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 1000 psig. Since all of the gases in the drywell are purges into the pressure suppression chamber air space during a loss-of-colant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber maximum allowable 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 54 psig which is below the design pressure of 56 psig. The minimum volume of 58,900 ft^3 results in a submergence of approximately 3 feet. Based on Humboldt Bay, Bodega Bay, and Marviken test facility data as utilized in General Electric Company document number NEDE-21885-P and data presented in Nutech document, Iowa Electric document number 7884-M325-002, the following technical assessment results were arrived at:

1. Condensation effectiveness of the suppression pool can be maintained for both short and long term phases of the Design Basis Accident (DBA), Intermediate Break Accident (IBA), and Small Break Accident (SBA) cases with three feet submergence.

2. There is no significant thermal stratification in the condensation oscillation regime after LOCA with three feet submergence.

3. There is some thermal stratification in the chugging regime for all break sizes. However, this will not inhibit the pressure suppression function of the suppression pool.

4. Seismic induced waves will not cause downcomer vent uncovering with three feet submergence.

5. Post-LOCA pool waves will not cause downcomer vent uncovering with three feet submergence.

6. Maximum post-LOCA drawdown will not cause downcomer vent uncovering and condensation effectiveness of the suppression pool will be maintained.

Therefore, with respect to downcomer submergence, this specification is adequate. The maximum temperature at the end of blowdown tested during the Humbolt Bay and Bodega Bay tests was 170°F and this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

Should it be necessary to drain the suppression chamber, this should only be done when there is no requirement for core standby cooling systems operability as explained in Basis 3.5.F.

Using a 50°F rise (Table 5.2-1, FSAR) in the suppression chamber water temperature and a minimum water volume of 58,900 ft^3, the 170°F temperature which is used for complete condensation would be approached only if the suppression pool temperature is 120°F prior to the DBA-LOCA. Maintaining a pool temperature of 95°F will assure that the 170°F limit is not approached.

2. Inerting

Safety Guide No. 7 assumptions for metal-water reactions result in hydrogen concentrations in excess of the Safety

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Director-Nuclear Generation and to the Chairman of the Safety Committee.

f. Review of those Reportable Occurrences requiring 24 hour notification to the Commission.

g. Review of facility operations to detect potential safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Chairman of the Safety Committee.

i. Review of the Plant Security Plan and implementing procedures.

j. Review of the Emergency Plan and implementing procedures.

6.5.1.7 Authority

The Operations Committee shall:

a. Recommend to the Plant Superintendent-Nuclear written approval or disapproval of items considered under Specification 6.5.1.6 (a) through (d) above.

c. Chemistry and radiochemistry.

d. Metallurgy.

e. Instrumentation and control.

f. Radiological safety.

g. Mechanical and electrical engineering.

h. Quality assurance practices.

i. Non-destructive testing.

j. Administration.

6.5.2.2 Composition

The Safety Committee shall be composed of persons who have been appointed in writing by the President to serve on a permanent basis and who collectively have or have access to applicable technical and experimental expertise in the areas listed in section 6.5.2.1, items a through j.

6.5.2.3 Alternates

All alternate members shall be appointed in writing by the President to serve on a permanent basis.

6.5.2.4 Consultants

Consultants shall be utilized as determined by the Safety Committee Chairman to provide expert advice to the Safety Committee.

6.5.2.5 Meeting Frequency

The Safety Committee shall meet at least once per calendar quarter during the initial year of facility operation following fuel loading and at least once per six months thereafter.

6.5.2.6 Quorum

A quorum of the Safety Committee shall consist of the Chairman or Vice Chairman and at least four members with a maximum of two alternates as voting members. No more than a minority of the voting members shall have line responsibility for operation of the facility.

i. Reports and meeting minutes of the Operations Committee.

6.5.2.8 Audits

Audits of facility activities shall be performed under the cognizance of the Safety Committee. These audits shall encompass:

a. The conformance of facility operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per 24 months.

b. The performance, training and qualifications of the entire facility staff at least once per 24 months.

c. The results of all actions taken to correct deficiencies occuring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per six months.

d. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10CFR50, at least once per 24 months.

e. The Emergency Plan and implementing procedures at least once per 12 months.

f. The Security Plan and implementing procedures at least once per 12 months.

g. Any other area of facility operation considered appropriate by the Safety Committee or the President.

h. Design change request safety evaluations at least once per 24 months.

i. The DAEC Fire Protection Program and implementing procedures at least once per 24 months.

6.5.2.9 Authority

The Safety Committee shall report to and advise the President on those areas of responsibility specified in Specifications 6.5.2.7 and 6.5.2.8.

6.5.2.10 Records

Records of Safety Committee activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each Safety Committee meeting shall be prepared, approved and forwarded to the President within 14 days following each meeting.

b. Reports of reviews encompassed by Specification 6.5.2.7 above, shall be prepared, approved and forwarded to the President within 14 days following completion of the review.

c. Audit reports encompassed by Specification 6.5.2.8 above, shall be forwarded to the President and to the management positions responsible for the areas audited within 30 days after completion of the audit.