Niagara Mohawk

Richard B. Abbott
Vice President
Nuclear Engineering

Phone: 315.349.1812
Fax: 315.349.4417

September 18, 2000
NMP1L 1532

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

RE: Nine Mile Point Unit 1
Docket No. 50-220
DPR-63

Subject: ***Proposed Use of ASME Code Section III, Appendix F Criteria***

Gentlemen:

This letter provides the results of analyses performed at Nine Mile Point Unit 1 (NMP1) in response to NRC Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," dated September 30, 1996. GL 96-06 identified the concern that thermally induced overpressurization of isolated water-filled piping sections in containment could jeopardize the ability of accident-mitigating systems to perform their safety functions and could also lead to a breach of containment integrity via bypass leakage.

In a February 7, 1997, letter, Niagara Mohawk Power Corporation (NMPC) identified ten penetrations at NMP1 with associated piping segments that were considered susceptible to thermally induced overpressurization. NMPC's letter indicated that long-term resolution of the overpressurization issue would require detailed analyses and any design changes, if required, would be implemented prior to restart from refueling outage number 15 (RFO15) in 1999. NMPC's subsequent letter dated December 16, 1997, revised the implementation date for required design changes from RFO15 to RFO16, which is scheduled to occur in 2001. In the meantime, during RFO14, which occurred in the Spring of 1997, NMPC modified three of the ten penetrations to provide overpressure protection capability, thus reducing to seven the number of penetrations that remained to be addressed.

By letter dated September 30, 1999, NMPC responded to an NRC supplemental request for additional information regarding the seven remaining penetrations. In that letter, NMPC stated that detailed analyses in accordance with the Appendix F (1986 edition) criteria of Section III of the ASME Boiler and Pressure Vessel Code had been performed to evaluate the capability of the piping to withstand overpressurization under design basis accident conditions. In all seven cases, the piping segment in question was found to be within Level D service

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limits. The letter noted that the original design code for the seven piping segments was ASA B31.1-1955, including Nuclear Interpretations, and the use of ASME Section III Appendix F criteria was not addressed in the NMP1 Updated Final Safety Analysis Report (UFSAR).

NMPC requests that the NRC approve the use of Appendix F criteria for evaluating thermally induced overpressurization of isolated water-filled piping sections on a generic and permanent basis. This request is consistent with the guidance provided in Supplement 1 to GL 96-06, as further elaborated in the Questions and Answers issued on January 28, 1998 (see excerpt below), from the NRC/Nuclear Energy Institute sponsored industry workshop on GL 96-06, which was held on December 4, 1997:

> *"Q4: Can a licensee use Appendix F beyond one outage?*
>
> *A4: Supplement 1 of GL 96-06 addresses this issue. The staff indicated that the use of ASME Code Appendix F is acceptable for interim operability determinations until permanent actions have been identified and approved by the NRC (as applicable) for resolving GL 96-06 issues. A licensee may request to use ASME Code Appendix F criteria for a permanent resolution of the thermal overpressurization issue by submitting a proposed FSAR amendment for staff review."*

Attachment 1 presents the methodology and results of the Appendix F analysis, as applied to the seven remaining NMP1 penetrations and the associated piping and isolation valves. These results provide reasonable assurance that under design basis accident conditions, the analyzed components will continue to perform their intended safety function and containment integrity will be maintained. Attachment 2 presents proposed UFSAR page changes for NRC review, reflecting the generic use of Appendix F.

NMPC requests NRC approval of the use of Appendix F by February 15, 2001, to support the scope of plant activities planned for RFO16, which is scheduled to begin in March 2001. The NMP1 UFSAR will be updated in accordance with 10 CFR 50.71(e) after receiving NRC approval for the use of Appendix F.

Very truly yours,

Richard B Abbott

Richard B. Abbott
Vice President Nuclear Engineering

RBA/IAA/kap
Attachments

xc: Mr. H. J. Miller, NRC Regional Administrator, Region I
Ms. M. K. Gamberoni, Section Chief PD-I, Section 1, NRR
Mr. G. K. Hunegs, NRC Senior Resident Inspector
Mr. P. S. Tam, Senior Project Manager, NRR
Records Management

ATTACHMENT 1

ATTACHMENT 1

ASME CODE SECTION III APPENDIX F ANALYSIS FOR OVERPRESSURIZATION

A. Introduction

The overpressurization scenario envisioned in Generic Letter (GL) 96-06 is a segment of piping, completely filled with water and isolated by closed, leak-tight valves. When the postulated loss of coolant accident (LOCA) occurs and the containment temperature increases, the heat transfer through the pipe wall to the water causes the water to expand. As there is no volume available for the water expansion, the pressure will rise significantly, possibly exceeding the code allowable. If the affected piping is at a containment penetration, this could impact the containment function of the piping or the nearby containment boundary.

As documented in the Nine Mile Point Unit 1 (NMP1) Updated Final Safety Analysis Report (UFSAR), safety-related piping was originally designed and analyzed in accordance with the requirements of ASA B31.1-1955. ASA B31.1-1955 allows for increased stress due to pressure on a temporary basis. However, the allowable increase (20%) is small, and the stresses due to thermal overpressurization during a LOCA would exceed the B31.1 acceptance criteria. No specific design criteria exist in the NMP1 design basis or the original piping code (ASA B31.1-1955) for evaluating isolated piping segments under these conditions. As a result, the loading conditions and the structural integrity evaluation criteria for such an event were not established during initial NMP1 licensing. Also, the particular sections of pipe and the corresponding penetrations that may be susceptible to overpressurization were not identified prior to the issuance of GL 96-06.

In response to GL 96-06, Reference 1 identified the ten penetrations listed in Table 1 as being susceptible to thermal overpressurization under post-LOCA conditions.

Table 1 - List of Susceptible Penetrations

Penetration	System	Inboard/Outboard Isolation Valves
X-7	Shutdown Cooling (to Reactor)	38-13/38-12
X-8	Shutdown Cooling (from Reactor)	38-01/38-02
X-25	Drywell Floor Drain	83.1-11/83.1-12
X-26	Drywell Equipment Drain	83.1-09/83.1-10

X-139[1]	Post-Accident Sampling	110-127/110-128
X-154	Reactor Water Cleanup System	33-01R/33-03
X-14	Core Spray System	40-10, 40-11/40-12
X-238	Core Spray High Point Vent	40-30, 40-31/40-32, 40-33
X-13A	Core Spray System	40-01, 40-09/40-02
X-9	Reactor Water Cleanup System	33-02R/33-04

[1]No Appendix F analysis was performed for this penetration, as clarified in the text.

A structural analysis using the method stipulated in ASME Code Section III, Appendix F (1986 edition), was performed for the penetrations listed above except penetration X-139. Penetration X-139, along with penetrations X-7 and X-8, was modified during RFO14 to provide overpressure protection capability (see Reference 2) so that Appendix F analysis is no longer required. However, NMPC included penetrations X-7 and X-8 in the structural analysis in order to provide a basis for exempting the installed relief valves from future maintenance testing. The structural analysis included the inboard and outboard isolation valves for each of the nine penetrations.

The analysis methodology and analysis results are described in the following sections.

B. Analysis Methodology

The analysis of potential overpressurization of containment penetrations was performed in accordance with ASME Code Section III, Appendix F, criteria for inelastic analysis (1986 edition). Results of the analysis were compared to Level D service limits to determine acceptability. This analysis approach is consistent with the recommendations in Supplement 1 to GL 96-06.

The analysis approach used the maximum water temperature, the piping stiffness, and the piping material properties to calculate the resulting internal pressure and material stress-strain state for the postulated accident scenario. Both elastic and plastic deflection of the piping due to pressure were considered. The deflections were assumed to follow the stress-strain behavior of the penetration material including strain hardening. Specific steps of the analysis methodology are summarized below:

1. The isolated piping section under consideration was reviewed to determine the pipe cross-sections. This review included the main piping run as well as branch piping located between the isolation valves. For penetrations with only one pipe cross-section and material, the analysis was performed on a unit length basis.

Membrane hoop stress due to pressure was calculated. For a given pressure, a piping section with the larger ratio of radius to thickness (r/t) will have a larger resulting membrane stress, and is therefore, limiting.

2. The maximum penetration temperature was based on conditions during a small break loss-of-coolant accident. This is because, for large breaks, the containment begins to cool prior to the pipe reaching maximum temperature. For small breaks, the containment temperature remains high for a longer period, thus resulting in the bounding maximum equilibrium containment temperature.

3. A bi-linear elastic-plastic stress-strain curve for the piping material was developed, considering both the elastic stretching of the pipe wall and plastic deformation of the pipe with strain hardening.

   The yield strength and ultimate strength of each material, along with the elastic and plastic strain values, are temperature dependent material properties. The stress-strain curve was developed based on material properties at the analyzed temperature.

4. An equation of state, relating pressure to temperature and specific volume for the water in the isolated piping conditions was developed. The equation of state is obtained by fitting a set of pressure-specific volume data from the ASME steam tables to a second-degree polynomial equation.

5. At this point, equations relating membrane stress, piping stress and strain, water specific volume and pressure have been defined. Two additional equations, relating specific volume to water mass and pipe radius, and pipe radius to original pipe radius and strain, were added to the above relationships, forming a set of equations which were solved simultaneously to determine final water pressure and piping stress intensity. The equation representing specific volume of the water per inch of pipe was also developed.

6. Steps 1 through 5 developed a set of five equations with five unknowns. The five equations represent the following five unknowns:

   a. Membrane Stress
   b. Stress - Strain Curve (both elastic and plastic)
   c. Equation of State (at maximum equilibrium containment temperature of 261° F)
   d. Specific Volume
   e. Final Pipe Radius

The equations were then solved simultaneously to determine the conditions at the maximum water temperature of 261° F.

In the case of penetrations X-25 and X-26, circumferential and longitudinal thermal expansion of the piping was also considered. Two penetrations, X-9 and X-154, contain multiple materials. For these penetrations, the equations were expanded to address the potential for fluid flow from piping with a stronger material cross section to piping with a weaker material cross section.

In performing the calculations discussed above, several assumptions were made. These assumptions, which are either conservative or have negligible effect on the results of the calculation, are discussed below:

- Several of the penetrations include attached small bore piping for vents, drains, etc. This piping is not included in the set of equations as another piping cross section. However, additional calculations were performed to verify that the small bore piping is stronger than the main run piping and that the expansion of the water in the small bore line would not affect the pressure in the main line.

- The penetration piping is bounded on each end by isolation valves. The isolation valve nozzles are thicker and stronger than the main piping. As a result, the strain in the valves will be less than in the piping. However, the amount of water in the valves is considered to be very small compared to that in the piping. Thus, the expansion of the water in the isolation valves and the possible strain of the valve body can be neglected.

- The valves, being generally stronger than the associated piping (see Section D below), will restrain the piping deflection at the valve connection, preventing the piping from fully yielding and straining (in comparison to the calculated values) at that point. The difference in pipe strain between the calculated value and the actual value due to end effects is neglected.

7. The calculated pressure stresses were combined with the longitudinal stresses from deadweight and seismic loads to determine the stress intensity.

8. The stress intensity due to the combined stresses was compared to the ASME Code Section III, Appendix F limit. Article F-1341.2 of Appendix F specifies an acceptance criterion of:

$$S_I < 0.7S_u$$

where S_I is the calculated stress intensity and S_u is the ultimate strength. Note that article F-1340 of Appendix F only requires the consideration of primary stresses. Therefore, the secondary stresses that would be present at branch connections, elbows, valves, etc. were not evaluated.

In addition to the stress limit specified in the ASME Code, this analysis placed a limit on the calculated strain. Article F-1322.5 of Appendix F states " In addition to the limits given in this Appendix, the strain or deformation limits (if any) provided in the Design Specification shall be satisfied." While none of the applicable Design Specifications identify a material strain limit, it is important that calculated strain remains low enough to ensure that failure will not occur. For this analysis, a limit of 5% strain was applied. Based on engineering judgement, none of the materials used in the penetrations will fail at strains of less than 10%, so the limit of 5% provides a factor of two margin. It should be noted, however, that the primary acceptance criterion utilized in this analysis was the stress limit specified in Appendix F. The specified strain limit is included only for completeness.

9. In the cases where an isolation valve is required to be operated under post-accident conditions, the ability of that valve to open under the calculated differential pressure was evaluated.

10. The structural adequacy of the isolation valves was evaluated as explained in Section D below.

C. Piping Analysis Results

The results of the piping analysis (see Table 2 below) demonstrate that calculated stresses and strains are within the acceptance limits of ASME Code Section III, Appendix F, in all cases.

Table 2. Summary of Piping Analysis Results

Penetration	Material[1]	Pressure (psi)	Stress Intensity (ksi)	Allowable Stress Intensity (ksi)	Strain (%)	Allowable Strain (%)
X-9[2]	CS	5,623	41.9	42.0	1.8	5.0
	SS	5,623	41.9	45.8	4.4	5.0
X-13A	SS	3,576	38.7	47.6	2.5	5.0

X-14	SS	3,576	38.7	47.6	2.5	5.0
X-25	CS	8,778	41.3	42.0	1.5	5.0
X-26	CS	5,391	41.8	42.0	2.0	5.0
X-154[2]	CS	5,591	41.8	42.0	1.8	5.0
	SS	5,591	41.7	45.8	4.4	5.0
X-238	SS	9,696	34.7	45.8	1.5	5.0
X-7	SS	3,586	38.8	47.6	2.5	5.0
X-8	SS	3,586	38.8	47.6	2.5	5.0

Note:

1. CS - Carbon Steel, SS - Austenitic Stainless Steel.
2. Penetrations X-9 and X-154 have sections of carbon steel and stainless steel pipe.

D. Isolation Valves Analysis and Results

The isolation valves will be subject to the same pressure as the interfacing isolated piping segment. The calculation addressed three valve components for structural adequacy under increased pressure: body, bonnet, and body-to-bonnet closure. Table 3 lists applicable valve material and geometry data. Table 4 provides the valve body versus piping strength comparison. This comparison was made by comparing the product of minimum wall thickness and material yield strength for the valve body with that for the piping. Table 4 shows that, with the exception of valves IV-83.1-11 and IV-83.1-12, all of the isolation valve bodies are at least as strong as the interfacing piping and are, therefore, considered structurally adequate.

Drywell floor drain sump discharge line isolation valves 83.1-11 and 83.1-12 (penetration X-25) could not be shown to be as strong as the pipe between the isolation valves. Therefore, a relief valve will be installed in the piping for penetration X-25 to preclude thermal overpressurization. This relief valve will be installed either during or prior to the next refueling outage (RFO16).

ASME/ANSI B16.34-1981, Section 5.1, requires that the valve bonnet be designed to be as strong as the valve body. Based on the reconciliation of NMP1 to B31.1-1986, which invokes ANSI B16.34-1981, the foregoing requirement is considered to have

been implemented in the design of the isolation valves. Therefore, where the valve bodies are structurally adequate, the valve bonnets are also considered structurally adequate.

The core spray inboard isolation valves (penetrations X-13A and X-14) have a post-LOCA safety function. However, under certain small break LOCA scenarios, it has been determined that these valves will not open under the pressure calculated in this analysis. Therefore, penetrations X-13A and X-14 will be modified during RFO16 to provide overpressure relief capability to ensure this post- LOCA safety function is maintained.

Table 3. Valve Data[1]

Valve Data					Valve Geometry and Material Data		
Valve ID	Size (in)	Pressure Class	Body Material	Ref.	d (in)	t (in)	Body S_y (ksi)
38-01 38-13	14	900	A351-CF8M	5	12.25	1.38	24.3
38-02	14	900	A351-CF8M	4	12.25	1.38	24.3
38-12	14	1500	A351-CF8M	6,7	11.37	2.19	24.3
33-01R, 33-02R	6	900	A351-CF8M	8	5.75	0.72	24.3
33-03	6	900	A479-316L	9,10	5.75	0.72	19.8
33-04	6	1500	A351-CF8	11	5.37	1.09	23.5
83.1-09	3	150	A216 WCB	12	3.00	0.22	32.3
83.1-10	3	150	A352 LCB	13	3.00	0.22	31.4
83.1-11	4	150	A216 WCB	14	4.00	0.25	32.3
83.1-12	4	150	A352 LCB	15	4.00	0.25	31.4
40-30 40-31 40-32 40-33	1	1500	A182-F316	16,19	0.87	0.26	24.3

[1] Table does not include valve data for penetrations X-13A and X-14, which will be modified in RFO16.

Table 4. Valve Body Versus Piping Strength Comparison[1]

Valve	Penetration	$S_{y,\ pipe}$ (ksi) (Ref. 1)	t_{pipe} (in) (Ref. 1)	$S_{y,\ valve}$ (ksi)	t_{valve} (in)	S_y*t pipe	S_y * t valve
38-01, 38-13	X-7/X-8	23.5	0.685	24.3	1.38	16.1	33.5
38-02	X-8	23.5	0.685	24.3	1.38	16.1	33.5
38-12	X-7	23.5	0.685	24.3	2.19	16.1	53.2
33-01R, 33-02R	X-9/X-154	19.8	0.432	24.3	0.72	8.6	17.5
33-03	X-154	19.8	0.432	19.8	0.72	8.6	14.3
33-04	X-9	19.8	0.432	23.5	1.09	8.6	25.6
83.1-09	X-26	31.4	0.216	32.3	0.22	6.8	7.1
83.1-10	X-26	31.4	0.216	31.4	0.22	6.8	6.9
83.1-11	X-25	31.4	0.438	32.3	0.25	13.8	8.1
83.1-12	X-25	31.4	0.438	31.4	0.25	13.8	7.9
40-30, 40-31 40-32, 40-33	X-238	19.8	0.179	24.3	0.26	3.5	6.3

[1] Table does not include strength comparison for penetrations X-13A and X-14, which will be modified in RFO16.

E. REFERENCES

1. NMPC letter dated February 7, 1997, NRC Generic Letter 96-06, "Assurance of Equipment Operability and Containment".

2. NMPC letter dated September 30, 1999, Supplemental Request for Additional Information Regarding Generic Letter 96-06.

ATTACHMENT 2

(Consists of Unit 1 UFSAR pages VI-23, VII-2, X-1, X-4, XVI-32, XVI-71, and P. 1 of 1 of Table V-4)

TABLE V-4

CODES FOR SYSTEMS CONNECTED TO THE REACTOR COOLANT SYSTEM
(UP TO THE OUTSIDE OF SECOND ISOLATION VALVE)

	Piping Vessel Nozzle to Second Isolation Valve	Isolation Valves
Shutdown Cooling	ASME Sec I*	ASME Sec I-1962
Cleanup	ASME Sec I*	ASA B31.1-1955 and certain requirements of ASME Sec IIIA-1965
Feedwater	ASME Sec I*	ASME Sec I-1962
Core Spray	ASME Sec I*	ASA 31.1-1955 and certain requirements of ASME Sec IIIA-1965
Liquid Poison	ASME Sec I*	ASME Sec I-1962

ASME Sec III
Appendix F
(1986 Edition)**

* With requirements of Sec III for nondestructive testing.
** For analyzing thermally induced overpressurization conditions between isolation valves

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 were ASME Section I-1965, 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.

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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.

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).

A. CORE SPRAY SYSTEM

1.0 Design Bases

The core spray system consists of two separate and independent core spray loops to prevent overheating of the fuel following a postulated loss-of-coolant accident (LOCA). Each loop has redundant active components within itself. This system is designed to accommodate the range of LOCAs from the smallest up to the largest line break as discussed in Section XV. An environmental qualification (EQ) program for electrical equipment has been conducted in accordance with 10CFR50.49. As a result of this program, electrical equipment in the core spray important to safety has been qualified to operate in the environment to which it is exposed.

2.0 System Design

2.1 General

The core spray system shown on Figure VII-1 is designed in accordance with the ASA B31.1-1955 Piping Code with certain requirements of the ASME Code, Section II-B-1965, for the piping and valves. The pump casings and discharge strainer housings (baskets removed) are designed in accordance with the ASME Code, Section III-B-1965. The system is of welded construction and where flanged connections are necessary double seals are used. Equipment and piping between the suppression chamber and the topping pumps are designed to 310 psig and for a temperature range of -10°F to 205°F. An exception to the 310 psig design pressure for this portion of the system is the core spray pump motor cooling coils which are designed to 100 psig. Cooling water for the motor coolers flows from the discharge side of each core spray pump, through pressure control valves set at 75 psig, through the motor cooling coils, and then to the suction side of the pump. Relief valves set at 85 psig are provided in the cooling water line to provide overpressure protection for the motor cooling coils. The piping and equipment after the topping pump to the containment system isolation check valve are designed to 470 psig and the same temperature range. The system outside the drywell is also qualified to a maximum torus bulk temperature of 165°F applied to a thermal stress analysis. These pressures include the pump shutoff heads and postaccident design pressure. The equipment and piping (carbon steel) between the suppression chamber and the isolation check valve will not be subjected to reactor pressure and temperature. Safety valves set at 349 psig are utilized for pressure protection of this section of the system. They also provide minimum flow for pump protection before vessel depressurization during small break LOCA. From the isolation check valve to the reactor, the system is fabricated of stainless steel with a design pressure of 1200 psig and design temperature of 575°F.

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Isolated piping sections considered susceptible to thermal overpressurization are analyzed in accordance with the criteria of the 1986 ASME Boiler & Pressure Vessel Code, Section III, Appendix F.

SECTION X

REACTOR AUXILIARY AND EMERGENCY SYSTEMS

A. REACTOR SHUTDOWN COOLING SYSTEM

1.0 Design Bases

This system, shown on Figure X-1, is designed to cool reactor water below temperatures and pressures at which the main condenser may be used as a heat sink following reactor shutdown. Once the reactor water has been cooled to about 350°F by the main condenser, the shutdown cooling system is used to cool the reactor water down to 125°F and maintain it at this temperature by removing fission product decay heat absorbed by the reactor water.

2.0 System Design

At or below a reactor pressure of 120 psig and a temperature of 350°F, the shutdown cooling system may be manually actuated from the main control room. The reactor water enters this system from the suction side of one of the reactor recirculation pumps, flows through the partial-capacity shutdown cooling system loops, then discharges into the discharge side of another reactor recirculation loop pump. Heat removal requirements are variable depending on the operating condition prior to shutdown. Usually two heat exchangers and two pumps are sufficient to provide a desirable cooldown rate (100°F/hr maximum allowable). If necessary, all three loops of the shutdown cooling system can be used. The heat exchangers are cooled by demineralized water from the reactor building closed loop cooling water (RBCLCW) system, described in Section X-D. Each of the three heat exchangers is designed for a heat load of 12,500,000 Btu/hr.

The equipment in this system is designed and constructed in accordance with ASME Pressure Vessel Code 1965, Section III, Class C, for the parts in contact with reactor water. Piping, valves other than those for isolation, and fittings are built to the requirements of ASA B-31.1-1955 with nuclear interpretations. The segment of the system from the recirculation loops to the isolation valves is designed for 1200 psig and 575°F.

All equipment in the shutdown cooling system is designed to withstand earthquake acceleration factors of 0.20g horizontal and 0.10g vertical. The heat exchanger tubes, piping, valves and fittings are constructed of stainless steel. The pump casing is made of carbon steel.

All components of the shutdown cooling system are located in a common concrete-shielded compartment designed to attenuate radiation levels to 5 mr/hr outside the compartment.

ADD

[Isolated piping sections considered susceptible to thermal overpressurization are analyzed in accordance with the criteria of the 1986 ASME Boiler & Pressure Vessel Code, Section III, Appendix F.]

of a portion of the cleanup flow, when effectiveness of the regenerative heat exchanger is reduced. Blowdown is normally used during reactor operation only to remove excess water from the reactor. This blowdown provision is also used as an alternate route for removing refueling water back to the condensate storage tanks (CST) via the main condenser and condensate demineralizers (CND).

A cleanup surge tank is provided to assure continuous submergence for the cleanup pumps and to provide a path for the pump recirculation flow.

The equipment in this system is designed and constructed in accordance with ASME Pressure Vessel Code, Section III-1965, Class C. Piping, valves other than those for isolation, and fittings are built to the requirements of ASA B-31.1-1955 with nuclear interpretations.

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All equipment in the cleanup system is designed to withstand earthquake acceleration factors of 0.20g horizontal and 0.10g vertical. The equipment and the piping subject to reactor water are constructed of stainless steel, rubber-lined carbon steel, or carbon steel. The primary side of the system subject to the reactor pressure is designed to withstand a pressure of 1300 psig and temperature of 575°F.

Operation of the cleanup system is controlled from the main control room. Conductivity of the water entering the cleanup filters and leaving the cleanup demineralizers is recorded in the control room. Filter backwash and resin sluicing operations are controlled from a local panel. Filters are automatically backwashed but are precoated by remote manual operation.

All equipment is shielded with concrete except for the main cleanup pumps, surge tank, flow control valve, filter aid tanks and pumps, and precoat tank. The bases for shielding design were determined by the estimated frequency of operating, inspecting and maintaining the various equipment and devices. The shielding is designed to reduce the radiation levels in the valve corridors to 30 mr/hr, and the levels in the access corridors around the cleanup system complex to 5 mr/hr. The unshielded equipment is located in controlled access areas.

3.0 System Evaluation

The dual full-capacity filters and demineralizers permit maintenance of these units during full-flow operation of the system through the spare units. The filter precoating and filter aid pumps are also duplicated for reliability of the system.

The system is provided with relief valves and instrumentation, shown on Figure X-2, to protect against overpressurization of the equipment and overheating of the resins. The cleanup system is

Isolated piping sections considered susceptible to thermal overpressurization are analyzed in accordance with the criteria of the 1986 ASME Boiler & Pressure Vessel Code, Section III, Appendix F.

2.8.6 Material Specification

Seamless carbon steel (ASTM - A106, Grade B) - Main steam and feedwater piping.

Seamless stainless steel (ASTM - A376, Type 304) - Other process piping.

Carbon steel (ASTM - A212, Grade B (Fire Box) to ASTM - A300) - Containment vessel nozzles and guard pipe.

Stainless steel (ASTM - A240, Type 321) - Bellows.

To satisfy NDTT requirements, all carbon steel process piping was subjected to a Charpy impact test at -10°F in accordance with Article 12, Section III, of the ASME Nuclear Vessel Code.

2.8.7 Applicable Codes

The applicable Codes for the containment penetrations were considered and subdivided in the following manner:

1. The process piping, including valves, was designed, constructed, and tested in accordance with the following: a) Code for Pressure Piping USAS B31.1, including the latest addenda and nuclear interpretations; b) Section I, the ASME Boiler and Pressure Vessel Code and Section III, Article N-324, Non-Destructive Examination, and Articles N-460 to N-469 inclusive, Design of Welded Construction.

2. The nozzles on the containment vessel are an integral part of the containment vessel and are designed in accordance with ASME Code, Section III, Subsection B. Stresses do not exceed limits stated in paragraph N-1312 (f).

3. All containment nozzle extension piping to the body of the external isolation valve was designed, constructed, and tested in accordance with the Code for Pressure Piping USAS B31.1, including the latest addenda and nuclear interpretations, and with ASME Code, Section III, Subsection B, Articles 11 through 14. The impact tests were conducted in accordance with Article 12 at -10°F. All full-penetration welded joints were fully radiographed in accordance with ASME Section III, Par. N-624, and Section VIII, UW-51.

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4. CRD containment penetrations are designed, fabricated, tested, and inspected in accordance with the rules for Class A vessels, ASME Section III, Summer 1967 addenda, as shown on the detail given in N-462.4 (d) of the Code.

Piping system segments penetrating containment and considered susceptible to thermally induced overpressurization in the event of an accident were analyzed in accordance with the criteria of the ASME Boiler & Pressure Vessel Code, Section III, Appendix F (1986 Edition).

Cables for reactor protection and engineered safeguards systems are routed in accordance with IEEE "Single Failure Criteria" via separate paths between the remote equipment and the control room equipment. Two routes are below grade, at el 261; two routes are above grade, at el 261; both sets run north-south at opposite ends of the turbine island. When below-grade cables must be extended to remote equipment terminations above grade (or above el 340 in the reactor building), individual rigid steel conduit is used for protection and isolation of the cables.

Graded scales of action, governed by weather conditions, progress from normal Station operation to emergency shutdown. Action is as follows:

Condition Green No U.S. Weather Bureau tornado warnings exist within a 60-mi radius of the Station.

Condition Yellow U.S. Weather Bureau tornado warning exists within a 60-mi radius of the Station.

Condition Red Sustained steady winds above 100 mph.

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5.0 Thermally Induced Overpressurization of Isolated Piping

NRC Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," identified the concern that thermally induced overpressurization of isolated water-filled piping sections in containment could jeopardize the ability of accident -mitigating systems to perform their safety functions and could also lead to a breach of containment integrity via bypass leakage. Piping system segments considered susceptible to thermally induced overpressurization are analyzed in accordance with the criteria of the ASME Boiler & Pressure Vessel Code, Section III, Appendix F (1986 Edition).