8506280050 850621
PDR ADOCK 05000293
P PDR

valve closure, are bypassed when reactor pressure is less than 600 psig; the low pressure main steam line isolation valve closure trip is bypassed, the reactor protection system is energized with IRM neutron monitoring system trips and control rod withdrawal interlocks in service.

2. Run Mode - In this mode the reactor system pressure is at or above 880 psig and the reactor protection system is energized with APRM protection and RBM interlocks in service.

3. Shutdown Mode - The reactor is in the shutdown mode when the reactor mode switch is in the shutdown mode position and no core alterations are being performed.

   a. Hot Shutdown means conditions as above with reactor coolant temperature greater than 212°F.

   b. Cold Shutdown means conditions as above with reactor coolant temperature equal to or less than 212°F.

4. Refuel Mode - The reactor is in the refuel mode when the mode switch is in the refuel mode position. When the mode switch is in the refuel position, the refueling interlocks are in service.

L. Design Power - Design power means a steady-state power level of 1998 thermal megawatts.

M. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

2. At least one door in each airlock is closed and sealed.

3. All blind flanges and manways are closed.

4. All automatic primary containment isolation valves are operable or at least one containment isolation valve in each line having an inoperable valve shall be deactivated in the isolated condition.

N. Secondary Containment Integrity - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1.0 DEFINITIONS (Cont'd)

U. Surveillance Frequency - Unless otherwise stated in these specifications, periodic surveillance tests, checks, calibrations, and examinations shall be performed, within the specified surveillance intervals. These intervals may be adjusted plus 25%. The total maximum combined interval time for any three consecutive tests shall not exceed 3.25 times the specified interval. The operating cycle interval is considered to be 18 months and the tolerances stated above are applicable.

V. Surveillance Interval - The surveillance interval is the calendar time between surveillance tests, checks, calibrations, and examinations to be performed upon an instrument or component when it is required to be operable. These tests may be waived when the instrument, component, or system is not required to be operable, but the instrument, component, or system shall be tested prior to being declared operable.

W. Fire Suppression Water System - A Fire Suppression Water System shall consist of: a water source(s); gravity tank(s) or pump(s); and distribution piping with associated sectionalizing control or isolation valves. Such valves shall include hydrant post indicator valves and the first valve ahead of the water flow alarm device on each sprinkler, hose standpipe or spray system riser.

X. Staggered Test Basis - A Staggered Test Basis shall consist of: (a) a test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals; (b) the testing of one system, subsystem, train or other designated components at the beginning of each subinterval.

Y. Automatic Primary Containment Isolation Valves - Are primary containment isolation valves which receive an automatic primary containment group isolation signal.

BASES:

3.2 In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operator's ability to control, or terminates operator errors before they result in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the core cooling systems, control rod block and standby gas treatment systems. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required by preserving its capability to tolerate a single failure of any component of such systems even during periods when portions of such systems are out of service for maintenance, and (ii) to prescribe the trip settings required to assure adequate performance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

Some of the settings on the instrumentation that initiate or control core and containment cooling have tolerances explicitly stated where the high and low values are both critical and may have a substantial effect on safety. The set points of other instrumentation, where only the high or low end of the setting has a direct bearing on safety, are chosen at a level away from the normal operating range to prevent inadvertent actuation of the safety system involved and exposure to abnormal situations.

Actuation of primary containment valves is initiated by protective instrumentation shown in Table 3.2.A which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement.

The low water level instrumentation set to trip at 129.5" above the top of the active fuel closes all isolation valves except those in Groups 1, 4 and 5. This trip setting is adequate to prevent core uncovery in the case of a break in the largest line assuming a 60 second valve closing time. Required closing times are less than this.

The low low reactor water level instrumentation is set to trip when reactor water level is 78.5" above the top of the active fuel (-49" on the instrument). This trip closes Main Steam Line Isolation

Valves, Main Steam Drain Valves, Recirc Sample Valves (Group 1) activates the CSCS subsystems, starts the emergency diesel generators and trips the recirculation pumps. This trip setting level was chosen to be high enough to prevent spurious actuation but low enough to initiate CSCS operation and primary system isolation so that no fuel damage will occur and so that post accident cooling can be accomplished and the guidelines of 10 CFR 100 will not be violated. For large breaks up to the complete circumferential break of a 28-inch recirculation line and with the trip setting given above, CSCS initiation and primary system isolation are initiated in time to meet the above criteria.

The high drywell pressure instrumentation is a diverse signal to the water level instrumentation and in addition to initiating CSCS, it causes isolation of Group 2 isolation valves. For the breaks discussed above, this instrumentation will initiate CSCS operation at about the same time as the low low water level instrumentation; thus the results given above are applicable here also. The low low water level instrumentation initiates protection for the full spectrum of loss-of-coolant accidents and causes isolation of Group 1 isolation valves.

Venturis are provided in the main steam lines as a means of measuring steam flow and also limiting the loss of mass inventory from the vessel during a steam line break accident. The primary function of the instrumentation is to detect a break in the main steam line. For the worst case accident, main steam line break outside the drywell, a trip setting of 140% of rated steam flow in conjunction with the flow limiters and main steam line valve closure, limits the mass inventory loss such that fuel is not uncovered, fuel temperatures remain approximately 1000°F and release of radioactivity to the environs is well below 10 CFR 100 guidelines.

Temperature monitoring instrumentation is provided in the main steam line tunnel and the turbine basement to detect leaks in these areas. Trips are provided on this instrumentation and when exceeded, cause closure of isolation valves. The setting of 170°F for the main steam line tunnel detector is low enough to detect leaks of the order of 5 to 10 gpm; thus, it is capable of covering the entire spectrum of breaks. For large breaks, the high steam flow instrumentation is a backup to the temperature instrumentation.

High radiation monitors in the main steam line tunnel have been provided to detect gross fuel failure as in the control rod drop acci-

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

Suppression Pool Water Volume and Temp.

1. At any time that the nuclear system is pressurized above atmospheric pressure or work is being done which has the potential to drain the vessel, the pressure suppression pool water volume and temperature shall be maintained within the following limits except as specified in 3.7.A.2 and 3.7.A.3.

   a. Minimum water volume - 84,000 ft^3

   b. Maximum water volume - 94,000 ft^3

   c. Maximum suppression pool bulk temperature during normal continuous power operation shall be $\leq$80°F, except as specified in 3.7.A.1.e.

   d. Maximum suppression pool bulk temperature during RCIC, HPCI or ADS operation shall be $\leq$90°F, except as specified in 3.7.A.1.e.

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment.

Specification:

A. Primary Containment

Suppression Pool Water Volume and Temp.

1. a. The suppression chamber water level and temperature shall be checked once per day.

   b. Whenever there is indication of relief valve operation or testing which adds heat to the suppression pool, the pool temperature shall be continually monitored and also observed and logged every 5 minutes until the heat addition is terminated.

   c. Whenever there is indication of relief valve operation with the bulk temperature of the suppression pool reaching 160°F or more and the primary coolant system pressure greater than 200 psig, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

   d. Whenever there is indication of relief valve operation with the local temperature of the suppression pool T-quencher reaching 200°F or more, an external visual examination of the suppression chamber shall be conducted before resuming power operation.

3.7 CONTAINMENT SYSTEMS (Con't)

e. In order to continue reactor power operation, the suppression chamber pool bulk temperature must be reduced to $\leq$80°F within 24 hours.

f. If the suppression pool bulk temperature exceeds the limits of Specification 3.7.A.1.d, RCIC, HPCI or ADS testing shall be terminated and suppression pool cooling shall be initiated.

g. If the suppression pool bulk temperature during reactor power operation exceeds 110°F, the reactor shall be scrammed.

h. During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 200 psig at normal cool down rates if the pool bulk temperature reaches 120°F.

i. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.17 psid, except as specified in j and k.

j. The differential pressure shall be established within 24 hours of placing the reactor in the run mode following a shutdown. The differential pressure may be reduced to less than 1.17 psid 24 hours prior to a scheduled shutdown.

k. The differential pressure may be reduced to less than 1.17 psid for a maximum of four (4) hours for maintenance activities on the differential pressure control system and during required operability testing of the HPCI system, the relief valves, the RCIC system and the drywell-suppression chamber vacuum breakers.

4.7 CONTAINMENT SYSTEMS (Con't)

e. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

f. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift when the differential pressure is required.

g. Suppression chamber water level shall be recorded at least once each shift when the differential pressure is required.

3.7 CONTAINMENT SYSTEMS (Con't)

l. If the specifications of Item i, above, cannot be met, and the differential pressure cannot be restored within the subsequent (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition in twenty-four (24) hours.

m. Suppression chamber water level shall be maintained between -6 to -3 inches on torus level instrument which corresponds to a downcomer submergence of 3.00 and 3.25 feet respectively.

n. The suppression chamber can be drained if the conditions as specified in Sections 3.5.F.3 and 3.5.F.5 of this Technical Specification are adhered to.

4.7 CONTAINMENT SYSTEMS (Cont'd)

3.7.A Primary Containment (Con't)

Primary Containment Integrity

2.a Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing "open vessel" physics test at power levels not to exceed 5 Mw(t).

Primary containment integrity means that the drywell and pressure suppression chamber are intact and that all of the following conditions are satisfied:

(1) All manual containment isolation valves on lines connected to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

(2) At least one door in each airlock is closed and sealed.

(3) All blind flanges and manways are closed.

(4) All automatic primary containment isolation valves and all instrument line flow check valves are operable except as specified in 3.7.A.2.b.

4.7.A Primary Containment (Con't)

Primary Containment Integrity

2.a The primary containment integrity shall be demonstrated by performing Primary Containment Leak-Tests in accordance with 10 CFR 50 Appendix J, as amended thru Sept. 22, 1980, with exemptions as approved by the NRC and exceptions as follows:

(1) The main steam line isolation valves shall be tested at a pressure ≥23 psig, and normalized to a value equivalent to 45 psig each operating cycle.

(2) Personnel air lock door seals shall be tested at a pressure ≥10 psig each operating cycle. Results shall be normalized to a value equivalent to 45 psig.

If the total leakage rates listed below are exceeded, repairs and retests shall be performed to correct the conditions.

(1) All double-gasketed seals: 10% L_t (x)

(2) All testable penetrations and isolation valves: 60% L_a (x)

(3) Any one penetration or isolation valve except main steam line isolation valves: 5% L_t (x)

(4) Any one main steam line isolation valve: 11.5 scf/hr @23 psig.

where x = 45 psig
L_t = .75 L_a
L_a = 1.0% by weight of the contained air @ 45 psig for 24 hrs.

3.7.A Primary Containment (Con't)

Primary Containment Isolation Valves

2.b. In the event any Automatic Primary Containment Isolation Valve becomes inoperable, at least one containment isolation valve in each line having an inoperable valve shall be deactivated in the isolated condition. (This requirement may be satisfied by deactivating the inoperable valve in the isolated condition. Deactivation means to electrically or pneumatically disarm, or otherwise secure the valve.)*

*Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under ORC approved administrative controls.

4.7.A Primary Containment (Con't)

Primary Containment Isolation Valves

2.b.1 The primary containment isolation valves surveillance shall be performed as follows:

a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for simulated automatic initiation and closure times.

b. At least once per quarter:

1. All normally open power operated isolation valves (except for the main steam line power operated isolation valves) shall be fully closed and reopened.

2. Trip the main steam isolation valves individually and verify closure time.

c. At least twice per week the main steam line power operated isolation valves shall be exercised by partial closure and subsequent reopening.

d. At least once per operating cycle the operability of the reactor coolant system instrument line flow check valves shall be verified.

2.b.2 Whenever an automatic primary containment isolation valve is inoperable, the position of the isolated valve in each line having an inoperable valve shall be recorded daily.

3.7.A Primary Containment (Con't)

4.7.A Primary Containment (Con't)

2.c Continuous Leak Rate Monitor

When the primary containment is inerted, the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements. This monitoring system may be taken out of service for maintenance but shall be returned to service as soon as practicable.

2.d Drywell Surfaces

The interior surfaces of the drywell and torus above the water line shall be visually inspected every refueling outage for evidence of deterioration.

3.7.A Primary Containment

5.b. Within the 24-hour period subsequent to placing the reactor in the Run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4% by volume and maintained in this condition. De-inerting may commence 24 hours prior to a shutdown.

6. If the specifications of 3.7.A.1 thru 3.7.A.5 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in Cold Shutdown condition within 24 hours.

4.7.A Primary Containment

3.7.A & 4.7.A Primary Containment

The integrity of the primary containment and operation of the core standby cooling system in combination limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception was made to this requirement during initial core loading and while the low power test program was being conducted and ready access to the reactor vessel was required. There was no pressure on the system at this time, thus greatly reducing the chances of a pipe break. Should this type of testing be necessary in the future, the reactor may be taken critical; however, restrictive operating procedures would be in effect again to minimize the probability of an accident. Procedures and the Rod Worth Minimizer would limit control worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the secondary containment and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep off-site doses well below 10 CFR 100 limits.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1035 psig. Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 45 psig which is below the maximum of 62 psig. Maximum water volume of 94,000 ft^3 results in a downcomer submergency of 4'-0" and the minimum volume of 84,000 ft^3 results in a submergence approximately 12-inches less. Mark I Containment Long Term Program Quarter Scale Test Facility (QSTF) testing at a downcomer submergency of 3.25 feet and 1.17 psi wetwell to drywell pressure differential shows a significant suppression chamber load reduction and Long Term Program analysis and modifications are based on the above submergence and Δ P.

Should it be necessary to drain the suppression chamber, provision will be made to maintain those requirements as described in Section 3.5.F BASES of this Technical Specification.

BASES:

3.7.A & 4.7.A Primary Containment

Experimental data indicates that excessive steam condensing loads can be avoided if the peak local temperature of the pressure suppression pool is maintained below 200°F during any period of relief-valve operation with sonic conditions at the discharge exit. Analysis has been performed to verify that the local pool temperature will stay below 200°F and the bulk temperature will stay below 160°F for all SRV transients. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high pressure suppression chamber loadings.

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

If a loss-of-coolant accident were to occur when the reactor water temperature is below approximately 330°F, the containment pressure will not exceed the 62psig code permissible pressure, even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperature above 212°F provides additional margin above that available at 330°F.

BASES:

3.7.A & 4.7.A Primary Containment

Primary Containment Testing

The primary containment pre-operational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The calculated peak drywell pressure is about 45 psig which would rapidly reduce to 27 psig following the pipe break. Following the pipe break, the suppression chamber pressure rises to 27 psig, equalizes with drywell pressure and therefore rapidly decays with the drywell pressure decay. The design pressure of the drywell and suppression chamber is 56 psig. The design leak rate is 0.5%/day at a pressure of 56 psig. Based on the calculated containment pressure response discussed above, the primary containment pre-operational test pressures were chosen. Also, based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than the individual components separately.

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.25%/day at 45 psig. Calculations made by the AEC staff with this leak rate and a standby gas treatment system filter efficiency of 95% for halogens and assuming the fission product release fractions stated in TID 14844, show that the maximum total whole body passing cloud dose is about 13 REM and the maximum total thyroid dose is about 110 REM at the site boundary over an exposure duration of two hours. The resultant doses that would occur for the duration of the accident at the low population zone distance of 4.3 miles are about 3 REM total whole body and 70 REM total thyroid. Thus, the doses reported are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected off-site dose and 10 CFR 100 guidelines.

The maximum allowable test leak rate is 1.0%/day at a pressure of 45 psig. This value for the test condition was derived from the maximum allowable accident leak rate of 1.25%/day when corrected for the effects of containment environment under accident and test conditions. In the accident case, the containment atmosphere initially would be composed of steam and hot air whereas under test conditions the test medium would be air at ambient conditions. Considering the differences in mixture composition and temperatures, the appropriate correction factor applied was 0.8 as determined from the guide on containment testing.

Establishing the test limit of 1.0%/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak-tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as built" condition is achieved by establishing the allowable operational leak rate. The allowable operational leak rate is derived by multiplying the maximum allowable leak rate or the allowable test leak rate by 0.75 thereby providing a 25% margin to allow for leakage deterioration which may occur during the period between leak rate tests.

BASES:

3.7.A & 4.7.A Primary Containment

The primary containment leak rate test frequency is based on maintaining adequate assurance that the leak rate remains within the specification. The leak rate test frequency is in accordance with 10 CFR 50 App. J as amended through Sept. 22, 1980.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a bolted double-gasketed penetration is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized. The personnel air lock is tested at 10 psig, because the inboard door is not designed to shut in the opposite direction.

Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss of coolant accident.

Group 1 - process lines are isolated by reactor vessel low-low water level in order to allow for removal of decay heat subsequent to a scram, yet isolate in time for proper operation of the core standby cooling systems. The valves in group 1 are also closed when process instrumentation detects excessive main steam line flow, high radiation,, low pressure, main steam space high temperature, or reactor vessel high water level.

Group 2 - isolation valves are closed by reactor vessel low water level or high drywell pressure. The group 2 isolation signal also "isolates" the reactor building and starts the standby gas treatment system. It is not desirable to actuate the group 2 isolation signal by a transient or spurious signal.

Group 3 - isolation valves can only be opened when the reactor is at low pressure and the core standby cooling systems are not required. Also, since the reactor vessel could potentially be drained through these process lines, these valves are closed by low water level.

Group 4 and 5 - process lines are designed to remain operable and mitigate the consequences of an accident which results in the isolation of other process lines. The signals which initiate isolation of group 4 and 5 process lines are therefore indicative of a condition which would render them inoperable.

3.7.A & 4.7.A Primary Containment

Group 6 - process lines are normally in use and it is therefore not desirable to cause spurious isolation due to high drywell pressure resulting from non-safety related causes. To protect the reactor from a possible pipe break in the system, isolation is provided by high temperature in the cleanup system area or high flow through the inlet to the cleanup system. Also, since the vessel could potentially be drained through the cleanup system, a low level isolation is provided.

These valves are highly reliable, have low service requirements and most are normally closed. The initiating sensors and associated trip channels are also checked to demonstrate the capability for automatic isolation. The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate. More frequent testing for valve operability results in a greater assurance that the valve will be operable when needed.

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

The primary containment is penetrated by several small diameter instrument lines connected to the reactor coolant system. Each instrument line contains a 0.25 inch restricting orifice inside the primary containment. A program for periodic testing and examination of the excess flow check valves is performed in a manner similar to that described in Amendment No. 23, Millstone Unit 1, Dkt. 50-245.

Primary Containment Painting

The interiors of the drywell and suppression chamber are painted to prevent rusting. The inspection of the paint during each major refueling outage, approximately every 18 months, assures the paint is intact. Experience with this type of paint at fossil fueled generating stations indicates that the inspection interval is adequate.

BASES:

3.7.A & 4.7.A Primary Containment

Vacuum Relief

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and reactor building so that the structural integrity of the containment is maintained. The vacuum relief system from the pressure suppression chamber to reactor building consists of two 100% vacuum relief breakers (2 parallel sets of 2 valves in series). Operation of either system will maintain the pressure differential less than 2 psig; the external design pressure. One valve may be out of service for repairs for a period of seven days. If repairs cannot be completed within seven days, the reactor coolant system is brought to a condition where vacuum relief is no longer required.

The capacity of the 10 drywell vacuum relief valves is sized to limit the pressure differential between the suppression chamber and drywell during post-accident drywell cooling to the design limit of 2 psig. They are sized on the basis of the Bodega Bay pressure suppression system tests. The ASME Boiler and Pressure Vessel Code, Section III, Subsection B, for this vessel allows a 5 psig vacuum; therefore, with two vacuum relief valves secured in the closed position and eight operable valves, containment integrity is not impaired.

Reactor operation is permissible if the bypass area between the primary containment drywell and suppression chamber does not exceed an allowable area. The allowable bypass area is based upon analysis considering primary system break area, suppression chamber effectiveness, and containment design pressure. Analyses show that the maximum allowable bypass area is 0.2 ft^2, which is equivalent to all vacuum breakers open 3/32". (See letters from Boston Edison to the Directorate of Licensing, dated May 15, 1973 and October 22, 1974)

Reactor operation is not permitted if differential pressure decay rate is demonstrated to exceed 25% of allowable, thus providing a margin of safety for the primary containment in the event of a small break in the primary system.

Each drywell suppression chamber vacuum breaker is equipped with three switches. One switch provides full open indication only. Another switch provides closed indication and an alarm on Panel C-7 should any vacuum breaker come off its closed seat by greater than 3/32". The third switch provides a separate and redundant alarm on Panel 905 should any vacuum breaker come off its closed seat by greater than 3/32". The two alarms above are those referred to in Section 3.7.A.4.a(3) and 3.7.A.4.d.

The water in the suppression chamber is used only for cooling in the event of an accident; i.e., it is not used for normal operation; therefore, a daily check of the temperature and volume is adequate to assure that adequate heat removal capability is present.

BASES:

3.7.A & 4.7.A Primary Containment

Inerting

The relatively small containment volume inherent in the GE-BWR pressure suppression containment and the large amount of zirconium in the core are such that the occurrence of a very limited (a percent or so) reaction of the zirconium and steam during a loss-of-coolant accident could lead to the liberation of hydrogen combined with an air atmosphere to result in a flammable concentration in the containment. If a sufficient amount of hydrogen is generated and oxygen is available in stoichiometric quantities, the subsequent ignition of the hydrogen in rapid recombination rate could lead to failure of the containment to maintain a low leakage integrity. The 4% oxygen concentration minimizes the possibility of hydrogen combustion following a loss-of-coolant.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss-of-coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration.

The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least twice a week the oxygen concentration will be determined as added assurance. Mark I Containment Long Term Program testing showed that maintaining a drywell to wetwell pressure differential to keep the suppression chamber downcomer legs clear of water significantly reduced suppression chamber post LOCA hydrodynamic loads. A pressure of 1.17 psid is required to sufficiently clear the water legs of the downcomers without bubbling nitrogen into the suppression chamber at the 3.00 ft. downcomer submergence which corresponds to approximately 84,000 $ft.^3$ of water. Maximum downcomer submergence is 3.25 ft. at operating suppression chamber water level. The above pressure differential and submergence number are used in the Pilgrim I Plant Unique Analysis.

Post LOCA Atmosphere Dilution

In order to ensure that the containment atmosphere remains inerted, i.e. the oxygen-hydrogen mixture below the flammable limit, the capability to inject nitrogen into the containment after a LOCA is provided. A minimum of 1500 gallons of liquid N_2 in the storage tank assures that a three-day supply of N_2 for post-LOCA containment inerting is available. Since the inerting makeup system is continually functioning, no periodic testing of the system is required.

The Post-LOCA Containment Atmospheric Dilution (CAD) System is designed to meet the requirements of AEC Regulatory Guides 1.3, 1.7 and 1.29, ASME Section III, Class 2 (except for code stamping) and seismic Class I as defined in the PNPS FSAR.

In summary, the limiting criteria are:

1. Maintain hydrogen concentration in the containment during post-LOCA conditions to less than 4%.

2. Limit the buildup in the containment pressure due to nitrogen addition to less than 28 psig.

3. To limit the offsite dose due to containment venting (for pressure control) to less than 300 Rem to the thyroid.

By maintaining at least a 3-day supply of N_2 on site there will be sufficient time after the occurrence of a LOCA for obtaining additional nitrogen supply from local commercial sources.[1] The system design contains sufficient redundancy to ensure its reliability. Thus, it is sufficient to test the operability of the whole system once per operating cycle. The H_2 analyzers will provide redundancy for the drywell i.e., there are two H_2 analyzers for the Unit. By permitting reactor operation for 7 days with one of the two H_2 analyzers inoperable, redundancy of analyzing capability will be maintained while not imposing an immediate interruption in plant operation. Monthly testing of the analyzers using H_2 will be adequate to ensure the system's readiness because of the design. Since the analyzers are normally not in operation there will be little deterioration due to use. In order to determine H_2 concentration, the analyzers must be warmed up 6 hours prior to putting into service. This time frame is acceptable for accident conditions because a 4% H_2 level will not be reached in the drywell until 16 hours following the accident. Due to nitrogen addition, the pressure in the containment after a LOCA will increase with time. Under the worst expected conditions the containment pressure will reach 28 psig in approximately 45 days. If and when that pressure is reached, venting from the containment shall be manually initiated per the requirements of 10CFR50.44. The venting path will be through the Standby Gas Treatment system in order to minimize the off site dose.

(1). As listed in Pilgrim Nuclear Power Station Procedure No. 5.4.6 "Post Accident Venting".

BASES:

3.7.C - Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

Initiating reactor building isolation and operation of the standby gas treatment system to maintain at least a 1/4 inch of water negative pressure within the secondary containment provides an adequate test of the operation of the reactor building isolation valves, leak tightness of the reactor building and performance of the standby gas treatment system. Functionally testing the initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Performing these tests prior to refueling will demonstrate secondary containment capability prior to the time the primary containment is opened for refueling. Periodic testing gives sufficient confidence of reactor building integrity and standby gas treatment system performance capability.

(next pg. is 177)

TABLE 6.9.1

	Area	Reference	Submittal Date
a.	Secondary Containment Leak Rate Testing (1)	4.7.C.1.c	Upon completion of each test (2)
b.	In-service Inspection Evaluation	4.6.G.	Five years after commercial operation
c.	(Deleted)		
d.	Gross Gaseous Release 0.05 Ci/sec for 48 Hours	4.8.B.	Ten days after the release occurs

NOTES:

1. Each integrated leak rate test of the secondary containment required by 4.7.C.1.c. shall be the subject of a summary technical report. This report should include data on the wind speed, wind direction, outside and inside temperatures during the test, concurrent reactor building pressure, and emergency ventilation flow rate. The report shall also include analyses and interpretations of those data which demonstrate compliance with the specified leak rate limits.

2. The report shall be submitted approximately 90 days after completion of each test. Test periods shall be based on the commercial service date as the starting point.

Attachment A

Reference

To aid in your review of this proposal we have included the following list of references:

1) NRC Letter dated 8/5/75 (BECo #1.75.75) - Requests evaluation of BECo Program vs. Appendix J
2) BECo Letter dated 10/10/75 - Exemption requests and proposed Tech. Specs.
3) BECo Letter dated 1/27/76 (BECo #2.76.11) - Exemption requests
4) BECo Letter dated 6/4/76 (BECo #2.76.48) - Revises letter of 1/27/76
5) NRC Letter 7/23/76 - Approves a portion of Proposed Tech Spec of 10/10/75
6) NRC Letter dated 8/12/80 (BECo #1.80.302) Requests additional information on BECo 1/27/76 Letter
7) BECo Letter dated 10/27/80 (BECo #2.80.275) BECo response to NRC 8/12/80 letter
8) NRC Letter dated 4/28/81 (BECo #1.81.111) Request to update Tech Spec
9) BECo Letter dated 9/15/81 (BECo #2.81.218) Additional information
10) NRC Letter 6/9/82 (1.82.160) Approval of one requested exemption
11) NRC Letter dated 7/2/84 (1.84.188) Request to revise Tech Specs
12) BECo Letter 10/5/84 (2.84.169) Request time extension for response to 7/2/84 due to RFO #6

Proposed Change

In light of present industry and regulatory work in the area of Technical Specification Improvement Programs, we propose this change to the PNPS Technical Specifications which incorporates the fundamentals of this program.

For example, in the area of Technical Specification Improvement, this change combines our present sections 3.7.D (Primary Containment Isolation Valves) into section 3.7.A.2 (Primary Containment). As such, requirements for primary containment are located in one section. Also, we were able to delete LCO 3.7.D.3 because it is redundant to LCO 3.7.A.6.

Major Changes

This proposed technical specification is comprised of four (4) major changes as described below. These major changes resulted in a variety of administrative changes to achieve consistency. To aid in your review of this proposal, we have summarized all changes on a table (Attachment B).

Change #15 - Combine Sections 3.7.A.2 and 3.7.D

Section 3.7.A.2 is Primary Containment, Section 3.7.D is Primary Containment Isolation Valves. Having two separate sections with Primary Containment requirements has caused some confusion at times. By combining these two sections, the requirements of primary containment are located in one place. The present requirements of 3.7.D have not been changed (except those which are redundant to 3.7.A.2 - see Change #30).

In reviewing for significant hazards considerations as defined in 48FR14870, this change (and the minor associated changes) are purely administrative - example (i) "to achieve consistency throughout the technical specifications." The minor changes associated with this change are numbers 1, 4, 6, 10, 17, 18, 23, 24, 29, 30, and 37.

Also, to ease reading of the technical specifications, we propose to add some additional headings in the LCO's, Surveillance Requirements and Bases. These headings are purely administrative under the auspices of 48FR14870 example (i). These additions are changes 7, 9, 14, 34 and 38.

Finally, when melding the bases of 3.7.D into 3.7.A (see change #32), a logical progression of information could not be found. Therefore, we have reformatted the order of paragraphs. No other changes in the content of the Bases have been made except where specifically shown (see changes 33, 34, 35, 36, 37, 38, 39, 40, 41, 45 and 46). Again, this is an administrative change, example (i) of 48FR14870.

Change #2 - Definition 1.0.M

In the current definition and the applicable LCO's (Change #11, 20 and 21) there is an intended discrepancy due to the design of our main steam isolation valve (MSIV) circuitry. The current definition states that the "valves are operable or deactivated...", the LCO's state the "valves are operable or shall be placed..." This is because each of the four (4) MSIV's (inboard or outboard) are protected by two (2) fuses (one AC and one DC); removal of the two (2) fuses causes all four (4) of the associated MSIV's to close.

Also, in the definition we have added the words "at least one containment isolation valve in each line having an inoperable valve shall be". These words account for the possibility of a valve failing in a non-isolated condition - we must close the other valve.

We believe the proposed rewording of the definition and the LCO's achieves consistency in the Tech. Specs. without changing the intent. As such, this is an administrative change for consistency as shown in example (i) of 48FR14870.

Change #31 - Delete Table 3.7.1

Table 3.7.1 as presently shown in Tech. Specs. needed to be updated due to physical changes made during our recent Refueling Outage #6.

The objective of this section (3.7.A) of Tech. Specs. is to assure primary containment integrity. In summary the LCO's are: if integrity cannot be met (LCO 3.7.A.2.a), then shut down (3.7.A.6).

In reviewing this table (Primary Containment Isolation Valves), we found it to be incomplete - it does not contain manual isolation valves. To incorporate the manual valves would not achieve any greater degree of safety than presently in Technical Specifications. Also, the surveillance requirements are not applicable to manual valves. We therefore believe this table is unnecessary in the Technical Specifications, to achieve the objective of assuring the integrity of primary containment.

The complete list of Primary Containment Isolation Valves is contained in station procedure 2.2.125. Changes in our procedures cannot be made without an approved Safety Evaluation (10CFR50.59) and must be reviewed and approved by our Operational Review Committee (ORC). Also, the daily operating procedure for surveillances (Oper. 9) further reference this procedure (2.2.125) for Tech. Spec. surveillance requirement 4.7.A.2.b.2. (Adding definition "M" to the LCO's, Change #11, further negates the need for this table.)

With respect to the associated valve closing times for automatic isolation valves, we believe this table is an unnecessary duplication of definition 1.0.M and LCO 3.7.A.2.a.4. Should a valve be declared inoperable, it can only be returned to service upon successful completion of operability testing. ASME Section XI requires closing time testing for operability. These times are in our ORC approved station procedures.

(Note: Although the list of primary containment valves, complete with valve timings, is in the PNPS-FSAR, we believe using the station procedure as the controlling document is more accurate because the FSAR is only updated annually.)

This change has been determined to be administrative (example i) based on our review of 48FR14870 for significant hazards considerations. This change achieves consistency between containment integrity (LCO 3.7.A.2.a) and Primary Containment Isolation Valves (LCO 3.7.A.2.b)

To make this change, several editorial changes needed to be made. These changes were administrative as shown in example (i) in 48FR14870 as far as significant hazards considerations. These changes are numbers 3, 16, 19, 25, 25a, and 46.

<u>Change #12 and #13</u> - Primary Containment Testing

This proposed change is in response to NRC letter dated September 2, 1984 (Reference 12).

The Pilgrim Station primary containment test program was written prior to the implementation of 10CFR50 Appendix J. Since Appendix J did not exist, all requirements were included in the Technical Specifications.

We propose to revise our Technical Specifications to include only those items <u>required</u> to be in Tech. Specs. by 10 CFR 50 App. J.

By adding the specific reference to App. J, we assure compliance with the rule, and can eliminate the detailed descriptions of the test methods. These descriptions are in our ORC approved station procedures and can only be changed with an approved safety evaluation (10 CFR 50.59) <u>and</u> ORC approval.

Our review of 48FR14870 for significant hazards considerations shows this change being similar to example (vii) - "a change to make the license conform to changes in the regulations" in that this change makes our license conform to the present regulations.

The changes made by #12 and #13 deleted several sections. Therefore, to avoid confusion we have changed the Surveillance Requirement number (only) of 4.7.A.2.g to 4.7.A.2.c (change #26) and 4.7.A.2.h to 4.7.A.2.d (change #27). These changes present no significant hazards consideration as they are administrative in nature as shown by example (i) of 48FR14870.

Also, due to changes in the surveillance requirements for Primary Containment Testing, the applicable bases required changes for consistency. These changes are #36 (delete references to Method A & B tests) and #45 (delete reference to AEC guide and add reference to 10 CFR 50 App. J). Changes for consistency are shown in example (i) of 48FR14870 and therefore present no significant hazards consideration.

Finally, specifically referencing 10 CFR 50 App. J, also required a change to Table 6.9.1 Note 2 (change #44). Adding the word "approximately" to the note makes our Tech. Spec. wording the same as the present rule. Therefore, the significant hazards consideration again is similar to example (vii) of 48FR14870 as stated above.

Other Changes

Since we are preparing the above "major" changes, there are several other changes (detailed below) which will provide consistency with present Technical Specifications, clarity and overall improvement.

Change #5 and #40

These proposed changes delete specific references to the FSAR. They are unnecessary and would require further Tech. Spec. changes should the FSAR section numbers change due to FSAR revisions (10 CFR 50.71(e)). This is an administrative change as shown in example (i) of 48FR14870.

Change #8

Change LCO #3.7.A.3 to 3.7.A.1.n - the present location in Tech. Specs. is inappropriate as it belongs in 3.7.A.1 (Suppression Pool) not 3.7.A.3 (Vacuum Breakers). This is an administrative change "correction of an error "as shown in example (i) of 48FR14870.

Change #22

Add note to LCO 3.7.A.2.b "Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under ORC approved administrative controls".

This change is necessary to allow operability testing of inoperable valves prior to returning them to service.

A review of 48FR14780 for significant hazards considerations shows that this change falls into example (vii) "A change to make a license conform to changes in the regulation..."

This change is consistent with the current wording of NRC approved BWR Standard Technical Specifications. The ORC approved administrative controls will assure that proper controls are in place to allow safe testing of this valve.

Change #28 and #39

Replace "operating cycle" with "refueling outage" is an administrative change for clarity as shown in 48FR14870 example (i). This surveillance is only performed during shutdown periods. Also, this change is consistent with previously approved changes.

Change #33

The present wording of these bases was changed to reflect the correct present and past tenses. The intent of the bases has not been changed. Therefore, this is an administrative change for consistency and presents no significant hazards consideration under example (i) of 48FR14870.

Change #35

The word "calculated" was added for clarity in the sentence. There is no significant hazard consideration as this is administrative under the auspices of example (i) of 48FR14870.

Change #41 and #42

Added note "the next pg is" to account for deleted pages.

Change #43

Corrected reference to read 4.7.C.1.c. This was an error in a previous submittal and presents no significant hazards consideration as shown by example (i) of 48FR14870.

Safety Considerations

This change has been reviewed and approved by the Operations Review Committee and reviewed by the Nuclear Safety Review and Audit Committee.

Significant Hazards Considerations (Summary)

The Commission has provided guidance for the application of the standards for determining whether a significant hazards consideration exists by providing examples of amendments not likely to involve significant hazards considerations (48FR14870).

These changes involve two (2) of the referenced types contained in 48FR14870: (i) administrative, and (vii) a change to conform to changes in the regulations.

For the reasons discussed above, (see proposed changes) the changes proposed herein do not present significant hazards consideration because the operation of Pilgrim Nuclear Power Station in accordance with these proposed changes would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

Schedule of Change

This change will be put into effect 60 days following BECo's receipt of its approval by the Commission.

Fee Determination

Pursuant to 10 CFR170.12(c), an application fee of $150.00 is included with the proposed amendment.

ATTACHMENT B

	PROPOSED	NEW PAGE	OLD PAGE	SECTION	CHANGE	REASON	SIGNIFICANT HAZARD CONSIDERATION
1	DELETE	NA	ii	TABLE OF CONTENTS	Delete 3.7.D	Section proposed to be moved.	Administrative (See #15)
2	CHANGE	3	3	DEFINITIONS	Definition "M", add words "at least one containment isolation valve in each line having an inoperable valve shall be."	To achieve consistency between definition and LCO's.	Administrative
3	ADDITION	5a	5a	DEFINITION	Add Def. Y.	To clarify LCO/ Surv. Req. 3/4.7.A.2.b.	Administrative (See #31)
4	DELETE	NA	68	BASES	Delete reference to 3.7.D.	Section proposed to be deleted.	Administrative (See #15)
5	DELETE	NA	69	BASES	Delete reference to FSAR Sections 6.5.3.1 and 14.6.5.	Unnecessary References.	Administrative
6	DELETE	NA	69	BASES	Delete references to Section 3.7.D.	Section proposed to be deleted.	Administrative (See #15)

	PROPOSED	NEW PAGE	OLD PAGE	SECTION	CHANGE	REASON	SIGNIFICANT HAZARD CONSIDERATION
7	ADDITION	152	152	LCO	Add title "Suppression Pool Water Volume and Temperature."	For ease in reading 3.7.A LCO's.	Administrative (See #15)
8	CHANGE	154	152B	LCO	Change LCO number from 3.7.A.3 to 3.7.A.1.n.	Present location in Tech. Specs. inappropriate. (Correction of an error)	Administrative
9	ADDITION	155	152B	LCO	Add title "Primary Containment Integrity."	For ease in reading 3.7.A LCO's.	Administrative (See #15)
10	CHANGE	155	152B	LCO	Change LCO number from 3.7.A.2 to to 3.7.A.2.a.	Due to addition of new material to this section.	Administrative (See #15)
11	ADDITION	155	152B	LCO	Added definition "M" to LCO 3.7.A.2.a.	For clarification of the intent of LCO.	Administrative (See #2)

	PROPOSED	NEW PAGE	OLD PAGE	SECTION	CHANGE	REASON	SIGNIFICANT HAZARD CONSIDERATION
12	CHANGE	155	152B, 153, 154 155	Surv. Req.	Revised 4.7.A.2.a thru f to reference 10CFR50 App J, as amended through Sept. 22, 1980, and approved NRC exceptions and exemptions.	For conformance to 10CFR50 App J.	A change to conform to in the regulations
13	DELETE	155	152B, 153, 154, 155	Surv. Req.	Deleted detailed descriptions of containment test program requirements (in Surv. Req. 3.7.A.2.a - f) except those required to be in tech specs. by by 10CFR50 App J.	The primary containment test program is controlled by ORC approved procedures. The added reference to 10CFR50 App J (above) assures compliance.	A change to conform to the regulations (See #12)
14	ADDITION	155a	NA	LCO	Add title "Primary Containment Isolation Valves."	For ease in reading 3.7.A LCO's.	Administrative (See #15)
15	CHANGE	155	160	LCO	Change LCO number from 3.7.D.1 to 3.7.A.2.a.4.	To simplify tech. specs. by combining 3.7.A.2 and 3.7.D, primary containment requirements are in one section.	Administrative

	PROPOSED	NEW PAGE	OLD PAGE	SECTION	CHANGE	REASON	SIGNIFICANT HAZARD CONSIDERATION
16	CHANGE	155	160	LCO	Delete reference in 3.7.A.2.a.4 to Table 3.7.1 and add "automatic primary containment."	Due to deletion of Table 3.7.1.	Administrative (See #31)
17	CHANGE	155	160	LCO	Change referenced LCO from 3.7.D.2 to 3.7.A.2.b.	To simplify tech. specs. by combining 3.7.A.2 and 3.7.D, primary containment requirements are in one section.	Administrative (See #15)
18	CHANGE	155a	160	LCO	Change LCO number from 3.7.D.2 to 3.7.A.2.b.	To simplify tech. specs. by combining 3.7.A.2 and 3.7.D, primary containment requirements are in one section.	Administrative (See #15)
19	CHANGE	155a	160	LCO	Delete reference in 3.7.A.2.b to Table 3.7.1 and add "automatic primary containment."	Due to deletion of Table 3.7.1.	Administrative (See #31)

	PROPOSED	NEW PAGE	OLD PAGE	SECTION	CHANGE	REASON	SIGNIFICANT HAZARD CONSIDERATION
20	CHANGE	155a	160	LCO	Delete "placed in in the" and add "deactivated."	For consistency with Definition "M" and LCO 3.7.A.2.a.3.	Administrative (See #2)
21	ADDITION	155a	NA	LCO	Add "this requirement may be satisfied by ... or otherwise secure the valve."	For consistency with Definition "M" and LCO 3.7.A.2.a.3.	Administrative (See #2)
22	ADDITION	155a	160	NOTE (*)	Add "Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under ORC approved administrative controls."	To allow operability testing of inoperable valve following repair.	A change to conform to the regulations.
23	CHANGE	155a	160	Surv. Req.	Change Surv. Req. number from 4.7.D.1.a - d to 4.7.A.2.b.1.a thru d.	To simplify tech. specs. by combining 3.7.A.2 and 3.7.D, primary containment requirements are in one section.	Administrative (See #15)
24	CHANGE	155a	160	Surv. Req.	Change Surv. Req. number 4.7.D.2 to 4.7.A.2.b.2.	To simplify tech. specs. by combining 3.7.A.2 and 3.7.D, primary containment requirements are in one section.	Administrative (See #15)

	PROPOSED	NEW PAGE	OLD PAGE	SECTION	CHANGE	REASON	SIGNIFICANT HAZARD CONSIDERATION
25	CHANGE	155a	160	Surv. Req.	Delete reference in 4.7.A.2.b.2 and add "automatic primary containment."	Due to deletion of Table 3.7.1.	Administrative (See #31)
25a	CHANGE	155a	160	Surv. Req.	Delete words "at least one other" and add "the isolated."	For clarity	Administrative (See #31)
26	CHANGE	155b	155	Surv. Req.	Change Surv. Req. number from 4.7.A.2.g to 4.7.A.2.c.	Due to deletion of other sections.	Administrative (See #12)
27	CHANGE	155b	155	Surv. Req.	Change Surv. Req. number from 4.7.A.2.h to 4.7.A.2.d.	Due to deletion of other sections.	Administrative (See #12)
28	CHANGE	155b	155	Surv. Req.	In 4.7.A.2.c, delete words "operating cycle" and replace with "refueling outage."	For consistency with operating practices. (Note: this surveillance is performed during refueling outages. This change has been approved in other sections).	Administrative

	PROPOSED	NEW PAGE	OLD PAGE	SECTION	CHANGE	REASON	SIGNIFICANT HAZARD CONSIDERATION
29	CHANGE	157a	157a	LCO	Change reference from 3.7.A to "3.7.A.1 thru 3.7.A.5."	To clarify reference.	Administrative (See #15)
30	DELETE	155a	160	LCO	Delete LCO 3.7.D.3.	To simplify tech. specs. by combining 3.7.A.2 and 3.7.D, LCO 3.7.D.3 is an exact duplicate of of LCO 3.7.A.6 and is therefore unnecessary.	Administrative (See #15)
31	DELETE	157a	161 thru 165	LCO	Delete Table 3.7.1.	To achieve consistency between LCO's 3.7.A.2.a and 3.7.A.2.6.	Administrative
32	CHANGE	160, 161, 162, 163, 164, 165, 166, 167	166, 167, 168, 168a, 169, 170, 171, 171a	BASES	Except as stated below, no changes were made to content of bases. The order of paragraphs was changed and pages renumbered.	When attempting to meld Bases 3.7.D into 3.7.A, a logical progression of information could not be found. By restructuring information, the bases order is more in line with Tech. Specs. 3/4.7.A.	Administrative (See #15)

	PROPOSED	NEW PAGE	OLD PAGE	SECTION	CHANGE	REASON	SIGNIFICANT HAZARD CONSIDERATION
33	CHANGE	160 and 166	166 and 171	BASES	Changed words to correct tense.	To bring Bases up to date.	Administrative
34	ADDITION	162	169	BASES	Added title "Primary Containment Testing."	For clarity.	Administrative (See #15)
35	ADDITION	162	169	BASES	Added word "calculated."	For clarity.	Administrative
36	DELETE	162	170	BASES	Deleted references to "Methods A & B tests."	Due to deletion of information in surveillance requirements.	Administrative (See #12)
37	CHANGE	163, 164	175, 176	BASES	Moved bases (in total) for primary containment isolation valves from 3.7.D to 3.7.A.	Due to changes in LCO/Surveillance Requirements.	Administrative (See #15)
38	ADDITION	164	NA	BASES	Added title "Primary Containment Painting."	For clarity.	Administrative (See #15)

	PROPOSED	NEW PAGE	OLD PAGE	SECTION	CHANGE	REASON	SIGNIFICANT HAZARD CONSIDERATION
39	CHANGE	164	169	BASES	Deleted words "once per year" to "every 18 months."	To be consistent with present operating cycle of 18 months.	Administrative (See #28)
40	DELETE	167	171	BASES	Deleted specific FSAR references.	Unnecessary references.	Administrative (See #5)
41	CHANGE	167	171	BASES	Added note" the next page is 172."	To account for deleted pages.	Administrative
42	ADDITION	175	175	NA	Added note "next page is 177."	To account for deleted pages.	Administrative
43	CHANGE	225	225	Table 6.9.1	Changed reference from 4.7.C.c to 4.7.C.1.c.	Correct an error.	Administrative
44	CHANGE	225	225	Table 6.9.1 Note 2	Added word "approximately".	To agree with 10CFR50 App J.	A change to conform with the regulations. (See #12)
45	CHANGE	163	170	BASES	Delete reference to AEC guide and Add reference to 10CFR50 App. J.	To achieve consistency in Tech. Specs.	Administrative (See #12)
46	DELETE	164	176	BASES	Delete references to valve timings.	Due to deletion of Table 3.7.1.	Administrative (See #31)