1.6 ACRS SPECIAL INTEREST ITEMS

1.6.1 GENERAL

This section describes the status of programs conducted for the investigation of items which were identified by the Advisory Committee on Reactor Safeguards (ACRS) as being of special interest and pertaining to all large water-cooled power reactors.

In carrying out these programs, information derived from research and development activities of the AEC or NRC and other organizations in the nuclear power industry was considered.

1.6.2 QUALITY ASSURANCE

The Baltimore Gas & Electric Company has traditionally retained full responsibility and maintained close control over all aspects of the design and construction of its power plants. This background of experience has been used by BGE in establishing a comprehensive quality assurance program to assure that the Calvert Cliffs Units 1 and 2 are designed, fabricated, and constructed in accordance with the requirements of applicable specifications and codes. The BGE program starts with the initial plant design and is continued through all phases of equipment procurement, fabrication, erection, construction, and plant operation. The program provides for review of specifications to assure that quality control requirements are included and for surveillance and audits of the manufacturing and construction efforts to assure that the specified requirements are met.

A summary description of the Calvert Cliffs quality assurance program is contained in the Quality Assurance Topical Report. This program fully meets the guidelines established by 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants."

1.6.3 FAILED FUEL DETECTION

Early detection of gross failure of fuel elements is important in limiting the consequences of fuel element failure. Early detection permits early application of protective action.

Combustion Engineering, Inc. has evaluated the following instruments for possible application as a failed fuel monitor:

a. Delayed-neutron monitor;

b. Ion exchange iodine monitor;

c. Cerenkov-detector monitor;

d. Gaseous-fission-product detector;

e. Differential gamma monitor;

f. Gross gamma plus specific isotope monitor.

Based on this instrument evaluation and a study of the expected fission and corrosion product activities in the reactor coolant, it has been concluded that the gross gamma plus specific isotope monitor provides a simple and reliable means for early detection of fuel failures.

The design bases of the detection system include the following:

a. Trends in fission product activity in the RCS are used as an indication of fuel element cladding failures. The minimum detectable activity is specified as 10^{-4} µc/cc I^{135};

b. There is a time delay of less than five minutes before the activity, emitted from a fuel-element-cladding failure, is indicated by the instrumentation. This time delay is a function of the location of the monitor;

c. The information obtained from this system is not used for automatic protective or control functions or for detecting the specific fuel assembly (or assemblies) which has failed;

d. The high activity alarm is supplemented with radiochemical analysis of the reactor coolant for fission products to provide positive identification for a fuel element failure.

The location and operation of the detector, designated as the process radiation monitor, is described in Section 9.1.3.

1.6.4 REACTOR VESSEL THERMAL SHOCK

Large quantities of ECC water are available to flood the core region in the event of a major LOCA. The Calvert Cliffs design uses a section of each of the RCS cold legs to conduct the water from the safety injection nozzles to the reactor vessel. This water then flows into the downcomer annulus and into the lower plenum of the reactor vessel before flooding the core itself. Analytical investigations were performed to provide assurance that the resultant cooling of the irradiated inner surface of the thick-walled reactor vessel will not induce or propagate cracks sufficient to cause the reactor vessel to fail.

A detailed analysis of the reactor response to thermal shock was performed. A report describing the results of the investigation, "Thermal Shock Analysis on Reactor Vessels Due to Emergency Core Cooling System Operation," A-68-9-1, March 15, 1968, was prepared and submitted to the AEC as part of Amendment 9 to the Maine Yankee license application (AEC Docket No. 50-309). The conclusion as stated in the report was that the CE reactor vessels are capable of sustaining the thermal shock imposed by ECCS operation without gross failure.

Further work was performed to refine the surface heat transfer coefficient and the brittle fracture model used in the evaluation. Combustion Engineering, Inc. first made an extensive review of temperature-quench data obtained during the heat treatment of several heavy section steel plates. With this background, additional quench tests were planned and conducted to develop experimental heat transfer coefficients to be used in the analysis of this problem. These tests were performed on a plate approximately 2'x2'x1/2" thick and instrumented with eleven thermocouples. The plate was heated to 550°F and quickly lowered into an agitated (turbulent) water bath at 80°F (nearly duplicating the temperature conditions present in the reactor). The temperature of all thermocouples were recorded throughout the cooldown of the plate. Subsequently, the transient temperature data were compared to a heat transfer computer model of the plate to obtain an effective heat transfer coefficient. A detailed report covering this work entitled, "Experimental Determination of Limiting Heat Transfer Coefficients During Quenching of Thick Steel Plates in Water," A-68-10-2, December 13, 1968, was submitted to the AEC and made part of the public record. The report concludes that an effective heat transfer coefficient of 300 Btu/hr-ft^2 F provides a realistic upper limit for thick steel plates quenched in highly agitated room temperature water.

Subsequent development of a more realistic brittle-fracture model involved the development of a finite-element analysis computer program. The finite element method was used to compute the stress near the tip of hypothetical axial and circumferential cracks in the vessel. The stress intensity factor for thermal, pressure and residual stress loadings was computed as a function of crack depth. A detailed report entitled, "Finite Element Analysis of Structural Integrity of a Reactor Pressure Vessel during Emergency

Core Cooling," A-70-19-2, January 1970, has been submitted to the AEC and is part of the public record. Accuracy of the finite element solution was confirmed by comparisons between a boundary collocation solution and the intensity factor as a function of crack depth for an edge cracked tensile specimen. The results confirm the conservatism of the approach used by CE and verify that cracks in the vessel will not grow during the thermal shock transient associated with ECC operation.

1.6.5 BLOWDOWN FORCES ON CORE AND REACTOR COOLANT SYSTEM COMPONENTS

In the event of a large break, the RCS would depressurize rapidly, developing local pressure differences and forces in excess of normal operating loads. The WATERHAMMER computer program was utilized to define the hydraulic transient during the initial subcooled portion of the blowdown; the MODFLASH-2 computer program was used to calculate the pressure variations during the saturated portion of the blowdown. The loadings on the system components were then calculated from the pressure forces so obtained.

The reactor vessel internals were evaluated on the basis of these transient loadings. All critical components were designed to withstand these loads so that the core will be kept in place and that there will be no significant interference with the subsequent cooling of the core. The analysis reported in Section 14.15 shows that the Calvert Cliffs reactor meets these design requirements; in addition, the analysis shows that all of the CEAs except those adjacent to the outlet nozzle nearest the break can be inserted into the core following the accident.

In order to further refine the capability in this area, CE is assessing the possible use of third generation computer programs like FLASH-3 and RELAP-3B. The progress of AEC- or NRC-sponsored experimental programs like the LOFT program is being monitored, and the information derived is being used to confirm the adequacy of the analytical techniques currently in use and under development.

1.6.6 EFFECT OF FUEL ROD FAILURE ON ECCS PERFORMANCE

Experimental results have indicated that the core conditions after a LOCA (high internal gas pressure and increasing clad temperature which decreases clad tensile strength) can induce deformation of the fuel cladding in the time interval between blowdown and refill. The deformation takes the form of localized swelling of a fuel rod continuing until the clad perforates or until the temperature transient is reversed. Analytical and experimental programs were undertaken by CE to provide assurance that this deformation will not significantly affect the ability of the ECCS to prevent fuel melting.

The analytical work program established the conditions within the core during the transient for various break sizes. The parameters which determine the extent of fuel deformation are the internal gas pressure and the clad temperature transient (the temperature, the rate of change and the duration). The experimental program was designed to establish the correlation between the variable (internal gas pressure and the temperature transient) and the extent of clad deformation and perforation.

The testing indicated the degree of clad deformation which may take place before perforation, as a function of clad heating rate and internal pressure. The clad swelling was observed to be a localized phenomenon, and the clad perforation was observed as a longitudinal split less than 1/2" long and 3/16" wide.