e. Local Areas

The computer programs used in the analyses of the various interior structures are capable of analyzing the effects of corners and general discontinuities. The vent region, as well as other major openings and penetrations, are analyzed using graded fine mesh finite element models. Boundary conditions for each of these large scale models are obtained from the general analysis discussed in Section 3.8.3.4.2.a.1. Local reinforcement or stiffening is provided around the openings or penetrations for the calculated stress concentrations.

f. Analysis of the Equipment Hatch, Personnel Door and Drywell Head

Analysis of the drywell equipment hatch, personnel door and drywell head is in accordance with the requirements of the ASME Code Section III, Division 1, Class MC, for the metallic components and proposed ASME Code Section III, Division 2, for concrete anchorage details. Figures 3.8-30, 3.8-31 and 3.8-32 give typical details of these appurtenances and the anchorage details. The STARDYNE computer program is used for the static analysis, and dynamic effects are evaluated by the use of floor response curves.

g. Variations in Material Properties and Assumptions

For a general discussion on the effects of variations in material properties and assumptions, see Section 3.8.1.4.3.d.

h. See attached. ← THIS CHANGE REQUEST

3.8.3.4.3 Expected Behavior Under Load

The methods of analysis and design used for these structures predict the behavior of the as constructed system. The analytical techniques allow for discontinuities, changes in section and materials such that these effects can be allowed for in the structural design. No impairment of

Add new sub-item "h" to Section 3.8.3.4.2 as follows:

h. Analysis of Drywell Personnel Airlock Shield Door Structural Support System

Potential Issue Form (PIF)# 96-0141 documented that analysis performed using design basis accident loading combinations resulted in certain limited components within the structural support system for the drywell personnel airlock shield doors being stressed beyond design basis allowables required elsewhere within USAR Section 3.8 when the shield doors are in an open position during Operational Conditions 1,2 or 3. The shield doors may need to be open for limited time periods during plant startup and shutdown to perform specific activities such as inspections for piping flange leaks. In the closed position, design basis stress allowables are satisfied within the door structural support system.

Subsequent structural analyses have confirmed that the overall design function of the doors with respect to plant nuclear safety has been maintained. More specifically, the shield door structural support system is adequate to ensure that:

1. Safety related systems/components supported by 620'-6" structural steel platform (which also supports the shield door monorail) are not affected in their ability to perform their intended design function.
2. The monorail system supporting the shield doors will preclude falldown of the doors under accident loading conditions.

The analyses supporting the above conclusions utilized certain alternate analytical techniques and acceptance criteria (as compared to pertinent design basis criteria within Sections 3.7 and 3.8) that are only applicable through plant operational cycle 6. The alternate criteria used operability guidance from NRC Generic Letter 91-18, as well as other interim design criteria changes such as the use of modified loading combinations. Refer to letter PY-CEI/NRR-2030L for a more detailed discussion of the alternate design criteria. A long-term resolution to this issue is required to be completed prior to restart from the sixth refueling outage.

PLANT OPERATING CONDITION	SIGNIFICANT EVENT	CYCLES	CONDITION DURATION (1)	TEMPERATURE °F	RELATIVE HUMIDITY (2) @ T FINAL (%)	PRESSURE	GAMMA RADIATION (3)		BETA RADIATION		SUPPLEMENTARY DATA
							MAX DOSE RATE RADS/HR	DOSE INTEGRATED RADS	MAX DOSE RATE RADS/HR	DOSE INTEGRATED RADS	
ZONE CT-1 (HARSH) AREA ABOVE REFUELING FLOOR (CONTINUED FROM B-022-020)											
ACCIDENT	LOCA INSIDE CONTAINMENT	1	1 HR 6 HRS 24 HRS 30 DAYS 100 DAYS 180 DAYS				3.8x10 ^{8}	1.5x10 ^{8} 5.3x10 ^{8} 1.2x10 ^{7} 2.8x10 ^{7} 3.1x10 ^{7} 3.1x10 ^{7}	2.1x10 ^{7}	9.2x10 ^{6} 1.3x10 ^{7} 2.5x10 ^{7} 6.5x10 ^{7} 8.8x10 ^{7} 1.1x10 ^{8}	(8)
ZONE CT-2 (HARSH) AREA ABOVE SUPPRESSION POOL											
NORMAL MAX MIN AVERAGE	NORMAL FULL POWER OPERATION	1	8363 HRS 3485 HRS 336,616 HRS	104 60 91	90 20 50	ATMOSPHERE ATMOSPHERE ATMOSPHERE	2.5x10 $^{-3}$ (14)	8.8x10 ^{2}	NEGLIGIBLE	NEGLIGIBLE	(4) (9)
ABNORMAL MAX MIN	LOSS OF HVAC	1	120 HRS	137	97 20	ATMOSPHERE ATMOSPHERE					(6)
ABNORMAL	SAFETY RELIEF VALVE DISCHARGE	[illegible]	0 MIN 30 MIN 90 MIN 2 HRS 3 HRS 4 HRS 5 HRS 6 HRS 7 HRS 8 HRS 9 HRS 10 HRS 11 HRS 12 HRS 13 HRS 17 HRS	95.0 96.6 103.0 112.7 117.7 119.7 120.0 118.1 116.9 115.9 115.1 114.7 109.2 105.0 101.7 90.0	90 100 100 100 100 100 100 100 100 100 100 100 100 100 100 100	ATMOSPHERE 0.0 PSIG 1.1 PSIG 1.4 PSIG 1.7 PSIG 1.9 PSIG 1.9 PSIG 1.8 PSIG 1.7 PSIG 1.6 PSIG 1.6 PSIG 1.6 PSIG 1.2 PSIG 1.0 PSIG 0.8 PSIG 0.0 PSIG	2.1x10 ^{1}	2.5x10 ^{3}	NEGLIGIBLE	NEGLIGIBLE	
ABNORMAL	TOTAL NON-ACCIDENT INTEGRATED DOSE							2.7x10 ^{5}			SUM OF SRVD TRANSIENTS PLUS NORMAL PLANT OPERATION RADIATION DOSES
ACCIDENT	LOCA INSIDE CONTAINMENT	1	0.0 SEC 1.5 SEC 2.0 SEC 3.0 SEC 5.0 SEC 6.0 SEC 10 SEC 300 SEC 16.7 MIN 30 MIN 80 MIN 3 HRS 18 HRS 10 DAYS 180 DAYS	90 90 92 95 97 98 100 125 150 165 180 184.6 160 120 90	90 100 100 100 100 100 100 100 100 100 100 100 100 100 100	0.0 PSIG 0.3 PSIG 6.6 PSIG 8.8 PSIG 6.8 PSIG 3.8 PSIG 4.3 PSIG 7.3 PSIG 10.3 PSIG 7.3 PSIG 10.3 PSIG 12.0 PSIG 6.3 PSIG 3.8 PSIG 0.0 PSIG					ACCIDENT ENVIRONMENTAL MEDIUM AIR, SPRAY (11) SUBMERGENCE AIR, SPRAY (11) SUBMERGENCE AIR, SPRAY (11) SUBMERGENCE AIR, SPRAY (11) SUBMERGENCE AIR, SPRAY (11) SUBMERGENCE AIR, SPRAY (11) SUBMERGENCE AIR, SPRAY (11) SUBMERGENCE AIR, SPRAY (11) SUBMERGENCE AIR, SPRAY (11) SUBMERGENCE AIR, SPRAY (11) SUBMERGENCE AIR, SPRAY (11) SUBMERGENCE AIR, SPRAY (11) SUBMERGENCE AIR, SPRAY (11) SUBMERGENCE AIR, SPRAY (11) SUBMERGENCE AIR, SPRAY (11) SUBMERGENCE
ACCIDENT	LOCA INSIDE CONTAINMENT	1	1 HR 6 HRS 24 HRS 30 DAYS 100 DAYS 180 DAYS				1.4x10 ^{8}	8.7x10 ^{6} 2.9x10 ^{6} 6.7x10 ^{6} 2.4x10 ^{7} 3.2x10 ^{7} 3.6x10 ^{7}	2.1x10 ^{7}	5.2x10 ^{6} 1.3x10 ^{7} 2.5x10 ^{7} 6.6x10 ^{7} 8.8x10 ^{7} 1.1x10 ^{8}	(8)

THIS CHANGE REQUEST

NOTES:

1. TIME DURATION SHOWN IS FOR EACH OCCURRENCE OF THE LISTED SIGNIFICANT EVENT. FOR NORMAL FULL POWER OPERATION, HOURS REPRESENT THE TIME FOR WHICH THE MAXIMUM, MINIMUM, AND NORMAL WEIGHTED AVERAGE TEMPERATURE OCCUR OVER THE POSTULATED 40 YEAR PLANT LIFE. THE ONE CYCLE LISTED FOR THE NORMAL FULL POWER OPERATION REPRESENTS THIS 40 YEAR DURATION MINUS ANY SIGNIFICANT TRANSIENTS. THE SUM OF THE NORMAL AND THE ABNORMAL DURATIONS SHOULD THEREFORE EQUAL 40 YEARS (350,400 HOURS).
2. CONDITION DURATION DOES NOT APPLY TO PERCENT RELATIVE HUMIDITY. PERCENT RELATIVE HUMIDITY IN RELATION TO SPECIFIED TIME DURATIONS HAS NOT BEEN POSTULATED EXCEPT FOR HIGH ENERGY LINE BREAK TRANSIENTS.
3. INTEGRATED GAMMA RADIATION IS OVER 40 YEARS FOR NORMAL PLANT OPERATION, AND 180 DAYS FOR ACCIDENT CONDITIONS. FOR ACCIDENT CONDITIONS, RADIATION DOSES ARE BASED ON RELEASES INSIDE CONTAINMENT ONLY, UNLESS OTHERWISE NOTED.
4. TEMPERATURES SHOWN OCCUR DURING NORMAL PLANT OPERATING MODES. THESE NORMAL PLANT OPERATING MODES INCLUDED THE FOLLOWING:
   a. NORMAL FULL POWER OPERATION
   b. HOT STANDBY (OCCURS APPROXIMATELY 5.5 TIMES PER YEAR FOR 5 DAYS)
   c. NORMAL SHUTDOWN (OCCURS APPROXIMATELY 2.7 TIMES PER YEAR FOR 10 DAYS)
   d. CONTINUATION OF SHUTDOWN (OCCURS APPROXIMATELY 1 TIME PER YEAR FOR 30 DAYS)
   e. TESTING (DURATIONS VARY)

   MINIMUM TEMPERATURES SHOWN ARE BASED ON SYSTEM OPERATION DURING THE ABOVE NORMAL PLANT OPERATING MODES. MINIMUM TEMPERATURES FOR THE "PLANT OFF", WHEN M-14 PURGE AND VENT SYSTEM OPERATING OR TOTAL SHUTDOWN CONDITION ARE NOT SHOWN. DURING THE PLANT OFF CONDITION OR WHEN M-14 SYSTEM OPERATING THE PLANT HEATING SYSTEMS ARE DESIGNED TO MAINTAIN A MINIMUM TEMPERATURE OF 50° F.
5. DELETED
6. LOSS OF HVAC IN THE CONTEXT OF THESE TABLES MEANS LOSS OF NON-SAFETY HVAC DUE TO A LOSS OF OFFSITE POWER (LOOP). THE TEMPERATURES SHOWN ARE BASED ON AN ASSUMED INITIAL TEMPERATURE EQUAL TO THE NORMAL PLANT OPERATING CONDITON MAXIMUM TEMPERATURE.
7. THIS AREA IS SUBJECT TO WATER SPRAY FROM THE CONTAINMENT SPRAY SYSTEM. THE WATER CHEMISTRY OF THE SPRAY IS AS NOTED IN SECTION 3.11.5.1.2 OF THE FSAR. DURATION OF THE CONTAINMENT SPRAYS IS CONTROLLED ADMINISTRATIVELY, HOWEVER PLANT TECHNICAL SPECIFICATIONS SHALL STATE THAT THE SPRAYS SHALL BE TERMINATED AT A MAXIMUM OF 30 HOURS POST ACCIDENT. SPRAYS MAY BE TURNED ON AGAIN IF REQUIRED TO LIMIT CONTAINMENT AIRSPACE TEMPERATURE 185° F. REACTIVATION OF SPRAYS COULD BE REQUIRED ANY TIME DURING THE POSTULATED 180 DAY ACCIDENT.
8. FOR QUALIFICATION OF EQUIPMENT TO AN ATWS EVENT, A RADIATION ENVIRONMENT EQUIVALENT TO 10% OF THE GIVEN LOCA DOSES MAY BE USED.
9. FOR TYPICAL NORMAL OPERATION RADIATION SOURCES REFER TO FSAR SECTION 3.11.5.2.1.
10. AREAS WITHIN THE ANNULUS WHICH ARE BELOW EL.599'-4" NEED NOT CONSIDER BETA RADIATION. THESE AREAS ARE SHIELDED WITH CONCRETE BY THE CONTAINMENT FIX AS SHOWN ON DWG E-015-012.
11. EQUIPMENT IS SUBJECT TO SUBMERGENCE BETWEEN ELEVATIONS 583'-6" AND 612'-6". FROTH AND SPRAY OCCUR FROM ELEVATIONS 612'-6" TO 623'-4". TIME DURATION IS 5 SECONDS.
12. PORTIONS OF ZONES CT-3 AND CT-4 DIRECTLY BELOW OPEN GRATING ON ZONES CT-7 AND CT-8 ELEVATIONS MAY BE SUBJECT TO RUN-OFF DRIPPING FROM CONTAINMENT SPRAYS.
13. ABOVE CONTAINMENT ELEVATION 730'-0" THE STRATIFIED TEMPERATURES FOLLOWING A RWCU LINE BREAK WILL BE 220° F FROM 0 TO 3 HOURS POST LOCA.

(Rev. 7 3/95)

REFERENCE DRAWINGS:

B-022-001
B-022-020

PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Environmental Conditions for Containment Building

Figure 3.11-21

(Dwg. B-022-021)

THIS CHANGE REQUEST

14. As Reactor power increases, dose rates at the Drywell Personnel Airlock Shield Doors will increase linearly. The gamma/neutron dose rate adjacent to the shield doors at 100% power, with the shield doors open, will be 45mrem/hr. Access to the 599' containment and drywell with the reactor at power and the shield doors open will be controlled in accordance with Health Physics procedures.