COOPER NUCLEAR STATION
TABLE 3.2.F
PRIMARY CONTAINMENT SURVEILLANCE INSTRUMENTATION

Instrument	Instrument I.D. No.	Range	Minimum Number of Operable Instrument Channels	Action Required When Minimum Condition Not Satisfied (1)
Reactor Water Level	NBI-LI-85A	-150" to +60"	2	A,B,C
	NBI-LI-85B	-150" to +60"		
Reactor Pressure	RFC-PI-90A	0 - 1200 psig	2	A,B,C
	RFC-PI-90B	0 - 1200 psig		
Drywell Pressure	PC-PI-512A	0 - 80 psia	2	A,B,C
	PC-PR-512B	0 - 80 psia		
Drywell Temperature	PC-TR-503	50 - 170°F	2	A,B,C
	PC-TI-505	50 - 350°F		
Suppression Chamber Air Temperature	PC-TR-21A	0 - 300°F	2	A,B,C
	PC-TR-23, Ch 1 & 2	0 - 400°F		
Suppression Chamber Water Temperature	PC-TR-21B	0 - 300°F	2	A,B,C
	PC-TR-22, Ch 1 & 2	0 - 400°F		
Suppression Chamber Water Level	PC-LI-10	(-4' to +6')	2	A,B,C
	PC-LR-11	(-4' to +6')		
	PC-LI-12	-10" to +10"	2	A,B,C,E
	PC-LI-13	-10" to +10"		
Suppression Chamber Pressure	PC-PR-20	0 - 2 psig	1	B,C
Control Rod Position	N.A.	Indicating Lights	1	A,B,C,D
Neutron Monitoring	N.A.	S.R.M., I.R.M., LPRM 0 - 100% power	1	A,B,C,D

NOTES FOR TABLE 3.2.F

1. The following actions will be taken if the minimum number of operable instrument channels as required are not available.

   A. From and after the date that one of these parameters is reduced to one indication, continued operation is permissible during the succeeding thirty days unless such instrumentation is sooner made operable.

   B. From and after the date that one of these parameters is not indicated in the control room, continued operation is permissible during the succeeding seven days unless such instrumentation is sooner made operable.

   C. If the requirements of A and B above cannot be met, an orderly shutdown shall be initiated within 24 hours.

   D. These surveillance instruments are considered to be redundant to each other.

   E. In the event that both channels are inoperable and indication cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in Hot Shutdown in six (6) hours and in a Cold Shutdown condition in the following eighteen (18) hours.

COOPER NUCLEAR STATION
TABLE 4.2.F
PRIMARY CONTAINMENT SURVEILLANCE INSTRUMENTATION
TEST AND CALIBRATION FREQUENCIES

Instrument	Instrument I.D. No.	Calibration Frequency	Instrument Check
Reactor Water Level	NBI-LI-85A	Once/6 Months	Each Shift
	NBI-LI-85B	Once/6 Months	Each Shift
Reactor Pressure	RFC-PI-90A	Once/6 Months	Each Shift
	RFC-PI-90B	Once/6 Months	Each Shift
Drywell Pressure	PC-PR-512A	Once/6 Months	Each Shift
	PC-PI-512B	Once/6 Months	Each Shift
Drywell Temperature	PC-TR-503	Once/6 Months	Each Shift
	PC-TI-505	Once/6 Months	Each Shift
Suppression Chamber Air Temperature	PC-TR-21A	Once/6 Months	Each Shift
	PC-TR-23, Ch. 1 & 2	Once/6 Months	Each Shift
Suppression Chamber Water Temperature	PC-TR-21B	Once/6 Months	Each Shift
	PC-TR-22, Ch. 1 & 2	Once/6 Months	Each Shift
Suppression Chamber Water Level	PC-LI-10	Once/6 Months	Each Shift
	PC-LR-11	Once/6 Months	Each Shift
	PC-LI-12	Once/6 Months	Each Shift
	PC-LI-13	Once/6 Months	Each Shift
Suppression Chamber Pressure	PC-PR-20	Once/6 Months	Each Shift
Control Rod Position	N.A.	N.A.	Each Shift
Neutron Monitoring (APRM)	N.A.	Once/Week	Each Shift

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

1. Suppression Pool

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 suppression pool water volume and temperature shall be maintained within the following limits except as specified in 3.7.A.2. and 3.5.F.5.

a. Minimum water volume - 87,650 ft^3

b. Maximum water volume - 91,100 ft^3

c. Maximum suppression pool temperature during normal power operation - 95°F.

d. During testing which adds heat to the suppression pool, the water temperature shall not exceed 10°F above the normal power operation limit specified in c. above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in c. above within 24 hours.

e. The reactor shall be scrammed from any operating condition if the pool temperature reaches 110°F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in c. above.

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

1. Suppression Pool

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

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.7.A (Cont'd)	4.7.A.2.F (cont'd)

4.7.A.2.F (cont'd)

4. Main steam line and feedwater line expansion bellows as specified in Table 3.7.3 shall be tested by pressurizing between the laminations of the bellows at a pressure of 5 psig. This is an exemption to Appendix J of 10CFR50.

5. The personnel airlock shall be tested at 58 psig at intervals no longer than six months. This testing may be extended to the next refueling outage (not to exceed 24 months) provided that there have been no airlock openings since the last successful test at 58 psig. In the event the personnel airlock is not opened between refueling outages, it shall be leak checked at 3 psig at intervals no longer than six months. Within three days of opening (or every three days during periods of frequent opening) when containment integrity is required, test the personnel airlock at 3 psig. This is an exemption to Appendix J of 10CFR50.

g. Deleted

h. Drywell Surfaces

The interior surfaces of the drywell and torus shall be visually inspected each operating cycle for evidence of torus corrosion or leakage.

3.7 & 4.7 BASES

3.7.A & 4.7.A PRIMARY CONTAINMENT

3.7.A.1 & 4.7.A.1 SUPPRESSION POOL

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 10CFR100 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 is made to this requiremen during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on th system at this time, thus greatly reducing the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. 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 reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep off-site doses well below 10CFR100 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.

As a result of the Mark I Containment Program, the District has completed the evaluation and requalification of the various containment structures and components at CNS. As a result of the requalification work, significant modifications were designed in accordance with the NRC acceptance criteria and installed. The Plant Unique Analysis Report, which was submitted on April 29,1982, and accepted on January 20, 1984, contains a detailed summary of the modifications installed. The maximum and minimum water volumes of 91,100 and 87,650 were not altered, but the downcomers were shortened by 1' 0½", so that their nominal submergence is now 3 feet and the initial volume of water in them is decreased proportionately. The acceptability of this is proven in "Mark I Containment Program Downcomer Submergence Functional Assessment Report", Task 6.6, NEDE - 21885-P, Class III, June, 1978.

Should it be necessary to drain the suppression chamber, this should only

3.7.A & 4.7.A BASES (cont'd)

be done when there is no requirement for core standby cooling systems operability as explained in bases 3.5.F.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. 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 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 change 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.

The maximum suppression pool temperature of 95°F is based on not exceeding the 200°F Mark I temperature limit as contained in NUREG-0661. This 95°F limit also prevents exceeding LOCA considerations, or ECCS pump NPSH requirements. The basis for these limits are contained in NEDC-24360-P.

3.7.A.2 & 4.7.A.2 CONTAINMENT INTEGRITY

The maximum allowable test leak rate is 0.635%/day at a pressure of 58 psig, the peak calculated accident pressure. Experience has shown that there is negligible difference between the leakage rates of air at normal temperature and a steam-hot air mixture.

Establishing the test limit of 0.635%/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, La, or the allowable test leak rate, Lt, by 0.75 thereby providing a 25% margin to allow for leakage deterioration which may occur during the period between leak rate tests.

The primary containment leak rate test frequency is based on maintaining adequate assurance that the leak rate remains within the specification. The leak rate test frequency is based on the NRC guide for developing leak rate testing and surveillance of reactor containment vessels. Allowing the test intervals to be extended up to 8 months permits some flexibility needed to have the tests coincide with scheduled or unscheduled shutdown periods.

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.

Table 3.7.4 identifies certain isolation valves that are tested by pressurizing the volume between the inboard and outboard isolation valves. This results in conservative test results since the inboard valve, if a globe valve, will be tested such that the test pressure is tending to lift the globe off its seat. Additionally, the measured leak rate for such a test is conservatively assigned to both of the valves equally and not divided between the two.

The main steam and feedwater testable penetrations consist of a double layered metal bellows. The inboard high pressure side of the bellows is subjected to drywell pressure. Therefore, the bellows is tested in its entirety when the drywell is tested. The bellows layers are tested for the integrity of both layers by pressurizing the void between the layers to 5 psig. Any higher pressure could cause permanent deformation, damage and possible ruptures of the bellows.

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 peak drywell pressure would be about 58 psig which would rapidly reduce to 29 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. Based on the calculated containment pressure response discussed above, the primary containment preoperational test pressure was 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 0.635%/day at 58 psig. Calculations made by the NRC staff with leak rate and a standby gas treatment system filter efficiency of 90% for halogens and assuming the fission product release fractions stated in NRC Regulatory Guide 1.3, show that the maximum total whole body passing cloud dose is about 1.0 REM and the maximum total thyroid dose is about 12 REM at 1100 meters from the stack over an exposure duration of two hours. The resultant 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 doses and 10 CFR 100 guidelines.

The water in the suppression chamber is used 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.

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

3.7.A.3 & 4 and 4.7.A.3 & 4 VACUUM BREAKERS

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 a pressure differential of less than 2 psi, the external design pressure. One valve may be out of service for repairs for a period of 7 days. If repairs cannot be completed within 7 days the reactor coolant system is brought to a condition where vacuum relief is no longer required.

The capacity of the 12 drywell vacuum relief valves are sized to limit the pressure differential between the suppression chamber and drywell during post-accident drywell cooling operations to well under the design limit of 2 psi. 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 2 psi differential; therefore, with three vacuum relief valves secured in the closed position and 9 operable valves, containment integrity is not impaired.

3.7.A.5 and 4.7.A.5 OXYGEN CONCENTRATION

Safety Guide 7 assumptions for Metal-Water reaction result in hydrogen concentration in excess of the Safety Guide 7 flammability limit. By keeping the oxygen concentration less than 4% by volume the requirements of Safety Guide 7 are satisfied.

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 period of time with significant leaks in the primary system, leak inspections are scheduled during 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.

3.7.A & 4.7.A BASES (cont'd)

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.

The 500 gallon conservative limit on the nitrogen storage tank assures that adequate time is available to get the tank refilled assuming normal plant operation. The estimated maximum makeup rate is 1500 SCFD which would require about 160 gallons for a 10 day makeup requirement. The normal leak rate should be about 200 SCFD.

3.7.B & 3.7.C STANDBY GAS TREATMENT SYSTEM AND 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 shut down 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. Secondary containment may be broken for short periods of time to allow access to the reactor building roof to perform necessary inspections and maintenance.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions. Both standby gas treatment system fans are designed to automatically start upon containment isolation and to maintain the reactor building pressure to the design negative pressure so that all leakage should be in-leakage. Should one system fail to start, the redundant system is designed to start automatically. Each of the two fans has 100 percent capacity.

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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.

In order to assure that the doses that may result from a steam line break do not exceed the 10CFR100 guidelines, it is necessary that no fuel rod perforation resulting from the accident occur prior to closure of the main steam line isolation valves. Analyses indicate that fuel rod cladding perforations would be avoided for main steam valve closure times, including instrument delay, as long as 10.5 seconds.

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 and an excess flow check valve outside the primary containment. A program for periodic testing and examination of the excess flow check valves is performed as follows:

1. Vessel at pressure sufficient to actuate valves. This could be at time of vessel hydro following a refueling outage.

2. Isolate sensing line from its instrument at the instrument manifold.

3. Provide means for observing and collecting the instrument drain or vent valve flow.

4. Open vent or drain valve.

   a. Observe flow cessation and any leakage rate.

   b. Reset valve after test completion.

5. The head seal leak detection line cannot be tested in this manner. This valve will not be exposed to primary system pressure except under unlikely conditions of seal failure where it could be partially pressurized to reactor pressure. Any leakage path is restricted at the source and therefore this valve need not be tested. This valve is in a sensing line that is not safety related.

6. Valves will be accepted if a marked decrease in flow rate is observed and the leakage rate is acceptable.