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REACTOR SAFETY ANALYSIS

By
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April 1966

Atomic Power Division
Westinghouse Electric Corporation
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REACTOR SAFETY ANALYSIS

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Plant Safeguards and Licensing

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1.0 INTRODUCTION

1.1 OBJECTIVES OF REACTOR SAFETY EVALUATION - The purpose of this report is to define the objectives and principles of a chemical shim pressurized water reactor, and to show by analytical simulation the adequacy of the various safety features of such a reactor.

Two types of safety issues are pertinent to the operation of a nuclear facility. The first concerns protection of the station facilities, including the reactor core, from damage and consequent economic loss. The second involves the prevention of hazards to the health and safety of station personnel and the general public. A systematic evaluation of safety issues in parallel with and in the context of the conceptual design of a facility is pertinent to both of these areas.

The human and economic interests are sometimes inseparable in the formulation of safety principles. Many published safety evaluations, in fact, make no distinction between these factors. Nevertheless, it is important to recognize the distinction where it does exist. If purely economic issues govern the design criteria for a given protective system, for example, the evaluation must consider the cost of protection as insurance against plant outage and fuel or component replacement. Where a potential radiation hazard of more than trivial magnitude is involved, a more conservative attitude generally prevails. Against such consequences redundant lines of protection are usually required, so that at least one contingent failure can be tolerated without propagating the accident. Operator failure to take corrective action is also assumed in such cases.

In this report, those aspects of system behavior which affect the potential for radiation exposure of persons or contamination of the environment are identified, and the protection philosophy is discussed.

1.2 SAFETY PRINCIPLES - The safety principles to which this nuclear power reactor must adhere are many and varied, but the most important can be summarized as follows:

- a. The technical standards of quality and performance incorporated in the reactor and plant components must be of a high order. In a general way these standards impart conservatism to the design, which is to say components which carry out some function related to nuclear safety would be expected to perform as well or better than required in any foreseeable condition of service.
- b. Assurance of safety is obtained also by requiring defense in depth against the possibility of component failure when needed to avert a possible hazard. Duplicate means are applied to fulfill safety requirements so that a single component can be assumed to fail to respond without sacrificing the protection which that component was

intended to provide. In some cases, where the component's function is to supply energy to an actuating circuit or device, the most probable failure of the component will itself cause the dependent circuit or device to trip in the direction of increased safety. The use of this feature, called "fail-safe" protection, is not required if component failure is tolerable on some other basis.

- c. The degree to which a protective system can be relied upon for safety is related to the ability of that system to be tested under conditions which simulate the requirements for actual service. Attention is therefore given to providing that ability. Components whose design and conditions of use dictate a testing frequency which is higher than the frequency of planned plant shutdown are designed to be tested without taking the reactor and generator out of service.
- d. All foreseeable reactor transients which could lead to departure from nucleate boiling (DNB) or fuel center melting actuate a reliable reactor trip before these constraints are reached. This action, combined with inherent strength and thermal properties of the fuel, ensure that such transients will not result in significant fuel damage over the design service life of the fuel. Significant damage is that which would result in fission product release to the coolant in excess of levels where one can operate within regulations governing personnel exposure and environmental contamination.
- e. To complement the design of the reactor protection system, the reactivity insertion rates possible with manual and automatic control action are limited mechanically. The limiting rates, as well as the extremes of reactor coolant conditions, flux level and distribution permissible under operating specifications, are analyzed to show the adequacy of reactor protection trip points.
- f. Should a failure or malfunction result in reactivity insertion at rates beyond the capability of the reactor protection system to limit fuel damage, the inherent reactivity limit of a single failure will preclude a transient severe enough to damage the reactor coolant system boundary. An example of such a reactivity limit is the worth of a single rod assuming it to be ejected from its position of maximum operational reactivity.
- g. The primary means of shutdown, control rod cluster insertion, is sufficient at all times to render the core subcritical with any single rod remaining withdrawn. A backup means of augmenting or supplanting this primary means is always available, i.e., injection of chemical neutron absorber in the coolant. Both of these shutdown systems are inherently reliable and would be capable of actuation under all foreseeable normal and abnormal service conditions.

- h. In the event of a loss of reactor coolant through a failure of that system boundary, emergency means of coolant injection are capable of limiting fuel melting and terminating the reaction of steam and fuel cladding so that the latter effects will not extensively augment the release of fission products from the fuel. Coolant system ruptures up to the severance of the largest reactor coolant loop pipe are considered under this requirement.
- i. Decay heat is accommodated following a loss-of-coolant accident or separation of the reactor from the main condenser heat sink without jeopardizing any remaining barrier to the release of fission products to the environment.
- j. The design of containment, waste storage, and any other vessels in which transmissible radioactive materials will or can be stored, will be such that maximum release in a single malfunction event is less than that which is acceptable at the site. The dose criteria of part 100 of AEC regulations are the upper limits for such releases in very improbable malfunctions. More restrictive limits are appropriate to malfunctions of higher probability.

1.3 SCOPE - Complete evaluation of reactor safety requires consideration of many environmental factors. These factors determine the atmospheric dispersion of airborne contaminants, the attenuation of gamma radiation emanating from a fixed source, and other physical limitations on the consequences of accidents, particularly as they would affect the public. As these factors are common to all types of reactors, this aspect of the evaluation of hazards is outside the scope of the Large Reactor Development Program. This study is concerned only with those aspects which are specific to the systems and components of the chemical shim PWR system typified by the San Onofre reactor. The scope of the hazards evaluation ends, therefore, with the definition of a radioactive source released or contained as a result of the malfunction and protective action considered. Where possible, the release is compared with that defined in the "Guide to the Calculation of Distance Factors for Test and Power Reactors," TID-14844, which is frequently used as a starting point for calculating potential public exposure due to a nuclear accident.

Section 2.0 of the report deals with malfunctions whose initial influence on the reactor is the addition of core reactivity. These "reactivity faults" include uncontrolled withdrawal of control rods or removal of chemical shim control, addition of cold water, adjustment of automatic control to an accidentally inserted (dropped) control rod, effect of a sudden large load increase, and the response to sudden ejection of a control rod.

Section 3.0 covers the class of mechanical accidents, such as rupture of the reactor coolant or steam system piping, loss of forced reactor coolant circulation, loss of power load, rupture of a steam generator tube, and various modes of failure resulting in unintentional discharge of radioactivity from sources external to the fuel.

Section 4.0 treats 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.

Section 5.0 is a discussion of the identity and magnitude of radiation sources considered as the potential release in the event of protection failures in various modes.

2.0 REACTIVITY FAULT PROTECTION

2.1 STARTUP

2.1.1 General

A startup incident or an uncontrolled rod withdrawal from a subcritical condition is extremely unlikely because of the protection provided. A continuous uncontrolled addition of positive reactivity during startup would result in an excessive rate of flux increase and a power excursion. Strict administrative controls will be followed at all times, and safety circuits, operated from nuclear instrumentation signals are provided to safely terminate any such transient without core damage. Analysis of the kinetic behavior of the reactor and of the effectiveness of the safety devices shows that reactivity faults during startup are highly improbable, and accidental core damage is effectively prevented.

The startup transient can proceed to a point requiring automatic reactor trip only if the control rods are continuously withdrawn or if the Chemical and Volume Control System malfunctions, and alarms are ignored. The range of reactivity insertion rates considered in the analysis of this transient includes the most severe conditions possible at startup.

While protection of the reactor core is the function of nuclear instrumentation in this case, pressure and level channels which respond to the power input to the reactor coolant would independently bring about a reactor trip in time to prevent an overpressure condition in the Reactor Coolant System, even if the protection from nuclear instrumentation were not considered. Multiple channels are employed in each means of protection, so that failure of any component will not prevent the desired signal from being given.

2.1.2 Facility Design Aspects

If a continuous control rod withdrawal were initiated, and the startup rate alarms and indications initiated from the source and intermediate range channels were not considered, the transient would be terminated by the following automatic safeguards:

a) Automatic Rod-stop Circuit -- actuated by high startup rate signals from the source and intermediate range nuclear instrumentation channels, will prevent further rod withdrawal. The startup rate is measured by two independent channels in the source range and two channels in the intermediate range. A high startup rate signal from any one of the appropriate channels is sufficient to cause a rod stop.

b) Automatic Reactor Trip -- actuated by a high startup rate signal from the intermediate range nuclear instrumentation provides further backup. This signal, which functions as a backup to the rod stop will also be given if either one of two independent channels senses a high startup rate.

c) Automatic Power Level Reactor Trip -- set for a fraction of full power is also provided. This trip will be effected if any two of four independent power range channels give a high power signal in coincidence.

Because of their reliable design, failure of all of these protective circuits is inconceivable.

The nuclear power response to a continuous reactivity insertion is characterized by a very fast rise terminated by the reactivity feedback effect of the negative fuel temperature coefficient. This self-limitation of the initial power burst results from a sizeable prompt negative fuel temperature coefficient (Dopplereffect) and is of prime importance during a startup accident since it limits the power to a tolerable level prior to external control action. After the initial power burst, the nuclear power is momentarily reduced and then if the accident is not terminated by a reactor trip, the nuclear power increases again, but at a much slower rate.

Termination of the startup transient by the high startup rate rod stop, high startup rate reactor trip, or the power range reactor trip, will prevent core damage. Other reactor protection channels, the variable low pressure trip, high pressure trip, and high pressurizer level trip (each actuated by coincidence of signals from any two of three independent channels), would terminate the startup transient independently before an overpressure condition could occur.

2.1.3 Assumptions and Method of Analysis

During a startup transient, the nuclear flux increases by many decades in a very short time interval. Analysis of this type of transient is by digital computation incorporating the neutron kinetics, including six delayed groups, and the core thermal and hydraulic equations. In addition to the nuclear flux response, the average fuel, clad and water temperatures, and also the hot spot heat flux response, are computed. Where applicable, the hot spot heat flux is used to calculate DNB ratios on the basis of the worst core power distribution expected during core life.

In the startup transient analysis, two distinct cases are studied, hot startup and cold startup. In the hot startup accident, the reactor is assumed to be initially at the normal operating pressure and the programmed temperatures for zero power. In the cold startup accident the temperature of the reactor is assumed to be 70 degrees Fahrenheit, and that pressure is assumed which results in the minimum terminal value of the DNB ratio. In either case the initial value of reactivity ($k = 0.99$) is selected from the range of expected values so that the greatest energy release will occur before any protective measures are taken.

In order to give conservative results for a startup transient, the following additional assumptions are made concerning the initial reactor conditions:

a) Doppler Coefficient -- since the magnitude of the nuclear power peak reached during the first part of the transient, for any given rate of reactivity insertion, is strongly dependent on the fuel temperature reactivity coefficient, the least negative design value is used for the startup accident ($\alpha_f = -1 \times 10^{-5} \delta k/^{\circ}F$).

b) Moderator Coefficient -- the contribution of the moderator reactivity coefficient is negligible during the initial part of the transient because the heat transfer time constant between the fuel and the moderator is much longer than the nuclear flux response time constant. However, after the initial nuclear flux peak, the succeeding rate of power increase is affected by the moderator reactivity coefficient. Accordingly, the most positive expected value is used for the moderator coefficient, since this yields the maximum rate of power increase ($\alpha_w = +1 \times 10^{-4} \delta k/^{\circ}F$).

c) Reactivity Insertion Rate -- for a given set of reactivity coefficients, the maximum energy release in the fuel corresponds to the maximum rate of reactivity insertion; thus, the reactivity insertion rate corresponding to withdrawal of the control rod group at the maximum velocity in the region of maximum reactivity worth is assumed in the analysis. This may be either a shutdown group or a control group, with the amount of reactivity inserted being limited by the maximum rod drive speed and the worth of the rod group. Reactivity insertion by rod withdrawal ($\delta k/sec = 2 \times 10^{-4}$) is inherently faster than that which can be produced by dilution of boron with the Chemical and Volume Control System. The significant aspects of reactivity addition by dilution of boron are covered in Paragraph 2.3, Boron Dilution.

d) Reactor Trip -- the behavior of the reactor is analyzed as though no protective trip were actuated until the overpower reactor trip set point is reached. The most adverse combination of instrument and set point errors, as well as delays for trip signal actuation and rod release are taken into account. Also, the rate of negative reactivity insertion corresponding to the trip action is computed based on the assumption that the highest worth rod is stuck in its fully withdrawn position.

2.1.4 Results

Cold Startup

The results of an uncontrolled control rod withdrawal with a constant insertion rate of $2 \times 10^{-4} \delta k$ per second from a cold coolant condition of 70 degrees Fahrenheit are shown in Figures 2-1 through 2-3. This reactivity insertion is the maximum rate for maximum control rod speed of 15 inches per minute and a total group worth of approximately 3.25 percent. This is a conservative assumption since no single programmed group of rod cluster control assemblies of this magnitude exists in the core. Also, no credit is taken in the transient for the fact that this maximum incremental worth would exist over a small portion of core travel. The maximum reactivity insertion rate achievable with dilution by the Chemical and

FIGURE 2-1
COLD STARTUP ACCIDENT, PERCENTAGE
OF NUCLEAR POWER

FIGURE 2-2
COLD STARTUP ACCIDENT, PERCENTAGE
OF THERMAL FLUX

FIGURE 2-3
 COLD STARTUP ACCIDENT, FUEL
 TEMPERATURE

Volume Control System is 1.0×10^{-5} δk per second (reference Paragraph 2.3 Boron Dilution), which is relatively insignificant rate compared to the 2.0×10^{-4} δk per second.

The reactor is assumed to be subcritical ($k_{eff} = 0.99$). The Doppler coefficient is assumed to be equal to the conservative value of -1.0×10^{-5} δk per degree Fahrenheit. (Reference paragraph 3.4) That section indicates a Doppler coefficient in excess of -2.0×10^{-5} δk per degree Fahrenheit at low effective temperatures. The reactivity feedback from Doppler provides the primary power cutback as shown in Figure 2-1 where there is a sharp break in the nuclear power transient at approximately 84 seconds. The effect of a reactor trip is also shown in this figure where the trip is conservatively assumed to occur at 118 percent power. The low power (25 percent) reactor trip will be in effect during subpower operation. It can be seen that the low power (25 percent) trip would occur at approximately 83.4 seconds while the 118 percent trip occurs at 83.8 seconds. The control rod insertion on reactor trip was assumed to follow the insertion versus time function described in Appendix A. The nuclear power decay is shown with two assumed values of shutdown reactivity and there is little significant difference except in the 0 to 10 percent power range. This shutdown is in addition to the reactivity inserted by the rods which are assumed to accidentally withdraw. The thermal flux response, of interest for DNB considerations, is shown in Figure 2-2. The beneficial effect of the inherent thermal lag in the fuel is evidenced by the peak flux equal to 42 percent of the full-load value reached during the transient. The heat transfer coefficient assumed from the fuel to coolant was a conservative high value, equivalent to assuming air in the gap between pellet and cladding rather than fission gases, thus yielding the most rapid heat transfer to the coolant. The fuel and cladding temperatures are shown in Figure 2-3. Also shown in this figure is the maximum coolant average temperature reached of 82 degrees Fahrenheit and thus there is no significant reactivity addition due to the positive moderator coefficient of reactivity. The positive moderator coefficient assumed for this case was $\alpha_w = +1.0 \times 10^{-4}$ δk per degree Fahrenheit, which is larger than the predicted value. The temperatures will not increase if the high startup rate trip occurs at 64 seconds.

There is a large margin to DNB during the transient since the rod surface heat flux remains low, and there is a high degree of subcooling at all times in the core.

Hot Startup

Figures 2-4 through 2-7 show the transient behavior for a startup accident from hot zero power conditions with average coolant temperature equal to 535 degrees Fahrenheit. Other plant conditions are the same as assumed for the cold startup case, i.e., $k_{eff} = 0.99$, $\alpha_f = -1 \times 10^{-5}$ δk per degree Fahrenheit and $\alpha_w = +1 \times 10^{-4}$ δk per degree Fahrenheit. The rate of reactivity insertion is also the same, 2×10^{-4} δk per second.

FIGURE 2-4
HOT STARTUP ACCIDENT, PERCENTAGE OF
NUCLEAR POWER

FIGURE 2-5
HOT STARTUP ACCIDENT, PERCENTAGE OF
THERMAL FLUX

FIGURE 2-6
HOT STARTUP TEMPERATURE TRANSIENTS

FIGURE 2-7
 STARTUP ACCIDENT, MAXIMUM ROD
 REACTIVITY RATE

The average thermal heat flux transient, assuming a reactor trip at 118 percent power, is shown in Figure 2-5. The peak average thermal heat flux is approximately the same as in cold startup case. The increase in average coolant temperature, as shown in Figure 2-6, assuming a reactor trip at 118 percent power, is nine degrees Fahrenheit, and is again insignificant with respect to positive moderator reactivity feedback. All the temperatures will remain at 535 degrees Fahrenheit if the high startup rate trip functions at 64 seconds as shown in Figure 2-7. The initial neutron source level is assumed to be 170 neutrons per cubic centimeter per second.

Since the peak average heat flux is only 42 percent of the steady state design value, the margin to DNB presents no problem during the accident. The W-3 DNB correlation is used to determine the minimum DNB ratio for the hot startup accident. The minimum DNB ratio occurs at approximately 86 seconds and is equal to 4.6, which is considerably higher than the minimum DNB ratio of 1.3 required. Were the accident to proceed without a reactor trip, it is conservatively calculated that a minimum DNB ratio of 1.3 would not be reached until approximately 95 seconds.

A parameter study was performed to investigate the sensitivity of peak nuclear power reached to initial multiplication factor, moderator coefficient of reactivity (α_w), Doppler coefficient of reactivity (α_f) and the heat transfer coefficient from the fuel to clad (UA)_f. The results are presented in Table 2-1. It can be seen that the Doppler coefficient provides the primary feedback mechanism to reduce the rapid nuclear power excursion. The effect of initial shutdown margin on the peak transients is insignificant. An initial multiplication factor of 0.99 is used for the accident.

2.1.5 Conclusions

Although very conservative assumptions were used in this analysis, the results indicate that the reactor core is safely protected from both the cold startup and the hot startup accidents. The peak thermal heat flux is well below the steady state design value and the minimum DNB ratio is considerably higher than the minimum DNB ratio of 1.3 required.

2.2 ROD WITHDRAWAL AT POWER

2.2.1 General

A continuous withdrawal of control rods at power without an increase in load in the turbine cycle is considered here. The response of the core is an increase in heat flux. Since the corresponding increase in heat extraction from the steam generator is a relatively small effect, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, this power mismatch and resultant coolant temperature rise would eventually result in DNB. To prevent the possibility of damage to the cladding, therefore, the Reactor Control and Protection System is designed to terminate any such transient with an adequate margin to DNB.

Table 2-1

Hot Startup Accident

Initial Multiplication Factor	α_w ($10^4 \delta k/F$)	α_f ($10^5 \delta k/F$)	(UA) _f (10^{-3} Btu/sec F)	Peak Nuclear Power Percent
1.0	+1	-1	1.65	186.5
1.0	+1	-1	0.7385	174.2
1.0	+1	-2	1.65	88.8
1.0	+0.5	-1	1.65	174.9
1.0	-1	-1	1.65	158.2
0.995	+1	-1	1.65	199.0
0.99	+1	-1	1.65	199.5

The third barrier to fission product release, the Reactor Coolant System is not subjected to excessive pressure in this transient, and protection is also provided by the Reactor Protection System. Reactor Protection System circuits independent of those provided for core protection would also terminate the transient before water level or pressure in the pressurizer could exceed safe limits. These include the high pressurizer pressure and level trip circuits.

A rod withdrawal transient could be initiated by a malfunction of the automatic Reactor Control and Protection System. In either case, both slow and fast reactivity additions, up to the maximum possible rate, must be considered because of the several ways (heat flux, temperature and pressure) in which the DNB condition can be approached.

2.2.2 Facility Design Aspects

The automatic features of the Reactor Control and Protection System which prevent core damage in a rod withdrawal transient at power include the following:

- a. Nuclear power range instrumentation actuates an automatic rod withdrawal stop signal if one out of four power range channels exceeds an overpower set point (105 percent).
- b. Nuclear power range instrumentation actuates a reactor trip if any two out of four power range channels exceed an overpower set point (109 percent).
- c. Reactor trip is actuated if any two out of three pressure channels indicate less than a variable low pressure set point. The low pressure trip set point is automatically adjusted for power and temperature to limit the amount of boiling in the core at the trip condition.
- d. A high pressure reactor trip, actuated from any two out of three pressure channels, is set at a fixed point (2,250 psia).

The manner in which the combination of overpower and variable low pressure trips provide protection over a full range of reactivity insertion rates is illustrated in Figure 2-8. This diagram represents the possible conditions of thermal power and coolant inlet temperatures in a two-dimensional plot. The boundaries of operation defined by the overpower trip and the variable low pressure trip are represented as "protection lines" on this diagram -- the overpower trip as a vertical line at the maximum value of nuclear power and the variable low pressure trip as family of sloping lines whose temperature intercepts increase with increasing pressure. The protection lines are drawn to include all adverse instrumentation and set point errors, so that under nominal conditions trip would occur well within the area bounded by these lines. A comparison between nominal and worst case trip conditions for this type of transient is presented in Paragraph 2.2.4. A maximum steady state operating condition for the reactor is shown as point O on Figure 2-8.

FIGURE 2-8
OVER-POWER, OVER-TEMPERATURE
PROTECTION DIAGRAM

A transient caused by reactivity insertion would result in movement of the operating point generally in the direction of increased power and inlet temperature, i.e., to the right and upward on Figure 2-8. A rapid reactivity addition can cause power increase which brings the reactor to the overpower limit before the inlet temperature can respond, i.e., along path O-A. A very slow reactivity addition can cause a net increase in inlet temperature to the variable low pressure trip point with only a smaller rise in thermal power, i.e., along path O-B. Other transients, corresponding to reactivity addition at intermediate rates, would be represented by paths lying between O-A and O-B.

The utility of the diagram just described is in the fact that the operating limit imposed by any given DNB ratio can be represented as a line on this coordinate system. In Figure 2-8, the line represents the locus of conditions for which DNBR equals 1.30 at 2,150 psia and 2,250 psia. All points below and to the left of this line have a DNB ratio greater than this value. The diagram shows that DNB is prevented for all cases if the area enclosed within the maximum protection lines is not traversed by the applicable DNB ratio line at any point.

The overpower trip representation on the diagram is invariant with pressure. The variable low pressure trip line at 2,150 psia lies below the 2,250 psia line, however, there being a smaller margin of subcooling at any operating temperature. Conversely, higher pressure raises the intercept of the variable low pressure trip line. At pressures greater than 2,250 psia, the limiting line falls below the low pressure trip line. Duplicate spray and relief valves are provided at the pressurizer to reliably limit transient overpressure to 2,250 psia for normal operating transients. For a complete loss of load without an immediate trip, or for a rod withdrawal transient from low power levels, the power mismatch can result in a rate of volume surge to the pressurizer greater than the capacity of the power operated relief valves to limit pressure to 2,250 psia. For these transients a fixed high pressure trip, detected by two out of three pressure channels, will terminate the transient and limit pressure to 2,250 psia. There is also a fixed lower limit for the variable low pressure reactor trip set at 1,600 psia.

From the arrangement of the protection lines in Figure 2-8, it is apparent that only those rod withdrawal transients which result in power-temperature conditions near the intersection of the protection lines would even approach the minimum DNBR limit. Considering the reliability of the protective circuits and the fact that reactor operation is permitted only when these circuits are functioning, the possibility of core damage or overpressure from rod withdrawal at power is extremely remote.

The overpower rod stop is provided to reduce the probability of an overpower trip caused by overshoot when moving control rods. This is achieved by stopping the rods at a flux level short of the trip set point.

The pressurizer high level trip, actuated by coincidence of two out of three high level signals, is a backup system provided to guard against water discharge from the pressurizer safety valves in the event of a sustained overpower condition. By preserving the required conditions for design safety valve capacity, the high level trip assures Reactor Coolant System protection independent of the means of core protection.

2.2.3 Assumptions and Method of Analysis

The control rod withdrawal accident analysis employs both analog and digital computer techniques. From the digital computer, the value of the variable low pressure trip set point is determined as a function of power, temperature and pressure such that trip is obtained over the range of interest before the DNB ratio becomes less than 1.30. This set point is then programmed, along with the overpower trip set point, with the full plant simulation on the analog computer.

Figures 2-9, 2-10, and 2-11 represent the results of a series of computations with varying input parameters. These curves are presented to show generally the effects of key parameters on the approach to DNB, and will clarify the basis for selection of pessimistic assumptions for the limiting case.

Figure 2-9 shows terminal DNB ratio as a function of reciprocal reactivity addition rate for various moderator coefficients in the range of importance. The limiting DNB ratio (i.e., the minimum approach to DNB) for each curve exists at a point of abrupt slope change corresponding to the intersection of the overpower and variable low pressure protection lines. The portion of the curves to the left of the abrupt slope change corresponds to termination by the overpower trip, while the portion to the right corresponds to termination by the variable low pressure trip. The minimum DNB ratio for the limiting condition is essentially independent of moderator coefficient, although the reactivity addition rate for minimum DNB ratio becomes slower as the coefficient becomes more positive.

Figure 2-10 illustrates the effect of pressure on terminal DNB ratio in rod withdrawal transients. In the curve marked without spray the pressure rises to the relief valve setting of 2,250 psia in the slower transients. With spray, pressure is suppressed to a value nearer the nominal value of 2,100 psia, and consequently, the variable low pressure trip occurs at a lower temperature. This lower temperature does not significantly affect the DNB ratio at trip since the trip occurs at the same exit quality in the core. Note that the limiting DNB ratio is unaffected, indicating that this condition is approached over a relatively small range of addition rates which are too rapid for spray to be effective. For limiting cases, the spray is assumed to be ineffective and the pressure is taken at the relief valve limit.

FIGURE 2-9
EFFECT OF MODERATOR COEFFICIENT ON
TERMINAL W3 DNB RATIO IN ROD
WITHDRAWAL TRANSIENTS

Initial Power = 103%

$$\alpha_f = -2 \times 10^{-5} \delta k / ^\circ F$$

$$\alpha_w = -2 \times 10^{-4} \delta k / ^\circ F$$

FIGURE 2-10
EFFECT OF PRESSURIZER SPRAY ON
TERMINAL W3 DNB RATIO IN ROD
WITHDRAWAL TRANSIENTS

FIGURE 2-11
 TERMINAL W3 DNB RATIO IN ROD
 WITHDRAWAL TRANSIENTS FROM REDUCED
 POWER LEVEL

Figure 2-11 shows, by comparison with Figure 2-9, the effect of initial power on terminal DNB ratio for the case of the most positive moderator coefficient ($\alpha_m = +0.5 \times 10^{-4} \delta k / ^\circ F$) and for the same fuel coefficient ($\alpha_f = -2.0 \times 10^{-5} \delta k / ^\circ F$). For the case where the transient starts from a partial power condition, it is seen that the limiting DNB ratio is essentially the same, that the variable low pressure trip is the governing protection over a wider range of addition rates, and that somewhat lower terminal DNB ratios are reached in those fast transients which are terminated by the overpower trip.

A typical result with a smaller fuel temperature coefficient, α_f , shown for comparison in Figure 2-11, indicates the limiting DNB ratio is not sensitive to changes in α_f over the range -1.0×10^{-5} to $-2.0 \times 10^{-5} \delta k$ per degree Fahrenheit. The essential effect of a less negative α_f is that the terminal DNB ratio resembles that of a slower reactivity addition rate.

For the case of an overpower trip, the time delay for breaker motion and rod release may allow an appreciable overshoot in nuclear power. This effect is counteracted by a much greater lag, however, between the nuclear flux and thermal flux levels due to thermal inertia of the fuel. For a slow reactivity accident which could result in a variable low pressure trip, the thermal flux may be following the nuclear flux more closely, but with the slower rate the delay in initiation of the trip is not significant. Thus, in either case, the results are not affected by the rate of negative reactivity insertion following trip.

In summary, to give conservative results, the initial operating point is assumed for analysis to be the maximum power level, the maximum inlet temperature, and the maximum primary pressure consistent with steady state operation. This will give the minimum initial margin to DNB and, since the thermal flux lags the nuclear flux, the minimum margin at the point of reactor trip for any reactivity addition rate.

2.2.4 Results

Three rod withdrawal transients are shown. Figure 2-12 shows the response of certain plant variables for the maximum reactivity insertion rate of $2 \times 10^{-4} \delta k$ per sec. Spray is assumed to be working but the transient is terminated so rapidly there is little spray effect. The maximum positive moderator coefficient of $+0.5 \times 10^{-4} \delta k$ per degree Fahrenheit was used and a minimum Doppler coefficient of $-1 \times 10^{-5} \delta k$ per degree Fahrenheit was assumed. The initial power level is 103 percent of full power ($1.03 \times 1,347$ Mwt.). The reactor trip occurs from high nuclear power (118 percent) in about five seconds and terminates the transient well before the coolant temperature and pressure have shown any increase. The rise in core inlet temperature and pressurizer pressure are a result of the turbine trip accompanying the reactor trip. Steam release to the atmosphere and condenser was not assumed. The DNB ratio barely changes during the transient. The initial margin to low pressure trip is conservatively assumed be the maximum value at that power level including instrument errors. This yields the greatest temperature and pressure changes before a low pressure trip will occur.

FIGURE 2-12
TRANSIENT RESPONSES FOR A FAST ROD
WITHDRAWAL ACCIDENT

Figure 2-13 shows the transient starting from the same initial conditions but with a much smaller reactivity insertion rate of 2×10^{-5} δk per second, a zero moderator coefficient of reactivity and a greater negative Doppler coefficient of 2×10^{-5} δk per degree Fahrenheit. For these conditions the transient is considerably lengthened and the reactor trip occurs from the variable low pressure trip signal. The effect of spray is seen in the reduction of the pressure rise at about 40 seconds. The surge rate then becomes too large for spray to handle alone and the pressure rises to the power operated relief valve setting and is then limited to 2,250 psia. There is a steady increase in core water inlet temperature until the low pressure trip occurs. For this case the minimum DNB ratio is slightly less than for the fast withdrawal and demonstrates the temperature pressure dependence of the correlation.

Figure 2-14 shows the transient starting from 80 percent power with $\alpha_w = +0.5 \times 10^{-5}$ δk per degree Fahrenheit, $\alpha_f = -1 \times 10^{-5}$ δk per degree Fahrenheit and for the low reactivity insertion rate of 2×10^{-5} δk per second. Spray was assumed to be ineffective for this case. Here, as in Figure 2-13, the transient is lengthened and the trip occurs from variable low pressure. The maximum pressurizer pressure reached is 2,250 psia where the relief valves are adequate to limit the pressure. There is a greater change in core inlet temperature than in Figure 2-13, but it started lower for the lower power level according to the programmed average temperature as a function of load. Here again, as for the slow transient, the minimum DNBR is similar to the full power case shown in Figure 2-13. Note, however, that the maximum instrumentation errors are assumed for the low pressure trip and the trip will occur in all probability much sooner. The transient also assumes no manual action although there would be alarms actuated for high average temperature, high ΔT across the steam generator, high pressurizer pressure and high pressurizer level before the trip occurs.

2.2.5 Conclusions

Transient conditions within the plant which could occur as a result of an uncontrolled rod withdrawal are constrained by a combination of the over-power, variable low pressure and fixed high pressure reactor trips to prevent DNB in the core. This protection has been shown to be adequate for all combinations of plant parameters which may exist at the time of the transient. Three different transients have been presented showing the time behavior of the plant during a rod withdrawal.

FIGURE 2-13
TRANSIENT RESPONSES FOR A SLOW ROD
WITHDRAWAL ACCIDENT

FIGURE 2-14
 TRANSIENT RESPONSES FOR A SLOW ROD
 WITHDRAWAL ACCIDENT FROM REDUCED
 POWER LEVEL

2.3 BORON DILUTION

2.3.1 General

Reactivity can be added to the reactor with the Chemical and Volume Control System by feeding a more dilute solution of boric acid into the Reactor Coolant System than is present in the reactor coolant. The possibility and consequences of causing an excessive reactivity insertion in this manner is considered in the design of the Chemical and Volume Control System. To verify the adequacy of the design, the limiting cases of reactor transients which could result from improper boron dilution are analyzed.

In general, the protection against uncontrolled reactivity addition by boron dilution is the same as that described for control rod withdrawal, the latter representing faster potential insertion rates. The maximum rate of boron dilution which can be attained with the Chemical and Volume Control System results in a maximum reactivity insertion rate of less than 10^{-5} δk per second. The Reactor Control and Protection System is designed to terminate reactivity insertion transients without core damage for all insertion rates up to 2×10^{-4} δk per second or 20 times the maximum boron dilution value.

2.3.2. Facility Design Aspects

A detailed description of the means of achieving and controlling the desired blend of boric acid and demineralized water in the reactor coolant makeup cycle is included in WCAP-3269-31 (1),* Section 4.0. As indicated in that description, the maximum rate of dilution is limited by the upper limit of the adjustable flow set point in the reactor makeup control.

Manual Operation

Three separate manual operator actions are required before the Chemical and Volume Control System can be used to dilute the reactor coolant boron concentration by any amount. First the operator must switch from the automatic makeup mode to the dilute mode. Second, the batching integrator must be set to some injection volume. Third, the start button must be depressed. Omitting any one step would prevent dilution. This makes the possibility of inadvertent dilution extremely small.

Information on the status of the reactor coolant makeup control is continuously available to the operator. Lights are provided on the control board to indicate the operating condition of valves and pumps in the Chemical and Volume Control System. Alarms are actuated to warn the operator if boric acid or demineralized water flow rates deviate from the preset values as a result of system malfunction.

During operation of the reactor, control rod position indication gives a sensitive and direct measure of any changes in boron concentration. Dilution of 50 ppm in boron concentration will require, on the average, a control group insertion of 20 inches to maintain a constant load and temperature.

* A list of references may be found at the end of each major section.

Improper Dilution

If the reactor is at power and under automatic control, the effect on shutdown reactivity margin of improper dilution would be control group insertion to maintain the desired average coolant temperature. The Reactor Control and Protection System automatically actuates a rod insertion alarm signal at a programmed limit where an adequate shutdown margin is assured. This LO alarm alerts the operator to borate using the normal boration procedure for the Chemical and Volume Control System. The additional boron will result in reduction of the coolant average temperature and the control group will withdraw to maintain the programmed temperature. If the dilution continues after this operation has been performed, the control group will continue to insert until a LO-LO alarm is actuated. At this point special procedures are initiated to inject borated water by either:

- 1) Aligning a boric acid transfer pump with the suction to the charging pump permitting a 45 gpm flow of 12 percent boric acid solution from the boric acid tank.
- 2) Aligning the refueling water tank discharge with the suction of the charging pump to supply borated water at 2,900 ppm to the Reactor Coolant System.

Both of these operations can be conveniently performed directly from the control room.

If the reactor is at power and under manual control, boron dilution will result in a slow rise in reactor power and coolant temperature. This slow transient would be terminated by the variable low pressure reactor trip, assuming the control board indications and alarms on coolant temperature were ignored.

Extended Boron Dilution

If boron dilution is continued for an extended period of time, a loss of shutdown margin would result. In view of the numerous alarms and indications of the impending condition, and considering that an adequate shutdown margin is to be maintained during normal power operation, there is considerable time to evaluate and correct the situation by several alternate means and thus maintain a safe shutdown margin. Details of the alarms and indications are described in WCAP-3269-45 (2).

During hot shutdown or cold shutdown conditions, greater shutdown is maintained, providing even more time to sense and correct any extended boron dilution effects. Under these shutdown conditions the boron concentration is periodically sampled. In addition, the source range channels will indicate any increase in neutron flux level. The rate at which boron dilution can occur with the Chemical and Volume Control System is very slow and any approach to criticality would occur at a much lower startup rate than in the case of the transient shown in Paragraph 2.1.

2.3.3 Assumptions and Method of Analysis

The lower control group insertion limit for each control group bank is monitored by its own shutdown margin monitor. The allowable insertion is established as a function of the reactivity requirements for power coefficient and moderator coefficient with the programmed average temperature, as explained in WCAP-3269-55, (3), Figure 55, in that report shows the lower insertion limit for Bank 2, for beginning of life. In this case Bank 1 is fully withdrawn. As the core proceeds through life and as boron is removed from the reactor coolant, the moderator coefficient becomes more negative. Additional reactivity is required in control rods to handle the programmed average temperature as a function of load. Bank 1 is then used in addition to Bank 2 to handle the hot to power reactivity variation. Figure 56 in the same reference shows the lower insertion limit for Banks 1 and 2 for end of life. The lower insertion limit is calculated as a linear function of measured average coolant temperature and of power, as determined by the average reactor coolant temperature differential. Thus any change in power or temperature, which could lead to reduction in shutdown margin, is indicated by an increase in the lower insertion limit requiring the addition of boron to withdraw the control group. The parameters, in the shutdown margin monitors, will be periodically adjusted throughout lifetime to reflect the change in moderator coefficient. Failure to make these changes would yield a premature indication of loss of shutdown margin.

2.3.4 Results

The normal dilution rate maintained with the Chemical and Volume Control System is 45 gpm, but provisions are made to permit dilution at 90 gpm. Figure 2-15 shows the effective reactivity addition rate for these dilution rates as a function of reactor coolant boron concentration. This figure includes the effect of increasing boron worth in reactivity at lower boron concentration. These rates are obtained for an average coolant temperature of 578°F.

At lower temperatures the boron worth is greater yielding a somewhat greater reactivity insertion rate as indicated by the point shown for a T_{avg} of 78°F and 2,500 ppm boron. Even for this condition the maximum reactivity insertion rate is less than 10^{-5} δk per second.

WCAP-3269-55 (3) summarizes the control rod requirements for shutdown. It is shown that at beginning of core life there is 2.9 percent shutdown with the highest worth rod stuck out including a 20 percent allowance for uncertainties. The shutdown at end of life is 2.2 percent. Therefore, at beginning of life, where the reactivity insertion resulting from boron dilution is greatest, it would require continuous dilution at the maximum rate of 90 gpm for almost 50 minutes at a conservatively high dilution reactivity rate of 10^{-5} δk per second to lose the 2.9 percent shutdown.

FIGURE 2-15
 REACTIVITY INSERTION RATE FOR BORON
 DILUTION AT $T_{\text{AVG}} = 578^\circ\text{F}$

With the reactor under manual control the reactivity insertion at 10^{-5} δk per second would result in a slow rise in coolant temperature and core power similar to the slow rod withdrawal transient until the variable low pressure trip is actuated. Figures 2-13 and 2-14, in conjunction with the control rod withdrawal at power, demonstrate the protection provided by the low pressure trip for this very slow insertion rate. The trip would occur within two to three minutes starting from full power operation.

As shown in WCAP-3269-55 (3) for the all rods in condition, there is an additional one percent shutdown. Therefore, for the hot shutdown or cold shutdown condition there is a possible additional dilution time of about 17 minutes, in addition to the previously mentioned 50 minutes, before criticality is reached.

2.3.5 Conclusions

The maximum boron dilution rate achievable with the Chemical and Volume Control System is limited by design to a rate very small with relation to the ability of the control rods to change reactivity. Excess shutdown reactivity is maintained at all times by the prescribed limits on control group insertion and two alarms are provided to alert the operator when these limits are reached. Numerous alarms and information are provided to alert the operator to the performance and status of the Chemical and Volume Control System. The excess shutdown, coupled with the inherent limit on reactivity insertion by dilution, provide the operator with adequate time to add boron as necessary by one of several alternate means to maintain shutdown margin.

The core is protected by the variable low pressure trip if boron dilution continues at power without compensating control group insertion.

2.4 COLD WATER

2.4.1 General

The addition of cold water to the core will result in an insertion of positive reactivity. The magnitude of the insertion is dependent upon the value of the negative moderator coefficient of reactivity. In a chemical shim plant the maximum value of the negative moderator coefficient is reduced due to the presence of boric acid dissolved in the coolant, thereby decreasing the potential reactivity insertion resulting from a cool-down. The most severe condition for a cold water transient in a chemical shim plant exists at the end of core life when the Reactor Coolant System is essentially boron free and the most negative moderator coefficient exists.

Three potential means of rapidly reducing the core water inlet temperature are considered in evaluating core protection against this type of reactivity insertion. They are:

- a. Startup of an inactive loop
- b. Addition of excess cold feedwater to the steam generators
- c. Sudden increase in steam discharge from the steam generators

Within the capability of the Reactor Control and Protection System, these transients can be accommodated without trip. For more severe cases, reactor protection in the form of the variable low pressure and overpower trips will terminate the transients without core damage.

2.5 STARTUP OF AN INACTIVE LOOP

2.5.1 Facility Design Aspects

Operation of the plant at power with a reactor coolant loop out of service (i. e., the reactor coolant pump shut down) results in a decrease in the coolant temperature in that loop. A subsequent restart of the idle reactor coolant pump without bringing that loop closer to system average temperature would then inject colder water, mixed with the flow from the active loops, into the core. The resultant reactivity insertion could exceed that which the Reactor Control System is able to follow. In the same manner as described in Paragraph 2.2 for rapid reactivity additions to control rod motion, the overpower trip will terminate the transient before the limiting margin to DNB is reached.

The plant has no valves to isolate a reactor coolant loop, therefore, operation of two reactor coolant pumps will result in backflow through the idle loop, in turn causing a reverse ΔT in that loop, i.e. the temperature of the normal hot leg falls below that of the reactor inlet. A temperature

interlock and alarm is provided for each loop to prevent starting a reactor coolant pump in an inactive loop if a reverse ΔT exists greater than a preset value. Restarting an idle loop can be accomplished by reducing load, which reduces the ΔT in the active loops to a value which will allow the idle loop pump to start.

Since there is no means of isolating a steam generator, the maximum idle loop temperature depression (with very small backflow) is limited by the temperature of the turbine cycle, thus providing an inherent limit on the severity of the transient. The range of possible inverse ΔT conditions is determined for a range of predicted backflow rates, and an analysis of the core response is performed for the most severe case assuming the ΔT interlock to fail.

2.5.2 Assumptions and Method of Analysis

The transient following startup of an inactive loop is calculated with a detailed plant simulation on an analog computer. A conservative calculation is made for low backflow and a conservative estimate of the overall heat transfer coefficient for the idle steam generator is made. These conditions yield the maximum idle loop temperature depression and the maximum initial power with two out of three loops. The reverse ΔT interlock and alarm will prevent a restart of the idle loop until the temperature difference is less than 5F. For this transient the interlock is assumed to be inoperative. The pump is assumed to accelerate to full flow instantaneously and the entire idle loop hot leg at the reduced temperature is swept into the core. A conservative maximum negative moderator coefficient is assumed. As a limiting condition the cold water is assumed to reach the core without mixing with the active loop flows, but is assumed to mix with the water in the inlet plenum. The core response is calculated assuming the temperature change to appear throughout the core. A low value of Doppler coefficient is assumed thus yielding the maximum power increase as a result of the cold water.

For the purposes of this transient calculation, the ratio of power level to coolant flow through the core is assumed to be a constant, since this yields the minimum steam pressure, and the greatest possible temperature decrease in the inactive loop. This assumption also results in the smallest initial margin to DNB. A high heat transfer coefficient from reactor cycle to turbine cycle to the inactive loop is used. This is approximately 85 percent of its normal full load value, whereas, the heat transfer coefficient will decrease a greater amount at the low load level at which the inactive loop is operating - approximately 18 percent of its normal full load value.

2.5.3 Results

A backflow rate of 20,800 gpm in an idle loop was determined for operation with two out of three reactor coolant pumps running. The flow rate through the active loops is 74,500 gpm each, an increase over the nominal flow with three loop operation. The resulting total core flow is then 128,200 gpm or

about 65 percent of normal three loop flow. The resulting cold leg temperature in all loops is 542°F and the hot leg temperature in the inactive loop is 505°F. The steam generator temperature in the inactive loop is 500°F so the inactive loop is approaching its minimum possible value using the high heat transfer coefficient. The hot leg temperature in the idle loop cannot go below the steam temperature. The assumed Doppler coefficient is -1×10^{-5} δk per °F and the moderator is -3.5×10^{-4} δk per °F. The transient results shown in Figure 2-16 are computed assuming the full 37°F temperature decrease is imposed at the inlet to the reactor vessel. Credit is taken for a mixing delay in the reactor inlet plenum but no credit for flow mixing with the two active loops.

The peak nuclear power occurs in about eight seconds and a peak thermal power of about 85 percent is reached. The reactor trip is from overpower at 100 percent. This is a conservative high value since the overpower trip at this power level would be set to give a maximum trip level of 80 percent of full power.

Since the peak thermal power is less than full power and the inlet temperature is less than the full power value there is margin to DNB. The minimum DNB ratio is 3.1 and occurs at about 10 seconds.

Figure 2-17 shows the effect of flow mixing with the active loops. The temperature change at the reactor vessel inlet plenum then becomes -12°F instead of -37°F. All other parameters are identical to those for the transient shown in Figure 2-16. The large reduction in transient peaks is evident from the two figures. The actual transient would lie somewhere between the two results and would most likely be represented by the results which assume flow mixing.

2.6 ADDITION OF EXCESS FEEDWATER

2.6.1 Facility Design Aspects

The addition of excess feedwater to a steam generator is also analyzed. The effect of the additional feedwater is similar to that of a load increase, causing a reduction in cold leg temperature of the affected loops. The temperature response to the cold feedwater is attenuated by the thermal capacity of the Reactor Coolant and Steam Systems.

Reactor control responds to the load increase within the transient capability of the plant. Since reactor control is based on Reactor Coolant System conditions and core power, it does not distinguish between a true load increase and a feedwater addition transient. If reactor control is unable to maintain plant conditions within the protection limits during the transient, the overpower or variable low pressure protection will cause a reactor trip without core damage.

FIGURE 2-16
TRANSIENT RESPONSE FOR STARTUP OF AN
INACTIVE LOOP WITH FULL TEMPERATURE
DECREASE

FIGURE 2- 17
TRANSIENT RESPONSE FOR STARTUP OF AN
INACTIVE LOOP WITHOUT FULL
TEMPERATURE DECREASE

2.6.2 Assumptions and Method of Analysis

The addition of excess feedwater is also analyzed by means of a detailed analog simulation. All three feedwater control valves are assumed to fully open together at full load. For purposes of analysis a complete loss of the air supply to the valves was assumed causing them to open. Since there are three instrument air compressors, providing a highly reliable air system, this occurrence is considered very improbable. However, this case was investigated to determine the effect on the core integrity. A large negative moderator coefficient is assumed which yields the maximum reactivity insertion. The transient is analyzed at full load where the additional available feedwater flow is limited by the feedwater system head-flow characteristics and at lower loads with only one feedwater pump running. For this case the transient is studied assuming the second pump is started and that all feedwater valves are in manual control.

2.6.3 Results

The most severe case of excess feedwater addition while at full power was analyzed. That is the sudden opening of all feedwater valves to their wide open position. The moderator coefficient of reactivity was assumed equal to -3.5×10^{-4} δk per $^{\circ}F$ and the Doppler to be -1×10^{-5} δk per $^{\circ}F$. No credit is taken for reactor control response. As seen in Figure 2-18 even for these very conservative assumptions in the first two minutes of the transient, prior to a trip of the unit, there is no significant reduction in core inlet temperature and there is very little response to this condition by the Reactor Coolant System at all. The first indications of the situation are high level alarms on the steam generators to alert the operator and it would be about two minutes before the upper level indicator is reached. It was assumed for this case that a manual trip is made at about two minutes.

Automatic steam release to condenser and atmosphere is neglected as a heat sink for this case. Following the turbine trip the steam generator pressure rises to the safety valve setting in about 10 seconds, thus limiting the peak core inlet temperature.

The transient behavior for startup of a feedwater pump with one running while at 50 percent power is shown in Figure 2-19. All three feedwater valves were assumed to be in manual control, a very remote possibility at this power level. The same Doppler and moderator coefficients were assumed as for Figure 2-18. The transients as shown in Figure 2-19 are very small.

The three steam generator high level alarms will annunciate within seconds from the time the pump was started. It is assumed that a manual trip is initiated 40 seconds from the time the feedwater pump was assumed to be started.

FIGURE 2-18
TRANSIENT RESPONSES FOR ALL THREE
FEEDWATER VALVES FULL OPEN

FIGURE 2-19
TRANSIENT RESPONSES FOR STARTUP OF
THE SECOND FEEDWATER PUMP

The Reactor Coolant System has essentially reached an equilibrium condition at this time with very little deviation from initial conditions as shown by the dashed lines in the figure showing coolant conditions without a trip. The core never approaches a DNB limit during this transient.

2.7 LARGE LOAD INCREASES

2.7.1 Facility Design Aspects

The Reactor Control and Protection System is designed to accommodate step load increases of the order of 10 percent without a reactor trip. A load tracking device is provided as a part of the turbine load limit control to maintain the load limit a preset amount above operating load but not to exceed rated full load. Any sudden load pickup from the turbine governor control is limited to this preset amount. At the end of core life this transient capability is increased because of the more negative moderator coefficient. Very large load increases, however, would cause an excessive reduction in reactor coolant temperatures, a large coolant contraction and resultant low pressure. The core is protected from DNB during this transient by the overpower-low pressure protection described in Paragraph 2.2. For the very large apparent increase in turbine cycle load caused by a steam line break this protection is provided during the initial transient. The behavior of the reactor during the steam line break is discussed in Section 3.4.

2.7.2 Assumptions and Method of Analysis

The turbine generator load limit with the automatic tracking feature will limit any load increase introduced either manually or automatically by turbine-governor control. Very large load increases, beyond the capability of the Reactor Control and Protection System, are protected by the combination of the variable low pressure trip and the nuclear overpower trip as described in Paragraph 2.2. Transient conditions following any load change will be constrained within the limits shown in Figure 2-8. If the reactor control is unable to maintain the plant conditions within these constraints following a load increase, a reactor trip will occur and no DNB can occur. The analysis presented here is determined from a detailed analog plant simulation and illustrates the operation of the reactor control and the automatic load tracking by the turbine load limit for a large step increase in load. A more severe pressure and temperature transient occurs in this case with a positive moderator coefficient, since a negative moderator coefficient serves to initially increase the reactor power level in response to a turbine load increase. With either a positive or negative coefficient the automatic control system must withdraw rods, to compensate for Doppler and moderator, to increase load and average temperature according to the temperature program. For the positive coefficient the withdrawal will be initiated earlier. A low incremental control group worth is also assumed.

2.7.3 Results

Figure 2-20 shows a transient response to a sudden requirement for 30 percent more load by the turbine governor control while operating at 70 percent load. The reactor control is assumed to be in automatic. A maximum positive moderator coefficient of $+0.5 \times 10^{-4} \delta k$ per $^{\circ}F$ and a low Doppler of $-1.2 \times 10^{-5} \delta k$ per $^{\circ}F$ are assumed which yield the most severe transient. The load tracking control is assumed to be set to allow a sudden 10 percent increase in load and it will then increase at a rate of 25 percent per minute to the desired load. The resulting transient is handled without a reactor trip and without excessive temperature or pressure excursions as shown in the figure. The increase of $14^{\circ}F$ in T_{avg} is to maintain the temperature program as a function of load.

Under manual control and without operator intervention, the coolant temperature and pressure, in the case of the positive moderator coefficient would continue to decrease until the variable low pressure trip is reached. The core is protected as shown in Figure 2-8.

With a negative moderator coefficient, a steady state condition may be achieved at a lower average temperature since the reduction in temperature will increase reactor power. This temperature reduction may also result in a large pressurizer pressure decrease and result in a low pressure trip.

2.7.4 CONCLUSIONS

Various means of introducing cold water into the core have been examined for end of life core conditions with the maximum negative moderator coefficient. There are no means of causing large and sustained temperature reductions at the core inlet with the exception of the assumed large steam line break described in Paragraph 3.4. The Reactor Control and Protection System will protect the core from DNB for other cold water transients associated with startup of an inactive loop or excess feedwater addition.

FIGURE 2-20
TRANSIENT RESPONSE TO 30 PERCENT
STEP LOAD INCREASE WITH LOAD LIMIT
TRACKING

2.8 DROPPED ROD

2.8.1 General

If one control rod should be released and drop into the core, the immediate effect would be a reduction of reactivity and a corresponding decrease in reactor power. Returning the reactor to normal power without withdrawing the dropped rod could result in local power densities in excess of design limits due to the distorted power distribution.

Protective Circuits

Two separate protective circuits are provided, each performing two separate functions, to preserve safe operating conditions following a rod drop. The first is to block automatic rod withdrawal, and the second is to initiate a turbine load cutback to a safe level.

Mode of Detection

These two functions are initiated by either of two independent signals: one signal from the nuclear instrumentation and the other signal from the rod position indicators. Either mode of detection is sensitive enough to respond to a rod drop at substantial power levels, and either mode of protection is adequate to prevent DNB. If only the rod withdrawal block function were actuated, the reactor power could reach equilibrium at a lower value. While the minimum steady-state DNB ratio would be reduced in this case, the variable low pressure reactor trip would still maintain core protection. If only the turbine load cutback were actuated the maximum reactor power would be limited to a safe value consistent with the worst dropped-rod nuclear flux distribution.

2.8.2. Facility Design Aspects

The probability of a rod cluster control assembly accidentally dropping is considered remote. The rod cluster control assemblies are positioned in the core by means of magnetic jack, hermetically sealed, rod drive mechanisms.

However, should a rod cluster control assembly drop into the core, the rod drop protection circuit and alarm are actuated from a rod bottom signal in the rod position indication system. The negative reactivity that would be inserted by a dropped rod will also cause a sudden decrease in neutron flux at one or more of the four power range nuclear instrumentation channels. The number of channels affected depends on the location of the dropped rod in the core. A flux difference circuit associated with each power range channel will initiate a rod drop alarm and actuate the rod drop protection circuit when this sudden decrease exceeds a preset amount. The flux difference set point will be 10 to 15 percent based on initial operating tests. The rod drop protection blocks automatic rod withdrawal and initiates a turbine load cutback to a preset load. This cutback is

accomplished by turbine load limit rundown which overrides the governor valve control. The power cutback occurs at the rate of approximately 50 per cent per minute.

To insure that the load cutback nearly matches the core power reduction resulting from the maximum negative reactivity insertion the final load for the power cutback is set according to the highest worth rod. This setting limits the severity of the reactor coolant temperature reduction, and hence pressure reduction, which could lead to a low pressure trip for a dropped rod while operating at or near full load. The power cutback reduces load to a present absolute value. Should a dropped rod occur from a power level below the cutback value. The operator must reduce load manually to compensate for the core power reduction to prevent a reactor trip. If the load is not reduced, the mismatch between core power and turbine load will result in a reactor coolant temperature and pressure reduction and a low pressure trip.

Should a dropped control rod insert less negative reactivity than the highest worth rod when the plant is at full load, the turbine load cutback will exceed the core power reduction at that time resulting in an increase in Reactor Coolant System temperatures and pressure. The Reactor Control and Protection System responds to insert the control group and restore equilibrium conditions at the reduced load. For a large difference between the core power reduction and the turbine load cutback, a short term steam release to the atmosphere and condenser may occur.

The highest worth rod is determined in the course of startup experiments and the power distribution effects verified. The protection is based on the fact that the maximum power tilt, resulting from a dropped rod, does not exceed the difference in load required for the highest worth rod. The power cutback required for the highest worth rod is therefore adequate to protect the core for the power tilt associated with any dropped rod. The exact magnitude of turbine load cutback is then set and is expected that the reduced set point will be between 75 and 85 percent of full load.

2.8.3 Assumptions and Method of Analysis

The transient following a dropped rod is determined using a detailed analog simulation of the plant. The dropped rod is assumed to cause a step decrease in reactivity and the core power generated is determined using a point-wise neutron kinetics model. The overall plant response is calculated by simulating the turbine load runback and blocking of automatic rod withdrawal. A low incremental worth (-2.5×10^{-4} δk per inch) is assumed for the control group as equilibrium conditions are restored. The analysis is made for the case where the power cutback is nearly equal to that required by the negative reactivity insertion from the rod (-2.5×10^{-3} δk) and for the case where the cutback is greater than required (-1.25×10^{-3} δk).

2.8.4 Results

Figure 2-21 illustrates the transient following a dropped rod worth -1.25×10^{-3} δk while operating at full power. For conservatism, the power cutback is assumed to reduce load to 75 percent at a rate of 25 percent per minute. This load cutback is greater than required for the dropped rod worth which gives a prompt reduction of about 15 percent in core power. Consequently, the coolant average temperature begins to increase. The programmed average temperature setpoint is reduced to the value corresponding to the reduced load. Consequently, there is a short term automatic steam release caused by the large temperature error signal and the equilibrium temperature is achieved under automatic control rod insertion in about four minutes. There is no rod withdrawal during the transient. The two curves compare typical beginning of life and end of life transients with a zero Moderator coefficient and a negative moderator coefficient of reactivity of -2.5×10^{-4} δk per $^{\circ}F$ respectively. The transient with the negative moderator coefficient exhibits a short term power increase as the average temperature is reduced. Note that there is an increase in the moderator margin to low pressure trip from the initial full power value.

Figure 2-22 shows the transient following a dropped rod worth -2.5×10^{-3} δk . In this case the prompt core power reduction is about 21 percent and the load cutback more nearly matches the core power. There is no steam release and equilibrium conditions are reached in four to six minutes.

2.8.5 Conclusions

Protection for a dropped control rod is provided by the combination of an automatic turbine power cutback and blocking of automatic rod withdrawal. the magnitude of the power cutback is to be determined experimentally in conjunction with initial startup tests. The protection system will in all cases protect the core and will when at or near full power prevent a reactor trip without requiring manual control.

FIGURE 2-21
 RESPONSE TO A DROPPED ROD WORTH OF
 $-1.25 \times 10^{-3} \delta k$ WITH POWER CUTBACK
 TO 75 PERCENT LOAD, INCREMENTAL
 CONTROL GROUP WORTH = -2.5×10^{-4}
 $\delta k / \text{INCH}$

FIGURE 2-22
 RESPONSE TO A DROPPED ROD WORTH OF
 $-2.5 \times 10^{-3} \text{ } \delta k$ WITH POWER CUTBACK
 TO 75 PERCENT LOAD, INCREMENTAL
 CONTROL GROUP WORTH = -2.5×10^{-4}
 $\delta k / \text{INCH}$

2.9 CONTROL ROD EJECTION ACCIDENT

2.9.1 General

The assumptions inherent for the postulation of a control rod ejection accident combine both the condition of a loss-of-coolant accident and the simultaneous insertion of reactivity associated with a given control rod. In order to assume a rapid ejection of the control rod from the core, a rupture of the control rod mechanisms must be postulated.

For this accident, it will be demonstrated that the effects are definitely localized, and that there is no resulting pressure surge that could further damage the Reactor Coolant System. Thus, it is assured that the ultimate consequences of the postulated accident would be less severe than an assumed double-ended rupture of the main coolant pipe.

In this postulated event, the control rod is ejected from the core in a very short time by full system pressure differential acting on the drive shaft. The resultant power excursion is limited by the Doppler reactivity effect of the increased fuel temperature, and terminated by reactor trip, actuated by high nuclear power signals. The amount of fuel damage that could result from such an accident is governed mainly by the peak power attained in the transient, which in turn depends on the worth of the ejected rod and the power distribution attained with the remaining control rod pattern.

In the following analysis, a series of assumed initial loading cases was studied. These included the rod ejection calculations reported in Section 4.2.3 of WCAP - 3269-55 (3). These calculations considered the power redistribution and rod worths of the highest worth rod removed from both the 29 and 17 rod control bank patterns. The nuclear analysis reviewed both the beginning-of-life and end-of-life configurations, because the control rod configuration changes with depletion of the core.

The results of these analyses indicated that there would be no fuel melting, and hence, no possibility of an excessive pressure transient in the Reactor Coolant System.

2.9.2 Facility Design Aspects

A failure of a control rod travel housing sufficient to allow a control rod to be rapidly ejected from the core is not considered credible for the following reasons:

- a) Each control rod drive mechanism is completely assembled and shop-tested at 6,300 psi.
- b) The mechanisms will be individually hydrotested to 3,750 psig as they are installed on the reactor vessel head to the head adapters, and again checked during the hydrotest of the completed Reactor Coolant System.

- c) Stress levels in the mechanism are not affected by system transients at power, or by thermal movement of the coolant loops. Because the mechanisms are restrained by cross-bracing at the top end, moments induced by the design earthquake can be accepted within the allowable primary working stress range specified by the ASME Code, Section III, for Class A components. The design further assures that failure would not occur with an earthquake of twice the design intensity.
- d) The pressure housing and the rod travel housing are each a single length of forged Type-304 stainless steel. This material provides excellent notch toughness at all temperatures that will be encountered. The joints between the mechanism and the head adapter, and between the pressure and rod travel housings, are threaded with a seal weld canopy.

Programming of all the rods in the shutdown and control groups will be under strict administrative supervision to ensure proper sequencing and withdrawal in going to power. All control rods have position indicating coils for indicating their position, thereby providing the means of surveillance over this operating limit. Separate and independent digital readout is also presented on the control board for the group or groups of rods used for control.

2.9.3 Assumptions

General

The nuclear analysis considered the control rod worth and power redistribution effects of various rods removed from both the 29 and 17 rod control group patterns. These analyses were performed at the beginning and end of core life. From the standpoint of this accident analysis, an ejected rod from the 45 rod pattern is trivial since the core would still be shut down by the remaining inserted rods.

Beginning of Life, Zero Power

The worth of the highest worth rod at beginning of life removed from the fully inserted 17 rod pattern was calculated as 0.4 percent $\Delta k/k$. The resultant power distribution for this case was only slightly higher than the full power design hot channel factors. Ejection of this rod would correspond essentially to the zero power condition, since this control bank is used alone in going to beginning of life power. It should be noted that this worth is substantially less than prompt critical. Thus the characteristic response would be a sudden jump of from two to two-and-one-half times the initial power, followed by a slow increase as delayed neutrons build up and as further reactivity changes are caused by fuel temperature and moderator density changes. Fuel temperatures approach the equilibrium values for the new power during the short time before reactor trip becomes effective. Thus the energy generated by the accident, for a given reactivity

addition, is approximately proportional to the initial power level. Practically no energy input results from an accident at zero power.

Beginning of Life Conditions, Full Power

For a short period at the beginning of the first core life, the moderator temperature and void coefficients will be slightly positive. During this time, the single seventeen rod control group provides enough control for a swing from zero-power to full-power. Ejection of the highest worth rod from the 29 rod control group pattern is again insignificant since there would be no return to criticality at beginning of life. Normally, the reactor would be at full power with the control banks in the upper portion of the core.

At power, the inserted worth of the entire control bank at beginning of life ranges from 0.2 to 0.8 percent $\Delta k/k$. The reactivity worth of a rod ejected from this pattern (approximately 0.04 to 0.06 percent $\Delta k/k$), with the control bank operating in the upper portion of the core, is substantially less than the 0.4 percent $\Delta k/k$ value calculated for the zero-power case. The rapid insertion of the small amount of reactivity associated with a single rod ejected from this full-power pattern would have little effect on the maximum fuel temperature.

End of Life Conditions, Zero Power

Later in life, as boron is removed from the coolant to compensate for burnup, the moderator temperature and void reactivity coefficients become increasingly negative. Because of this, greater control rod worth is required to shut down from full power than at beginning-of-life conditions, to make up for the moderator reactivity gained by cooling down from full-power temperature to zero-power temperature. In the extreme, at the end of life, the moderator temperature coefficient will be approximately $-2 \times 10^{-1} \Delta K/K-F$; the hot zero-power condition would have the seventeen-rod bank fully inserted; and the twelve-rod bank would be almost fully inserted at zero power. Ejection of the highest worth control rod from this hot zero-power condition ($10^{-6} \times 1,347 \text{ Mwt}$) would result in a heat flux hot channel factor of almost 10, and a calculated maximum ejected rod reactivity of 0.8 percent $\Delta k/k$.

As an extremely pessimistic approach, this accident at zero power was studied using a zero-moderator coefficient, and a low estimate for the end-of-life delayed neutron fraction, 0.0050. A fuel temperature reactivity weighting factor of 2.0; a fuel-to-clad gap heat transfer coefficient of 2,000 Btu/hr-ft²-F; a prompt neutron lifetime of 13 microseconds; rod ejection time of 0.1 second; time to start effective reactivity insertion following reactor trip of 1.0 second; and a trip rod reactivity of 1.8 percent were used. The weighting factor is the ratio of the fuel reactivity feedback to the reactivity, calculated from the change in core average fuel temperature.

End of Life, Full Power

At the end of life, both control banks must be utilized to go to full power. With the established withdrawal sequence, as the control banks are successively withdrawn, the worth of an ejected rod is reduced and also the ejected rod hot channel factors are improved. At full power, towards end of life, the rods will be operated substantially in the upper portion of the core with very little inserted worth. The maximum ejected rod reactivity insertion calculated for full power is 0.04 percent $\Delta k/k$. The total hot channel factor is not significantly greater than the design value. Thus the rapid insertion of this amount of reactivity would, as in previously described case, cause only a slight rise in fuel temperature.

2.9.4 Methods of Analysis

Reactor Kinetics

The nuclear power transients were calculated using the CHIC-KIN code, developed, by the Bettis Atomic Power Laboratory for similar analyses (J. A. Redfield, "CHIC-KIN -- A FORTRAN Program for Intermediate and Fast Transients in a Water Moderated Reactor", WAPD-TM-479, January, 1965). This code solves the point reactor kinetics equations, with feedback from an axially-segmented fuel element. Both local and bulk boiling voids are accounted for, providing a conservative extreme for the positive void reactivity case. For this study, the fuel rod was divided into eight radial increments, with a ninth increment for the clad. A fuel-to-clad gap heat transfer coefficient of 2,000 Btu/hr-ft²-F was used as a conservatively high value. The calculated fuel temperature reactivity was based on the core average transient fuel temperature rise, with a fuel temperature reactivity weighting factor of 2.0 for the 0.8 percent insertion case.

In order to obtain an extreme positive moderator reactivity feedback for the beginning-of-life case only, the entire core power increase was assumed to be concentrated in one-fourth of the core. This is consistent with iterative physics calculations that indicated this pattern of energy input of the coolant would give the maximum positive reactivity.

The physics calculations consisted of performing two-dimensional analyses of the core, with different amounts of core assigned lower water density than is normally the case. These calculations were performed for both the unrodded core, the core containing one control bank with its highest worth rod ejected entirely out of the core. Analyses were also carried out for two different values of water density lower than the normal full-power operating value.

All studies indicated that, for a core divided into a normal full-power outer region and a center region at some lower water density, the size of this center region, corresponding to the maximum reactivity insertion, is 25 percent of the total core. In order to obtain the value of the maximum reactivity that could be inserted in this fashion, the same kind of two-dimensional calculations were performed for different values of water density in the center quarter of the core.

The results are presented in Figure 2-23 for the unrodded core, and for the core containing one control bank less its highest worth rod where the multiplication factor is plotted as a function of water density in the center quarter of the core. Results are given for a range of densities from normal full power to 15 percent void content in this center region. It can be seen that the greatest moderator density reactivity insertion is 0.19 percent in the unrodded core, and that the value in the case of the ejected rod is only 0.07 percent (i.e., the maximum Keff minus the Keff at operating density. From the curve, for the unrodded core, the figures are $1.00100101 - .00014 = .00187 = 0.19$ percent. For the curve with one control bank in the core, the figures are $.99978 - .99909 = .00069 = 0.07$ percent.) In the rod ejection transient calculations, the moderator feedback curve for the unrodded core was used, as a conservative extreme.

The resultant calculated energy storage in the fuel center was then normalized to the calculated worst hot-channel factor to obtain the maximum core fuel temperature. (Because heat continues to be added for about a second after the initial power burst, the maximum fuel temperature reached for all cases studied is in the fuel center).

Fission Product Release

Assumptions governing the determination of fission product released to the coolant, as used in this analysis, are as follows:

- a) Cladding failure in any fuel rod releases the equilibrium fission products in the pellet-cladding gap. Cladding failure is assumed to accompany DNB.
- b) A "burst" of fission products, conservatively estimated to be equal to that produced in 180 hours of diffusion from the fuel at normal operating temperature, occurs due to transient-induced reorientation of the surface layer of uranium oxide.
- c) Diffusion in the pellets whose average temperature exceeds $2,900^{\circ}\text{F}$, but where center melting is not attained, occurs at an accelerated rate, 10^7 times normal. This rate, which is actually characteristic (of the diffusion at temperature very near the melting point (i.e. $5,000^{\circ}\text{F}$), is assumed to persist for the duration of the fuel temperature transient.

FIGURE 2-23
EFFECTIVE MULTIPLICATION FACTOR
VERSUS MODERATOR DENSITY FOR DENSITY
CHANGES IN 25 PERCENT OF THE CORE -
2500 PPM BORON

- d) In computing the fission product inventory of the fraction of the core subject to each type of release, the steady state power distribution is compounded to give the maximum possible fission product release.

System Pressure Surge

Calculations were also performed to determine whether the pressure surge, resulting from the rapid increase in heat transfer through the fuel cladding, might cause a coolant volume increase sufficient to exceed the allowable system pressure. A digital computer code was used that accounted for steam void formation in the core and mixing in the hot leg; pressure drop in the pressurizer surge line; and load rejection due to turbine trip caused by the overpower reactor trip. To achieve even more conservative calculations, no condensation of steam by the water stored in the reactor vessel outlet plenum was assumed. For this calculation, the release of coolant through the head penetration for the ejected control rod was neglected.

The pressure increase is a function of the heat transferred to the water in the coolant; the initial coolant temperature; the initial heat transferred to the steam generator; and the initial pressurizer level. For a given increase in reactor heat flux, the highest pressure would result if the plant were initially at full power, because at that time all these parameters would be at a maximum. At full power, however, the possible ejected rod worth, and thus heat flux increase, is much smaller than at low power.

2.9.5 Results

Power Increases and Fuel Temperatures

At beginning of life, with the calculated worth of rods from zero-power and full-power configurations, there was little increase in core power for the zero-power case and only a slight increase in fuel temperature for the full-power case.

At full power, toward the end of core life, the control rods will be operated substantially in the upper portion of the core with very little inserted worth. The maximum ejected rod worth calculated for full power was 0.04 percent. The core power distribution was only slightly higher than the design value. The rapid insertion of this amount of reactivity would cause only a slight rise of approximately 100°F in fuel temperature.

Figure 2-24 demonstrates the nuclear power transient resulting from an ejected rod at the end of life, with the control rod configuration at zero power, but with zero moderator, void, and temperature reactivity coefficients. A reactivity insertion of 0.8 percent $\Delta k/k$ and a resulting total hot channel factor of 10.0 were assumed. Even under these severe conditions, the resulting maximum hot-spot fuel center temperature only reached 4,060°F, well below the fuel melting temperature. The temperature of the outer part

FIGURE 2-24
CONTROL ROD EJECTION END OF CORE LIFE

of the pellet did not get as hot, and was cooling as the center reached its peak. Even in this isolated case, there was no danger of molten fuel being expelled into the coolant. The fraction of rods in DNB was calculated to be 11 percent, while the fraction of fuel pellets with $T_{avg} \geq 2,900$ F was only 1.0 percent.

Fission Product Release

The fission product release was conservatively calculated to be equal to or less than one percent of that equivalent to a complete core meltdown.

Pressure Surge

The core fuel-to-water heat flux calculated for the zero power conditions is shown in Figure 2-25. The maximum calculated pressure resulting from this transient, only 2,316 psia, would occur 9.7 seconds after the accident.

2.9.6 Conclusions

The rod ejection analysis was reviewed for the control rod configurations specified for zero and full power conditions at the beginning and end of life. The results of this analysis indicated that there was no melting of the fuel, even for the conservative assumptions postulated. For the fullpower cases, where the center temperature was initially high, the resulting temperature increase was small. The operation of the plant precludes the insertion at power of any rod of substantial worth. The zero power case, at end of life, resulted in the highest increase in the center temperature of the fuel; however, this was still substantially below the melting temperature.

Thus there is no chance of any rapid expulsion of finely dispersed molten fuel with the coolant and further propagation of the accident. Experimental results show that gross fuel melting, preferentially located in the vicinity of the clad surface, must occur before fragmentation of the fuel pellet clad system could occur. Since this analysis indicated that there was a substantial margin to any gross fuel pellet melting, it was concluded that there could not be a substantial pressure surge in the Reactor Coolant System. The maximum pressure due to the increased heat flux was determined in all cases to be equal to or less than 2,316 psia, well below the design pressure of 2,500 psia.

The maximum size break for this postulated accident is on the order of four inches. The total break and resultant rate of energy released to the containment sphere was postulated to be much less severe than those cases studied for the hypothetical accident described in Section 3.0.

The amount of fission products released to the containment also resulted in an amount well below that determined for the hypothetical accident described in Section 3.0 even with the conservative assumption that there was no retention of the fission products in the coolant during the relatively slow blowdown of the coolant.

CONTROL ROD EJECTION
CORE HEAT FLUX CURVE

ZERO POWER, $k = 0.8\%$

FIGURE 2-25
CONTROL ROD EJECTION CORE HEAT FLUX
CURVE

2.10 REFERENCES

1. WCAP-3269-31, "Development of Auxiliary Systems," by H. J. Cordle, February 1965.
2. WCAP-3269-45, "Reactor Control and Protection Systems, Functional Requirements and Performance," by W. King, March 1965.
3. WCAP-3269-55, "Core Physics Characteristics of the First Loading of the San Onofre Nuclear Generating Station," by D. L. Miller, November 1965.

3.0 PROTECTION AGAINST MECHANICAL MALFUNCTION

3.1 LOSS OF COOLANT

3.1.1 General - Loss of coolant is a condition of accidental discharge of fluids from the reactor coolant system, which is a breach in the third of the barriers to fission product transport. Reactor protection involves tripping the reactor and supplementing the supply of coolant to prevent or limit damage to the fuel cladding. Public protection is ensured by the containment sphere.

The following types of coolant loss are treated in this evaluation:

- a. Rupture of any line or nozzle connecting to the reactor coolant system.
- b. Improper opening of a valve, or of valves in series, allowing uncontrolled discharge from the high-pressure reactor coolant system to a low-pressure system.
- c. Material failure in a pipe or component of the reactor coolant system, resulting in an opening with an effective size not greater than the largest in categories a and b above.

A special case of loss of coolant, treated in Paragraph 3.5, is the rupture of a steam generator tube. For this failure, one must consider the possibility of transport through the steam system of some of the activity present in the reactor coolant. The analysis and description of consequence-limiting features must take the steam system operation into account and is therefore considered separately. Similarly, a severance of a control rod drive pressure housing with resultant ejection of the rod cluster from the core, while constituting a loss-of-coolant fault, is treated separately in Paragraph 2.9 because of the special conditions introduced by the reactivity insertion.

3.1.2 Facility Design Aspects

Preventive Features - The importance of the loss-of-coolant class of mechanical failures merits consideration of the many features designed to prevent the occurrence of a loss of coolant. These features provide assurance of the integrity of the reactor coolant barrier by:

- Conformance to high standards of materials and construction
- Control of mechanical and pressure loads
- Control of the environment of material exposure
- Surveillance

The manner in which each of these areas affects the integrity of the reactor coolant system is amplified in the following paragraphs.

a) Material and Construction Standards - A summary of the applicable codes and standards, including special nuclear case rulings, under which the components of the reactor coolant system and the reactor vessel itself are designed is contained in Table 3-1, Component Code Requirements. Within the rules of the above codes, configurations and thicknesses of pressure-bearing members are established by specified calculations to provide acceptable margins of safety against overstress during design loading conditions.

The applicable design code specifies acceptable materials, fabrication procedures, and test requirements which give assurance that the properties inherent in the application of the code rules are obtained.

In addition, provisions of the ASA B31.1 piping code are applied to loop piping in the reactor coolant system to limit the allowable stress range for cycle loadings. Otherwise, the basis for design against fatigue-type failures is that recommended in the "Tentative Design Basis for Reactor Pressure Vessels and Directly Associated Components" (PB 151987), or a modification of this basis utilizing the latest appropriate data from Section III, ASME Boiler and Pressure Vessel Code, "Nuclear Vessels." This basis incorporates a low-cycle fatigue design method which assures that allowable usage of the fatigue life of the material is not exceeded, even in the unlikely event that the full number of cycles assumed for design are actually imposed on the plant.

A piping flexibility analysis was performed for the reactor coolant piping to show that the maximum stress range induced in the pipe and fittings by the constraint of thermal distortions are within the limits of the piping code. A similar analysis was performed with seismic and dead loads imposed on the piping. Each of these analyses confirmed that the location of hangers and component supports, and the dimensional allowances for unrestrained deflection are adequate.

In the structural analysis of other reactor coolant system components, piping end forces, dead loads and seismic loads are applied to show that the design of supports and stops is adequate to code allowable stress limits. The criteria applied are highly conservative in that seismic forces are considered as primary loads, i.e., continuous rather than transient as is customary in seismic design of structures. Seismic forces in components essential to the nuclear safety of the plant are determined by a spectrum response method which takes into account the natural period and damping characteristics of the structures.

Table 3-1

Component Code Requirements

REACTOR COOLANT PUMP

ASA B 16.5 - injection and
leakage control line
flanges
ASME Boiler and Pressure
Vessel Code Section VIII -
guide for design of pressure
containing parts
Casing will comply with ASME
Boiler and Pressure Vessel
Code Section VIII, par.
P-299(a) and P-1(a)

PRESSURIZER

ASME Boiler and Pressure
Vessel Code Section VIII,
par. U-1-e. Also Special
Rulings 1270N, 1273N -
vessel
ASME Boiler and Pressure
Vessel Code Section VIII,
Nuclear Code Case 1273N-4,
par. 6(a) (2) and 6(a) (3) -
instrumentation welded to
inside wall of vessel
Tentative Structural Design
Basis for Reactor Pressure
Vessel and Directly Assoc-
iated Components, ID 151987 -
nozzle stresses fatigue
conditions
ASME Boiler and Pressure Ves-
sel Code Section VIII, par.
UG-31b - heater sheathes
Unfired Pressure Vessel Safety,
Orders, State of California-
vessel

Table 3-1 (Continued)

REACTOR COOLANT PIPE

ASME Boiler and Pressure
Vessel Code Section I
except code stamping is
not permitted (cyclic
loading per ASA B31.1)

REACTOR VESSEL

ASME Boiler and Pressure
Vessel Code Section VIII,
Cases 1270N, 1273N (Refer
to Reactor Vessel Design
Report, July 1965)

STEAM GENERATORS

ASME Boiler and Pressure
Vessel Code Section VIII,
par. U-1-3. Also Special
Rulings 1270N, 1273N and
1274N

Unfired Pressure Vessel
Safety Orders, State of
California

Tentative Structural Design
Basis for Reactor Pressure
Vessel and Directly Assoc-
iated Components, PB 151987 -
nozzle stresses and fatigue
conditions

b) Control of Pressure, Thermal and Mechanical Loads - Pressure loads are limited in accordance with the applicable design codes by the provision of self-actuated safety valves of specified characteristics. These valves are sized to control the pressure surge to the code limit of 110 percent of the maximum allowable working pressure of 2,485 psig in the improbable condition where 100 percent load rejection occurs without reactor trip or feedwater addition. All operational transients and load rejections are less restrictive than this case to the design of the pressure relieving devices to the reactor coolant system.

The largest pressure surges incurred under design load transients are controlled below 2,185 psig at the pressurizer by operation of either of two power relief valves. Smaller transients are controlled within this limit by pressurizer spray.

The possibility of reactor faults giving rise to pressure surges in excess of power relief valve setting is discussed in Sections 2 and 3. In general, all credible reactor faults are terminated by the reactor protection system before an overpressure condition arises, even if one failure is assumed to occur in the protection system itself.

Thermally induced local stresses are limited by the provision of thermal sleeves at all inlet connections where introduction of cold fluid may occur a significant number of times during the life of the plant. The connections so protected are the surge line and spray connections at the pressurizer, and the reactor coolant loop-inlet connections of the charging line and residual heat removal loop. Temperature changes in the reactor coolant system which occur as a consequence of load changes, reactor trips, and planned heatup and cooldown are less severe and the expected cycles far less numerous than the values assumed for the purpose of fatigue analysis. Control of these transients is inherent in the reactor control system response characteristics, except for the heatup and cooldown cycle which is manually controlled. Heatup and cooldown rates will be adjusted to ensure that thermal stress in the reactor vessel shell conforms to limits imposed by the temperature and radiation history of the vessel.

Mechanical loads on the reactor coolant system piping and components are controlled by proper design of supports, hangers, and limit stops to accommodate thermal expansion and to resist seismic and dead loads. The restraining forces exerted by these structures are considered in the mechanical design analysis of the respective components.

c) Control of the Environment - Environmental control is the principal means of protection against corrosion-type failures. The pressure bearing materials exposed to the reactor coolant environment, austenitic stainless steel and Inconel, are highly resistant to general surface corrosion in water because they both form tightly adhering oxide films which inhibit mass transfer to and from the base metal. The rate of attack by other mechanisms, i.e. galvanic effects and stress corrosion cracking, is

sensitive to the presence of chlorides and oxygen in solution. Control of reactor coolant chemistry, limits the concentration of chlorides and dissolved oxygen. These precautions are taken to assure a margin of about ten to one below the minimum concentrations of these agents which have been known to exist in cases of stress corrosion attack.

The corrosion rates which have been measured under the conditions of reactor coolant system operation, including average concentrations of boric acid, are such that an integrated material loss of the order of two mils for stainless steel and four mils for Inconel would be expected in the life of the plant. Corrosion at this rate will have no effect on the integrity of the reactor coolant system.

d) Surveillance - Surveillance of the reactor coolant system to assure continued integrity will be carried out through the following programs:

1. Continuous monitoring of reactor and coolant system operating variables; corrective action can be taken according to planned procedures even before operating limits are reached.
2. Periodic sampling of reactor coolant assures continuity of the chemical environment which precludes corrosive deterioration of the system.
3. Periodic removal and testing of reactor vessel material specimens exposed to the environment and incident flux levels seen by the vessel itself at the core midplane, to verify the adequacy of provisions for the anticipated shift in the nil ductility transition temperature.
4. Removal and inspection at refueling intervals of the reactor vessel head studs; visual and dye penetrant examinations to check the continued integrity of these items.

Protection of Reactor Core

The occurrence of a loss of coolant, unless of a very small magnitude, will bring into operation one or more means of supplementing the supply of reactor coolant to prevent or limit possible damage to the fuel. The means of supplying this protection will depend on the magnitude of the rate of coolant loss. In the order of increasing loss rate, these actions are:

- a) For small breaks, pressurizer level control adjusts the net charging-letdown rates to maintain a minimum level in the pressurizer. Depletion of coolant in the volume control tank is compensated by automatic makeup control at reactor coolant boron concentration.

b) For larger breaks, or in case of malfunctions of automatic makeup, an emergency low-level signal from the volume control tank-level instrumentation enables the charging pumps to take suction from the refueling water storage tank instead of the volume control tank. Valving to accomplish this transfer is duplicated for reliability.

c) For still larger breaks, i.e., those for which the discharge rate exceeds the delivery of the charging pumps for a prolonged period of time, the level in the pressurizer will continue to decrease, eventually exceeding the low-level trip point for safety injection. A change in the pressurizer steam volume, will result in an accompanying decrease in system pressure. The coincidence of low-pressurizer pressure with low pressurizer level, requiring signals from two out of three channels, will actuate the safety injection system. This actuation is fully automatic and consists of:

1. Tripping the reactor.
2. Isolation of the turbine cycle feedwater-condensate system from the feedwater pumps by valve action.
3. Starting the two safety injection pumps, both feedwater pumps (if not already running), and the standby charging pump.
4. Stopping the four condensate pumps and two heater drain pumps.
5. Aligning charging pump suction with the refueling water storage tank (if not already accomplished).
6. Opening the safety injection isolation valves at the reactor coolant loops.
7. Opening the safety injection isolation valves at the feedwater pump discharge (only after the condensate isolation valves at the feedwater pump suction are verified closed).

Actuation relays and related circuitry to accomplish these steps are fully duplicated for reliability. Completion of safety injection actuation results in pressurization of the injection header and an open path for delivery of borated water from the refueling water storage tank through two independent safety injection trains and three independent injection lines, one to each reactor coolant loop. In addition, both charging pumps are running, and delivering borated refueling water to the charging line.

The flow rate of refueling water delivered to the reactor coolant system by the combination of pumps just described will vary with the pressure head against which they are delivering. The delivery characteristics of the system, shown in Figure 3-1, are based on pump manufacturers' performance curves, with allowances for the calculated flow resistances of the various

CONDITIONS

	CASE NO.				
	<u>I</u>	<u>II</u>	<u>III</u>	<u>IV</u>	<u>V</u>
No. of S. I. Trains Functioning	2	1	2	1	2
No. of S. I. Lines Injecting	3	2	2	2	1
No. of S. I. Lines Spilling to Containment	0	0	1	1*	1
No. of S. I. Lines Blocked	0	1	0	0	1

* Flow limited to 500 gpm plus the higher of the two injecting lines.

FIGURE 3-1
SAFETY INJECTION DELIVERY
CHARACTERISTICS

delivery paths. Five cases are shown, representing full system operability and four partial operability modes representing single failure conditions. These cases are identified in the following manner:

- Case I - Both safety injection trains operating; delivery through three injection lines.
- Case II - One safety injection train operating; delivery through two injection lines, with third line isolated.
- Case III - Two safety injection trains operating; delivery through two injection lines, with third line spilling to the containment.
- Case IV - One safety injection train operating; delivery through two injection lines, with third line spilling.
- Case V - Two safety injection trains operating; one injection line operating; one injection line isolated; one injection line spilling to the containment.

Note that Case I represents full system capability. Case II allows for the failure of one train to operate, whether by malfunction of a pump, valve or electrical circuit; it assumes one injection line isolated, denoting either (1) failure to open one injection line isolation valve or (2) operation of the flow comparator to limit the loss through a broken safety injection line. Case III assumes failure of the flow comparator, allowing full flow through a broken injection line to be discharged to the containment. Normal functioning of the flow comparator would isolate the broken injection line unless the flow in that line were within 500 gpm of the other two lines. Case IV assumes that the flow through the broken line is within the range where the flow comparator is non-sensitive (less than 500 gpm); hence, no isolation signal would be expected. The failure assumed in this case is the loss of one pumping train. Case V represents another condition for a loss of coolant involving isolation valve failing in the closed position blocking delivery through one of the undamaged lines. The flow comparator is prevented from closing the valve in the spilling line in this case, but pumping through both trains is assumed. Credit for charging flow has not been assumed in the analysis for all cases.

In the general case, injection maintains sufficient water in the reactor to dissipate the residual heat of the core, either by boiling (with steam discharging through the break) by heat transfer to the steam generator tubes, or by the net enthalpy increase of injection water spilling from the break. Reactor Coolant System pressure decreases as the temperature is gradually reduced. Operation of the recirculation loop of the Safety Injection System is started by manual action to return borated water from the Containment Sphere sump to the Reactor Coolant System by way of the recirculation heat exchanger.

Loss-of-coolant analysis is performed to show conformance to the following criteria:

- a. Under no condition would there be fission products released from the fuel rods as a result of a Reactor Coolant System rupture of credible size (i.e., not exceeding the effective size of the ten-inch severed connecting line).
- b. Protection against larger size ruptures shall be effective in substantially limiting the fission product release.
- c. The reliability of the protection described in a and b above shall be such that the criteria are met even in the event of a failure of any single component of the protection system to respond actively in the prescribed manner.

3.1.3 Assumptions and Method of Analysis - The analysis of the loss-of-coolant blowdown transient is performed with the digital computer. The mathematical model used is summarized in Paragraph 3.1.6.

Simplifying Assumptions and Initial Conditions

- a. The Reactor has been operating at full-rated power for an infinite time prior to the break.
- b. The break occurs instantaneously, but nuclear heat generation and heat removal by the steam generators continue at full-rated power until the requisite conditions of low pressurizer pressure and low pressurizer level are reached, and until an additional instrument response and reactor trip time of 1.5 seconds has elapsed. After reactor trip, heat generation follows the calculated residual power relation, which will be described in Paragraph 3.1.6.
- c. Heat removal by the steam generator is assumed until the liquid-vapor interface falls below the level of the reactor nozzles. An expression for the difference between residual heat and steam generator heat-removal rate is included in the energy balance equations.
- d. Sensible heat released by cooling the core and one-third of the vessel and thermal shield, assuming the temperature of these masses to fall at the same rate as the coolant temperature, is transferred to the coolant during blowdown.

- e. Expulsion of the coolant through the break is governed by the mechanisms of subcooled flow until the calculated system pressure falls to saturation at the mass-average temperature.
- f. After saturation is reached, two-phase flow through the break occurs with no slip. System pressure remains at saturation level, with flow limited by sonic choking as long as the pressure ratio across the break exceeds the critical value for the mixture.
- g. When the steam-water interface falls below the level of the break, pure steam flow is assumed if the calculated entrainment is less than one percent. The point in the transient at which this steam flow occurs is obtained by calculation.
- h. Following initiation of the safety injection signal and a time delay for repositioning valves, injection flow is added to the system at a rate determined by the applicable delivery curve for the system pressure at that instant.

In those cases where partial or complete uncovering of the fuel rods occurs, the temperature transient of the core is calculated using the method described in Paragraph 4.6. This analysis consists of a heat balance on the fuel rod, with heat input from both residual heat and metal-water reaction at the cladding surface, internal heat transfer among three pellet regions and a cladding region, and no heat loss from the cladding surface while the core is exposed to a steam atmosphere.

The pellet and cladding temperatures in each of ten core regions, representing various fractions of the peak steady-state power density, are evaluated by this method as functions of time after break. The maximum temperature of the cladding in each core region is compared with a calculated failure temperature to determine the extent of clad rupture, if any, resulting from the accident.

The aggregate steady state thermal power of those core regions determined to undergo clad bursting is used to calculate the release of volatile fission products from the pellet-cladding gap, as detailed in Paragraph 5.2.

Similarly, the maximum temperature attained by the fuel pellet in each of the core regions is determined. The thermal power of each core region, and its maximum fuel pellet temperature is used to calculate the fission product evolution from the UO_2 , following the method described in Paragraph 5.2.

3.1.4 Results of Blowdown Analysis - Figures 3-2 through 3-8 show the calculated pressure and volume histories for a complete range of break sizes and for the various safety injection performance cases described earlier.

Break sizes corresponding to six nominal connecting pipe diameters (single-ended rupture of 1-, 2-, 4-, 6-, 8-, and 10-inch lines at the point of connection) and the hypothetical double-ended rupture of the main reactor coolant loop were considered. Certain combinations of break size and assumed safety injection performance are not compatible and were excluded from the analysis for reasons listed below.

Case I	- Hypothetical
27-1/2 inch
break size	Severance of a main loop could cause the associated injection line to discharge directly to the containment through the break.
Case II	- Hypothetical
27-1/2 inch
break size	Flow comparator would not be sensitive to flow differential; third injection line would not be isolated but would instead lose its flow via the loop break (covered by Case IV).
Case III,
IV &
V	- 1-, 2-, and
4-inch break
sizes	Complete loss of the injection flow of one 6-inch injection line cannot be assumed if the break size is less than 6 inches

It will be noted from the curves that in no case involving a 10-inch break or smaller does the calculated liquid volume fall below that required to cover the bottom 18 percent of the core. In the worst case (10-inch Case V), the mid-plane of the core becomes uncovered for 21 seconds, beginning 83 seconds after the break. Assuming the hot-spot power density to exist at the mid-plane, and a surface heat transfer coefficient of 20 Btu/ft²-F-hr, the maximum clad temperature reached in this example is calculated to be less than 1,050°F. The calculated short-term bursting temperature of Zircaloy cladding at this condition is greater than 1,100°F, and that of stainless steel cladding is greater than 1,300°F. It is concluded that there would be no clad bursting, hence no fission product release, from either a Zircaloy or a stainless clad core in the accident.

Note that the case which results in maximum core exposure of all of the credible cases, the 10-inch Case V, is extremely improbable. It assumes the occurrence of a 10-inch equivalent break circumscribing a 6-inch safety injection line juncture and compounds this event with a failure of the flow comparator. The nominal, or "design" case of a 10-inch rupture, which is severance of the 10-inch pressurizer surge line, is more nearly represented by 10-inch Case IV, in which the core mid-plane is not exposed.

FIGURE 3-2
LOSS OF COOLANT - ONE-INCH (NOMINAL)
PIPE BREAK

FIGURE 3-3
LOSS OF COOLANT - TWO-INCH (NOMINAL)
PIPE BREAK

FIGURE 3-4
LOSS OF COOLANT - FOUR-INCH
(NOMINAL) PIPE BREAK

FIGURE 3-5
LOSS OF COOLANT - SIX-INCH (NOMINAL)
PIPE BREAK

FIGURE 3-6
LOSS OF COOLANT - EIGHT-INCH
(NOMINAL) PIPE BREAK

FIGURE 3-7
LOSS OF COOLANT - 10-INCH (NOMINAL)
PIPE BREAK

Time After Rupture - Sec.

FIGURE 3-8
LOSS OF COOLANT - 27.5-INCH
HYPOTHETICAL MAIN LOOP BREAK

In the hypothetical main loop severance cases, complete uncovering of the core occurs in 7.3 seconds, and recovery of the bottom, in the worst case (IV), 64 seconds after the break. The mid-plane is recovered 20 seconds later, and the top of the core at 104 seconds after the break. This case represents the minimum degree of core cooling afforded by the safety injection system under the single failure criteria for the hypothetical loss-of-coolant accident. The degree of core damage is assessed in the following paragraph. Note that in the more probable case (III) where both injection trains function and two lines deliver injection flow to the reactor while the third delivers ineffectively to the severed loop, the bottom of the core is recovered in 46 seconds.

3.1.5 Results of Core Damage Assessment - Hypothetical Loss-of-Coolant Case -
In the hypothetical Case IV, in which the maximum core exposure occurs, the maximum pellet and cladding temperature were calculated for both stainless steel and Zircaloy clad cores. The fission product release was predicted using the method and data described in Paragraph 5.2.

The fission product activities released to the containment sphere, obtained by the method outlined in Paragraph 5.2 are listed in Table 3-2. The total fission product release for the stainless steel clad core and the Zircaloy clad core corresponds to approximately 6 percent and 25 percent, respectively, of that equivalent to a complete core meltdown.

3.1.6 Coolant Blowdown Calculation

General Description

The discharge of reactor coolant from a system rupture is analyzed by means of the digital computer, which solves a series of mass, energy, and volume balance equations for small time intervals defined by small changes in pressure. The results of the computation are expressed as system pressure and liquid volume remaining as functions of time following the break.

Mathematical Model

For the purpose of constructing a suitable mathematical model, the blowdown process is separated into four distinct phases:

- a) Pressurizer Blowdown - Beginning immediately after the occurrence of the rupture and continuing until the pressurizer is emptied of liquid.
- b) Subcooled Blowdown - Consisting of the further expansion of the reactor coolant to saturation pressure conditions.

Table 3-2 Fission Product Activities Released to Containment Sphere

DIRECT SOURCES		
Energy Group, Mev	Initial Source Strength, Mev/sec	
	Stainless Steel Clad Core	Zircaloy Clad Core
0.4	3.2×10^{16}	1.4×10^{17}
0.8	2.9×10^{17}	1.2×10^{18}
1.3	8.4×10^{16}	3.5×10^{17}
1.7	4.9×10^{17}	2.1×10^{18}
2.5	3.3×10^{17}	1.4×10^{18}
INHALATION SOURCES		
Isotope	Initial Activity, Curies	
	Stainless Steel Clad Core	Zircaloy Clad Core
I-131	1.1×10^6	4.5×10^6
I-132	1.6×10^6	6.5×10^6
I-133	2.3×10^6	9.5×10^6
I-134	2.9×10^6	1.2×10^7
I-135	2.1×10^6	8.8×10^6

c) Two-Phase Blowdown - Starting when the reactor coolant reaches saturation condition and two-phase flashing flow begins. This phase continues at least until the steam-water interface falls below the level of the break.

d) Steam Blowdown - Beginning when the interface falls below the break, if water entrainment is low. Steam flow continues until the pressure in the Reactor Coolant System reaches equilibrium with the pressure in the vapor container.

A more detailed description of each phase, and the assumptions and methods used in the calculations are presented below.

Phase 1 - Pressurizer Blowdown

The first phase begins when a rupture occurs, exposing the interior of the reactor coolant system to the low-pressure environment of the containment sphere. At this instant the water in the system is in a sub-cooled state. The first discharge through the break will thus consist of compressed water, impelled by the expansion of the steam volume in the pressurizer. This phase continues until the pressurizer is empty of liquid. It is assumed that the coolant comes to its weighted average temperature and thus, the only significant pressure gradient in the system is at the break. Other friction effects, including the losses in the pressurizer surge line, are neglected.

Phase 2 - Sub-Cooled Blowdown

At the end of the first phase the pressure in the entire system is higher than the saturation pressure of the remaining reactor coolant. The second phase consists of the expansion of the coolant to saturation pressure. The only difference between this phase and the previous one is a shift in emphasis from the pressurizer to the reactor coolant system proper. Actually, the two phases may occur with some overlapping, but in order to achieve more flexibility in the calculation, they are separated. The division makes it possible to instruct the code to ignore the pressurizer blowdown phase for the calculation of conditions following a break of the pressurizer surge line.

At some time during the first or second phases a reactor trip is initiated by a low-pressure signal. The trip results in insertion of control rods. Until the trip is completed, heat addition to the system continues at 100 percent of rated core power, and is assumed to be removed at the same rate by the steam generators. After the trip, the nuclear heat input for each interval of the calculation is computed by a special subroutine using the residual heat relations:

$$r(t) = 0.051 e^{-0.004t} + 0.177 e^{-0.1t} \text{ for } t < 60 \text{ sec} \quad (1)$$

$$r(t) = 0.28 + 0.023 e^{-0.009t} \text{ for } t > 60 \text{ sec} \quad (2)$$

All of the sensible heat in the core and one-third of the vessel and thermal shield is assumed to be transferred to the coolant with a constant temperature difference between the core and coolant.

The rate of discharge from the system during Phases 1 and 2 is calculated for each interval by the use of the following equation based on the investigation of Burnell and others: (J. G. Burnell - "Flow of Boiling Water Through Nozzles, Orifices and Pipes," Journal, The Institution of Engineers, Australia, March 1946).

$$G = \sqrt{2gp [P - (1 - C) P_{sat}]} \quad (3)$$

The surface tension factor C is determined from an empirical expansion given by Burnell. Fluid properties are obtained from standard steam table sub-routines included in the code.

Phase 3 - Two Phase Blowdown

In the third phase the nature of the discharge changes radically, as flashing flow occurs at the break and a two-phase mixture flows through the hole. For most of this phase the pressure difference across the break will exceed the critical pressure ratio and choking flow will exist. It is interesting to note at the beginning of this phase, if the discharge through the break is small, the pressure in the system may actually rise because the volume of steam generated by the residual heat production is greater than the volume leaving the system through the break.

The discharge rates for two-phase flow are calculated using an equation derived from the general energy equation by Benjamin and Miller. (M. W. Benjamin and J. C. Miller - "The Flow of a Flashing Mixture of Water and Steam through Pipes," Transactions ASME, October 1942).

$$G^2 = \frac{-\int_{P_1}^{P_2} \frac{dP}{v}}{\ln \left(\frac{v_2}{v_1} \right) + \frac{fL}{2D}} \quad (4)$$

where the subscripts 1 and 2 refer to the beginning and end of the flow path, respectively.

This equation is based on the assumption of a homogeneous flow of the two phases (no slip). At low pressures, where slip does occur, the assumption is questionable validity. However, the results are correspondingly conservative and represent an upper limit for flashing flow. Note that this equation includes a term $\frac{fL}{2D}$ for pressure losses in the piping and components. This

information is supplied as part of the input data. To be certain that the blowdown calculation is conservative, the total resistance used consists entirely of the discharge coefficient at the break.

Generally speaking, the pressure at the end of the path, P_2 , is the critical pressure and is unknown. An iterative subroutine is therefore used in the code to determine the maximum (critical) flow rate for each interval for which the calculation is performed.

Throughout the first three phases, credit is taken for heat removal by the steam generators. The reactor decay heat in the energy balance equations is replaced by the difference between the reactor decay heat and steam generator heat removal rate. Steam generator heat removal continues until either the reactor or turbine cycle side of the steam generator is emptied of water.

Phase 4 - Steam Blowdown

For small breaks the mechanism of two-phase flow will cease when the steam-water interface falls below the break. After this time the flow consists of saturated steam carrying some amount of entrained water.

The calculation obtains the liquid volume point at which the steam blowdown begins. The criterion for steam discharge is that no more than one percent liquid be entrained in the steam flow from the liquid space in the reactor vessel.

For pure steam flow the following equation is developed for an adiabatic expansion from the boiling interface to the break:

$$G = .94 \sqrt{2g \rho^2 J (h_1 - h_2)} \quad (5)$$

Since the flow passes through what may be considered an orifice, a flow coefficient of 0.94 has been included in the theoretical flow equation. This is a conservative value and will yield higher flow rates than would actually be expected.

Nomenclature

- C = surface tension factor
- D = inside diameter of pipe at rupture point, ft
- f = friction factor
- g = gravitational constant, ft/sec²
- G = mass velocity, lb/ft² sec
- h = specific enthalpy, Btu/lb

J = mechanical equivalent of heat, ft-lb/Btu
 L = length of flow path, ft
 P = system pressure, psf
 P_{sat} = saturation pressure, psf
 r = ratio of residual heat generation to full power
 t = time, sec
 v = specific volume, ft³/lb
 p = specific density, lbs/ft³

3.2 LOSS OF COOLANT FLOW

3.2.1 General - The response to a reduction in coolant flow while the reactor is at substantial power is a corresponding increase in the reactor coolant temperature. If the increase in coolant temperature is large enough, DNB will occur. Depending on the heat flux and flow conditions at the point of DNB, cladding temperature rise may result in plastic deformation or rupture unless corrective action is taken. To avoid such damage, a reactor trip is actuated under those conditions of reduced flow and initial heat flux which could result in excessive cladding temperature. In this context, excessive temperature is that which might result in plastic deformation of the cladding.

Public safety is not affected by a loss of flow, even if clad damage should occur, since any activity released from the fuel is retained in the Reactor Coolant System. Independent reactor trips are provided that would terminate the coolant temperature rise without exceeding system design pressure. Instrumentation provided to detect the loss of flow condition and initiate reactor trip is discussed in WCAF-3269-45.⁽¹⁾

A loss of coolant flow would be initiated by loss of voltage on the pump supply buses, a circuit breaker trip, an electrical or mechanical failure in one of the pumps or pump motors, or by an operator manually tripping a pump when reactor power is above the prescribed limit for such action.

3.2.2 Facility Design Aspects - In designing the reactor coolant pumps, the flywheel inertia was selected on the basis of providing sufficient coastdown flow to protect the fuel cladding in the case of coincident loss of power to all three pumps with immediate reactor trip from full load. After pump coastdown is completed, natural circulation of coolant through the core and reactor coolant loops builds up due to the thermal syphoning effect of the reactor and steam generator. The resulting flow rate is sufficient to remove the residual heat of the core without damage to the fuel or cladding.

If the loss of flow is caused by a loss of bus voltage, the reactor is tripped by the low bus voltage pump circuit breaker trip or the low flow trip circuits. If the loss of flow results from a pump breaker being opened by the operator, the breaker trip or the low flow trip circuits would again trip the reactor. A pump failure without a breaker trip would actuate the low flow trip circuit in the affected loop.

The low bus voltage trip circuit consists of two detection channels, one on each reactor coolant pump bus, which opens a pump breaker on detection of loss of bus voltage. The pump circuit breaker trip circuit and low flow trip circuit each consist of one detection channel per reactor coolant loop. Above the loss of flow permissive power level, a signal from either of these circuits indicating a loss of flow in any one loop will actuate a reactor trip; below this power level, coincident signals from either circuit in any two loops are required to actuate a trip.

Either the circuit breaker trip or the low flow trip will protect against core damage in any loss-of-flow transient. If none of these circuits were assumed to function, coolant temperature would increase until the variable low pressure trip, the high pressure trip, or high pressurizer level trip points were reached. In such case, the Reactor Coolant System would not be overpressured. Each of these circuits consists of three channels; signals from any two out of three in coincidence will actuate a reactor trip.

Finally, manual reactor trip by the operator is an additional means of limiting the consequences of loss of flow, independent of the automatic trip schemes.

3.2.3 Method of Analysis and Selection of Parameters - The loss of coolant flow accident is analyzed by a combination of computational techniques. A full plant simulation is used to compute the core average and hot spot heat flux transient responses including flow coastdown, temperature, reactivity and control rod reactivity insertion effects. These flux responses are used as the input to a detailed thermal-hydraulics computation to compute the margin to DNB using the continuity, momentum and energy equations of fluid flow together with the W-3 DNB correlation described in WCAP-5584.(2)

For a trip initiated from a turbine protective device or a reactor protective device, the generator is maintained connected to the electrical system for one minute. The reactor coolant pumps remain connected to the generator after the one minute delay. Continued forced circulation is maintained on turbine-generator inertia for three to four minutes.

In order to assure the conservatism of the results, the following assumptions are made concerning the reactor conditions at the start of the transient:

a) Operating Point - The operating point is assumed to be most adverse with respect to the margin-to-DNB. The conditions are the maximum steady state power level, minimum steady state primary pressure, and maximum steady state core water inlet temperature are as follows:

$$\begin{aligned}\text{Power} &= (1.03) (1,347 \text{ MWt}) \\ \text{Pressure} &= 2,100-30 = 2,070 \text{ psia} \\ \text{Inlet} &= 553 + 4 = 557^{\circ}\text{F}\end{aligned}$$

b) Moderator and Doppler Coefficient - The smallest negative expected value of Doppler reactivity coefficient and the most positive expected value of moderator reactivity coefficient are assumed since these result in the minimum hot spot heat flux decrease during the initial part of the transient, the time when the minimum DNB ratio is reached. The low flow trip is assumed to be initiated at 90 percent of full flow. The reactor trip is completed in less than three seconds, however, and these temperature feedback effects are not very significant.

$$\begin{aligned}\alpha_f &= -1 \times 10^{-5} \delta k/^{\circ}\text{F} \\ \alpha_w &= +0.5 \times 10^{-4} \delta k/^{\circ}\text{F}\end{aligned}$$

c) Reactor Trip - Since the low flow trip results in a longer delay than the bus undervoltage or breaker trips, it is the low flow trip which is assumed to be actuated. After the trip signal is given, a conservative instrumentation delay is assumed before the rods are inserted. The assumptions for reactor trip are summarized in Appendix A. Credit is taken for the reactivity added during the rod run-in time. Minimum available reactivity worth in withdrawn rods is assumed, and the most reactive rod is assumed to be stuck in its fully withdrawn position. The trip worth is four percent δk for this case.

d) A conservative residual heat rate equal to eight percent of full power is assumed. The reactor inlet temperature is assumed constant. This is a reasonable assumption, since the minimum DNB ratio in the core is reached in about three seconds, which is shorter than the transport time between the steam generator outlet and the core inlet.

3.2.4 Results - The reactor coolant flow coastdown curve, shown in Figure 3-9, was determined using a conservative high estimate of loop pressure losses. The extended coastdown time illustrates the effect of flywheel inertia added to the reactor coolant pump. The reactor trip is completed in about three seconds where there is still between 70 and 80 percent of full flow. There is no approach to DNB, even for the complete and simultaneous loss of power to all the reactor coolant pumps.

FIGURE 3-9
REACTOR COOLANT FLOW FOLLOWING
COMPLETE LOSS OF PUMP POWER

Figure 3-10 shows the neutron power response, while Figure 3-11 shows the average channel and hot channel heat flux response. The effect of the time delay associated with the fuel to water heat transfer is evident when compared to the neutron power response.

Figure 3-12 shows the DNB ratio as a function of time at the point of minimum DNBR, 65 percent up the channel from the core inlet. It is seen that the minimum DNB ratio is 1.62, which is well above the 1.30 limit.

The loss of flow in one or two loops from full power operation is less severe because there is more total core flow during the same core heat flux conditions following reactor trip.

3.2.5 Conclusions - The core is protected from DNB for the transient resulting from a complete loss of power to all three reactor coolant pumps while operating at the maximum initial steady state operating conditions. This protection is provided by a direct reactor trip on loss of flow, combined with the extended flow coastdown provided by high inertia reactor coolant pumps.

3.3 FUEL HANDLING

3.3.1 General - In the design of reactor components, fuel handling facilities, and fuel handling methods, numerous precautions have been taken to preclude accidental nuclear excursions, release of fission products, and overexposure of personnel during refueling. Safeguards have been adopted with particular attention to the following potential sources of hazard:

- Improper withdrawal of rod cluster control assemblies
- Failure to maintain subcritical geometry
- Failure to maintain adequate shielding for personnel
- Failure to maintain adequate cooling of fuel assemblies
- Mechanical damage to cladding

All refueling operations will be conducted in accordance with prescribed procedures under direct surveillance of a technically-trained supervisor.

Before any refueling operations begin, shutdown procedures for the Safety Injection System will be completed. The procedures are discussed in WCAP-3269-31. (3)

Monitors equipped with audible alarms, will be maintained in continuous operation in the immediate work vicinity to advise personnel of any increased activity.

FIGURE 3-10
NEUTRON POWER FOLLOWING COMPLETE
LOSS OF PUMP POWER WITH REACTOR
TRIP

FIGURE 3-11
CORE HEAT FLUX FOLLOWING COMPLETE
LOSS OF PUMP POWER WITH REACTOR
TRIP

FIGURE 3-12
MINIMUM DNB RATIO VERSUS TIME
FOLLOWING COMPLETE LOSS OF PUMP
POWER WITH REACTOR TRIP

3.3.2 Fuel Handling Precautions

Control Rod Cluster Withdrawal

The following safeguards will be employed to avoid accidents due to inadvertent rod withdrawal:

- a. Prior to disconnecting drive mechanism power supplies, the drive mechanism will be disengaged from the drive shaft.
- b. Prior to removal of the reactor vessel head, boron concentration of water in the reactor and its coolant loops will be increased to refueling concentration and verified by analysis. Refueling concentration is sufficient to maintain the fully-loaded, cold, clean core subcritical by 10 percent $\Delta k/k$ with all rod cluster control assemblies inserted. This concentration is sufficient to maintain the core subcritical even if all rods are removed.
- c. During removal of the reactor vessel head, a continuous audible signal from source range nuclear instrumentation will be monitored to verify that the core is not approaching a critical multiplication factor. As the vessel head is raised, a visual check is made to verify that the drive shafts are free in the mechanism housings.
- d. After the vessel head is removed, the control cluster drive shaft assemblies are removed from their respective control clusters using the manipulator crane and special drive shaft unlatching tool. A spring scale will be used to determine that the drive shaft is free of the control cluster as the lifting force is applied.
- e. During reactor refueling operations, a continuous audio count rate signal from the source range nuclear instrumentation will be provided in the control room and within the sphere in the refueling cavity area.

Transfer and Storage of Spent Fuel Assemblies

Storage racks in the spent fuel pit are arranged so that spent fuel assemblies cannot attain a critical geometry even with unborated water in the spent fuel pit. This is applicable in storage, during transfer, or in the process of loading the fuel shipping cask. The racks are spaced in a square pitched pattern, 20 inches on centers with 12 inches between assemblies. Water in the spent fuel pit is borated to 2,900 ppm because there can be an interchange of water with the refueling cavity.

Water Shielding

Fuel handling manipulators and hoists are designed to reach their physical limit on lift before raising a fuel assembly or control cluster above a height consistent with adequate water shielding. This safeguard applies to handling facilities in both the Containment Sphere and the spent fuel pit area. Radiation levels at the refueling water surface in the spent fuel pit and in the refueling canal are monitored continuously to warn personnel of any increased activity.

Heat Removal

Adequate cooling of fuel during underwater handling is accomplished by convective heat transfer to the surrounding water, where the assembly is immersed continuously while in the refueling cavity or spent fuel pit. Analysis has shown that forced circulation through the transfer tube is not required. In the event the carrier, containing a spent fuel assembly, became stuck in the transfer tube, adequate cooling will be maintained by natural convection. If some boiling did occur, there would be no danger of cladding failure at the existing conditions.

Protection Against Mechanical Damage

Gripping devices and handling fixtures used to lift fuel assemblies and other core internals are designed to avert accidental breach of cladding during fuel handling. Equipment is designed to withstand seismic stresses of 0.5 g in the horizontal and vertical directions resulting from ground accelerations. The new fuel and spent fuel assembly handling tools are designed so that engagement of the fuel assembly is accomplished with positive, cam-actuated latching fingers. A manually-operated, spring-loaded safety catch must be actuated to ensure positive engagement with the assembly before it can be lifted. The manipulator crane gripper is monitored by an electric load cell indicator which will stop the winch drive from moving in the up direction if the load exceeds a preset value. The gripper is also interlocked through the weight sensing device and a mechanical spring lock to prevent opening when supporting a fuel assembly.

Further protection against an accidental breach of cladding is provided by making it impossible to pass the fuel shipping cask above the spent fuel storage rack locations.

3.3.3. Assumptions and Method of Analysis - Due to fuel handling precautions, actual operating experience, and the inherent strength of fuel assemblies and individual fuel rods, it is highly improbable that a fuel handling accident involving a breach of clad integrity will occur. Even if clad damage was assumed to occur, the potential hazard would be small because of the low temperature of UO_2 (enhancing its retentivity for volatile fission products) and the

tendency of the refueling water to absorb any non-gaseous fission products which might escape as the gas bubbles through it.

For purposes of analysis, it has been assumed that an entire row of fuel rods (14) has been damaged and the cladding breached. It takes approximately 90 hours after a reactor shutdown before a fuel assembly is accessible for removal from the core. The resulting fuel rod gap noble gas activity release is based on data presented in Paragraph 5.5 Spent Fuel Sources.

3.3.4 Results - The noble gas activity released from the fourteen damaged fuel rods is as follows:

<u>ENERGY GROUP</u>		<u>NOBLE GAS ACTIVITY, γ/SEC</u>
I.	$E \leq 0.4$ Mev	4.26×10^{13}
II.	$0.4 \text{ Mev} < E \leq 0.8$ Mev	8.0×10^7
III.	$0.8 \text{ Mev} < E \leq 1.7$ Mev	Negligible
IV.	$E > 1.7$ Mev	Negligible

<u>ISOTOPE</u>	<u>CURIES</u>
Kr-85	120
Kr-85m	Negligible
Kr-87	Negligible
Kr-88	Negligible
Xe-133m	6.9
Xe-133	1000
Xe-135	Negligible

3.4 STEAM LINE BREAK

3.4.1 General - The integrity of the turbine cycle steam system affects the safety of the plant in two principal ways:

a) A rupture in this system increases the rate of heat extraction by the steam generators and causes cooldown of the reactor coolant. If the nuclear power required to meet this apparent load increase is excessive, or if the cooldown and depressurization result in inadequate subcooling, the reactor is tripped to ensure the minimum desired margin to DNB in this phase of the transient.

b) Cooldown after trip, due to continued discharge of steam, reduces the shutdown reactivity margin. In the more severe cases, injection of borated water is required to terminate this reactivity transient within safe limits.

In both respects, analysis of the conditions ensuing from a steam break shows that the safety of the general public is assured by the continued integrity of the Reactor Coolant System barrier, and that the possibility of core damage exists only in the most extreme and highly improbable cases. The analysis covers the full range of possible break sizes, up to and including severance of main steam piping at the point which yields the highest postrupture rate of steam flow. All cases of uncontrolled steam discharge by means other than pipe rupture, for example improper opening of the steam dump or bypass valves, are well within this range.

3.4.2 Facility Design Aspects

Effect of Steam Break

A steam break is reflected in the Reactor Coolant System as a sudden increase in turbine cycle load causing a reduction in Reactor Coolant System temperature, pressure and pressurizer level. During the first phase of the transient, the reactor control system responds to increase reactor power in attempting to maintain constant average temperature. Normal overpower trip protection will terminate this power increase without core damage. If the break is so large that the reactor control system cannot compensate for the decrease in temperature and pressure, a low pressure reactor trip may occur before the overpower trip.

During the second phase of the transient, which follows reactor trip, continued steam flow through the break further reduces the Reactor Coolant System temperature, pressure, and pressurizer level. With a negative moderator coefficient, an increase in reactivity accompanies this cooldown, thereby reducing the shutdown margin. Transfer of the reactor coolant makeup control to the borate mode could be made in order to help compensate for this reactivity gain.

If the cooldown rate is sufficiently rapid, coolant contraction will cause the coincident low pressurizer pressure and low pressurizer level condition which automatically initiate borated water addition by the Safety Injection System. Injection continues until the pressurizer is refilled with water or until the reactor coolant pressure rises to the shutoff pressure of the Safety Injection System. Regardless of break size, the amount of boron injected into the system in this manner is sufficient to maintain the core in a subcritical condition.

Effect of Break Inside Containment Sphere

If the break is inside the containment sphere, the Reactor Coolant System cooldown is terminated when the steam pressure in the steam generator approaches the equilibrium containment sphere pressure. If the break is outside the sphere, cooldown is terminated when the water in the feedwater side of all steam generators is completely boiled away.

For steam breaks ranging from small size up to that equivalent to the inadvertent opening of all steam dump valves, the protection described above prevents core damage in all cases, including those which combine the following most adverse conditions:

1. Highest worth control rod stuck in the withdrawn position
2. Most negative moderator coefficient
3. Minimum shutdown margin available with the remaining control rods

To minimize the probability of this transient occurring due to inadvertent opening of steam dump valves, the air operators of these valves are designed so that loss of supply air pressure would cause these valves to remain closed. An interlock is provided in the steam dump controller to reduce the probability of a spurious steam dump. This interlock requires a large turbine-generator load decrease in coincidence with high reactor coolant average temperature to actuate steam dump.

Under nominal conditions, even the highly improbable case of a main steam pipe severance would not result in core damage. Should a very large area break occur in coincidence with the adverse conditions itemized above, however, a momentary return to criticality could occur. If the flux distribution resulting from the single stuck rod is sufficiently distorted, DNB may occur, causing clad deformation and possible bursting in a limited core region. Public and plant personnel safety is assured by the fact that the transient is self-limiting (rising temperatures, coolant voids, and continued boron addition act to terminate the transient), and by the fact that any fission products released are confined to the Reactor Coolant System.

3.4.3 Assumptions and Method of Analysis - Steam break analysis is performed by utilizing a detailed analog simulation of the plant.

The steam flow through the break is determined as a function of steam generator pressure, with allowance for pressure losses up to the location of the break. Critical flow occurs at the break during most of the transient. The flow is proportional to steam generator pressure. The station operates with an increasing steam pressure during decreasing load. Thus the maximum initial steam flow through the break occurs with the plant at low power.

The transient temperature, power and pressure behavior of both the Reactor Coolant System and the steam system is determined. The reactor coolant pumps are conservatively assumed to maintain full flow during the cooldown transient, resulting in the greatest rate of heat transfer to the turbine cycle, and thus the most rapid cooldown of the Reactor Coolant System. The initial full power heat transfer coefficient from the Reactor Coolant System to the Turbine Cycle System also is assumed to exist during the transient. Once the pressurizer is emptied, the Reactor Coolant System pressure is determined analytically, using the temperature output from the analog study combined with a mass and volume balance in the system.

A core water density calculation is then made to determine the net reactivity. For the most pessimistic reactivity calculation, the end-of-life (most negative) moderator coefficient is used, and the variation of coefficient with temperature is included. Minimum shutdown reactivity in rods is assumed, with the highest worth rod remaining fully withdrawn after reactor trip.

The analysis incorporates a time delay for the injection of borated water, consisting of the following:

- a. Time required for cooldown to the point where the requisite safety injection trip conditions are reached
- b. Response time for instrumentation to initiate the safety injection signal
- c. Time for operation of valves, starting of pumps, and the sequencing of interlocks in the Safety Injection System
- d. Transport delay for the flow of borated water through the intervening pipe from the Safety Injection System to the reactor core

The rate of injection of borated water is determined as a function of the reactor coolant pressure according to the injection system head-flow characteristics. The injected water is assumed to mix with the reactor coolant flow past the inlet connection, and mixed boron concentration and enthalpy are determined. The boron concentration of the coolant in the core increases at intervals corresponding to the transit time of the coolant loop as the incoming safety injection water mixes with the previously borated loop flow.

In accordance with the design philosophy of the Safety Injection System as applied to loss of coolant protection, safety criteria are met if any single malfunction occurs in conjunction with the steam break. Consistent with this philosophy, protection is evaluated on the conservative assumption that only one of the two trains of the Safety Injection System will operate.

Another single malfunction case evaluated is one which could result in the injection of unborated water into the Reactor Coolant System. Interlocks are provided on each feedwater train to assure that

- a. Borated water has been flushed through the feedwater pump and its suction and discharge piping
- b. Condensate isolation valves have completely closed before the safety injection header isolation valves can start to open

Furthermore, the condensate and heater drain pumps, which normally deliver unborated water to the feedwater pump suction, are automatically stopped by the safety injection signal. With this arrangement, the most adverse case of single malfunction causes injection of only that quantity of unborated water (approximately 23 cubic feet) initially contained between the feedwater pump suction and the discharge isolation valve in each of the safety injection trains. In this case, the reactivity effect of this malfunction is shown to be within acceptable limits, considering that the unborated flow is mixed with the borated water pumped by the other safety injection branch, and is preceded into the Reactor Coolant System by approximately six times its own volume of fully borated safety injection water which is always maintained in the delivery pipes during reactor operation. The Safety Injection System then continues to supply fully borated refueling water.

If, in a particular case, the cooldown results in a temporary return to criticality, the equilibrium power level reached is conservatively determined from an estimate of the minimum (absolute value) power coefficient. The rate of reactivity addition due to cooldown is slow enough to permit calculation of power on an equilibrium reactivity basis, i.e., there is no rapid nuclear excursion as a result of a return to critical.

The degree of possible core damage is calculated, assuming DNB to occur over the entire core. A heat transfer coefficient for film boiling is calculated and used, together with the stuck rod power distribution and calculated equilibrium power level, to determine cladding temperatures. The calculated cladding temperature would cause internal gas pressure to burst the rod. The percentage of rods exceeding this clad temperature is then calculated.

3.4.4. Results

General

The maximum initial steam flow is calculated for a postulated break in the main 24-inch pipe. The break is assumed to occur at the tee connecting the steam generator for Loop B, from which the flow splits to the ring supply header. Flow is assumed to occur from all three connections to the tee.

The total flow through the break is directly proportional to the initial steam generator pressure. Since the steam pressure increases with decreasing load, maximum initial steam flow through the break exists at low power. For this break during full power operation, the steam flow from the steam generators would suddenly increase to approximately 400 percent of full load flow, including 100 percent flow continuing to the turbine until the reactor and turbine trips occur. From the no-load condition at a higher initial steam pressure (923 psia), the steam flow would suddenly increase to 385 percent of full load steam flow, with no flow to the turbine. This initial steam flow, without the compensating

effect of a reduction of flow with turbine trip, results in a more rapid reactor coolant cooldown rate. The no-load condition is therefore the most severe steam break condition.

Figures 3-13 and 3-14 show the reactor coolant and steam generator temperatures as a function of time after the break for the no-load condition. The total steam flow from the steam generators is also shown which decreases as the steam generator pressure decreases. Figure 3-15 shows the pressurizer pressure and water volume versus time. The Safety Injection System is actuated on a coincidence of low pressurizer pressure and low pressurizer level 14 seconds after the break. For the no load case the feedwater pumps are assumed to be in a standby condition and a starting delay of 10 seconds is assumed to bring the feedwater pump up to speed. A seven-second delay is also assumed for the operation of the safety injection valves admitting safety injection water to the Reactor Coolant System. The total delay between actuation and the time safety injection water reaches the reactor coolant cold leg is 17 seconds. An additional transport delay of three seconds is assumed before the borated water reaches the core.

Figure 3-16 shows the reactivity insertion in the core resulting from the cooldown followed by safety injection. The core is assumed to be just critical at the initiation of the transient. The increase in reactivity is calculated for the most negative moderator coefficient, which occurs at the end of core life when the Reactor Coolant System is essentially boron free. The decrease in reactivity at 16 seconds is a result of a reactor trip actuated by the safety injection signal. The assumed shutdown is 2.2 percent δk based on the end of life value assuming the highest worth rod stuck out and including margin for uncertainties. The reactivity continues to increase with continued cooldown until the borated water from the safety injection reaches the core at about 34 seconds. This figure shows the effect with one safety injection train and feedwater interlocks assumed to work. The small increase in reactivity at about 42 seconds is the effect of the 3.4 cubic feet of clean water which cannot be swept out of the system. The second rapid decrease in reactivity occurs at 45 seconds and is the result of the transport of borated water around the loop which then mixes with the safety injection stream essentially doubling the boron concentration. The continuum of this curve follows the same general trend. For this case there is a short term power increase to less than 50 percent power, and following reactor trip the core is maintained in a subcritical condition. If both safety injection trains function, the transient up to 34 seconds is the same and the reduction resulting from safety injection will be essentially twice the value shown and similarly at 45 seconds.

Several combinations of interlock failures were investigated for the no load steam break transient. Figure 3-17 shows the reactivity transient for one safety injection train assuming the valve interlock fails leading to the addition of approximately 23 cubic feet of clean water after an initial

Maximum Steam Break
Hot Zero Power
One Safety Injection Train and
Interlocks Function

FIGURE 3-13
CORE WATER AND STEAM GENERATOR
TEMPERATURES RELATED TO TIME AFTER
BREAK

Maximum Steam Break From Hot Zero Power,
Steam Generator

FIGURE 3-14
STEAM GENERATOR FLOW AND
TEMPERATURES RELATED TO TIME
AFTER BREAK

FIGURE 3-15
PRESSURIZER PRESSURE AND WATER
VOLUME RELATED TO TIME AFTER BREAK

FIGURE 3-16
REACTIVITY INSERTION IN CORE
RESULTING FROM COOLDOWN

FIGURE 3-17
REACTIVITY TRANSIENT FOR ONE SAFETY
INJECTION TRAIN

injection of 105 cubic feet of borated water at 2,900 parts per million. The clean water addition is limited by the automatic condensate pump trip. Since the valve interlock is assumed to fail, the borated safety injection water reaches the core seven seconds earlier. The short-term spike in reactivity at 35 seconds results from the unborated water addition from safety injection mixing with the original unborated reactor coolant prior to a complete loop transport of the borated water. No credit is taken for mixing in the core inlet plenum. Figure 3-18 shows other combinations of failures, none of which results in a return to criticality following the reactor trip. With both safety injection trains operating, and even if pump speed interlocks were assumed to fail, the unborated water addition effect is attenuated because the piping of two trains is not symmetrical in regard to the initial borated water addition or to the subsequent unborated water addition. There is, therefore, a mixing of borated water in one train with the unborated water from the other train.

Hypothetical Injection of Unborated Water

In order to estimate what limits may exist in the addition of unborated water during a steam break, a hypothetical calculation was performed. A shutdown of 1 percent δk instead of the calculated 2.2 percent δk was assumed. Both trains of the Safety Injection System were assumed to inject unborated water for an assumed period of time after the initial borated water is swept out of the injection lines. The resulting reactivity and power transients were calculated using the CHIC-KIN code (J. A. Redfield, "CHIC-KIN" - A Fortran Program for Intermediate and Fast Transients in a Water Moderated Reactor, WAPD-TM-479, January 1965) including void reactivity shutdown effects. The inputs to the code as a function of time were core inlet temperature, system pressure, reactivity from control rods following reactor trip and boron concentration changes resulting from safety injection. The external reactivity addition is shown in Figure 3-19 where the effect of reactor trip and boron concentration changes from safety injection are shown. The reactivity effects of Doppler and moderator are computed by the code.

A reactor trip is assumed to be effective at 16 seconds. Borated water reaches the core at 34 seconds followed by unborated water at 42 seconds. The effect of the loop transport delay is seen in the reactivity decrease at 34 seconds followed by another decrease at 45 seconds. The resultant heat flux transient is shown in Figure 3-20. The effect of the peak heat flux was analyzed assuming the most adverse power distribution with a control rod stuck in the fully withdrawn position. The corresponding radial power distribution is shown in Figure 3-21 with hot channel factor given as a function of number of rods exceeding the factor.

A conservative estimate of core damage resulting from this hypothetical case is obtained by assuming DNB to occur and by determining the resultant cladding temperatures as a function of the power distribution. A conservative low film coefficient is calculated for the low pressure condition in the core; the value

Maximum Steam Break
Hot Zero Power
Both Safety Injection Trains Operating

FIGURE 3-18
REACTIVITY TRANSIENTS FOR BOTH
SAFETY INJECTION TRAINS

FIGURE 3-19
REACTIVITY INPUT TO CHIC-KIN CODE
FOR "HYPOTHETICAL" CLEAN WATER
ADDITION DURING STEAM BREAK

FIGURE 3-20
MAXIMUM STEAM BREAK "HYPOTHETICAL"
CLEAN WATER ADDITION WITH ONE
PERCENT SHUTDOWN FROM HOT ZERO
POWER

FIGURE 3-21
RADIAL POWER DISTRIBUTION WITH ONE
STUCK ROD

Nuclear Radial Hot Channel Factor

used was 600 Btu per hour per square foot per $^{\circ}\text{F}$. A cladding temperature of 1,300 degrees Fahrenheit was calculated as a conservative upper limit for possible clad bursting. For these conditions, the peak heat flux at 44 seconds would yield no bursting. The total amount of unborated water that would be inserted into the reactor coolant up to about 52 seconds is approximately 450 cubic feet. After 52 seconds it is assumed the injected water is borated. Note that this is a hypothetical case with only one percent shutdown assumed and a total unborated water addition of about 450 cubic feet. The reactivity transient with the available 2.2 percent shutdown is shown in Figure 3-22 which shows there is no return to criticality after the reactor trip. These considerations show that there is no severe consequence in adding as much as 450 cubic feet of unborated water during the steam break even under conservative assumptions for shutdown margin.

Figures 3-23, 3-24 and 3-25 show the transient for the maximum steam break flow from initial operation at full power. The reactor is tripped at about five seconds from the overpower trip. There is an overshoot in nuclear power but no approach to DNB since the thermal flux is lagging and the coolant temperatures are decreasing. The reactivity transient is shown for 2.2 percent shutdown, and there is no return to criticality. One safety injection train is assumed to work and all interlocks are assumed to function. The initial cooldown rate for the full load case is less than that for the no load case since the change in steam flow through is less. There is a sudden reduction in steam generator steam flow at 6 seconds when the turbine is tripped.

The effect of an even smaller initial steam flow is shown in Figure 3-26. The initial flow for this case is equivalent to a sudden opening of all steam dump valves yielding a sudden increase of 40 percent of full load steam flow from the no load condition. With this smaller additional flow the pressurizer pressure and level do not reach the safety injection trip point until about 60 seconds. The net reactivity insertion at this time is essentially the same as that introduced at safety injection trip point for the most severe break. The subsequent insertion rate due to cooldown is slower, however, for the smaller steam flow, and there is no return to criticality. The assumption made for this case was that both safety injection trains are working but both speed interlocks fail leading to the addition of about 45 cubic feet of unborated water.

3.4.5 Conclusions - The effect of uncontrolled heat extraction from the Reactor Coolant System resulting from a steam supply system rupture was analyzed. For the break resulting in the maximum steam flow the Safety Injection System together with the shutdown available in rods is sufficient to prevent core damage. This analysis was performed for the most severe conditions with respect to reactivity addition, i.e., end of core life with the most negative moderator coefficient of reactivity.

2.2% Shutdown from Hot Zero Power

FIGURE 3-22
MAXIMUM STEAM BREAK WITH
"HYPOTHETICAL" CLEAN WATER
ADDITION

Maximum Steam Break
103% Power
One Safety Injection Train
and Interlocks Function

FIGURE 3-23
CHANGES IN STEAM FLOW AND
TEMPERATURE RELATED TO TIME
AFTER BREAK

Maximum Steam Break
103% Power
One Safety Injection Train and Interlocks Function

FIGURE 3-24
CORE WATER AND STEAM GENERATOR
TEMPERATURES RELATED TO TIME
AFTER BREAK

Maximum Steam Break
103% Power
One Safety Injection Train and Interlocks Function

FIGURE 3-25
REACTIVITY RELATED TO TIME
AFTER BREAK

FIGURE 3-26
ACCIDENTAL OPENING OF ALL STEAM
DUMP VALVES FROM NO LOAD
CONDITION

Several cases of assumed Safety Injection System interlock failures were investigated leading to various amounts of clean water addition. These analyses showed that in all cases a single interlock and in most cases multiple interlock failures can be tolerated during the steam break transient without adverse effects. A hypothetical case of reduced shutdown margin was studied with uncontrolled clean water addition following the initial injection of boroated water contained in the safety injection piping. The results showed that even under these circumstances, the injection of 450 cubic feet of unbored water resulted in no core damage.

3.5 STEAM GENERATOR TUBE RUPTURE

3.5.1 General - The integrity of the steam generator reactor to turbine cycle barrier is of significance from a radiological safety point of view because it prevents the transport of contaminants from the Reactor Coolant System to the turbine cycle. Should penetration occur, the steam piping, turbine, condenser, condensate, and feedwater systems would confine the non-gaseous material that might be carried by the steam. Gaseous activity, however, would be released through the air ejectors to the plant stack.

A complete severance of a steam generator tube has been analyzed to show the effectiveness of plant operating limits (such as maximum permissible coolant activity) and emergency shutdown procedures in restricting the stack release to safe levels, in this limited case of tube rupture.

Buildup of non-gaseous activity in the steam generator shell-side liquid following tube rupture is not of immediate significance, because there is no direct path for the release of this activity. This liquid is sampled as a safeguard against improperly discharging blowdown wastes containing reactor coolant activities above permissible levels.

Because the steam generator tube rupture results in a loss of reactor coolant, it imposes a requirement on supplemental coolant makeup, by charging flow and possibly safety injection, to prevent pump cavitation and eventual uncovering of the core. Protection against a loss of coolant of the magnitude of a tube rupture was analyzed in the category of small breaks in Paragraph 3.1. That analysis shows that there would be no release of fission products to the turbine cycle other than those initially present in the reactor coolant prior to the break.

3.5.2 Facility Design Aspects - In the design and manufacture of the steam generator, prevention of tube rupture or separation at the tube sheet joint is implemented by:

- rigid quality control of the Inconel tubing material
- conformance to Section VIII of the ASME Boiler and Pressure Vessel Code, and

- hydrostatic testing at a pressure differential between tube and shell of 2.3 times that which exists in normal service.

Crevice corrosion tests have been performed under conditions much more severe than would be expected in the plant. The tests showed that Inconel is highly resistant to this type of attack.

Tube support plates are designed to withstand the hydraulic forces associated with five times full rated steam flow without structural damage.

A rupture or pullout failure of a steam generator tube has never occurred in a unit designed to these or equivalent standards, despite several hundred unit-years of service.

Should such a failure occur, however, the following plant features and procedures would be employed to limit the consequences:

a) The air ejector radiation alarm would be actuated about 16 seconds after rupture by the appearance of Nitrogen-16, a shortlived reactor coolant activation product. Proper functioning of the radiation monitor would be verified, and then the reactor would be manually tripped. Other indications of tube rupture would include: stack radiation alarm; steam generator blowdown sample radiation alarm; loss of normal level in pressurizer; and volume control tank alarms.

b) Maximum available reactor coolant makeup capacity is brought into operation automatically as a result of falling water level in the pressurizer. This automatic action includes starting the standby charging pump. If these measures are inadequate to balance the rate of coolant loss, the Safety Injection System is automatically actuated. In either case, sufficient supplementary water is added from the refueling water storage tank to prevent uncovering the top of the core. The same condition which initiates safety injection would cause an automatic reactor trip if manual trip had not been executed.

c) To minimize the discharge of reactor coolant through the ruptured tube, the pressure differential across the break would be reduced by manually initiating cooldown and depressurization of the Reactor Coolant System. In this procedure, boron concentration is brought to the cold shutdown level or greater by the addition of refueling water through the charging pumps, and/or safety injection. Shell-side pressure is manually regulated by operating the steam bypass valves which direct steam to the condenser; primary pressure is regulated by manually operating the pressurizer spray valves.

d) Cooldown is accomplished under manual control at a rate not exceeding the temperature transient rate established for reactor vessel thermal stress limitation. When the appropriate pressure and temperature are reached, the residual heat removal loop is placed in operation. Further requirement for steam bypass having been eliminated, the reactor coolant pumps are stopped, the steam bypass valves are kept closed, and any additional leakage through the break is confined to the steam generator.

3.5.3 Assumptions and Method of Analysis - The consequences of a steam generator tube rupture can be analyzed by determining the maximum reactor coolant discharge rate for reactor and turbine cycle pressure conditions, as well as temperature conditions encountered during the recovery procedure outlined above. The net effect of discharge and makeup from available sources is integrated over the term of the recovery.

To determine the maximum transfer of reactor coolant to the turbine cycle, it is conservatively assumed that the operator on duty will neglect to take action when radiation alarms first indicate the occurrence of a break. Instead, it is assumed that the initial action will be only by automatic control. Reserve charging flow would be initiated by low charging header pressure (the condition that arises when the pressurizer level control calls for more flow than the charging pump can deliver). The reactor trip and safety injection would be actuated when the requisite two-out-of-three low pressure and low pressurizer level signals occur.

After safety injection trip, it is assumed that cooldown and depressurization would be executed by manual control at the allowable rate, with the leak rate during this period based on a differential pressure of 200 psi across the ruptured tube. A net positive pressure differential of about 200 psi is maintained to prevent cavitation of the reactor coolant pumps. Therefore, it would not be practical to eliminate the driving force for leakage across the break during this phase.

Because no cladding rupture would occur, the radiological consequences can be conservatively assessed by assuming that all of the radioactive material initially present in the reactor coolant would be transferred through the break in the course of the recovery period. This assumption is pessimistic because some of the initial coolant activity will remain in the Reactor Coolant System, diluted by the makeup water.

It is assumed that the gaseous fission products, xenon and krypton, would separate from the condensate without holdup in the turbine cycle. These gases would be carried to the plant stack through the air ejectors as long as it was necessary to maintain condenser vacuum. Vacuum would be maintained throughout the detection and depressurization phases, until the residual heat removal loop is placed in operation.

Non-volatile fission and corrosion products would largely concentrate in the feedwater of the faulty steam generator. A fraction of this activity would be carried over the moisture droplets to the steam piping, thence to the turbine moisture separators (prior to trip) and the condenser. During a brief interval after reactor trip, automatic pressure control in the steam system would actuate a steam dump to the atmosphere. The reactivity associated with moisture droplets (estimated at 0.25 percent of the steam mass flow) is assumed to be discharged as airborne particles in a ground-level cloud. During the remainder of the recovery period, non-volatile activity in carryover would accumulate in the condenser liquid phase. Partition of the radio-iodine between the liquid and the non-condensibles leaving the condenser is estimated based on a decontamination factor of 10,000. It is assumed that the resulting vapor fraction would go to the stack by way of the air ejectors.

The magnitude of the radioactive discharges calculated in this manner are a measure of the worst-case consequences of a tube rupture for a given level of activity in the reactor coolant. The environmental consequences of such a release, show the relationship of normal and maximum coolant activity to consequences considered limiting for public safety in this incident.

3.5.4 Results - The analysis outlined above was performed for the hypothesis of a double-ended rupture of one steam generator tube, occurring while the plant is at full rated power with a maximum steam bypass requirement. The calculated fission product releases, based on a reference reactor coolant activity level corresponding to one percent defective fuel rods, were as follows:

- a) During the period of direct steam bypass to the atmosphere, which lasts approximately 60 seconds, the release of entrained iodine activity as equivalent I-131 is calculated to be 0.005 curies. In evaluating the release of a mixture of iodine isotopes with respect to permissible limits under 10 CFR 20, each isotope can be treated as an equivalent amount of I-131, obtained by dividing the actual number of curies by the ratio:

$$\frac{\text{Maximum Permissible Concentration of the Isotope in Question}}{\text{Maximum Permissible Concentration of I-131}}$$

A similar procedure is followed in the case of a noble gas mixture, using Xe-133 as the reference activity.

Noble gas activity released during the same interval by way of the atmospheric bypass would be 850 curies as equivalent Xe-133. This release would begin approximately two minutes after the occurrence of the tube rupture, corresponding to the time the reactor trip were initiated by the safety injection signal. If the reactor were manually tripped prior to this signal, the activity release would be less, due to the reduced accumulation of reactor coolant impurities in the feedwater at the time the steam bypass was opened.

b) Throughout the remainder of the depressurization procedure, noble gas and entrained soluble activities would be carried to the condenser by the intermittent steam flow required for cooldown and residual heat dissipation. The total carryover was calculated for the end-of-plant life case (longest cooldown, hence maximum integrated steam flow). The resulting release at the plant stack, integrated over the 4.3-hour cooldown period, was 26,443 curies equivalent Xe-133, and less than 0.031 curies equivalent I-131.

3.5.5 Conclusions - The result for other fuel defect conditions would be directly proportional to the assumed percentage of defective rods. The magnitude of the release derived above for the reference case of one percent defective fuel rods is not hazardous at the San Onofre site. In fact the xenon release is only 3% and the iodine release only 0.01% of that which could be released within the yearly limits of 10 CFR 20 with unfavorable atmospheric dispersion conditions.

3.6 LOSS OF LOAD

3.6.1 General - A loss of load analysis is performed to determine the effects of a rapid and large reduction in generator load which would in turn cause a rapid and large reduction in the heat extracted from the Reactor Coolant System. If the decrease is greater than that which the system is designed to accept, with no protective action the resultant increase in reactor coolant temperature and pressure could exceed the normal core operating limits. To preserve the integrity of the cladding, the Reactor Control and Protection System initiates a reactor trip in time to prevent DNB. The trip will also limit the maximum pressure rise in the Reactor Coolant System; however, ultimate overpressure protection by the pressurizer safety valves is capable of limiting the Reactor Coolant System pressure to a value below the trip point, even for the maximum possible loss of load. Protection of these barriers is therefore independent.

3.6.2 Facility Design Aspects - The station is designed on a conservative basis to accept, without reactor trip but with automatic steam release to the atmosphere and condenser, a 30 percent step reduction from 450-MWe generator load and a 50-percent step reduction from 395 MWe load. A 10-percent step reduction of 450-MWe load would not require steam release.

The greater the magnitude of the sudden load decrease the more severe are the system transients. To prevent the possibility of clad damage in severe cases beyond 30-percent reduction from 450-MWe load, the Reactor Control and Protection System is designed so that a reactor trip initiated by conditions exceeding either the variable low pressure set point, or in some cases, the high pressurizer pressure or high pressurizer level set points, will prevent DNB. As described in WCAP-3269-45(1) Reactor Control and Protection System, each of these channels functions on a two-out-of-three coincidence principle.

The most likely source of a complete loss of load is a turbine generator trip. In this case there is a direct reactor trip signal derived from turbine auto-stop oil pressure (a two-out-of-three circuit) and there is no temperature or pressure increase in the Reactor Coolant System. The plant behavior is, however, evaluated for a complete loss of load from 450 MWe without a reactor trip, primarily to show the adequacy of the pressure relieving devices, as well as to show that no core damage occurs. The Reactor Coolant System and Turbine Cycle System pressure relieving capacities are designed for this condition.

A partial loss of load may result from turbine governor response to a sudden significant increase in frequency of the distribution network - a remote possibility in view of the electrical characteristics of the Southern California Edison Company and San Diego Gas & Electric Company systems to which the San Onofre plant is connected. In this case there is an increase in reactor coolant and steam temperature, and system pressures will increase.

The Reactor Control and Protection System and steam release to the atmosphere and condenser, if in automatic control, will respond to try to limit these excursions. If the load decrease is within their capability, equilibrium system conditions are restored at the reduced load without further incident. If the load decrease exceeds the capability of these systems, or if the systems are in manual control, a reactor trip will be initiated to protect the core. Automatic reactor control and/or steam dump are not required as a design basis for the safety of the plant.

3.6.3 Method of Analysis and Selection of Parameters - The loss of load transient is analyzed as part of the complete and detailed plant analog simulation study. The objectives of this analysis are twofold: to determine margins to core protection limits, and to establish pressure relieving requirements for the reactor coolant and turbine cycle systems.

Conservative results for the loss of load transient are assured by the following assumptions:

- a) Initial operating conditions - the initial power level, coolant temperature and pressure are assumed to be at their maximum values consistent with steady-state operation, including allowances for calibration and instrument errors. This assumption results in the maximum power difference for the load loss, and the minimum margin to core protection limits at the initiation of the loss of load.

The initial conditions are:

$$\text{Power} = (1.03 (1,347 \text{ MWt}))$$

$$T_{in} = 553^{\circ}\text{F} + 4^{\circ}\text{F} = 557^{\circ}\text{F}$$

$$\text{Pressure} = 2,100 + 30 = 2,130 \text{ psia}$$

- b) Moderator coefficient of reactivity - assumed to be the most positive value expected during core life, i.e., beginning of life, while operating at power; this results in the minimum power decrease during the transient, and may result in a power decrease ($\alpha_w = +0.5 \times 10^{-4} \text{ } \delta k/oF$).

- c) Doppler coefficient of reactivity - assumed to be the smallest (absolute value) negative coefficient, again resulting in the maximum power during the transient ($\alpha_f = -1.0 \times 10^{-5} \delta k/^\circ F$).
- d) Reactor control - the reactor is assumed to be in manual control; no control rods are assumed to be inserted, either manually or automatically, prior to the initiation of reactor trip.
- e) Steam release - steam release to the atmosphere and condenser is assumed to be inoperative.
- f) Pressurizer spray and power operated relief valves - no credit is assumed for the beneficial effect of spray and relief valves in reducing or limiting pressurizer pressure.

3.6.4 Results - The transient response for a complete loss of load from 450-MWe operation is shown in Figures 3-27 and 3-28. The assumptions for initial conditions and parameters are as given in Paragraph 3.6. Loss of Load. The results are primarily shown to indicate the effective capacity of plant pressure relieving devices without an immediate reactor trip and with a positive moderator coefficient. For this case, the reactor is assumed to continue at power until the high pressurizer water level trip is reached at 34 seconds. The reactor would actually be tripped immediately upon a turbine trip from a signal from low autostop oil pressure (two-out-of-three circuit) and there would be no temperature or pressure increase in the Reactor Coolant System. For the case assumed above, the maximum volume insurge before trip is conservatively estimated, considering an initial minimum pressurizer water level, including adverse instrument and control errors, and considering a maximum level trip point, again including adverse instrument errors. Even under these assumptions there is only a 25-cubic foot overshoot in volume surge. The maximum surge is therefore well below filling the 1,300-cubic foot pressurizer volume. An instrument delay of two seconds was assumed for trip actuation. The steam generator temperature (hence pressure) rises to the safety valve setpoint (nominal 1,000 psia) in about seven seconds where the safety valves have sufficient capacity to maintain pressure throughout the transient. The pressurizer pressure increases to the safety valve setpoint (nominal 2,500 psia) in 12 seconds, where the capacity is sufficient to maintain pressure.

The analysis of plant protection is based upon the fixed high pressure reactor trip actuated at 2,250 psia, which will limit the transient at approximately seven seconds and prevent DNB in the core. With the high pressure reactor trip the minimum DNB ratio achieved during the transient is 1.55. Note the increase in reactor power due to the positive moderator coefficient. No control rod motion is assumed to occur before the reactor trip. The maximum overpower trip level with errors is 118 percent, with a minimum possible trip point of 103 percent. Thus the overpower could also result in a trip to limit the transient.

FIGURE 3-27
TRANSIENT RESPONSE FOR A FULL LOSS
OF LOAD FROM 450 MWE POSITIVE
FACTOR

FIGURE 3-28
TRANSIENT RESPONSE TO A FULL LOSS
OF LOAD FROM 450 MWE

With a negative moderator coefficient of reactivity the transient is less severe since core power level will decrease before the trip, as shown in Figures 3-29 and 3-30. The conditions are the same as those in Figures 3-27 and 3-28, with the exception of the moderator coefficient equal, in this case to -0.5×10^{-4} δk per degree Fahrenheit. The resulting transient shows a high pressure reactor trip at about eight seconds, and the minimum DNB ratio is 1.96. With the most adverse errors leading to the maximum volume insurge before the pressurizer high water level reactor trip (300 cubic feet), a level trip would not occur. For this case the level trip was assumed to occur at 250 cubic feet. Note that for this case, equilibrium conditions are being approached, with a resultant very small rate of insurge to the pressurizer.

In actual operation the pressurizer pressure would be rapidly reduced to approximately 2,150 psia by the power operated relief valves and by pressurizer spray, which are both duplicated. The reduced pressure, in conjunction with the elevated reactor coolant temperatures, would result in a variable low pressure reactor trip, a further protective feature for the transient.

The minimum DNB ratio reached, when the high level trip is assumed, is 1.68, showing that the fixed high pressure reactor trip is not necessary for DNB protection when a negative moderator coefficient exists, causing a core power reduction during the transient. The pessimistic combination of most positive moderator coefficient and least negative Doppler coefficient was selected for the transient shown in Figures 3-27 and 3-28.

3.6.5 Conclusions - The complete loss of load, without a direct and immediate reactor trip, presents no hazard to the integrity of the Reactor Coolant System and turbine cycle, even under the very conservative assumptions. Pressure relieving devices incorporated in the two systems are more than adequate to limit the maximum pressures. The integrity of the core is maintained by the fixed high pressurizer pressure trip for the loss of load transient under conditions with a positive moderator coefficient. The high pressure trip is not required for core protection with a negative moderator coefficient.

FIGURE 3-29
TRANSIENT RESPONSE FOR A FULL LOSS
OF LOAD FROM 450 MWE NEGATIVE
FACTOR

FIGURE 3-30
NEGATIVE TRANSIENT RESPONSE TO A
FULL LOSS OF LOAD FROM 450 MWE

3.7 REFERENCES

1. WCAP-3269-45, "Reactor Control and Protection Systems - Functional Requirements and Performance," by W. King, March 1965.
2. WCAP-5584, "DNB Prediction for an Axially Non-Uniform Heat Flux Distribution," by L. S. Tong, September 1965.
3. WCAP-3269-31, "Development of Auxiliary Systems," by H. J. Cordle, February 1965.
4. WCAP-3269-55, "Core Physics Characteristics of the First Loading of the San Onofre Nuclear Generating Station," by D. L. Miller (to be published).

4.0 CHEMICAL REACTIONS

4.1 INTRODUCTION - Exothermic chemical reactions between commonly used fuel cladding metals (principally Zircaloy) and a steam atmosphere have been shown to occur under conditions possible in the event of a major loss-of-coolant accident. Although the initial core cladding material is stainless steel, it is expected that future cores will employ Zircaloy cladding. Consequently, the plant systems and Containment Sphere have been designed to accommodate entire cores clad with Zircaloy. The method of analysis of potential zirconium-water reactions is discussed in this chapter.

4.2 THE ZIRCONIUM-WATER REACTION - The zirconium-water reaction is described by the chemical equation:

The amount of heat liberated in this reaction is approximately 2,800 Btu per pound of reacting zirconium. The heat release in calories-per-gram-mole of solid zirconium reacted as a function of absolute reaction temperature on the Kelvin scale is given by the expression,⁽¹⁾

$$\Delta H = -149,120 + 8.5\% - 1.44 \times 10^{-3} T^2 + \frac{2.41 \times 10^5}{T}$$

For reaction of molten zirconium, the heat of fusion, given as 5,500 calories-per-gram-mole⁽¹⁾ (9,900 Btu. per pound) must also be added.

The hydrogen produced in the reaction is evolved at temperatures much greater than the ignition temperature, and burns as it comes in contact with the air. The energy release from the hydrogen combustion equivalent may be as much as an additional 2,150 Btu per pound of zirconium reacted. For purposes of sphere analysis, the hydrogen is assumed to be immediately released from the reactor coolant system as soon as it is generated.

4.3 PARABOLIC RATE EQUATION - Following a loss-of-coolant accident in which the core is exposed to a steam atmosphere, zirconium undergoes a corrosion-type reaction with steam. The reaction achieves significant rates only after the cladding has been heated by residual heat generation to temperatures in excess of 1,800°F. The rate of reaction is dependent upon the availability of steam, the thickness of the zirconium oxide layer developed at the cladding surface, and the temperature of the metal exposed to the steam. The following expression, known as the parabolic rate equation, relates weight of zirconium reacted to time and temperature, when an excess of steam is available:

$$W^2 = 33.3 \times 10^6 (t) \exp - \left(\frac{45,500}{RT} \right)$$

where

W = weight of metal reacted, mg/cm²

t = time, seconds

R = universal gas constant, 1,986 calories per mole, K

T = absolute temperature, K

This equation has been verified in several independent tests. The constants used were evaluated by Baker and Just from experimental findings of several investigators.⁽²⁾

4.4 STAINLESS STEEL REACTIONS - Stainless steel reactions with steam are less significant than zirconium-steam reactions, with regard to both rate and quantity of energy release to the sphere. The stainless steel reaction liberates between 259 and 455 Btu per pound of metal reacting.⁽³⁾ Excluding the energy of hydrogen-oxygen recombination the stainless steel reaction releases less than one-fifth of the energy of the zirconium-water reaction. Thus, the effect of reaction on the rate of rise of cladding temperature is less important with stainless steel. Also, stainless steel reacts much more slowly than zirconium. The extent of reaction has been observed to be significant only at or above the melting point of approximately 2,500°F.

Therefore, the total amount of stainless steel reaction over the periods of time that are of interest will be much less than for zirconium. Less cooling is required to stop the reaction, since the energy released from the reaction of stainless steel is not regenerative to the extent of the zirconium-water reaction. Consequently, the metal-water reaction analyses have been based upon the use of Zircaloy cladding.

4.5 ZIRCONIUM-WATER REACTION ENERGY RELEASE CALCULATIONS - Evaluation of the energy release associated with the zirconium-water reaction required application of the parabolic rate equation considering the transient rise in temperature of the core following the loss-of-coolant accident. The availability of steam to continue the reaction is also considered in evaluating the reaction energy release. The boil-off of water remaining below the core after the initial blowdown is governed by conduction of heat from the lower reactor vessel structure and the thermal and gamma radiation heating from the core. The rate of reaction is limited by the parabolic rate law early in the transient where an excess of steam is available and cladding temperatures are low. The reaction rate is limited by the rate of steam evolution later in the transient, when cladding temperatures are high, and when the parabolic rate equation indicates a rate of reaction higher than the rate at which steam can be generated to support the reaction.

4.6 LOCAL TRANSIENT CALCULATIONS - Where an excess of steam is available, a digital computer code has been used to solve the controlling equations for local points in the core. Overall core conditions are determined by dividing the core into several volumes, each of which is represented by an average initial power level expressed as a fraction of the peak heat flux. The local transient calculation is performed for each value of initial heat flux. The total zirconium-water reaction energy is the sum of the energy release transients determined for each fraction of core volume.

4.6.1 Heat Transfer Model - The starting point for the calculation of the local cladding and fuel temperature transients is the initiation of the loss-of-coolant accident. The calculation is based on a heat balance which considers heat transfer between fuel pellet and cladding as well as heat transfer between the cladding surface and coolant fluid. A lumped parameter technique is employed for a unit length of fuel rod assuming no axial heat conduction. The heat transfer model utilizes three nodes in the fuel pellet and one node to represent the cladding. Physical properties (e.g., thermal conductivity, specific heat of the fuel, cladding, and oxide layer) are varied as a function of temperature. The model accounts for the thermal energy initially stored in the fuel at the beginning of the transient, the residual heat generation in the fuel during the transient, and the heat of reaction in the cladding based upon the parabolic rate equation.

4.6.2 Calculation Assumptions - Conservative assumptions which follow are employed throughout the calculation:

- a) During the initial phases of blowdown, a cladding surface, heat transfer coefficient of 1,000 Btu per hour per degree Fahrenheit per square foot is used. This heat transfer coefficient is approximately one-third of the value for normal operation and is assumed to drop to zero once the surface of reactor coolant falls below the core. This assumption of no heat transfer from the cladding surface is then used for the entire period of the transient until the reaction is halted when safety injection causes water to rise into the core. The resumption of cooling at this point causes cladding surface temperature to fall rapidly and halts the metal-water reactions.
- b) The hydrogen evolved is assumed to present no barrier for diffusion of the steam reactant to the cladding surface.
- c) In the event the cladding temperature exceeds the melting point of zirconium, the zirconium oxide layer is assumed to prevent the molten cladding from flowing into cooler regions where the reaction could stop due to a lowering of temperature. Under this assumption, the reaction continues at the rate determined by the parabolic rate equation.

d) Should the cladding temperature exceed the melting temperature of the oxide (approximately 4,800°F), the cladding is assumed to run off and drop into any water remaining in the bottom of the reactor vessel. An additional 10 percent of this metal will react on contact with the water.

e) If the rate of steam evolution from heat transferred to the water below the core is equal to or less than the consumption rate determined by the parabolic rate equation, the assumption is made that all the steam evolved takes part in the reaction.

4.7 RESULTING ZIRCONIUM-WATER REACTION ENERGY - Results of zirconium-water reaction calculations for a loss-of-coolant accident with a core containing 272 pound-moles of zirconium are shown in Figure 4-1. Representative core parameters are as follows:

Number of fuel rods	28,260
Length of fuel rod	10 ft
Cladding OD	0.422 in.
Cladding thickness	0.0243 in.
Fuel pellet diameter	0.3669 in.
Reactor power	1,347 MWt

Curve A in Figure 4-1 shows the results of the energy release calculations over the period of interest before the reaction is terminated by safety injection. There is no appreciable reaction during the first 20 seconds following the hypothetical pipe rupture. This period elapses before the hottest portions of the cladding are heated significantly above the reaction threshold temperature by residual heat generation. Initially, the reaction rate is determined by the parabolic rate equation. The boil-off rate of water remaining below the core in the reactor vessel exceeds requirements at this time. As larger portions of the core are heated above the reaction threshold, the reaction proceeds at a rate limited by the availability of steam. When the reaction is terminated by water rising above the bottom of the core (64 seconds after the break as defined in Paragraph 3.1), a maximum of 2.4 percent of clad has reacted, adding approximately 1.6 million Btu of reaction energy to the residual heat stored in the core, and liberating approximately 1.3 million Btu directly to the containment sphere from combustion of the evolved hydrogen.

Curve B in Figure 4-1 is an extension of the energy release curve, assuming an excess of steam where the reaction rate is limited only by the parabolic rate equation.

FIGURE 4-1

It should be noted that the 64-second period prior to termination of the reaction is based upon the following assumptions of safety injection system performance:

- a. One safety injection train is functioning and one of the three injection lines spills its flow.
- b. The reactor vessel has no water remaining after the initial blow-down (a conservative assumption, considering that the reaction calculation is based on the existence of water below the core, which is heated to form the vapor necessary to sustain the reaction).

In the more likely event that all three valves open, both pumping trains function, and one line delivers ineffectively into the ruptured loop, the refilling time is reduced from 64 seconds to 46 seconds for the hypothetical break. A corresponding reduction in total reaction to less than 0.7 percent of the cladding would be obtained.

4.8 REFERENCES

1. WAPD-SC-543, "Zirconium-Water Reaction Data and Application to PWR Loss-of-Coolant Accident," by B. Lustman, May 1957.
2. ANL-6548, "Studies of Metal-Water Reactions at High Temperatures, III Experimental and Theoretical Studies of the Zirconium-Water Reaction," by L. Baker and L. Just, Argonne National Laboratory, May 1962.
3. "U. S. Reactor Containment Technology," W. B. Cottrell and A. W. Savolainen.

5.0 SOURCES OF RADIOACTIVITY

5.1 INTRODUCTION - This chapter defines the quantities of fission product and other radioactive materials subject to release under the conditions of various postulated accidents as a basis for the calculation of potential radiation dosage.

The following sources are considered:

<u>Source</u>	<u>Related Accident Analysis</u>
Core Fission Products	Rod ejection (limited fuel damage), Loss of coolant (limited fuel damage), Hypothetical core meltdown
Reactor Coolant	Loss-of-coolant without fuel damage, Steam generator tube rupture
Auxiliary Systems Sources	Waste gas release, Volume control tank rupture
Spent Fuel Sources	Fuel handling accident

The effect of plant features which limit the extent of the release (for example, the fraction of the core which undergoes a given type of damage) is discussed in the section of this report dealing with the particular accident or malfunction assumed to cause the release.

5.2 CORE FISSION PRODUCTS

5.2.1 Fission Product Inventory - The buildup of core fission products is calculated with the digital computer which numerically solves the following differential equations of production and removal for all significant mass-number chains:

For parent nuclides in the pellet,

$$\frac{dN_{pi}}{dt} = F y_i P - (\lambda_i + \nu_i + \phi \sigma_i) N_{pi} \quad (1)$$

For daughter nuclides in the pellet,

$$\frac{dN_{pj}}{dt} = F y_j P + \lambda_i N_{pi} - (\lambda_j + \nu_j + \phi \sigma_j) N_{pj} \quad (2)$$

For parent nuclides in the gap,

$$\frac{dN_{gi}}{dt} = \nu_i N_{pi} - (\lambda_i + \phi\sigma_i)N_{gi} \quad (3)$$

For daughter nuclides in the gap,

$$\frac{dN_{gj}}{dt} = \nu_j N_{pj} + \lambda_i N_{gi} - (\lambda_j + \phi\sigma_j) N_{gj} \quad (4)$$

where

- F = fission rate per megawatt-second thermal
- N = population of nuclide
- P = power level, megawatts thermal
- t = irradiation time, sec.
- y = fission yield, atoms per fission
- λ = radioactive decay constant, sec^{-1}
- ν = escape coefficient for diffusion out of UO_2 , sec^{-1}
- ϕ = thermal neutron flux, neutrons/ $\text{cm}^2\text{-sec}$
- σ = thermal neutron capture cross section, cm^2
- g refers to pellet-clad gap
- i refers to parent nuclide
- j refers to daughter nuclide
- p refers to fuel pellets

By applying proper boundary conditions, the computation is made for the separate power histories of the three core regions, and irradiation time is selected to give the maximum aggregate fission product inventory.

5.2.2 Calculation of Meltdown Release - For use in the analysis of the hypothetical core melt-down accident, the isotopes are grouped according to their relative tendency to be released from the core at elevated temperature. The following classification and release fractions are used:

- a. From any fuel rod in which the peak clad temperature exceeds an assumed failure limit, all noble gases and halogens present in the pellet-clad gap are assumed released.

- b. From any pellet whose average temperature exceeds 2,500°F, the TID-14844 release fractions are assumed. (The TID-14844 release fractions are: 100 percent of the noble gases, 50 percent of the halogens, and one percent of the remaining species.)

Tables 5-1 and 5-2 present results of the aggregate core fission product calculation. For consideration of direct gamma radiation, Table 5-1 presents the source strength of those fission products subject to release from the fuel pellets and from the pellet-clad gap, respectively. Source strength is expressed in Mev/sec for each of five photon energy groups, computed on a megawatt thermal basis for the portion of the core exceeding the limits described.

Inhalation hazard following a fission product release is primarily due to radioactive iodine. Table 5-2 gives the curie inventories of the iodine isotopes subject to release from the fuel pellets and from the pellet-clad gap, respectively. Like the direct gamma sources, these data are also given on a megawatt thermal basis for the involved portions of the core.

To calculate the source release in a case where only a part of the core exceeds the pellet release criteria, the distribution of core power as a function of core volume is employed. The gross fission product distribution follows that of the thermal power. In Figure 5-1 the average thermal power of the hottest volume fraction ϕ_v , expressed as a fraction of equilibrium core power, is plotted as a function of ϕ_v .

In a given accident case, the core volume increment reaching a given temperature is determined by the method described in Section 4, Chemical Reactions. From this core volume increment (ϕ_v), the fraction of equilibrium core power is obtained from Figure 5-1. The product of the fraction of equilibrium core power and the gross thermal power of the core gives the power produced by the volume increment under consideration. The fission product release is then obtained from the data of Tables 5-1 and 5-2 for that power.

The fraction of the core reaching clad failure temperature will generally exceed that in which the average pellet temperature exceeds 2,500°F. The releases under these two criteria are determined separately and added to obtain the total release.

5.3 REACTOR COOLANT ACTIVITY - The inventory of radioactive material in the reactor coolant may be divided into three classes: fission products, activated corrosion products, and products of the activation of water and dissolved air. Only the first two are significant as accident sources, since the products of coolant and dissolved air activation decay within a few seconds to levels which are quite low in comparison with the others.

Table 5-1 Direct Radiation Sources Following Fuel Meltdown

	TIME AFTER SHUTDOWN						
	0	1 Hr	2 Hr	8 Hr	1 Day	1 Wk	1 Mo
Source and Energy Group	Radiation Sources Released, Mev/sec-Mwt						
Part A: Fission Products from Fuel Pellets							
0 - 0.4 Mev/γ	3.3×10^{14}	3.1×10^{14}	3.1×10^{14}	3.0×10^{14}	2.0×10^{14}	3.5×10^{13}	6.0×10^{12}
0.4 - 0.8 Mev/γ	3.6×10^{15}	2.7×10^{15}	2.3×10^{15}	1.7×10^{15}	1.2×10^{15}	2.7×10^{14}	1.5×10^{14}
0.8 - 1.3 Mev/γ	1.0×10^{15}	0.7×10^{15}	6.1×10^{14}	3.7×10^{14}	2.2×10^{14}	6.3×10^{13}	4.6×10^{13}
1.3 - 1.7 Mev/γ	6.1×10^{15}	1.7×10^{15}	1.5×10^{15}	0.7×10^{15}	2.5×10^{14}	2.7×10^{13}	5.9×10^{12}
>1.7 Mev/γ	4.0×10^{15}	2.0×10^{15}	1.5×10^{15}	3.5×10^{14}	0.9×10^{14}	2.2×10^{13}	1.5×10^{11}
Part B: Fission Products from Pellet-Clad Gap							
0 - 0.4 Mev/γ	7.0×10^{13}	7.0×10^{13}	7.0×10^{13}	6.8×10^{13}	6.0×10^{13}	4.2×10^{13}	1.0×10^{13}
0.4 - 0.8 Mev/γ	8.0×10^{13}	8.0×10^{13}	7.5×10^{13}	6.4×10^{13}	5.6×10^{13}	4.8×10^{13}	4.8×10^{13}
0.8 - 1.7 Mev/γ	8.0×10^{12}	7.2×10^{12}	6.2×10^{12}	4.0×10^{12}	1.9×10^{12}	9.5×10^{11}	7.0×10^{11}
>1.7 Mev/γ	6.2×10^{12}	5.8×10^{12}	5.0×10^{12}	2.3×10^{12}	9.0×10^{11}	5.5×10^{11}	4.5×10^{11}

Table 5-2 Radioactive Iodine Sources Following Fuel Melting

Source and Release Fraction		Time After Shutdown						
		0	1 Hr	2 Hr	8 Hr	1 Day	1 Wk	1 Mo
Part A: 50% Iodine from Fuel pellets	Isotope	Iodine Released, Curies/Mwt						
	I-131	1.3×10^4	1.3×10^4	1.3×10^4	1.2×10^4	1.1×10^4	6.7×10^3	3.0×10^2
	I-132	1.9×10^4	1.4×10^4	1.1×10^4	1.7×10^3	1.3×10^1	-	-
	I-133	2.8×10^4	2.8×10^4	2.7×10^4	2.2×10^4	1.3×10^4	1.1×10^2	-
	I-134	3.4×10^4	1.6×10^4	7.2×10^3	6.2×10^1	-	-	-
	I-135	2.6×10^4	2.4×10^4	2.1×10^4	1.1×10^4	2.2×10^4	-	-
Part B: 100% Iodine from Pellet-Clad Gap	I-131	3.2×10^2	3.2×10^2	3.2×10^2	3.0×10^2	2.7×10^2	1.7×10^2	0.7×10^1
	I-132	2.0×10^2	1.5×10^2	1.2×10^2	2.5×10^1	5.1	1.4	-
	I-133	7.7×10^1	7.7×10^1	7.5×10^1	6.1×10^1	3.6×10^1	-	-
	I-134	4.1	1.9	-	-	-	-	-
	I-135	2.3×10^1	2.1×10^1	1.9×10^1	1.0×10^1	-	-	-

FIGURE 5-1
DISTRIBUTION OF THERMAL POWER AS A
FUNCTION OF CORE VOLUME

5.3.1 Fission Products - The fission product content of the reactor coolant is computed with the digital code FIPCO-2 described in Paragraph 5.2.1. For this result, the code makes use of the following differential equations:

For parent nuclides in the coolant,

$$\frac{dN_{wi}}{dt} = Dv_i N_{pi} - \left(\lambda_i + R\eta_i + \frac{B'}{B_o - tB'} \right) N_{wi} \quad (5)$$

For daughter nuclides in the coolant,

$$\frac{dN_{wj}}{dt} = Dv_j N_{pj} - \left(\lambda_j + R\eta_j + \frac{B'}{B_o - tB'} \right) N_{wj} + N_i N_{wi} \quad (6)$$

where

- D = fraction of fuel rods having defective cladding
- R = purification flow, coolant system volumes per sec.
- B_o = initial boron concentration, ppm
- B' = boron concentration reduction rate by feed and bleed, ppm per sec
- η = removal efficiency of purification cycle for nuclide
- w = refers to coolant

Boundary conditions assumed to obtain a numerical solution include the following:

- a. The purification demineralizer operates at a constant volumetric flow rate, with an assumed removal factor of 10 for all isotopes except noble gases, yttrium, molybdenum, and cesium for which no removal effect is assumed.
- b. Use of feed and bleed to the radwaste disposal system at a variable volumetric rate, as required to reduce boron concentration at a rate of 4 ppm per day.
- c. Degassing and dilution of the reactor coolant with the refueling water at each refueling shutdown.
- d. Reactor shutdown, with no fission yield or neutron activation, during the appropriate phase of each core cycle.

5.3.2 Corrosion Products - Corrosion product activities are calculated by the digital computer which calculates the concentration of active nuclides formed by neutron irradiation and subsequent exchange of core-deposited crud with that in suspension. Removal by purification, deposition, and decay are taken into account.

Tables 5-3 and 5-4 give the maximum concentrations and total system inventories of the significant fission and corrosion products, expressed in terms of isotopic quantities and as photon energy groups for direct gamma source application. The table is based on a maximum power rating of 1347 MWt, and on a reference defective fuel rod population of one percent.

5.4 AUXILIARY SYSTEMS SOURCES

5.4.1 Volume Control Tank - In the volume control tank, maximum activity level occurs when the chemical and volume control system is operating with the purification demineralizer bypassed. In this case, the contained source consists of reactor coolant activity essentially equal to that listed in Table 5-3 on a mass basis, but greater by a factor of 1.3 on a volumetric basis due to the density change. The vapor space of the volume control tank contains gaseous fission products in equilibrium with the entering reactor coolant. The specific activity of the vapor phase is calculated using Henry's Law constants for the volatile fission products (noble gases and halogens).

The resulting liquid and vapor phase activities are listed in Table 5-5 for the reference condition of one-percent fuel defects, 1,347 MWt and the minimum steady-state liquid volume in the tank (to yield a maximum volatile fission product source).

5.4.2 Waste Gas Decay Tanks - The other auxiliary system location in which it is possible to accumulate a significant quantity of radioactive material with a potential for release is the waste gas decay tanks.

The maximum inventory of the waste gas decay tanks is calculated by assuming the reactor undergoes a cold shutdown at the point in core life when reactor coolant activity is a maximum, and that in the process all gaseous fission products are stripped, compressed, and stored in the gas decay tanks. The quantities of these gases can be obtained from Table 5-3. In so doing, the effect of decay during the shutdown and stripping operation is neglected. Table 5-6 lists separately the noble gas activities pertinent to the waste gas decay tank source, both as isotopic quantities and as photon energy group source strength.

Table 5-3

Maximum Fission and Corrosion Product Isotopic Inventories of
Reactor Coolant for Reference One Percent Fuel Defect Case
(Based upon 570°F Coolant)

Isotope	Specific Activity Microcuries/cc	Coolant Inventory, Curies
Br-84	0.16	31
Kr-85	4.2	800
Kr-85 (m)	1.3	250
Kr-87	0.73	140
Kr-88	2.1	400
Rh-88	2.1	400
Rh-89	5.0×10^{-2}	10
Sr-89	2.0×10^{-3}	0.4
Sr-90	1.0×10^{-4}	0.02
Y-90	1.9×10^{-4}	0.04
Sr-91	1.0×10^{-3}	0.2
Y-91	2.8×10^{-2}	5.3
Mo-99	2.5	477
Te-129	1.6×10^{-2}	3
I-129	2.4×10^{-8}	0.5×10^{-5}
I-131	1.4	267
Te-132	0.13	25
I-132	0.50	95
I-133	1.9	363
Xe-133	158	30,200
Te-134	1.4×10^{-2}	2.7
I-134	0.26	50
Cs-134	0.68	130
I-135	0.99	189
Xe-135	4.4	840
Cs-136	7.0×10^{-2}	13
Cs-137	15.7	3000
Xe-138	0.32	61
Cs-138	0.52	100
Ba-140	2.1×10^{-3}	0.4
La-140	7.1×10^{-4}	0.1
Co-60	1.5×10^{-3}	0.29
Fe-59	1.9×10^{-3}	0.36
Co-58	0.5×10^{-3}	1.6
Mn-56	2.3×10^{-2}	4.4
Mn-54	4.4×10^{-3}	0.84

Table 5-4 Maximum Fission and Corrosion Product Gamma Sources of
Reactor Coolant for Reference One Percent Fuel Defect Case

	TIME AFTER SHUTDOWN						
	0	1 Hr	2 Hr.	8 Hr	1 Day	1 Wk	1 Mo
Source and Energy Group, Mev	Radiation Sources Released, Mev/sec						
0.4	1.0×10^{14}	1.0×10^{14}	9.9×10^{13}	9.4×10^{13}	8.0×10^{13}	3.6×10^{13}	1.9×10^{12}
0.8	9.2×10^{13}	8.0×10^{13}	7.8×10^{13}	7.6×10^{13}	7.4×10^{13}	6.9×10^{13}	6.9×10^{13}
1.7	1.5×10^{13}	9.7×10^{12}	6.5×10^{12}	5.2×10^{12}	2.9×10^{11}	2.9×10^{10}	1.9×10^{10}
2.5	4.8×10^{13}	3.1×10^{13}	2.3×10^{13}	2.9×10^{12}	9.0×10^{11}	-	-
Isotope	Iodine Activities, Curies						
I-131	267.0	267.0	267.0	260.0	244.0	147.0	20.0
I-132	88.0	66.0	50.0	8.6	0.086	neg	neg
I-133	363.0	351.0	340.0	279.0	163.0	13.8	neg
I-134	50.2	22.5	9.6	0.096	neg	neg	neg
I-135	189.0	172.0	154.0	82.3	15.7	neg	neg

Table 5-5 Direct Gamma Sources Contained in Volume Control
Tank for Reference One-Percent Fuel Defect Case

Energy Group, Mev	SPECIFIC ACTIVITY Mev/cc per sec		GROSS ACTIVITY Mev/sec	
	Liquid Phase	Vapor Phase	Liquid Phase	Vapor Phase
0.4	7.5×10^5	8.3×10^5	9.0×10^{11}	9.9×10^{11}
0.8	6.2×10^5	3.2×10^5	7.4×10^{11}	3.8×10^{11}
1.7	10.5×10^4	2.5×10^5	12.5×10^{10}	3.0×10^{11}
2.5	3.2×10^5	1.4×10^6	3.9×10^{11}	1.7×10^{12}

Table 5-6 Direct Gamma Sources and Isotopic Inventory of Waste Gas Decay Tanks for a Reference One-Percent Fuel Defect Case

Energy Group, Mev	Gross Activity Mev/sec
0.4	1.1×10^{14}
0.8	5.0×10^{12}
1.7	4.0×10^{12}
2.5	2.8×10^{13}
Isotope	Inventory, Curies
Kr-85 M	265
Kr-85	2556
Kr-87	149
Kr-88	428
Xe-133	32,200
Xe-135	897
Xe-138	65

5.5 SPENT FUEL SOURCES - The potential for release of fission product volatile activity contained in the pellet-cladding gap region of an irradiated fuel assembly is calculated for fuel which has undergone the maximum design burnup. It is assumed that a decay period of 90 hours elapsed in the process of shutting down the reactor, removing the head, and transferring the first fuel assembly to the spent fuel building.

For use in an evaluation of cladding damage in a spent fuel handling incident, the gap inventory of noble gas isotopes is listed in Table 5-7 on a per-megawatt basis. The potential release in the event a given number of fuel rods undergo clad ruptures would be obtained by multiplying the table value by the thermal power rating of the involved fuel rods. Non-gaseous isotopes are excluded, because of their essentially complete retention by the fuel and clad surfaces at the low temperatures of the spent fuel pit, and by the scrubbing effect of the fuel pit water should a release occur.

Table 5-7 Direct Gamma Sources and Isotopic Inventory for the Spent Fuel Gap Activity 90 Hours Following Shutdown

Energy Group, Mev	Noble Gas Activity (α /sec) (14 Fuel Rods)
0.4	4.26×10^{13}
0.8	8.0×10^7
1.7	Negligible
2.5	Negligible
Isotope	Inventory, Curies/Mwt
Kr-85	1.8×10^2
Kr-85m	1.2×10^{-4}
Kr-87	10^{-10}
Kr-88	10^{-10}
Xe-133m	10.4
Xe-133	1.5×10^3
Xe-135	5.8×10^{-3}

APPENDIX A

GENERAL ASSUMPTIONS FOR ACCIDENT ANALYSES

REACTOR TRIP

A reactor trip signal acts to open the two series trip breakers feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the control rods which are then inserted, by gravity fall, into the core. There are various instrumentation delays associated with each tripping function including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time the signal reaches the trip level, depending on the particular transient, to the time the rods are free and begin to fall. The time delay assumed for each tripping function is given in Table A-1.

Table A.1 Assumed Time Delays

<u>Tripping Function</u>	<u>Time Delay (Seconds)</u>
Startup rate	3.5
Overpower (nuclear)	0.32
Variable low pressure (includes delay in temperature detectors)	2.82
High pressurizer pressure	2.0
High pressurizer level	2.0
Low reactor coolant flow (from loop flow detectors)	0.8

The reactivity insertion following a reactor trip is a function of the acceleration of the control rods and the variation in rod worth as a function of rod position. The control rod position is determined experimentally as a function of time in a simulated reactor trip using an actual prototype assembly under simulated flow conditions. Control rod position versus time is shown in Figure A.1. Reactor coolant flow simulated for this case was conservatively set at 120 percent of normal flow.

Control rod reactivity insertion as a function of rod position in the core is shown in Figures A.2 through A.3 in terms of percentage of total worth. This function is shown for both hot zero power and hot full power conditions. These functions when combined with the position versus time function

FIGURE A.2
CONTROL ROD REACTIVITY INSERTION
VERSUS ROD POSITION FOR HOT ZERO
POWER CONDITIONS

FIGURE A.3
CONTROL ROD REACTIVITY INSERTION
VERSUS ROD POSITION FOR HOT FULL
POWER CONDITIONS

REACTOR TRIP (continued)

given in Figure A.1 result in the reactivity insertion versus time functions shown in Figures A.4 and A.5. The zero power function was conservatively applied for all reactor trips since this results in the longest delay before an effective trip.

HOT CHANNEL CONDITIONS

Unless stated otherwise in the sections describing specific accident conditions, the margins to DNB were determined on the basis of design hot channel factors F_q (heat flux engineering rise hot channel factor) = 3.23 and $F_{\Delta H}$ (enthalpy rise engineering hot channel factor) = 1.88. The conservative basis for the various subfactors which make up these total factors are presented in References 1 and 2. As shown in Reference 1, the computed nuclear hot channel factors are substantially below the design objectives consistent with the total factors given above. The in-core instrumentation system will be employed to check the initial power distributions and to follow the power distribution throughout core lifetime. The axial flux distribution is conservatively assumed to be an extended cosine distribution with a maximum to average ratio of 1.78.

REFERENCES

1. WCAP-3269-55 "Core Physics Characteristics of the First Loading of the San Onofre Nuclear Generating Station" by D. L. Miller (to be published).
2. WCAP-5584, "DNB Prediction for an Axially Non-Uniform Heat Flux Distribution", by L. S. Tong, September 1965.

FIGURE A.4
REACTIVITY INSERTION VERSUS TIME
FOR REACTOR TRIP, HOT ZERO POWER
CONDITIONS, 120 PERCENT REACTOR
COOLANT FLOW

REACTIVITY INSERTION
PERCENT OF TOTAL WORTH

FIGURE A.5
REACTIVITY INSERTION VERSUS TIME
FOR REACTOR TRIP, HOT FULL POWER
CONDITIONS, 120 PERCENT REACTOR
COOLANT FLOW