SMUD

SACRAMENTO MUNICIPAL UTILITY DISTRICT

RANCHO SECO NUCLEAR GENERATING STATION
UNIT NO. 1

PRELIMINARY SAFETY ANALYSIS REPORT

Volume IV

NOVEMBER 1967

Docket No. 50-312
February 2, 1968

AMENDMENT NO. 1

SACRAMENTO MUNICIPAL UTILITY DISTRICT

RANCHO SECO NUCLEAR GENERATING STATION

UNIT NO. 1

Amendment No. 1 to the Sacramento Municipal Utility District's Preliminary Safety Analysis Report includes both replacement pages and new pages and tabs. All pages to be inserted are identified as Amendment 1. Any technical text material changed by this amendment is coded in the outside margin by a black bar and the numeral one.

Before inserting the Amendment 1 material (contained in this new Volume V) in the different volumes, it is suggested that the Appendix 5 material be removed from Volume IV to provide space. After the Amendment 1 material has been inserted, Appendix 3 should be the first amendment in the new Volume V. The List of Effective Pages should be checked to verify the completeness of Volumes I thru V.

It should be noted that License Application page 4 is replaced with a new page 4 plus two new additional pages, 8 and 9.

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SACRAMENTO MUNICIPAL UTILITY DISTRICT

RANCHO SECO NUCLEAR GENERATING STATION

UNIT NO. 1

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February 2, 1968
Amendment No. 1

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Docket No. 50-312
April 15, 1968

AMENDMENT NO. 2

SACRAMENTO MUNICIPAL UTILITY DISTRICT

RANCHO SECO NUCLEAR GENERATING STATION

UNIT NO. 1

Amendment No. 2 to the Sacramento Municipal Utility District's Preliminary Safety Analysis Report includes both replacement pages and new pages and tabs. All pages to be inserted are identified as Amendment 2, except the reprinted appendices. Any technical text material changed by this amendment is coded in the outside margin by a black bar and the numeral two.

Before inserting the Amendment 2 material in the different volumes, it is suggested that Appendices 2A, 2C, 2D and 2E be removed from Volume IV, discarded and replaced with the new reprinted appendices 2A, 2C, 2D, and 2E. Additionally, remove Appendices 3 and 4 (including tabs) from Volume V and place at the back of Volume IV. The list of Effective Pages should be checked to verify the completeness of Volumes I thru V.

It should be noted that three new additional pages, 10, 11 and 12 are to be added to the License Application.

The response to letter from Peter A. Morris, Director, Division of Reactor Licensing to E. K. Davis, General Counsel, Sacramento Municipal Utility District, dated March 21, 1968, is arranged in the question order of the above letter. For convenience a cross reference of the AEC DRL question number and SMUD response number is presented below. Response to questions are to be inserted into the volumes according to the assigned SMUD number.

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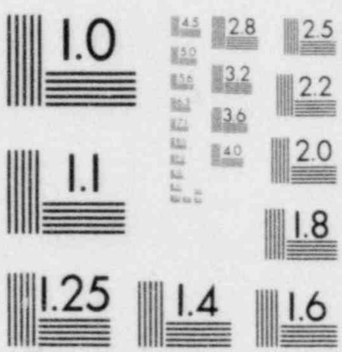
IMAGE EVALUATION
TEST TARGET (MT-3)

6″

MICROCOPY RESOLUTION TEST CHART

IMAGE EVALUATION
TEST TARGET (MT-3)



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MICROCOPY RESOLUTION TEST CHART

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APPENDIX 1

015

1 A ANSWERS TO QUESTIONS

QUESTION 1A.2

Discuss in detail the scope of the following research, development, or test programs including projected completion dates for various phases of the programs and test equipment descriptions. To the extent possible, results of the programs to date should be stated.

1A.2-1 Thermal design, including DNB and flow distribution. (Will loss of a core barrel check valve be simulated in the flow tests?)

1A.2-2 Control rod drives.

1A.2-3 Steam generator including blowdown tests. (Discuss the desirability of insulating or otherwise maintaining the shell at a high temperature to simulate the thermal transient that might be experienced in the actual generator during secondary system blowdown.)

1A.2-4 Core barrel check valves. (Discuss the program for testing the valves or a scaled prototype under operational and accident flow and temperature conditions including vibrational effects during operation and mechanical forces during blowdown.)

1A.2-5 Material tests at high burnup. (Discuss which material properties are critical, the results expected and the manner in which the results will be used. Could significant data of a confirmatory nature be obtained by removing and testing fuel from the reactor environment at intervals in the future. If other test programs, currently in progress, are relied on for fuel rod failure mechanisms, describe the scope and schedule of these tests and compare your requirements in detail.)

ANSWER

Refer to 3.3.2

1A.2-1 <u>Thermal Design</u>

a. <u>Departure from Nucleate Boiling Heat Transfer Investigation</u>

In the late fall of 1961, The Babcock & Wilcox Company began the design and construction of a large heat transfer facility for the purpose of doing DNB testing at power reactor operating conditions. In this facility, which is located at the B&W Research Center, Alliance, Ohio, testing over a wide range of variables covering practically all of the situations one might expect to encounter during normal and expected transient operation of water cooled reactors is possible.

QUESTION 1A.1

What is your criterion for a minimum shutdown margin during operational transients?

ANSWER Refer to 1.4.29

Minimum Shutdown Margin - Operational Transients

The reactor is designed to meet the criterion that it can be shut down to the hot subcritical condition with a margin of at least 1% $\Delta k/k$ with one control rod stuck out of the core. The evaluation of operational transients - such as moderator dilution without rod motion, loss of pumping power, and rod withdrawal - has shown that this margin is not changed by these transients, because the reactor returns to the hot subcritical condition at the end of the transient. This margin at the hot shutdown condition also provides sufficient shutdown reactivity to keep the reactor subcritical in accident-induced transients which cool the reactor coolant to lower temperatures, such as a steam line failure.

cold walls, inter-channel mixing, and instabilities. Data from these assemblies are still being analyzed, and work is progressing on a new DNB correlation. Results to date indicate that the analytical methods used in the design of the reactor core are conservative and that no critical areas exist.

The Babcock & Wilcox Company intends to maintain an active and aggressive program in the field of DNB heat transfer. Some of the principal programs which will be conducted in the near future are outlined below along with a tentative schedule:

(1) A 9-rod assembly with the same rod diameter, pitch spacing and spacer grid used in the fuel assembly will be tested. A nonuniform axial power generation profile will be employed over six feet of the bundle length. The power profile will be representative of that portion of the core experiencing the most severe heat transfer conditions and the most probable location of a DNB. Testing of this assembly is scheduled for the first two quarters of 1968.

(2) An annular specimen with nonuniform power distribution on the outside tube will be tested for additional verification of the effects of length and nonuniform power generation. Power may be supplied to various portions of the specimen so that length effects up to the full 12-foot long test region of the specimen may be examined. Testing for the annular specimen is expected to begin in the fourth quarter of 1968.

(3) A 9-rod bundle test employing nonuniform radial power distribution will be tested in 1969. A definitive program and schedule for this series of tests is not formulated.

(4) Depending upon the results obtained from the previous tests, additional tests will be devised as part of the continuing basic heat transfer and core optimization program. Tests under consideration are for additional radial and axial power distributions, larger test assemblies, investigation of different grid designs, and transient simulation.

b. Mixing Studies

Related to the studies for DNB are additional programs conducted to determine the degree of mixing in the fuel rod channels. Flow tests involving a 4-rod assembly have been conducted to determine mixing effects. Flow tests on a mockup of the outer two rows of fuel rods and the can panels

The facility is supplied with 1.8 megawatts of d-c power and a fully automated data acquisition system. It can be operated within the following limits:

Pressure - 100 to 2,700 psia

Inlet subcooling - 20 to 250 F

Mass velocity - 0.2×10^6 to 3.5×10^6 $lbs/hr\text{-}ft^2$ in a 9-rod assembly.

Present specimen size - 9-rod assembly with a heated length of six feet.

DNB detection with thermocouples (resistance measurement back-up)

Flow - 150 gpm at 295 ft head

Since the loop has been completed, a variety of experiments have been performed to gain better understanding of the DNB phenomema and to develop empirical relationships necessary for the design of water reactors. Among the experiments completed to date have been tests on:

(1) Single tubular specimens with both uniform and nonuniform power distributions. Nonuniform axial peak-to-average powers as high as 1.9, simulating inlet and outlet peak locations, have been included in the tests. These tests were conducted as a function of shape, length, and system parameters. On the basis of these tests, power shape factors for application of the test results to reactor design have been determined, and it was concluded that simulation of reactor axial power shapes could be achieved with confidence in test bundles of shorter length than actual reactor fuel assemblies.

(2) Annular specimens with various combinations of inner and outer wall heat generation and nonuniform axial power distributions were also tested. It was determined that the results obtained with the annular data correlated very well with data taken on bundles. Analytical work done on the tubular and annular specimens has formed the basis for the bundle size and power generation shape to be used in a future test bundle described below.

(3) A 9-rod test assembly with a uniformly heated length of six feet, simulating the reactor fuel rod diameter, pitch spacing, and spacer grid details, has been tested. This was the first test approaching actual geometrical conditions as well as operating conditions for the core. Of principal interest were the effects of spacer grids,

top and bottom of each assembly. Pressure sensors and thermocouples are provided in all fuel assemblies to determine the flow distribution at the core inlet. Additional pressure sensors and thermocouples are provided in other portions of the vessel and core so that overall mixing and pressure drop determinations may be made.

Preliminary investigations in the model and the analysis described in the appendix to Section 3.2.4 herein (Item 5.1.5) indicate that it will not be necessary to simulate the internal check valve construction, operation, or any malfunctions in the vessel model flow tests.

Testing should be completed in mid-1968.

Refer to 3.3.3

1A.2-2 Control Rod Drives

a. Component Tests

The purpose of this program is to seek out potential material and/or design problems prior to production unit testing. The component test program consists of:

(1) Evaluation of various grades of Graphitar-bearing materials in an autoclave at 1,600 psi, 600 F, and water chemistry with 13,000 ppm H_3BO_3. The bearing materials are statically loaded against a 17-4 PH shaft such that the developed stress is greater than will be present in the actual control rod drive.

(2) Environmental dynamic gear and bearing tests under loads equivalent to the control rod drive operating conditions. In this test, the bevel gears, the pinion, and the bearings supporting these gears are being tested in an autoclave at 2,250 psi, 400 F, and reactor water chemistry. The purpose of this test is to obtain wear characteristics of the gear material combinations and projected life of the bearings.

(3) Simulated drive test. A complete mechanism, which simulates the drive with the exception of its overall length, is being tested under no-flow reactor operating conditions of temperature and pressure in an autoclave. An accelerated wear and life test through a short stroke will be completed in conjunction with the life-testing of the prototype mechanism.

(4) Autoclave testing at reactor operating temperature and pressure of buffer seal, splines, and bearings. In this test, the spline joints of the drive rod assembly are being tested under static, no-load conditions for corrosion.

of two adjacent fuel assemblies have been conducted to determine the friction effects at the perforated wall boundary. These tests have confirmed that the larger flow cells at the periphery of the bundle compensate for higher equivalent friction adequately. These effects are shown numerically in the appendix to Section 3.2.3.2.4j herein. Additional tests to extend the investigation to larger sizes, and more elaborate geometry for the purpose of confirming the analytical model and value for mixing coefficients, are described below.

(1) A 9-rod mixing test assembly, of the same bundle geometry as the DNB bundle described previously, has been constructed to determine the degree of mixing present during the DNB investigations. Testing with this assembly is currently in progress and is expected to be completed in the first two quarters of 1968.

(2) A 16-rod assembly with the simulated juncture of four perforated, fuel assembly cans meeting at the corner is under construction. Testing with this assembly will enable one to determine the degree of mixing which occurs between fuel assemblies, and will give more detailed information on velocity distributions and mixing in the peripheral cells of the fuel assembly than did the 4-rod tests. The current core analysis considers only mixing within a fuel assembly and does not take credit for mixing external to the assembly. It is expected that testing with this assembly will begin in January 1968.

(3) A facility large enough to accept a 64-rod assembly is currently under construction. Tests for this facility are not yet firm, but it is expected that some of the preliminary work for calibration of in-core thermocouples and pressure differential instrumentation will be done in this facility. Initial plans were to construct a low pressure facility large enough to accept a full size, cross section fuel assembly. This has currently been replaced with the 64-rod assembly, and its need will be re-evaluated. Testing in this facility is scheduled for the second quarter of 1968.

c. <u>Vessel Model Flow Tests</u>

A 1/6 scale model of the reactor vessel, the internals, and the reactor coolant piping from the pumps to the reactor vessel is currently being tested at the Research Center. Portions of the reactor vessel and internals are constructed of transparent plastic to facilitate visual observation of flow patterns within the vessel. The reactor core is simulated in the model with individual fuel assemblies constructed of perforated sheet material and calibrated orifices at the

In addition to the testing described in the above reference, secondary system blowdown tests have been carried out, and a reactor coolant (primary) side blowdown is planned when schedule commitments permit. The hot water facility of the Research Center is shared by the Control Rod Drive Tests, the Steam Generator Tests, and other experiments and tests.

Three secondary system blowdown tests have been completed. The results of these tests have demonstrated the integrity of the steam generator under conditions of rapid depressurization and large (greater than 200 F), tube-to-shell temperature differentials.

In addition, the results of these tests are used in the development and verification of analytical models for steam system blowdown analyses.

The construction of the test steam generator (including insulation) is such that the thermal time constant of the shell is lower than that of a full-scale unit. This lower time constant results in more rapid cooling of the shell during steam system blowdown than would occur in a full-size unit.

The primary side blowdown test will provide temperature conditions which simulate a thermal transient greater than that for the full-scale unit secondary blowdown as well as simulation of the thermal transient for primary blowdown.

Refer to 3.3.4

1A.2-4 Core Barrel Vent Valves

The core barrel vent valves will be designed to relieve the pressure generated by steaming in the core following the LOCA so that the core will remain sufficiently covered. The valves will also be designed to withstand the forces resulting from rupture of either a reactor coolant inlet or outlet pipe. Testing of the valves will consist of the following:

a. A full-size valve assembly (seat, locking mechanism, and socket) will be tested at steady-state conditions at the maximum pressure expected to result during the blowdown.

b. Sufficient tests will be conducted at zero pressure to determine the frictional loads and clearances in the hinge assembly, the inertia of the valve cover, and the deflections resulting from impact of the cover so that the valve response to cyclic blowdown forces may be determined analytically.

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(5) Autoclave testing of shortened drive rod assembly under static load conditions. This test is similar to the spline testing with the addition of the bevel gear set at the lower end of the drive rod.

(6) Autoclave testing at reactor temperature and pressure of the bevel gears, bearings, shortened rack, and pinion gear under vibratory loading of the rack to determine the fretting characteristics of the gear train. This test is a static load test.

b. Full Scale Prototype Testing Under No Flow Conditions

This test will be performed in an autoclave permitting full stroking in room temperature water and at reactor operating conditions of temperature, pressure, and water chemistry. The cold tests will be utilized only as an initial checkout of the drive prior to temperature and pressure testing. The control rod will be simulated with a dummy weight. Misalignment will be introduced to note its effect on wear and overall performance of the drive mechanism.

c. Full Scale Prototype Testing at Reactor Operating Conditions of Temperature, Pressure, and Flow

The full life test program as defined in the PSAR will be conducted under this test. A prototype control rod and fuel assembly will be used in order to establish the complete drive train assembly.

Inasmuch as the rack and pinion drive concept described in Section 3.2.4.3.2 herein is somewhat different from the first rack and pinion drive tested at the Research Center, the test program which has been outlined above provides an extension of previous tests to establish verification of drive performance and adequacy. The previous test program verified the basic material selection, the snubber design, and the buffer seal concept for use with a rack and pinion drive.

The components test program (items a and b) is scheduled for completion by the end of December 1967. Hot loop testing (item c) at full flow conditions of the prototype drive will begin in October 1967, and continue until the end of December 1967.

Refer to 4.4 — 1A.2-3 Steam Generator

The basic steam generator test program is discussed in detail in Appendix 4A of the Duke Power Company PSAR (Dockets 50-269, 270, and 287).

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The fuels irradiation program will test fuel specimens at design temperatures and at exposures in excess of those obtained in the fuel rod. The specimens irradiated to the design burnup are scheduled to be completed in mid-1969, well in advance of reactor operation. The program will provide information on the swelling rate of UO_2 as a function of burnup, density, heat rate, and cladding restraint. Fuel specimens will be operated at heat rates up to 21.5 kw/ft, which is in excess of the peak specific power in the core. The burnup will range up to 75,000 MWD/TU. The fuel rods will operate with a cladding surface temperature of 650 F.

A program has been carried out to determine the effects of irradiation on the mechanical properties of Zircaloy-4.[1] Tests were conducted to temperatures as high as 775 F. The summary of results from this program is as follows:

a. The room temperature tensile and yield strengths of Zircaloy-4 increased with total neutron exposure for irradiation temperatures up to 650 F. The rate of increase was greater at lower irradiation temperatures. This increase in strength was accompanied by a decrease in the total and uniform elongations.

b. The room temperature yield and tensile strengths of the specimens irradiated at 775 F were somewhat lower than those of the specimens before irradiation. These changes in properties, however, were not significantly different from those observed in specimens aged out-of-pile for like periods of time.

c. The room temperature uniform elongation values for both annealed and cold-worked material were approximately 2 percent after neutron irradiation at 130 F to an exposure of 4.5×10^{19} nvt (E > 1 Mev).

d. A difference in irradiation behavior was noted for the longitudinal and transverse specimens, particularly after irradiation at 775 F. At this temperature the tensile strength in the transverse direction continued to increase whereas in the longitudinal specimens the strength decreased.

A summary of capsule specimens is given in Table 1A.2-1 and a tentative schedule is presented in Figure 1A.2-1.

The following is a description of the research now in progress at B&W that is related to the current reactor design.

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c. The valve assembly will be pressurized to determine what pressure differential is required to cause the valve to begin to open. A determination of the pressure differential required to open the valve to its maximum open position will be simulated by mechanical means.

d. A valve assembly will be installed and removed remotely in a test stand to judge the adequacy of handling equipment.

Since the temperature differential existing across the valve assembly during normal operation in the reactor is only approximately 55 F, and since the same material is used for the valve seat, socket, and cover, there is no need to conduct tests at elevated temperatures.

The valves are located in a region of relatively low velocity and turbulence, and preliminary analysis indicates that there is insufficient energy in the coolant to cause vibrational problems. Therefore, no testing to prove the vibrational adequacy of the valve is planned.

Testing should be completed by January 1969.

Refer to 3.2.4.2.2

1A.2-5 High Burnup Fuel Tests

The design of fuel rods for pressure cycles and thermal gradients are amendable to analysis, based on out-of-pile properties. In determining the behavior of materials under the influence of accumulated irradiation the properties of interest are uranium dioxide growth rates under restraint by tubular cladding, and the ability of the cladding to absorb strain without failure at reactor operating conditions.

A detailed report of sources of information for the irradiation of clad and fuel has been presented in the PSAR, 3.2.4.2.2 plus references. In addition to the PSAR references, irradiation of fuel assemblies or partial fuel assemblies with Zircaloy-clad UO_2 is in progress in the Saxton and Big Rock Point reactors. These data will demonstrate the behavior of fuel assemblies under the combined effects of irradiation, pressure cycles, thermal gradients, reactor coolant environment, and fuel-clad restraints.

B&W is conducting a program to obtain a better understanding of fuel growth rates and irradiation effects on cladding, the influence of hydrogen on cladding, and fission gas release at high burnup for the specific design burnup projected for peak power regions in the reactor.

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Task III - High Burnup Fuel Irradiation

The primary purpose of the High Burnup Program is to determine the swelling rate of UO_2 as a function of burnup using fuel rods of the same design as the core. In addition to determining the swelling rate, the effect of several other variables including the density, heat rate, and cladding restraint will be investigated.

The program consists of capsules some of which will operate at a heat rate of 18 kw/ft and others at a heat rate of 21-1/2 kw/ft. The pellets, other than U-235 content, will conform to the reactor fuel specifications. The burnup will range from 10,000 to 75,000 MWD/TU with six capsules exceeding 45,000 MWD/TU. The capsules will not operate with an external pressure. However, two different cladding thicknesses, 0.015 and 0.025 in. will be used to vary the restraint offered by the cladding. The fuel rods will operate with a cladding surface temperature of 650 F. The diametral gaps between the pellets and cladding will vary from 4-5 to 7-8 mils, to give smeared densities of about 92.3 and 90.8 percent, respectively. These gaps and smeared densities are consistent with the fuel rod specifications. The insertion date for the first capsule was September 5, 1967.

The tests are oriented toward the determination of the behavior of materials in an irradiation environment and to determine the optimum geometric and material properties for the specific application. The information is essential for advancement of the art, but is not considered critical in the sense that all of the programs must be completed to insure safe operation.

Removal and testing of fuel taken from operating reactors at intervals during operation is not considered necessary. The data on hand, plus programs which are currently under way, should satisfactorily provide the information necessary for assurance of safe operation within the limits required.

REFERENCE

[1] Mechanical Properties of Zircaloy-4 After Irradiation at 130, 650, and 775 F, TP-299, April 1967.

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Material Irradiation Testing Program

The purpose of this program is to determine the effects of irradiation on the core components of a central station power reactor. The program is divided into three tasks:

Task I   Low Burnup Fuels Irradiation

Task II   Zircaloy-4 Irradiation

Task III   High Burnup Fuels Irradiation

Task I - Low Burnup Fuels Irradiation

The primary objective of this task is to investigate the dimensional stability of pellet-type fuel rods when irradiated at current and future PWR operating conditions.

The program consists of capsules, some of which are designed to operate at 21-1/2 kw/ft. The cladding for these capsules will operate at a surface temperature of about 640 F. All of the capsules will be irradiated, when possible, for one complete cycle of the BAWTR. Under normal operation, this will amount to about 25 EFPD and a burnup of about 3,500 to 4,000 MWD/TU.

The irradiation of capsules, initially operated at a heat rate of 25-25.7 kw/ft, has been completed. Some capsules received as much as 609 power cycles at 22.8 to 24.6 kw/ft. Hot cell examination is underway.

Task II - Zircaloy-4 Irradiations

The Zircaloy-4 cladding in the core operates with outside and inside surface temperatures as high as 650 and 800 F, respectively. A program was therefore designed to determine how the mechanical properties of Zircaloy-4 are affected by irradiation at these temperatures.

Longitudinal specimens cut from 0.425-in. diameter Zircaloy-4 tubing are used to determine the properties in the longitudinal direction. Ring specimens and flattened rings conforming to dimensions of the longitudinal specimens are used to determine the properties in the transverse direction. Some of the tensile specimens were charged with 250 to 400 ppm hydrogen prior to irradiation.

Irradiation of the two 300-day capsules is continuing without any operational difficulties. As of June 30, 1967, these capsules had achieved an exposure of 306 EFPD.

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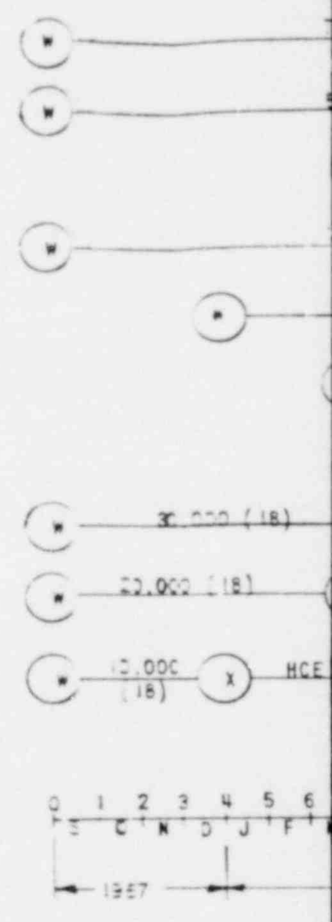
QUESTION 1A.3 We believe that research and development above that which you have indicated will be required to justify the use of core barrel check valves as a solution to the steam bubble problem. Further consideration should be given to testing (1) vibration effects on the valves (caused by core barrel vibrations) and (2) flow characteristics in the reactor after loss of a valve. We believe that if the loss of a valve is not detectable, the DNB ratio at the overpower condition after loss of a valve must be not less than 1.3 (based on the W-3 correlation).

ANSWER Refer to 3.3.4 In order to investigate vibration of the vent valves caused by core barrel vibrations, it was assumed that the core support shield would excite the disc at a frequency where the shield mode shape corresponded to an 8-valve configuration. This frequency is 125 Hz and is substantially below the lowest resonant frequency of the disc, i.e. 1500 Hz. This large difference in frequency indicates that vibratory motions transmitted from the core support shield to the disc will not be amplified by the disc and will not exceed transmitted motions from the shield, which our preliminary analysis indicates will be less than 0.005 inch. Other more rigorous, but more time consuming, analytic methods are being pursued in order to confirm the vibratory motion of the shield. Assuming the worst case of the disc being force-excited at 125 Hz, the amplitude of the disc would have to exceed 0.025 inch in order to develop an inertial force which would exceed the pressure load of 2-1/2 tons (based on 31.5 psi) which acts to keep the valve shut, at full flow. Therefore, it is not possible to transmit sufficient high frequency vibratory power from the coolant stream to cause the shield to vibrate at an amplitude of 0.025 inch. It is concluded that even under the most pessimistic assumptions, excitations from the core support shield cannot cause the valve to open or vibrate. Therefore, it is not necessary to perform a vibration test which would attempt to vibrate the valve by simulating the postulated excitation.

The DNB ratio in the hot channel at the maximum overpower with a vent valve disc off will be high enough to insure that there is a 99 percent confidence that at least 94.5 percent of the population of all such channels are in no jeopardy of experiencing a DNB. This degree of protection is consistent with Paragraphs 3.1.2.3 and 3.2.3.1.1 of the PSAR. It will be demonstrated in the final design that the DNB ratio in the hot channel with the flow resulting from the loss of one vent valve disc will not be less thar 1.3 using the W-3 correlation.

A preliminary sensitivity analysis using postulated worst case parameters has been made for the reduced flow. The results of this analysis are described in the appendix to Section 3.2.4



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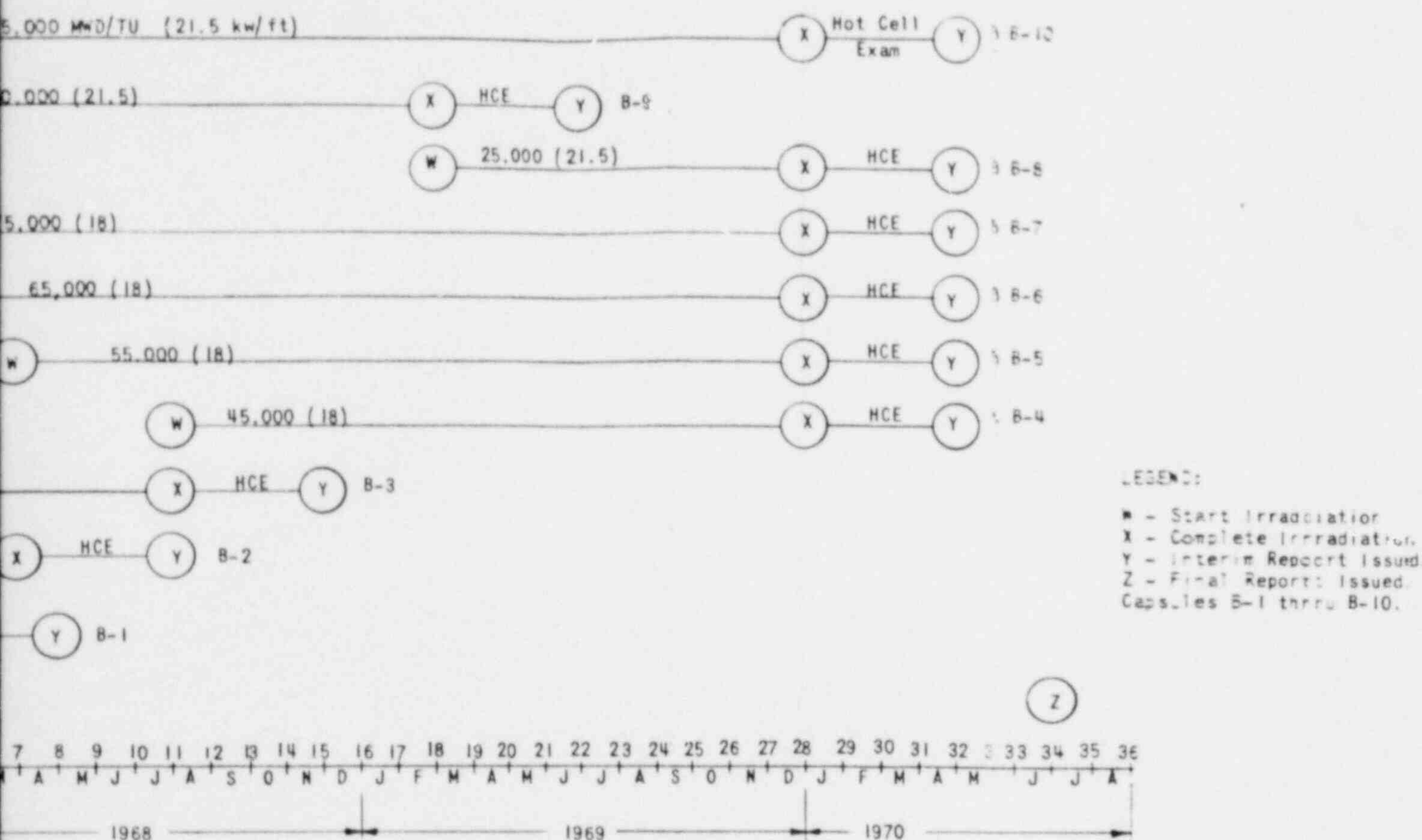

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FIGURE 1A.2-1
HIGH BURNUP IRRADIATION PROGRAM
SCHEDULE (BASED ON BAW-TM-192)

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Amendment 1

herein where the DNB ratios for full and reduced flow are as follows for various reactor powers:

Percent Rated Power	DNBR (Full Flow)	DNBR (Reduced Flow)
100	1.76	1.68
107.5	1.53	1.44
112	1.40	1.30
114	1.34	1.24

The minimum DNB ratio of 1.24 resulting from the analysis at 114 percent power for the postulated worst case is large enought to ensure a DNB ratio of not less than 1.30 for final design conditions. The postulated worst case, used for sensitivity analysis, is not the design condition but a case with heat transfer and mechanical conditions much more severe than expected in the final design. This is demonstrated by a comparison of the nominal and postulated worst case as shown in the appendix to Section 3.2.3.2.4 herein where the W-3 DNB ratios are as follows at rated flow conditions and 114 percent power:

Cell Type	Nominal DNB	Postulated Worst Case DNB
Corner	1.85 (1.71)	1.34 (1.24)
Wall	1.89	1.38
Unit	1.89	1.46

The minimum DNB ratios occurring in the corner cell for the two conditions at reduced flow due to loss of a vent valve disc are shown in parentheses above. The final design DNB will be within the limits of 1.71 to 1.24 shown. It is expected that a value greater than 1.30 will result from final evaluation of the combination of the following significant factors:

(1) Mixing coefficient of 0.03 to 0.07 at design conditions compared to 0.01 used in the preliminary analysis.

(2) Statistical determination of mechanical tolerances in lieu of minimum conceivable dimensions.

(3) A more accurate determination of the hot channel local peaking factor of 1.095 shown in Figure 3.2-55 of the PSAR considering: (a) the statistically determined water gap, and (b) the excess metal in the solid can section surrounding the corner pin. The final value is expected to be about 1.06.

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(4) Application of final vessel and core flow distribution tests results instead of the hot to average fuel assembly flow ratio of 85 percent assumed for the worst postulated case.

(5) The statistical comparison of the multiple rod fuel assembly heat transfer test data with the single channel data that currently forms the basis for the W-3 correlation.

A consideration of the final thermal-hydraulic design compared with the preliminary postulated worst case and the mechanical integrity of the vent valve indicates that it is very unlikely that the core will be subject to an unsatisfactory heat transfer condition.

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QUESTION 1A.4 (DRL 1.1) Update the discussion of your proposed design with respect to its conformance to the Commission's Proposed General Design Criteria. Include in this discussion the impact of the several design changes made in your facility.

ANSWER Response to the Commission's Proposed General Design Criteria including discussion on the impact of the several design changes made are presented in Section 1.4 of the PSAR. Those criteria which reflect changes are 7, 10, 11, 22, 38, 44, 46, 52, 59, 61 and 62.

QUESTION 1A.5 (DRL 1.2) Describe each of your research and development programs with a proposed schedule for obtaining the desired information. Include, as appropriate, when the design of the associated feature must be frozen in order to meet the schedule for construction of the Rancho Seco Plant.

ANSWER Research and development programs that will provide information to complete the final detail design of some of the components or to demonstrate the capability of the design for future operation at a higher power level are summarized in Section 1.5 of the PSAR. Further discussion of research, development or test program is provided in the answer to Question 1A.2 in Appendix 1A of Amendment 1. Additional information and discussion is provided below:

a. Once-Through Steam Generator

Testing necessary to prove the adequacy of the once-through steam generator design for service at the initial power level and to confirm the size and configuration of the units has been completed. These programs were described in Appendix 4A of the Oconee PSAR (Docket Nos. 50-269, 270 and 287) and in the Rancho Seco PSAR, Appendix 1A, Question 1A.2. Steady state and load changing operations using once-through steam generator models have demonstrated the ability of the unit to follow transients and the interaction of the control system with the water level, steam pressure and flows. Primary and secondary blowdown tests on the models have demonstrated the integrity of the units under conditions of rapid depressurization and large tube-to-shell temperature differentials. The results of the blowdown tests are being used in the development and verification of analytical models for steam system blowdown analyses.

b. Control Rod Drive Unit

These programs have been described in Section 3.3.3.4 and Appendix 1A, Question 1A.2 of the PSAR. Some of the results of those programs will be discussed in this reply. The development and testing of the rack and pinion drive is being conducted under three separate programs:

1. Full-scale prototype testing under no-flow conditions.

2. Full-scale prototype testing at reactor operating conditions of temperature, pressure, and flow.

3. Components testing.

The no-flow prototype testing is performed in an autoclave in which the reactor conditions of control rod stroke, temperature, pressure, and water chemistry are duplicated. The tests are performed with a dummy weight equivalent to the weight of the control rod assembly attached to the rack.

The objectives of this testing were to verify the design concept and to obtain a preliminary verification of the trip insertion time.

The mechanism was subjected to approximately 100 full-stroke cycles and 100 trip cycles simulating both hot and cold reactor conditions.

This testing confirmed that the design and the mechanical arrangement met the objectives. The time for 2/3 insertion was less than 1.2 seconds; the snubber design worked properly, and the buffer seal did not impair trip capability.

Further testing was conducted which included a complete life test of full-stroke cycles and trip cycles simulating reactor operating conditions with maximum tolerance misalignment. Examination of components after the test indicated that the wear observed was acceptable on all components except the miter gear which although badly worn continued to operate satisfactorily.

The control rod drive life testing program will be continued after the mechanism has been refurbished and modified to incorporate a new miter gear utilizing 17-4 PH nitrided or Haynes 25 metal.

The second life test will be conducted with different stroking specifications than those used on the first life test.

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Other prototype testing was conducted in another autoclave in which all reactor operating conditions except radiation are duplicated. The complete driveline is established with prototype components, i.e., the fuel assembly, control rod, upper guide tube of the reactor internals, and the drive mechanism.

This testing concentrated mainly on the performance characteristics under coolant flows ranging from zero to full flow at reactor conditions of temperature, pressure, and water chemistry. The objective of these tests was to determine the compatibility of the mechanism trip time with the specification requirements of 1.4 seconds for 2/3 insertion. After some modification of the pattern of flow holes in the shroud of the upper guide tube, the trip time ranged from 1.37 to 1.4 seconds.

Selected components testing was performed prior to and in addition to the life testing programs in order to resolve potential material or design problems. These component test programs produced the following results:

1. Provided the basis for the selection of Graphitar bearing material.

2. Ascertained the buffer seal injection flow rate.

3. Assured acceptable wear from the revised miter gear combination.

4. The corrosion product buildup in the static test of the splines and bearings has not noticeably affected the resistance to rotation of the system.

The program will be completed by August, 1968. By that date the prototype mechanism will have completed the life test program as outlined in the PSAR and all material problems for the production type mechanisms will have been resolved.

c. <u>In-Core Neutron Detectors</u>

This program consists of basic physics parametric studies of the detector and mechanical insertion - withdrawal tests of the assembly. The development program has been outlined in Section 7.3.3.3 of the PSAR. Mechanical testing of the assembly has been completed. All parametric studies have been completed except the long term radiation effects and the depletion effects. Results to date have been satisfactory and the performance of the detectors has been demonstrated. As of April 1968 detectors have been irradiated in the Big Rock Point Reactor for about 34 months and in The Babcock and Wilcox Test Reactor for about 23 months. These lifetime tests are continuing.

d. Core Thermal and Hydraulic Design

These programs have been discussed in Section 3.3.2 and Appendix 1A, Question 1A.2 of the PSAR. The initial core power level has been justified on the basis of the W-3 correlation which has been approved for the design of several similar pressurized water reactors. With the use of that correlation, only the reactor vessel flow model test data is necessary to further substantiate the core thermal and hydraulic design. Test runs already completed without check valves in the internals have demonstrated the ability to provide adequate flow distribution. Tests including check valves will be completed in 1968.

The Departure from Nucleate Boiling (DNB) and mixing studies described in the PSAR are being conducted to support the final thermal design margin on the basis of the B&W correlation and to provide for an increase in its rated power output when that increase is requested.

Due to the fact that the information produced by these programs has been, is being, or will be used to finalize the detail design of components for Oconee and Three Mile Island units which are scheduled to precede the Rancho Seco unit into commercial operation by about two years, the information needed from these programs will be available long before it is needed to freeze design details for the Rancho Seco unit.

QUESTION 1A.6 (DRL 1.3) If not specifically included in 1.2, describe your program including schedule and acceptability criteria for vibration testing of the core barrel check valves.

ANSWER The testing program for the core barrel check valves (internals vent valves) was discussed in Appendix 1A, Question 1A.2. In addition to the testing discussed there B&W is presently working with the valve designer-manufacturer and a vibration testing laboratory on the details of the vibration test of a full scale prototype vent valve. The prototype valve will be mounted in a test fixture which duplicates the method of valve mounting in the core support shield and simulates this local area of the internals. The test fixture with valve installed will be attached to a vibration test machine and excited sinusoidally through a range of frequencies representative of those anticipated for the core support shield during reactor operation. The relative motion between the valve disc and seat will be monitored and recorded during test. The test results will be evaluated and, if required, the valve design will be modified prior to valve production to eliminate any adverse disc vibration problems. All testing will be completed by January 1, 1969.

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QUESTION 1A.7 (DRL 1.4)

If not specifically included in 1.2, discuss the programs currently in progress that will assure fuel element capability for 55,000 MWD/MTU burnup at the design power densities.

ANSWER

A high burnup fuel irradiation test program is in progress at B&W, and is described in Question 1A.2-5, Appendix 1A of the PSAR. A schedule for the program is shown in Figure 1A.2-1. This program includes fuel specimens with representative cladding thickness, fuel-clad gaps and UO_2 densities. Heating rates are representative of maximum heating rates and temperatures in the core. Post-irradiation examination will include profilometer scans to determine permanent clad strain, fission gas release, metallographic examination of fuel and cladding, and confirmation of burnups estimated from flux monitors during the test. Maximum target burnup is 75,000 MDW/MTU. Examination will be made at several stages of burnup between 10,000 and 75,000 MWD/MTU to determine the behavior of the fuel and cladding as a function of burnup.

The damage criteria for the high burnup test program are that the cladding will not allow fission product release, or the entrance of coolant into the fuel rod which could lead to further damage. Other experiments in the industry have shown that the limit of permanent strain in the cladding is approximately 1.5%.[1] This, therefore, represents the current upper limit to avoid damage associated with excessive clad strain. Design limits are set at approximately 1%. (See PSAR 3.1.2.4.2.c).

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The consequence of burnup on fuel rods is that continued fuel growth and fission product release will eventually lead to clad failure due to progressive clad strain. The point of failure is influenced by irradiation-induced changes in the cladding. The program is designed to better understand the limit of burnup and allowable strain which can be achieved without clad failure. The program will also assure fuel element capability for 55,000 MWD/MTU burnup at the design power densities. It will also give a better understanding of the burnup limit, or "margin of safety," for fuel rods of representative design when tested at maximum heating rates.

REFERENCE

[1]Fracture of Cylindrical Fuel Rod Cladding due to Plastic Instability, WAPD-TM-651, April 1967.

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QUESTION 1A.8 (DRL 1.5)

Submit the staffing and training plans for SMUD's Nuclear Project Engineering Staff.

ANSWER

The staffing and training plans for SMUD's Nuclear Project Engineering Staff are presented in Appendix 1C. The program presented will provide the District with a technically qualified engineering staff both during construction and after the plant is operational.

QUESTION 1A.9 (DRL 1.6)

Discuss the principal design decisions yet to be made that require nuclear and steam plant knowledge and which affect nuclear power plant safety. Indicate the approximate dates by which these decisions must be made and to what extent reliance will be placed upon contractors for making decisions. Indicate how the training plans for SMUD personnel are orientated toward these requirements.

ANSWER

The principal design decisions which affect nuclear power plant safety have been made and are presented in the PSAR. However, studies are in progress as defined in Section 1.5 of the PSAR which may affect the final design. SMUD will review the results of these studies, with the aid of consultants if necessary, and initiate any action required to ensure a safe plant.

As a matter of policy, SMUD does not rely on contractors to make decisions on matters of safety. Additionally, SMUD does not rely on contractors to make decisions concerning plant reliability, maintainability or operability. SMUD, working with its consultants and contractors, identifies problem areas and calls for proposed solutions. The proposals are then evaluated by SMUD with expert support as necessary from its consultants. Training of SMUD personnel toward these requirements is set forth in Appendix 1C of the PSAR.

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QUESTION 1A.10 (DRL 1.7) Your Amendment No. 1 provided the SMUD response to applicable questions raised during the review of a similar plant (Metropolitan Edison). This response used information that was available through November, 1967. Please update your response to these questions by considering applicable information that became available in January, 1968.

ANSWER The PSAR has been updated to be responsive through Metropolitan Edison Company's PSAR Amendment No. 6 (Docket No. 50-289).

QUESTION 1A.11 (DRL 16.6) Discuss and evaluate your program to experimentally study vibrations in the check valves.

ANSWER (See the response to Question 1.3)

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