

Docket No. 50-423
B13429

Attachment 2
Millstone Nuclear Power Station, Unit No. 3
Proposed Revision to Technical Specifications
Containment Pressure

February 1990

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

- a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves or operator action during periods when containment isolation valves are opened under administrative control,** and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions.
- b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3; and
- c. After each closing of each penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at a pressure not less than P_a , 53.27 psia (38.57 psig), and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than $0.60 L_a$.

* Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

** The following manual valves may be opened on an intermittent basis under administrative control. 3FPW-V661, 3FPW-666, 3SSP-V13, 3SSP-V14, 3HCS-V2, 3HCS-V3, 3HCS-V9, 3HCS-V10, 3HCS-V6, 3HCS-V13, 3SAS-V875, 3SAS-V50, 3CHS-V371, 3CCP-V886, 3CCP-V887, 3CVS-V13.

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

- a. An overall integrated leakage rate of less than or equal to L_a , 0.65% by weight of the containment air per 24 hours at P_a , 53.27 psia (38.57 psig);
- b. A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to P_a ; and
- c. A combined leakage rate of less than or equal to 0.042 L_a for all penetrations identified in Table 3.6-1 as Enclosure Building bypass leakage paths when pressurized to P_a .

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

ACTION:

With the measured overall integrated containment leakage rate exceeding 0.75 L_a , or the measured combined leakage rate for all penetrations and valves subject to Type B and C tests exceeding 0.60 L_a , or the combined bypass leakage rate exceeding 0.042 L_a , restore the overall integrated leakage rate to less than 0.75 L_a , the combined leakage rate for all penetrations subject to Type B and C tests to less than 0.60 L_a , and the combined bypass leakage rate to less than 0.042 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using methods and provisions of ANSI N45.4-1972 (Total Time Method) and/or ANSI/ANS 56.8-1981 (Mass Point Method):

- a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at a pressure not less than P_a , 53.27 psia (38.57 psig) during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection;
- b. If any periodic Type A test fails to meet 0.75 L_a , the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a , a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a at which time the above test schedule may be resumed;

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- c. The accuracy of each type A test shall be verified by a supplemental test which:
- 1) Confirms the accuracy of the test by verifying that the supplemental test results, L_s , minus the sum of the Type A and the superimposed leak, L_o , is equal to or less than $0.25 L_a$;
 - 2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test; and
 - 3) Requires that the rate at which gas is injected into the containment or bled from the containment during the supplemental test is between $0.75 L_a$ and $1.25 L_a$.
- d. Type B and C tests shall be conducted with gas at P_a , 53.27 psia (38.57 psig), at intervals no greater than 24 months except for tests involving:
- 1) Air locks
- e. The combined bypass leakage rate shall be determined to be less than or equal to $0.042 L_a$ by applicable Type B and C tests at least once per 24 months except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a , 53.27 psig (38.57 psig), during each Type A test;
- f. Air locks shall be tested and demonstrated OPERABLE by the requirements of Specification 4.6.1.3;
- g. Purge supply and exhaust isolation valves shall be demonstrated OPERABLE by the requirements of Specifications 4.6.3.2.c and 4.9.9.
- h. The provisions of Specification 4.0.2 are not applicable.

TABLE 3.6-1

ENCLOSURE BUILDING BYPASS LEAKAGE PATHS

<u>PENETRATION</u>	<u>DESCRIPTION</u>	<u>RELEASE LOCATION</u>
14	N ₂ to Safety Injection Tanks	Ground Release
15	Primary Water to Pressurizer Relief Tanks	Ground Release
35	Vacuum Pump Suction	Plant Vent
36	Vacuum Pump Suction	Plant Vent
37	Air Ejector Suction	Plant Vent
38	Chilled Water Supply	Plant Vent
45	Chilled Water Return	Plant Vent
52	Service Air	Turbine Building Roof Exhaust
54	Instrument Air	Turbine Building Roof Exhaust
56	Fire Protection	Ground Release
59	Fuel Pool Purification	Ground Release
60	Fuel Pool Purification	Ground Release
70	Demineralized Water	Ground Release
72	Chilled Water Supply	Plant Vent
85	Containment Purge	Ground Release
86	Containment Purge	Plant Vent
116	Chilled Water Return	Plant Vent
124	Nitrogen to Containment	Plant Vent

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 The containment air lock shall be OPERABLE with:

- a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and
- b. An overall air lock leakage rate of less than or equal to $0.05 L_a$ at P_a , 53.27 psia (38.57 psig).

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

ACTION:

- a. With one containment air lock door inoperable:
 1. Maintain at least the OPERABLE air lock door closed* and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed,
 2. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days,
 3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, and
 4. The provisions of Specification 3.0.4 are not applicable.
- b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Except during entry to repair an inoperable inner door, for a cumulative time not to exceed 1 hour per year.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

- a. 1) Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying no detectable seal leakage by pressure decay when the volume between the door seals is pressurized to greater than or equal to P_a , 53.27 psia (38.57 psig), for at least 15 minutes;

or
 - 2) Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying that the seal leakage is less than 0.01 L_a as determined by precision flow measurements when measured for at least 30 seconds with the volume between the seals at a constant pressure of greater than or equal to P_a , 53.27 psia (38.57 psig);

or
 - 3) Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by completing an overall air lock leakage test per 4.6.1.3.b.
- b. By conducting overall air lock leakage tests at not less than P_a , 53.27 psia (38.57 psig), and verifying the overall air lock leakage rate is within its limit:
- 1) At least once per 6 months,* and
 - 2) Prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.**
- c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

*The provisions of Specification 4.0.2 are not applicable.

**This represents an exemption to Appendix J, paragraph III.D.2.(b)(i), of 10 CFR Part 50.

CONTAINMENT SYSTEMS

CONTAINMENT PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment pressure shall be maintained greater than or equal to 10.6 psia and less than or equal to 14.0 psia.

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

ACTION:

With the containment pressure less than 10.6 psia or greater than 14.0 psia, restore the containment pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment pressure shall be determined to be within the limits at least once per 12 hours.

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3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guidelines of 10 CFR Part 100 during accident conditions and the control room operators dose to within the guidelines of GDC 19.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a . As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to $0.75 L_a$ during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 and 3/4.6.1.5 AIR PRESSURE and AIR TEMPERATURE

The limitations on containment pressure and average air temperature ensure that: (1) the containment structure is prevented from exceeding its design negative pressure of 8 psia, and (2) the containment peak pressure does not exceed the design pressure of 60 psia during LOCA conditions. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature. The limits on the pressure and average air temperature are consistent with the assumptions of the safety analysis. The minimum total containment pressure of 10.6 psia is determined by summing the minimum permissible air partial pressure of 8.9 psia and the maximum expected vapor pressure of 1.7 psia (occurring at the maximum permissible containment initial temperature of 120°F).

CONTAINMENT SYSTEMS

BASES

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 60 psia in the event of a LOCA. A visual inspection in conjunction with the Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 42-inch containment purge supply and exhaust isolation valves are required to be locked closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the Containment Purge System. To provide assurance that these containment valves cannot be inadvertently opened, the valves are locked closed in accordance with Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevents power from being supplied to the valve operator.

The Type C testing frequency required by 4.6.1.2d is acceptable, provided that the resilient seats of these valves are replaced every other refueling outage.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 and 3/4.6.2.2 CONTAINMENT QUENCH SPRAY SYSTEM and RECIRCULATION SPRAY SYSTEM

The OPERABILITY of the Containment Spray Systems ensures that containment depressurization and iodine removal will occur in the event of a LOCA. The pressure reduction, iodine removal capabilities and resultant containment leakage are consistent with the assumptions used in the safety analyses.

3/4.6.2.3 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 7.0 and 7.35 for the solution recirculated within containment after a LOCA. This pH band minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

Docket No. 50-423
B13429

Attachment 3

Millstone Nuclear Power Station, Unit No. 3

Description of the Proposed Technical Specification
Changes and Significant Hazards Considerations Discussion

February 1990

Millstone Nuclear Power Station, Unit No. 3
Description of the Proposed Technical Specification
Changes and Significant Hazards Consideration Discussion

Description of the Proposed Technical Specification Changes

Technical Specification 3/4.6 (Containment Systems) and associated bases are being changed to allow the containment pressure to increase to 14.0 psia during Modes 1 through 4. The purpose of the containment pressure increase is to reduce the potential for personnel injury when entering containment due to crossing the pressure boundary and due to oxygen deficiency. The proposed containment pressure change is based on the results of a recent containment analysis performed by Stone and Webster under the direction of Northeast Nuclear Energy Company (NNECO).

The proposed changes to Technical Specification 3/4.6 affects the following:

1. The peak calculated containment pressure (P_a) is changed to 53.27 psia (38.57 psig) in Sections 4.6.1.1.c, 3.6.1.2.a, 4.6.1.2.a, 4.6.1.2.d, 4.6.1.2.e, 3.6.1.3.b, 4.6.1.3.a.1 and a.2, 4.6.1.3.b. This is based on the results of a revised containment analysis.
2. The integrated leak rate at P_a , containment leak rate (L_a) is changed from 0.9 weight percent per day to 0.65 weight percent per day in Section 3.6.1.2.a.
3. The combined bypass leakage rate is changed from $0.01 L_a$ to $0.042 L_a$ in Sections 3.6.1.2 ACTION and 4.6.1.2.e.
4. The operating containment pressure of 14.0 psia is specified in Section 3.6.1.4. In addition, the maximum and minimum limit for the containment pressure is specified as total containment pressure instead of air partial pressure.
5. Figure 3.6.1 is deleted as the containment pressure will be read directly from the main control board indicators.
6. Bases for Sections 3/4.6.1.1, 3/4.6.1.4, 3/4.6.1.5, 3/4.6.2.1, and 3/4.6.2.2 are revised to reflect the above changes.
7. Index of Technical Specifications has been revised to reflect the above changes.

Containment bypass penetrations are lines that come out of the primary containment and run through the enclosure building to areas outside the plant. Leakage through the containment isolation valves (CIV) in these penetrations could bypass the secondary containment afforded by the enclosure building and

go into the environment during a design basis accident (DBA). The following changes to Table 3.6-1 represent the results of refinements in previous analyses which identified bypass penetrations. It will improve containment integrity by deleting testing of penetrations that are not potential bypass paths and refocusing this testing on penetrations that really do have the potential for being bypass leakage paths.

The proposed changes also correct the bypass penetration listing of Technical Specification Table 3.6.1 as follows:

1. Penetrations Z-28 and Z-29 (aerated drains and gaseous vents) are being deleted.
2. Penetrations Z-59, Z-60, and Z-124 (fuel pool purification and nitrogen supply to containment) are being added.
3. Table 3.6.1 has been revised to include description for each penetration.

Significant Hazards Consideration

In accordance with 10CFR50.92, NNECO has reviewed the proposed Technical Specification changes and has concluded that they do not involve a significant hazards consideration. The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed changes do not involve a significant hazards consideration because the changes would not:

1. Involve a significant increase in the probability of occurrence or consequences of an accident previously analyzed.
 - a. The increase in containment pressure affects the following:
 - (1) The temperature and pressure in the containment due to a spectrum of postulated loss-of-coolant accidents (LOCA), control rod ejection accidents (CREA), and secondary system steam and feedwater line breaks.
 - (2) The external pressure to which the containment is subjected.
 - (3) The range and accuracy of instrumentation that is provided to monitor and record containment conditions during and following an accident.
 - (4) Containment heat removal system.
 - (5) Minimum containment pressure analysis for emergency core cooling system performance capability studies (LOCA).
 - (6) Subcompartment analysis.

- (7) Mass and energy release analysis for postulated LOCAs and secondary system pipe ruptures.
 - (8) Combustible gas concentration.
 - (9) Containment leakage testing.
 - (10) Determination of leakage paths.
- b. The increase in containment pressure impacts the consequences of the DBAs listed above as follows:
- (1) The pressures in the containment increase. The containment pressure/temperature response was evaluated (see Attachment 1) assuming a maximum operating pressure of 14.2 psia using the same methods and models described in Section 6.2.1 of the Final Safety Analysis Report (FSAR). The maximum peak containment pressure was recalculated to be 38.57 psig, which shows an increase from the current containment pressure peak of 36.09 psig (References 1 and 2). (Note: The current Technical Specification does not reflect the current analysis.) The containment long-term depressurization transient was also recalculated (see Attachment 1). The containment pressure does not return to subatmospheric pressure, and leakage is assumed to continue for 30 days (see Attachment 1). The current analysis discussed in References 1 and 2 shows the containment pressure returns to atmospheric pressure within 1 hour post-LOCA, at which time the containment leakage paths are assumed to stop. To help compensate for the increased release of radioactivity, the allowable Technical Specification leak rate, L , is being reduced from 0.9 percent per day to 0.65 percent per day. In spite of the reduction in allowable leak rate, however, some of the calculated dose consequences of the LOCA and CREA increase (see Attachment 1). The new P_c of 38.57 psig is well below the design containment pressure of 45 psig. The containment pressure reduces to less than 50 percent of the peak containment pressure within 24 hours after the postulated accident (Standard Review Plan 6.2.1.A). The calculated radiation doses for the exclusion area boundary (EAB), low-population zone (LPZ), and operating personnel remain well within the 10CFR100 limits, the GDC 19 limits, and the Standard Review Plan acceptance criteria (see Attachment 1).
 - (2) The external pressure to which the containment is subjected following, for example, inadvertent operation of the containment heat removal system is unchanged or in some cases decreased.

- (3) The range and accuracy of instrumentation that is provided to monitor and record containment conditions during and following an accident is not changed. Currently transmitters 3LM3*PT934, 935, 936, and 937, which have a range 0 to 60 psia, are used to perform High 1, 2, and 3 containment isolation. The range of this transmitter is too large for operations to maintain containment pressure within the proposed Technical Specifications. The two narrow-range transmitters 3LMS*PT43A/43B (8.5 to 14.5 psia) that provide indication on the main control board will be utilized to set and maintain containment pressure to the proposed Technical Specifications. The total probable error of the reading during normal plant condition was determined to be $\pm .167$ psi when using the plant process computers. This error is incorporated in the proposed Technical Specifications. Use of other methods of reading pressure will be readjusted for total error. The electrical equipment qualification for 10CFR50.49 is not impacted by the increase in containment pressure (see Attachment 1). For normal environment conditions, the EEQ program is based on a normal containment pressure range of 9.5 to 14.7 psia, which bounds the proposed containment pressure of 14.0 psia. For accident environment conditions, the EEQ program is based on the pressure and temperature envelope (Millstone Unit No. 3 FSAR Section 3.11), which bounds the calculated new pressures and temperatures indicated in Attachment 1. For post-DBA environment conditions from 1 hour to 1 year, the containment pressure value of 1.75 psig, although not bounded by the existing envelope included in the Millstone Unit No. 3 FSAR Section 3.11, has no impact on the EEQ qualification because the pressure is not an aging parameter which causes degradation of material. The proposed change will not impact the existing accident radiation qualification of EEQ equipment. Although the proposed increase in containment pressure results in some increase in the radiation consequences following a DBA (Attachment 1), the equipment qualification remains valid with adequate margin.
- (4) The proposed change has no effect on the containment heat removal system effectiveness. The containment heat removal systems have been shown to be capable of reducing rapidly the containment pressure and temperature following a LOCA (Attachment 1).
- (5) The results of the minimum containment pressure analysis are favorably impacted since higher containment pressures yield higher core flooding rates during LOCA and subsequent lower fuel peak cladding temperatures (Standard Review Plan 6.2.1.5).

- (6) The results of subcompartment analysis are favorably impacted since higher containment pressures minimizes the resultant differential pressure (Standard Review Plan 6.2.1.2).
- (7) The FSAR LOCA long-term mass and energy analysis, Section 6.2, will remain valid. The release rate of the flow into the containment will change due to the initial containment pressure, but not the total effluent available.
- (8) The proposed change has no effect on the current evaluation of hydrogen generation and control (see Attachment 1). An increase in the containment operating pressure causes an increase in the mass of air in the containment. Because the rate of generation of hydrogen is unchanged, the concentration of hydrogen is lower.
- (9) The containment leak testing will reflect tighter containment leakage limits due to the increase in operating containment pressure. The containment leak rate (L_c) will be reduced from 0.9 percent per day to 0.65 percent per day. The secondary containment bypass leakage from the containment will be increased from 0.01 L_c to 0.042 L_c . With the reduction in L_c , this is actually an increase from 0.009 percent per day to 0.028 percent per day. As explained earlier in this section, one of the consequences of this change is that the containment pressure does not return to subatmospheric pressure following a LOCA (see Attachment 1). The current analysis on the subatmospheric design of Millstone Unit No. 3, however shows that all containment leakage terminates within 1 hour. The Technical Specification leak rate, L_c is being reduced from 0.9 percent per day to 0.65 percent per day to compensate for the increased time in leakage release. The proposed changes in containment leakage meet the requirements of 10CFR50, Appendix J. However, it requires administrative revision of containment Type B and C leak testing procedures.
- (10) Currently, any preexisting bypass leak in the Millstone Unit No. 3 containment resulting from human error, such as valves left inadvertently open, can be detected shortly upon initial isolation. The proposed Technical Specification change to increase the operating containment pressure could cause a leakage path to go undetected for a longer period of time. It has been concluded that a 3/4-inch line is the smallest line that could be bypassed. NNECO has determined that at the maximum containment pressure of 14.2 psia, it could take 6.41 hours to detect a .1 psi change without the containment vacuum pumps operating. Since the instrument error on containment pressure measurement is .167 psi, the time to detect a .267 psi change would be about 17.12 hours. Currently,

Millstone Unit No. 3 has a 10^{-4} chance of early containment bypass. It has been estimated that when a leak from a 3/4-inch line is being left open and assumed to go undetected for 5 days, the probability of early containment bypass remains in the order of 10^{-4} .

In summary, the increase in normal operating containment pressure may increase the duration of containment radiation leakage following a LOCA. In spite of a proposed reduction in allowable leak rate, some of the calculated doses following LOCA and CREA increase. The 10CFR100 limits, GDC 19 limits, and Standard Review Plan acceptance criteria, however, are still satisfied.

- c. The changes in Table 3.6.1 are limited to changes in the designations of containment bypass penetration. Bypass penetrations are piping systems that come out of the primary containment and run through the enclosure building to areas outside of the plant. Leakage through the containment valves in these penetrations after a DBA could bypass the secondary containment afforded by the enclosure building. However, a change to the bypass penetration listing does not constitute an increase in potential off-site consequences due to a DBA. The leakage limit is applied to all bypass CIVs regardless of their number (i.e., the total bypass leakage limit is shared by all the CIVs). In addition, the refinement of the listing of the penetrations that are potential bypass paths does not affect the probability of occurrence of a DBA.
2. Create the possibility of a new or different kind of accident from any previously analyzed. The proposed increase in normal operating pressure is within the existing design conditions of the equipment. The proposed changes would not impact the plant response to the point where a new accident is created. No new failure modes are introduced by these proposed Technical Specification changes that would allow the containment to remain at 14.0 psia during Modes 1 through 4 and that would refine the listing of bypass penetrations.
 3. Involve a significant reduction in a margin of safety.
 - a. The proposed increase in operating containment pressure to 14.0 psia does not impact the safety limits for the protective boundaries. The calculated P_a is well below the containment design pressure of 45 psig. The containment pressure reduces to less than 50 percent of the peak containment pressure within 24 hours after the postulated accident, thus satisfying the Standard Review Plan Section 6.2.1A. The calculated radiation dose in the EAB, LPZ, and for operating personnel remain well within 10CFR100 limits, the General Design Criterion 19 limits, and the Standard Review Plan acceptance criteria for the postulated LOCA and CREA (see Attachment 1). Since safety limits are not impacted, the margin of safety between the

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safety limits and protective boundary failure is not impacted. Therefore, the proposed changes do not result in a reduction of any safety margin.

- b. The proposed changes to Table 3.6.1 are the result of refinements in previous analyses which identified bypass penetrations. Since the proposed changes do not impact the safety limits, the proposed changes do not result in a reduction of any margin of safety.

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References

- (1) Millstone 3 Safety Evaluation Report, NUREG-1031, Supplement No. 5, January 1986.
- (2) Millstone Unit No. 3 FSAR, Amendment 17.

References

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- Reference 2 Letter, E. J. Mroczka (Northeast Utilities) to W. T. Russell (NRC Region I), B12863, "Report of Substantial Safety Hazard" (March 25, 1988).
- Reference 3 Letter, Rosemount, Inc. (Steve Wanck) to Northeast Utilities (Director of QA), "Notification Under 10 C.F.R. 21" (February 7, 1989).
- Reference 4 REF 89-10, Completed Reportability Evaluation for Millstone Unit 3 (February 22, 1989).
- Reference 5 REF 89-09, Completed Reportability Evaluation for Millstone Unit 2 (March 10, 1989).
- Reference 6 REF 89-08, Completed Reportability Evaluation for Millstone Unit 1 (March 14, 1989).
- Reference 7 Letter, E. J. Mroczka (Northeast Utilities) to U.S. Nuclear Regulatory Commission, B13178, "Rosemount Transmitters," (April 13, 1989).
- Reference 8 NRC Information Notice 89-42, "Failure of Rosemount Models 1153 and 1154 Transmitters," (April 21, 1989).
- Reference 9 NRC Inspection Report 50-423/89-04, dated June 28, 1969.
- Reference 10 GSP-89-299, "Operability Determination for Rosemount Pressure and Differential Pressure Transmitters at Millstone Unit 3" (July 31, 1989).
- Reference 11 Letter, E. J. Mroczka (Northeast Utilities) to U.S. Nuclear Regulatory Commission, A08132, "Response to Inspection 50-423/89-04" (August 1, 1989).
- Reference 12 Letter, E. J. Mroczka (Northeast Utilities) to U.S. Nuclear Regulatory Commission, B13366, "Rosemount Transmitters" (October 31, 1989).
- Reference 13 Letter, D. L. Fuller (Westinghouse) to A. R. Roby (Northeast Utilities), NEU-90-518, "Rosemount Transmitter Review" (February 15, 1990).

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