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UC-80 Reactor Technology
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SAN ONOFRE NUCLEAR GENERATING STATION

REFERENCE CORE DESIGN

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Reactor Development

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I. ABSTRACT

The objective of the research and development program under Contract No. AT(30-1)-3269 is the performance of research and development 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 reactor core. 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.

Nuclear Power Plant to be
Situat^{ed} at Camp Pendelton, California

Reference Core Design - March, 1965

II. INTRODUCTION

This report presents the reference design for the San Onofre Nuclear Generating Station. The core of the San Onofre plant is to release 1346 MW of heat to produce from the plant a gross electrical output of 450 MWe.

The San Onofre reactor will be a closed-cycle, pressurized light water moderated and cooled reactor. The closed cycle feature provides absolute separation of the radioactive reactor systems from the essentially conventional turbine generator plant. This separation enhances the safety and reliability of the plant.

The reference core for this reactor is made up of fuel assemblies, composed of metallic tubes containing uranium dioxide, arranged to form a unit that is roughly cylindrical in shape. Type 304 stainless steel tubes will be used in the fuel assemblies of the first three regions (initial loading); with the successful completion of certain development work it is anticipated that the stainless steel cladding will be replaced with Zircaloy-4 for subsequent cycles.

The core is arranged to form a unit that is roughly cylindrical in shape. The core has an active fuel length of 10 feet and an equivalent diameter of 9.2 feet resulting in a length to diameter ratio of 1.09. The core contains 157 fuel assemblies as shown in Figure No. 1. The fuel in each region is of different enrichment.

The reference plan for refueling is in accordance with a modified out-in loading schedule. The basic out-in loading schedule calls for removing at the end of each cycle of core operation, the innermost region from the core, moving the intermediate region into its place, then moving the outermost region into the place assigned to the intermediate region, then installing new fuel around the core periphery. In the modified out-in loading schedule, some new or less depleted fuel is placed in the inner regions to improve the overall power distribution. In both the basic and modified out-in loading schedules, one-third of the total fuel assemblies are removed and changed at each refueling cycle.

This core arrangement, which is referred to as a "three-region, three-cycle loading", is superior in several ways to a uniformly enriched core. The ratio of peak-to-average power level is significantly reduced. This flattening of power distribution contributes to a substantial increase in amount of heat extracted per unit area of heat transfer surface and per unit weight of coolant flow. Also, the fuel economics are improved in the cycled core. The inventory of fissile material is reduced below that for a "batch-loaded" core.

The core loading is 143,600 lbs of uranium dioxide; 57,400 kg of uranium. For the equilibrium core the average burnup is 27,000 megawatt days per metric ton, resulting in a core life of approximately 27,600 effective full power hours, or 9,200 hours per refueling cycle. Based on a 0.9 plant factor, this gives a period of about 14 months between refueling shutdowns. The first core average burnup is 20,000 megawatt days per metric ton of contained uranium; the first cycle average burnup is 13,500.

Reactivity control is provided by neutron absorbing control rods (that are lifted out of the reactor to increase reactivity and fall into the reactor by gravity in the event of a reactor scram) and by a soluble chemical neutron poison (boric acid) in the reactor coolant stream. The concentration of the boric acid is varied as necessary during the life of the core to compensate for: (1) changes in reactivity which occur with change in temperature of the primary coolant from shutdown to the hot operating, zero power condition; (2) changes in reactivity associated with the relatively half-lived fission product poisons xenon and samarium; and (3) reactivity losses associated with depletion of fissile inventory and buildup of long-lived fission products (other than xenon and samarium). The mechanical control system (neutron absorbing control rod clusters) provides reactivity control for: (1) shutdown; (2) reactivity changes associated with the programmed increase in the average coolant temperature in the core above hot zero power; and (3) reactivity changes associated with the power coefficient of reactivity.

The mechanical design of the core is based on a rod type of fuel element packaged into fuel assemblies which may be separately loaded in the core. Each fuel assembly provides 16 open lattice positions (provided by the removal of fuel rod elements) to accept a control rod absorber design which has been developed under this contract. This new concept replaces the previous "standard" cruciform control element (absorber and follower) which travels in guide slots between fuel assemblies. In the new design, each control element consists of a cluster of a number of cylindrical absorber rods, which are connected at the top end by a spider arrangement which is coupled to the control rod drive shaft. Each rod in the cluster, however, moves in its own tubular guide, which is located symmetrically within a single fuel assembly as shown in Figures 1 and 2. The guide thimbles replace fuel rods within the fuel assembly lattice.

III. MECHANICAL DESIGN

A. General

The San Onofre core will approximate in shape a right circular cylinder 9.2 ft. in diameter and 10 ft. in height giving a length to diameter ratio of 1.09. The core will consist of 157 fuel assemblies of identical design. These may be thought of as being divided into three concentric regions of 52 fuel assemblies each plus a central fuel assembly. The fuel assemblies are square in cross section, and the fuel rods composing the fuel assembly are arranged in a close packed square lattice. The lattice array is 14 x 14; of the 196 lattice positions, 180 are occupied by fuel rods and 16 positions are vacant to accommodate the RCC control elements.

Figure 5 is a cross section of the San Onofre Reactor showing the three loop vessel with the inlet and outlet nozzle arrangement, the thermal shield and core configuration. Indicated in the core configuration are the 45 control rod locations which match the penetrations in the vessel head provided for the top-mounted control rod drive mechanisms. An enlarged view in Figure 1 also indicates the arrangement of the fuel rods and guide thimbles, within four typical fuel assemblies. Another enlarged detail indicates the relative diameters of the absorber rods and their guide thimbles.

The 157 individual, replaceable fuel assemblies will be held in position between a lower core plate and upper core plate. The weight of the fuel assemblies plus the additional down loading of four Inconel

springs at the top of the assembly is carried on four integral short legs. Their position in the core (in the horizontal plane) is determined by the positioning of two holes in two of the four support legs which mate with two pins mounted in the lower core plate. Angular positioning of the upper end of each fuel assembly is assured by two integral pins mounted in the top core plate which are inserted in two mating holes in the fuel assembly top nozzle when the upper core plate is lowered in place. The four Inconel springs mounted to the top nozzle are also pre-loaded at this time to prevent the individual fuel assemblies from lifting due to hydraulic flow. Holes are provided in the core plates for the coolant inlet and discharge. In the upper core plate, the coolant discharge holes also provide space, in the proper locations, for admitting the rod cluster control elements; no similar openings are necessary in the lower core plate since the control elements do not have followers.

There are in the reference design 45 (nominal) rod cluster control elements. Each cluster (element) consists of 16 individual neutron absorbing rods. The axis of the control cluster is parallel to the vertical axis of the core; each rod cluster control element will be actuated by a separate control rod drive mechanism above the core. Reactivity of the core is increased by lifting the control cluster out of the core in a vertical direction. For refueling purposes, the control rod drive shafts are disengaged from the rod cluster elements leaving the control clusters fully inserted in their assemblies. After

removal of the upper support package, the fuel assemblies may be removed individually. Where a fuel assembly contains a rod cluster element, the rod cluster element is removed with the fuel assembly and is either cycled to a new control position or the control rod removed and inserted into another assembly.

The core consisting of 157 fuel assemblies is surrounded by a form-fitting baffle which confines all but a small portion of the upward flow of coolant within the fuel bearing zone. Outside the baffle, and surrounding it, there is a core barrel. A small amount of coolant is allowed to flow in an upward direction between the barrel and the baffle for cooling. The barrel is the structural member which sustains the differential hydraulic pressure between the downward flow of coolant outside the barrel and the upward flow of coolant within the core. The water between the form-fitting baffle and the barrel acts as a neutron reflector.

B. Fuel Assemblies

Each fuel assembly has a total length of approximately 137.25 inches and an active fuel length of 120 inches. The fuel assemblies are square in cross section, 7.76 inches by 7.76 inches. See Fig. 2.

The basic element in the fuel assemblies is the fuel pellet. This pellet is 0.383 inch in diameter and approximately 0.6 inch long. The pellets will be manufactured by sintering a powder compact of uranium dioxide to 94/94/93% of theoretical density,* (three region core). After sintering, the individual pellets will be centerless ground to

* 94% for the center and intermediate regions and 93% for the outer region.

obtain the required tolerance on the diameter. The ends of the pellets will be dished to accommodate the high rate of expansion of the center of the fuel pellet due to the temperature rise thru the pellet during power operation. Three enrichments of uranium will be used in the first core; these enrichments will be in the ratio of 7.5/8/9. Approximately 200 pellets are loaded into the stainless fuel rod to yield the active fuel length of 120 inches.

The reference design fuel clad is a tube 0.422" O.D. made from type 304 stainless steel cold worked approximately twelve percent. The design wall thickness of 0.0165" assures that the cladding will be "free standing", that is, it will not collapse onto the fuel pellets at the design pressure of 2500 psia at reactor operating thermal conditions. All clad tubing is subjected to extensive quality control procedures during manufacturing to assure absence of defects, including ultrasonic testing, chemical ladle analysis, and check analysis of the flat rolled stock or tube of each heat. Additional characteristic requirements are verified by "Room Temperature Tensile Flattening and Reverse Flattening, Interangular Corrosion, and Microstructural Tests".

The fuel is retained in the clad by type 308 stainless steel end plugs welded to the cladding to provide a hermetic seal to contain fission gases released during core operation. A fission gas space of 5-5/16" length or 0.880 cu. in. is provided at the top of each fuel rod to limit fission gas pressure and to permit axial differential expansion of the fuel within the cladding. The fuel is held in position against the lower end plug by means of a spring between the upper end of the fuel column and the top end plug. The spring is thermally insulated from the fuel column by one or more aluminum oxide (Al_2O_3) discs.

The fuel rods will be assembled on a square lattice with a center to center distance of 0.556 inches to make a fuel assembly. Only one design of fuel assembly will be used having 196 lattice positions in a square array, 14 x 14. Fuel rods will be omitted from each fuel assembly, as shown in Fig. 1, to provide for the passage of the 16 absorber rod elements of the rod cluster control, thereby reducing the number of fuel rods from 196 to 180. The passage-ways for the individual neutron absorber rods of the rod cluster control element are defined by guide thimbles which are 16 in number.

The guide thimbles are 0.535" O.D. by 0.012" wall thickness and are made from type 304 stainless steel. The lower ends of the guide thimbles are sealed by means of end plugs and are reduced in diameter to 0.470" over the lower 20-13/16" to provide the proper clearance for shock absorber action on the control rods. The thimbles are perforated for a distance of approximately 1/2" above the lower 21-5/16" to permit free flow of water out of the thimble during scram and to permit adequate flow for cooling of the control rods within the thimbles during full or partial insertion in the core. The thimbles are anchored to the lower end plate of the fuel assembly by welding. The RCC Fuel Assembly is shown in Figure 2.

The fuel rods and rod cluster element guide thimbles are positioned in the lattice by means of grid assemblies of an "egg-crate" configuration. Each unit cell of the grid or 180 of the 196 positions in the lattice contain two holding springs on adjacent sides with two dimples on each of the two opposite adjacent sides. The two dimples opposite each spring serve a

dual purpose by providing a definite fixed stop or centering of the fuel rod in the unit cell, and also providing a dual support, similar to a beam simply supported at both ends with a fixed load at mid-point being applied by the holding spring. This provides a six point suspension for each rod in each unit cell. The remaining 16 positions in the lattice have four radial contoured tabs each, two on the upper and two on the lower side of each unit cell opposite each other. These tabs provide the anchor for each guide thimble to the grids and are welded to the thimble. The grid straps are made from two thicknesses of material, the internal straps all being from 0.016 inch thick with the four external straps of 0.020 inch thick material. Age hardened Inconel 718 was chosen for the grid material because of its more uniform and higher spring rate for the holding springs.

In fabricating the fuel assemblies, seven grids will be attached to the guide thimbles by welding as previously described. The top and bottom grids are spaced so that the unsupported overhang of the fuel rod will be less than $3/4$ inch. The second grid from the bottom is approximately $17-5/8$ inches with the remainder of the grids spaced evenly at $21-1/3$ inches. The bottom nozzle is then attached and squared to the 16 guide thimbles by inserting a guide pin at the end of each thimble into a matched hole in the nozzle and plug welding. At this point 180 fuel rods will be loaded into the assembly from the top end with a hydraulic or mechanical ram device. After completion of loading the top nozzle is slid in place over the 16 guide thimbles squared up and welded to each thimble with a fusion weld of the thimble to a concentric sleeve attached to the nozzle plate. Final impact loads from the scram stroke are absorbed in this

top plate, with the initial loads being transferred hydraulically to the bottom nozzle thru the thimbles.

The advantage of this construction is minimizing thermal bow of the fuel assembly. Since the holding springs are not physically bound to the fuel rods, the rods are free to expand axially between end nozzles and grids thereby largely eliminating thermal bowing due to differential expansion of the fuel rods which arises from power gradients across the fuel assembly. Additional advantages are the use of cold worked stainless steel cladding with significant reduction of wall thickness. This is attainable due to the higher strength of the cold worked condition as contrasted with the annealed condition required in a brazed Yankee-Rowe type fuel assembly.

C. Control Elements

The control elements for the San Onofre reactor core each consist of a cluster of 16 cylindrical absorber rods attached to a "spider" at the top and free to move into or out of a unit fuel assembly. The cylindrical absorber elements are made of 80% Ag 15% In 5% Cd alloy rods with a diameter of 0.390 inches by 10 feet long hermetically sealed in "free standing" stainless steel tubes .432" O.D. by .019" wall.

Each absorber element in the cluster moves up and down in its own stainless steel tubular guide thimbles which are part of the fuel assembly. The 16 absorber elements are located in a symmetrical pattern within a single fuel assembly. As noted above, the guide thimbles replace fuel rods within the fuel assembly lattice and the guide thimbles are positioned by the grid assembly.

All the absorber rods of one control cluster are attached to a spider connector above the core so that all absorber elements of one control cluster move together as shown in Fig. 3. The elements of the cluster are guided above the core to prevent hydraulically induced vibration and to retain the proper orientation of the cluster elements in the withdrawn position. When the control cluster is withdrawn from the core, water replaces the absorber elements in the guide thimbles, there are no follower elements. The ends of the control rods still project a short distance into the fuel assembly guide thimbles in the fully withdrawn position to prevent excessive bypass flow through the thimbles and to assure proper alignment for insertion during scram.

The spider consists of 12 radial blades attached to a hub. Pods are attached to the blades to match the guide thimble locations in the fuel assembly. The tops of absorber rods are attached to the pods on the blades. The top portion of the hub is bored out to accept the coupling of the drive shaft of the control rod drive mechanism. The bottom portion of the hub contains an energy absorbing spring pack to elastically absorb the energy associated with the terminal velocity of

approximately 2 ft/sec at the end of dashpot travel. This device impacts on the upper retaining plate of the fuel assembly.

There are several advantages associated with this RCC element. In making absorber rods dimensionally similar to the fuel rods, similar flexibilities, deflections, forces and design conditions prevail on all the in-core elements. Potential misalignments and relative rotations between fuel assemblies do not affect the control rod clearances. In contrast, the usual, relatively rigid cruciform between four adjacent fuel assemblies must allow for the relative motion, twist and misalignments of each of them in the form of larger water gaps. Distribution of absorber material within a fuel assembly does not disturb the basic lattice pitch and results in less neutron flux "water gap" peaking. The net result is that control rod followers may be eliminated without adverse affect on local power distribution.

Other advantages of this rod cluster control scheme are:

1. Shorter reactor vessel due to omission of control rod followers.
2. Reduction in the number, mass and complexity of control rod and drive train parts.
3. Elimination of mechanical dashpot.

4. Simplification of fuel assemblies.

5. More control rod worth per unit weight of absorber material.

Figure 4 illustrates the comparable cruciform core cross section for the San Onofre reactor. It will be noted that the fuel assembly size and control rod locations are strictly compatible with the RCC design. The cruciform design utilizes 156 instead of 157 fuel assemblies in a slightly different core pattern than the RCC core. The compatibility feature made it possible to proceed with the reactor vessel head fabrication before the decision to use the RCC design was made.

D. Internals

The internals of the reactor as shown in Figure 5 can be divided into two major components:

1. The core barrel assembly -- this portion of the internals supports the core and remains in the reactor pressure vessel during refueling.
2. The upper internals structure -- this portion contains the control rod guide tubes, upper core plate and in-core instrumentation package. This structure must be removed prior to any fuel assembly removal.

The core barrel assembly is of an all welded Type 304 stainless steel construction and consists of a flange, a core barrel and baffle assembly, and a lower support structure. The flange is welded to the core barrel and transmits the core loads to the reactor vessel at the vessel flange region. The core barrel, because of its length, is rolled and fabricated in sections. The upper section of the core barrel contains exit nozzles which line up with the exit nozzles of the reactor vessel and which direct the coolant flow from the upper plenum to exit without mixing with the inlet flow. The lower section of the core barrel consisting of two sections welded together houses the baffle assembly, which defines the boundaries of the core. The lower core barrel section is then welded to the upper portion of the barrel and to the lower core support structure to form the core barrel assembly.

The lower support structure consists of the lower core plate attached to a lower support casting through extension columns. The lower core plate positions the lower ends of the fuel assemblies and transmits the core loads through the columns to the support casting, and back into the core barrel. The support casting is far enough removed from the core so that relatively thick sections, perforated with combination coolant flow entrant and cooling holes, can be used and still maintain relatively low thermal stresses from gamma heating. Sandwiched between the lower core support casting and the lower core plate (shown on the preliminary drawing, Figure 5) is a hydraulic distribution plate attached to the columns to improve the inlet flow distribution.

The upper internals structure, which is removed at each refueling, consists of an upper support plate, control rod guide tubes, an upper core plate and an instrumentation assembly. The upper support plate rests on top of the core barrel assembly flange and carries the weight of the guide tubes, the upper core plate and the instrumentation assembly. The control rod guide tubes, which are bolted to the upper support plate are located directly over the 45 fuel assemblies which contain control rods.

The lower 30 inches of the guide tubes form an open, slotted structure to permit the fuel assembly flow to exit from the tubes. The slots are naturally formed by the full length guides which sheath the RCC control element to prevent it from vibrating due to cross flow as the coolant exits in a radial direction. These same sheaths are also slit vertically on the innermost side to allow the spider to pass thru. Above the slotted area, the guide tube contains guide plates approximately 6 inches apart to locate and guide the control rod in this upper region. These guides act to prevent lateral buckling displacements of the long slender control rod elements.

The upper support package is completed by bolting the upper core plate to the guide tubes. This plate, as previously mentioned, positions the fuel assemblies relative to the core barrel and guide tubes. Proper angular orientation between the vessel, the vessel head, and the various internals is accomplished by means of dowel pins or keys. This assures

accurate alignment of the control rod drive mechanisms, guide tubes and fuel assemblies.

The lower end of the core barrel assembly (at the lower core plate level) is radially positioned in the reactor vessel by the thermal shield support lugs. Rigid positioning is obtained after the primary system is brought to temperature by differential expansion between the stainless steel internals and the carbon steel pressure vessel. This feature minimizes barrel and flange stresses during a possible earthquake and due to hydraulic forces during normal operation.

The principal loadings to which the internals are subjected during reactor operation are:

- (a) Core weight
- (b) Control rod scram load
- (c) Coolant flow pressure drop

In addition to the above loadings, other conditions exist to cause stresses in the internals components; principal among these is gamma heating of the components. It will be noted that the internals structure are all designed to remove the major load carrying members as far from the core as feasible to take maximum advantage of the shielding afforded by the water. All structures close to the core are made in thin sections to minimize gamma heating stresses.

Primary and secondary source rods will be located in guide thimbles in a fuel assembly which does not have a control rod.

IV. CORE THERMAL-HYDRAULIC DESIGN

The basic objective of the thermal-hydraulic design is to remove the energy generated in the core at the maximum coolant temperature without exceeding safe fuel clad temperatures at any position in the core, even under the most severe postulated transient conditions.

The main thermal-hydraulic criterion for core design is the establishment of the point of Departure from Nucleate Boiling, commonly abbreviated as DNB, and of the safety margin to be used in the design. DNB is undesirable because once it occurs, a vapor film insulates the fuel cladding from the reactor coolant. The consequence is a rapid rise in the cladding temperature, and the result may be melting of the cladding (burnout).

The measure of the DNB safety margin is the DNB ratio. Two DNB ratios are normally considered: The enthalpy rise DNB ratio and the heat flux DNB ratio. The enthalpy rise DNB ratio (ΔH -DNBR) is evaluated by first calculating the enthalpy rise that would cause DNB in the hottest channel under specified design conditions (usually an overpower condition), and then by dividing this value by the actual design enthalpy rise in the hottest channel. The heat flux DNB ratio (q'' -DNBR) is evaluated by first calculating the heat flux that would cause DNB under specified design conditions, and then by dividing this value by the actual design heat flux value.

Actual design values of enthalpy rise and heat flux in the hottest channel take into account non-uniformities in power distribution.

These non-uniformities are expressed quantitatively by "hot channel factors", which are the peak-to-average ratio of the parameter of interest. The enthalpy rise hot channel factor, $F_{\Delta H}$, is defined as the ratio of the enthalpy rise in the hottest channel to the enthalpy rise in the average channel. The heat flux hot channel factor, F_q , is defined as ratio of the heat flux at the hottest point to the average heat flux in the core.

The hot channel factors, $F_{\Delta H}$ and F_q , are each the product of a number of subfactors. These subfactors are usually divided into two principal groups, engineering and nuclear. The engineering subfactors are basically statistical in nature and account for mechanical non-uniformities in the core and fuel rod. These non-uniformities include variations in fuel pellet diameter; in diameter, pitch, and straightness of fuel rods; in fuel density; and in fuel enrichment. The subfactors related to flow distribution, redistribution, and mixing are not of a statistical nature. The nuclear subfactors take into account the overall power distribution in the radial and axial directions, as well as local peaks caused by the presence of control rods and other disturbing factors.

The hot channel factors for the reference design are:

<u>Engineering</u>	<u>F_{ΔH}</u>	<u>F_q</u>
1. (a) Fuel pellet diameter, density, and enrichment	1.08	1.04
(b) Fuel rod diameter, pitch and bowing		
2. Plenum maldistribution	1.03	-
3. Flow redistribution	1.05	-
4. Flow mixing	0.92	-
<u>Total Engineering</u>	1.075	1.04
<u>Nuclear</u>		
5. Radial, max.-to-average	1.75	1.75
6. Axial, max.-to-average	-	1.78
<u>Total Nuclear</u>	1.75	3.11
<u>Total Factor</u>	1.88	3.23

The calculated values of enthalpy rise and heat flux DNB are obtained from "DNB Correlations". These correlations have been established by analysis of over 3,000 data points collected during many investigations by Westinghouse and others of the phenomena of heat transfer to water.

Two separate correlations are used to determine DNB ratios. One (ΔH_{DNB}) is applicable when the coolant enthalpy is at or above the saturation enthalpy corresponding to the coolant pressure; and the other ($q''\text{-DNB}$) is applicable when the coolant enthalpy corresponds to the coolant pressure.

These two correlations are:

(a) DNB Correlation in Quality Region

$$\Delta H_{\text{DNB}} = 0.529 (H_f - H_{\text{in}}) + (0.825 + 2.36e^{-204D_e}) H_{fg} e^{-1.5 G/10^6} \\ - 0.41 H_{fg} e^{-0.0048 L/D_e} - 1.12 H_{fg} \rho_g / \rho_f + 0.548 H_{fg}$$

(b) DNB Correlation in Subcooled Region

$$q''_{\text{DNB}} = (0.23 \times 10^6 + 0.094G) (3.0 + 0.01 \Delta T_{\text{sc}}) \\ (0.435 + 1.23e^{-0.0093 L/D_e}) \times (1.7 - 1.4e^{-a})$$

where

$$a = 0.532 \left[(H_f - H_{\text{in}}) / H_{fg} \right]^{3/4} (\rho_g / \rho_f)^{-1/3}$$

By basing design values of enthalpy rise and heat flux on these hot channel factors and by basing calculated values of DNB enthalpy rise and heat flux on correlations having a high confidence level, the resulting DNB ratio is conservatively established. To give a further margin of safety during reactor transients and to provide a margin for uncertainties, the reactor is designed so that the ratio at the design overpower is not less than 1.25.

A second thermal-hydraulic criterion for reactor core design is that there shall be no melting of the uranium dioxide fuel. At the reactor power of 1346 MWt, the average rate of heat release in the fuel is 4.64 kw per linear ft of fuel rod. With a value of 3.23 for F_q , the maximum rate of heat release is 15 kw per ft. Experimental evidence has been accumulated which shows that, for the fuel rod diameter chosen, center melting does not occur until the rate exceeds 20 kw per ft.

Thermal and hydraulic design parameters are given in Appendix A.

A third design criterion for design of the SCE reference core is that bulk boiling is permissible in the hot channel during steady state conditions, and that, the quality at the outlet of the hot channel not exceed 2%. In the reference design, the coolant at the outlet of the hot channel is 2.2°F below saturation temperature (642.8°F) at 2100 psia. The coolant temperature rises in the core and hot channel were computed using an effective heat transfer flow rate of 70×10^6 lb/hr, which is 91% of the total flow rate, and an enthalpy

rise hot channel factor of 1.88. The 9% factor is used to account for the coolant flow that completely bypasses the core and that is not effective for direct heat transfer of the fuel rods.

Hydraulically, the SCE reactor is a single-pass, upward flow core. Coolant enters the reactor vessel from the primary system piping nozzles at the top and flows downward through the annuli between the core barrel and the vessel wall. The thermal shield divides this flow into two channels. At the bottom of the vessel, the coolant changes direction by 180° and flows upward through the core. The reactor vessel outlet nozzles are also located at the top of the vessel.

The coolant velocity along the fuel rods is approximately 13.1 feet per second. The core pressure drop is about 18.8 psi. This includes, in addition to fuel rod friction, the pressure loss caused by the 7 grids and the fuel assembly end plates. The total pressure drop through the reactor vessel was estimated to be 33 psi.

V. NUCLEAR DESIGN

The reactor of the San Onofre Nuclear Generating Station is a closed cycle pressurized water reactor with a nominal rating of 1346 MWt. The reactor will employ non-uniform, multiregion loading and fuel cycling with shim control by chemical poison. Three major advantages of this mode of operation are:

- (1) More uniform power distributions.
- (2) Reduced fissionable material loading for a given fuel burnup.
- (3) Reduced control rod requirements.

The fuel loading scheme of the reference first core is based upon a division of the core into three equal volume concentric regions with preferential fissionable material loading towards the outside of the core in the ratio of 7.5:8:9.

In a core in which all the control is supplied by control rods, the maximum burnup is limited by the maximum number of rods that can be inserted in the core. When chemical shim is used to control the excess reactivity required for burnup, however, this limitation no longer exists. For increased burnup in a chemical shim core the extra control is achieved by the addition of more boron, and the number of control rods is not affected except for a small decrease in rod worth with increasing boron concentration. This restriction being removed, much higher burnups are possible in a chemical shim core. This fact allows the average first core burnup in SCE to be 20,000 MWD/MTU instead of approximately 13,000 MWD/MTU if control were entirely by rods.

In order to achieve 20,000 MWD/MTU for the first core, the first three cycles will be scheduled for burnups of approximately 13,000, 7,000 and 9,000 MWD/MTU, respectively. This burnup schedule will be continued through to the equilibrium cycle and an equilibrium discharge burnup of about 27,000 MWD/MTU will be achieved. In order to obtain the scheduled burnups, the fuel enrichments (weight per cent U-235) in the first cycle are estimated to be 3.2 w/o, 3.4 w/o and 3.8 w/o in the center, intermediate and outer core regions, respectively. The feed enrichment for all following cycles is expected to be 4.2 w/o for a steel clad core.

As fuel is cycled (post core reloading), the gradation of fuel burnup supplies an effective, non-uniform fissionable material loading to provide power flattening. It has been found that the description of region boundaries for non-uniform loading is dependent on the core burnup. For burnups as high as 27,000 MWD/MTU, the simple, three-region loading described above for the first cycle is not adequate with normal out-in fuel movement in an equilibrium cycle. This is because the central region becomes so heavily depleted for long burnups that the power sharing by the inner region is not acceptable. The maximum-to-average power density would increase by more than 30% from the first cycle to the equilibrium cycle with the simple, three-region loading.

Because of the high burnup that the fuel in the center region has seen in its two previous locations in a normal out-in movement, it is not sufficiently reactive. In order to improve the power distribution in the equilibrium core, the center region is made more reactive. This is accomplished by moving some of the highly depleted fuel towards the outside of the core and some of the lesser depleted fuel towards the center of the core. If, at the beginning of each cycle, several slightly depleted fuel assemblies are placed near the center of the core, replacing the more highly depleted assemblies, the power sharing is quite good as shown in Figure 6. This interchange of fuel between regions will be a function of burnup and, therefore, each transition cycle may require a different form of interchange.

An initial effective multiplication of 1.254 is calculated for the cold, clean core at beginning of life. Of this total, 0.049 is allotted for the increment from cold to hot zero power conditions, 0.020 for the change from zero to full power, and 0.033 for equilibrium poisons. The remaining 0.152 is provided for load following control and for long term depletion of reactivity with burnup.

Control to render the reactor subcritical at temperatures below the operating range is provided by chemical poison (boron). It is estimated that 3900 ppm boron will provide adequate shutdown for fueling the core at room temperature with all rods in the core. This concentration is equivalent to approximately two per cent boric acid (H_3BO_3), and is well within the solubility limits at ambient temperature. Other characteristic boron concentrations are included in the table of Nuclear Design Parameters, Appendix B. The values are estimates only; exact concentrations will be determined during core design and during pre-operational full core critical tests.

Even though boron is used to control the excess reactivity for burnup and the reactivity associated with equilibrium xenon and samarium in the San Onofre core instead of control rods, control rods do serve an important function. Since the soluble poison concentration requires a significant time for adjustment, it cannot be employed for the compensation of rapid reactivity variations. The reactivity requirements which must be controlled by rods include the variation with power resulting from Doppler broadening of U-238 resonances, the variation

which will arise from the programmed T_{avg} with load, as well as the normal shutdown margin with the assumption of one control rod stuck in the full out position. These requirements are in excess of any operational control band requirements which may be imposed. In addition, control rods may be employed for limited power shaping to resolve fuel management problems, particularly those involving non-equilibrium cycles. The number of control rods that will fulfill these requirements in the San Onofre core is 45.

The temperature coefficient in a core controlled by chemical shim is less negative than the coefficient in a rodged core. This results, first, from the fact that control rods contribute a negative increment and, in a chemical shim core, the rods are nearly out of the core. Secondly, since water density decreases with an increase in temperature, the chemical poison density is decreased concurrently which gives rise to a positive component of the moderator temperature coefficient. The second effect is directly proportional to the poison concentration and, since the chemical poison is used to control the excess reactivity for burnup, the beginning of life temperature coefficient becomes less negative as the design burnup is increased.

It has been found that the stability of the core is not adversely affected even with a zero or positive moderator temperature coefficient as long as the net coefficient (Doppler plus moderator) remains negative or only slightly positive. The Doppler (fuel temperature) coefficient, which is the basis for determining the power coefficient is an expression of the Doppler broadening of absorption resonances in U-238 with increasing temperature. The value of the fuel temperature coefficient will range from -1 to $-2.5 \times 10^{-5} (^{\circ}\text{F})^{-1}$. The range of the moderator temperature coefficient is from + 0.1 to $-3.5 \times 10^{-4} (^{\circ}\text{F})^{-1}$. The moderator coefficient becomes more negative as fission product poisons are built up in the core, and therefore the most positive condition occurs at the beginning of the first cycle.

The trend toward an algebraically more positive moderator coefficient (smaller negative coefficient) affects plant operation in two respects. Normally, part of the reactivity insertion due to a partial load reduction is compensated by the negative temperature coefficient of the moderator. As this coefficient becomes ineffectively small or slightly positive, more rod motion is required to compensate for load reduction. A failure to compensate does not mean an unsafe condition as the consequence is to initiate a scram and shut the plant down. Also it is necessary to maintain a small moderator temperature control band at full power. Since a stepping type control rod mechanism is employed, the finite reactivity insertion per step will result in the temperature overshooting the control band if the

temperature coefficient is too small in magnitude. The consequence of this condition is control rod mechanism hunting in which the rod group alternates back and forth between two step positions. The large thermal inertia of the primary loop results in the period of this oscillation being on the order of several minutes and no undue hardship is imposed upon the mechanism.

The safety of the San Onofre plant is not affected by a possibly small positive moderator temperature coefficient because of the long time delay in heat transfer from the oxide fuel to the coolant. The time constant is on the order of five seconds. The total scram time is not more than half that value or 2.5 seconds, so that the response to an abnormal transient is determined to a large extent by the prompt Doppler coefficient which is unaffected by the addition of chemical poison.

Other coefficients of reactivity are the void and pressure coefficients. The void coefficient defines the reactivity response to an increase in steam voids in the core lattice. The expected value is in the range + 0.01 to - 0.3%k/% void. The pressure coefficient is a positive reactivity effect, but the magnitude is quite small. The expected range is from 0.2 to $3.5 \times 10^{-6} (\text{°F})^{-1}$.

The delayed neutron fraction and prompt neutron lifetime are important parameters in determining the kinetic behavior of the core, particularly when analyzing the response to large reactivity insertions. At the beginning-of-life the delayed neutron fraction is 0.007 and the prompt neutron lifetime is 1.8×10^{-5} seconds. Since the fraction of delayed neutrons in Pu-239 fission is somewhat smaller than in U-235 and U-238 (fast) fission, the effect of plutonium buildup with core life is to reduce the average delayed neutron fraction and reduce the reactivity insertion necessary to incur prompt criticality. The minimum value of the mean delayed neutron fraction is estimated to be 0.5 per cent at the end-of-life of the equilibrium cycle. The effects of poison buildup on the prompt neutron lifetime is to reduce it to 1.2×10^{-5} seconds.

In relatively large thermal reactor cores, an instability due to oscillatory redistribution of xenon is theoretically possible. Criteria for the occurrence of this phenomenon are: (1) high flux levels such that burnout of xenon is important relative to decay, and (2) large core size such that the reactivity effects of leakage is small compared to that of xenon poison. In the San Onofre core, xenon oscillations will be strongly damped by the substantial local negative power coefficient, and it is expected that analysis will demonstrate that significant fluctuations of this type cannot occur.

Important nuclear design parameters of the core are shown in the table of Nuclear Design Parameters, Appendix B.

VI. CORE MATERIALS SELECTION

A. Fuel Rod Materials

The fuel will be in the form of pressed and sintered UO_2 pellets, produced to close tolerances on chemical and isotopic analysis. The control of pellet density is very important, as density is related to maximum fuel burnup. As burnup increases, swelling of the UO_2 from fission fragment accumulation increases, and without proper design and control of certain parameters, expansion of the fuel against the cladding could cause clad rupture. Figure 7 shows the original percent void volume necessary to prevent distortion and possible rupture of the cladding for various amount of peak fuel burnup. The peak burnup anticipated in the reference design is between 45 and 50,000 MWD/MTU. This corresponds to a total original void volume of 8%. The 8% will comprise 2% diametral fuel-clad gap and 6% voids in the pellets. Therefore, the density of the fuel pellets will be specified as 94% of theoretical density.*

The cladding will be type 304 stainless steel tubing cold-worked approximately 12% by a final draw. This will produce a cladding with about 85,000 psi yield strength and good ductility. The durability of cold-worked type 304 fuel cladding in pressurized water reactor environment has been well established by reactor service and in-pile experiments. There have been no failures, such as stress corrosion cracking of this type of material in service or simulated service where the coolant chemistry has been under the type of close control normal to closed-cycle water reactors.

* 94% for the center and intermediate regions and 93% for the outer region.

B. Control Rod Materials

The control rods will be silver-indium-cadmium alloy, encapsulated in cold-worked type 304 stainless steel tubes. The 80% silver, 15% indium, 5% cadmium alloy has a nuclear worth equivalent to that of hafnium. The silver and indium through their complimentary resonance peaks control the epithermal spectrum, while the cadmium has a high thermal neutron cross section. Furthermore, it has been shown that the silver-indium-cadmium alloy does not swell or undergo any deleterious dimensional changes as a result of irradiation.

However, this alloy is subject to slight, but significant corrosion in reactor coolant. The chief disadvantage resulting from corrosion is the sloughing off of particles of highly radioactive material. Therefore, the absorber rods will be hermetically sealed in stainless steel tubing.

The grids for positioning the fuel rods and control rod guide thimbles consist of a brazed "egg-crate" configuration. The grid material is Inconel-718, selected for its good high temperature strength properties. Each lattice position contains two dimples on each of two adjacent sides and two holding springs on each of the other two adjacent sides.

The brazing alloy is a nickel-chromium-phosphorous alloy, MICROBRAZE-50, which has excellent strength and corrosion resistance.

NOTE:

1. .850 DIA. THRU SPIDER ASSEMBLY & ABSORBER ROD USING PLOT HOLE IN SPIDER AS A TEMPLATE.
2. THIS ASSEMBLY MUST PASS THRU INSPECTED AND ACCEPTED ASSEMBLY GR. 1 OF DWG 685ED07 WITHOUT RESTRICTION OR BINDING.

BILL OF MATERIAL				REV	
ITEM	QTY	DESCRIPTION	REV	DATE	BY
1	1	SPIDER ASSEMBLY	682015 G4		
2	1	ABSORBER ROD	74C915 G4		
3	1	SPRING	500803 IT		
4	1	SPRING TENSION BOLT	500804 IT		
5	1	SPRING RETAINER	500805 IT		
6	1	SPRING	500806 IT		
7	1	DOWEL PIN	AISI - 504 SST

Figure 3. Spider and Control Rod Assembly

FIGURE 7 MAXIMUM UO₂ FUEL BURN-UP AT ONSET OF CLAD SWELLING
VS INITIAL FUEL ELEMENT VOID VOLUME AT OPERATING
TEMPERATURE.

APPENDIX A

THERMAL AND HYDRAULIC CHARACTERISTICS OF

THE SAN ONOFRE REFERENCE CORE

February 1965

Total Heat Output, MW	1346
Btu/hr	4595×10^6
Heat Generated in Fuel, Per cent	97.4
System Pressure, Psia	
Nominal	2100
Minimum in Steady State	2075
Hot Channel Factors in Steady State	
Heat Flux, F_q	3.23
Enthalpy Rise, $F_{\Delta H}$	1.88
Coolant Conditions:	
Total Flow Rate, lbs/hr	76.9×10^6
Effective Flow Rate for Heat Transfer, lbs/hr	70×10^6
Flow Area for Heat Transfer Flow (Unit Cells), Ft^2	33.2
Average Velocity Along Fuel Rods, ft/sec	13.1
Average Mass Velocity Along Fuel Rods, $lbs/hr-ft^2$	2.1×10^6
Coolant Temperature, °F	
Nominal Inlet	552.8
Maximum Inlet Due to Instrumentation	
Error and Dead Band	556.8

Average Rise in Vessel	44.8
Average Rise in Core	48.8
Average in Vessel	575.2
Average in Core	577.2
Nominal Outlet of Hot Channel	637.8
Maximum Outlet of Hot Channel	640.6
Maximum Outlet Enthalpy of Hot Channel, Btu/lb	679.3
Saturation Enthalpy at Minimum Steady State	
Pressure, Btu/lb	680.4
Average Film Coefficient, Btu/hr-ft ² °F	5080
Average Film Temperature Difference, °F	28

Heat Transfer

"Active" heat transfer surface area, ft ²	
Fuel Assembly Rods	31,200
Average Heat Flux, Btu/hr-ft ²	143,400
Maximum Heat Flux, Btu/hr-ft ²	463,100
Maximum Thermal Output, kw/ft	15
Maximum Rod Surface Temperature at Nom. Pressure, °F	648

DNB Ratios (Normal Cell)

Local q" - DNBR at 100% power	1.82
ΔH - DNBR at 100% power	*

* Not applicable; no boiling in channel

Fuel Rod (Cold Dimensions)

Outside Diameter, inches	0.422
Clad Thickness, inches	0.0165
Diametral Gap, inches	0.0055
Pellet Diameter, inches	0.3835
Density of UO_2 , % T.D.	94/94/93*
Fuel Length (Pellets Only), inches	120
Pitch, inches	0.556
Rod Array in Assembly	14 x 14
Rods per Assembly	180
Total Number Fuel Rods in Assemblies	28,260
Hydraulic Equivalent Diameter of Unit Cell, ft	0.0426
Additional Water Gap Between Assemblies, inches	0.019

* 94% for the center and intermediate regions and
93% for the outer region.

General

Total Core Area (Inside Core Baffle) ft^2	66.4
Equivalent Core Diameter, ft.	9.2
Maximum Diameter of Core, inches	119.4
Core Length, between Fuel Ends, ft	10
Length to Diameter Ratio of Core	1.09
Water to Uranium Ratio, Unit Cell	3.03
Fuel Weight, Kg U	57,400
Pressure Drop, psi	
Across Core	18.8
Across Vessel, including Nozzles	33
Core Power Density	
kw/liter of Core	71.6
kw/kg of U	23.4

APPENDIX B

NUCLEAR DESIGN PARAMETERS FOR THE SAN ONOFRE REFERENCE CORE

February 1965

(A) STRUCTURAL CHARACTERISTICS

1.	Core Material	Density (lb/in ³)	Volume (in ³)	Atomic Density**	Weight (lbs)
	Fuel (UO ₂)	<u>.370</u>	<u>387,500</u>	<u>.00757</u>	<u>143,600</u>
	Water	<u>.026</u>	<u>666,100</u>	<u>.01393</u>	<u>17,300</u>
	Stainless Steel (304)	<u>.285</u>	<u>77,100</u>	<u>.00679</u>	<u>22,000</u>
	Inconel	<u>.297</u>	<u>5,100</u>	<u>.000372</u>	<u>1,500</u>
	Void	—	<u>11,300</u>	—	—
	Total		<u>1,147,100</u>		<u>184,400</u>
1.1	Core Diameter				<u>110.9 in</u>
	Core Height				<u>120.0 in</u>
	Reflector Thickness and Composition				
	Top		H ₂ O	{	<u>~ 10 in</u>
	Bottom		+		<u>~ 10 in</u>
	Side		Steel		<u>~ 15 in</u>
2.	H ₂ O/U - cold				<u>3.59</u>
	at power*				<u>2.59</u>
3.	Number of Fuel Assemblies				<u>157</u>
4.	UO ₂ Rods per Assembly				<u>180</u>

* volume ratio if $\rho_{H_2O} = 1$ and $\rho_U = 18.7$

** (No/barns-cm)

(B) PERFORMANCE CHARACTERISTICS

1. Heat Output, Mw				<u>1346</u>
2. Loading Technique			<u>3-Region: Non-Uniform</u>	
3. Full Power Lifetime, hrs				<u>13,800</u>
4. Average First Cycle Burnup, MWD/MTU				<u>13,500</u>
5. Initial Weight of U per Charge, kg*				
	Region 1**			<u>19,450</u>
	Region 2			<u>19,080</u>
	Region 3			<u>18,800</u>
6. Discharge Weight of U in First Charge, kg*				<u>18,630</u>
(52 assemblies from Region 1)				
7. U-235 Initial Enrichment, w/o*	<u>3.2</u>	<u>3.4</u>	<u>3.8</u>	
8. U-235 Discharge Enrichment, w/o	<u>1.84</u>			
9. Pu produced per charge, kg	<u>126</u>			
% Pu-240	<u>15</u>			
% Pu-241	<u>9</u>			
10. Initial Conversion Ratio, Clean Fuel				<u>.50</u>
11. Fractional Energy Production				
	<u>Region I</u>	<u>Region II</u>	<u>Region III</u>	<u>Total</u>
U-235	.28	.25	.20	.73
U-238	.03	.03	.02	.08
Pu	.08	.08	.03	.19
Total	.39	.36	.25	1.00

* Listed for each region starting at center of core.

** Region 1 contained one more assembly than the other regions.

(C) CONTROL CHARACTERISTICS

1. Effective Multiplication (beginning-of-life)	
Hot, Power, Equilibrium Poison	<u>1.152</u>
Hot, Power, Unpoisoned	<u>1.185</u>
Hot, No Power, Unpoisoned	<u>1.205</u>
Cold, No Power, Unpoisoned	<u>1.254</u>
2. Number of Potential Control Rod Locations	<u>69</u>
3. Number of RCC Elements per Assembly	<u>16</u>
4. O.D. of Poison Element	<u>0.375</u>
5. Control Rod Material	<u>Ag-In-Cd</u>
6. Nominal Number of Control Rods	<u>45</u>
7. Estimated Control Rod Worth	<u>7%</u>
8. Boron Concentrations, ppm	
Refueling Shutdown; Rods in; $k = 0.90$	<u>3900</u>
Cold Normal Shutdown; Rods in; $k = 0.97$	<u>2900</u>
Hot Normal Shutdown; Rods in; $k = 0.97$	<u>2700</u>
Hot Full Power; No Rods; No Poison	<u>2800</u>
Hot Full Power; No Rods; Xe Poison	<u>2300</u>
Hot Full Power; No Rods; Xe and Sm	<u>2200</u>
End-of-Life; No Rods, Xe and Sm	<u>≈ 0</u>

(D) NUCLEAR CHARACTERISTICS AT POWER, CLEAN CORE

1. Groupwise Multiplication Factors for Various Enrichments

Enrichment w/o	3.2	3.4	3.8	4.2
k_1	0.091	0.092	0.092	0.093
k_2	0.014	0.015	0.017	0.019
k_3	0.166	0.175	0.193	0.210
k_4	0.927	0.929	0.936	0.941
k_∞	1.218	1.231	1.258	1.283
k_{eff}	1.198	1.211	1.238	1.263

2. Neutron Age (τ), cm^2	<u>52.2</u>
3. Thermal Diffusion Area (L^2), cm^2	<u>2.57</u>
4. Geometric Buckling (B^2), cm^{-2}	<u>.000362</u>
5. Average Flux Levels, Neutrons/ cm^2 -sec	

Thermal
Fast

1.8×10^{13}
 1.3×10^{14}

(E) NEUTRON BALANCE

Fraction

Fast Capture	<u>0.28</u>
Fast Leakage	<u>0.02</u>
Fast Fission	<u>0.12</u>
Thermal Capture	<u>0.21</u>
Thermal Fission	<u>0.37</u>
Thermal Leakage	<u>0.00</u>
Total	<u>1.00</u>

(F) KINETIC CHARACTERISTICS

1. Temperature Coefficients ($^{\circ}\text{F}^{-1}$)

Fuel	<u>$-1.0 \text{ to } -2.5 \times 10^{-5}$</u>
Coolant	<u>$+0.1 \text{ to } -3.5 \times 10^{-4}$</u>
Pressure	<u>$0.2 \text{ to } 3.5 \times 10^{-6}$</u>
Coolant void %k/ % void	<u>$+0.01 \text{ to } -0.3$</u>

2. Delayed Neutron Fraction, %

0.5 to 0.7

3. Prompt Neutron Lifetime

$1.2 \text{ to } 1.8 \times 10^{-5} \text{ sec}$