitoring Program for larger bore CCCW piping not subject to inspection under another program.

3. Establish periodic monitoring, trending, and evaluation of performance parameters for the RBCLC and control room HVAC systems.

4. Establish the frequencies to inspect for degradation of components in CCCW systems, including heat exchanger tube wall thinning.

5. Perform a heat removal capability test for the control room HVAC system at least every 5 yr.

6. Expand periodic chemistry checks of CCCW systems consistent with the guidelines of EPRI TR-107396.

7. Provide the controls and sampling necessary to maintain water chemistry parameters in CCCW systems within the guidelines of EPRI Report TR-107396.

8. Ensure acceptance criteria are specified in the implementing procedures for the applicable indications of degradation.

The enhancements will be completed prior to the period of extended operation.

C.1.14 Compressed Air Monitoring Program

The Compressed Air Monitoring Program manages aging effects for portions of the compressed air systems within the scope of license renewal, including cracking and loss of material due to general corrosion, by controlling the internal environment of systems and components. Program activities include air quality checks at various locations to detect contaminants that would affect the system's intended function. Additional visual inspections are credited for identification and monitoring of degradation for air compressors, receivers, and air dryers. The Compressed Air Monitoring Program is based on GL 88-14 and recommendations presented in Institute of Nuclear Power Operations (INPO) Significant Operating Event Report (SOER) 88-01. The program also includes good practice elements of the general maintenance and inspection activities for the compressor, receiver, and air drier discussed in EPRI TR-108147 (revision to EPRI NP-7079) and ASME OM-S/G-1998, Part 17. However, specific exception is taken to any maintenance recommended in EPRI TR-108147 that is not also endorsed by the equipment manufacturers, and to the preservice and inservice testing guidelines of ASME OM-S/G-1998, Part 17. This is an exception to the program described in NUREG-1801. Unit 1 also takes exception to the use of ISA-S7.0.01-1996 for air quality standards. The system air quality is monitored and maintained in compliance with the requirements of ANSI/ISA-S7.3-1975, which meets or exceeds the quality requirements for dew point, hydrocarbons, and particulate of Section 4.4 of EPRI TR-108147 and ISA-S7.0.01-1996.

Enhancements to the Compressed Air Monitoring Program include the following revisions to existing activities that are credited for license renewal:

1. Develop new activities to manage the loss of material, stress corrosion cracking, and perform periodic system leak checks.

2. Expand the scope, periodicity, and inspection techniques to ensure that the aging of certain subcomponents of the dryers and compressors (e.g., valves, heat exchangers) is managed.

3. Establish activities that manage the aging of the internal surfaces of carbon steel piping and that require system leak checks to detect deterioration of the pressure boundaries.

4. Expand the acceptance criteria to ensure that the aging of certain subcomponents of the dryers and compressors (e.g., valves, heat exchangers) is managed.

5. Develop and implement the activities to address the failure mechanism of stress corrosion cracking in unannealed red brass piping in Unit 1.

Enhancements will be completed prior to the period of extended operation.

C.1.15 Environmental Qualification Program

The Environmental Qualification (EQ) Program manages thermal, radiation, and cyclical aging for electrical equipment important to safety and located in harsh plant environments at Unit 1. Program activities 1) identify applicable equipment and environmental requirements; 2) establish, demonstrate, and document the level of qualification (including configuration, maintenance, surveillance, and replacement requirements); and 3) maintain (or preserve) qualification. The EQ Program employs aging evaluations based on 10CFR50.49(f) qualification methods. Components in the EQ Program must be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation. Important attributes for the reanalysis of an aging evaluation include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met).

C.1.16 Fatigue Monitoring Program

The Fatigue Monitoring Program (FMP) manages the fatigue life of RCPB components by tracking and evaluating key plant events. The FMP monitors operating transients to date, calculates cumulative usage factors (CUF) to date, and directs performance of engineering evaluations to develop preventive and mitigative measures in order not to exceed the design limit on fatigue usage.

The FMP will be enhanced with guidance for the use of the FatiguePro software package and updated methodology for environmental fatigue factors in establishing updated fatigue life calculations for components, and to add safety relief valve (SRV) actuations for Unit 1 as a monitored transient. These enhancements will be completed prior to the period of extended operation.

C.1.17 Fire Protection Program

The Fire Protection Program provides guidance for performance of periodic visual inspections to manage aging of the various materials comprising rated fire barriers. These include 1) sealants in rated penetration seals (subject to shrinkage due to weathering); 2) concrete and steel in fire-rated walls, ceilings, and floors (subject to loss of material due to flaking and abrasion; separation and concrete damage due to relative motion, vibration, and shrinkage); and 3) steel in rated fire doors (subject to loss of material due to corrosion and wear or mechanical damage). In addition, this program requires testing of the diesel-driven fire pump to verify that it is performing its intended function. This activity manages aging of the fuel oil supply line to, and the exhaust system from, the diesel engine, both of which may experience loss of material due to corrosion. Inspection and testing is performed in accordance with the guidance of applicable standards.

There are two exceptions to the Fire Protection Program as described in NUREG-1801. Inspections on hollow metal fire doors will be performed on a plant-specific schedule, and valve lineups will not be used for aging management of fire suppression systems. These exceptions are consistent with NRC Interim Staff Guidance (ISG) 04.

The Fire Protection Program will be enhanced to include the following: 1) periodic visual inspections of piping and fittings in a non-water environment in the Halon and carbon dioxide (CO_2) fire suppression systems components to detect signs of degradation; 2) periodic functional tests of the diesel-driven fire pump will be enhanced to include inspection of engine exhaust system components to verify that loss of material is managed; and 3) the fire door inspection frequency will be determined by a plant-specific analysis. These enhancements will be completed prior to the period of extended operation.

C.1.18 Fire Water System Program

The Fire Water System Program manages aging of water-based fire protection systems due to loss of material and biofouling. Program activities include periodic maintenance, testing, and inspection of system piping and components containing water (e.g., sprinklers, nozzles, fittings, valves, hydrants, hose stations, standpipes). Inspection and testing is performed in accordance with the guidance of applicable National Fire Protection Association (NFPA) Codes and Standards and the Nuclear Electric Insurance Limited (NEIL) Members' Manual. Enhancements to the Fire Water System Program include the following revisions to existing activities that are credited for license renewal:

1. Incorporate inspections to detect and manage loss of material due to corrosion into existing periodic test procedures.

2. Specify periodic component inspections to verify that loss of material is being managed.

3. Add procedural guidance for performing visual inspections to monitor internal corrosion and detect biofouling.

4. Develop new procedures and preventive maintenance tasks to implement sprinkler head replacement and/or inspections to meet NFPA 25, Section 5.3.1 (2003 edition) requirements.

5. Add requirements to periodically check the water-based fire protection systems for microbiological contamination.

TABLE C-1

COMMITMENTS

Item	Commitment	Source	Schedule
1	Incorporate Appendix A1 into the UFSAR.	· LRA Section A.0	Completed
2	In accordance with 10CFR54.21(b), during NRC review of this application, provide an annual update to the application to reflect any change to the current licensing basis that materially affects the contents of the LRA.	· LRA Section 1.2.1	Completed - Letters dated December 20, 2005, and March 23, 2006
3	Apply for relief from reactor vessel circumferential weld inspections for the period of extended operation. Supporting analyses, procedural controls, and operator training will be completed prior to the period of extended operation to support and confirm that the RPV circumferential weld failure probability remains acceptable for the period of extended operation.	· LRA Section 4.2.3 · LRA Appendix A.1.2.1.3	Completed
4	Supporting analyses will be completed prior to the period of extended operation to confirm that the failure probabilities for the limiting RPV axial welds remain bounded for the period of extended operation.	· LRA Section 4.2.4 · LRA Appendix A.1.2.1.4	Completed
5	For those locations where additional fatigue analysis is required to take advantage of the implicit margin, and to more accurately determine CUF, the EPRI FatiguePro fatigue monitoring software will be implemented prior to the period of extended operation.	· LRA Section 4.3 · LRA Appendix A.1.2.2 · LRA Appendix B.3.2	Completed
6	For the critical reactor vessel components locations shown in Table 4.3-3 of the LRA, additional usage will be added to the baseline CUF using one of the methods described in Section 4.3 of the LRA.	· LRA Section 4.3.1 · LRA Appendix A.1.2.2.1	Completed
7	Transients contributing to fatigue usage of the feedwater nozzles will be tracked by the FMP, with additional usage added to the baseline CUF using the stress-based fatigue method described in Section 4.3 of the LRA.	· LRA Section 4.3.3 · LRA Appendix A.1.2.2.2	Completed
8	Develop a baseline CUF for the specified portions of the following systems: 1. Feedwater/HPCI; 2. Core spray; 3. RWCU (piping inside the RCPB); and 4. Reactor recirculation (and associated shutdown cooling systems lines). If the baseline CUF for a specified portion of a system exceeds 0.4, the limiting locations may require additional monitoring to demonstrate compliance over the period of extended operation.	· LRA Section 4.3.4 · LRA Appendix A.1.2.2.3	Completed

TABLE C-1 (Cont'd.)

Item	Commitment	Source	Schedule
9	Assess the impact of the reactor coolant environment on a sample of critical component locations, including locations equivalent to those identified in NUREG/CR-6260, as part of the FMP. These locations will be evaluated by applying environmental correction factors (F_{en}) to existing and future fatigue analyses.	· LRA Section 4.3.6 · LRA Appendix A.1.2.2.5 · LRA Appendix B.3.2	Completed
10	The FMP will track transients specific to the emergency cooling system with additional usage added to the baseline CUF for the emergency condensers as described in Section 4.3 of the LRA.	· LRA Section 4.3.7 · LRA Appendix A.1.2.2.6	Completed
11	Enhance the FMP to: 1. Ensure that fatigue usage of the torus-attached piping and other torus locations does not exceed the design limits, add ERV lifts as a transient to be counted by the FMP; and 2. Add the two highest usage torus-attached piping locations, the 12-in core spray suction line for core spray pump 111 that enters the torus at penetration XS-337, and the 3-in containment spray line that enters the torus at penetration XS-326 as fatigue monitoring locations.	· LRA Section 4.6.2 · LRA Appendix A.1.2.4.2 · LRA Appendix B.3.2	Completed
12	The RPV weld flaw evaluations will be revised to consider additional fatigue crack growth and the effects of additional irradiation embrittlement (for beltline materials) associated with operation for an additional 20 yr (i.e., out to at least 46 EFPY) and submitted for NRC review and approval no later than 2 yr prior to the period of extended operation. If the revised calculation shows the identified flaws cannot meet the applicable acceptance criteria, the indications will be reexamined in accordance with ASME Section XI requirements.	· LRA Section 4.7.4 · LRA Appendix A.1.2.5.1	Completed
13	Enhance the BWRVIP to address: 1. BWRVIP-18 open item regarding the inspection of inaccessible welds for core spray system. As such, Nine Mile Point will implement the resolution of this open item as documented in the BWRVIP response and reviewed and accepted by the NRC; 2. The inspection and evaluation guidelines for steam dryers are currently under development by the BWRVIP committee. Once these guidelines are documented and reviewed and accepted by the NRC, the actions will be implemented in accordance with the BWRVIP program; 3. The baseline inspections recommended in BWRVIP-47 for the BWR lower plenum components will be incorporated into the appropriate program and implementing documents; and 4. The reinspection scope and frequency for the grid beam going forward will be based on BWRVIP-26A guidance for plant-specific flaw	· LRA Appendix B.2.1.8	Completed

TABLE C-1 (Cont'd.)

Item	Commitment	Source	Schedule
	analysis and crack growth assessment. The maximum reinspection interval for the grid beam will not exceed 10 yr consistent with standard BWRVIP guidance for the core shroud. The reinspection scope will be equivalent to the UT baseline 2005 inspection scope. In addition, the reinspection scope will include an EVT-1 sample inspection of at least two locations with accessible indications within the initial 6 yr of the 10-yr interval. The intent of the EVT-1 is to monitor the known cracking to confirm flaw analysis crack growth assumptions.		
14	Enhance the OCCWS Program to: 1. Ensure that the applicable commitments made for GL 89-13, and the requirements in NUREG-1801, Section XI.M20, are captured in the implementing documents for GL 89-13, "Service Water System Problems Affecting Safety Related Equipment Program Plan;" 2. Incorporate into the OCCWS Program the requirements of NUREG-1801, Section XI.M20, that are more conservative than the GL 89-13 commitments; and 3. Revise the preventive maintenance and heat transfer performance test procedures to incorporate specific inspection criteria, corrective actions, and frequencies.	· LRA Appendix B.2.1.10	Completed
15	Enhance the CCCWS Program to: 1. Expand periodic chemistry checks of the system consistent with the guidelines of EPRI TR-107396; 2. Direct periodic inspections to monitor for loss of material in the piping of the CCCWS; 3. Implement a Corrosion Monitoring Program for larger bore CCCW piping not subject to inspection under another program; 4. Establish the frequencies to inspect for degradation of components in CCCWS, including heat exchanger tube wall thinning; 5. Perform a heat removal capability test for the control room HVAC system at least every 5 yr; 6. Establish periodic monitoring, trending, and evaluation of performance parameters for the RBCLC and control room HVAC; 7. Provide the controls and sampling necessary to maintain water chemistry parameters in CCCWS within the guidelines of EPRI Report TR-107396; and 8. Ensure acceptance criteria are specified in the implementing procedures for the applicable indications of degradation.	· LRA Appendix B.2.1.11	Completed

TABLE C-1 (Cont'd.)

Item	Commitment	Source	Schedule
16	The Boraflex Monitoring Program will be enhanced to: 1. Require periodic neutron attenuation testing and measurement of boron areal density to confirm the correlation of the conditions of test coupons to those of Boraflex racks that remain in use during the period of extended operation; and 2. Establish monitoring and trending instructions for in-situ test results, silica levels, and coupon results.	· LRA Appendix B.2.1.12	Prior to period of extended operation
17	Revise applicable procedures related to the Crane Inspection Program to add specific direction for performance of corrosion inspections, with acceptance criteria, for certain hoist-lifting assembly components.	· LRA Appendix B.2.1.13	Completed
18	Enhance the Compressed Monitoring Program to: 1. Develop new activities to manage the loss of material, stress corrosion cracking, and perform periodic system leak checks; 2. Expand the scope, periodicity, and inspection techniques to ensure that the aging of certain subcomponents of the dryers and compressors (e.g., valves, heat exchangers) are managed; 3. Develop and implement activities to address the failure mechanism of stress corrosion cracking in unannealed red brass piping; 4. Establish activities that manage the aging of the internal surfaces of carbon steel piping and that require system leak checks to detect deterioration of the pressure boundaries; and 5. Expand the acceptance criteria to ensure that the aging of certain subcomponents of the dryers and compressors (e.g., valves, heat exchangers) are managed.	· LRA Appendix B.2.1.14	Completed
19	Enhance the Fire Protection Program to: 1. Incorporate periodic visual inspections of piping and fittings located in a non-water environment, such as Halon and CO_2 fire suppression systems components, to detect evidence of corrosion and any system mechanical damage that could affect its intended function; 2. Expand the scope of periodic functional tests of the diesel-driven fire pump to include inspection of engine exhaust system components to verify that loss of material is managed; 3. Perform an engineering evaluation to determine the plant-specific inspection periodicity of fire doors.	· LRA Appendix B.2.1.16	Completed

TABLE C-1 (Cont'd.)

Item	Commitment	Source	Schedule
20	Enhance the Fire Water System Program by revising applicable existing procedures to: 1. Incorporate inspections to detect and manage loss of material due to corrosion into existing periodic test procedures; 2. Specify periodic component inspections to verify that loss of material is being managed; 3. Add procedural guidance for performing visual inspections to monitor internal corrosion and detect biofouling; 4. Add requirements to periodically check the water-based fire protection systems for microbiological contamination; 5. Measure fire protection system piping wall thickness using non-intrusive techniques (e.g., volumetric testing) to detect loss of material due to corrosion; 6. Establish an appropriate means of recording, evaluating, reviewing, and trending the results of visual inspections and volumetric testing; 7. Define acceptance criteria for visual inspections and volumetric testing; and 8. Develop new procedures and PM tasks to implement sprinkler head replacement and/or inspections to meet National Fire Protection Association (NFPA) 25, "Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems," Section 5.3.1 (2003 edition) requirements.	· LRA Appendix B.2.1.17	Completed
21	Enhance the Fuel Oil Chemistry Program to: 1. Establish a requirement to perform quarterly trending of water and sediment; 2. Provide guidelines for the appropriate use of biocides, corrosion inhibitors, and/or fuel stabilizers to maintain fuel oil quality; 3. Add requirements to periodically inspect the interior surfaces of the emergency diesel fuel oil storage tanks for evidence of significant degradation, including a specific requirement that the tank bottom thickness be determined by UT or other industry-recognized methods; 4. Add a requirement for quarterly trending of particulate contamination analysis results; 5. Ensure acceptance criteria are specified in the implementing procedures for the applicable indications of potential degradation; 6. Establish a requirement for periodic opening of the diesel fire pump fuel oil day tank drain; and 7. Establish a requirement to remove water, if found.	· LRA Appendix B.2.1.18	Completed

TABLE C-1 (Cont'd.)

Item	Commitment	Source	Schedule
22	Enhance the Reactor Vessel Surveillance Program to: 1. Incorporate the requirements and elements of the ISP, as documented in BWRVIP-116 and approved by NRC, or an NRC-approved plant-specific program, into the Reactor Vessel Surveillance Program, and include a requirement that if Nine Mile Point surveillance capsules are tested, the tested specimens will be stored in lieu of optional disposal. When the NRC issues a final SER for BWRVIP-116, Nine Mile Point will address any open items and complete the SER action items. Should BWRVIP-116 not be approved by the NRC, a plant-specific Reactor Vessel Surveillance Program will be submitted to the NRC 2 yr prior to commencement of the period of extended operation; and 2. Project analyses of USE and P-T limits to 60 yr using methods prescribed by RG 1.99, Revision 2, and include the applicable bounds of the data, such as operating temperature and neutron fluence.	· LRA Appendix B.2.1.19	Completed
23	Develop and implement a One-Time Inspection Program, which also includes the attributes for a Selective Leaching of Materials Program.	· LRA Appendix B.2.1.20 · LRA Appendix B.2.1.21	Prior to period of extended operation
24	Develop and implement a Buried Piping and Tank Inspection Program which includes a requirement that before entry into the period of extended operation, if an opportunistic inspection has not occurred, Nine Mile Point will excavate Unit 1 degradation susceptible areas to perform focused inspections. Upon entering the period of extended operation, Nine Mile Point will perform a focused inspection within 10 yr, unless an opportunistic inspection occurred within this 10-yr period.	· LRA Appendix B.2.1.22	Completed
25	An augmented VT-1 visual examination of the containment penetration bellows will be performed using enhanced techniques qualified for detecting SCC, per NUREG-1611, Table 2, Item 12.	· LRA Appendix B.2.1.23	Prior to period of extended operation
26	Enhance the Structures Monitoring Program to: 1. Expand the program to include the following activities or components in the scope of license renewal but not within the current scope of 10CFR50.65: a. The steel electrical transmission towers required for the SBO and recovery paths. 2. Expand the parameters monitored during structural inspections to include those relevant to aging effects identified for structural bolting; and 3. Implement regularly scheduled groundwater monitoring to ensure that a benign environment is maintained.	· LRA Appendix B.2.1.28	Completed

TABLE C-1 (Cont'd.)

Item	Commitment	Source	Schedule
27	Develop and implement a Non-EQ Electrical Cables and Connection Program.	· LRA Appendix B.2.1.29	Completed
28	Enhance the Non-EQ Electrical Cable and Connections Used in Instrumentation Circuit Program to: 1. Implement reviews of calibration or surveillance data for indications of aging degradation affecting instrument circuit performance. The first reviews will be completed prior to the period of extended operation and every 10 yr thereafter; and 2. In cases where a calibration or surveillance program does not include the cabling system in the testing circuit, or as an alternative to the review of calibration results described above, provide requirements and procedures to perform cable testing to detect deterioration of the insulation system, such as insulation resistance tests or other testing judged to be effective in determining cable insulation condition. The first test will be completed prior to the period of extended operation. The test frequency of these cables shall be determined based on engineering evaluation, but the test frequency shall be at least once every 10 yr.	· LRA Appendix B.2.1.30	Completed
29	Enhance the Preventive Maintenance Program to: 1. Expand the PM Program to encompass activities for certain additional components identified as requiring aging management. Explicitly define the aging management attributes, including the systems and the component types/commodities included in the program; 2. Specifically list those activities credited for aging management; 3. Specifically list parameters monitored; 4. Specifically list the aging effects detected; 5. Establish a requirement that inspection data be monitored and trended; and 6. Establish detailed parameter-specific acceptance criteria.	· LRA Appendix B.2.1.32	Completed
30	Enhance the System Walkdown Program to: 1. Train all personnel performing inspections in the Systems Walkdown Program to ensure that age-related degradation is properly identified and incorporate this training into the site Training Program; and 2. Specify acceptance criteria for visual inspections to ensure aging-related degradation is properly identified and corrected.	· LRA Appendix B.2.1.33	Completed

TABLE C-1 (Cont'd.)

Item	Commitment	Source	Schedule
31	Enhance the Non-Segregated Bus Inspection Program to: 1. Expand visual inspections of the bus ducts, their supports and insulation systems; 2. Create new provisions to perform as an alternative to either thermography or periodic low-range resistance checks of a statistical sample of the bus ducts accessible bolted connections, a visual inspection for the connections that are covered with heat shrink tape, sleeving, insulating boots, etc., and 3. Define acceptance criteria for inspection of the bus ducts, their support and insulation systems, and the low-range ohmic checks of connections.	· LRA Appendix B.2.1.34	Completed
32	Develop and implement a Fuse Holder Inspection Program.	· LRA Appendix B.2.1.35	Completed
33	Enhance the Bolting Integrity Program to: 1. The Structures Monitoring, PM, and Systems Walkdown Programs will be enhanced to include requirements to inspect bolting for indication of loss of preload, cracking, and loss of material, as applicable; 2. Include in administrative and implementing program documents references to the Bolting Integrity Program and industry guidance; and 3. Establish an augmented inspection program for high-strength (actual yield strength ≥ 150 ksi) bolts. This augmented program will prescribe the examination requirements of Tables IWB-2500-1 and IWC-2500-1 of ASME Section XI for high-strength bolts in the Class 1 and Class 2 component supports, respectively.	· LRA Appendix B.2.1.36	Prior to period of extended operation
34	Enhance the Protective Coating Monitoring and Maintenance Program to: 1. Specify the visual examination of coated surfaces for any visible defects includes blistering, cracking, flaking, peeling, and physical or mechanical damage; 2. Perform periodic inspection of coatings every refueling outage versus every 24 months; 3. Set minimum qualifications for inspection personnel, the inspection coordinator, and the inspection results evaluator; 4. Perform thorough visual inspections in areas noted as deficient concurrently with the general visual inspection; 5. Specify the types of instruments and equipment that may be used for the inspection; 6. Pre-inspection reviews of the previous two monitoring reports before performing the condition assessment; 7. Establishment of guidelines for prioritization of repair areas and monitoring these areas until they are repaired; and	· LRA Appendix B.2.1.38	Completed

TABLE C-1 (Cont'd.)

Item	Commitment	Source	Schedule
	8. Require that the inspection results evaluator determine which areas are unacceptable and initiate corrective action.		
35	Develop and implement a Non-EQ Electrical Cable Metallic Connections Inspection Program.	· LRA Appendix B.2.1.39	Prior to period of extended operation
36	As acknowledged by the NRC, the ASME Code Committee is evaluating the acceptability of roll/expansion techniques as a permanent repair for CRD stub tubes via Code Case N-730. Nine Mile Point will continue to follow the status of the proposed ASME Code case and will implement the final Code case, as conditioned by the NRC, once it has been approved. If the Code case is approved by ASME but not yet listed in RG 1.147, Unit 1 will seek NRC approval of the Code case on a plant-specific basis as conditioned by the NRC. It will be programmatically required that during the period of extended operation, should a CRD stub tube rolled in accordance with the provisions of the Code case resume leaking, Nine Mile Point will implement one of the following zero leakage permanent repair strategies prior to startup from the outage in which the leakage was detected: 1. A welded repair consistent with BWRVIP-58-A, "BWRVIP Internal Access Weld Repair" and Code Case N-606-1, as endorsed by the NRC in RG 1.147. 2. A variation of the welded repair geometry specified in BWRVIP-58-A subject to the approval of the NRC using Code Case N-606-1. 3. A future developed mechanical/welded repair method subject to the approval of the NRC.	· LRA Appendix B.2.1.8	August 22, 2009
37	Enhance the program to evaluate component susceptibility to loss of fracture toughness. Assessments and inspections will be performed, as necessary, to ensure that intended functions are not impacted by the aging effect.	· LRA Appendix B.2.1.8	Prior to period of extended operation
38	An EVT-1 examination of the Unit 1 feedwater sparger end bracket welds will be added to the BWRVIP. The inspection extent and frequency of the end bracket weld inspection will be the same as the ASME Section XI inspection of the feedwater sparger bracket vessel attachment welds.	· NMP Letter NMP1L 2005, December 1, 2005	Completed
39	The Masonry Wall Program (as managed by the Structures Monitoring Program) will be enhanced to provide guidance for inspecting Unit 1 non-reinforced masonry walls that do not have bracing and are within scope of license renewal more frequently than the reinforced masonry walls.	· NMP Letter NMP1L 2005, December 1, 2005	Completed

TABLE C-1 (Cont'd.)

Item	Commitment	Source	Schedule
40	Unit 1 will perform an EVT-1 inspection of the thermal shield to flow shield weld starting in 2007 and proceeding at a 10-yr frequency thereafter consistent with the ISI inspection interval.	· NMP Letter NMP1L 2005, December 1, 2005	Completed
41	The NRC review of BWRVIP-76 is not yet complete. When the NRC review of BWRVIP-76 is complete, Nine Mile Point will evaluate the NRC SER and complete the SER action item(s), as appropriate.	· LRA Appendix B.2.1.8	Prior to period of extended operation
42	Nine Mile Point will perform volumetric examinations on the Unit 1 drywell shell during the 2007 refueling outage, and an engineering evaluation will be performed to determine the actions necessary for Unit 1 operation through the period of extended operation in accordance with the Drywell Supplemental Inspection Program.	· NMP Letter NMP1L 2037, April 4, 2006	Completed