NMC
Committed to Nuclear Excellence

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10 CFR 50.90

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

Ladies/Gentlemen:

DOCKETS 50-266 AND 50-301
LICENSE AMENDMENT REQUEST 223
CONTAINMENT PRESSURE
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

In accordance with the provisions of 10 CFR 50.90, Nuclear Management Company, LLC (NMC) is submitting a request for an amendment to the Technical Specifications (TS) for Point Beach Nuclear Plant, Units 1 and 2.

The proposed amendment would revise TS 3.6.4, Containment Pressure, to reduce the maximum allowable pressure from three psig to two psig. The changes are needed to address a nonconservatism that was identified during reviews of the PBNP accident analyses.

Attachment I provides a description, justification and safety analysis, and No Significant Hazards Consideration for the proposed change. Attachment II provides the existing TS pages marked up to show the proposed change. Attachment III provides the existing TS Bases pages marked up to show the proposed change (for information only). Attachment IV provides revised (clean) TS pages.

NMC requests approval of the proposed License Amendment by July 2002, with the amendment being implemented within 45 days. The approval date was administratively selected to allow for NRC review.

Sincerely,

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A001

Subscribed to and sworn before me
on this 11 day of January, 2002

Paula C. Marcec
Notary Public, State of Wisconsin

My Commission expires on 10/24/2004.

JG/kmd

Attachments:
I - Description and Assessment
II - Proposed Technical Specification Changes
III - Proposed Technical Specification Bases Changes
IV - Revised Technical Specification Pages

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DESCRIPTION AND ASSESSMENT OF CHANGES

LICENSE AMENDMENT REQUEST 223

CONTAINMENT PRESSURE

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

1.0 INTRODUCTION

This proposed License Amendment Request (LAR) is made pursuant to 10 CFR 50.90 to modify Technical Specification (TS) 3.6.4, Containment Pressure, to reduce the maximum allowable pressure from three psig to two psig. The changes are needed to address a nonconservatism that was identified during reviews of the PBNP accident analyses. The nonconservatism in the analysis consisted of not addressing the case of a single failure of a feedwater regulating valve (FRV) at full power.

2.0 BACKGROUND

A reanalysis of the Main Steam Line Break (MSLB) containment integrity analysis has been performed to confirm that the peak calculated pressure is maintained below the containment design pressure of 60 psig. The reanalysis was performed to address the case of a single failure of a feedwater regulating valve (FRV) at full power. This case was determined to be the worst case with respect to the peak containment pressure.

System Information

The Containment System structure is described in Section 5.1 of the Point Beach Nuclear Plant Final Safety Analysis Report. The structure is a right cylinder with a flat base and a shallow domed roof. A ¼ in. thick welded steel liner is attached to the inside face of the concrete shell to insure a high degree of leak tightness. The base liner is installed on top of the slab and is covered in concrete. The structure provides biological shielding for both normal and accident situations.

Design Basis

The Containment System structure completely encloses the entire reactor and reactor coolant system and ensures that an acceptable upper limit for leakage of radioactive materials to the environment is not exceeded even if gross failure of the reactor coolant system occurs. The containment structures of Units 1 and 2 are designed to maintain leakage no greater than 0.4% per 24 hours of containment air weight at a design pressure of 60 psig and 286°F.

System Design And Operation

The Containment System structure is designed to maintain its integrity and the design basis leakage limit as described above. It is designed to withstand accident pressure loading at least 50% greater than those calculated for the postulated loss-of-coolant accident (LOCA) alone, or 25% greater when combined with a design basis earthquake or hypothetical wind loading.

In the event of a design basis accident such as a LOCA or a MSLB, the internal containment pressure will rapidly increase as fluid is released from the reactor coolant system or the secondary system, respectively. The systems, structures, and components (SSCs) that mitigate the effects on the containment structure of this pressure transient include the containment fan coolers (CFCs), the containment spray (CS) system, and the containment structure itself. As the fluid blowdown progresses and the internal temperature increases, the components that make up the containment structure (concrete, steel liner, etc) will act as a heat sink and begin to absorb the released energy. Engineered safeguards systems (CFCs and CS) will actuate after a time delay, and also begin to remove energy from the containment atmosphere. As the fluid blowdown decreases, a peak containment pressure is reached at the point when the energy addition and energy removal rates are approximately equal. Analysis is performed to demonstrate that the peak containment pressure remains below the design pressure of 60 psig, thus meeting the containment design pressure conditions described above. Continued heat removal results in a gradual decrease in containment pressure, and analyses are typically run out for some time interval after the peak pressure to demonstrate a stable decreasing trend.

Note that in the case of a MSLB with a single failure of a FRV, it is assumed that both trains of containment safeguards (CS and CFCs) function during the accident.

3.0 DESCRIPTION OF CHANGE

The proposed amendment would revise TS 3.6.4, Containment Pressure, to reduce the maximum allowable containment pressure from 3.0 psig to 2.0 psig.

The proposed amendment revision is as follows (additions are double-underlined; deletions are strikethrough):

LCO 3.6.4 Containment pressure shall be ≥ -2.0 psig and ≤ +~~3~~2.0 psig.

The proposed Bases revisions for TS B 3.6.4 support the proposed TS change. The revised Bases remove information regarding the previous analysis of the steam line break (SLB) case and add information describing the revised analysis results.

The initial pressure condition used in the SLB containment reanalysis was 16.7 psia (2.0 psig). This resulted in a maximum peak pressure from the limiting SLB inside containment of 59.8 psig. The new limiting SLB case assumed the failure of a feedwater regulating valve at 102% of rated thermal power. The SLB containment analysis shows that the maximum peak calculated containment pressure results from this limiting SLB case. Neither the limiting SLB case nor the limiting LOCA case exceed the containment design pressure of 60 psig.

A statement was added to specify that measurement uncertainties for containment pressure are not included in the new 2.0 psig limit.

The Bases for TS 3.6.5, Containment Air Temperature were also similarly revised to reflect the results of this new analysis.

4.0 ANALYSIS

4.1 Technical Justification

The proposed amendment will limit the maximum allowed containment pressure during normal reactor operation to two pounds per square inch. The existing Technical Specifications permit three pounds per square inch. Reducing the maximum allowed pressure is conservative, in that this action reduces the peak pressure that will occur in containment. An accident initiated with an initial containment pressure of 2.0 psig will result in a lower peak containment pressure than would result from an accident initiated with an initial containment pressure of 3.0 psig. Reduction of the maximum allowed containment pressure will result in increased assurance that the containment will meet its design and licensing bases under any credible post-accident condition.

Impact on Environmental Qualification of Equipment

Based on the previous analysis, the calculated peak containment pressure was less than 53 psig. The revised analysis (using the 2 psig limit) predicts a peak containment pressure of 59.8 psig. Although both values are less than the containment design pressure of 60 psig, a review of equipment inside containment was performed to provide assurance that the equipment was qualified for higher calculated peak pressure.

A query of all current Equipment Qualification Summary Sheets (EQSS) was performed to identify the Environmental Qualification components which may be affected by a change in pressure inside containment. Upon reviewing the items listed as being inside containment, the following items were found to be in question: Okonite T-95 & #35 tapes, Scotch 130C & 33+ tapes, and Hardline Coaxial Cable. A review of the qualification documentation data for these items concluded that they are sufficient to be qualified to the new pressure requirement inside containment of 59.8 psig.

Conclusions

NMC has concluded that reducing the maximum allowed containment pressure limit provides additional conservatism for containment integrity and that there is reasonable assurance that the calculated peak containment pressure change will not impact equipment qualification. Therefore, the reduction in maximum allowed pressure for the containment from 3.0 psig to 2.0 psig is justified and supports this amendment for the incorporation of the changes to the PBNP Technical Specifications.

4.2 Safety Analysis

The proposed amendment will reduce the maximum containment pressure allowed under normal operating conditions from 3.0 psig to 2.0 psig. This change is requested to allow the initial condition of the containment pressure assumed in the Main Steam Line Break (MSLB) containment response analysis (Reference 1) to be set to 2.0 psig. Sensitivity studies performed by the reactor vendor, Westinghouse, indicated that a reduction in the initial containment pressure would be needed to limit the peak containment pressure during a MSLB inside containment assuming a failed open feedwater regulating valve (FRV), to less than the design pressure of 60 psig.

Limiting Design Basis Accident Analysis

The current analysis of record for the MSLB containment response was not prepared as a PBNP specific analysis. It was based on a LOFTRAN/COCO analysis of a reference 2-loop plant. A parameter by parameter comparison of the input assumptions for the reference 2-loop plant and PBNP was performed, with benefits or penalties assigned based on the differences (Reference 1). This analysis assumed that the worse case was a double ended rupture (DER) at hot zero power with a single failure of a train of containment safeguards equipment. However, later studies performed for PBNP by Westinghouse for the power uprate program, indicated that the worse case was actually a DER at hot full power with a single failure of the FRV to close (Reference 2). In evaluating this scenario, it was determined that this was also the worse case for the current licensed power level (Reference 3). This case is limiting because of the relatively high energy transfer to the faulted steam generator and the early flashing of the large quantity of feedwater in the unisolable feed line. Therefore, the NMC contracted Westinghouse to prepare a PBNP specific analysis of the MSLB containment response at hot full power assuming a single failure of the FRV. The following discussion summarizes the new MSLB containment integrity analysis performed by Westinghouse. (Reference 10)

Mass and Energy Release Analysis

The steam line break mass and energy releases are generated using the NRC-approved LOFTRAN code (Reference 4). LOFTRAN is used for studies of the transient response of a PWR system to specified perturbations in process parameters. The code simulates a multi-loop system including the reactor vessel, hot and cold leg piping, steam generator (shell and tube sides), and the pressurizer. A neutron point kinetics model is used and the reactivity effects of the moderator, fuel, boron, and rods are included. The secondary side of the steam generator is modeled as a homogeneous saturated mixture. Protection and control systems are simulated, as well as the Emergency Core Cooling System. The calculation of secondary side break flow is based on the Moody critical flow correlation (Reference 5) with $fL/D = 0$.

The analysis was performed using the Westinghouse steam line break mass and energy release methodology documented in WCAP-8822, "Mass and Energy Release Following a Steam Line Rupture" (Reference 6). WCAP-8822 forms the basis for the assumptions and models used in the calculation of the mass and energy releases resulting from a steam line rupture.

Major assumptions affecting the mass and energy releases to containment are summarized below.

- The initial power level is 102% of 1524.5 MWt.
- The initial RCS average temperature is 575.6°F, which includes a +5.6°F uncertainty.

- The core nuclear power transient due to the cooldown following the steam line rupture is based on end-of-core life conditions with the most reactive control rod stuck out of the core. The credited shutdown margin is 3.1%Δk. LOFTRAN's point kinetics core model was confirmed via statepoints with a detailed 3D neutronics model.
- Two sources of latent energy to the reactor coolant system are modeled: the reactor vessel and primary system piping thick metal, and the fluid inventory in the intact steam generator.
- Offsite power is assumed to remain available. The largest effect of this assumption is the continued operation of the reactor coolant pumps, which maintains a high heat transfer rate to the steam generators.
- Minimum flowrates are modeled from ECCS injection, to conservatively minimize the amount of boron that provides negative reactivity feedback. The flowrates correspond to a single train of ECCS with 10% pump head curve degradation. Note that this is a significant conservatism since the assumed single failure of the FRV would allow credit of both trains of ECCS.
- A high initial steam generator mass is assumed. The initial level corresponds to 64% NRS + 4% uncertainty.
- The main feedwater modeling accounts for an increase from the initial flowrate due to the depressurization of the faulted steam generator and the opening of the FRV in response to the increased steam flow. Main feedwater is terminated by the trip of the main feedwater pumps. The pump speed is assumed to linearly decrease over 40 seconds; the ΔP across the pumps decreases based on the square of the pump speed, and pumped flow to the faulted steam generator terminates before the main feedwater pump has completely stopped.
- Feed line flashing occurs when saturated conditions are reached in the 1198 ft^3 unisolable volume between the faulted steam generator and the main feedwater pump discharge valves. The homogenous flashing model in LOFTRAN was used, but with two separate volumes to account for the water that was heated to 430°F by the feedwater heaters, and the water upstream of the heaters at a temperature of 350°F.
- Maximum flowrates of auxiliary feedwater were assumed, with the auxiliary feedwater conservatively modeled at the time of the SI signal, with no delay. The auxiliary feedwater is assumed to be manually re-aligned at 600 seconds to prevent further water addition to the faulted steam generator.
- The steam in the unisolable volume of 1650 ft^3 between the faulted steam generator and the steam line non-return check valve comprises the reverse flow from the break.
- The break effluent is assumed to be dry, saturated steam throughout most of the transient. However, when a large double-ended break first occurs, it is expected that there will be a significant quantity of liquid in the break effluent. An evaluation was done based on reviewing the NRC-approved entrainment analyses performed with the TRANFLO code (Reference 7) for other steam generator types. The results of large double-ended breaks were found to be largely insensitive to the steam generator design. Therefore, entrainment (i.e., break quality less than 1.0) input is included in this analysis, and includes an uncertainty of 0.10 quality compared to the TRANFLO-calculated values. The result is that LOFTRAN models less than 60% of the integrated liquid flowrate that was predicted by TRANFLO, and the break effluent is assumed to return to all vapor within the first 25 seconds.

- The elevated containment pressure is modeled within LOFTRAN as a function of time. An elevated containment pressure causes the transition from critical to non-choked break flow to occur earlier in the transient, reducing the break flowrate. The elevated containment back pressure is modeled in the steam line break analyses to make the interface between the steam line break analysis and the containment response analysis more consistent and realistic.

A final assumption in the steam line break mass and energy releases is regarding the heat transfer to the faulted steam generator, which is different than identified in WCAP-8822 (Reference 6). As discussed in Reference 6, the film coefficient on the outside of the tubes and the forced convection from the reactor coolant pumps will typically maintain a large secondary side heat transfer coefficient. The only mechanism for reducing the heat transfer capability to the steam generator is to lower the effective heat transfer area. Such a reduction occurs when sufficient mass is lost from the steam generator to lower the water level below the top of the tube bundle. To conservatively force a high heat transfer rate to the faulted steam generator, the Westinghouse practice in steam line break inside containment analyses is to typically model tube uncovery well after the time that it is anticipated.

In both WCAP-8822 and Supplement 2 to WCAP-8822 (Reference 8), there are discussions of small variations when the tube bundle uncovery time is delayed. The assessments focus on the effect on the integrated mass and energy released, and also show a relatively small effect on peak containment pressures that occur after 600 seconds. For Point Beach, the containment pressure peaks earlier, and thus there may be a heightened sensitivity to the timing of the event. While tube uncovery has little effect on the integrated mass and energy released during the event, it does affect the transient rate of release. Thus, more realistic time of tube uncovery is modeled in the same manner as was used in Reference 6 for "predicted tube uncovery" cases. In addition, the heat transfer to the uncovered tube region is modeled, which may result in superheated steam.

Containment Response Analysis

The COCO computer code (Reference 9) is used to analyze the containment pressure and temperature transient response following the postulated steam line break accidents presented in this amendment. COCO is a mathematical model of a generalized containment; the proper selection of various options in the code allows the creation of a specific model for the particular containment design.

Initial conditions, as listed in Table 1, are selected to maximize the containment pressure response. The initial pressure has a direct relationship on the peak containment pressure, and thus is maximized. The initial temperature is maximized because the steady-state temperature of the containment heat sinks are assumed to be the same as the containment air temperature. The higher initial heat sink temperature causes them to be less effective in removing heat. The initial humidity is conservative when it is assumed to be low, since this maximizes the amount of air initially assumed in the containment. The moles of air are non-condensable, and thus will maximize the containment pressure response as the containment temperature increases.

Table 1 Containment Initial Conditions	
RWST water temperature for containment sprays (°F)	100
Initial containment temperature (°F)	120
Initial containment pressure (psia)	16.7
Initial relative humidity (%)	20
Net free volume (ft^3)	1.0×10^6

The containment fan coolers each have a fan which draws in the containment atmosphere and the steam/air mixture is routed through the enclosed fan cooler unit, past service water cooling coils. The fan then discharges the air back to the containment. Note that all 4 fan coolers are credited; a limiting single failure of the FRV has already been modeled in the mass and energy release calculation.

The containment spray system flowrate was modeled as function of containment pressures. During the steam line break blowdown, the containment spray pumps draw water from the refueling water storage tank (RWST) and spray it into the containment through nozzles mounted high above the operating deck. A conservatively high temperature of 100°F has been assumed as the temperature of the spray water.

Finally, the heat transfer through, and heat storage in, interior and exterior walls of the containment structure are considered. Structural heat sinks, consisting of steel and concrete, are modeled as slabs having specific areas and layers of varying thickness. The thermal conductivity, density and specific heat of each layer are specified.

Results and Conclusion

The analysis shows that the peak containment pressure of 59.8 psig is reached at 276 seconds. The peak containment pressure is less than the containment design pressure of 60 psig, and is therefore acceptable.

Note that sensitivities were also performed to determine the effect of the containment spray temperature and the initial containment temperature (applied to the air, heat sinks, and outside environment). It was found that if the containment spray temperature was decreased by 20°F (from the assumed value of 100°F), the peak pressure would be reduced by approximately 0.5 psig. If the initial containment temperature was decreased by 20°F (from the assumed value of 120°F), the peak pressure would be reduced by approximately 0.9 psig. Typically, the values for these two parameters are well below the assumed values. This demonstrates that there is additional margin to the containment design pressure during most of the operating cycle.

The analysis also shows that the peak containment temperature of 285°F is reached at 276 seconds. This value is below the peak containment temperature predicted for the LOCA of 291°F, and is thus bounded by the LOCA.

Based on the above discussion, implementation of the proposed Technical Specification change is consistent with the MSLB containment response analysis, and demonstrates that the peak containment pressure is maintained below the containment design pressure.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Determination

In accordance with the requirements of 10 CFR 50.90, Nuclear Management Company (licensee) hereby requests amendments to facility operating licenses DPR-24 and DPR-27, for Point Beach Nuclear Plant, Units 1 and 2, respectively. The purpose of the proposed amendments is to revise Technical Specifications to reduce the maximum allowed containment pressure limit at Point Beach.

Nuclear Management Company has evaluated the proposed amendments in accordance with 10 CFR 50.91 against the standards in 10 CFR 50.92 and has determined that the operation of the Point Beach Nuclear Plant in accordance with the proposed amendments presents no significant hazards. Our evaluation against each of the criteria in 10 CFR 50.92 follows.

1. **Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant increase in the probability or consequences of any accident previously evaluated.**

The operability of containment ensures that radionuclides are contained within allowable limits during and following all credible accident conditions. The inoperability or failure of containment is not a design basis accident initiator or precursor. Therefore, the probability of an accident previously evaluated will not be significantly increased as a result of the proposed change. Because design limitations continue to be met and the integrity of the containment system pressure boundary is not challenged, the assumptions employed in the calculation of the offsite radiological doses remain valid. In addition, the radiological consequence analysis for the main steam line break (MSLB) is performed assuming the MSLB is outside of the containment. Therefore, the operability of the containment structure does not affect the results of the offsite dose or control room dose consequences.

Therefore, the consequences of an accident previously evaluated will not be significantly increased as a result of the proposed change.

2. **Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a new or different kind of accident from any accident previously evaluated.**

The possibility for a new or different type of accident from any accident previously evaluated is not created as a result of this amendment. The evaluation of the effects of the proposed changes indicate that all design standards and applicable safety criteria limits are met. These changes, therefore, do not cause the initiation of any new or different accident nor create any new failure mechanisms.

Equipment important to safety will continue to operate as designed. Component integrity is not challenged. The changes do not result in any event previously deemed incredible being made credible. The changes do not result in more adverse conditions or result in any increase in the challenges to safety systems. Therefore, operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments does not result in a significant reduction in a margin of safety.

The containment functions to mitigate the effects of accidents. There are no new or significant changes to the initial conditions contributing to accident severity or consequences. The proposed modification will not otherwise affect the plant protective boundaries, will not cause a release of fission products to the public, nor will it degrade the performance of any other SSCs important to safety. Reducing the maximum allowed containment pressure limit is conservative in that it reduces the peak containment pressure that could result in the event of an accident. Therefore, reducing the maximum allowed containment pressure limit will not reduce the margin of safety. The added conservatism provides improvement to the design pressure margin resulting from the proposed change and will enhance protection against conditions resulting from a design basis accident, which will therefore provide a net benefit to radiological health and reactor safety.

Conclusion

Operation of the Point Beach Nuclear Plant in accordance with the proposed amendments will not result in a significant increase in the probability or consequences of any accident previously analyzed; will not result in a new or different kind of accident from any accident previously analyzed; and, does not result in a significant reduction in any margin of safety. Therefore, operation of PBNP in accordance with the proposed amendments does not result in a significant hazards determination.

5.2 Commitments

There are no actions committed to by NMC in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments.

6.0 ENVIRONMENTAL EVALUATION

NMC has determined that the information for the proposed amendments does not involve a significant hazards consideration, authorize a significant change in the types or total amounts of effluent release, or result in any significant increase in individual or cumulative occupational radiation exposure. Therefore, we conclude that the proposed amendments meet the categorical exclusion requirements of 10 CFR 51.22(c)(9) and that an environmental impact appraisal need not be prepared.

7.0 REFERENCES

1. FSAR Section 14.2.5, "Rupture of a Steam Pipe", 06/2001.
2. Ohkawa, D.K., "Wisconsin Electric Power Company Point Beach Nuclear Plant, Units 1 and 2 Steam line Break and Containment Integrity Analysis", WCAP-15153 (Proprietary), December 1998.
3. Condition Report 01-2026, "Containment Design Pressure Issue", 6/7/01.
4. Burnett, T.W.T., et. al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary) and WECAP-7907-A (Non-Proprietary), April 1984.
5. Moody, F. J., "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Journal of Heat Transfer, 87, 134 (1965).
6. Land, R. E. , "Mass and Energy Releases Following a Steam Line Rupture," WCAP-8822 (Proprietary), WCAP-8860 (Non-Proprietary), September 1976.
7. Land, R. E., "TRANFLO Steam Generator Code Description," WCAP-8821-P-A (Proprietary), WCAP-8859-A (Non-Proprietary), September 1976, approved version June 2001.
8. Butler, J. C., "Mass and Energy Releases Following a Steam Line Rupture, Supplement 2 – Impact of Steam Superheat in Mass/Energy Releases Following a Steam line Rupture for Dry and Subatmospheric Containment Designs," WCAP-8822-S2-P-A (Proprietary), WCAP-8860-S2-A (Non-Proprietary), September 1986.
9. "Containment Pressure Analysis Code (COCO)," WCAP-8327 (Proprietary), WCAP-8326 (Non-Proprietary), July 1974.
10. Swigart, S., "Containment Response to Steam line Break at 1524.5 MWt NSSS Power - Final Report", WEP-01-060, October 29, 2001

PROPOSED TECHNICAL SPECIFICATION CHANGES

(additions are double-underlined; deletions are strikethrough)

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -2.0 psig and ≤ +~~3~~2.0 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment pressure not within limits.	A.1 Restore containment pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify containment pressure is within limits.	12 hours

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES

(additions are double-underlined; deletions are strikethrough

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside the upper containment pressure limit coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB. The LOCA and SLB containment integrity evaluations ~~is~~ are accomplished by use of the digital computer code, COCO. ~~The SLB containment pressure calculation is a parameter by parameter comparison of a reference 2-loop plant to Point Beach. Each parameter is evaluated to determine if the Point Beach value is conservative, non-conservative or nominal. The effects of non-conservative parameters are quantified using a conservative heat balance to determine how much they increase peak containment pressure. Non-conservative parameters quantified in the calculation include additional FW and AFW, higher initial containment pressure, longer fan cooler delay time and lower fan cooler heat removal rates. The effect of one conservative parameter, containment heat sink surface area, is also quantified to determine how much it decreases peak containment pressure. Quantified increases and decreases are added to and subtracted from the most limiting result from the reference 2-loop plant analysis. Another conservative parameter is the trip reactivity worth for PBNP. The excess trip reactivity worth is used to show that there is no return to criticality during a steam line break. Avoiding a return to criticality can significantly reduce the mass and energy release rate to containment. The calculation uses the fact that there is no return to criticality to eliminate the need to evaluate many parameters that affect reactivity and the amount of energy created by a~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~return to criticality. By comparing and quantifying the effects of the conservative and non-conservative parameters, it is shown that the peak containment pressure resulting from a SLB is 51.3 psig.~~

The initial pressure condition used in the containment LOCA analysis was 14.7 psia (0.0 psig). This resulted in a maximum peak pressure from a LOCA of between 52 and 53 psig. ~~The containment analysis (Ref. 1) shows that the maximum peak calculated containment pressure, P_a, results from the limiting LOCA. The maximum containment pressure resulting from the worst case LOCA, between 52 and 53 psig,~~ The initial pressure condition used in the SLB containment analysis was 16.7 psia (2.0 psig). This resulted in a maximum peak pressure from the limiting SLB inside containment of 59.8 psig. The limiting SLB case assumed the failure of a feedwater regulating valve at 102% of rated thermal power. The SLB containment analysis shows that the maximum peak calculated containment pressure results from this limiting SLB case. The limiting SLB case does not exceed the containment design pressure, of 60 psig.

The containment was also designed for an external pressure load equivalent to -2.0 psig. This limit is sufficient to accommodate increases in atmospheric pressure and decreases in containment temperature after the establishment of containment integrity without the use of the containment purge valves.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 3).

Containment pressure satisfies Criterion 2 of the NRC Policy Statement.

BASES

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. The ~~3~~2.0 psig positive containment pressure limit was chosen based upon ~~engineering judgment~~ analysis. ~~The containment pressure analysis assumes an initial containment pressure of 14.7 psia (0.0 psig) resulting in a worst case DBA containment pressure of between 52 and 53 psig.~~ Measurement uncertainties for the containment pressure are not included in the 2.0 psig limit. A ~~3~~2.0 psig positive pressure limit is sufficiently low to prevent exceeding the containment design pressure (60 psig) in the event of a DBA, while allowing the operational flexibility to accommodate containment pressure increases resulting from evolutions such as plant heat ups and atmospheric pressure changes, in addition to instrument air leakage and operation of air operated valves. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure that containment integrity is maintained, the LCO is applicable in MODES 1, 2, 3 and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the limits established to ensure that containment design pressures are not exceeded. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that unit operation remains within the limits established to ensure that containment design pressures are not exceeded. The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room,

REFERENCES

1. FSAR, Section 14.
2. FSAR, Section 5.5.2.
3. 10 CFR 50, Appendix K

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during unit operations. The total amount of energy to be removed from containment by the structural heat sinks and Containment Spray and Cooling systems during post accident conditions is dependent upon the energy released to the containment due to the event, as well as the initial containment temperature and pressure. Higher initial containment temperatures result in higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1).

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB. The DBA LOCA and SLB are ~~is~~ analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. ~~The SLB containment pressure calculation is a parameter by parameter comparison of a reference 2-loop plant to Point Beach. Each parameter is evaluated to determine if the Point Beach value is conservative, non-conservative or nominal. The mass and energy release from a SLB is less than that calculated for a LOCA; therefore, the containment pressure and temperature analysis for the LOCA bounds the SLB event.~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs LOCA is ~~are~~ analyzed with regard to Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train each of the Containment Spray System, Residual Heat Removal System, and Containment Cooling System being rendered inoperable. The postulated DBA SLB was similarly analyzed, except that both trains of the Containment Spray System and the Containment Cooling System are assumed operable. This is acceptable since the DBA SLB analysis assumed a single failure of the feedwater regulating valve as the worst case single failure for the containment integrity analysis.

The limiting DBA for the maximum peak containment air temperature is a LOCA. The initial containment average air temperature assumed in the design basis analyses (Ref. 1) is 120°F. This resulted in a maximum containment air temperature of 291°F. The design temperature is 286°F.

The temperature limit is used to establish the environmental qualification operating envelope for containment. The maximum peak containment air temperature was calculated to exceed the containment design temperature for only a few seconds during the transient. The basis of the containment design temperature, however, is to ensure the performance of safety related equipment inside containment (Ref. 2). Thermal analyses showed that the time interval during which the containment air temperature exceeded the containment design temperature was short enough that the equipment surface temperatures remained below the design temperature. Therefore, it is concluded that the calculated transient containment air temperature is acceptable for the DBA LOCA ~~SLB~~.

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is a SLB ~~LOCA~~. The temperature limit is used in this analysis to ensure that in the event of an accident the maximum containment internal pressure will not be exceeded.

Containment average air temperature satisfies Criterion 2 of the NRC Policy Statement.

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the containment design temperature. As a result, the ability of containment to perform its design function is ensured.

REVISED TECHNICAL SPECIFICATION CHANGES

(incorporating proposed changes)

3.6 CONTAINMENT SYSTEMS

3.6.4 Containment Pressure

LCO 3.6.4 Containment pressure shall be ≥ -2.0 psig and ≤ +2.0 psig.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Containment pressure not within limits.	A.1	Restore containment pressure to within limits.	1 hour
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.4.1	Verify containment pressure is within limits.	12 hours

B 3.6 CONTAINMENT SYSTEMS

B 3.6.4 Containment Pressure

BASES

BACKGROUND

The containment pressure is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB). These limits also prevent the containment pressure from exceeding the containment design negative pressure differential with respect to the outside atmosphere.

Containment pressure is a process variable that is monitored and controlled. The containment pressure limits are derived from the input conditions used in the containment functional analyses and the containment structure external pressure analysis. Should operation occur outside the upper containment pressure limit coincident with a Design Basis Accident (DBA), post accident containment pressures could exceed calculated values.

APPLICABLE SAFETY ANALYSES

Containment internal pressure is an initial condition used in the DBA analyses to establish the maximum peak containment internal pressure. The limiting DBAs considered, relative to containment pressure, are the LOCA and SLB. The LOCA and SLB containment integrity evaluations are accomplished by use of the digital computer code, COCO.

The initial pressure condition used in the containment LOCA analysis was 14.7 psia (0.0 psig). This resulted in a maximum peak pressure from a LOCA of between 52 and 53 psig. The initial pressure condition used in the SLB containment analysis was 16.7 psia (2.0 psig). This resulted in a maximum peak pressure from the limiting SLB inside containment of 59.8 psig. The limiting SLB case assumed the failure of a feedwater regulating valve at 102% of rated thermal power. The SLB containment analysis shows that the maximum peak calculated containment pressure results from this limiting SLB case. The limiting SLB case does not exceed the containment design pressure of 60 psig.

The containment was also designed for an external pressure load equivalent to -2.0 psig. This limit is sufficient to accommodate increases in atmospheric pressure and decreases in containment temperature after the establishment of containment integrity without the use of the containment purge valves.

BASES

APPLICABLE SAFETY ANALYSES (continued)

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative. In particular, the cooling effectiveness of the Emergency Core Cooling System during the core reflood phase of a LOCA analysis increases with increasing containment backpressure. Therefore, for the reflood phase, the containment backpressure is calculated in a manner designed to conservatively minimize, rather than maximize, the containment pressure response in accordance with 10 CFR 50, Appendix K (Ref. 3).

Containment pressure satisfies Criterion 2 of the NRC Policy Statement.

LCO

Maintaining containment pressure at less than or equal to the LCO upper pressure limit ensures that, in the event of a DBA, the resultant peak containment accident pressure will remain below the containment design pressure. The 2.0 psig positive containment pressure limit was chosen based upon analysis. Measurement uncertainties for the containment pressure are not included in the 2.0 psig limit. A 2.0 psig positive pressure limit is sufficiently low to prevent exceeding the containment design pressure (60 psig) in the event of a DBA, while allowing the operational flexibility to accommodate containment pressure increases resulting from evolutions such as plant heat ups and atmospheric pressure changes, in addition to instrument air leakage and operation of air operated valves. Maintaining containment pressure at greater than or equal to the LCO lower pressure limit ensures that the containment will not exceed the design negative differential pressure.

APPLICABILITY

In MODES 1, 2, 3, and 4, a DBA could cause a release of radioactive material to containment. Since maintaining containment pressure within limits is essential to ensure that containment integrity is maintained, the LCO is applicable in MODES 1, 2, 3 and 4.

In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining containment pressure within the limits of the LCO is not required in MODE 5 or 6.

BASES

ACTIONS

A.1

When containment pressure is not within the limits of the LCO, it must be restored to within these limits within 1 hour. The Required Action is necessary to return operation to within the limits established to ensure that containment design pressures are not exceeded. The 1 hour Completion Time is consistent with the ACTIONS of LCO 3.6.1, "Containment," which requires that containment be restored to OPERABLE status within 1 hour.

B.1 and B.2

If containment pressure cannot be restored to within limits within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.4.1

Verifying that containment pressure is within limits ensures that unit operation remains within the limits established to ensure that containment design pressures are not exceeded. The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room,

REFERENCES

1. FSAR, Section 14.
2. FSAR, Section 5.5.2.
3. 10 CFR 50, Appendix K

B 3.6 CONTAINMENT SYSTEMS

B 3.6.5 Containment Air Temperature

BASES

BACKGROUND

The containment structure serves to contain radioactive material that may be released from the reactor core following a Design Basis Accident (DBA). The containment average air temperature is limited during normal operation to preserve the initial conditions assumed in the accident analyses for a loss of coolant accident (LOCA) or steam line break (SLB).

The containment average air temperature limit is derived from the input conditions used in the containment functional analyses and the containment structure external pressure analyses. This LCO ensures that initial conditions assumed in the analysis of containment response to a DBA are not violated during unit operations. The total amount of energy to be removed from containment by the structural heat sinks and Containment Spray and Cooling systems during post accident conditions is dependent upon the energy released to the containment due to the event, as well as the initial containment temperature and pressure. Higher initial containment temperatures result in higher peak containment pressure and temperature. Exceeding containment design pressure may result in leakage greater than that assumed in the accident analysis. Operation with containment temperature in excess of the LCO limit violates an initial condition assumed in the accident analysis.

APPLICABLE SAFETY ANALYSES

Containment average air temperature is an initial condition used in the DBA analyses that establishes the containment environmental qualification operating envelope for both pressure and temperature. The limit for containment average air temperature ensures that operation is maintained within the assumptions used in the DBA analyses for containment (Ref. 1).

The limiting DBAs considered relative to containment OPERABILITY are the LOCA and SLB. The DBA LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients.

No two DBAs are assumed to occur simultaneously or consecutively. The postulated DBA LOCA is analyzed with regard to Engineered Safety Feature (ESF) systems, assuming the loss of one ESF bus, which is the worst case single active failure, resulting in one train each of the Containment Spray System, Residual Heat Removal System,

BASES

APPLICABLE SAFETY ANALYSES (continued)

and Containment Cooling System being rendered inoperable. The postulated DBA SLB was similarly analyzed, except that both trains of the Containment Spray System and the Containment Cooling System are assumed operable. This is acceptable since the DBA SLB analysis assumed a single failure of the feedwater regulating valve as the worst case single failure for the containment integrity analysis.

The limiting DBA for the maximum peak containment air temperature is a LOCA. The initial containment average air temperature assumed in the design basis analyses (Ref. 1) is 120°F. This resulted in a maximum containment air temperature of 291°F. The design temperature is 286°F.

The temperature limit is used to establish the environmental qualification operating envelope for containment. The maximum peak containment air temperature was calculated to exceed the containment design temperature for only a few seconds during the transient. The basis of the containment design temperature, however, is to ensure the performance of safety related equipment inside containment (Ref. 2). Thermal analyses showed that the time interval during which the containment air temperature exceeded the containment design temperature was short enough that the equipment surface temperatures remained below the design temperature. Therefore, it is concluded that the calculated transient containment air temperature is acceptable for the DBA LOCA.

The containment pressure transient is sensitive to the initial air mass in containment and, therefore, to the initial containment air temperature. The limiting DBA for establishing the maximum peak containment internal pressure is a SLB. The temperature limit is used in this analysis to ensure that in the event of an accident the maximum containment internal pressure will not be exceeded.

Containment average air temperature satisfies Criterion 2 of the NRC Policy Statement.

LCO

During a DBA, with an initial containment average air temperature less than or equal to the LCO temperature limit, the resultant peak accident temperature is maintained below the containment design temperature. As a result, the ability of containment to perform its design function is ensured.