3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

b. Model Change

1. Core CCFL pressure differential - 1 psi - Incorporate the assumption that flow from the bypass to lower plenum must overcome a 1 psi pressure drop in core.

2. Incorporate NRC pressure transfer assumption - The assumption used in the SAFE-REFLOOD pressure transfer when the pressure is increasing was changed.

A few of the changes affect the accident calculation irrespective of CCFL. These changes are listed below.

a. Input Change

1. Break Areas - The LBA break area was calculated more accurately.

b. Model Change

1. Improved Radiation and Conduction Calculation - Incorporation of CHASTE 05 for heatup calculation.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Bases Table B 3.2.1-1.

For plant operation with a single recirculation loop, the MAPLHGR limits specified in the CORE OPERATING LIMITS REPORT are multiplied by the smallest of $MAPFAC_f$, $MAPFAC_p$ or ~~0.85 (Reference 2). The constant factor, 0.85,~~ is derived from LOCA analyses initiated from single loop operation to account for earlier boiling transition at the limiting fuel node compared to standard LOCA evaluations.

3/4.2.2 APRM SETPOINTS [DELETED]

the numerical factor specified for single recirculation loop operation in the CORE OPERATING LIMITS REPORT (Reference 2). The numerical factor

CONTAINMENT SYSTEMS

BASES

3/4.6.3 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the drywell and containment pressure will not exceed the design pressure of 30 psig and 15 psig, respectively, during primary system blowdown from full operating pressure.

The suppression pool water volume must absorb the associated decay and structural sensible heat released during a reactor blowdown from 1040 psia. Using conservative parameter inputs, the maximum calculated containment pressure during and following a design basis accident is below the containment design pressure of 15 psig. Similarly the drywell pressure remains below the design pressure of 30 psig. The maximum and minimum water volumes for the suppression pool are 150,230 cubic feet and 146,400 cubic feet, respectively. These values include the water volume of the containment pool, horizontal vents, and weir annulus. Testing in the Mark III Pressure Suppression Test Facility and analysis have assured that the suppression pool temperature will not rise above 185°F for the full range of break sizes.

Should it be necessary to make the suppression pool inoperable, this shall only be done as specified in Specification 3.5.3.

Experimental data indicates that effective steam condensation without excessive load on the containment pool walls will occur with a quencher device and pool temperature below 200°F during relief valve operation. Specifications have been placed on the envelope of reactor operating conditions to assure the bulk pool temperature does not rise above 185°F in compliance with the containment structural design criteria.

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

The containment spray system consists of two 100% capacity trains, each with two spray rings located at different elevations about the inside circumference of the containment. RHR A pump supplies one train and RHR pump B supplies the other. RHR pump C cannot supply the spray system. Dispersion of the flow of water is effected by ~~251 nozzles in each train,~~ enhancing the condensaton of water vapor in the containment volume and preventing overpressurization. Heat rejection is through the RHR heat exchangers. The turbulence caused by the spray system aids in mixing the containment air volume to maintain a homogeneous mixture for H_2 control.

249 nozzles in Train 'A' and 251 nozzles in Train 'B',

3/4.6.5 DRYWELL POST-LOCA VACUUM RELIEF VALVES

Drywell vacuum relief valves are provided on the drywell to pass sufficient quantities of gas from the containment to the drywell to prevent an excess negative pressure from developing in the drywell.

3/4.6.6 SECONDARY CONTAINMENT

The secondary containment completely encloses the primary containment, except for the upper personnel hatch. It consists of the fuel building, gas control boundary, and portions of the auxiliary building enclosed by the extension of the gas control boundary and the ECCS cubicles and areas as described in FSAR Figure 6.2-132. The standby gas treatment system (SGTS) is designed to achieve and maintain a negative 1/4" W.G. pressure within the secondary containment following a design basis accident. This design provides for the capture within the secondary containment of the radioactive releases from the primary containment, and their filtration before release to the atmosphere.

Establishing and maintaining a vacuum in the secondary containment with the standby gas treatment system once per 18 months, along with the surveillance of the doors, hatches, dampers, and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment. The inleakage values are not verified in the surveillances since no credit for dilution was taken in the dose calculation. As noted however, adequate drawdown is verified once per 18 months. The acceptance criteria specified in Figure 4.6.6.1-1 for the drawdown test is based on a computer model, verified by actual performance of drawdown tests, in which the drawdown time determined for accident conditions is adjusted to account for performance of the test during normal plant conditions. The acceptance criteria indicated per Figure 4.6.6.1-1 is based on conditions corresponding to power operation (with the turbine building ventilation system in operation) and wind speeds less than or equal to 10 mph. The acceptance criteria for plant conditions other than those assumed will be adjusted as necessary to reflect the conditions which exist during performance of the surveillance test.

The OPERABILITY of the standby gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Continuous operation of the system with the heaters OPERABLE for 10 hours during each 31-day period is sufficient to reduce the buildup of moisture on the absorbers and HEPA filters. The specified heater dissipation is based on a bus voltage of 460 volts. Heater test results shall be adjusted to account for actual bus voltage.

3/4.6.7 ATMOSPHERE CONTROL

The OPERABILITY of the systems required for the detection and control of hydrogen gas ensures that these systems will be available to maintain the hydrogen concentration within the containment below its flammable limit during post-LOCA

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 SHUTDOWN SERVICE WATER SYSTEM

The OPERABILITY of the shutdown service water system ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

The ultimate heat sink (UHS) specification ensure that sufficient cooling capacity is available for continued operation of safety-related equipment for at leat 30 days to permit safe shutdown and cooldown of the reactor. The surveillance requirements ensure that quantities maintained are consistent with the assumption used in the accident analysis as described in the FSAR and the guidance provided in Regulatory Guide 1.27, January 1976

3/4.7.2 CONTROL ROOM VENTILATION SYSTEM

The OPERABILITY of the control room ventilation system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all design basis accident conditions. Continuous operation of the system with the heaters OPERABLE for 10 hours during each 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix "A", 10 CFR 50. Surveillance testing provides assurance that system and component performances continue to be in accordance with performance specifications for Clinton Unit 1, including applicable parts of ANSI N509-1980.

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

The reactor core isolation cooling (RCIC) system is provided to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without requiring actuation of any of the Emergency Core Cooling System equipment. The RCIC system is conservatively required to be OPERABLE whenever reactor pressure exceeds 150 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring the RCIC system.

The RCIC system specifications are applicable during OPERATIONAL CONDITIONS 1, 2 and 3 when reactor vessel pressure exceeds 150 psig because RCIC is the primary (non-ECCS) source of emergency core cooling when the reactor is pressurized.

With the RCIC system inoperable, adequate core cooling is assured by the OPERABILITY of the HPCS system and justifies the specified 14 day out-of-service period.

The specified heater dissipation is based on a bus voltage of 460 volts. Heater test results shall be adjusted to account for actual bus voltage.