1.5 RESEARCH AND DEVELOPMENT REQUIREMENTS

1.5.1 INTRODUCTION

The design of the Calvert Cliffs Nuclear Plant is based upon concepts which have been successfully applied in the design of pressurized water reactor power plants. However, certain programs of theoretical analysis or experimentation (constituting "research and development" as defined in the Atomic Energy Act, as amended, and in AEC or NRC regulations) have been undertaken to aid in plant design and to verify the performance characteristics of plant components and systems. This section describes the results and status of those analytical and test programs which were conducted or in progress at the time of operating license application including experimental production and testing of models, devices, equipment and materials. No attempt is made to include all pertinent research and development programs conducted since operating license application.

In carrying out these programs, information which is derived from research and development activities of the AEC or NRC and other organization in the nuclear industry has been taken into account.

1.5.2 CEA TESTING

1.5.2.1 Critical Experiments

An experimental program was completed to confirm techniques for calculating CEA worth and local nuclear peaking associated with the fuel assembly design. The work was performed in the CRX facility of the Westinghouse Reactor Evaluation Center at Waltz Mill, Pennsylvania, between June and August 1967. The basic core configuration was a 30x30 square array of Zr-4 clad UO_2 fuel rods with an enrichment of about 2.7 wt% U-235; fuel rods were removed to create internal water holes or channels to accommodate absorber elements.

The experiments demonstrated that the current CE methods of analysis (Section 3.0) accurately predict the CEA worth and local peaking in the small critical assemblies. This lends support to the use of these methods in the design of the Calvert Cliffs core.

The significant conclusions which were drawn on the basis of the test program include the following:

a. The standard CE design methods are capable of calculating clean, room temperature, critical lattices (lattices which contain no CEAs, water slots, or other heterogeneities) to an accuracy of within 0.03% reactivity on the average;

b. The worths of various arrays of cylindrical absorbers containing boron carbide were predicted within 2% of the worth on the average, with errors for individual cases ranging from +6% to -2.2% of the worth. The arrays had worths ranging from 6 to 8% reactivity;

c. In assemblies containing water holes, the calculated and measured power peaking agree within 2%. Occasional differences between calculated and measured power of up to 4% are seen, but only in fuel rods of low power, usually near the reflector of these highly buckled, small cores.

1.5.2.2 Mechanical Testing

A series of tests were completed on single and dual CEAs to satisfy the following objectives:

a. To determine the mechanical and functional feasibility of the CEA concept;

b. To experimentally determine the relationship between CEA drop time and CEA drop weight, annular clearance between CEA fingers and guide tubes, and coolant flowrate within the guide tube;

c. To experimentally determine the relationship between flowrate and pressure drop within the guide tube as a function of CEA axial position and of finger-to-guide tube clearance;

d. To determine the effects on drop time of adding a flow restriction or of plugging the lower end of a guide tube (as might occur under accident conditions);

e. To determine the effects of misalignment within the CEA guide tube system on drop time.

Both types of CEAs were tested by operation for over 1000 hours at reactor operating temperature, pressure, and water chemistry with flows in excess of those anticipated in the reactor. This test and other tests demonstrated that the five-finger CEA concept is mechanically and functionally feasible and that the CEA meets criteria established for drop time under the most adverse condition. Both types of CEAs were examined after conclusion of these tests and no significant wear was observed. At no point were there any wear marks in excess of .001" in depth. The testing also verified that the analytical model used for predicting drop time gives uniformly conservative results.

The effects on drop time of all possible combinations of frictional restraining forces in the CEDM, angular and radial misalignment of the CEDM, misalignments between the CEA and the guide tubes of as much as 0.4", misalignments of the CEA was experimentally investigated and defined. The conditions tested simulated all the effects of tolerance buildup, dynamic loadings, and thermal effects. The tests demonstrated that misalignments and distortions in excess of those expected from tolerance buildup or any other anticipated cause would still result in acceptable drop times.

The results of the cold CEA testing were analytically extended into the hot operating range. Further testing of a design verification nature was performed scheduled on a complete CEDM-CEA system in the hot test facility at operating plant temperature, pressure, and flow.

1.5.3 CONTROL ELEMENT DRIVE MECHANISM TESTING

The development of the magnetic jack CEDM was carried out over a period of approximately three years. As such, it was a closely coupled program consisting of design, testing and fabrication. The early design effort resulted in a prototype which was tested extensively and was modified during the testing process to improve operation. The results of this early testing led to the design and fabrication of a second improved prototype mechanism. This mechanism was again tested extensively in a manner similar to the first, and from this series of tests coupled with design modifications that Maine Yankee prototype mechanism resulted. The specifications for the Calvert Cliffs CEDM was similar to that used for the construction of the Maine Yankee mechanisms.

The following design requirements were imposed on the CEDMs with regard to operation under severe service conditions:

a. The pressure housing of the CEDM, for the replacement RVCH, is designed for service as a Class 1 appurtenance and code stamped for service at 2,500 psia and 650°F. Normal CEDM operating conditions are 2,250 psia and 608°F.

b. The CEDMs are designed to withstand the combined mechanical loads associated with normal and abnormal operating pressures, temperatures and transients, plus the maximum earthquake, with no loss of function.

c. When DBE reactor coolant pipe rupture loads are added to the loads given in (b) above, the CEDM need not function normally during the application of the DBE loads, but must prevent ejection of its CEA from the core and be capable of normal operation following the DBE.

d. The CEDM is capable of performing its normal function after an inactive period of one month with the CEDM in the hold mode and the plant at normal pressure and temperature.

e. Under the ambient conditions inside the containment building following a postulated main pipe rupture (DBE), the CEDM is capable of driving the CEA to fully inserted position from the fully withdrawn position, and transmitting all position indication signals for 15 minutes after rupture occurs.

f. The CEDM is capable of withstanding complete loss of cooling service for a four-hour period with the plant at normal operating temperature and pressure. CEDM operation under these conditions is restricted to the "HOLD" and "SCRAM" modes. Upon restoration of cooling service the CEDM is capable of normal operation.

The CEDM must be capable of passing the following tests:

a. Single CEA - Accelerated life tests of at least 30,000' of travel and 200 full-height gravity drops at simulated reactor operating conditions and ambient external conditions.

b. Dual CEA - Accelerated life test of at least 15,000' of travel and 200 full-height gravity drop tests at simulated reactor operating conditions and ambient external conditions.

After installation and prior to operation, each CEDM was tested in the field to ascertain that the system, as constructed, meets all of the design requirements, as discussed in Section 13.1.

1.5.4 FUEL ASSEMBLY DESIGN

1.5.4.1 Prototype Tests

Full size Zircaloy fuel assemblies filled with depleted UO_2 have undergone high temperature (600°F), high pressure (2,250 psig) full flow tests in typical reactor chemistry coolant. These tests were performed in connection with a full-size CEA and a CEDM.

Full-size prototype fuel assemblies loaded with depleted UO_2 were subjected to mechanical testing to evaluate their reaction to applied loads. Axial and lateral loading of assemblies supported in air between simulated upper and lower support plates, as well as free end twisting and lateral motion typical of refueling operation, were performed. Previous testing of a similar nature on CE fuel assemblies had shown them to be stable and mechanically sound for all expected reactor operating, casualty and refueling conditions. These tests were repeated for the Calvert Cliffs type fuel assemblies to verify that no significant mechanical characteristics changed.

In 1966, a series of single-phase tests on coolant turbulent mixing was run on a "prototype" fuel assembly which was geometrically similar to the Palisades assembly. The model enabled determination of flow resistances and vertical subchannel flow rates using pressure instrumentation and the average level of eddy flow using dye-injection and sampling equipment. The tests yielded the value of inverse Peclet number characteristic of eddy flow (0.00366). The value was shown during the course of the tests to be insensitive to coolant temperature and to vertical coolant mass velocity. The design value of the inverse Peclet Number was established at 0.0035 on the basis of the experimental results.

As part of a CE-sponsored research and development program, a series of single phase dye injection mixing tests were conducted in 1968. The tests were performed on a model of a portion of a CEA-type fuel assembly which was sufficiently instrumented to enable measurement (via a data reduction computer program) of the individual lateral flows across the boundaries of twelve subchannels of the model. Although these tests were not intended for that purpose, some of the test results could be used to determine the average level of turbulent mixing in the reference design assembly. The inverse Peclet number calculated from the average of 56 individual turbulent mixing flows (two for each subchannel boundary) obtained from the applicable data was 0.0034. With respect to general turbulent mixing, therefore, the more recent study on the CEA assembly verifies the constancy of the inverse Peclet number for moderately different fuel assembly geometries and confirms the design value of that characteristic.

1.5.4.2 Assembly Flow Distribution Tests

Velocity and static pressure measurements were made in an oversize model of a CEA fuel assembly in order to determine the flow distributions present in that geometry. The effect of the distributions on thermal behavior and margin were evaluated, where necessary, with the use of CE's CORAL code, which is an extensively revised version of the COBRA thermal and hydraulic code. Subjects investigated include the following:

a. Assembly inlet flow distribution, as affected by the core support plate and lower end fitting flow hole geometry. The flow distribution was measured and indicated that the desirable uniform condition is achieved within 10% of core height. The effect of the initial non-uniform condition on thermal behavior was analyzed;

b. Assembly inlet flow distribution as affected by a blocked core support plate flow hole. The flow distribution was measured and indicated that flow has recovered to at least 50% of the uniform nominal value at an elevation corresponding to 10% of core height. The effect of the non-uniform flow pattern on thermal behavior was analyzed;

c. Flow distribution within the assembly, as affected by complete blockage of one to nine subchannels. The flow distributions were measured and indicated very little upstream effect of such blockage, followed by recovery to normal subchannel flow conditions within 10 to 15% of core height, depending upon the number of subchannels blocked;

d. Flow distribution below the upper end fitting as affected by the upper end fitting and fuel bundle alignment plate flow hole geometry and by the presence of the CEA shroud. Measurement of the flow pattern in the absence of the shroud showed no appreciable upstream effect of the flow holes in the active core region.

1.5.4.3 DNB Testing on the Mark V CEA Fuel Assembly, 1969-1970

In 1968, CE initiated a series of tests at Columbia University on the departure from nucleate boiling (DNB) phenomenon. One purpose of the tests was to obtain experimental DNB data for verifying the combined accuracy of the thermal and hydraulic COSMO design code and the empirical W-3 DNB correlation in predicting the DNB condition for the CEA fuel assembly.

The tests were conducted on a nine-foot long exact scale portion of a Mark Volt CEA fuel assembly, consisting of one guide tube and 21 electrically-heated "fuel" rods arranged five-by-five. There were three distinct test sections, one with a 7' heated length and a uniform lateral power distribution, and one with a 4' heated length and a non-uniform lateral power distribution. The axial power distribution was uniform for all test sections. Test conditions comprised a coolant inlet temperature range of 450°F to 650°F, a mass velocity range of $1x10^6$ to $3x10^6$ lb/hr-ft^2 and a system pressure range of 1,500 to 2,200 psia.

Approximately 90 data points were obtained from all three test sections. The COSMO/W-3 combination was used for predicting the corresponding Critical Heat Flux values for the experimental conditions. The measure of the accuracy of prediction was defined as the average value of the ratio of experimental to predicted Critical Heat Flux. The value was 0.983 with a sample standard deviation of 0.58; these compare satisfactorily with corresponding values in the literature. The result implies that COSMO and W-3 are acceptable by present standards for describing DNB in the CEA geometry.

The remainder of the DNB analytical and experimental program was devoted in part to further aspects of predicting DNB for the reference design assembly. This program was comprised of:

a. Refinement of the W-3 correlation or development of a new correlation to reduce the statistical error attendant on the prediction;

b. Investigation of the case of the small systematic deviation;

c. Investigation of DNB behavior over a wider range of system pressure and flow conditions.

1.5.4.4 Dynamic Loop and Vibration Testing

Considerable testing was performed to evaluate the effects of assembly and fuel rod vibration or fretting. Dynamic loop testing under simulated reactor operating conditions and mechanically-induced autoclave vibration tests were carried out.

Over 18,000 hours of test time were accumulated on subsize assemblies and over 14,000 hours on full-size test assemblies in dynamic test loops. Test conditions duplicated reactor temperature, water chemistry, pressure, and flow velocity. Intentional cross flow and forced bundle vibrations were used to accentuate any vibration between the fuel rods and the spacer grids. In addition, the spacer grid spring tabs were individually set to simulate relaxed spring conditions.

In addition to the dynamic loop tests, forced vibration tests were also performed. Fuel rods supported by spacer grids were vibrated at various frequencies and amplitudes. The tests were conducted in a static autoclave at operating pressure and temperature. Test variables included, in addition to the vibration frequency and amplitude, spring preset of the spacer grids, and time under test. The spring tabs were varied from design interference fits to gaps. These tests did not reproduce reactor flow conditions but were designed to develop trends in the degree of fretting as a function of the test variables.

Even under unreasonably severe conditions (i.e., high frequencies, large amplitudes, and gaps between the grid spring and fuel tube) no serious fretting was observed in these tests.

1.5.5 MODERATOR TEMPERATURE COEFFICIENT

Analytical studies were completed as part of the detailed plant design to define the least negative MTC for the Calvert Cliffs reactor. The factors which affect the MTC are discussed in Section 3.0 of this report.

Analyses of MTC for the Connecticut Yankee reactor compared with measurements made during the course of the start-up experiments are shown Table 1-2. It will be observed from the data that the measured coefficient is at most $0.16x10^{-4}$ $\Delta\rho/°F$ more positive than the calculated value. This good agreement lends confidence in the ability of the methods used to predict MTCs.

1.5.6 FUEL ROD CLADDING

A substantial amount of information was generated in the course of CE's continuing test program on Zr-4 cladding.

Creep collapse tests on unsupported Zr-4 specimens with t/OD ratios of between 0.050 and 0.071 were performed at 650°F and 750°F. All tests were performed at an external pressure of 2400 psia. Results of tests performed at 750°F show that specimens with a t/OD of 0.059 (the reference design of Calvert Cliffs) collapse between 100 and 1000 hours at this temperature. Tests conducted at 650°F show collapse will occur between 5000 and approximately 30,000 hours.

Zircaloy-4 specimens supported with plenum springs and mandrels with machined defects to simulate chipping and separated pellets were creep collapse tested at 750°F. Zircaloy-4 specimens with plenum springs were creep collapse tested at 650°F. Tests at 650°F after 19,000 hours showed no indications of the cladding deforming into the spaces between the springs. Specimens tested at 750°F accumulated in excess of 10,000 hours and the cladding showed some deformation into the intentional defects. Grooves 3/16" wide representing separated pellets caused a maximum of 2.0 mils deformation of the clad into the groove. Grooves 1/16" wide showed no measurable deformation after 10,000 hours at 750°F.

Long-term corrosion tests were performed on Zr-4 fuel cladding under simulated reactor coolant conditions. Results of weight gain and hydrogen pickup were evaluated with respect to results obtained using demineralized water. The tests were conducted in coolant which had approximately 1100 ppm boron and less than 10 ppm NH_4OH added for pH control. Results after 12,000 hours of test at 650°F showed no effect of the coolant additives on corrosion rates or hydrogen pickup over similar tests conducted in demineralized water. Specimens tested included as-received Zr-4 tubing and 750°F steam autoclaved samples. Similar tests performed using LiOH as a pH control additive accumulated in excess of 8000 hours. Under the test conditions, the corrosion characteristics exhibited by the material were equivalent to those observed in demineralized water.

Dynamic corrosion tests were also conducted at 600°F to determine the effect of contamination on non-autoclaved material corrosion rates. Samples intentionally contaminated with dilute acid and machine oils were tested. Results after 4,300 hours of tests showed no deleterious effect of these conditions on the corrosion behavior of the Zr-4. Samples tested included as-received and 750°F steam autoclaved Zr-4 tubing.

In addition to the corrosion studies mentioned above, CE participated in two studies being conducted by American Society for Testing and Materials (ASTM).

The first study, "Corrosion Testing Zr-4 and its Effect on Hydrogen Absorption," was performed by the ASTM G Committee (Corrosion of Metals), Subcommittee No. VIII (Corrosion of Zirconium in Water Systems).

The second study, "Task Force on Hydride Orientation," was performed by the ASTM B10 Committee (Reactive and Refractory Metals and Alloys), Subcommittee No. II (Zirconium and Hafnium).

Areas of concern to the committee included methods of hydriding and effect of fuel tubing fabrication on platelet orientation.

Calvert Cliffs began phasing in the Westinghouse ZIRLO cladding starting with Unit 1 Cycle 16 (Batch 1V). The AREVA M5® cladding is first used beginning with Unit 2 Cycle 19 (Batch 2Z) and Unit 1 Cycle 21 (Batch AB). AREVA changed to Framatome in January 2018.

1.5.7 REACTOR VESSEL FLOW TESTS

Tests were conducted with one-fifth scale models of CE reactors to determine hydraulic performance. The first tests were performed for the Palisades plant which has a RCS similar to that of Calvert Cliffs. The test investigated flow distribution, pressure drop and the tracing of flow paths within the vessel for all four pumps operating and various part-loop configurations. Air was used as the test medium.

Similar one-fifth scale model tests were performed for Maine Yankee, which has a core similar to that of Calvert Cliffs. These tests were conducted in a cold water loop. All components for the model were geometrically similar to those in the reactor except for the core where 217 cylindrical core tubes were substituted for the fuel bundles. The core tubes contained orifices to provide the proper axial flow resistance.

Combustion Engineering, Inc. also conducted tests on a one-fourth scale model of the Fort Calhoun reactor using air as the test medium.

Flow characteristics for Calvert Cliffs were determined by taking into consideration similarities between Calvert Cliffs and other CE reactors, in conjunction with the experimental data from the flow model programs.

1.5.8 INCORE INSTRUMENTATION TESTS

Tests on incore thermocouples and flux detectors were performed to insure that the instrumentation will perform as expected at the temperature to be encountered and that it does not excessively vibrate and cause excessive wear or fretting. Cold flow testing was completed; no adverse vibrations or wear effects were encountered. Hot flow testing was also completed; after 2,000 hours at 590°F and 2,100 psig in a test loop, no breach of mechanical integrity was observed.

Mechanical tests of the insertion and removal equipment and instrumentation were performed to determine the necessary forms and procedures. The top entry incore instrumentation design provides a means of eliminating the need for handling instrument assemblies separately, thus minimizing down-time and personnel exposure. A full scale mock-up was built to accommodate three incore instrumentation thimble assemblies. Major components and subassemblies of the mock-up included:

a. An incore instrumentation test assembly, including the upper guide structure support plate, three thimble guide sleeves, fuel alignment plate, three fuel bundle guide tubes, and the core support plate.

b. A thimble assembly consisting of the instrument plate, three incore instrumentation thimbles and the lifting sling.

c. An upper guide tube, with the guide tube attached to the thimble extension and the detector cable partially inserted in the guide tube.

Insertion and withdrawal tests were performed to determine the frictional forces of a multi-tube instrument thimble assembly during insertion and withdrawal from a set of fuel bundles. This test simulated the operation that will be performed during the refueling of the reactor. To determine whether jamming of the thimbles would occur during this operation, bending loads were applied to the thimble assembly by tilting the instrument plate 0.5° increments up to a total of 5° from horizontal. Guide tubes were filled with water. The assembly was raised and lowered approximately five times for each tilt setting. Results showed no discernible difference in the friction forces for the various tilt settings, however, the friction forces varied during withdrawal and insertion, reaching a maximum value of 8 lbs. The tests demonstrated that the repeated insertion and withdrawal of incore instrumentation thimble assemblies into the fuel bundle guides can be accomplished with reasonable insertion forces.

Life cycle tests were performed to determine if the frictional forces increase as a result of 40 insertions and withdrawals. An automatic timer was installed in the crane electrical circuitry to automatically cycle the thimble assembly between the fully inserted and withdrawn position. The instrument plate was set for 5° tilt and the assembly was cycled 60 times. The insertion and withdrawal forces were measured during the first and last five cycles. No discernible difference was noticed.

An off-center lift test was performed to determine if the thimble assembly can be withdrawn from the core region while lifting the assembly from an extreme off-center position. For a lifting point 11" off-center, insertion was accomplished without incident. The flexibility of the thimble is such that jamming of the assembly due to off-center lifting does not occur.

Cable insertion tests were performed to determine forces required to completely insert and withdraw a detector cable from the incore instrumentation thimble assembly. The guide tube routing included several 5" radius bends. The detector cable was passed through the wet guide tubing and into the thimble. For 540° of 5" radius bends, an insertion force of 15 lbs and a withdrawal force of 37 lbs was required. This force is reasonable for hand insertion.

1.5.9 MATERIALS IRRADIATION SURVEILLANCE

Surveillance specimens of the reactor vessel shell section material are installed on the inside wall of the vessel to monitor the Nil Ductility Transition Temperature (NDTT) of the material during reactor operating lifetime. Details of the program are given in Section 4.1.5.

TABLE 1-2

ANALYSIS OF MODERATOR TEMPERATURE COEFFICIENTS IN THE CONNECTICUT YANKEE REACTOR AT START-OF-LIFE

REACTOR TEMPERATURE (°F)	DISSOLVED BORON CONCENTRATION (ppm)	ROD WORTH INSERTED (% $\Delta\rho$)	MODERATOR TEMPERATURE COEFFICIENT (10^{-4}/°F)	
			CALCULATED	MEASURED
260	2040	-	0.46	0.57
560	2305	-	0.84	1.00
551	2045	1.8	0.37	0.47
561	1730	4.5	-0.23	-0.25
551	1610	5.6	-0.30	-0.30