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WCAP-3269-75

LARGE CLOSED-CYCLE WATER REACTORRESEARCH AND DEVELOPMENT PROGRAMSUMMARY REPORT

OF DESCRIPTIONS OF TOPICAL REPORTS

PREPARED UNDER AEC CONTRACT AT(30-1)-3269

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E. Weingart

Project Administration

December 1967

Prepared for the U. S. Atomic Energy Commission
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(Sub-task LRD-100)

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INTRODUCTION

This summary report is a bibliography of the topical reports produced by the Westinghouse Electric Corporation for the U. S. Atomic Energy Commission under Contract AT(30-1)-3269. The reports represent developments in reactor technology subsequent to June 30, 1962, in continuation of the effort initiated December 1, 1960 under a prior contract. The reports for the initial contract were summarized in WCAP-3728.

Brief descriptions of the reports have been collated into the subtasks corresponding to the main work areas. These areas are as follows:

- 100 - Project Administration
- 200 - Reactor Physics
- 300 - Core Mechanics and Hydraulics
- 400 - Chemistry and Corrosion Effects
- 500 - Development of Plant Systems
- 600 - Reactor Safety Analysis
- 700 - Critical Experiments

The earlier program developed basic considerations and methods required for development of a large, chemical shim, closed cycle, pressurized water reactor. The later program extended these developments, and further applied them to a larger nuclear power reactor.

In the physics area, computing methods were developed and core characteristics were evaluated. Core power distribution and control rod reactivity worths were calculated, and fuel management for large reactors received consideration. Critical studies were made comparing calculated results with experimental data to establish validity of methods.

The core mechanics effort was largely directed towards use of Zircaloy fuel cladding. Hydraulic designs for the large reactor were produced and tested.

The chemistry emphasis shifted from basic studies of the shim solutions to the effects of these solutions on corrosion and wear of reactor materials. Studies were also conducted on the dynamic behavior of reactors with chemical shim solutions.

Reference core design parameters were established, and plant systems designs were outlined.

Periodic reports presenting information in addition to that contained in the topical reports are included in the following list of "Progress Reports" for the "Large Closed-Cycle Water Reactor Research and Development Program":

<u>Report No.</u>	<u>For the Period</u>
WCAP-3718	July 1 to September 30, 1962
3721	October 1 to December 31, 1962
3738	January 1 to March 31, 1963
3739	April 1 to June 30, 1963
3743	July 1 to September 30, 1963
3750	October 1 to December 31, 1963
WCAP-3269-2	January 1 to March 31, 1964
3	April 1 to June 30, 1964
5	July 1 to September 30, 1964
6	October 1 to December 31, 1964
12	January 1 to March 31, 1965
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19	July 1 to September 30, 1966
20	October 1 to December 31, 1966
21	January 1 to March 31, 1967
22	April 1 to June 30, 1967
23	July 1 to September 30, 1967
67	October 1 to December 31, 1967

DESIGNATION AND TITLE OF TASKS

FUNDED UNDER THE LRD PROGRAM

LRD

100 PROJECT ADMINISTRATION

200 REACTOR PHYSICS DEVELOPMENT

201 Light Water Reactor Core Physics Calculation Methods

202 Local Power Disturbances

203 Nuclear Design - Fuel and Control Management

204 Analysis of Nuclear-Thermal-Hydraulic Interactions

205 Development of Nuclear Instrumentation System

300 REACTOR CORE - DESIGN AND FABRICATION DEVELOPMENT

301 Development of Advanced Fuel Assembly Fabrication

301.1 Mechanical and Physical Properties of Zircaloy Cladding

301.2 Hydriding Studies of Zircaloy-4 Fuel Element Cladding Materials

301.6 Fabrication of Advanced RCC Fuel Assembly

301.7 Determination of Biaxial Strain Fatigue Limits of Cladding Materials

301.8 Stress Corrosion of Stainless Steel Fuel Element Cladding

301.9 Burnable Poison Fuel Mix Process Development

302 Irradiation Proof Tests of Advanced Fuel Assembly Prototype

302.6 Irradiation Testing of Advanced Canless Fuel Assembly with
Improved Grids

302.7 Improved Cladding Materials Development

302.9 Fuel Swelling Studies

303 Development of Advanced Fuel Assemblies - Design and Hydraulic Testing

303.3 RCC Dual-Assembly Hydraulic Testing

303.6 Development of Advanced RCC Canless Fuel Assembly Design

304 Control Rod Material Selection and Prototype Fabrication

304.1 Metallurgical Characteristics of Control Rod Materials

305 Fuel Bearing Control Rod Followers - Design Development

306 Fuel Bearing Control Rod Followers - Fabrication Development

307 Core Support and Reactor Internals-Mechanical Development

308 Reactor Evaluation Channel Operation and Maintenance

308.6 Reactor Evaluation Channel Operation and Special Equipment
Maintenance

309 Reactor Flow Model Test and Flow Mixing Between Fuel Assemblies

310 Flow Stability and Burnout Studies in an Open Lattice PWR Core

400 CHEMICAL SHIM DEVELOPMENT

401 Out-of-Pile Corrosion Testing and Studies

402 In-Pile Chemistry Testing

403 Concentration Control of Soluble Neutron Poisons

404 Continuous Measurement of Soluble Neutron Poison Concentrations

405 Wear Testing of Movable Parts

406 Chemical Studies of Soluble Neutron Poisons

407 Chemical Shim Kinetic Experiments

DESIGNATION AND TITLE OF TASKS (Continued)

500	PLANT DEVELOPMENT
501	System Stability and Control Studies
501.7	Loss of Coolant Flow Analysis
502	Analysis of Steam Generator Transient Behavior
503	Development of Auxiliary Systems
504	Development of Reactor Coolant System
504.4	Development of Refueling Concept
505	Radiation Analysis
506	Development of Sequencing, Variable Speed Rod Drive Control System
600	REACTOR HAZARDS ANALYSIS
700	CRITICAL EXPERIMENTS
701	Planning and Analysis of Critical Experiments
702	Performance of Critical Experiments
702.7	Fine Structure Critical Experiments

WCAP-3269-7 SAN ONOFRE NUCLEAR GENERATING STATION REFERENCE
CORE DESIGN - W. E. ABBOTT, MARCH 1965

The research and development program under Contract No. AT(30-1)-3269 is directed towards the design, construction, test and operation of the San Onofre Nuclear Generating Station, which will be jointly owned by the Southern California Edison Company and the San Diego Gas and Electric Company. This nuclear power plant will employ a closed cycle water reactor which will operate at 1346 MW(t) to produce 450 MWe gross.

This report presents the reference design parameters for the pressurized light water moderated and cooled core for the San Onofre reactor. Pertinent mechanical, thermal and hydraulic, and nuclear design data are presented, and considerations affecting the selection of the design concepts and specific values of design parameters are given. Also included are considerations leading to the selection of core materials.

WCAP-3269-66 COMPARISON OF PREDICTED AND MEASURED NUCLEAR DESIGN
PARAMETERS AT BEGINNING OF LIFE OF CYCLE 1 IN THE
SAN ONOFRE REACTOR - A. J. IMPINK, JR., J. R. EASTER,
DECEMBER 1967

A program of reactor physics tests has been carried out at beginning of life of the first fuel cycle in the San Onofre Nuclear Generating Station in support of the nuclear design effort which preceded construction and operation of the station. Final design methods and analytically derived predictions are reviewed and the results of the experimental program, spanning reactor operations from first criticality to initial operation at 395 MWe, are presented. Finally, comparison is made between predicted and measured values and conclusions concerning the validity of the analytical procedures are drawn.

WCAP-3269-34 THEORETICAL CALCULATIONS OF INELASTIC
SCATTERING MATRICES FOR MULTIGROUP
REACTOR ANALYSIS - C. P. BHALLA,
SEPTEMBER 1964

The treatment of neutron slowing-down is of significant importance in the design of thermal power reactors. There are two main mechanisms by which fission neutrons lose energy: (1) the elastic scattering and (2) the inelastic scattering processes. This report discusses only the inelastic scattering of neutrons, by which it is implied that a neutron loses part of its energy by exciting the nucleus (in the ground state) to one of its excited levels. The probability of this process depends upon the specific nuclear levels of a nucleus. All the nuclear levels, which can in principle be excited, are not known. The relevant information like spins, parities and excitation energies may be known experimentally only for a few nuclear levels. Consequently, a nuclear model has to be used in such calculations.

This report describes a calculational procedure for obtaining the inelastic scattering probabilities, which are required for MUFT-4 - A Multigroup Fast Neutron Spectrum Code. The relevant theoretical considerations are summarized. The formulation of Hauser and Feshbach is used when known nuclear levels can be excited, but in all other cases, the statistical model is recommended. To elucidate the procedure an illustration is given for Fe^{56} . In addition, general inelastic scattering probabilities are presented as a function of nuclear temperatures, and a critical discussion of the theoretical developments in the statistical model is given.

WCAP-3269-52 EVALUATION OF A COMPUTATIONAL SYSTEM APPLICABLE
TO THE NUCLEAR ANALYSIS OF CRITICAL EXPERIMENTS
AND POWER REACTORS - W. J. EICH, H. T. WILLIAMS,
R. E. RADCLIFFE, JUNE 1965

This report describes a computational system which has proven to be extremely consistent in the nuclear analysis of practical configurations involving light water moderated lattices. The assemblies which have been analyzed consist of either UO_2 or U-metal clad in aluminum, steel or zirconium; single enrichment, multi-region and borated room temperature experiments, as well as actual reactor (Yankee Core I) zero power configurations, have been studied.

The computational scheme applied to the analysis of these experiments makes use not only of a refined specification of lattice cell reaction rates but also of auxiliary multigroup calculations to account for the various spectral and transport effects always present near lattice discontinuities such as the core-reflector interface in criticals or the control rods in a large reactor.

Neutron multiplication factors for most of the room temperature criticals subjected to analysis average to the value of 0.9979 with a standard deviation of 0.0019. Neutron multiplication factors which are outside this band correspond to two sets of experiments. One set is a group of high buckling experiments with 0.300" diameter fuel rods. The second set involves the Yankee reactor at a temperature of 514⁰F.

WCAP-3269-40 AN EXPERIMENTAL EVALUATION OF THE POWER
COEFFICIENT IN SLIGHTLY ENRICHED PWR
CORES - W. T. SHA, APRIL 1965

Because of the low thermal conductivity of UO_2 fuel pellets, the pellet temperature changes appreciably with power level, and a large reactivity variation results from the Doppler broadening of U-238 resonance.

Knowledge of fuel pellet temperature as a function of power is important for the following reasons:

1. Its effect on the steady state neutron multiplication factor, as well as the reactivity feedback in reactor kinetics behavior, is essential to reactor design.
2. It is essential in predicting accurately the conversion ratio of U-238 and, therefore, the lifetime of a core.
3. It is essential in providing sufficient control rods to meet local variation requirements.
4. It is essential in the evaluation of the power coefficient as a means of damping spatial xenon oscillations for large power reactors.

A correlation is presented of the effective UO_2 fuel pellet temperature (T_{eff}) for the Doppler broadening of U-238 resonance, based on the experimental power coefficient data from Yankee, Saxton, BR-3, and Selni. The correlation can predict the total power defect of reactivity from hot zero power to full power within experimental uncertainties, and T_{eff} at any power level within $\pm 80^\circ\text{F}$, a magnitude which constitutes no more than $\pm 6\%$ of the value at full power for current PWR design. With the $\pm 80^\circ\text{F}$ uncertainty in T_{eff} , the power coefficient of reactivity can vary as much as $\pm 30\%$ in the low power region and $\pm 20\%$ in the high power region. This study indicates the temperature drop in the gap between the pellet and clad to be much lower than the value calculated for it in the past.

WCAP-3269-46 IN-CORE INSTRUMENTATION FOR THE SAN ONOFRE
NUCLEAR GENERATING STATION - W. D. LEGGETT,
MARCH 1965

In-core instrumentation for the San Onofre Nuclear Generating Station consists of thirty Aeroball flux monitoring channels and thirty-five fuel assembly outlet thermocouples. This equipment is not intended to serve as operating instrumentation, but to provide supplementary core monitoring to be used from time to time to confirm reactor performance.

The Aeroball system pneumatically moves 1/16" steel balls into the core in tubes located within the flux thimbles, which are steel tubes about 0.3" OD running parallel to the fuel rods from the fuel assembly bottom, then out through the pressure vessel upper head. Under irradiation the manganese content of the balls becomes radioactive to a degree proportional to the local neutron flux. Following irradiation, the balls are pneumatically moved from the core to a scintillation detector which gives a signal proportional to activity and hence flux.

In-core instrumentation was added subsequent to the original plant design, in order to permit raising the power rating from 395 MWe to 395 - 450 MWe. In this report, the instrumentation, the relevant plant characteristics, and the complementary ex-core instruments are described. The bases for the choice of instruments and instrument locations are discussed, the procedure for data reduction is delineated, and the anticipated accuracy of the system is stated.

WCAP-3269-48 XENON INDUCED CORE INSTABILITIES - L. E.
STRAWBRIDGE, E. C. ALLARD, C. P. BHALLA,
FEBRUARY 1965

In an extensive analysis of axial xenon oscillations in a pressurized water reactor, using a one-dimensional diffusion theory analysis, the effect of core height, power distribution, Doppler coefficient, and moderator coefficient on xenon oscillations has been evaluated. The results indicate:

1. Stability to xenon oscillations definitely decreases with increasing core size.
2. Stability is quite sensitive to the power distribution, with flat or dished power shapes tending to be less stable than normal cosine distributions.
3. The Doppler coefficient is extremely important and is the principal damping mechanism in slightly enriched pressurized water reactors.
4. The moderator temperature coefficient has a relatively small effect on xenon oscillations.

Threshold analyses using the procedures of Randall and St. John were performed. A revised threshold calculation was developed during the program, which eliminates certain approximations in the previous procedure and produces results in substantial agreement with the diffusion theory results.

Methods of controlling reactors to prevent xenon oscillations were studied. It was demonstrated that either partial length or full length control rods can satisfactorily be used to maintain stability in cores that would otherwise be unstable.

It is concluded that the Doppler coefficient is sufficiently negative to prevent divergent axial xenon oscillations in a 10 ft. pressurized water reactor. Nevertheless, if a xenon oscillation should be initiated, it has been demonstrated that the control system is capable of controlling the reactor without sacrificing power capability.

WCAP-3269-25 CALCULATION OF LATTICE PARAMETERS AND CRITICALITY
FOR UNIFORM WATER MODERATED LATTICES - L. E.
STRAWBRIDGE, SEPTEMBER 1963

This report describes in detail a method of calculating criticality which has been programmed in the LEOPARD code, and which improves over the previous method in the following ways:

1. Elimination of the loop model in which the U-238 absorption was divided into two parts with the assumption that all U-235 epithermal fissions occurred at the breakpoint. Two empirical constants were used to describe the split in U-238 absorption and fast leakage. These constants are no longer necessary.
2. Introduction of a more general U-238 resonance integral correlation which is applicable to both uranium metal and uranium dioxide and which inherently includes the temperature effect.
3. Improved thermal calculations in which modified Amouyal-Benoist disadvantage factors are calculated at every SOFOCATE energy level.
4. Introduction of a heterogeneous fast fission effect to replace the homogeneous calculation. (This effect was found to be small.)

In evaluating each change to the calculational procedure, the neutron multiplication factor (K) was calculated for 49 critical experiments using UO_2 fuel, and, in some cases, 20 exponential experiments using uranium metal fuel.

Improvement over the previous methods is evidenced by a reduction in the standard deviation about the mean for several groups of experiments of particular interest in the design of pressurized water reactors. The proposed method was found to be in good agreement with a more rigorous calculation which requires approximately 50 times as much computer time.

WCAP-3269-26 LEOPARD- A SPECTRUM DEPENDENT NON-SPATIAL DEPLETION
CODE FOR THE IBM-7094 - R. F. BARRY, SEPTEMBER 1963

The LEOPARD computer program determines fast and thermal spectra, using only basic geometry and temperature data, based on a modified MUFT-SOFOCATE model. The code optionally computes fuel depletion effects for a dimensionless reactor and recomputes the spectra before each discrete burnup step. LEOPARD was written to fill two needs:

1. to automate the spectra calculations defined by Strawbridge in WCAP-3269-25.
2. to account for the variation of neutron spectra with fuel depletion.

LEOPARD is written in FORTRAN-II and FAP and operates under the FORTRAN MONITOR SYSTEM. Basic requirements are a 32-K, two channel, 12 tape IBM-7094 with off-line card/tape equipment. From an operational standpoint, an important feature of LEOPARD is the inclusion of a buffered tape-handling package written by Sanford and Hoodes, discussed briefly in this report. The package allows asynchronous transmission of data to and from tape and consequently improves program efficiency. Running time for LEOPARD is one-two complete spectra calculations per minute.

The LEOPARD program itself is described in this report. The modified MUFT-SOFOCATE model is described in a companion report, WCAP-3269-25.

WCAP-3269-33 CONTROL ROD WORTH IN A HETEROGENEOUS MOCKUP
OF A LARGE PRESSURIZED WATER REACTOR CORE -
P. G. LACEY, JULY 1964

This report presents the results of a control rod worth evaluation program which included both analytic and experimental investigations. The critical experiments were performed at the Westinghouse Reactor Evaluation Center (WREC) using stainless steel clad UO_2 fuel rods in light water.

This program had two main purposes:

1. to measure absolute and relative reactivity worth of different types^{/a} and geometries^{/b} of control rods in a simulated hot PWR core, and
2. to permit evaluation of analytic methods and data used in control rod worth calculations in the nuclear design of large PWR cores.

The analytic methods and procedures used to calculate control rod worth in large PWR cores were applied to the analysis of critical experiment configurations in which control rod worth was measured. The results of this correlation are given along with complete descriptions of the analytic and experimental methods. Reactivity worths are given for cruciform shaped control rods of Ag-In-Cd and B_4C in a simulated hot-core fuel assembly mockup. Experimental local-fission power distributions obtained in this series of measurements are given in a companion report WCAP-3269-35, along with the analytic power distributions.

a) (Ag-In-Cd and B_4C); b) (solid blade and rodded cruciforms)

WCAP-3269-35 LOCAL FISSION POWER DISTRIBUTIONS AROUND
CONTROL RODS AND CONTROL ROD FOLLOWERS IN
SIMULATED PWR CORES - P. G. LACEY,
SEPTEMBER 1964

Local fission power distributions around cruciform shaped control rods and control rod followers are given for a typical pressurized water reactor fuel assembly design. Experimental data taken by gamma scanning typical fuel rods around the control rod channel in a simulated core cross section are compared with calculated fission power distributions.

All cores were critical experiments using Type 304 stainless steel clad UO_2 fuel rods. A majority of the data apply to fully inserted or fully withdrawn, centrally located control rods, however, some of the data apply to an off-center control rod inserted half way into the core to permit observation of axial power shape as a function of radial position from the control rod channel.

The report includes discussion of the analytic and experimental techniques, as well as the correlation between experiment and theory.

WCAP-3269-47 AN ANALYSIS OF REACTIVITY WORTH OF THE ROD CLUSTER
CONTROL (RCC) ELEMENTS AND LOCAL WATER-HOLE POWER
DENSITY PEAKING - W. T. SHA, MAY 1965

The reactivity worth and local power disturbances associated with a reactivity control system consisting of clusters of small cylindrical poison elements is investigated, and the results of the experimental program for RCC evaluation are presented.

In the rod worth computation, a method to specify the boundary condition of the rod cluster (RCC) elements in two-group, two-dimensional diffusion theory is presented. A semi-empirical design procedure is developed for extrapolation in size and temperature to the operating PWR core. It appears that rod worth can be correlated within 3.5% for the practical RCC arrangement by using this method.

A comparison is also made between the calculated values of power-peaking and experimental measurements. It is concluded that RCC local water-hole power density can be predicted within $\pm 2\%$. The proposed procedure is compared to a technique which attempts to eliminate the consequence of empiricism in extrapolation in order to establish the probable accuracy of the extrapolation.

WCAP-3269-49 OPTIMUM CORE LATTICE PARAMETERS FOR THE SAN ONOFRE
NUCLEAR GENERATING STATION - L. E. STRAWBRIDGE,
R. F. BARRY, FEBRUARY 1965

This optimization study, applicable to large steel-clad pressurized water reactors, uses the extensive physics data reported in WCAP-3733, coupled with detailed fuel cost equations, to determine the core parameters corresponding to minimum fuel costs. It describes the LOCK code, written for the IBM-7094, which can optimize simultaneously the three parameters W/U, pellet size, and enrichment, and it presents many results obtained with the LOCK code.

Relatively loose lattices were found to be economically attractive, with the optimum W/U generally greater than 4.5. (W/U is defined as the volume of water in the active core divided by the volume of uranium, if the uranium existed in metallic form.) The optimum pellet diameter was found to be approximately 0.36 inches, and the optimum burnup was in the range of 25,000 to 30,000 MWD/MTU. It should be noted that the detailed consideration of plant components and burnout heat fluxes (which was beyond the scope of this study) tends to reduce the optimum water-to-fuel ratio.

The effect on the optimum lattice parameters was determined for changes in such parameters as fabrication costs, reprocessing cost, separative work charge, load factor and kw/ft limit, and the effect of a nuclear economy based on private fuel ownership was evaluated. The results, presented in graphical form, show that the optimum lattice parameters change very little as economic ground rules change. If reactor lattices are optimized for present economic ground rules, redesign probably will not be justified when ground rules consistent with private fuels ownership become effective.

WCAP-3269-53 FUEL MANAGEMENT IN A LARGE CHEMICAL SHIM
CLOSED-CYCLE WATER REACTOR - S. RAJE,
D. H. PETRIE, MAY 1966

This report presents the development of a fuel management program to obtain maximum design power rating with maximum utilization of fuel for a large chemical shim closed-cycle water reactor. The methods developed are generally applicable to large pressurized water reactors with chemical shim control, even though the data used pertain to a specific reactor core.

The rated power output of a reactor with non-uniform power density distribution is limited by physical and thermal conditions which exist in those fuel rods which have the highest power density and are surrounded by the hottest coolant. A more uniform distribution would allow more of the fuel rods to operate close to their thermal-hydraulic design limit, resulting in a higher permissible power output for the core. The most non-uniform power distribution which will be obtained during the life time of the core determines the design maximum power level of the reactor.

A fuel management program termed the modified Out-In technique which meets design goals is presented in detail in this report. Fuel loading patterns are given for the first four operating cycles in which the calculated maximum-to-average power density ratios in the x-y plane are 1.50 or less.

The report includes a detailed statement of the problem of fuel management, the various fuel management schemes considered, a comparison of the results obtained using the different methods, and the procedures used in calculation of power density distribution and fuel depletion with the different fuel cycling methods.

WCAP-3269-55 CORE PHYSICS CHARACTERISTICS OF THE FIRST
LOADING OF THE SAN ONOFRE NUCLEAR GENERATING
STATION - D. L. MILLER, OCTOBER 1966

The nuclear design characteristics of the San Onofre first core are presented here. Design data is given for reactivity, power distributions, control by chemical shim and rods, and for all reactivity coefficients. Along with the nuclear characteristics of the San Onofre core, the methods of analysis of these data are presented. Finally, experimental justification of these methods is given where such experimental data exist.

In evaluating the nuclear characteristics of the core, a two-step approach has been taken. First, detailed nuclear parameters are presented for every phase of operation of the first cycle. Where applicable, the calculated characteristics are compared with design limits to show that adequate margin of safety exists in all cases, and the method of calculation is given for each parameter calculated. Second, a justification is given for the methods of calculation which have been employed.

Experimental measurements from criticals or operating reactors or both are used to validate the methods employed for design calculations.

It is shown that the calculations can predict core operational conditions, and that the nuclear characteristics of the San Onofre core which are presented have a high confidence level of being obtained.

WCAP-3269-37 A NEW ZIRCALOY END CLOSURE WELD JOINT
CONFIGURATION - R. DUNCAN, SEPTEMBER 1965

Development of improved end closure techniques for Zircaloy-clad fuel elements in water reactors in general, and in the San Onofre plant in particular, was undertaken to obtain increased reliability at reduced processing cost.

Reliability in a welded closure is improved if the two joint members have uniform and reproducible thermal conditions. This is best accomplished by making the effective sizes of the two sections equivalent. Costs are reduced by decreasing the number of welding operations required for each element. Combining attachment of the end plug to the Zircaloy cladding with closure of the gas vent hole reduced the number of welding operations per element from three to two.

The new closures were made under normal weld box conditions to avoid unusual corrosion effects which would complicate final evaluation of loop test results.

WCAP-3269-30 A STUDY OF THE ZIRCALOY-STEAM REACTION IN A
SIMULATED REACTOR LOSS-OF-COOLANT ACCIDENT -
C. J. KUBIT, R. N. STANUTZ, D. N. FULTONBERG,
SEPTEMBER 1965

An experiment was performed to study the kinetics and extent of reaction between Zircaloy-2 and steam under conditions of a simulated pressurized water reactor loss-of-coolant accident.

Prior to the experiment, the expected temperature transients of the clad surface of stainless steel fuel rods after a loss-of-coolant accident were obtained from computer code runs. An attempt was then made to duplicate the calculated transients by means of a programmed heat input to the specimens in an autoclave containing 150 psi, 600°F steam.

A comparison of the experimental temperature transient of the Zircaloy-2 specimen with that of a stainless steel control sample indicates that there was no autocatalytic effect and that the Zircaloy-steam reaction was negligible up to 1750°F, the highest temperature reached in the experiment.

WCAP-3269-41 PHYSICAL AND MECHANICAL PROPERTIES OF ZIRCALOY
2 AND 4 - D. B. SCOTT, MAY 1965

This report assembled information on the physical and mechanical properties of Zircaloy-2 and Zircaloy-4, primarily for engineering design. Extensive use was made of the published literature, and where published information was not available, tests were initiated to determine the necessary properties.

The following properties were determined on representative Zircaloy-4 fuel cladding and are presented as a function of temperature:

- 1) Coefficient of thermal expansion, from 70°F to 1850°F in the longitudinal direction, from 70°F to 1200°F in the diametrical direction
- 2) Thermal conductivity, from 70°F to 1450°F
- 3) Specific heat, from 70°F to 1850°F
- 4) Elastic modulus, from 70°F to 1400°F
- 5) Poisson's Ratio, from 70°F to 750°F
- 6) Longitudinal tensile properties, from 70°F to 1400°F
- 7) Longitudinal tensile properties for transient temperature conditions, from 950°F to 1350°F
- 8) Compressive strength and modulus, from 70°F to 1050°F
- 9) Transverse tensile properties, from 70°F to 800°F

The data are assembled and expressed graphically where possible. There was a particular lack of high temperature mechanical information for Zircaloy-4 tubing which was needed both for conventional design purposes and for safety problems, such as a loss of water accident.

WCAP-3269-56 CORROSION AND HYDRIDING OF ZIRCALLOY UNDER
HEAT TRANSFER CONDITIONS - E. N. AQUA,
R. J. ALLIO, K. C. THOMAS, AUGUST 1967

The effect of heat transfer and isothermal conditions on the corrosion and hydriding of Zircaloy-4 tubing has been studied in a pressurized water system. In all of the systems studied, the oxidation was directly proportional to the clad surface (metal-oxide interface) temperature. Nucleate boiling, achieved with a maximum heat flux of 20,000 BTU/ft²hr, caused increased oxidation as compared to heat transfer systems with forced convection. The oxidation rate was inversely proportional to the difference between bulk water and clad surface temperatures. The hydrogen absorption was the same in the nucleate boiling systems as in the forced convection heat transfer systems, and homogeneous throughout the tube cross-sections. No abnormal platelet clusters were observed.

WCAP-3269-42 BIAXIAL STRAIN FATIGUE OF ZIRCALOY-4 -
R. E. SCHREIBER, OCTOBER 1965 (TWO VOLUMES)

Fatigue data were developed for Zircaloy-4 to implement a study of thin cladding for large pressurized water reactor cores. Conventional clad design is free-standing; i.e., it does not collapse under coolant pressure. Thin clad would be collapsed, hence subjected to strain cycling throughout the entire core life. The economic incentives of more compact core and reduced material requirement warranted study of the extent to which the material would resist fatigue failure.

Part A - Experimental and Analytical Techniques

The design and operation of the equipment used to obtain uniaxial and biaxial fatigue data are stressed. The method of strain analysis in wrinkled tubular biaxial specimens is presented, and an example of preliminary data analysis given.

Part B - Analysis of Test Results, Development of Strain Fatigue Model, and Application to Design

Part B describes the uniaxial tests and shows how their strains were analyzed. Push-pull specimens with uniform cross section over their gage length were deliberately cyclically buckled. The stress vs cycles and the direct strain vs cycles are examined, as well as the effects of mean strain and plastic strain range on cycles-to-failure.

The biaxial equipment and specimen descriptions are reviewed prior to a discussion of the test results. Biaxial strain fatigue data are reported for 10-15% cold worked Zircaloy-4 tested at 715°F. A mathematical model of strain fatigue was developed and applied to a design problem involving fuel rod cladding for the San Onofre Reactor. The predicted fraction of fatigue life consumed was 30%.

WCAP-3269-50 STRESS CORROSION IN ALLOYS FOR FUEL ELEMENT
CLADDING - STAINLESS STEEL AND ZIRCALOY -
K. C. THOMAS, R. J. ALLIO, SEPTEMBER 1965

The purpose of the present study was to determine the factors responsible for the stress-corrosion failure of stainless steel and Zircaloy fuel element cladding so that they can be controlled, and so that the susceptibility to failure can be minimized.

The failure of austenitic stainless steel in high temperature water systems is well documented and is attributed to concentration of chloride and hydroxyl ions. These ions should not be a problem in water reactor systems where extremely high purity water is used, but failures encountered in stainless steel fuel cladding suggest that although the purity of the bulk coolant is high, undesirable concentration of the ions at heat transfer surfaces can lead to conditions conducive to stress corrosion attack.

Similar failures have been noted in zirconium in chloride solutions at low temperatures, and also in Zircaloy in a reactor circuit. However, there are only two instances reported for this class of material.

The report describes a series of experimental studies which included tests on stressed 304 SS wire in concentrated aqueous Mg, Ca, and Li chloride solutions, tests on 304 SS strip at various stress levels under autoclave conditions reproducing the pressurized water environment, a study of crack initiation stress levels on 304 SS wire in boiling Mg Cl_2 solution, a study of the effect of different alloy additions to a basic 16% Cr/20% Ni material on time to failure in Mg Cl_2 solution, a study of dislocation densities and arrangements in 304 SS and Incoloy 800 which postulated a theory of stress corrosion cracking based on the concept of ordering and reordering in an alloy lattice, and tests of stressed Zircaloy wire in 20% Fe Cl_3 solution.

WCAP-3269-36 FABRICATION AND IRRADIATION OF THIN-WALLED
TYPE 304 STAINLESS STEEL CLAD FUEL RODS IN
THE SAXTON REACTOR - D. R. McCLINTOCK,
JANUARY 1965

Ten thin-walled (9.7 mils) Type 304 stainless steel clad fuel rods were irradiated as part of the Saxton core to a maximum burnup of 8900 MWD/MTU at a maximum heat flux of 410,000 Btu/hr/ft². The corresponding integrated fast flux exposure (> 1 Mev) was 1.1×10^{21} nvt. Post-irradiation examinations revealed no evidence of failure or crack formation although the thin-walled clad was subjected to plastic deformation. The successful operation of this engineering proof test emphasizes the satisfactory performance of stainless steel in a borated pressurized water environment.

WCAP-3269-59 DESIGN, FABRICATION AND IRRADIATION OF A SAXTON
SPECIAL 9x9 ASSEMBLY - A. F. MATTIS, D. R. McCLINTOCK,
MAY 1967

A special 9 x 9 assembly was irradiated in the Saxton reactor to obtain statistically significant data on prototype designs and materials considered for use in a large plant chemical shim environment.

The assembly operated as part of the reactor core for a total of 155 EFP days at a maximum heat flux of 525,000 BTU/Hr-Ft². The assembly experienced a peak burnup of ~8,000 MWD/MTU and a corresponding fast flux exposure (>1 Mev) of 1.0×10^{21} nvt.

In-pile performance was satisfactory and a visual examination revealed no anomalies. Exposure did not increase the force required to move the RCC absorber rods.

WCAP-3269-43 PERFORMANCE OF ZIRCALOY CLAD FUEL RODS IN SAXTON -
D. R. McCLINTOCK, DECEMBER 1967

This report evaluates performance of Zircaloy-4 clad UO_2 fuel rods which operated in chemical shim environment in the Saxton reactor.

The work includes a comparison of the effects of varying pre-irradiation surface treatments on fuel rods in a special removable 3x3 assembly, representing design power levels up to 16 Kw/ft; and it examines the behavior of defected UO_2 fuel rods to ascertain effects of boric acid on fuel-water reaction, or on hydriding of the internal clad surface.

WCAP-3269-64 THE EFFECT OF HIGH POWER DENSITY ON UO_2 ROD TYPE
ELEMENTS - THE SAXTON SPECIAL 2x2 ASSEMBLY -
D. R. McCLINTOCK, DECEMBER 1967

Stainless steel clad UO_2 fuel rods were operated at power densities approaching center melting in the Saxton reactor. The in-pile performance of these rods is evaluated in this report.

This fuel represents burnup approaching 10,000 MWD/MTU.

WCAP-3269-61 IN-PILE SWELLING OF UO_2 FUELS - C. J. KUBIT

This report describes the failures obtained in swelling experiments on UO_2 fuels under irradiation in the ETR. Some conclusions are drawn on the causes for the failures.

WCAP-3269-62 HYDRAULIC DESIGN OF A ROD CLUSTER CONTROL FUEL
ASSEMBLY - R. A. DEAN, SEPTEMBER 1965

A test program was conducted to evaluate the hydraulic characteristics of the RCC control rod system. Two fuel assemblies, one containing the RCC control rod system, were positioned side by side in a single test cell.

Tests were run which determined the resistance required to balance the outlet flows from the two assemblies. Flow patterns, made visible by dye injection, indicated the lack of mixing between the outlet of the fuel assembly and the upper core plate. A dual purpose mixing device was developed which balanced the outlet flows in addition to mixing the coolant so that the average outlet temperature could be recorded by the in-core instrumentation.

An analytical model for predicting the pressure loss coefficients for components with rounded or chamfered entrances is presented. Pressure loss coefficients for the various core components were obtained and agree well with this model.

Pressure drop measurements were also made to determine the friction factor for flows in annuli with the radial gaps varying from 0.014" to 0.053". These data are presented for flows in the laminar region and for turbulent flows with Reynolds numbers up to 15,000.

WCAP-3269-29 METALLURGICAL EVALUATION OF SILVER-INDIUM-CADMIUM
CONTROL ROD ALLOYS - W. R. SMALLEY, D. B. SCOTT,
AUGUST 1964

Tin, platinum, and palladium have been evaluated as alloying additions to the basic 80% silver-15% indium 5% cadmium alloy in an effort to improve mechanical properties and corrosion resistance. It was found that a combined addition of 2.5% Pd and 0.5% Sn was the most effective in increasing elevated temperature yield and creep strengths, and in improving the resistance to internal oxidation in hot borated water.

Experiments run to determine the relative effects of irradiation on the basic Ag-In-Cd and a 0.5% Sn modification showed that the ternary alloy was more resistant to irradiation-induced changes in mechanical properties and dimensions.

Extended exposure of Ag-In-Cd specimens at 725°F, both in an inert atmosphere and in steam, showed that the alloy was dimensionally stable under the most severe temperature and environmental conditions expected in service.

WCAP-3269-8 HYDRAULIC TESTS OF THE SAN ONOFRE REACTOR
MODEL - G. HETSRONI, JUNE 1964

Experimental studies were conducted on a one-seventh scale model of the San Onofre Reactor. Seven lower plenum configurations were tested and compared with regard to the pressure drop of various components and with regard to the flow distribution at the inlet to the core.

Based on these studies, and the change to Rod Cluster Control which requires no control rod followers, the vessel length of the San Onofre plant was reduced by seventy inches and other changes were made in the internal components.

In addition, the inlet flow distribution subfactor in the engineering hot channel factor was reduced, and the estimates for the pressure drop in the vessel were revised.

WCAP-3269-24 THE SAXTON CHEMICAL SHIM EXPERIMENT -
J. WEISMAN, S. BARTNOFF, JULY 1965

An extensive program to demonstrate the feasibility of operation with chemical shim control has been conducted at the Saxton reactor. The experimental data obtained do not indicate any significant boron accumulation on the core surface at power or any decrease in core lifetime under normal operating conditions.

Boric acid remained in the core continuously from May 23, 1963, to January 20, 1964. After a brief period of non-borated operation, chemical shim operation resumed on March 6 and continued until April 20 when the reactor was shutdown temporarily for replacement of experimental fuel rods. Total average core burnup was then 6330 MWD/MTU, of which 4521 MWD/MTU was incurred during chemical shim operation. Peak burnup was 15,600 MWD/MTU.

Alkali addition for pH control was satisfactory in both pure water and the chemical shim solution. The expected hot channel factors were achieved.

WCAP-3725 CORROSION AND CORROSION PRODUCT BEHAVIOR IN
CHEMICAL SHIM SOLUTIONS - D. D. WHYTE,
G. R. TAYLOR - DECEMBER 1964

A comparison between metal release rates and net metal deposition rates for 304 stainless steel at 600°F was made in a 304 stainless steel loop system, in boric acid (1000 ppm as Boron) and partially neutralized boric acid (10^{-4} M LiOH), using radiotracer techniques. These experiments indicated that for a two-month exposure the metal release rates were higher in the partially neutralized boric acid ($7 \text{ mg/dm}^2\text{-mo}$ compared to $1.5 \text{ mg/dm}^2\text{-mo}$), but decreased rapidly with time. The net deposition rates were also higher in the partially neutralized boric acid solutions ($6.8 \text{ mg/dm}^2\text{-mo}$ compared to $1 \text{ mg/dm}^2\text{-mo}$).

Deposition rates were determined for 410 stainless steel, Inconel, chrome-plate on 410 stainless steel, and Zircaloy-2. The rates varied with the material, being highest for 410 stainless steel, and lowest for Zircaloy-2 and chrome-plate. Exposure of specimens prior to testing decreased the net deposition for all materials tested. Velocity variation between 6 and 20 feet per second had no effect on deposition rate in boric acid or partially neutralized boric acid.

Ion-exchange performance was evaluated from detailed observations of chemical and radioactivity distribution in resin columns. Essentially no solids could be freed from the ion-exchange material by washing. Axial distribution in the column was normal. Equilibrium uptake at the influent end was determined as 4.7 milliequivalents of trivalent iron or 3.14 milliequivalents of divalent iron per dry gram of resin in boric acid solution.

WCAP-3269-57 THE POST-IRRADIATION EXAMINATION OF SAXTON FUEL
CLADDING CORROSION PRODUCTS - L. F. PICONE,
W. D. FLETCHER, MARCH 1966

Fuel assemblies from the Saxton Core I were examined after exposure at various times to coolant chemistries and nuclear conditions to ascertain the corrosion product deposition on fuel clad surfaces.

It was established that no significant difference existed between deposition on stainless and Zircaloy fuel cladding, and further that the actual amount of deposition was small under normal PWR operating conditions.

The results of the visual examinations of these assemblies together with some chemical and radiochemical data obtained from deposit samples are discussed in relation to the factors which effect corrosion product deposition.

WCAP-3269-63 FISSION PRODUCTS FROM FUEL DEFECT TEST AT SAXTON -
W. D. FLETCHER, L. F. PICONE, APRIL 1966

An experimental 3 x 3 fuel rod assembly, with one rod intentionally defected with a 15 mil hole through the cladding, was placed in the Saxton reactor which was operated at and near full power for 57 days. Primary coolant samples were analyzed for fission products, and escape rate coefficients were estimated for iodines and fission gases which indicated that coefficients used in previous design calculations were very conservative.

The primary coolant system was purified by a potassium-borate ion exchange mixed bed which was effective for fission product iodines and tramp sodium, but not for fission product cesium.

WCAP-3269-9 EFFECTS OF CHEMICAL SHIM ENVIRONMENTS ON THE
WEAR OF REACTOR MATERIALS - D. G. SAMMARONE,
MAY 1964

The effects of chemical shim environments on the wear of reactor material were investigated at two temperatures, 600°F and 200°F, and in three water chemistries, demineralized water, boric acid, and boric acid plus lithium hydroxide. Six material combinations normally employed for wear applications in nuclear power reactors were used in this study in 36 individual tests run on a piston-cylinder wear test rig.

Combined and individual wear factors, as well as specific wear values, were determined. Results showed that the amount of wear with chemical shim solutions was generally the same as with demineralized water. Changes in wear due to temperature could not be detected except for the 410 SS vs. 304 SS piston-cylinder combination. In this case the amount of wear was higher at the 200°F temperature than at the 600°F temperature.

WCAP-3269-51 THE CALCULATED pH OF AQUEOUS BORIC ACID
SOLUTIONS AS A FUNCTION OF TEMPERATURE AND
ADDED BASE CONTENT - M. E. MEEK, MARCH 1965

Calculations of pH of chemical shim solutions at various temperatures are made using the equilibrium equations for conservation of total boron, charge and total base.

The pH is plotted as a function of total boron concentration and concentration of total base ion, and as a function of the total base (lithium or ammonia) and total boron concentration. Crossplots of pH versus total boron concentration at constant base concentration and plots of pH versus temperature at constant base and boron concentration are also presented.

A method is described for an iterative graphical solution for the pH of a solution of boron and a neutralizing mixture of bases at constant temperature.

WCAP-3269-58 REACTOR FOULING EXPERIMENT - L. F. PICONE,
G. R. TAYLOR, A. J. IMPINK, V. RAJAGOPAL,
JULY 1966

An experiment was conducted at the Saxton reactor to investigate a procedure for determining the presence of boron concentrations on operating in-core surfaces. Although boric acid for chemical shim control had been generally accepted, a question remained as to hideout, previously noted out-of-pile, and defined as the tendency for a solute to concentrate or precipitate at a heat transfer surface. A power excursion which could be produced by sudden release of concentrated neutron poison from hideout areas could be a potential reactor hazard.

This report describes the out-of-pile development for the reactor test, as well as the reactor experiment itself. Procedures were developed to prepare the $\text{Fe}(\text{OH})_2$ slurry, and to add it to the high temperature, high pressure system to foul the heat transfer surfaces. Distribution of the synthetic corrosion products on system surfaces was determined, and the safety of the test was established.

In the experiment at Saxton, the fuel rods were heavily fouled, and a reactivity effect was observed associated with the $\text{Fe}(\text{OH})_2$ deposit itself and considered independent of boron. Boron hideout of a type which was power reversible by transient from nucleate to non-nucleate boiling, and which could be detected by reactivity follow methods, did not occur during the fouling experiment.

WCAP-3269-28 SAXTON KINETIC EXPERIMENTS - V. RAJAGOPAL,
JUNE 1964

Several oscillation tests were performed on the Saxton reactor to determine the transfer functions, and to study the effects of chemical shim operation on the dynamic behavior of the reactor. The tests consisted of producing pseudo-random, random or sinusoidal changes in reactivity through an oscillator and measuring the reactor flux and temperature responses by use of correlation techniques. A cross-correlation method with a single series of pseudo-random input permitted analysis of data with small signal to noise ratios and over a wide range of frequencies.

The transfer functions determined at a power level of 20 MWt under both borated and non-borated operations showed no evidence of any power reversible boron hideout at the test conditions. Slight increases in fuel time constant and Doppler reactivity effects and a slight decrease in moderator reactivity effects were observed with chemical shim operation.

Noise measurements during this series were carried out mainly to determine the changes in nucleate boiling characteristics during pH tests. No conclusive relationship was observed between nucleate boiling characteristics and pH value of the coolant at the test conditions.

Techniques for on-line monitoring of the changes in nucleate boiling characteristics have been developed.

WCAP-3269-44 is the final report on these experiments.

WCAP-3269-44 SAXTON CHEMICAL SHIM KINETIC EXPERIMENTS -
V. RAJAGOPAL, G. E. SWEN, JUNE 1965

In order to study the effect of Chemical Shim Operation (boric acid in the main coolant for shim control) on the dynamic behavior of a reactor, a series of measurements was performed on the Saxton reactor using noise analysis and frequency response techniques.

Measurements of the fluctuations in nuclear power, coolant flow and temperatures have been performed at various reactor operating conditions. Oscillation tests consisted of producing pseudo-random, random or sinusoidal changes in reactivity through an oscillator and measuring the reactor flux and temperature response at different operating conditions by correlation techniques. For analysis, an analog power and cross-power density spectrum analyzer was employed.

The first study was an experimental investigation of the inherent fluctuations (noise) in the reactor to determine applications for the dynamic characterization of power reactors. A significant result of this study was to detect a well defined resonance in the power spectrum of the nuclear power fluctuations, and experimentally establish its relationship to nucleate boiling. Other useful applications of noise measurements were found in determining reactor and instrumentation time constants and flow characteristics.

The second study was an investigation of different techniques for obtaining frequency response data and of developing methods for obtaining kinetic parameters from the frequency response. Cross-power density spectral method with a single series of a pseudo-random input was found to be the best way to determine frequency response characteristics. With a simplified model, it was possible to determine kinetic parameters from the frequency response data. These measurements showed no evidence of any power reversible hideout under chemical shim operation.

WCAP-3269-10 STEAM GENERATOR WATER LEVEL CONTROL -
A. N. NAHAVANDI, A. BATENBURG, DECEMBER 1964

A combined digital-analog mathematical model for the dynamic analysis of vertical U-tube natural circulation steam generators is presented. The application of this model to the optimal design of a water level controller for a steam generating unit is demonstrated.

It is shown that a control system consisting of standard proportional and reset controls on water level deviation from a desired set point and the difference between the steam and feedwater mass flow rates can be successfully employed for the control of water level in such a plant. The optimum values, as well as the range of the controller parameter settings for which the steam generator exhibits a desired stable response, are determined.

WCAP-3269-45 REACTOR CONTROL AND PROTECTION SYSTEMS FUNCTIONAL
REQUIREMENTS AND PERFORMANCE - W. KING, MARCH 1965

This report presents the results of the development of control and protection systems for a 395 MWe closed-cycle water reactor with chemical shim control. The additional requirements and limitations for operation at 450 MWe stretched power are also included.

The functional requirements and performance of reactor control and protection systems are discussed in detail. The results of parametric study to demonstrate the sensitivity of operating margins as a function of design parameters are summarized.

In the control system study, emphasis is placed on the reactor plant transient responses to various specified plant disturbances such as $\pm 10\%$ step load changes, 50% step load rejection, loading and unloading, turbine trip and scram, and electrical system frequency dip. In the protection system study, emphasis is placed on the protection analysis for various postulated accidents, including: reactivity insertion, steam break, loss of load, and loss of flow.

WCAP-3269-54 AUTOMATIC CONTROL OF ROD POSITION WITH BORON
 IN A CHEMICAL SHIM NUCLEAR PLANT - A. BATENBURG,
 G. R. BUIS, M. A. MANGAN, J. A. STRITTMATTER,
 MARCH 1966

A model has been derived to describe the chemical and volume control system of the nuclear plant. The analog simulation of this system has been combined with an accurate simulation of a pressurized water reactor plant in order to study the behavior of the overall process. The important transport time delays associated with the chemical and volume control system were simulated on a digital computer.

The study showed that it is feasible to maintain the control rods at a required position, and remain at essentially constant power, by varying the concentration of boron in the reactor coolant system. This feature can be used to increase the load following capabilities of the reactor by minimizing core power distribution variations resulting from long term operation with control rods inserted. In order to maintain effective stable control, however, a control compensator is required to account for the relatively long system delays in effecting a desired boron concentration due to piping and storage delays in the chemical and volume control system.

WCAP-3269-31 DEVELOPMENT OF AUXILIARY SYSTEMS - H. J. CORDLE
FEBRUARY 1965

This report describes the development of auxiliary systems for a 395 MWe closed-cycle water reactor with chemical shim control.

Significant areas of development are discussed, and descriptions of the system conceptual designs are included for the Chemical and Volume Control System, the Sampling System, the Auxiliary Coolant System and the Safety Injection System. Requirements for waste disposal and vapor container ventilation for the reference plant are included.

WCAP-3269-27 FRICTION-SPEED CHARACTERISTICS OF THRUST BEARINGS
FOR REACTOR COOLANT PUMPS - J. BOYD, A. A. RAIMONDI,
OCTOBER 1963

A test program was formulated to determine friction-speed characteristics of Kingsbury-type water-lubricated thrust bearings in low speed ranges. The purpose of the investigation was to obtain data correlations to aid in predicting the existence of a hydrodynamic film (and adequate bearing lubrication) when a de-energized pump is subjected to reverse flow in a loop without check valves.

Tests of model bearing pads were conducted in a specially constructed apparatus under varying conditions of water temperature, bearing load, and speed. Results of the tests indicated a low probability of prolonged operation of a bearing in a speed range in which a full hydrodynamic film is not established.

WCAP-3269-32 DEVELOPMENT OF REACTOR COOLANT SYSTEM -
H. J. CORDLE, SEPTEMBER 1964

This report summarizes the significant areas of development leading to the final conceptual design of the Reactor Coolant System, which in a closed-cycle water reactor transfers heat from the reactor core to the steam producing secondary system. The system consists of identical heat exchange loops connected in parallel to the reactor vessel, each containing a reactor coolant circulating pump and a steam generator. The system also includes the pressurizer, pressurizer relief tank, connecting piping, valves and instrumentation necessary for operational control.

The basic objectives of this study were: (a) a plant conceptual design consistent with operating criteria for a chemical shim core, and (b) optimum system parameters and plant layout concepts with a view toward reduction of equipment cost. The development of conceptual designs for the various tools and equipment needed for refueling a core using the Rod Cluster Control (RCC) concept was also included in this study.

The reference conceptual design evolved as a three-loop plant using the more efficient controlled-leakage reactor coolant pumps and having a gross rated output of 395 MWe based upon a thermal power of 1176 MW.

WCAP-3269-60 REACTOR SAFETY ANALYSIS - S. N. ENRENPREIS,
J. D. McADOO, APRIL 1966

This report defines the objectives and principles of a chemical shim pressurized water reactor, and shows by analytical simulation the adequacy of the various safety features of such a reactor. Those aspects of system behavior which affect the potential for radiation exposure of persons or the potential for contamination of the environment are identified. The protection philosophy is discussed, but the study is limited to those aspects which are specific to the systems and components of the chemical shim PWR system typified by the San Onofre reactor. The hazards evaluation ends with the definition of a radioactive source released or contained as a result of the malfunction, and the protective action considered.

The report deals with malfunctions such as uncontrolled rod withdrawal whose initial influence on the reactor is the addition of core reactivity, with the class of mechanical accidents such as rupture of system piping resulting in unintentional discharge of radioactivity from sources external to the fuel, with the special problem of analysis to determine the amount and rate of release of energy from a zirconium-steam reaction which might ensue from a loss of coolant and core meltdown accident; and it includes a discussion of the identity and magnitude of radiation sources considered as the potential release in the event of protection failures in various modes.

WCAP-3269-38 THE ANALYTIC REPRESENTATION OF ROD CLUSTER CONTROL
ELEMENTS IN NUCLEAR DESIGN CALCULATIONS - W. J. EICH,
H. T. WILLIAMS, H. AISU, OCTOBER 1964

A series of reactivity measurements related to the Rod Cluster Control design was initiated at the Westinghouse Reactor Evaluation Center. Elements of this type consist of a cluster of 12 to 20 cylindrical Ag-In-Cd absorber rods, having approximately the same dimensions as fuel rods, which move as a unit within thimbles that are an integral part of every fuel assembly. The absorber rods are distributed within an approximately annular region, with two fuel rods between absorbers.

Control elements in previous light water reactors designed by WAPD were cruciform arrangements, either of poison slabs or of poison rods in linear arrays so closely packed as to be equivalent to slabs.

The absolute worths in a variety of critical configurations were derived using partial moderator height techniques. The consequent experimental values provided criteria against which methods of cluster representation could be tested.

Analytic schemes for the representation of Rod Cluster Control elements in reactor core design calculations were developed and compared with experiment. Theoretical bases and limitations of these methods are discussed in this report. Procedures were provided to expedite the application of these methods to design.

In the course of this comparison between analysis and experiment, a fundamental improvement was made in the analytic form used in the "worth" comparison.

WCAP-3269-39 CRITICAL EXPERIMENTS PERFORMED WITH CLUSTERED
AND UNIFORM ARRAYS OF RODDED ABSORBERS -
R. D. LEAMER, D. F. HANLEN, G. N. HAMILTON,
E. G. TAYLOR, NOVEMBER 1965

A series of critical experiments was undertaken in which the reactivity worths of clusters and uniform arrays of Ag-In-Cd absorber rods were measured in water-moderated uranium-oxide cores of low enrichment.

Experimental parameters included enrichment, lattice pitch, diameter of poison and fuel rods, and absorber rod cladding thickness. Reactivity worths were obtained by measuring the critical water heights of the test and reference cases, and integrating the differential water worth curve between those limits.

These experiments were performed using a variety of fuel including stainless steel clad fuel rods of 2.7 w/o, 3.7 w/o and 5.7 w/o enrichments, and Zircaloy-4 clad rods of 2.72 w/o enrichment.

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WCAP-3269-8 Hydraulic Tests of the San Onofre Reactor Model - G. Hetsroni (LRD-309)	11/6/64
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