Westinghouse Technology Systems Manual

Chapter 2

CORE CHARACTERISTICS

Section

2.0 Core Characteristics
2.1 Reactor Physics Review
2.2 Power Distribution Limits

Westinghouse Technology Systems Manual

Section 2.0

Core Characteristics

2.0 CORE CHARACTERISTICS

This chapter provides an introduction to some of the terms, theories, operations and controls that are used in association with the pressurized water reactor core. The two sections in this chapter are Core Physics, Section 2.1, and Power Distribution Limits, Section 2.2.

Section 2.1, Core Physics, introduces the fission process which is the heart of the heat production process in the core. This section discusses the neutron life cycle from the production of the neutron until it is absorbed within the core or it escapes. The fission process can be affected by both naturally occurring, and operator initiated factors. The origin and control of these factors will also be discussed.

Section 2.2, Power Distribution Limits, addresses the need for establishing limits on the power produced in the core. Two major areas of concern in this section are the thermal-hydraulic characteristics and the neutron (power) distribution within the core. In order to preclude any damage to the core during normal operations and to limit the amount of core damage under accident conditions, licensing limits are placed upon the operation of the core. These limits are discussed in this section.

Westinghouse Technology Systems Manual

Section 2.1

Reactor Physics Review

TABLE OF CONTENTS

2.1 REACTOR PHYSICS REVIEW 2.1-1

2.1.1 Introduction 2.1-1

2.1.2 Fission Process 2.1-1

2.1.3 Moderation 2.1-2

2.1.4 Nuclear Cross Section 2.1-3

2.1.5 Neutron Multiplication 2.1-4
2.1.5.1 Fast Fission Factor 2.1-5
2.1.5.2 Fast Nonleakage Factor 2.1-5
2.1.5.3 Resonance Escape Probability 2.1-5
2.1.5.4 Thermal Nonleakage Factor 2.1-6
2.1.5.5 Thermal Utilization Factor 2.1-6
2.1.5.6 Neutron Production Factor 2.1-6

2.1.6 Reactivity and Reactivity Coefficients 2.1-7
2.1.6.1 Fuel Temperature Coefficient 2.1-7
2.1.6.2 Moderator Temperature Coefficient 2.1-9
2.1.6.3 Void Coefficient 2.1-11
2.1.6.4 Pressure Coefficient 2.1-12
2.1.6.5 Power Coefficient and Power Defect 2.1-12

2.1.7 Poisons 2.1-12
2.1.7.1 Uncontrollable Poisons 2.1-12
2.1.7.2 Controllable Poisons 2.1-15

2.1.8 Reactor Kinetics 2.1-15

2.1.9 Subcritical Multiplication 2.1-16

LIST OF TABLES

2.1-1 Particles and Energy Produced per Fission Event 2.1-19
2.1-2 Neutrons per Fission 2.1-19
2.1-3 Typical Neutron Balance for U-235 .... 2.1-20

LIST OF FIGURES

2.1-1	Fission Neutron Energy
2.1-2	Flux Distribution
2.1-3	Total Cross Section for U-235
2.1-4	Cross Section Curve
2.1-5	Doppler Temperature Coefficient, BOL and EOL
2.1-6	Doppler Only Power Coefficient, BOL and EOL
2.1-7	Doppler Only Power Defect, BOL and EOL
2.1-8	Moderator Temperature Coefficient
2.1-9	Total Power Coefficient, BOL and EOL
2.1-10	Total Power Defect, BOL and EOL
2.1-11	Fission Yield versus Mass Number
2.1-12	Equilibrium Xenon Worth vs Percent of Full Power
2.1-13	Xenon Transients
2.1-14	Xenon Transients Following a Reactor Trip
2.1-15	Xenon Transients Following a Reactor Trip and Return to Power
2.1-16	Samarium Transients
2.1-17	Samarium Transients Starting with a Clean Core
2.1-18	Samarium Transients Starting with a Clean Core
2.1-19	Samarium Shutdown Transients
2.1-20	Integral and Differential Rod Worth
2.1-21	Reactivity versus Startup Rate

2.1 REACTOR PHYSICS REVIEW

Learning Objectives:

1. Define the following terms:

   a. K_{eff}
   b. Reactivity
   c. Reactivity coefficient
   d. Power defect
   e. Poison
   f. Critical
   g. Supercritical
   h. Subcritical
   i. Startup rate

2. Describe the following reactivity coefficients and explain how their values change with core life and reactor power level:

   a. Moderator temperature
   b. Fuel temperature (Doppler)
   c. Void
   d. Power

3. Explain the relative effects of the following poisons in plant operations:

   a. Xenon
   b. Samarium

4. Explain how the following controllable poisons affect core reactivity:

   a. Control rods
   b. Chemical shim

2.1.1 Introduction

This section represents a summary of basic nuclear physics and nuclear reactor design principles and terminology. The material presented is broader in scope than can be conveniently covered in the classroom time allotted; therefore, all the written material will not be covered in detail. Basic explanations and definitions of concepts will be given in the classroom. It is emphasized that the written material is only a summary of the subject.

2.1.2 Fission Process

Nuclear fission is the splitting of the nucleus of an atom into two or more separate nuclei accompanied by the release of a large amount of energy. The reaction can be induced by a nucleus absorbing a neutron, or it can occur spontaneously, because of the unstable decay nature of some of the heavy isotopes. Very few isotopes have been excited to the state where the fission reaction occurs. These have been in the heavy elements, generally uranium and above on the chemical scale.

Nearly all of the fissions in a reactor are generated in the fuel by neutron absorption, which can result in the splitting of the fissionable atoms that make up the fuel. Only a few of the heavy isotopes are available in quantities large enough or present a sufficient probability of fission to be used as reactor fuel. These are uranium-233 (U-233), uranium-235, (U-235), uranium-238 (U-238) for fast or high-energy fission only, plutonium-239 (Pu-239), and plutonium-241 (Pu-241). Several other isotopes undergo some fission but their contribution is always extremely small. U-235 and U-238 are naturally occurring isotopes with very long half-lives; they are generally the fuels used for reactors. Artificially produced fuels include U-233, which is produced by the irradiation of thorium-232 (Th-232) in a reactor, and Pu-239 (produced by irradiation of U-238 in a reactor). Th-232 and U-238 are called fertile materials and are generally placed in the core or in a blanket surrounding the reactor for the express purpose of producing fuel (fissionable material) as the original fuel is used up in fission. The ratio of the amount of fuel that is produced in a reactor to the amount that is used during any period of time is called the conversion ratio of the reactor. If the amount of fuel produced is greater than the amount consumed, then the excess fuel produced is called a breeding gain. The fission nucleus absorbs a neutron, and almost immediately a fission occurs. In the case of U-235 the reaction is represented by the following:

U-235 + n --> (U-236)*

(U-236)*--> FP_1 + FP_2 + 2.43 n + Energy

where FP = Fission Product
n = neutron
"*" indicates the isotope is unstable.

In atomic studies it has become the practice to express energies in "electron volt" units, abbreviated "eV". It has been determined that gamma energies are frequently on the order of a million electron volts (MeV) so that the MeV has become a convenient unit for stating these (and related) energies.

The fission of any of the fissionable isotopes produces gammas, neutrons, betas, and other particles. The total energy released per fission is about 207 MeV for U-235; this energy is distributed as shown in the following table:

The energy of the neutrinos, which accompany the radioactivity, is not available for producing power because these particles do not interact appreciably with matter; thus, the net energy available is 197 MeV, or roughly 200 MeV per fission. The Table 2.1-1 lists the particles and energy each particle produces per fission event.

Neutron production (neutrons per fission) varies with the different fissionable isotopes and with the energy at which the fission reaction is caused to take place. Table 2.1-2 shows some relative values for neutrons per fission for some of the common fuels that are now being used in reactors:

More than 99.3% of the total number of neutrons produced are produced within 10^{-14} seconds and are called prompt neutrons. It should be noted that each fission event does not produce the same number of neutrons. The number of neutrons per fission given in the above table represent an average number produced per fission.

The neutrons released from fission are not monoenergetic neutrons (all of one energy); they vary in energy from essentially thermal energy up to about 15 MeV. The energy distribution of these prompt neutrons is shown in Figure 2.1-1. The horizontal axis shows the range of prompt neutron energy distribution in MeV and the vertical axis shows the fractional neutron distribution in an incremental band (delta) around a selected energy level. The units for the vertical axis are fractional distribution per MeV. It can be seen that the area under the curve in an incremental band around 0.65 MeV yields the highest fraction of neutrons. Using the area under the curve it can also be seen that approximately 98% of all prompt neutrons are born at an energy level less than 8 MeV and the average prompt neutron energy is approximately 2 MeV.

2.1.3 Moderation

In an actual operating reactor the probability of fission for typical reactor fuels is dependent on the energy of the incident neutrons. Fission neutrons are born fast (at high energies), and the probability that they will cause a fission in U-235 at that energy is very small. It is necessary to reduce this kinetic energy in order to increase the chance that they will cause fission. This is accomplished by interposing relatively non-absorbing nuclei as collision media to absorb the kinetic energy of fission neutrons through the process of elastic scattering. This medium is called the "moderator." It acts to slow down or "thermalize" the fission neutrons.

Typical moderators are hydrogen, beryllium, and carbon. It should be clear that fewer collisions are necessary in a hydrogen medium to cause complete moderation than in carbon since the nuclear mass of hydrogen is smaller and therefore more likely to absorb kinetic energy of the neutron by the elastic scattering process.*

The amount of moderator in a multiplying system (a reactor, for instance) greatly influences the degree of slowing down that occurs. If there is too little moderator, the neutrons are not adequately slowed down. Therefore, the probability of fission is small when compared with optimum. If there is too much moderator, then the probability that a thermal neutron will be captured by the moderator (or some other non-fissionable material) is greatly increased. Figure 2.1-2 shows a snapshot of the neutron energy spectrum for a light water moderated reactor. The horizontal axis is neutron energies in electron volts (eV) on a logarithmic scale while the vertical axis is the neutron flux per unit energy and is in units of nv/eV. As shown in this figure two peaks occur. The first about the fast neutron energies due to the prompt neutron production (within 10^{-14} sec) of the fission events. The second peak occurs at thermal neutron energy levels. This is caused by the diffusion of the thermal neutrons until they are absorbed by either a poison or the U-235 fuel. The time it takes a neutron to slow down from fast to thermal energy is relatively short, 5 microseconds, as compared to the thermal diffusion time of 210 microseconds. This results in a relatively low number of intermediate energy neutrons and the
peak at the thermal energy level.

2.1.4 Nuclear Cross Section

The previous discussions of neutron reactions alluded to the fact that they have different probabilities of occurrence. A measure of the relative probability that a given reaction will occur is defined as the cross section of the nucleus for that specified reaction. More precisely, this measure is referred to as the microscopic cross section. In an approximate sense, the microscopic cross section may be considered as the effective area for interaction that the nucleus presents to the neutron.

The microscopic cross section has units of area (cm^2) and it is often expressed in units of barns (1 barn = 10^{-24} cm^2) for ease of manipulation. The microscopic cross section is represented by the symbol σ. It is made up of several component parts. The total cross section, σ_T is a combination of σ_c (capture), σ_s (scattering) and sometimes σ_f (fission), given by:

$$\sigma = \sigma_T = \sigma_c + \sigma_s + \sigma_f$$

where

σ_c is the area presented for neutron capture; it means that a neutron approaching an atom is exposed to an area of this apparent size

σ_s is the area of the nucleus that will scatter or deflect the neutron

σ_f is present in only a few of the many hundred nuclei, but it is the area that is presented for a neutron to strike the nucleus and cause a fission to occur

These component cross sections are a function of the target nucleus (U-235, boron-10, etc.) and incident neutron's energy (thermal, epithermal, or fast). In general, the probability of a given reaction will be determined by the neutrons energy. The probability of absorption of neutrons for fission tends to follow an inverse velocity trend. Figure 2.1-3 shows this general trend in U-235. The horizontal axis is neutron energy in electron volts (eV) on a logarithmic scale, while the vertical axis is total neutron cross section (fission, capture, and scattering) in barns. For energy levels above 100 eV, the predominant cross section is scattering. The neutrons collide with and bounce off the nucleus but no interaction takes place.

The peaks in the middle of the curve are due to resonance capture. A qualitative explanation of resonance capture is that nuclei have discrete excited states. If the energy of the incident neutron is such that the energy of the resultant compound nucleus is equal to one of these states, then the neutron has a high probability of being captured with no resultant fission. These discrete energies determine where these isotopes will have an abrupt increase in their capture cross section. These spikes in the cross section curve are referred to as resonance peaks.

Light element resonances are not as observable, since their peaks are very broad and give the effect of a smooth curve. Below 1 eV the predominant cross section is fission. The probability of fission increases as the energy level of the neutron decreases. This is referred to as the 1/v region and is approximately a linear function.

The total effective cross section presented by all of the nuclei of a given isotope in a cubic centimeter is referred to as the macroscopic cross section (Σ):

$$\Sigma = N\sigma$$

where N = the number of individual atoms (or nuclei) per cubic centimeter.

The macroscopic cross section has dimensions of reciprocal length (cm^{-1}). It is sometimes convenient to consider the macroscopic cross section as the probability of neutron interaction per unit track length. The reciprocal of Σ is called the neutron mean free path (denoted by λ) and is a measure of the average distance a neutron will travel in the substance before it experiences the nuclear reaction under consideration. If a material contains several isotopes (i.e., a compound or a naturally occurring material with several isotopes), then the effective macroscopic cross section is:

$$\Sigma_T = \Sigma_1 + \Sigma_2 + \bullet\bullet\bullet + \Sigma_i$$

$$= N_1\sigma_1 + N_2\sigma_2 + \bullet\bullet\bullet + N_i\sigma_i$$

2.1.5 Neutron Multiplication

We have discussed the fission process, moderation, and cross sections. In this section, these three topics are combined into the quantity called the "multiplication factor", which describes all the possible events in the life of a neutron and effectively describes the state of a nuclear reactor.

The multiplication factor in a nuclear system is a measure of the change in the fission neutron population from one neutron generation to the subsequent generation. If the multiplication factor for a reactor core (or any nuclear assembly) is less than 1.0, a condition known as subcritical, then the system is decaying or dying out and will never be self-sustaining. With a multiplication factor greater than 1.0, a condition referred to as supercritical, a nuclear system is producing more neutrons than are needed to be self-sustaining and is subjected to an increasing chain reaction that must be controlled by some exterior factor. The stable or critical condition of a nuclear system occurs when the multiplication factor is equal to 1.0 and there is no change in neutron population from one generation to the next. The effective multiplication factor is defined as:

$$K_{eff} = \varepsilon L_f p L_t f \eta$$

These symbols are defined and considered in detail in the following paragraphs.

In reactor operation, K_{eff} is the most significant property with regard to reactor control. At any specific power level or condition of the reactor, K_{eff} is kept as near to the value of 1.0 as possible. At this point in operation, the neutron balance is kept to exactly one neutron completing the life cycle for each original neutron absorbed in the fuel. An example of this balance is shown in Table 2.1-3.

The operational factors that affect reactor control are all-important because of the way they change the factors that make up K_{eff}. As seen in the previous table if any one of these factors making up K_{eff} changes, the the ratio of 1.0 will not be maintained. This resultant change in K_{eff} will either make the reactor subcritical or supercritical.

2.1.5.1 Fast Fission Factor

The fast fission factor, ε, is the contribution to neutron multiplication from the fissions that occur at higher-than-thermal energies. This contribution is from the fast fission in U-235 and U-238. The probability for a fission reaction in U-238 is relatively low, but there is so much of this isotope in the reactor core that there is a contribution to the multiplication factor. The fast fission factor is defined as the ratio of the neutrons produced by fissions at all energies to the number of neutrons produced in thermal fission. As core temperature is increased, the value of e is increased because more fast neutrons are present to fission the U-238 due to poorer moderating properties of the water. There is only a slight, almost insignificant, change over the core lifetime due to loss of U-238 by conversion to Pu-239.

2.1.5.2 Fast Nonleakage Factor

The fast nonleakage factor, L_f, is the fraction of neutrons that is not lost due to leakage from the core system during the slowing down process from fission energies to thermal energies. It is also the probability that a neutron will remain in the core and become a thermal neutron without being lost by fast leakage. It is represented by:

$$L_f = e^{-\tau B^2}$$

for the continuous slowing down model, or by:

$$L_f = \frac{1}{1+\tau B^2}$$

for the two-group model. "Two-group" indicates that the core can be described as consisting of only thermal and one representative group of fast, or epithermal, neutrons. Fermi age, τ, is a measure of how far fast neutrons travel before being thermalized. B^2 is called buckling and depends on the shape and size of the core. Small cores have larger buckling than large cores.

As the temperature of the core increases, L_f decreases because of the increase in the numerical value of τ from the decreasing density of the water. The change during core life is almost insignificant because it is primarily due to a change in metal-to-water ratio, which is constant with core age.

2.1.5.3 Resonance Escape Probability

The resonance escape probability, symbolized by p, is the probability that a neutron will be slowed to thermal energy and will escape resonance capture. It is also the fraction of neutrons that escape capture during the slowing-down process. It is always less than 1.0 when there is any amount of U-238 or Pu-240 present in the core, which means that high-energy capture by these isotopes always removes some of the neutrons from the neutron life cycle.

As the reactor temperature increases, the resonance escape probability decreases in value because of the decrease in the ratio of the water-moderating atoms to fuel atoms and the broadening of the resonance capture cross sections. The resonance escape probability increases with core lifetime due to the decrease in fuel temperature. These changes in the resonance escape probability will be discussed further in Section 2.1.6.1, "Fuel Temperature Coefficient".

2.1.5.4 Thermal Nonleakage Factor

The thermal nonleakage factor, L_t, is the fraction of the thermal neutrons that do not leak out of the core during thermal diffusion but remain to contribute to the chain reaction. L_t is also the probability that a thermal neutron will remain and be utilized in the core. It is a calculated value for each condition of the core and is represented by the equation:

$$L_t = \frac{1}{1 + L^2 B^2}$$

where L^2 is the thermal diffusion length squared and B^2 is the geometric buckling of the system. The value of L_t decreases as the temperature of the core increases; the effect can be seen from the value of L^2, which is a measure of how far thermal neutrons travel before absorption. When temperature is increased, the values of all absorption cross sections decrease. This increases L^2, which in turn decreases L_t. Buckling [B^2] does not change over the range of interest. As the core is operated and fuel is consumed, the value of L_t decreases as a result of fuel burnup.

2.1.5.5 Thermal Utilization Factor

The thermal utilization factor, f, is the ratio of the probability that a neutron will be absorbed in the fuel to the probability that the neutron will be absorbed in all the material that makes up the core. This factor is the one that the plant operator has the greatest control over. It is described by the following equation:

$$f = \frac{\Sigma_a(fuel)}{\Sigma_a(fuel) + \Sigma_a(other)}$$

where Σ_a= macroscopic absorption cross section, which is the sum of the capture cross section, Σ_c, and the fission cross section, Σ_f.

$$\Sigma_a = \Sigma_c + \Sigma_f$$

An examination of the thermal utilization factor shows that the Σ_a (fuel) comprises only the absorption by the U-235 at the beginning of core life. As the amount of Pu-239 increases because of the irradiation of U-238 in the core, it is necessary to consider the change of fuel concentration in determining the value of f at different times in the core lifetime. The reactor operator can change Σ_aã(other) by positioning of the control rods and by addition or removal of boric acid from the moderator.

2.1.5.6 Neutron Production Factor

The neutron production factor, η, is the average number of neutrons produced per thermal neutron absorbed in the fuel. It is based on physical measurement for each type of fuel used in a reactor.

The numerical value of η does not change with core temperature over the range considered for most reactors. There is essentially no change in η over the lifetime of the reactor core because the values for U-235 and Pu-239 are very close. As the reactor operates for a period of time, and Pu-239 begins to contribute to the neutron economy of the core, the average effect of η is expressed by:

$$\eta = \frac{(\nu\Sigma_f)^{235} + (\nu\Sigma_f)^{238} + (\nu\Sigma_f)^{239}}{\Sigma_a^{235} + \Sigma_a^{238} + \Sigma_a^{239}}$$

where ν is the number of neutrons per fission.

2.1.6 Reactivity and Reactivity Coefficients

In reactor physics, it is more convenient to use a term called reactivity rather than K_{eff} to describe the state of the reactor core. Reactivity (ρ or ΔK/K) is defined in terms of K_{eff} by the following equation:

$$\rho = \frac{K_{eff}-1}{K_{eff}}$$

Based on this equation, when K_{eff} is equal to 1.0, the reactivity of the core is zero, and the reactor is said to be critical. If K_{eff} is less than 1.0, reactivity of the core is negative, and the reactor is subcritical. Values of K_{eff} greater than 1.0 make the reactivity positive, and the reactor would be supercritical.

Mathematically, reactivity is a dimensionless quantity, so a label is used to identify the quantity known as reactivity. The most common labels are % ΔK/K and pcm. A reactivity of .01 could be expressed as 1 % ΔK/K or 1000 pcm.

As any operating condition of moderator or fuel (temperature, pressure, or voids) changes, the reactivity of the core changes accordingly. It is difficult to change any operating parameter and not affect every other property of the core. Once a change has been made to the core, it is necessary to make some compensating change to maintain criticality at the same power. For instance, assume the reactor is critical at 50% power and the reactor operator wants to increase power to 75%. The operator must first bring the reactor supercritical by, making a positive reactivity change (control rod withdrawal or boron dilution) to initiate the power increase. As the power increases the core parameters such as moderator temperature and fuel temperature change, causing a negative reactivity effect. If this offsets the original positive reactivity change, then the reactor will return to the critical condition and power will stabilize at the new value. These compensating changes are usually made by manual devices or means, like control rods. There are also inherent features in a reactor system that automatically change upon a change of some other feature of the core. These are generally described by reactivity coefficients.

A reactivity coefficient is defined as the rate of change of reactivity with respect to a change in some operating parameter of the reactor. The coefficients that are significant in the reactor are the fuel temperature (Doppler) coefficient, moderator temperature coefficient, void coefficient, and pressure coefficient. In addition to these coefficients, one other coefficient is considered. It is called the power coefficient. The power coefficient is the combination of the first three coefficients. Most of the reactivity coefficients for the core are important to the safety of reactor operation because they act in a negative manner to oppose any power change in the nuclear system.

The response of the reactor core to plant conditions or operator adjustments during normal operation, as well as the response to abnormal or accidental transients, is evaluated by means of a detailed plant simulation. In these calculations, reactivity coefficients are required to couple the response of the core neutron multiplication to the variables which are set by conditions external to the core. Since the reactivity coefficients change during the life of the core, a range of coefficient values are established to ensure the correct response of the plant throughout life.

2.1.6.1 Fuel Temperature Coefficient

The fuel temperature coefficient, also called Doppler coefficient of reactivity, is the reactivity change that results from a change in the resonance cross section of the fuel due to a change of fuel temperature. It is brought about by the nature of some fuel materials, principally U-238 and Pu-240. Figure 2.1-4 is a plot of the total cross section vs. energy levels for the resonance region of U-238. The solid line represents the cross section of the fuel at cold temperatures (547°F). The dashed line represents the cross section of the fuel at hot temperatures (1500°F). As the temperature of the fuel increases, the resonance peaks decrease and the base of the peaks broaden. Although the total area under the curve stays the same, the number of neutrons in this region increases. This results in more neutrons being lost to the fission process through resonance absorption.

Figure 2.1-5 is a plot of Doppler coefficient vs. effective fuel temperature. The effective fuel temperature is a weighted average temperature. The temperature profile across the fuel pellet makes this temperature lower than the average temperature and closer to the temperature the neutrons will be exposed to when they interact with the fuel. The terms BOL and EOL represent beginning of life and end of life. For this discussion and all others that follow, this is equivalent to new unirradiated fuel (BOL) , and the fuel at the end of the first core cycle (EOL). Note that the EOL case is more negative than the BOL case for cooler fuel temperatures. This is caused by the buildup of plutonium-240 which has a higher cross section at lower temperatures. The difference at higher temperatures is minimal due to the broadening of the resonance peaks. Note that the Doppler coefficient is always negative.

In the event of an addition of a positive reactivity to the reactor core, the Doppler coefficient of reactivity would be the first and most important effect in controlling that addition. It is effective almost instantaneously, because an addition of positive reactivity causes the fission rate to increase and produce more heat in the fuel, which in turn causes an addition of negative reactivity. As quickly as the heating rate is increased, the Doppler coefficient becomes effective.

Since the effective fuel temperature can not be measured, the Doppler-only power coefficient used for reactor transients is a function of power. The Doppler coefficient used in the remainder of this text is defined as the change in core reactivity as the result of a change in the reactor power level. It is in units of $\Delta\rho/\Delta\%$power or change in reactivity per percent change in power. Figure 2.1-6 shows the Doppler-only power coefficient as a function of power. The results presented do not include any moderator coefficient even though the moderator temperature changes with power level.

The numerical value of the Doppler-only power coefficient changes with both reactor temperature and core lifetime. As fuel temperature is increased, the U-238 resonance peaks broaden. However, the rate of broadening diminishes at higher temperatures yielding a Doppler-only power coefficient with a smaller negative value at higher reactor power levels. Three important factors influence the Doppler-only power coefficient during core lifetime; these are: thermal conductivity of the gases in the gap between the fuel and the cladding, plutonium production, and fuel-clad gap reduction. Dealing with these factors one at a time will lead to the conclusion that the Doppler-only power coefficient is more negative at BOL than it is at EOL.

1. Initially the fuel rods are pressurized with helium gas which yields a given fuel-clad gap thermal conductivity coefficient, but as fission gases such as xenon and krypton are produced, they tend to pollute the helium gas causing a reduction in the fuel-clad gap thermal conductivity coefficient. The result of this is

an increased fuel temperature for any given power level, and the Doppler-only power coefficient becomes more negative as the core ages.

2. Plutonium buildup is the result of the conversion of U-238 to the plutonium isotopes. Pu-240 has a large thermal resonance for parasitic capture of neutrons. As Pu-240 builds in, the Doppler- only power coefficient becomes more negative as the core ages.

3. The fuel-clad gap reduction is the most dominant effect, outweighing the other two factors combined. The reduction of the gap between the fuel pellets and the cladding is a result of swelling of the fuel pellets and clad creep. Fuel pellet swelling occurs because fission gases cause the pellet to swell and crack. The pellet occupies a larger volume. At the same time, the cladding is distorted by outside pressure. These two effects result in direct fuel-clad contact. With direct contact, the overall thermal conductivity increases due to conductive heat transfer, which results in a lower effective fuel temperature for the same power level over core life. This causes the Doppler-only power coefficient to become less negative over core life. In the beginning of life case, the fuel temperature will rise approximately 1000°F for a 0-100% power change or 10°F/% power. For the end of life case, with direct fuel-clad contact the temperature rise will be approximately 800°F, or 8°F/%power.

As a result of the three combined effects, the Doppler-only power coefficient will be more negative early in core life, and become less negative throughout the rest of core life, resulting in the EOL value being slightly less negative than the BOL value. Consequently, it is less effective for controlling power increases at EOL.

Figure 2.1-7 is a plot of the Doppler-only power defect vs. power. For this discussion and all others that follow, "Defect" is defined as the total amount reactivity added to the core due to a change in power. The differences in the BOL and EOL values have been explained in the preceding paragraph. An operator would use this graph to determine the total amount of reactivity added to the core from Doppler only associated with a desired load change. For example, changing load from 80% to 100% at EOL would result in a Doppler-only reactivity defect of -225 pcm. This value is calculated as follows:

Final reactivity - Initial reactivity = Defect
{-1025pcm - (-800pcm)}= -225pcm

2.1.6.2 Moderator Temperature Co-efficient

The next important reactivity coefficient is the moderator temperature coefficient (MTC). MTC is defined as; "A change in reactivity per degree change in moderator temperature," and is expressed in pcm/°F.

When the moderator temperature increases, its density decreases, so fewer moderator molecules are available to slow fast and epithermal neutrons and bring them to thermal energy. Decreasing the density of the moderator has two effects. First, if the density of the moderator is less, it increases the leakage probability of the neutrons. Second, if the neutrons are not slowing, they stay at a higher energy for a longer period which increases the probability of non-fission capture of these neutrons. Both effects together cause a net decrease in the neutron population that adds negative reactivity to the core.

Moderator density changes are not linear. At high temperatures an increase in the moderator temperature causes a larger reduction in density than an identical increase at low moderator temperatures. This is illustrated by the zero (0) ppm curve (bottom curve) in Figure 2.1-8. Notice how the slope of this curve becomes more negative as the temperature of the moderator increases. This is a graphical representation of the non-linear density change of the moderator. This curve also displays the reactivity effect due to a 1°F change in the temperature of the moderator. As shown, if the moderator temperature changes from its present value (horizontal axis), a certain amount of reactivity, in pcm (vertical axis), will be added to the core. As an example, using the 0 ppm curve, if the moderator temperature is initially at 500°F and its temperature increased by 1°F; -17 pcm's worth of reactivity would be added to the core.

Commercial reactors sold by Westinghouse are designed with a water volume to fuel volume ratio so the value of MTC is negative. A negative MTC plus the negative fuel temperature coefficient ensures stable power operations of the reactor core. As explained in the previous paragraph, if the temperature of the moderator is increased, negative reactivity is added to the core. This negative reactivity causes reactor power to decrease preventing any further increase in temperature or power. Therefore, a condition that could cause an increase in reactor power is limited by the addition of negative reactivity added because of the increase in the moderator temperature.

Operating with a negative MTC is desired due to the negative feedback mechanism discussed above and to its favorable operational characteristics during power changes. When the operator increases the load on the turbine, governor valves supplying steam to the high pressure turbine open. This action increases the steam demand causing steam pressure, saturation temperature in the steam generators, and the moderator temperature to decrease. Reducing the temperature of the moderator adds positive reactivity to the core causing reactor power to increase.

If the secondary load is reduced by the operator, the governor valves on the turbine close which reduces the steam demand of the turbine. The reduced steam demand causes the following to increase; steam pressure, saturation temperature and the moderator temperature. Increasing the temperature of the moderator adds negative reactivity which reduces reactor power. Therefore, the two above examples display how reactor power follows secondary steam demand due to the negative reactivity feedback caused by the MTC.

When the reactor is designed, excess reactivity in the form of fuel is added to the core so it can operate at 100% power for an extended period. With excess fuel added to the reactor, it could be taken critical and power escalated with the rods greatly inserted into the core. However, various operating limits require the rods to be fully withdrawn from the core. To get all rods out (ARO) of the core, an additional poison must be added to the core. This poison addition is accomplished by adding soluble boron to the reactor coolant. By adding boron to the reactor coolant, negative reactivity is inserted uniformly throughout the core. Since boric acid is dissolved in the coolant, the excess reactivity in the core can be controlled by boron changes without creating flux distribution problems associated with the control rods.

Let us now explore what happens to MTC when the moderator contains both light water and boron. Boron is a poison. When it is added to the reactor coolant (moderator), it adds negative reactivity to the core in a similar fashion as the control rods. By adding a certain amount of boron to the reactor coolant, the power of the reactor can be maintained at its desired values of power and temperature with the control rods fully withdrawn from the core.

Although it's true that adding boron to the moderator has a negative reactivity effect on the cores' reactivity, it has the opposite effect on MTC. As previously discussed, increasing the temperature of the moderator (0 ppm curve) has a negative reactivity effect. Now consider what happens when the temperature of the boron solution is increased.

During a temperature increase boric acid is expanded out of the core along with the moderator. Since boric acid is a poison, and it is expanding out of the core, positive reactivity is added. (The positive reactivity addition due to the expansion of boron out of the core offsets the negative reactivity addition due to the expansion of the moderator out of the core). As an example, using the 500 ppm curve on Figure 2.1-8, it is shown that for an initial temperature of 500°F that a 1°F increase adds -8 pcm worth of reactivity.

As explained earlier, -17 pcm was added to the core due to the density decrease of the moderator. For the conditions in the above example the same -17 pcm worth of reactivity was added to core via the moderator expansion out of the core, but due to the expansion of boron out of the core the net effect was -8 pcm.

The amount of positive reactivity added to the core depends upon the initial concentration of boron in the moderator. With high concentrations of boric acid, the positive effect could be large enough to overcome the negative effect of the moderator loss. In other words, a net positive reactivity addition could result from a moderator temperature increase. At boric acid concentrations greater than approximately 1400 ppm, MTC is positive.

Once all Westinghouse designed, nuclear power plants were required by technical specifications to operate with a negative MTC. This effectively limited the boric acid concentration of the reactor coolant. To reduce the dissolved poison requirement for control of excess reactivity, burnable poisons were incorporated in the core design as discussed in Chapter 3.1. However, today, utilities are demanding the vendors to design cores that can operate from 18 to 24 months at full power. To operate this long, more fuel and burnable poisons must be added to the core. In addition, the boron concentration also had to increase. The increase in boron has been of such an amount that the MTC has become positive. This resulted in technical specification changes that allow operation of the plant with a positive MTC.

Operating with a positive MTC makes the operating characteristics of the core less stable. Ignoring the effect of the fuel temperature coefficient, consider what would happen if MTC is positive. With a positive MTC, a 1°F increase in moderator temperature will add positive reactivity to the core causing reactor power to increase. As reactor power increases, more energy is added to the moderator. As the moderator temperature increases, more positive reactivity is added to the core, further increasing power that raises moderator temperature, etc. In this discussion reactor power would be self-escalating. Fortunately the fuel temperature coefficient, which is always negative and stronger than MTC, cannot be ignored. Therefore the scenario described above, reactor power self-escalating, will not occur.

2.1.6.3 Void Coefficient

The next coefficient of reactivity that acts to reduce the results of a reactivity insertion is void coefficient. It is defined as the change in reactivity as the result of boiling of the moderator in the core region and has the units of $\Delta\rho/\Delta\%$void.

The formation of voids in the core has the same effect as the temperature increase of the moderator (decreasing the density of the moderator.) Therefore, the moderating ability of the core is reduced. If the moderator is borated, the decrease in moderator density also reduces the boron concentration. In general, void formation could have a positive or negative effect on reactivity depending on both boron concentration and the departure from the optimum fuel-to-moderator ratio. The void content of the core is about one-half of one percent. The void coefficient varies from -30 pcm/%void at BOL and at low temperatures to -250 pcm/%void at EOL and operating temperatures.

2.1.6.4 Pressure Coefficient

The pressure coefficient is a change in reactivity caused by a change in pressure on the primary system. It is effected by the same mechanism that changes the moderator temperature and void coefficients. As pressure increases, the only change that takes place is in the density of the cooling and moderating water. The pressure coefficient is given in units of $\Delta\rho/\Delta P$. The pressure coefficient of reactivity has a slight positive effect on reactivity as the pressure of the system is increased if boron concentration is very low or absent. At high boron concentrations, an increase in pressure produces a slightly negative effect on reactivity.

2.1.6.5 Power Coefficient and Power Defect

The power coefficient combines the Doppler, moderator temperature, and void coefficients. It is expressed as a change in reactivity per change in percent power, $\Delta\rho/\Delta\%$ power. It is negative at all times in core life but is more negative at the end of life primarily due to the change in the moderator temperature coefficient. Figure 2.1-9 shows the values of the power coefficient vs. percent power at BOL and EOL. This graph can be used to calculate the reactivity change associated with 1% incremental changes in power. Figure 2.1-10 is the integrated power coefficient, or power defect, vs percent power at BOL and EOL. This graph is more useful to the operator for calculating the total reactivity change for any power change.

Operators need to be concerned about the effect of the power defect. As power is increased the power defect has a negative reactivity effect to the core. Therefore an equal amount of positive reactivity must be added to keep the reactor critical or near critical. This positive reactivity will be in the form of rod withdrawal or boron dilution. When power is decreased quickly, as it will after a trip, power defect is a positive reactivity effect to the core, and the resulting rod insertion must be sufficient to make the reactor subcritical. An alternate method, boron addition, is not rapid enough to overcome this positive reactivity addition.

To insure that the rods can shut down the reactor, they must be maintained above a minimum rod height. Since the magnitude of the power defect is dependent on total power, the minimum rod height is also increased with increasing power.

2.1.7 Poisons

In a reactor system, poisons absorb neutrons without causing fission. These neutrons are no longer in the neutron life cycle. Control rods and soluble boron are examples of poisons which are controllable. Xenon and samarium are examples of poisons which are uncontrollable.

2.1.7.1 Uncontrollable Poisons

In the fission process the nucleus absorbs a neutron and the resulting nucleus breaks into two or more parts called fission fragments. A detailed investigation of the thermal neutron fission of U-235 has shown that the resultant nucleus splits in many different ways, yielding more than 350 primary fission products (or fission fragments). In Figure 2.1-11, the percent fission yields of U-235 are plotted against the mass numbers. Note that each mass number represents several isotopes.

During power operation, the isotope xenon-135 (Xe-135) is formed as one of fission products. This isotope has an extremely large capture cross section (2.6 million barns). Xe-135 is formed in two ways; directly as a fission product (0.3% of the total fission products) and indirectly from the radioactive decay of tellurium-135 (5.9% of the total fission products). The radioactive decay chain of tellurium-135 is as follows:

$$\text{Te-135} \xrightarrow[19.2\text{ sec.}]{\beta} \text{I-135} \xrightarrow[6.6\text{ hr.}]{\beta} \text{Xe-135} \xrightarrow[9.1\text{ hr.}]{\beta} \text{Cs135 (stable)}$$

Xe-135 is lost in two ways; it can capture a thermal neutron, forming Xe-136, which is stable and has a very small capture cross section, or it can decay to Cs-135. When the reactor is first brought to power, the concentration of Xe-135 (atoms/cm^3) is slowly built up to an equilibrium value. This is due primarily to the relatively long half-lives of iodine-135 (6.6 hours) and Xe-135 (9.1 hours). Because of the high thermal neutron cross section of Xe-135, as the concentration of the isotope increases, so does the macroscopic absorption cross section of the core increase.

Operationally, as Xe-135 builds up, other poisons in the core (control material such as control rods or boric acid) must be removed to maintain criticality. Provided there is enough control material to remove during this xenon buildup, equilibrium will be reached after approximately 48 hours of power operation. After this 48 hour time period a condition is reached where the production of Xe-135 is equal to the removal of Xe-135 by neutron capture plus the loss of Xe-135 through radioactive decay.

The equilibrium value for the amount of xenon in the core at any time is a function of the reactor's neutron flux (power) Figure 2.1-12. Since a neutron absorber (poison) is added to the reactor when xenon is built into the system, its effect on the chain reaction can be described in terms of reactivity. The multiplication factor is lowered primarily through the decrease in thermal utilization with a secondary effect on the thermal nonleakage probability caused by decreasing the thermal diffusion length. The reactivity effect can be shown to be approximately equal to the ratio of the macroscopic absorption cross section of the xenon to that of the fuel.

The following tabulation (shown on the next page) lists the equilibrium values of xenon poisoning (negative reactivity) during reactor operation as a function of the average neutron flux level.

A change in power will cause a transient in xenon concentration. At the end of the transient, which will take about 2 days, the xenon concentration will reach its new equilibrium, assuming that power is left constant after the change. Three cases will be discussed; power increase, power decrease, and a shutdown.

Referring to Figure 2.1-13, following a xenon-free startup, a reactor has reached an equilibrium xenon concentration (time = 48 hrs.), and then power is increased. There is only a small increase in production of xenon because the production as a direct fission product is small (about .3%), and the iodine concentration cannot change very quickly, so the production by decay of iodine remains practically constant. Removal of xenon is increased significantly due to increased burnout (xenon absorbs neutrons). The decay of xenon to cesium remains almost constant initially.

These are only the initial effects, but it can be seen that the removal of xenon has increased more than the production, so xenon concentration decreases. This will continue for a few hours until the iodine concentration has increased enough that xenon production by iodine decay is greater than xenon removal. Eventually, the xenon concentration will reach a higher equilibrium value than that existing before the transient. This will happen when production and removal rates are again equal (time=96 hrs.).

Later when power is decreased (time=150 hrs.), the production is again changed very little initially, but removal by burnout is reduced and xenon concentration increases initially. As iodine concentration decreases, production of xenon will decrease until removal is greater than production. Then xenon concentration will decrease until a new lower equilibrium is reached (time= 200 hrs.). A shutdown is similar to a power decrease, but more severe (refer to Figure 2.1-14.)

Production of xenon directly from fission and removal by burnout almost cease. What is left is radioactive decay of iodine to produce xenon, and radioactive decay of xenon to remove it. Xenon concentration will increase initially, just as in the power decrease case. However, the increase will continue longer and reach much higher levels before decreasing. The peak occurs about 8 or 9 hours after shutdown. Since iodine is no longer being produced its concentration will continue to decrease as it decays to xenon. The xenon will be almost gone in about 3 days.

In some cases Xenon-135 presents moderately sensitive reactor operating conditions. Consider a reactor that has operated at power long enough to establish a high xenon concentration. Assume that the reactor trips and that a rapid recovery to full power is achieved simultaneously with the point in time at which "peak" xenon occurs. In returning the reactor to power, the negative reactivity associated with the xenon is overcome, as indicated above, by withdrawing the control rods to a higher position than usual. With the high flux of full power operation acting to "burnout" the accumulated Xe-135 and the decay of Xe-135 due to the shutdown, the concentration quickly falls off, and the result is a positive reactivity insertion. It should be pointed out that properly designed power reactors will overcome the xenon burnout quite adequately with proper control system operation.

Figure 2.1-15 shows reactor trips and return to power from various power levels. Note the drop off in xenon concentration during the restart. At EOL there is a possibility that there is not enough positive reactivity remaining in the fuel to overcome the negative reactivity added by xenon. In cases as this the reactor restart will have to be delayed until the xenon has decayed to lower values.

Another important fission product poison encountered in reactor operation is samarium-149 (Sm-149). This stable isotope has a capture cross section of about 4 x 10^4 barns. It enters the system as the end product of the following decay chain:

$$\text{Nd-149} \xrightarrow[\text{1.7 hr.}]{\beta} \text{Pm-149} \xrightarrow[\text{53 hr.}]{\beta} \text{Sm-149 (stable)}$$

(Nd: neodymium, Pm: promethium)

These products occur in about 2.1% of U-235 fissions. Because Sm-149 is stable, its equilibrium value in any reactor produces about -.65% $\Delta K/K$ or -650 pcm reactivity, independent of the neutron flux (power level) as shown in Figure 2.1-16. The rate at which Sm-149 approaches this equilibrium is related to the Pm-149 (53 hour) half-life. Upon reactor shutdown, Figure 2.1-17, the Sm-149 builds up to an asymptotic value dependent on the equilibrium level of promethium which is power level dependent.

2.1.7.2 Controllable Poisons

Two controllable conditions existing in a nuclear reactor greatly affect the reactivity balance within the system. These are control rod position and soluble poison concentration.

Control rods are made of a material or materials with an extremely high capture cross section for neutrons. Consider a just-critical reactor in which a control rod is partially inserted. If the rod is withdrawn slightly, K_{eff} will increase to a value slightly greater than 1.0. By withdrawing the control rod, neutron absorbing material is removed from the core. The effect is an increase in the thermal utilization factor, and (as seen in the six-factor formula discussed earlier) K_{eff} is increased. Control rod reactivity effects vary with their location in the core. If a control rod is located at the mid-plane of the core, its worth per step of movement is greater than if it were almost totally withdrawn or inserted.

A typical differential control rod worth ($\Delta\rho/\Delta$ rod position) curve is shown in Figure 2.1-20(a). Note that the maximum differential worth is found when the rod is positioned at about 40% withdrawn. This curve can be integrated to provide a determination of total control rod worth, which is defined as that amount of reactivity change associated with a given total movement of the control rod. Figure 2.1-20(b) shows the integrated control rod worth curve for the given differential rod worth.

Reactivity control is also maintained by using soluble boron poison in the moderator or coolant that passes through the reactor core. Addition of boron reduces the thermal utilization factor and requires that the control rods be further removed from the core to sustain the just-critical system. In most large pressurized water reactors today, large amounts of positive reactivity are held down by a soluble poison such as boric acid. Boron-10 is the principal absorber in this solution. When used as gross control, the plant can operate with almost all control rods fully removed. This works to optimize the power distribution and heat transfer through the core and at the same time enhances the available shutdown safety margin.

Soluble poisons are injected through the makeup portion of the chemical and volume control system. As reactor fuel is consumed and the installed excess reactivity drops, the soluble poison is removed through the purification system, compensating for the reactivity loss due to fuel depletion.

2.1.8 Reactor Kinetics

Consider a reactor in which K_{eff} is exactly 1.0. With equilibrium established, the neutrons entering into the self-sustaining reaction are a mixture of prompt and delayed neutrons. The fraction of this mixture that are delayed neutrons at steady state is known as β. Each fuel has a characteristic β. due to its yield of delayed neutron producers. A mixture of fuels, such as what would be present in a commercial reactor after it has operated, has a β which is a weighted average of the βs for the fuels present. For a reactor with only U-235, β is about .007 or .7%. Even though this is such a small fraction of the neutrons, delayed neutrons are essential to reactor control. Delayed neutrons do not appear until about 13 seconds after the fission that results in a delayed neutron producer. The long time decay is because the delayed neutron producer must decay before a neutron is released.

After the release of the delayed neutron, it will go through the same life cycle with the prompt neutrons, which were released immediately after the fission. The lifetime for a prompt neutron is about 10^{-4} seconds, but the lifetime for a delayed neutron is about 13 seconds ($13 + 10^{-4} \sim 13$). The life cycle time or generation time is a weighted average of the lifetimes for prompt and delayed neutrons. Since the delayed neutron fraction is β, then the prompt neutron fraction is $(1-\beta)$. These are used as the weighting factors. The generation time is about .1 seconds. The reactor would be uncontrollable with a generation time of 10^{-4} seconds but is controllable with .1 seconds as can be seen in the following discussion.

As previously stated , if the fuel is U-235 and the reactor is critical, .7% of the fission neutrons are delayed. That means that 99.3% are prompt. Assume now that an adjustment has been made so that K_{eff} is increased to 1.001 or 0.1% supercritical. The system is now multiplying at a given rate proportional to the number of neutrons present within the system. After the transients effects decay out, the rise in the neutron level can be described by a single exponential equation:

$$n(t) = n_0 e^{t/T}$$

where n_o = the neutron level at time zero
t = the time in seconds
T = the stable reactor period (the time it takes the neutron level to increase by a factor of e)

At the instant the reactor is made supercritical, the neutrons in the core are a mixture of 0.7% delayed and, in this instance about 99.4% prompt neutrons (99.3 + 0.1%).

If K_{eff} were increased to 1.007, then the reactor would be critical on prompt neutrons alone, no waiting for delayed neutrons would be necessary, and it would be uncontrollable by normal means. The effect of the delayed neutrons on generation time is lost. To put it another way, the power would be increasing at such a rapid rate that it would be impossible to control the reactor short of spontaneous shutdown on its own inherent mechanisms.

The relationship between reactivity and the reactor period can be expressed mathematically. The period of the reactor is useful in equations for reactor analysis. This term for reactor operation is called startup rate (SUR). The startup rate is related to the reactor period by the following equation:

$$SUR = 26/T$$

where SUR = startup rate in decades per minute
T = period in seconds.

Therefore, if the reactor power level increases by a factor of 10 every minute, then the reactor SUR is 1 decade per minute, which is equal to a period of 26 seconds. Figure 2.1-21 is a plot of reactivity addition versus reactor startup rate. If a reactor operator wished to establish a 1 dpm startup rate from a critical condition it would require an addition of 175 pcm of positive reactivity. The startup rate can also be used to predict power changes if the time the SUR will be in affect and the starting power level are known.

The power of the reactor can be expressed by:

$$p = p_0 10^{SUR\, t}$$

where: p_o = power at time zero
t = time in minutes

2.1.9 Subcritical Multiplication

There is a third category of neutrons that are not fission neutrons, but are very important during startup and shutdown operations. These are source neutrons and they come from sources other than neutron-induced fission. This includes spontaneous fission of certain nuclides and source assemblies loaded into the core to provide a sufficient count rate on source range nuclear instruments.

When the reactor is subcritical, the neutron population will decrease exponentially with a startup rate of -1/3 decades per minute, which is the result of the delay of the longest-lived delayed neutron producers. Without source neutrons, the decrease would continue indefinitely. But the neutron population eventually levels off because the source neutrons make up for the losses in the neutron life cycle. The source neutrons enter the life cycle and are affected, just as fission neutrons are. When equilibrium is reached, the count rate measured at the source range nuclear instruments is:

$$CR = \frac{S}{1-K_{eff}} \text{ for } K_{eff}<1$$

where S = source strength

The source strength includes a constant to account for the detector location. Note: This equation applies only to subcritical reactors.

The amount of time it takes to reach equilibrium increases as K_{eff} approaches 1. As criticality is approached, it takes longer for count rate to level off. The reason can be seen by expressing the count rate formula as a series.

$$CR = \frac{S}{1-K_{eff}} = S(1+K_{eff}+K_{eff}^2+\cdots)$$

Each term in the series represents a generation. Larger values of K_{eff} have more significant terms, and therefore it takes more generations to reach equilibrium.

One use of the count rate formula is to predict the point of criticality as fuel assemblies are added or reactivity is increased. However, because CR gets infinitely large as K_{eff} approaches 1, the inverse of CR is plotted. As criticality is approached, 1/CR approaches zero.

Table 2.1-1
Particles and Energy Produced per Fission Event

Instantaneous Energy Release	MeV/fission
*K.E. of fission fragments	165
Prompt γ energy	5
Capture γ energy	5
*K.E. of prompt neutrons	7
Total	182
Delayed Energy Release	
β energy from fission products	7
γ energy from fission products	8
Neutrinos	10
Total	25

Table 2.1-2
Neutrons per Fission

Isotope	Neutrons
U-233	2.51
U-235	2.43
U-238 (fast fission)	2.47
Pu-239	2.90
Pu-241	3.06

Table 2.1-3
Typical Neutron Balance for U-235

Function	Number	Factor
Initial thermal neutrons	1000	
Fission neutrons produced	2430	η
Fast fission	73	ε
Fast Leakage	-47	L_f
Resonance capture	-247	p
Thermal leakage	-44	L_τ
Captured in the moderator	-68	f
Fission product capture	-44	f
Captured by poisons	-201	f
Control rods	-38	f
Neutrons for conversion	-627	f
Neutrons lost in fuel capture	-187	f
Total neutrons produced	2503	η
Total neutrons lost in cycle	-1503	
Neutrons for new cycle	1000	

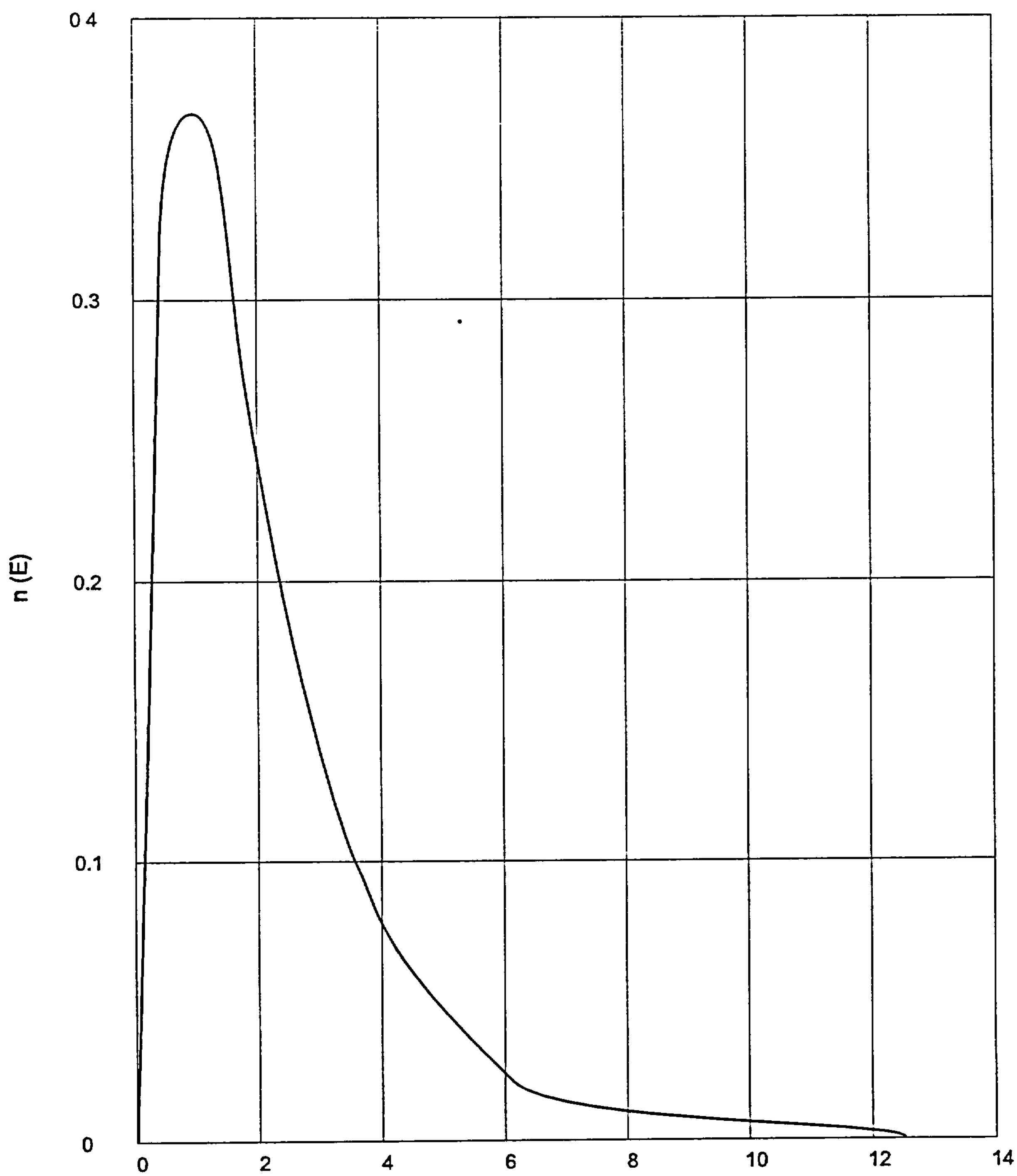


Figure 2.1-1 Fission Neutron Energy

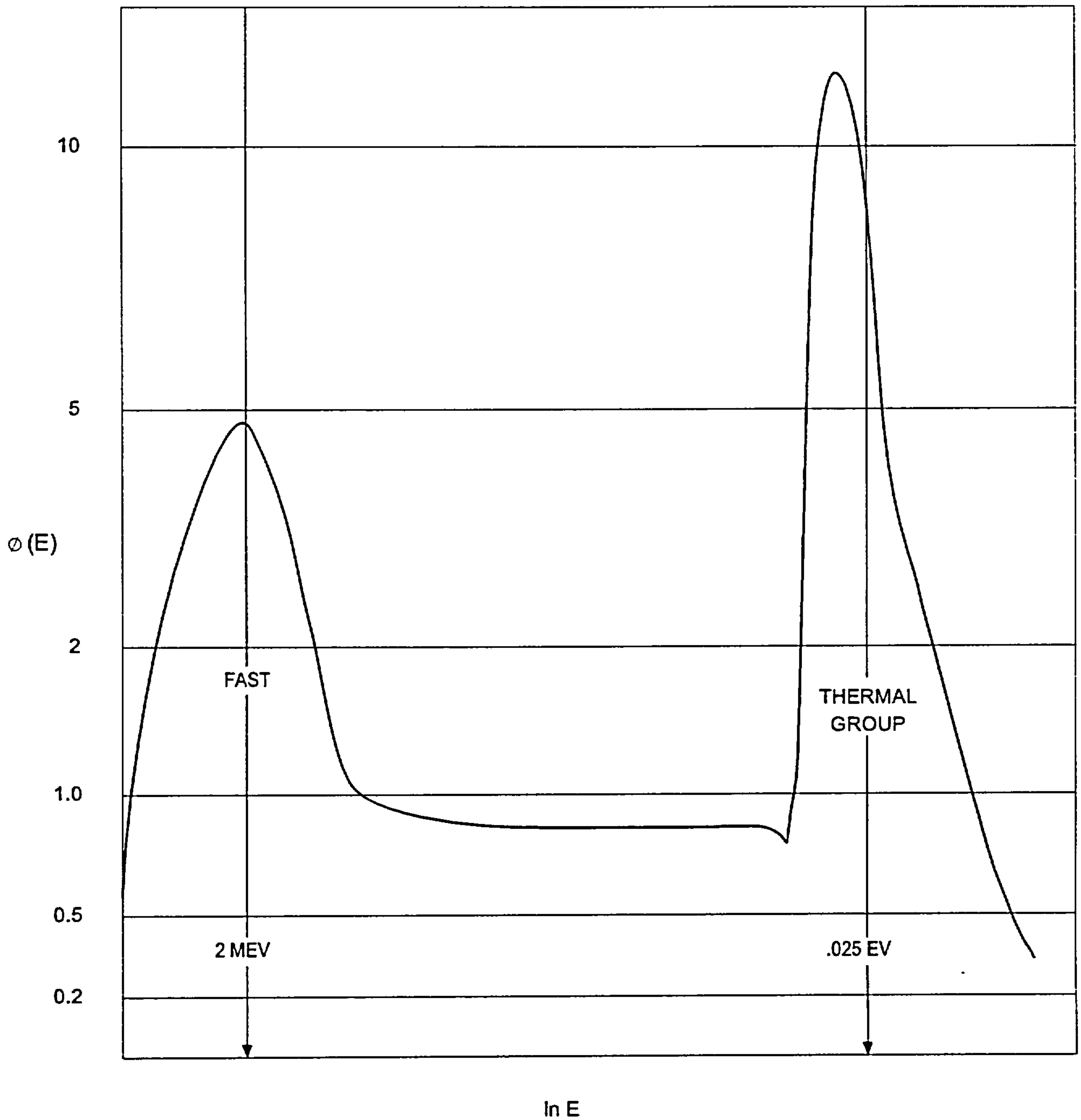


Figure 2.1-2 Flux Distribution

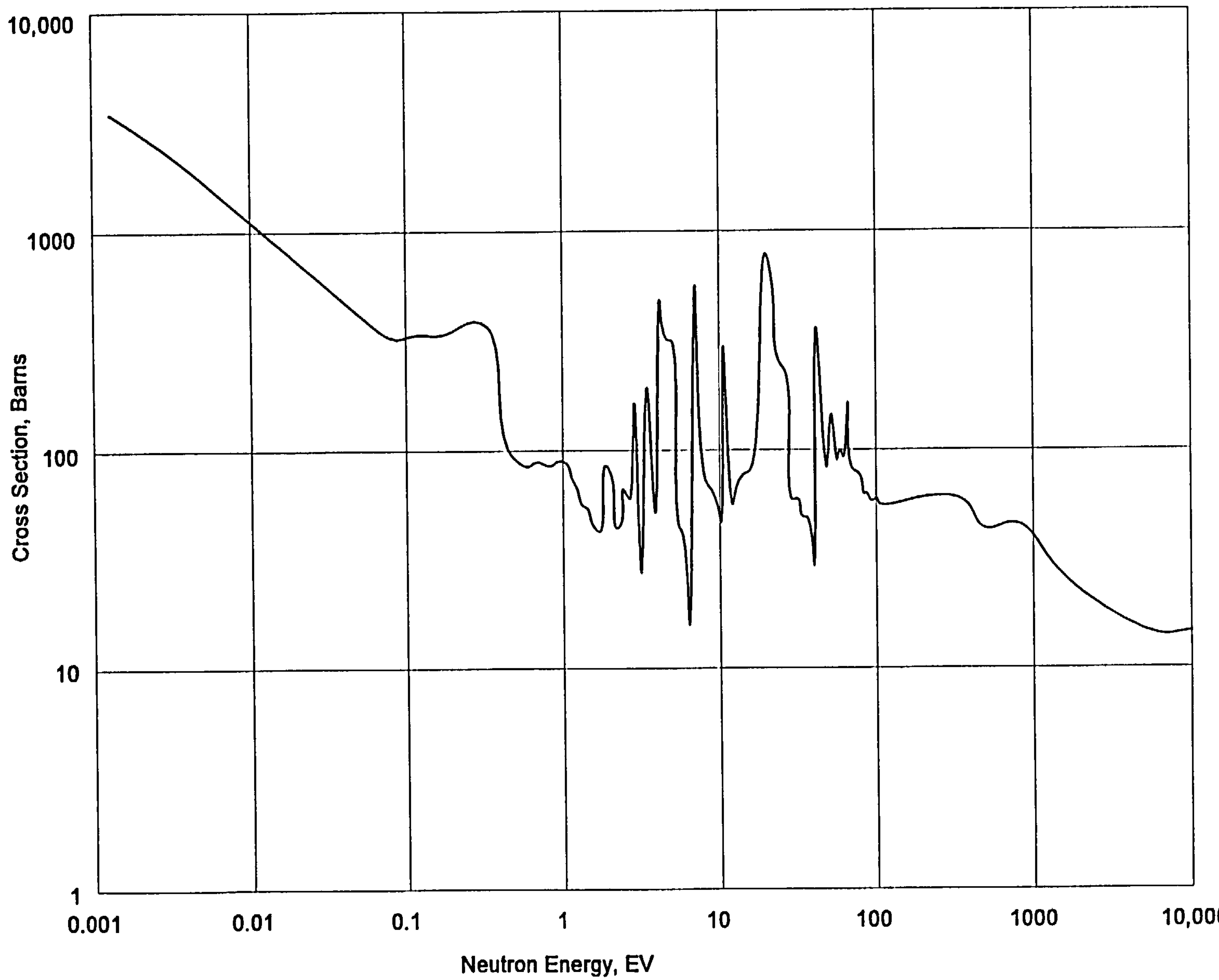


Figure 2.1-3 Total Cross Section for U-235

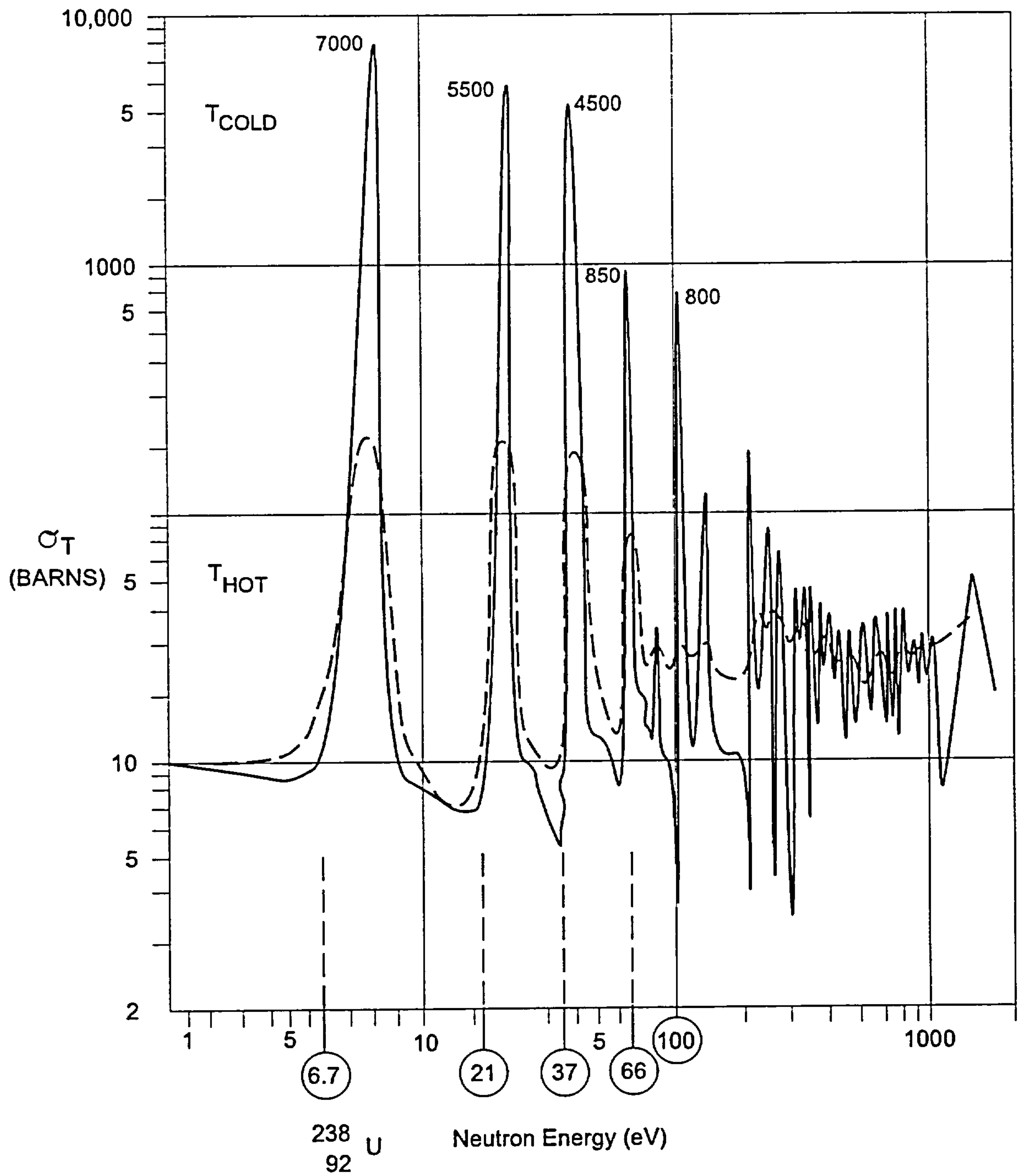


Figure 2.1-4 Cross Section Curve

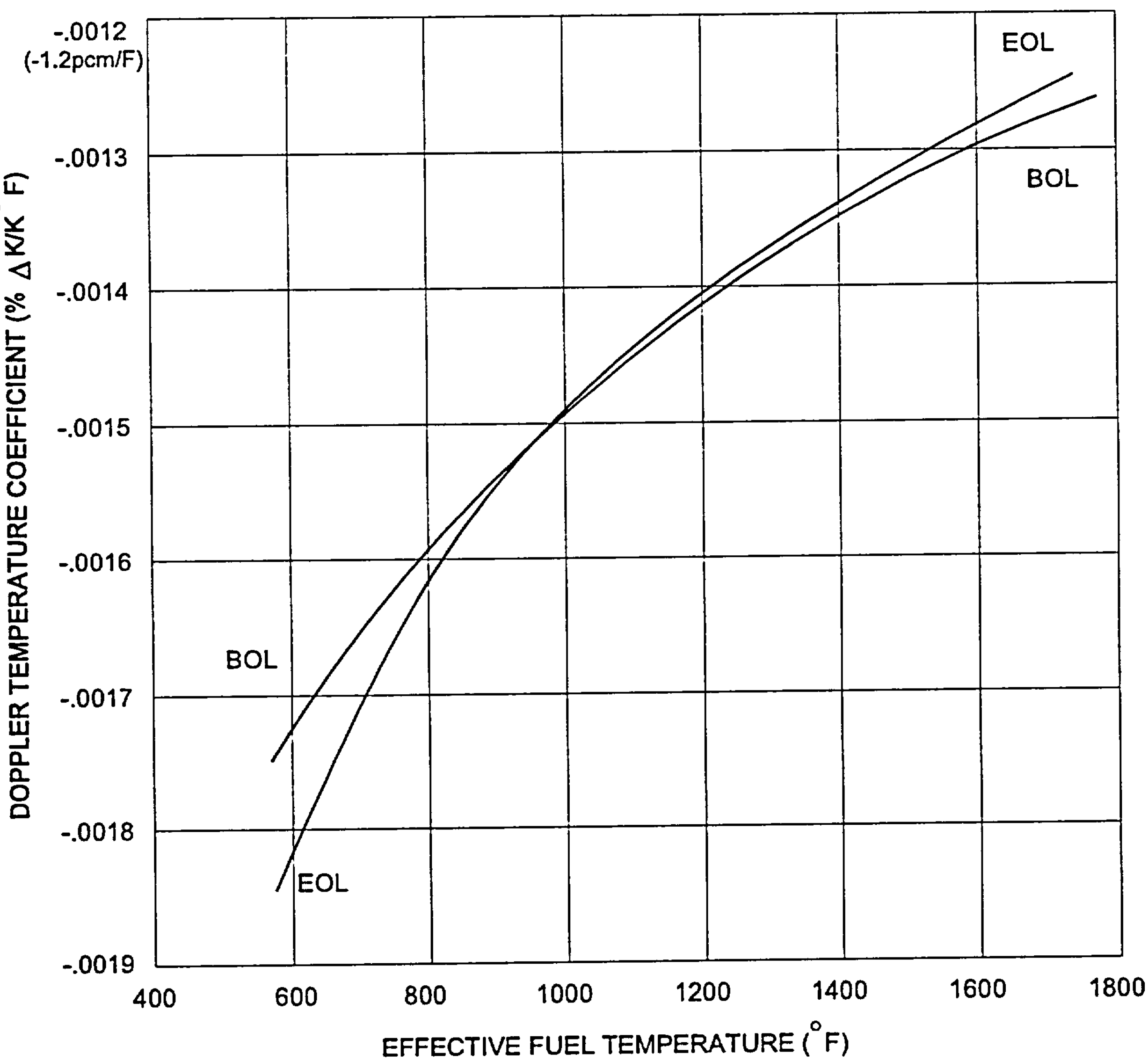


Figure 2.1-5 Doppler Temperature Coefficient, BOL and EOL

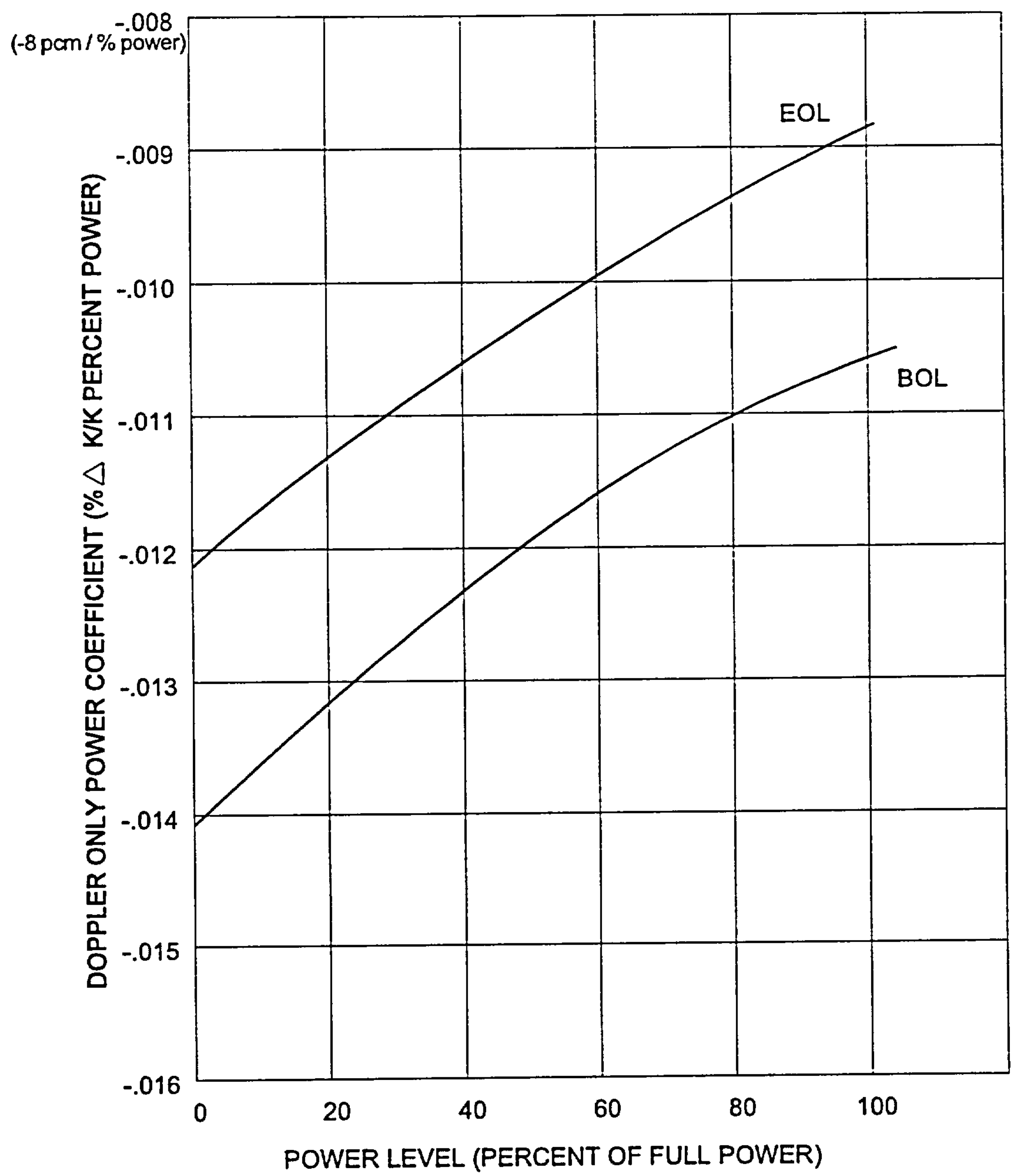


Figure 2.1-6 Doppler Only Power Coefficient, BOL and EOL

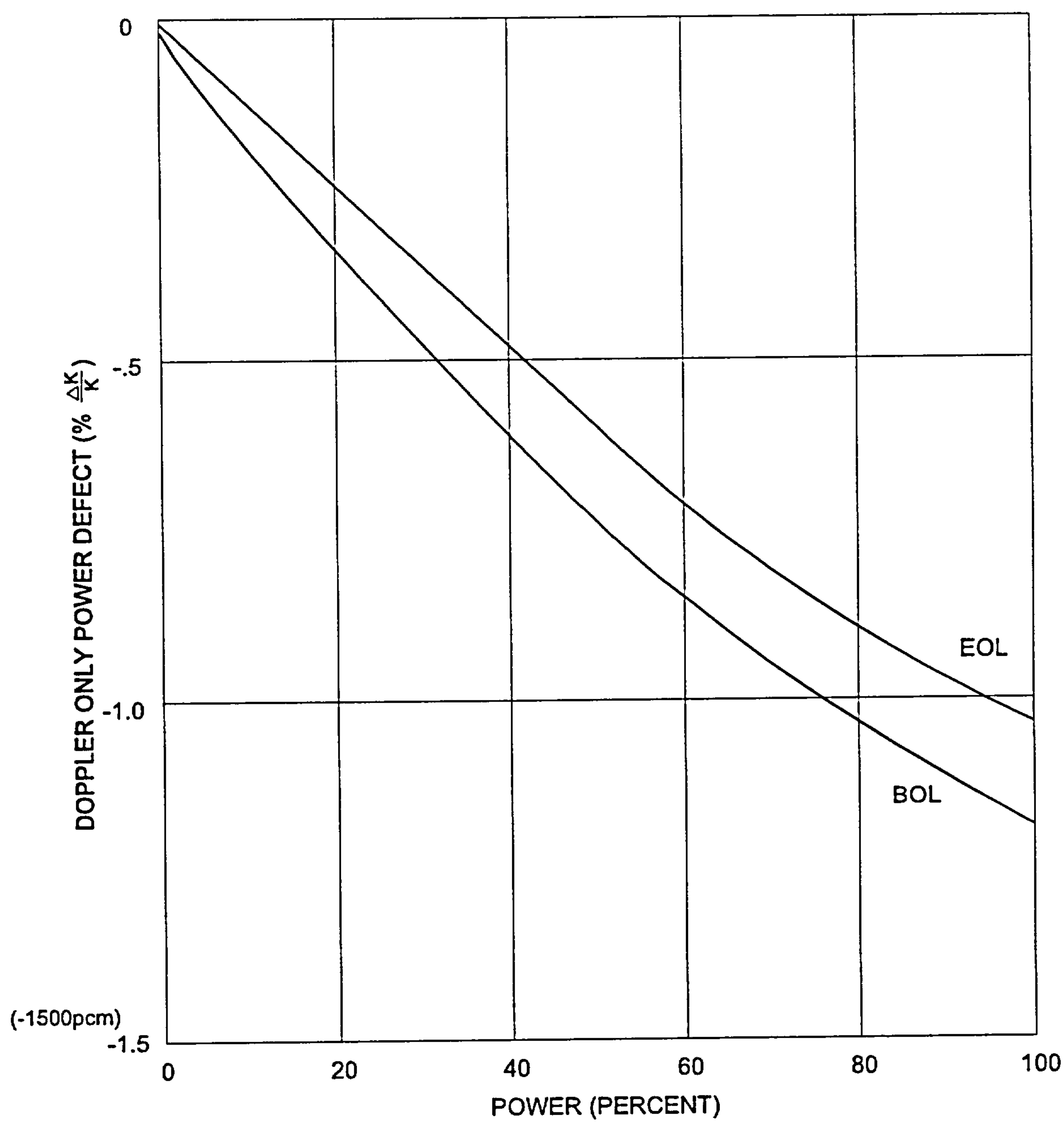


Figure 2.1-7 Doppler Only Power Defect, BOL and EOL

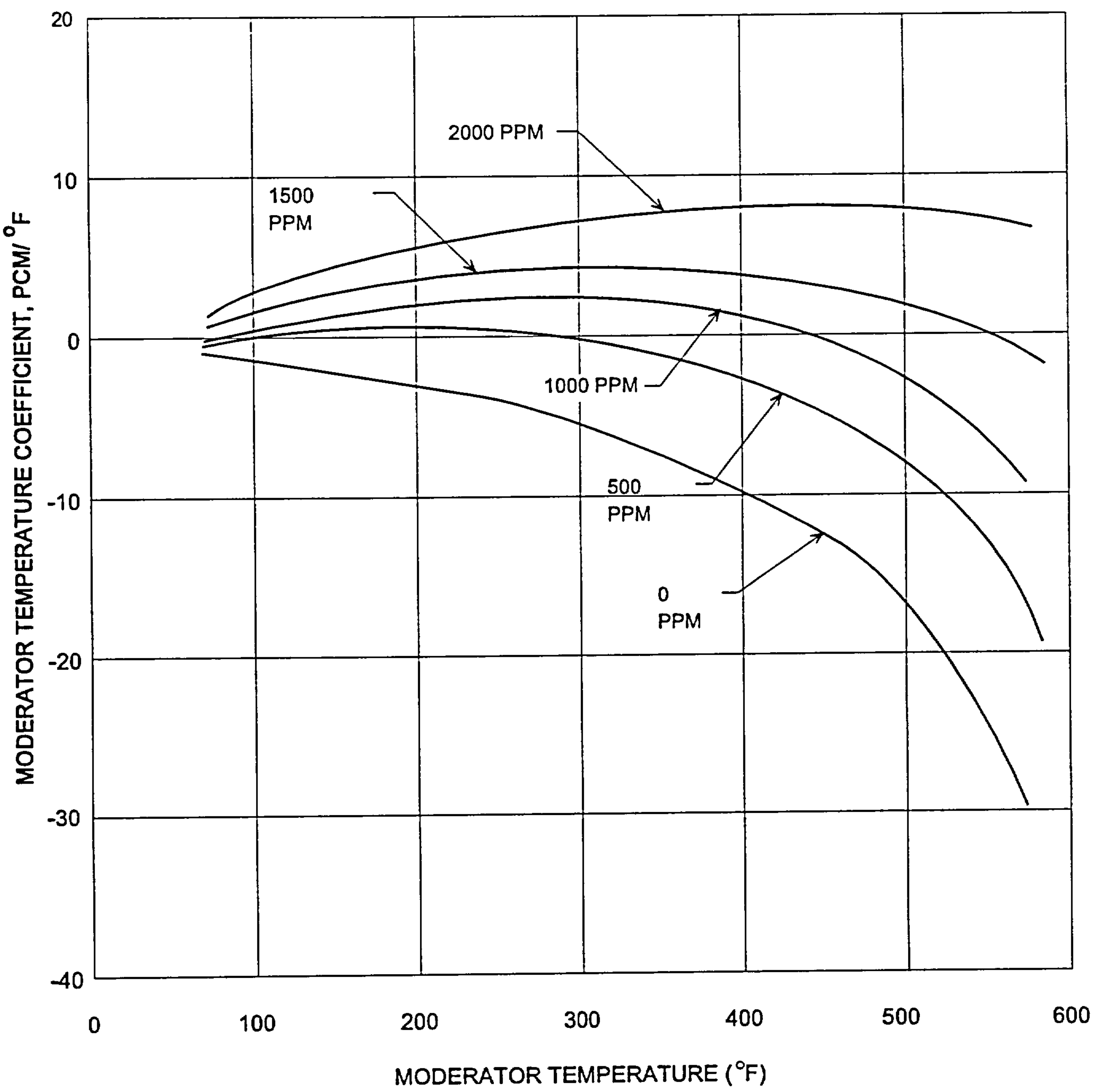


Figure 2.1-8 Moderator Temperature Coefficient

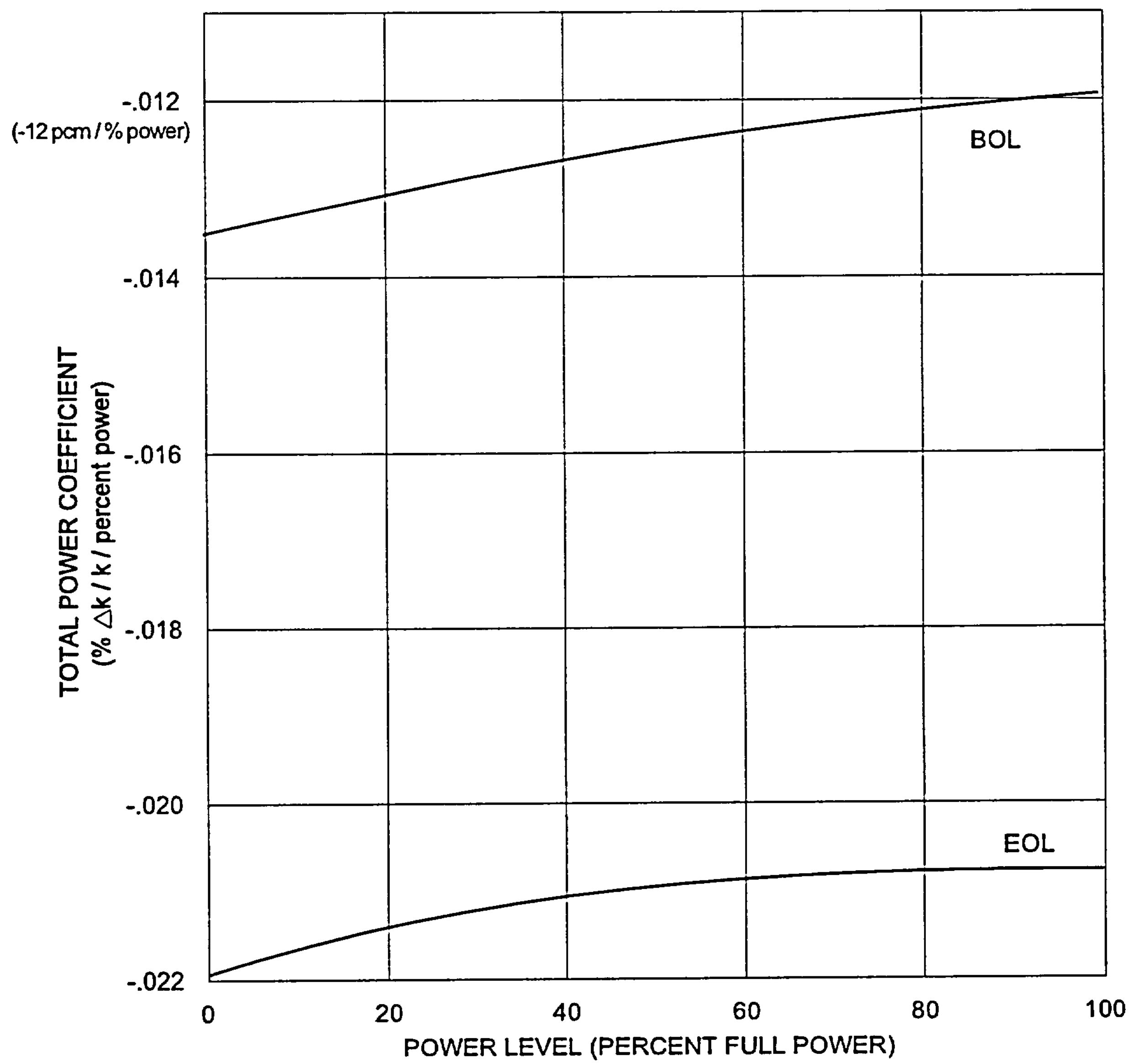


Figure 2.1-9 Total Power Coefficient, BOL and EOL

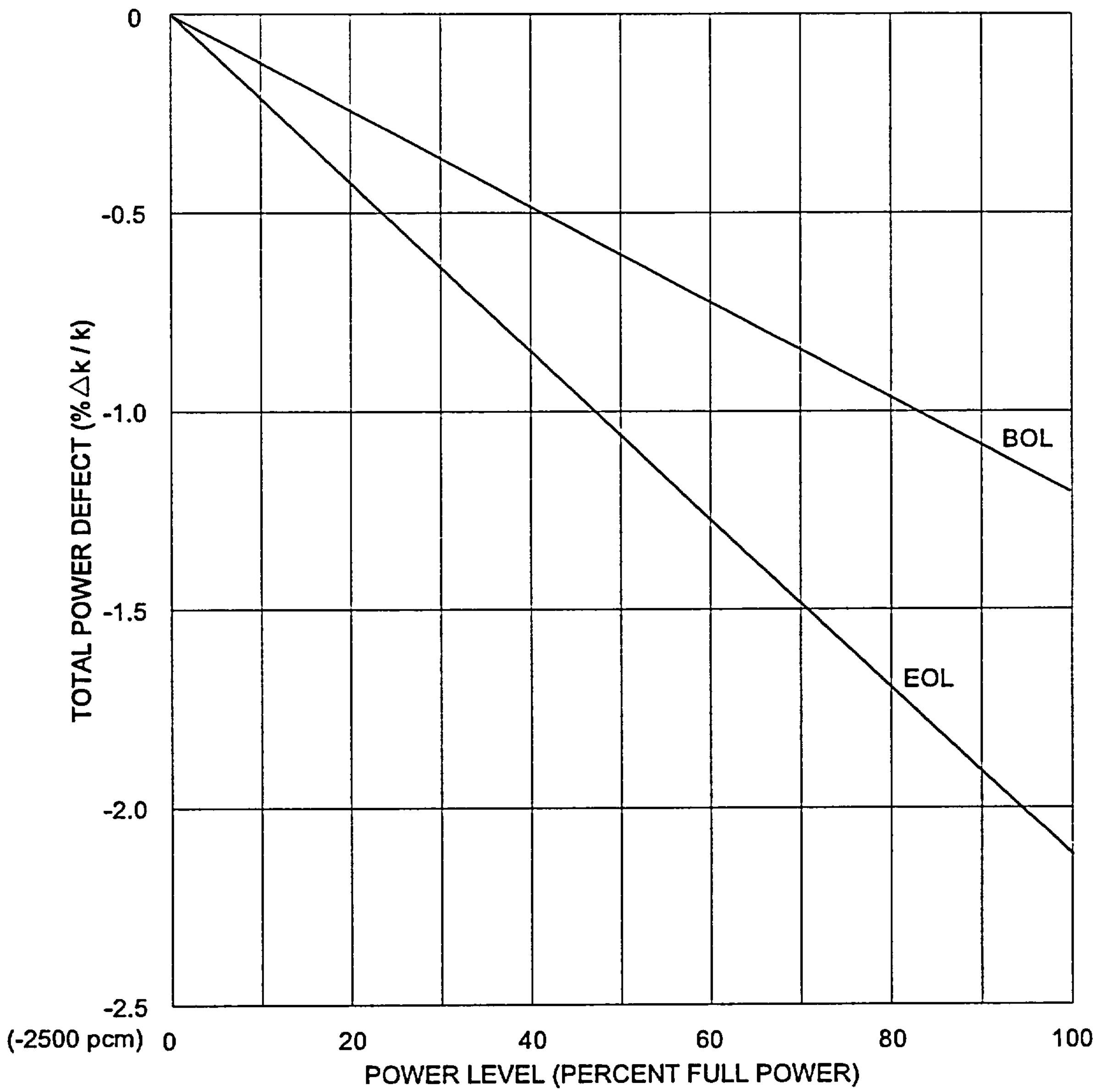


Figure 2.1-10 Total Power Defect, BOL and EOL

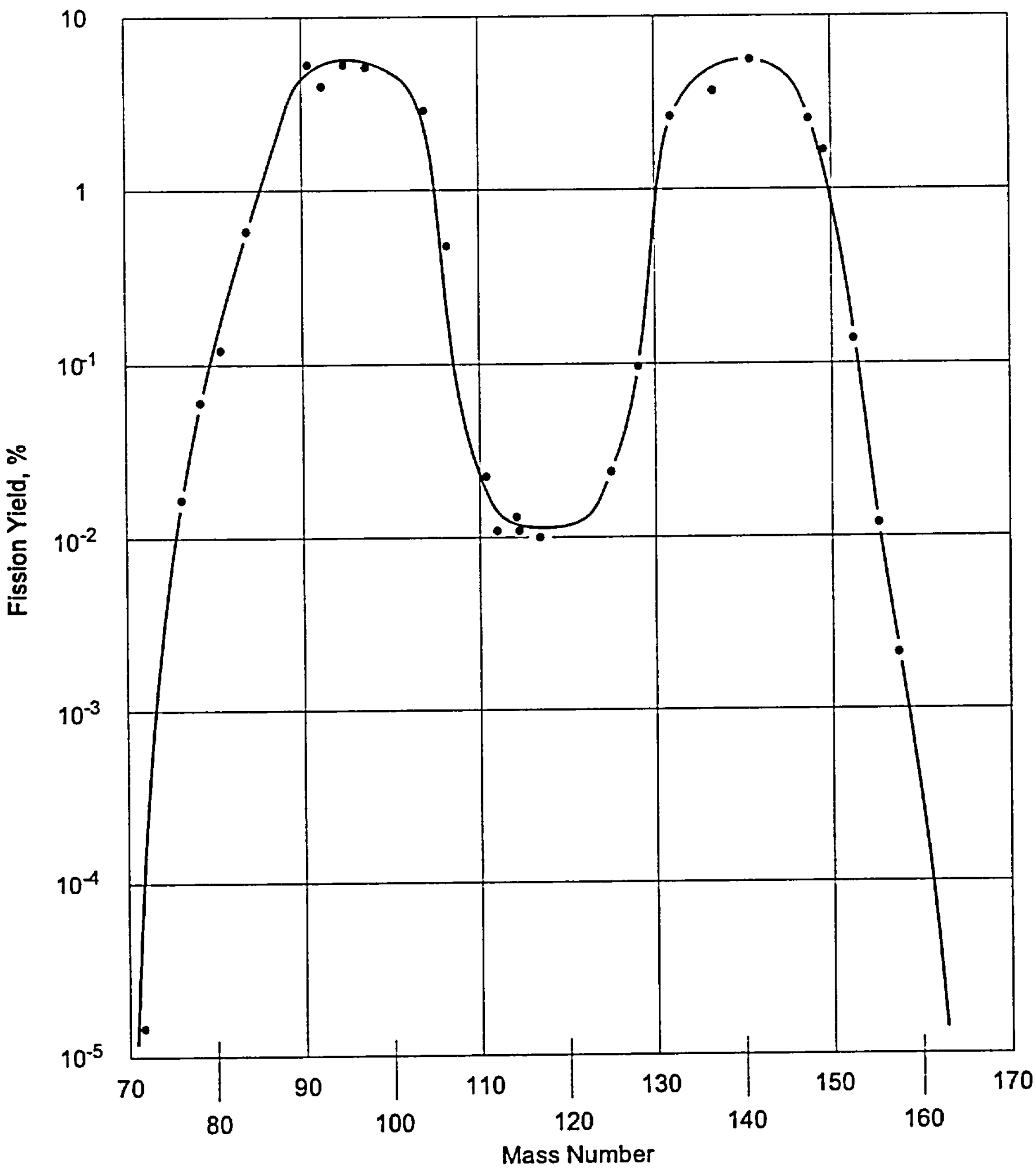


Figure 2.1-11 Fission Yield versus Mass Number

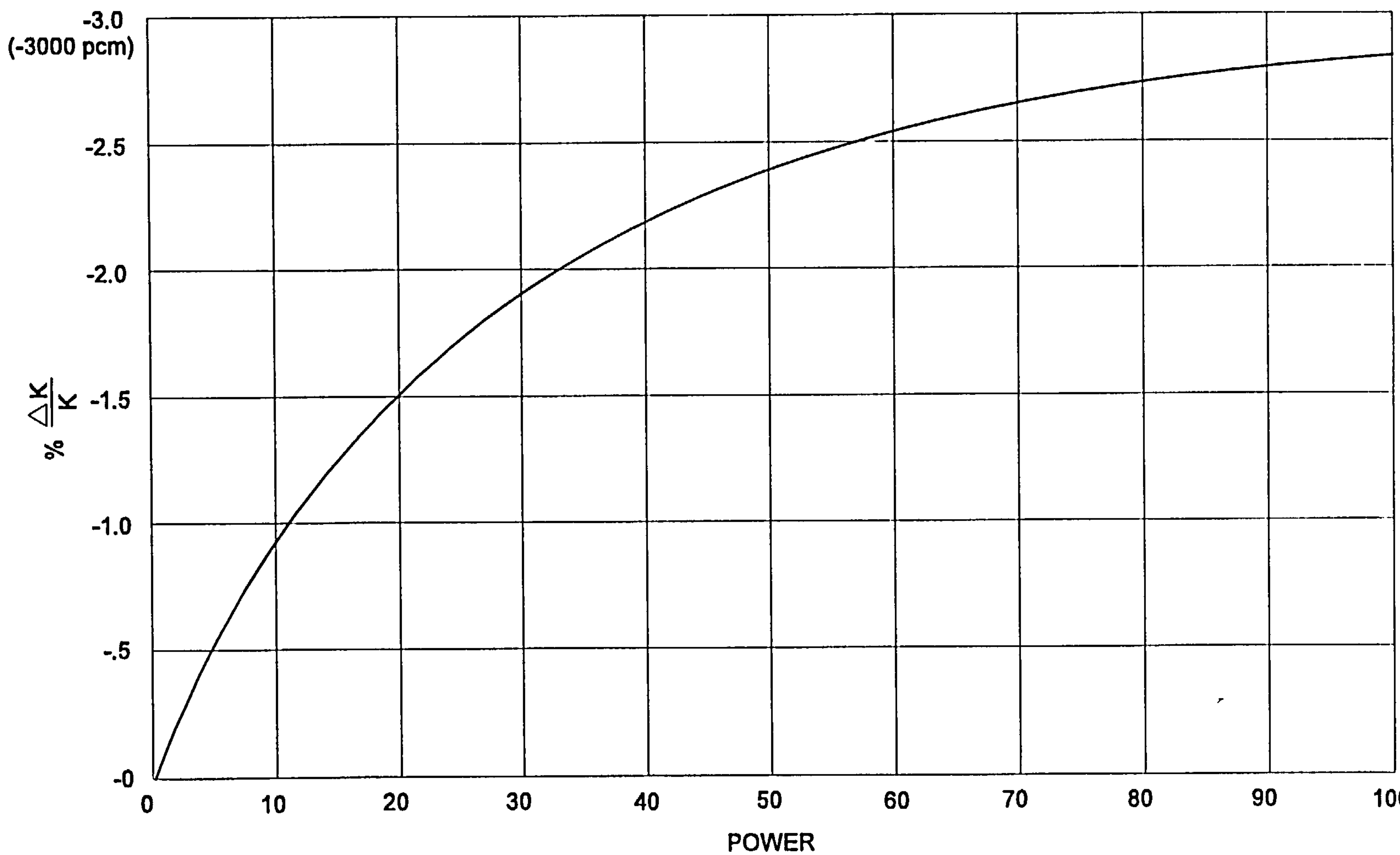


Figure 2.1-12 Equilibrium Xenon Worth vs Percent of Full Power

0199-3

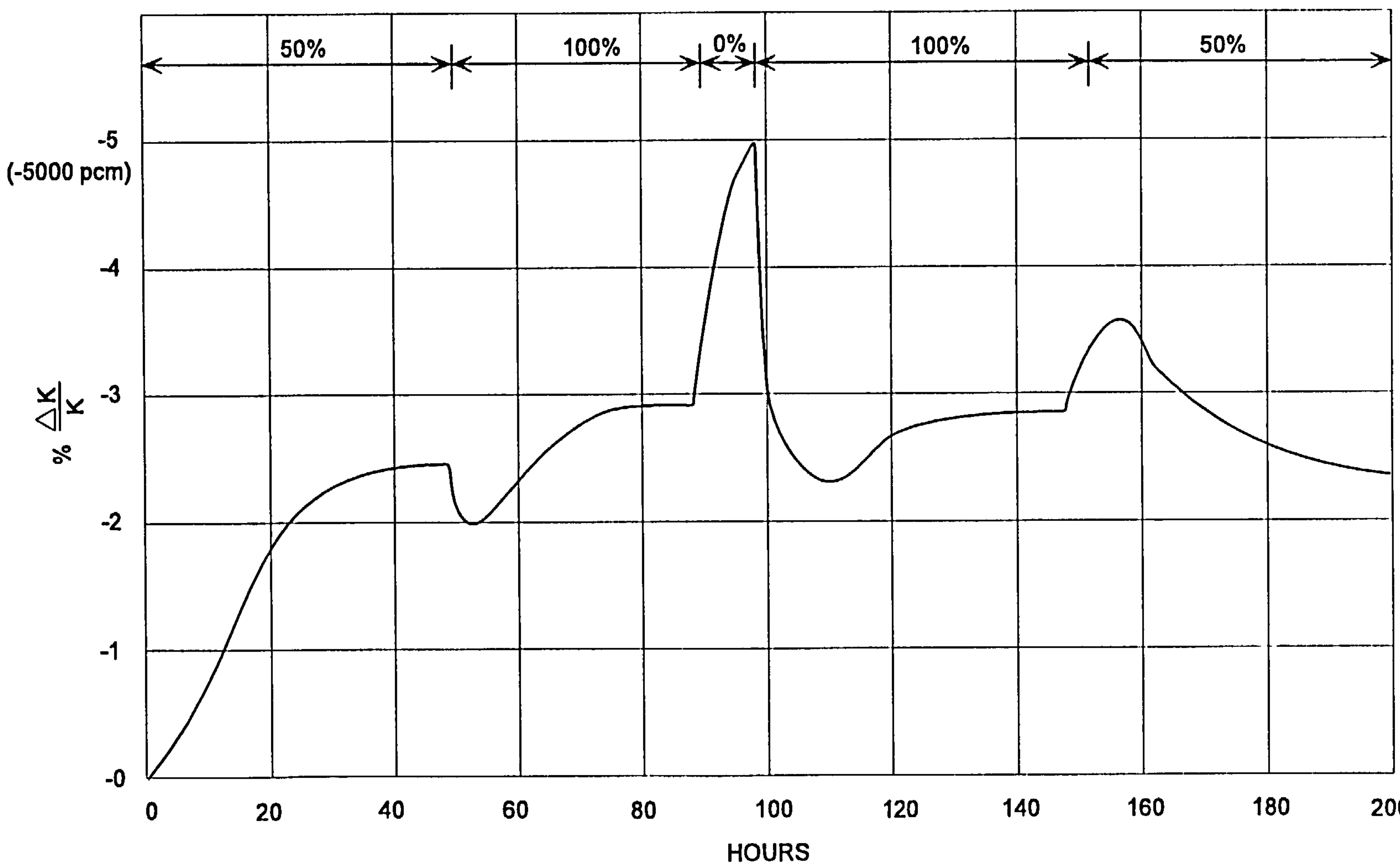


Figure 2.1-13 Xenon Transients

0199-3

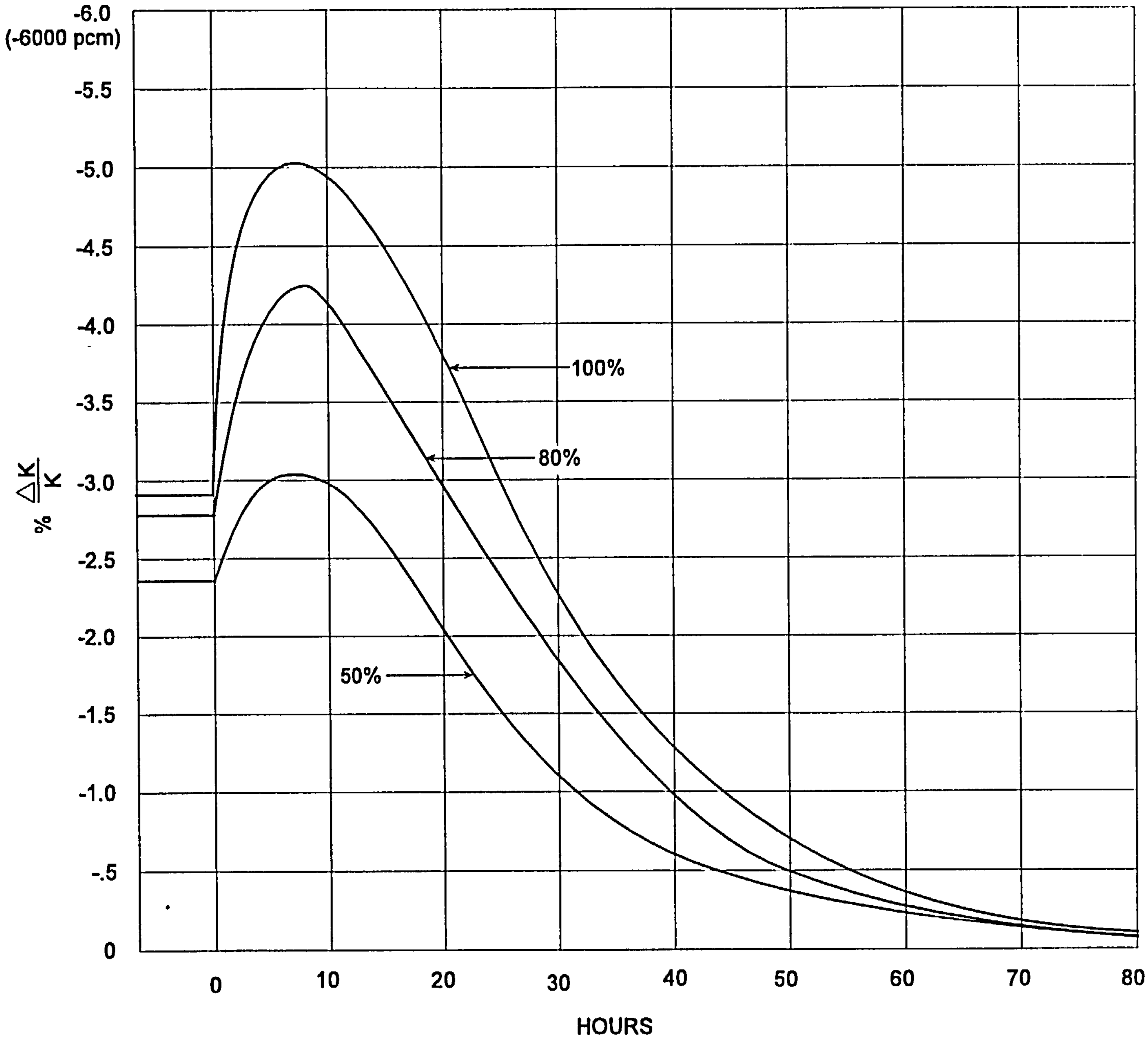


Figure 2.1-14 Xenon Transients Following a Reactor Trip

0199-3

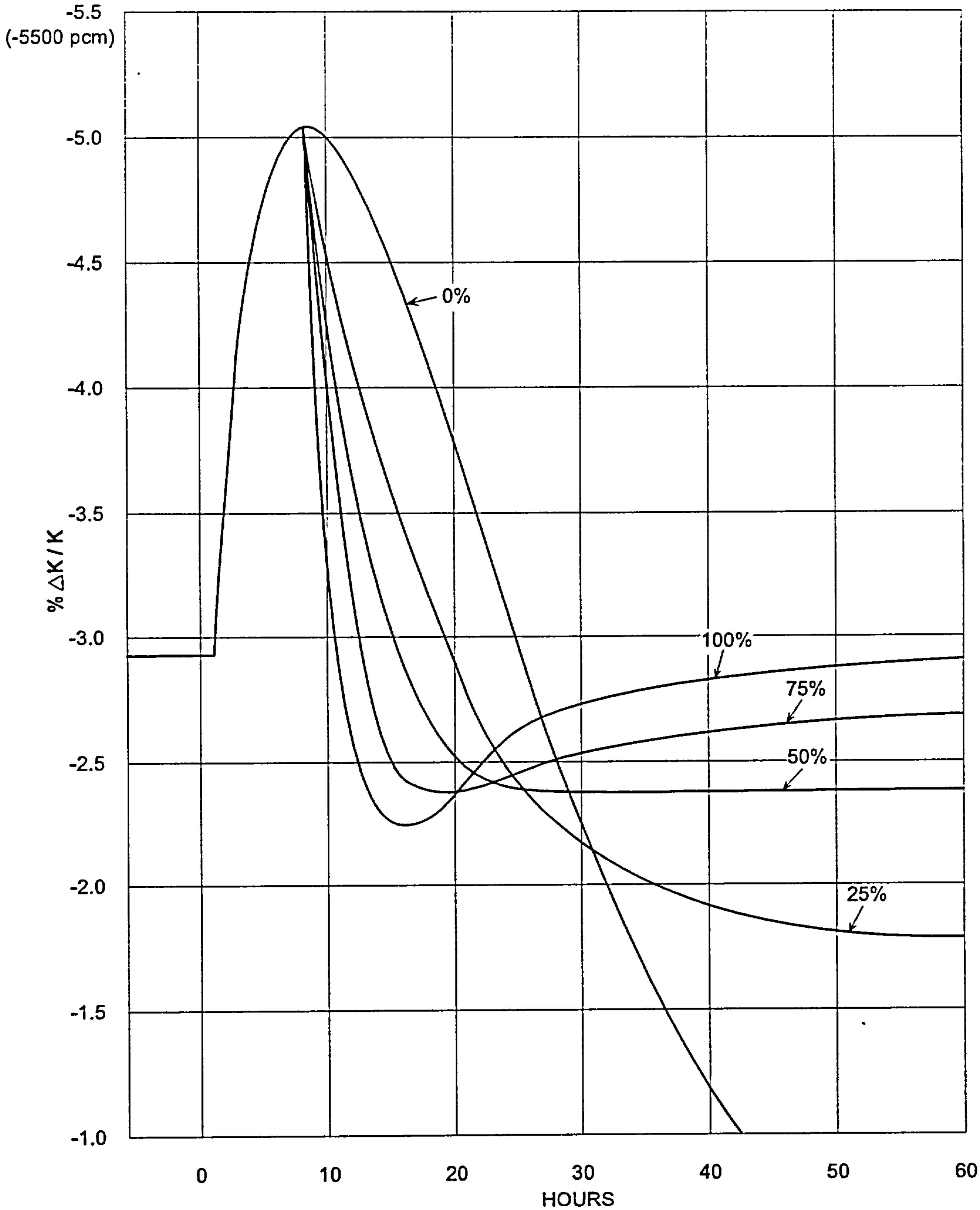


Figure 2.1-15 Xenon Transients Following a Reactor Trip and Return to Power

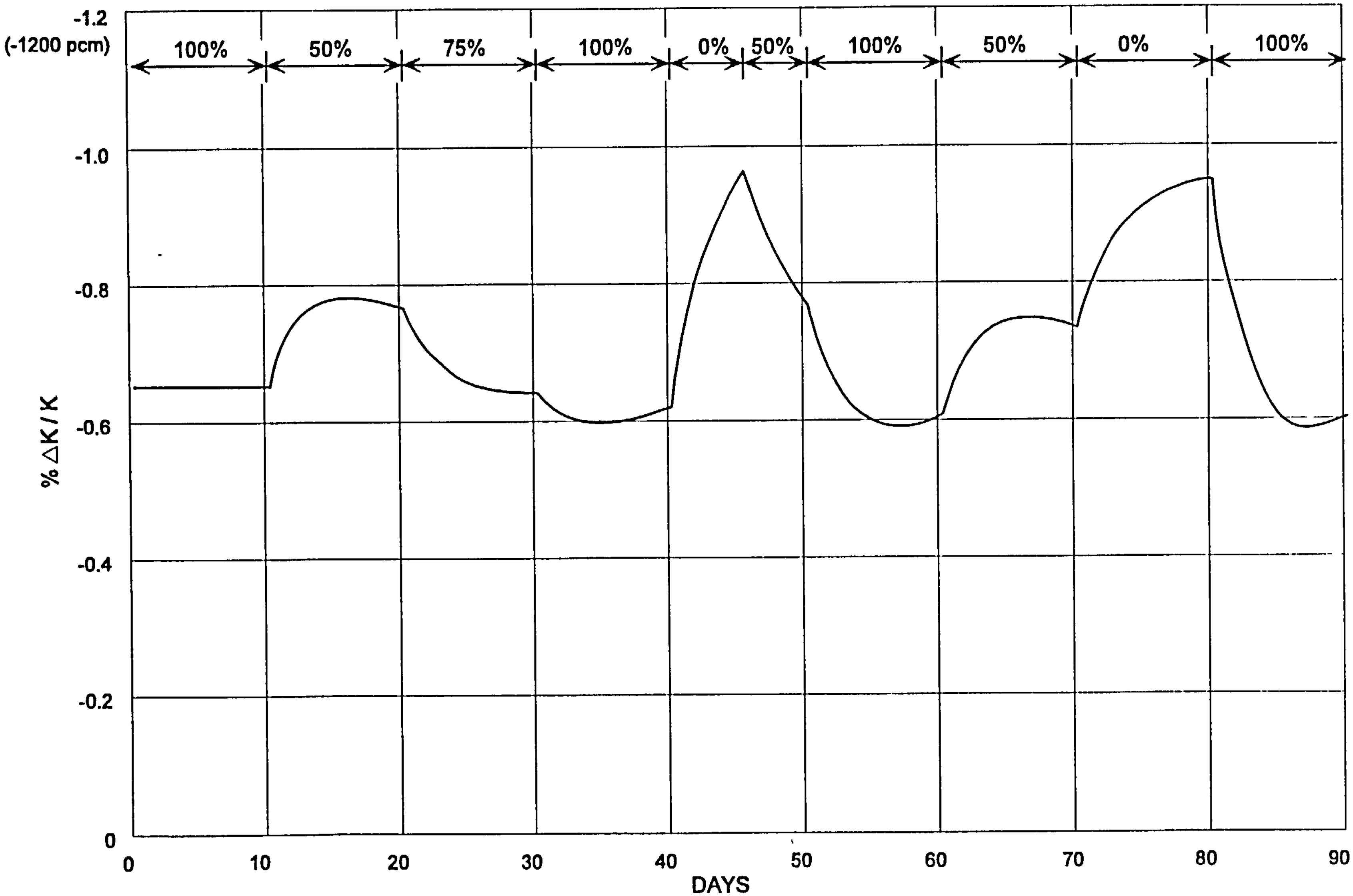


Figure 2.1-16 Samarium Transients

0199-3

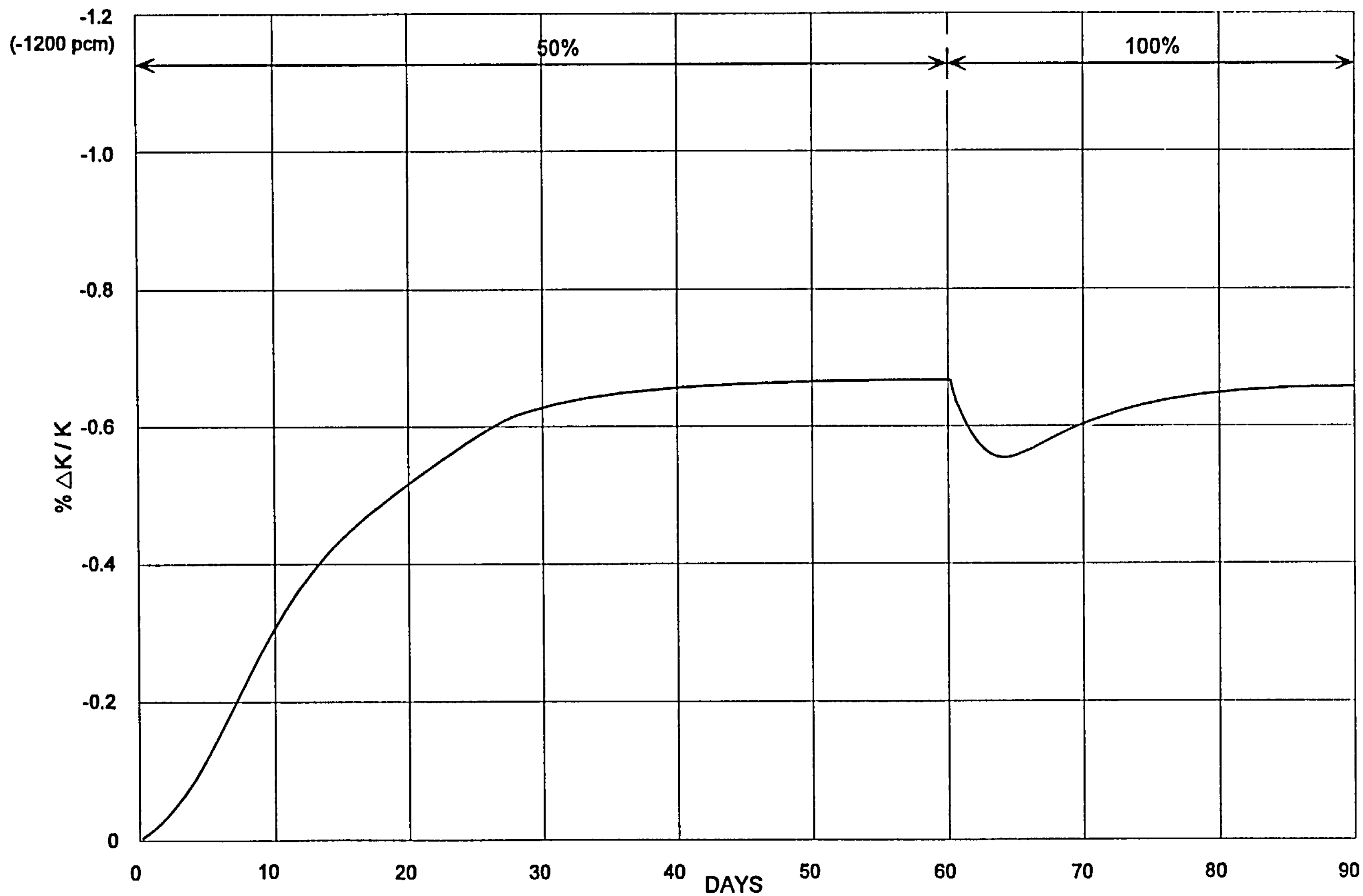


Figure 2.1-17 Samarium Transients Starting with a Clean Core

0199-3

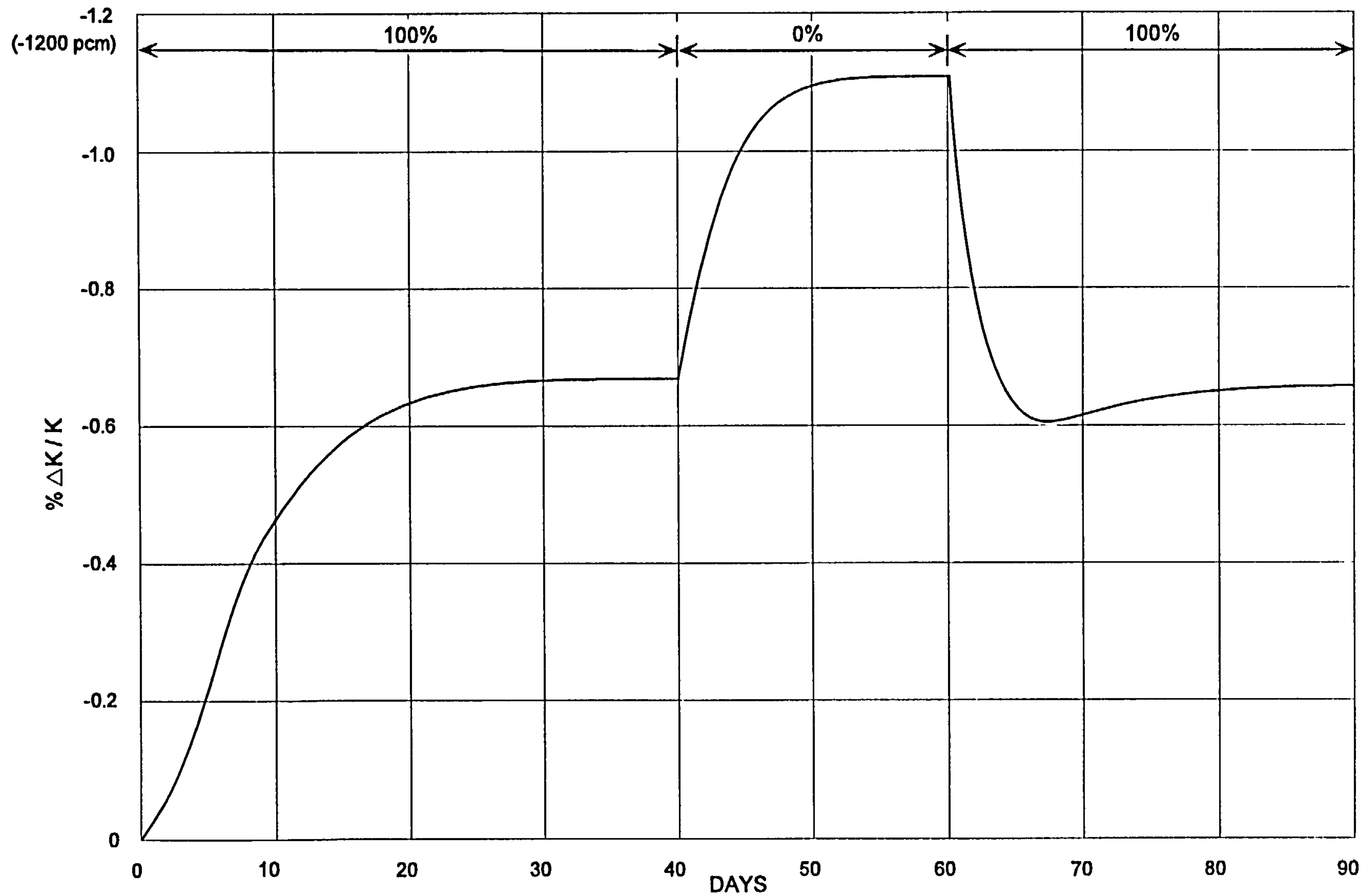


Figure 2.1-18 Samarium Transients Starting with a Clean Core

0199-3

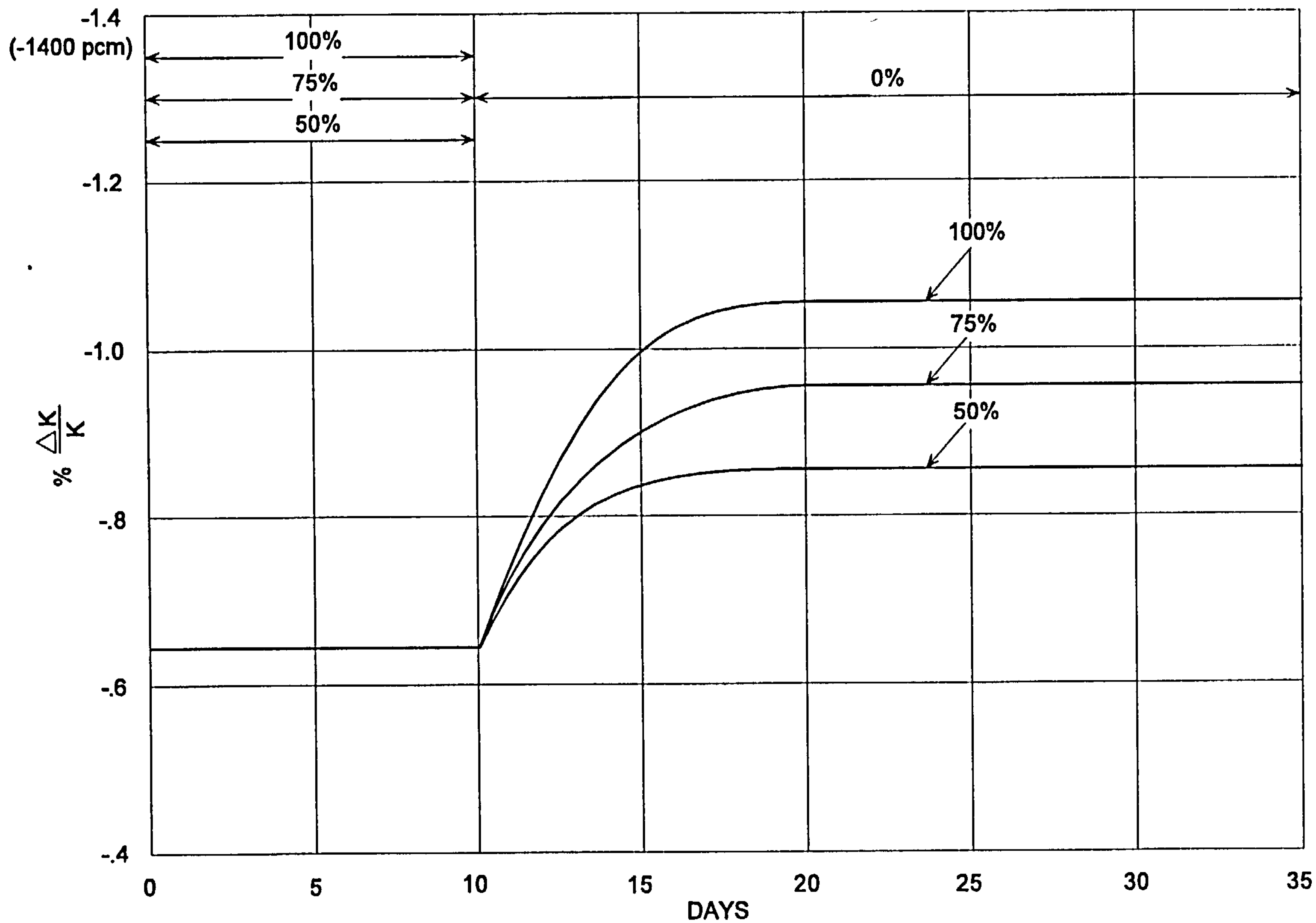


Figure 2.1-19 Samarium Shutdown Transients

0199-3

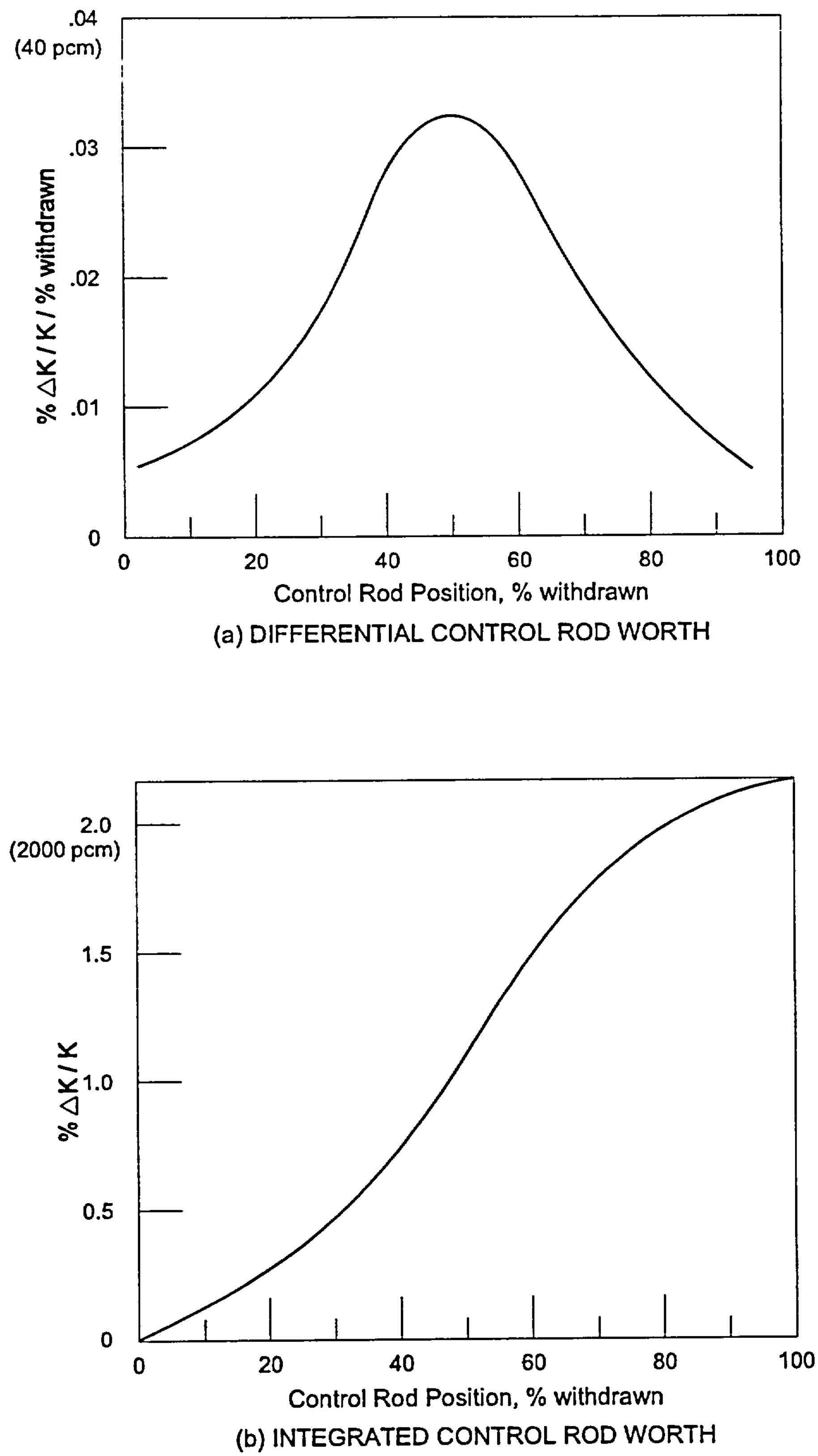


Figure 2.1-20 Integral and Differential Rod Worth

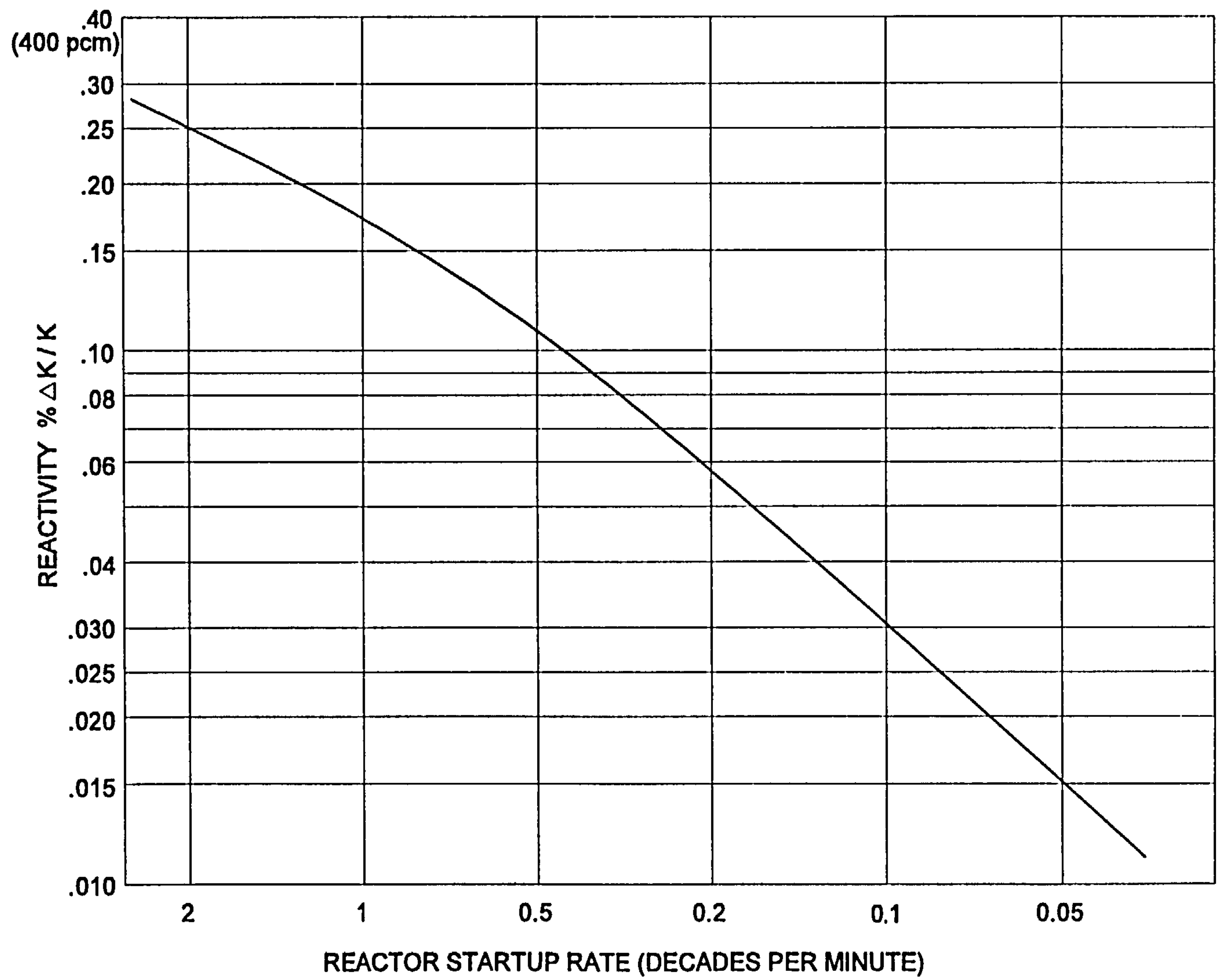


Figure 2.1-21 Reactivity versus Startup Rate

0199-3

Westinghouse Technology Systems Manual

Section 2.2

Power Distribution Limits

TABLE OF CONTENTS

2.2 POWER DISTRIBUTION LIMITS 2.2-1

2.2.1 Introduction 2.2-1

2.2.2 Thermal Hydraulic Considerations 2.2-2
2.2.2.1 Heat Generation Process 2.2-2
2.2.2.2 Heat Transfer Process 2.2-3
2.2.2.3 Fluid Heat Transfer 2.2-3
2.2.2.4 Departure from Nucleate Boiling 2.2-4
2.2.2.5 Departure from Nucleate Boiling Ratio (DNBR) 2.2-5

2.2.3 Nuclear Power Distribution Considerations 2.2-6
2.2.3.1 Peak Power Limits 2.2-7
2.2.3.2 Power Distribution Measurement 2.2-8
2.2.3.3 Hot Channel (Peaking) Factors 2.2-9
2.2.3.4 Peaking Factor Correction Terms 2.2-10
2.2.3.5 Changes to Peaking Factor Limits 2.2-11
2.2.3.6 Height Dependency Correction Term K(Z) 2.2-12
2.2.3.7 Enthalpy Rise Hot Channel Factor 2.2-13
2.2.3.8 Quadrant Power Tilt Ratio 2.2-13
2.2.3.9 Axial Flux Difference and Xenon Transients 2.2-14

2.2.3 Summary 2.2-15

LIST OF FIGURES

2.2-1 .... Local Radial Temperature and Velocity
2.2-2 .... Heat Rate Versus Temperature Difference
2.2-3 .... Measured versus Predicted Critical Heat Flux
2.2-4 .... K (Z) Correction Term
2.2-5 .... Axial Flux Difference Limits

2.2 POWER DISTRIBUTION LIMITS

Learning Objectives:

1. Define the following terms:

   a. Departure from nucleate boiling (DNB)
   b. Departure from nucleate boiling ratio (DNBR)
   c. Power density (linear heat generation rate)
   d. Heat flux hot channel factor ($F_Q(Z)$)
   e. Axial flux difference (AFD)
   f. Enthalpy rise hot channel factor ($F_{\Delta H}^{N}$)
   g. Quadrant power tilt ratio (QPTR)

2. Explain why DNBR is required to be greater than a specific limit.

3. Explain why the F_Q limit is varied as a function of core height.

4. List the 4 operational requirements that ensure the F_Q limits are not exceeded between surveillance intervals.

5. Explain how AFD limits ensure F_Q is not exceeded.

2.2.1 Introduction

The concept of placing limits on "hot channel factors" or "peaking factors" was introduced to limit the maximum power produced in the fuel to a value consistent with fuel design limitations. These limitations fall into two basic categories:

1. thermal hydraulic design considerations
2. nuclear power distribution considerations

The two are not easily separated, but will be discussed separately for ease of clarification. The fuel design limitations provide adequate heat transfer (thermal hydraulic considerations) compatible with heat generation distribution (nuclear or power distribution considerations) in the core. These design limitations ensure adequate heat removal by the reactor coolant system during normal conditions or by appropriate engineered safety features during emergency conditions. The general performance and safety criteria are:

1. Fuel damage is not expected during normal operation and operational transients (Condition I) or any transient conditions arising from faults of moderate frequency (Condition II).

2. The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged. However, sufficient fuel damage might occur to preclude immediate resumption of operation and result in considerable outage time.

3. The reactor can be brought to a safe state and the core configuration can be kept subcritical with acceptable heat transfer characteristics following a transient arising from Condition IV events.

Since 1970, Westinghouse has been using the classification of plant conditions as described in ANSI Standard 18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," which amplifies the General Design Criteria of 10 CFR Part 50, Appendix A. This standard divides plant operating conditions into four categories in accordance with the anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

1. Condition I: Normal Operations
2. Condition II: Faults of Moderate Frequency
3. Condition III: Infrequent Faults
4. Condition IV: Limiting Faults

The basic principle applied in relating design requirements to each of the conditions is that the most frequent occurrences must yield little or no radiological risk to the public and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur.

Condition I occurrences are those which are expected frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action.

Condition II faults, at worst, result in a reactor shut down with the plant capable of returning to normal operation. By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., a Condition III or IV category accident. In addition, Condition II events are not expected to result in fuel rod failures.

Condition III occurrences are faults which may occur very infrequently during the life of the plant. They will be accommodated with the failure of only a small fraction of the fuel rods, although sufficient fuel damage might occur to preclude resumption of reactor operation for a considerable outage time. A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the reactor coolant system or containment barriers.

Condition IV occurrences are faults which are not expected to take place, but are postulated, because their consequences would include the potential for the release of significant amounts of radioactive material. They are the most drastic events which must be designed against and thus represent limiting design bases. Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of 10 CFR Part 100 limits. Detailed descriptions of and the transients associated with each of these conditions may be found in Chapter 5.0.

2.2.2 Thermal Hydraulic Considerations

The design of the core and its heat transfer system (reactor coolant system) must be compatible. That is, the heat transfer must be equal to or greater than the heat generation rate or overheating and possible damage to the fuel may occur. Both processes (heat generation and heat transfer) should be understood.

2.2.2.1 Heat Generation Process

Of the total heat energy released during reactor operations, 97% is transferred from the fuel to the coolant (energy of all fission fragments, beta particles, some neutrons, and some gammas) and 3% is released directly into the reactor coolant, from the pressure vessel internals, and secondary shield as a result of the heating of these materials by neutrons and gammas. The energy released by the nuclear fission process in a PWR is emitted as heat from the fuel rods. The reactor coolant is circulated through the core and removes the generated heat. The temperature of the coolant increases continuously as it passes through the core. Some local boiling occurs at the fuel rod-coolant interface, although the reactor coolant system is pressurized to prevent bulk boiling during normal operation. The heat energy added to the coolant is measured by the change in its enthalpy, in Btu per pound.

2.2.2.2 Heat Transfer Process

The heat generated in the fuel must be transferred through the fuel pellet, across the pellet-clad gap, and then through the clad to the coolant (Figure 2.2-1). The heated coolant is circulated out of the core and used to boil the water in the secondary side of the steam generators. The thermal conductivity (ability to transfer heat) of the fuel is quite low. The fuel is manufactured in a form that is ceramic or nonmetallic in structure. This results in good fission product retention and a high melting point. Unfortunately, it also results in poor heat transfer characteristics.

Because of this resistance to heat transmission, the temperature gradient within the fuel must be high to achieve a satisfactory rate of heat transfer. Fortunately, the melting point of UO_2 is high enough (5080°F for unirradiated fuel, approximately 4890°F for expended fuel) so that acceptable heat transfer rates can be attained.

Another resistance to heat transfer from the fuel pellet to the coolant is presented by the gap between the pellet and the interior walls of the cladding. The gap conductance is dependent upon the size of the gap, the nature of the gas in the gap, and the extent of direct pellet-to-clad contact. When the fuel is cold, there is clearance between the fuel pellets and the inner walls of the zircaloy rods.

At operating conditions, the highest temperature pellets will be in at least partial contact with the cladding's inner wall. As irradiation progresses, the pellets swell and crack to some extent, increasing the amount of pellet-to-clad contact. Fission product gases mix with the helium gas originally present in the fuel rods. All these factors make the gap conductance variable and difficult to predict. Compared with the other thermal resistances, the cladding presents the least thermal resistance, and the temperature gradient from the inner to outer wall of the cladding is very small.

2.2.2.3 Fluid Heat Transfer

The transfer of heat from the cladding surface to the coolant must also be considered in the heat transfer process. This requires an introduction to thermal hydraulics terminology and concepts.

1. Evaporation is the conversion of a liquid to a vapor.
2. Boiling is the evaporation of a liquid occurring within the body of the liquid by the mechanism of bubble formation.
3. Convection is the transfer of heat from one location to another by fluid motion between regions of unequal density that result from nonuniform heating.
4. Radiation is the transfer of thermal energy by means of electromagnetic waves, with no material medium playing an essential role in the process of transmission.
5. Conduction is the transfer of heat through the conducting medium without perceptible motion of the medium itself.

In a liquid, the types of heat transfer can be broken down into four categories, known as "regimes." These may be best explained by referring to experiments performed with an electrically heated wire submerged in a pool of liquid, a situation similar to the transfer of heat by boiling from any heated surface to a pool.

This experiment relates the heat flux per unit area (or Q/A) to the temperature difference between the surface of the wire and the saturation temperature of the liquid. At low heat transfer rates between the wire surface and the liquid, heat is transferred via natural convection. Natural convection is shown by Regime I on Figure 2.2-2. The heated liquid rises to the surface, where evaporation without the formation of steam bubbles occur. Since temperature, or ΔT, is the "driving force" for transferring heat, the amount of heat transferred increases with the temperature differential while in the natural convection regime.

At a temperature difference of about 10°F, small vapor bubbles start to form at various points along the wire surface. These small bubbles then move into the liquid surrounding the wire and collapse in the cooler water. This agitates the wire/liquid interface promoting better heat transfer. This region of small bubble formation is known as the "nucleate boiling regime" and is designated as Regime II in the figure. In this regime the increase in the amount of heat transferred from the wire surface is due primarily to the increased convective heat transfer due to agitation of the liquid at the heated surface, not due to the heat actually carried away as enthalpy of the vapor in the form of steam bubbles.

When Regime III (partial film boiling regime) is reached, there is a reduction in the amount of heat transfer. This reduction in heat transfer is due to the insulating effect of the formerly mobile nucleate steam bubbles becoming larger and combining to form stationary steam bubbles. Heat removal by radiation through the steam is much less efficient than that of convection or conduction to a liquid. The thermal energy is passed from molecule to molecule in the course of purely thermal motion, with no mass motion of the medium.

When the bubbles become somewhat stagnant, they tend to form an insulating layer between the heated surface and the liquid. In the partial film boiling regime, the film itself is unstable. It spreads over a part of the heated surface and then breaks down. Under these conditions, some areas of the surface exhibit violent nucleate boiling, while film boiling occurs in other areas.

The precise point where the mobile, agitating nucleate boiling ends and where partial film boiling begins is referred to as the "critical heat flux" or "departure from nucleate boiling" (DNB) and will be discussed later.

The fourth area, Regime IV, is the film boiling regime. Here, as in the partial film boiling regime, steam blanketing hinders the transfer of heat. With the increasing ΔT, however, the film becomes stable and the heat transfer mechanism is by radiation and conduction, neither of which is very efficient. Heat transfer by radiation requires an extremely large ΔT, which is not attainable in existing commercial reactors without producing considerable fuel damage.

This experiment and its results are not fully applicable to the conditions existing in the reactor coolant system with its forced circulation. Conditions in the reactor coolant system are much more complicated than in the experimental model. It is, however, a close enough approximation that it can be enhanced by computer modeling and actual data input to be used by the designers to predict core conditions with a reasonable certainty.

2.2.2.4 Departure from Nucleate Boiling (DNB)

Pressurized water reactors are designed to operate in the nucleate boiling region at high power levels and are not allowed to operate, at any time, with partial film boiling. Due to the reduction in heat transfer capability that occurs with partial film boiling, excessive fuel rod surface temperatures would actually lower the heat being transferred. This would lead to a larger ΔT between the fuel rods and the coolant, which would lower the heat transfer rate even further. Figure 2.2-2 illustrates this effect. It further illustrates that once the conditions leave the nucleate boiling regime, (point "a"), the ΔT must increase from less than 100°F to slightly more than 1000°F (point "b") before a net increase in heat flux is realized. This rise of more than 900°F in ΔT occurs as a rise in the temperature of the heated surface.

Once the critical heat flux is reached, the heat flux cannot be increased without a large increase in ΔT in the form of fuel rod surface temperature. This could result in a failure of the fuel. For this reason, the heat flux of the fuel rods must be limited to some value below the critical heat flux.

In an actual core, many things have an influence on the actual point at which DNB or the critical heat flux occurs. Core flow rate, coolant pressure, coolant temperature, coolant channel cross section, and localized variations in the fission rate along the length of the fuel rod can all cause changes in the critical heat flux value. Since many factors vary the point of critical heat flux, it is impossible to predict it with 100% accuracy.

Westinghouse initially used a calculational model known as the "Westinghouse W-3 DNB Correlation" and later used the WRB-1 correlation to assure that core conditions would not result in exceeding the critical heat flux. These correlations employ a computer-assisted program to examine the relationship between the many physical variables and the critical heat flux.

Using variables such as pressure, mass velocity, heated length to point of the critical heat flux, various hydraulic parameters, and grid designs, the WRB-1 correlation predicts the heat flux required to cause DNB. Actual tests were conducted at Columbia University using a range of conditions as described above. Figure 2.2-3 illustrates the difference between the heat flux at DNB predicted by the WRB-1 correlation and the actual heat flux at DNB as determined from the tests. Over 1100 points are plotted on this figure. If the prediction were perfect, all points on this graph would fall on the 45-degree line. However, the prediction of DNB is not exact and in most cases either over- or under-predicts the actual point of DNB.

As an example, using Figure 2.2-3, if the predicted heat flux to cause DNB (horizontal axis) for a given set of conditions is 800,000 BTU/hr-ft^2 and the measured value (vertical axis) for the same set of initial conditions was 700,000 BTU/hr-ft^2, a point is placed at the intercept of these two values. In this case DNB occurs at 100,000 BTU/hr-ft^2 less than the predicted value.

Since, in an operating core, the critical heat flux is predicted and never actually attained, a degree of safety margin or conservatism must be applied to the correlation to ensure that the core is operated below the departure from nucleate boiling point. It was concluded, to meet this design criterion, that the limit for DNBR (DNBR is defined as the heat flux required to reach DNB divided by the actual local heat flux) should be set at 1.17 (as predicted by the WRB-1 correlation). This value is displayed as the 0.85-slope line on Figure 2.2-3. This line constitutes the limiting DNBR criterion, and at least 95% of the plotted points must fall above this line.

2.2.2.5 Departure from Nucleate Boiling Ratio (DNBR)

The design criterion established for DNB is -- there will be at least a 95 percent probability that departure from nucleate boiling will not occur on "the" limiting fuel rod during normal operation and operational transients and any transient conditions arising from faults of moderate frequency (Condition 1 and 2 events), at a confidence level of 95 percent. For Condition 3 and 4 events (limiting faults), a limited number of fuel rods are allowed to violate the 95/95 DNB criterion. The actual limit depends upon a given plant's offsite radiation dose release criteria.

The above design criterion ensures safe core operation as discussed below. By preventing departure from nucleate boiling, adequate heat transfer is assured between the fuel cladding and the reactor coolant, which prevents cladding damage as a result of inadequate cooling. The DNBR concept was developed as a measure of the margin of safety existing between the critical heat flux and the actual or existing heat flux. DNBR is defined as:

$$DNBR = \frac{\textit{the heat flux required to reach DNB}}{\textit{the actual local heat flux}}$$

Example: If the actual heat flux present at some instant is only ½ of that heat flux which could produce a departure from the nucleate boiling regime, then

$$DNBR = \frac{1.0}{0.5} = 2.0$$

In this instance the DNBR is equal to 2.0.

The minimum allowed DNBR varies depending upon which correlation is used for the calculation. Often the W-3 DNB correlation is used and places the DNBR limit at 1.3. This correlation uses a single tube as the reference point, with correction factors for unheated walls and non-uniform axial heat flux. Further modifications have been made to the W-3 DNB correlation to incorporate L-grid and R-grid fuel assembly designs (Section 3.1), which lower the DNBR limit to 1.24 and 1.28, respectively.

Predicting DNB with greater accuracy was accomplished when Westinghouse developed the WRB-1 (Westinghouse Rod Bundle) correlation. This correlation uses full-length tubes in either 4x4 or 5x5 bundle arrays. The calculations or predictions include both L-grid and R-grid configurations, along with uniform and non-uniform heat flux distributions. Using the WRB-1 correlation, a DNBR limit of 1.17 was shown to meet the 95 x 95 criteria explained earlier.

2.2.3 Nuclear Power Distribution Considerations

As previously discussed, thermal-hydraulic considerations place constraints on the design and operation of the core and its support systems. In addition, nuclear power distribution must be held within limits to ensure the integrity of the fuel cladding. The specific design criteria to ensure the integrity of the zircaloy fuel cladding are as follows:

1. DNBR >1.30 as calculated by the Westinghouse W-3 correlation or >1.17 using the WRB-1 correlation. The expected minimum value of DNBR at nominal operating conditions is 2.08.

2. Fuel center line temperature below the melting point of the UO_2 ceramic fuel pellets. This condition is imposed because the change from solid to liquid is accompanied by swelling which could crack the clad. The melting temperature of UO_2 is assumed to be 5080°F minus 58°F for each 10,000 MWD/MTU of burnup. The expected peak value of the centerline temperature at nominal operating conditions is 3275°F.

3. Cladding stress less than the zircaloy yield stress. (Stress is the force applied per unit area.)

4. Cladding strain less than 1%. (Cladding strain is a measure of how much the cladding has been stretched past its ability to recover elastically. A strain of 1% means that it has been deformed permanently a total of no more that 1% of its original diameter.)

Cladding stress and strain are minimized by limiting the internal fission gas pressure to less than the external reactor coolant system pressure of 2250 psia, and limiting the average cladding temperature to less than 850°F. Above this temperature the minimum ultimate yield strength reduces to the design yield strength.

Some stress and strain occur when the fuel is in contact with the interior wall of the clad. This is due to the fuel having roughly twice the thermal expansion coefficient of the clad. As the power level changes, the temperatures of both the fuel and the clad change. The uneven expansion and contraction of the fuel and clad cause stress and strain. Linear power density, kilowatts of power produced per foot of fuel rod, must also be limited during normal operations (Condition I) so that in the event of a worst-case loss of coolant accident (LOCA, a Condition IV event), the criteria of 10 CFR Part 50.46 would be met.

If a LOCA should occur, it is expected to be accompanied by some fuel cladding failure. The idea, then, is to limit the amount of fuel failure that could be expected to occur rather than to prevent cladding failure altogether. In addition to imposing limits on the fuel design, the NRC also imposes minimum design criteria on the Emergency Core Cooling System (ECCS). Each ECCS design must demonstrate the capability to maintain core conditions within five general limits in the event of a worst-case LOCA. These are stipulated in 10 CFR 50.46 and listed below:

1. Peak clad temp. < 2200°F
2. Clad oxidation < 17% clad thickness
3. H_2 generation <1% hypothetical maximum
4. Coolable core geometry maintained
5. Long term cooling maintained

2.2.3.1 Peak Power Limits

Peak power limits are placed on the core to avoid a boiling crisis and to eliminate the conditions which could cause fuel pellet melt. The boiling crisis puts a physical limit on the amount of heat that can be extracted from a fuel rod. The melting point of the fuel material places a limit on the amount of heat that can be generated by the fuel rod. For safety purposes, the license limit on the heat flux is set well below these physical limits.

The most economical way to operate the reactor would be to have the heat flux and power level at all points just equal to the maximum allowable to get the maximum power from each pound of fuel. This cannot be achieved because the neutron flux and the resulting power distribution, is non-uniform. Although attempts are made to "flatten" the radial power distribution with fuel enrichment and burnable poison rods, the flux decreases near the edge of the core. This means that many assemblies around the outer edge of the core operate below license limits. Their power output cannot be increased excessively without violating limits toward the center of the core. Secondly, the heat flux cannot be equal to the maximum allowable at all points along the vertical axis of any channel because the power output at the ends of the channel is lower. If the maximum allowable heat flux is reached near the ends of a channel, it may be exceeded at the middle of the channel.

These and other factors produce differences in power levels throughout the core. In addition, many localized conditions exist to further complicate the situation. These include the location of fuel assembly grid straps, minor manufacturing differences in fuel pellet enrichment and density, gaps between adjacent pellets, a partially inserted control rod, etc.

Since the potential exists for localized "hot spots," those hot spots must be accounted for. Westinghouse has demonstrated by actual experiments and computational models that the four specific design criteria to ensure the fuel cladding integrity (listed in section 2.2.3) can be met if the maximum power output during normal operations does not exceed 13.6 kW/ft of fuel rod. The value of 13.6 kW/ft then becomes, in effect, the limit to be imposed on peak power output in the fuel.

2.2.3.2 Power Distribution Measurement

Power is proportional to the fission rate, which is in turn proportional to the thermal neutron flux. Thus, local power in the fuel is often taken to be proportional to the thermal neutron flux at the point in question.

If the fuel is to be protected from localized high power conditions, it is necessary to be able to locate and measure these "hot spots." If each individual point of each fuel pin could be monitored, then an upper limit of 13.6 kw/ft could be set on local fuel pin power. Monitoring each fuel pin is, however, a physical impossibility.

Since it is impossible to measure each foot of each individual rod, the next best approach is to predict, as accurately as possible, the condition of each location using available information. This consists of the read-out from the incore monitoring system, which is collected by the plant computer. The incore system consists of 6 separate miniature flux detectors that can be driven into the hollow center support thimbles in 58 of the 193 fuel assemblies. As the movable incore detector moves from the bottom to the top of a fuel assembly, and then back down, it provides an electrical output which is transmitted to the plant computer. The computer receives and stores this information once per second. Since the probe travels the full length of the fuel assembly in one minute, the result is a "stack" of points monitored. Instead of a single picture of the core, there are 61 separate pictures collected in each of 58 different fuel assemblies. Having many individual pictures allows a more precisely detailed examination of the core.

Since only 6 detectors are provided, a total of 12 "passes" must be run to monitor all the available assemblies. The small size of the incore movable detectors enables them to "see" highly localized neutron flux conditions. This information is the basis for the computer-assisted calculations to follow. The information generated by the incore system is still in the form of fuel assembly-specific information. It must be extrapolated to include the other, unmonitored, fuel assemblies. The flux level or power detected in any fuel assembly is heavily influenced by the fuel assemblies surrounding the one being monitored. This general diffusion is accounted for in the extrapolation model.

After the information received from the instrumented 58 assemblies has been extrapolated to calculate what is occurring in the other 135 assemblies, a further extrapolation is performed. This involves calculating what each fuel rod is contributing to the power levels, either measured or calculated, for each fuel assembly. With this computer generated information, the fuel design engineer now has an idea about what is occurring in each fuel assembly and in each fuel pin. Not only is the total power known (calculated) for each fuel pin, but also the power level for each elevation of each pin. Remember that the incore detector provided outputs to the plant computer as it traversed the length of the core and provided data at each core slice.

Since the six probes travel together and their locations are known to the computer, the information can be made to reflect relative power levels at any and all core elevations. The information is now in its final form and power distribution throughout the core is known. Every fuel pin has been identified and measured.

The only thing left to do is to make sure that none of the core locations is producing more than the limit of 13.6 kw/ft.

2.2.3.3 Hot Channel (Peaking) Factors

The problem with the technique described above for locating and measuring local power distribution throughout the core lies in the fact that it is not an on-line system. Typically, incore flux mapping is performed every 30 effective full power days (EFPD). That is, this data is not available on a real-time basis. It takes one or two hours to run the incore system through the flux-mapping routine. In addition to this time delay, the information collected by the on-site computer may have to be transmitted to an off-site computer of sufficient capacity to run the computations and extrapolations. All this can take several days to perform, so the information received is historical. Therefore, other methods are employed to ensure safe operation of the core between flux mapping runs.

To ensure the core is operated within the prescribed limits, technical specifications require that four operational requirements be monitored. These ensure that the peaking factor limits are not exceeded between the required surveillance intervals. These operational requirements are as follows:

1. Control rods within a group move together with no individual rod differing by more than 12 steps from its demanded position.
2. Control rod groups are sequenced with overlapping groups.
3. Rod insertion limits are maintained.
4. Axial power distribution expressed in terms of axial flux difference (AFD) is maintained within its limits.

In the early development stages of its core design, Westinghouse found that, in an unrodded core, a relatively constant relationship exists between the peak power in the core and the average power. Since this condition yields a natural flux shape, even during a power level change in which the peak and average values are changing, the ratio remains relatively constant. In other words:

$$\frac{peak}{average} = K, \text{ where } K \text{ is a constant}$$

The average linear heat generation rate (kw/ft) or power density can be calculated by dividing the total power of the core by the total active rod length. For the plant under discussion, the thermal megawatt rating is 3411 Mwt, 97.4% of which is produced in the fuel. The other 2.6% is contributed by radiation heating of vessel materials. The total number of kw are:

$$3411\ Mw \times \frac{1000\ kw}{Mw} \times .974$$

$$= 3,322,314\ kw$$

A similar calculation yields the number of active feet of fuel rods:

$$193 \textit{ assemblies } X \frac{264 \textit{ rods}}{\textit{assembly}} X \frac{11.97 \textit{ ft}}{\textit{rod}}$$

$$= 609{,}895 \textit{ ft}$$

The average heat generation rate at full power is:

$$\frac{3{,}322{,}314 \textit{ kw}}{609{,}895 \textit{ ft}} = 5.45 \textit{ kw/ft}$$

This is the average power in one foot of fuel rod when operating at 100% power.

Safety analysis has determined that fuel damage will not result if the peak power does not exceed 13.6 kw/ft. Then at 100% power, the core can be shown to be safe if the ratio of the peak power to average power is 2.5 or less:

$$F_Q \textit{ Limit} = \frac{\textit{peak}}{\textit{average}} = \frac{13.6 \textit{ kw}}{5.45 \textit{ kw}} = 2.50$$

In other words, the plant can be safely operated if the peak/avg ratio does not exceed 2.50. In fact, core power distribution limits are placed on this ratio of peak to average. A typical peak power value at 100% power would be 10.9 kw/ft resulting in a ratio of 2.0 (i.e.,10.9 / 5.45), which is well below the limit of 2.50.

2.2.3.4 Peaking Factor Correction Terms

Since calculated values are used instead of actual measurements, and since manufacturing tolerances preclude having perfect, defect-free fuel pellets and pins, the measured values have been conservatively increased. Experiments to match predictive versus actual conditions have demonstrated that the actual peak power is no more than 4.58% greater than the calculated peak power. This was rounded off to 5% and is called the "measurement uncertainty factor."

In addition to the uncertainty in measuring the magnitude of peak local power density, there is also some uncertainty in precisely locating the peak local power density. This is due to the fact that the measured flux shape produced by the incore detectors and the computer programs is based on the assumption that all fuel rods are identical. Variations from rod to rod in fuel pellet enrichment, density and diameter; in the surface area of the cladding; and in the eccentricity of the pellet-to-clad gap could all make the actual peak power density greater than the measured peak.

Statistical checks indicate that due to the tight quality assurance standards imposed on the fuel rod manufacturing process, it is almost certain that the magnitude of the actual peak local power density is no more than 3% greater than the peak predicted by the incore system and its computational model. This 3% is known as the "engineering uncertainty factor."

These uncertainty factors are included in the measured value of F_Q. After the measured F_Q is found by the measurement and extrapolation techniques previously described, it is increased by the amount of the uncertainty factors.

F_Q = Total Measured Heat Flux Hot Channel Factor (with correction terms applied)

$$F_Q = \frac{\textit{maximum kw/ft}}{\textit{average kw/ft}}$$

$$F_Q = F_Q^N \times F_U^N \times F_Q^E$$

where:

F_Q^N = measured and extrapolated nuclear peaking factor, peak nuclear flux to average nuclear flux ratio.

F_U^N = nuclear uncertainty factor, which accounts for possible errors in the measurement techniques involved. This "measurement uncertainty factor" is 1.05.

F_Q^E = engineering uncertainty factor, which accounts for variations in the manufacturing processes and the subsequent deviations in pellet density and diameter, fuel rod eccentricity, and pellet enrichment. This "engineering uncertainty factor" is 1.03.

Example: To illustrate these terms, assume a plant initiated a set of incore data runs. After the data was collected, an off-site computer with the approved math model found, through extrapolation, that the highest measured peak to average ratio was 2.0. Using the base formula:

$$F_Q = F_Q^N \times F_U^N \times F_Q^E$$

$$F_Q = 2.0 \times 1.05 \times 1.03 = 2.16$$

The measured F_Q with its correction terms is shown above to be 2.16 and is then compared to the F_Q limit. As shown in this example the corrected measured value is within the imposed F_Q limit of 2.50.

2.2.3.5 Changes to Peaking Factor Limits

Since the introduction of the peaking factors concept there have been several major changes in the measurement methods, in the math models, and in the basic concept itself. Along with these changes there have been significant changes in the core structure and design. As a result, the peaking factors have undergone several revisions.

As the state of the art has advanced, more accurate and realistic analyses of core behavior under transient and accident conditions have caused several changes in the allowable limits. At one time a maximum of 18.0 kw/ft was considered safe. Using an average of 5.45 kw/ft at 100% power as calculated earlier, this yielded an F_Q limit of 3.30 (18.0/5.45 = 3.30). Later analysis, however, revealed some errors in the original assumptions. For instance, the early models did not include the possibility that the assumed LOCA could cause fuel swelling and clad burst. This could impede the flow of water from the ECCS which refills the reactor vessel and refloods the core after the initial LOCA blowdown. Delaying or impeding ECCS flow could cause more damage than originally calculated. Since the potential for rod burst could not be eliminated, the only alternative was to reduce the peak allowed kw/ft for normal operation. Therefore, the assumed LOCA would start with a lower peak power.

Similar reductions have occurred due to such diverse conditions as the practice of plugging leaking steam generator tubes (causing an increase in resistance to flow, and heat transfer area reduction), hotter than expected upper head temperatures, math errors found in the zirconium clad/water reaction rates, and fuel densification which causes localized neutron flux peaks.

As a result of the above considerations the F_Q limit at most Westinghouse plants is 2.32. To calculate the equivalent peak power allowed for this value is as follows:

$$F_Q^N = \frac{Peak\ Power\ (Kw/ft)}{Average\ Power\ (Kw/ft)} = 2.32$$

therefore,

$$2.32 = \frac{Peak\ Power\ (Kw/ft)}{5.45\ (Kw/ft)}$$

solving for Peak Power :

peak power = 2.32 x average power
= 2.32 x 5.45 kw
= 12.64 kw/ft

After all the correction terms and factors have been applied, a peak linear power density of only 12.64 kw/ft is allowed for a core originally having been deemed safe operating with a peak of 18.0 kw/ft.

2.2.3.6 Height Dependency Correction Term K(Z)

In addition to modifying the "measured" F_Q as discussed in section 2.2.3.4, it became necessary to modify the F_Q limit to account for another problem. Studies, experiments and computer models revealed that the fuel damage resulting from a LOCA is strongly height dependent. Specifically, the upper areas of the core will experience more damage than the lower areas. This is due to the nature of both the initial blowdown (upper areas uncovered first) and the reflood (upper areas reflooded last).

Additionally, for smaller (3" - 4" diameter) breaks, experiments and computer runs have indicated that reflooding of the top 10" - 12" of the core could be delayed for a significant time. The small break LOCA causes a backpressure which reduces the reflood flow rate.

Rather than reducing the peaking factor limit throughout the core, it was decided to make the limit more restrictive at the higher core elevations. Figure 2.2-4 shows the correction term that must be applied to the F_Q limit. Instead of a single limit that applies from top to bottom, the core now has a specific limit for each core elevation. The new measured peaking factor is $F_Q(Z)$ which is defined as:

$$F_Q(z) = \frac{maximum\ kw/ft\ at\ elevation\ z}{average\ kw/ft\ in\ the\ core}$$

For instance, the correction term for elevations between 0 and 6 ft is 1.0. This means that the $F_Q(z)$ limit for all locations below 6 ft. is the F_Q limit multiplied by the height correction term K(z), or

$F_Q(z)$ limit = F_Q limit x K(z) =
2.32 x K(z) =
2.32 x 1.0 = 2.32

There is no reduction or penalty imposed on the limit at core elevations less than 6.0 ft. For higher elevations, however, the correction term varies with core height. Between 6.0 feet and 11.0 feet, the correction term reduces linearly to approximately 0.94. This reduction accounts for a large LOCA, which would uncover the upper half of the core first and reflood it last. Imposing a more restrictive limit precludes operating the core with power tilted toward the top. A combination of an upward power tilt and a subsequent LOCA could exceed the fuel design limits. With more restrictive limits, this possibility is reduced. For example:

At 11.0 ft, K(z) = 0.94
Therefore, at 11.0 ft the $F_Q(z)$ limit is:

$F_Q(z)$ limit = 2.32 x 0.94 = 2.18

The highly restrictive values of K(z) above 11.0 ft elevation are to preclude high power levels in the last 12 inches of the core. In this area, the "small break LOCA" analysis required an additional limitation to account for the damage resulting from backpressure increases which could oppose the reflood rate (i.e., reducing the reflood rate to less than 1 inch/sec as defined in 10 CFR 50 App. K). This restriction could occur if the break size were large enough to cause a blowdown of the core area but not large enough to allow the displacement of steam out the break when the ECCS starts reflooding the core. This impeding of the reflood water occurs to the extent where additional core damage results in the last 10" - 12" of the core. At the 12' level the correction term is 0.666.

$$F_Q(Z) \text{ limit} = 2.32 \times K(z) = 2.32 \times 0.66 = 1.53$$

It should be understood that all of these power distribution limits are operating limits. They ensure acceptable power distribution at the start of the accident, so that the accident does not cause fuel damage in excess of that stipulated by the ECCS design criteria.

2.2.3.7 Enthalpy Rise Hot Channel Factor

The limits on the heat flux hot channel factor, $F_Q(z)$, ensure that the peak power density does not exceed its limit and that the peak cladding temperature after a LOCA will not exceed 2200°F. However, limiting $F_Q(z)$ does not in itself ensure that DNB will not occur. DNB depends not only on the local power density but also on the local enthalpy and flow rate of the coolant. Consider a case where $F_Q(z)$ is under the limit at each core elevation, but the maximum heat flux at several core elevations occurs in the same coolant channel. In this case, the heat flux is limited at each location, but the coolant enthalpy increases more in that coolant channel than in any other. DNB is more likely to occur in that channel because the heat flux that causes DNB is lower when coolant enthalpy is higher. Therefore, another peaking factor is needed to protect against such cases. This peaking factor is called the enthalpy rise hot channel factor, and is defined as:

$$F_{\Delta H}^{N} = \frac{\textit{maximum integrated rod power}}{\textit{average integrated rod power}}$$

A typical technical specification limit for the enthalpy rise hot channel factor is:

$$F_{\Delta H}^{N} \leq 1.49[1 + .3(1 - P)]$$

where P = fraction of full power

2.2.3.8 Quadrant Power Tilt Ratio

Quadrant power tilt ratio (QPTR) is defined in technical specifications as:

The ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

The four upper and lower excore power range detectors are the instruments used to calculate the QPTR. If one of these detectors is inoperable the remaining three detectors will be used for computing the average.

The limit that is placed on the QPTR is 1.02. If the QPTR exceeds the limit and cannot be restored below its limit, thermal power must be reduced to restore calculated safety margins. If the QPTR were to exceed its limit it would most probably be due to a misaligned control rod.

2.2.3.9 Axial Flux Difference and Xenon Transients

One of the few conditions which could cause flux and power levels to be tilted toward the top of the core, and exceeding the restrictive $F_Q(z)$ limits in that portion of the core, is a xenon transient. The concentration of xenon at any given location in the core is transient in nature. If the neutron flux levels in the core are allowed to be high at some elevations and low in others, the xenon concentrations will also be of varying magnitudes. The following is an example of how a xenon transient could be initiated:

1. One bank of control rods is inserted into the core to the mid-plane while maintaining full power. This condition exists for 24 hours. While the rods are in the core, they depress the neutron flux in the upper half of the core and force a high flux to exist in the lower half. This initially causes the xenon in the lower half of the core to experience a high burn-up rate, and some hours later a new, higher xenon concentration is reached, consistent with the higher neutron flux and fission rate. The upper half of the core undergoes the opposite change: an initial increase in xenon concentration followed by a lower equilibrium xenon concentration.

2. At the end of the 24-hour period, the rods are withdrawn from the core. Since the two halves of the core now have different xenon levels, the flux will be depressed in the lower half of the core and will increase in the upper half of the core. This effect will eventually reverse itself in a cyclic manner, with each swing of xenon and flux being of a smaller magnitude. Over a period of approximately 48 hours, the transient should dampen itself out.

The problem, however, is in the initial swing of flux levels which displace the flux upward. With the highly restrictive $F_Q(z)$ limits at the higher elevations of the core, the limits would probably be exceeded.

To prevent exceeding these limits, operating conditions have been imposed which will minimize xenon transients. The most important of these are the axial flux difference (AFD) limits. AFD is a measure of the imbalance between the upper and lower halves of the core in terms of power or flux (φ). AFD is defined as:

$$AFD \ or \ \Delta\varphi = \frac{\varphi_{top} - \varphi_{bottom}}{\varphi_{top} + \varphi_{bottom} \ at \ 100\% \ power}$$

The AFD is determined from the outputs of the upper and lower excore neutron detectors.

For plants operating with "constant axial offset control," the AFD limit involves a target band, as shown in Figure 2.2-5, and the collection of "penalty minutes" for time operated outside the target band. This target band,+5% and -5% around the target, defines, the allowed variation from the natural flux profile of an unrodded core. As an example, assume the core is operating at 100% power with all rods out, and the delta flux target is -10%. The core would then be able to operate with a delta flux of -15% to -5% without collecting any penalty minutes. The + 5% and -5% flux difference around the target allows for a small amount of movement of the control rods.

Since the xenon distribution and concentration is time and flux dependent, the longer the core operates outside its target band, the greater the chance of initiating a xenon transient. Therefore, after 60 penalty minutes have accumulated, in a sliding 24-hour period, operation above 50% power is not allowed until the potential for a xenon transient has abated. In addition, if the delta flux exceeds some maximum value, as shown by the maximum AFD limit line, the delta flux must be reduced. The delta flux must be reduced to a value less than this absolute limit within 15 minutes or the power of the core must be reduced to less than 50%. This lower power level provides additional margin to core thermal limits while the xenon transient is dampened.

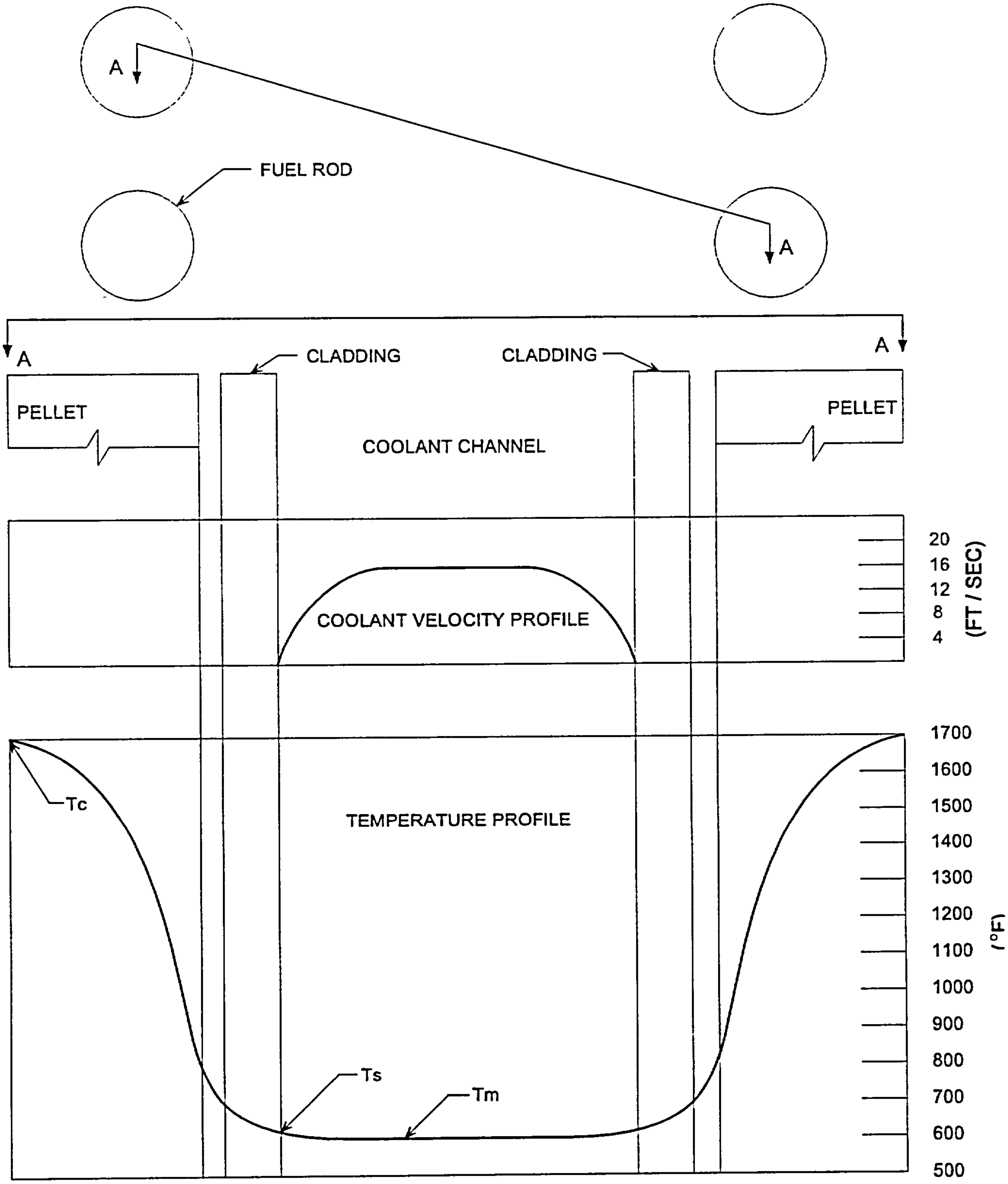
For some plants an analysis has been completed which allows "relaxed axial offset control." For these plants, the axial flux difference target band is no longer applicable and the axial flux difference is limited to the area defined by the absolute limits indicated on Figure 2.2-5. However, for plants incorporating the relaxed axial offset, the absolute limit is somewhat modified in size and goes to 100% power.

2.2.4 Summary

The heat generated in the core must be removed by the coolant in order to prevent or to minimize fuel damage. To be reasonably sure that the power of the core does not exceed the power removal capability of the coolant, limitations have been placed on peak power density and DNBR. In order to meet these limits, certain peaking factors and operational requirements must be maintained.

If the peaking factors are limited and the operational requirements are met, there is a reasonable assurance that the fuel will not be damaged during normal and transient operations and limited fuel damage during accidents. To ensure the peaking factors are met between surveillences four operational limits have been incorporated into the plants Technical Specifications. These limits are as follows:

1. The rods are maintained above the rod insertion limits.
2 The rods are sequenced and overlapped.
3. The rods within a group are aligned within 12 steps from their demanded position, and finally.
4. The axial flux limits are maintained.



Tc - Pellet Centerline Temperature

Ts - Rod Surface Temperature

Tm - Moderator Temperature

Figure 2.2-1 Local Radial Temperature and Velocity

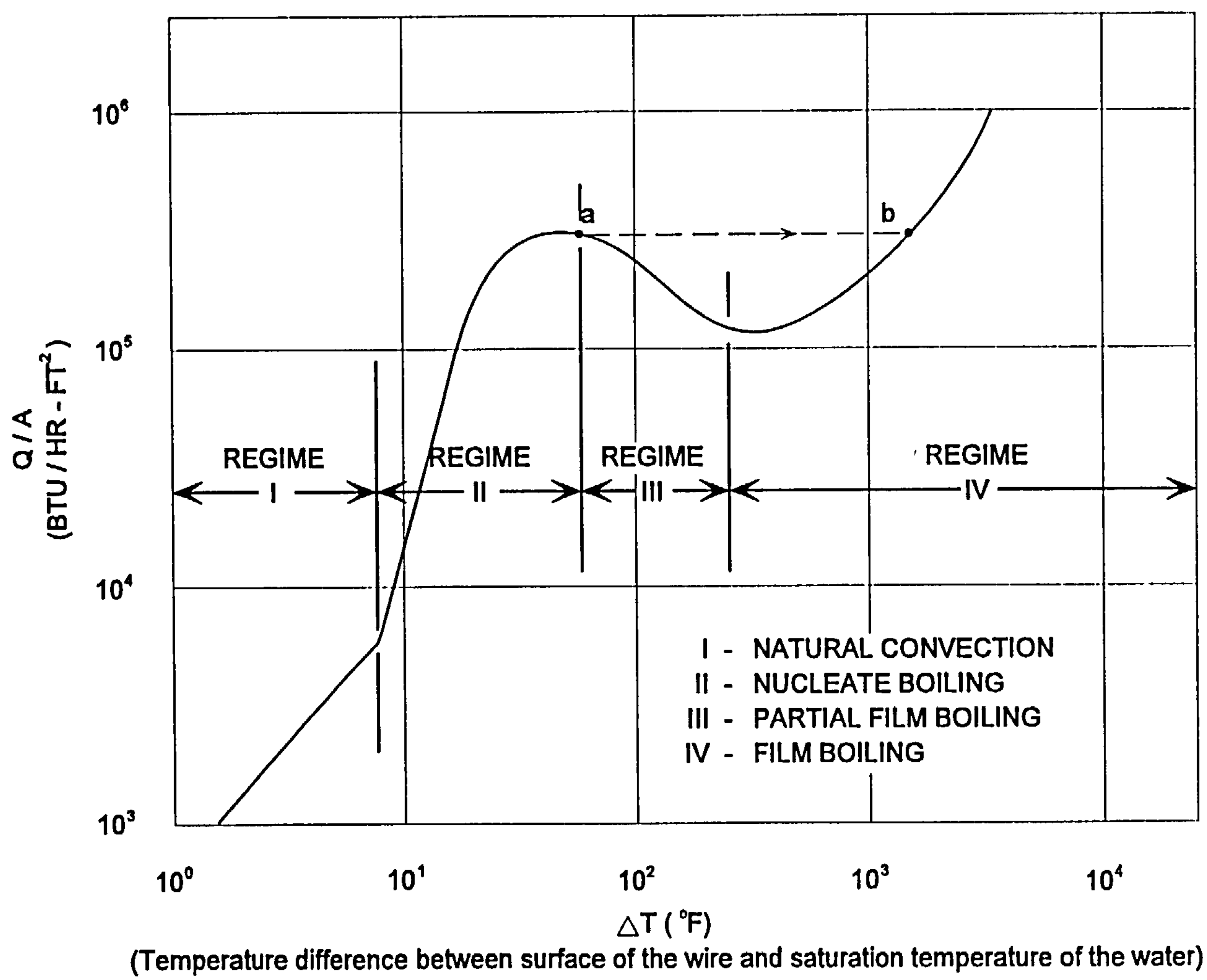


Figure 2.2-2 Heat Rate Versus Temperature Difference

0199-3

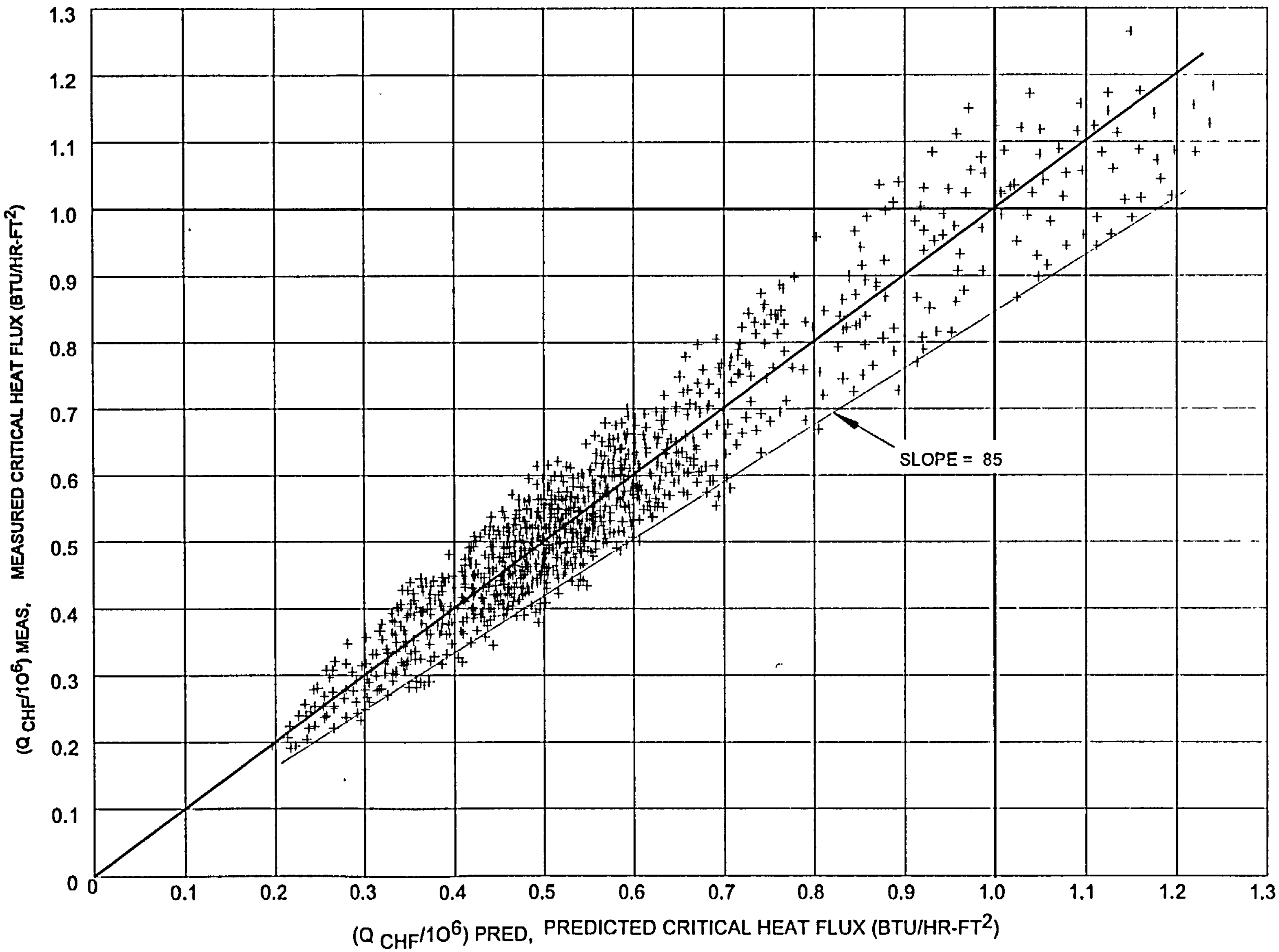


Figure 2.2-3 Measured versus Predicted Critical Heat Flux

0199-3

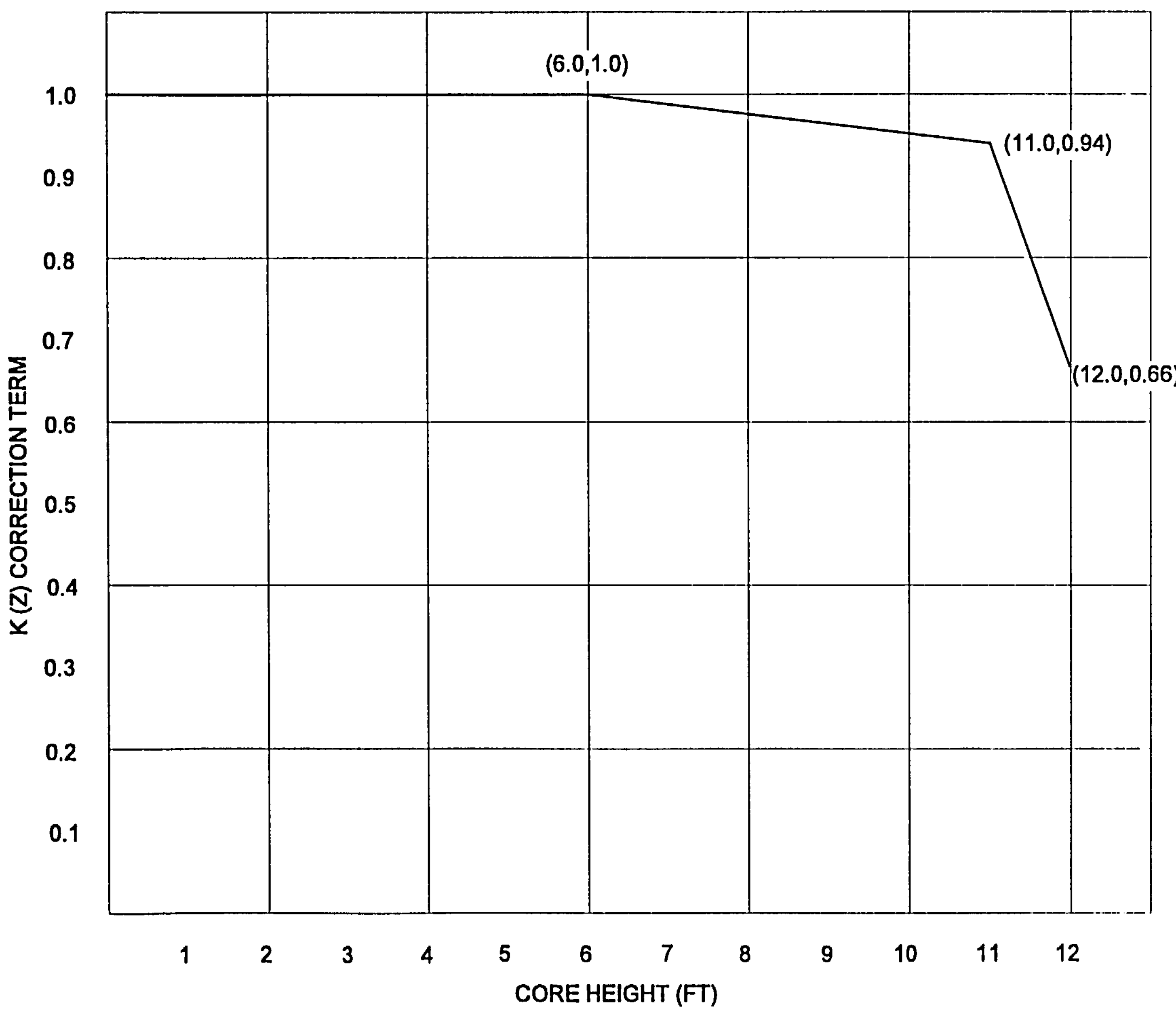


Figure 2.2-4 K (Z) Correction Term

0199-3

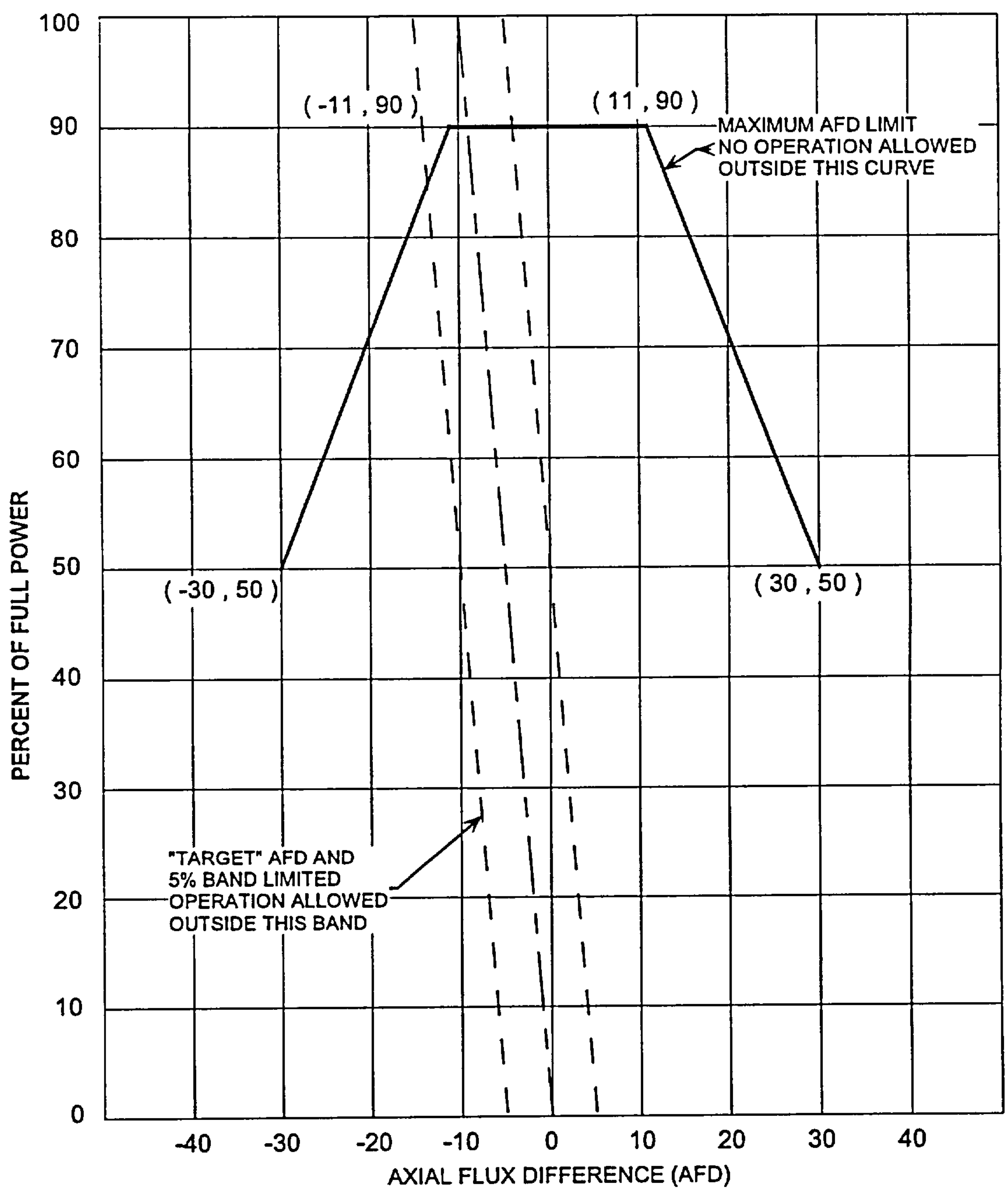


Figure 2.2-5 Axial Flux Difference Limits

Westinghouse Technology Systems Manual

Chapter 3

REACTOR COOLANT SYSTEM

Section

3.0 Reactor Coolant System
3.1 Reactor Vessel and Internals
3.2 Reactor Coolant System

Westinghouse Technology Systems Manual

Section 3.0

Reactor Coolant System

TABLE OF CONTENTS

3.0 REACTOR COOLANT SYSTEM 3.0-1

LIST OF FIGURES

3.0-1 . . . . Reactor Coolant System

3.0 REACTOR COOLANT SYSTEM

Introduction

The Reactor Coolant System (RCS) as shown in Figure 3.0-1, consists of the reactor vessel, which contains the nuclear fuel, and four parallel heat transfer loops. Each loop contains a steam generator, a reactor coolant pump, associated piping and valves, and instrumentation for both control and protection. In addition, the system includes a pressurizer, a pressurizer relief tank and a number of penetrations for connection of auxiliary systems and components necessary for normal or accident operations of the Reactor Coolant System. All RCS components are located inside the containment building.

The reactor vessel as discussed in Section 3.1 is part of the Reactor Coolant System pressure boundary and is capable of accommodating the temperatures and pressures associated with operational transients. The reactor vessel supports the reactor core and control rod drive mechanisms.

The Reactor Coolant System as discussed in Section 3.2, provides sufficient heat transfer capability to transfer the heat produced both during power operations and when the reactor is shutdown. In addition, heat may be transferred to the steam and power conversion system during the first phase of cooldown.

The system heat removal capability under power operations and normal operational transients, including the transition from forced to natural circulation, assures no fuel damage within the operating limits permitted by the reactor control and protection systems.

This system provides the light water used as the neutron moderator and reflector and as a solvent for chemical shim control. The reactor coolant maintains the homogeneity of the soluble neutron poison concentration and controls the rate of change of coolant temperature such that uncontrolled reactivity changes do not occur.

The pressurizer maintains the reactor coolant system pressure during various modes of operation, and is designed to limit pressure transients. During an increase or decrease in plant load, the reactor coolant volumetric changes are accommodated via the surge line to the pressurizer.

Reactor coolant pumps provide the reactor core with sufficient coolant flow to remove the heat that is being generated as a result of the fission process. This coolant then flows to the steam generators and transfers this heat to the secondary water, thereby generating steam to be used by the turbine generator.

Steam generators are provided to supply high quality steam to the turbine. The tube and tube sheet boundaries are designed to prevent the transfer of reactor coolant system activity to the secondary system. The layout of the RCS, with the heat sink (steam generators) located above the heat source, assures natural circulation capability following a loss of forced flow.

The RCS serves as a boundary for containing the coolant under operating temperature and pressure conditions and for limiting leakage to the containment atmosphere. The RCS piping contains demineralized light water which is circulated at a flow rate and temperature consistent with achieving the design reactor core thermal and hydraulic performance.

The RCS pressure boundary is defined as those systems and/or components which are subjected to full reactor coolant system pressure and include the following:

1. Reactor vessel including the control rod drive mechanism housings
2. Reactor coolant side of the steam generators
3. Reactor coolant pumps
4. Pressurizer
5. Pressurizer safety and relief valves
6. Interconnecting piping, valves and fittings between major components
7. Piping, fittings and valves connecting the auxiliary or support systems up to and including the second isolation valve (from the high pressure side) in each line.

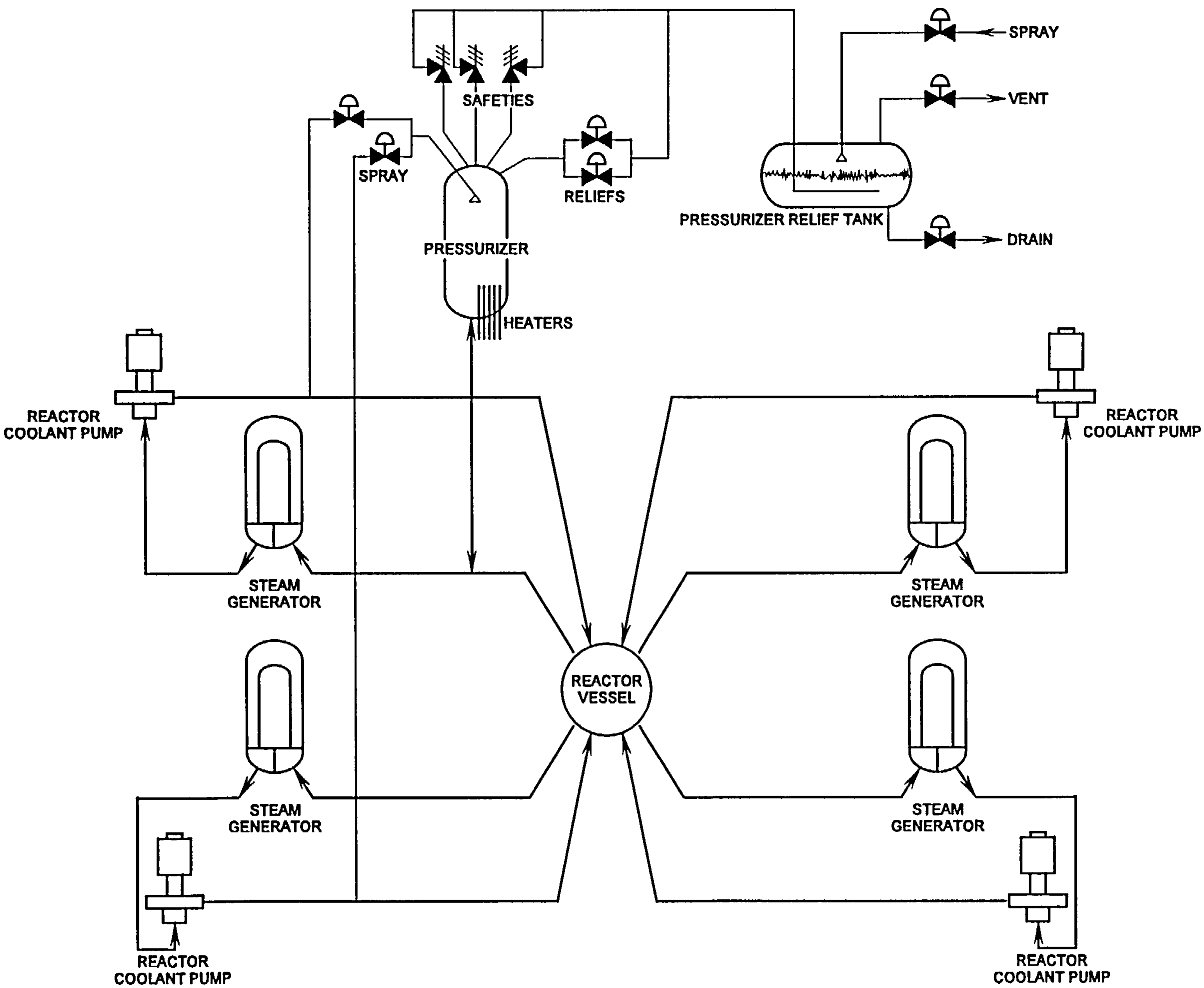


Figure 3.0-1 Reactor Coolant System

Westinghouse Technology Systems Manual

Section 3.1

Reactor Vessel and Internals

TABLE OF CONTENTS

3.1 REACTOR VESSEL AND INTERNALS 3.1-1

3.1.1 Introduction 3.1-1

3.1.2 System Description 3.1-1

3.1.3 Component Description 3.1-2
3.1.3.1 Reactor Vessel 3.1-2
3.1.3.2 Reactor Internals 3.1-5
3.1.3.3 Reactor Core Components 3.1-9
3.1.3.4 Control Rod Drive Mechanisms (CRDM) 3.1-17

3.1.4 System Interrelationships 3.1-19
3.1.4.1 Reactivity Control 3.1-19
3.1.4.2 Vessel Flowpath 3.1-19

3.1.5 Summary 3.1-22

LIST OF TABLES

3.1-1 Reactor Vessel Design Parameters 3.1-23
3.1-2 Fuel Assembly Design Parameters 3.1-24
3.1-3 Fuel Rod Design Parameters 3.1-24
3.1-4 RCCA Design Parameters 3.1-25
3.1-5 BPRA Design Parameters 3.1-25

LIST OF FIGURES

3.1-1 .... Reactor Vessel Cutaway
3.1-2 .... Reactor Vessel Internals
3.1-3 .... Reactor Vessel Cross Section
3.1-4 .... Core Barrel Flange
3.1-5 .... Core Loading Arrangement
3.1-6 .... Reactor Vessel Construction
3.1-7 .... Reactor Vessel Seal
3.1-8 .... Reactor Vessel Head Venting System
3.1-9 .... Reactor Vessel Supports
3.1-10 .... Lower Core Support Structure
3.1-11 .... Instrument Guide and Secondary Core Support Assembly
3.1-12 .... Upper Core Support Structure

LIST OF FIGURES
(Continued)

3.1-13 .... Control Rod Guide Tube Assembly
3.1-14 .... Upper Core Support Plate
3.1-15 .... In-Core Instrumentation
3.1-16 .... Fuel Assembly and RCCA Cutaway
3.1-17 .... 17 X 17 Fuel Assembly Cross Section
3.1-18 .... 17 X 17 Fuel Assembly Outline
3.1-19 .... Upper Fuel Assembly and RCCA Spider
3.1-20 .... Fuel Rod
3.1-21 .... Spring Grid Clip Assembly
3.1-22 .... Spring Grid Assembly
3.1-23 .... Rod Cluster Control Assembly
3.1-24 .... RCCA Drive Shaft
3.1-25 .... Full Length Rod with Interfacing Components
3.1-26 .... Burnable Poison Rod Assembly
3.1-27 .... Burnable Poison Rod
3.1-28 .... Wet Annular Burnable Absorber
3.1-29 .... Primary Source Assembly
3.1-30 .... Secondary Source Assembly
3.1-31 .... Thimble Plug Assembly
3.1-32 .... Control Rod Drive Mechanism
3.1-33 .... Control Rod Drive Mechanism Assembly
3.1-34 .... Core Flow Paths

3.1 REACTOR VESSEL AND INTERNALS

Learning Objectives:

1. State the purpose of the following major reactor vessel and core components:

   a. Internals support ledge
   b. Neutron shield pad assembly
   c. Secondary support assembly
   d. Internals packages
   e. Neutron sources
   f. Burnable poisons
   g. Thimble plug assemblies
   h. Irradiation specimens

2. Describe the flow path of reactor coolant from the inlet nozzles to the outlet nozzles of the reactor vessel.

3. List the core bypass flow paths.

4. Describe the physical arrangement of the following assemblies including the purposes of the component parts listed:

   a. Fuel assembly
      - fuel rods
      - spring clip grid assembly
      - guide thimbles
      - top and bottom nozzles
   b. Control rod assembly
      - rodlets
      - spider
      - hub
      - drive shaft
   c. Rod drive mechanism
      - magnetic coils
      - gripper latches
      - pressure boundary

5. Describe the reactor vessel head seal arrangement.

6. Describe how the reactor vessel is supported.

3.1.1 Introduction

The reactor vessel and its internals contain the heat source for the nuclear steam supply system in the form of the fuel assemblies in the core area. The cladding of the fuel assemblies provide the first barrier to the release of fission products to the environment. The fuel assemblies are supported and held in alignment by the internals packages within the reactor vessel. Additionally, the internals packages provide flow paths for the coolant to remove the heat from the fuel and distribute it to the coolant loops for circulation.

3.1.2 System Description

A simplified diagram of the reactor vessel and internals is shown on Figure 3.1-1. More detailed cutaway and cross-sectional diagrams can be found on Figures 3.1-2, 3.1-3, and 3.1-4, respectively, and are used for the following descriptions.

The reactor vessel is cylindrical, with a welded hemispherical bottom head and a removable, flanged and gasketed, hemispherical upper head. The vessel contains the reactor core, core support structures, control rods and other components directly associated with the core.

The vessel inlet and outlet nozzles are located in a horizontal plane just below the reactor vessel flange but above the top of the core. Coolant enters the vessel through the inlet nozzles and flows down the core barrel-vessel wall annulus, turns at the bottom and flows up through the core to the outlet nozzles.

The reactor internals are comprised of the upper support structure, the lower core support structure, and the incore instrumentation support structure. They are designed to support, align. and guide the core components, direct the coolant flow to and from the core, and to support and guide the incore instrumentation. The fuel assemblies are arranged in a roughly circular cross-sectional pattern.

The fuel is in the form of slightly enriched uranium dioxide ceramic pellets. The pellets are stacked within zircaloy-4 tubular cladding which is plugged and seal welded at the ends to encapsulate the fuel. The fuel rods are internally pressurized with helium during fabrication. Heat generated by the fuel is removed by demineralized light water which flows upward through the fuel assemblies and acts as both moderator and coolant.

The fuel assembly loading arrangement for an operating cycle is shown on Figure 3.1-5. Refueling is typically accomplished by removing part of the core and replacing it with new fuel. The remaining parts of the core are shuffled. The exact arrangement of the core will depend upon nuclear engineering calculations and vessel embrittlement concerns. For example, to reduce the neutron flux on the vessel, the older fuel can be placed on the outside of the core and the new fuel loaded in the center.

Movable neutron absorbing control rods, designated as Rod Cluster Control Assemblies (RCCAs), are provided to accomplish large rapid reactivity additions for reactor control or shutdown purposes. Each RCCA is identical and consists of a number of individual absorber rods attached to a common spider and hub assembly. These absorber rods fit within hollow guide thimbles in the fuel assemblies.

As shown in Figure 3.1-1 and 3.1-2, the fuel assemblies are positioned and supported vertically in the core between the upper and lower core plates. These plates are manufactured with pins which fit into holes in the fuel assembly top and bottom nozzles to provide lateral support. The pins maintain the fuel assembly alignment which ensures free movement of the RCCAs in the fuel assembly without binding or restriction between the rods and their guide surfaces.

3.1.3 Component Description

3.1.3.1 Reactor Vessel

The reactor vessel cylinder is made of welded rolled plate and ring forgings with a welded hemispherical bottom and a removable, flanged and gasketed, hemispherical upper head. The vessel is designed to provide the smallest and most economical volume required to contain the core, core support structures, control rods, and flow directing members of the internals package. The electro-mechanical control rod drive mechanisms and incore temperature instrumentation supports are attached to the reactor vessel head. The bottom of the vessel contains penetrations for the incore nuclear instrumentation. All penetrations are welded to their respective head to minimize coolant leakage.

Inlet and outlet nozzles are located in a horizontal plane below the reactor vessel flange but above the top of the fuel assemblies. Coolant enters the reactor vessel through the inlet nozzles and flows down the annulus between the vessel wall and the core barrel of the core support structure, then turns at the bottom and flows up through the core to the outlet nozzles. All reactor vessel internals are supported from the internals support ledge which is machined into the reactor vessel flange. Reactor vessel design parameters are listed in Table 3.1-1.

Reactor Vessel Construction

The reactor vessel and head are constructed of a manganese molybdenum alloy steel with all surfaces in contact with the reactor coolant clad with weld-deposited stainless steel for corrosion resistance. The method of construction is a number of ring forgings and rolled plates welded together to form the large reactor vessel as shown on Figure 3.1-6.

Samples of reactor vessel and weld materials are provided to evaluate the effect of radiation on the fracture toughness of the reactor vessel. The reactor vessel surveillance program uses a number of specimen capsules located in irradiation specimen guides (guide baskets) attached to the outside of the neutron shield pad (Figure 3.1-3). There are two single holders and two double holders. The specimens are positioned at about core midplane height. Dosimeters are also included to evaluate the level of flux seen by the specimens and vessel wall.

The bottom head of the vessel contains 58 penetration nozzles for connection and entry of the incore nuclear instrumentation. Each nozzle is attached to the inside of the bottom head by a partial penetration weld and extends 12 inches up into the vessel. The nozzles join to the lower internals assembly with a slip fit. Stainless steel conduit is welded to these nozzles below the reactor vessel and is an extension of the Reactor Coolant System (RCS) pressure boundary up to the seal table. At the seal table, a mechanical seal is made to the incore instrument guide thimble which is inserted inside the conduit. The guide thimble is sealed and dry while the space between the conduit and guide thimble is subject to reactor vessel pressure. Details of the incore nuclear instrumentation may be found in Chapter 9.2.

The cylindrical portion of the reactor vessel below the refueling seal ledge and the vessel bottom head is permanently insulated with a metallic reflective-type insulation. This insulation consists of inner and outer sheets of stainless steel spaced three (3) inches apart with multiple layers of stainless steel and air between the sheets acting as the insulating agent. Removable panels of this insulation are provided for the reactor vessel flange and head area and also at the vessel inlet and outlet nozzles to facilitate refueling operations and inservice inspections.

Reactor Vessel Flange

The reactor vessel flange is a ring forging that is 32.5 inches wide and contains 54 threaded holes for the studs which hold the vessel head in place. The reactor internals hang from a ledge on the inside surface of the flange. A sealing area is provided on the outside surface of the flange so that a temporary seal may be made between the reactor vessel and the refueling canal liner.

Reactor Vessel Head

The reactor vessel closure head assembly (Figures 3.1-1 and 3.1-2) is a flanged hemispherical component bolted to the vessel by 54 large threaded studs. The head flange is sealed to the vessel flange by two concentric self-energizing, metallic O-rings which fit into grooves machined in both flanges (Figure 3.1-7). The O-rings are attached to the head by clips to ensure proper alignment during installation. Seal leakage is detected by means of two leak off connections: one between the inner and outer O-rings (normal alignment); and one outside the outer O-ring (must be manually aligned). Leakage is detected by a leak off line temperature detector which will alarm in the control room. The leakage is directed to the Reactor Coolant Drain Tank (RCDT) via check valves which require two (2) psid to unseat. The RCDT high pressure setpoint of six (6) psig ensures that the level in the leakoff line does not rise to the point where there could be flow out of the vessel flange.

The reactor vessel closure head studs are loosened and retorqued as required using hydraulic stud tensioners. These devices (normally three (3) are used simultaneously to torque the head evenly) grip the stud which is threaded into the reactor vessel flange and stretch it. Then the nut is tightened and the tension released producing a correctly torqued stud. The normal procedure is to torque the head in two stages to eliminate warping. Any time the head is removed, new O-rings are installed to ensure a good seal. Proper alignment of the head during installation is accomplished by the use of long guide studs threaded into several of the holes in the vessel flange. These guides are then aligned with the proper holes in the closure head flange as the head is lowered into place during reactor assembly. When the head is tightened, the flange holds the reactor internals in place.

Reactor vessel head penetrations include:

1. Control Rod Drive Mechanism (CRDM) adapters: These penetrations are attached by partial penetration welds to the underside of the closure head. The upper end of the adapter contains threads for attachment of a control rod drive mechanism housing. The housing and penetration are then sealed by a welded flexible canopy seal. A long closed tube called the rod travel housing is attached to the top of the control rod drive mechanism housing to accept the control rod drive shaft as control rods are withdrawn. The rod travel housing is, therefore, an extension of the reactor vessel pressure boundary and is filled with high pressure reactor coolant during normal operation. Vent valves are installed at the top of each rod travel housing but are not normally used due to possible leakage problems.
2. Reactor vessel head vent: This connection is a 3/4 inch outside diameter inconel tube which passes through the upper head and is increased in size to one (1) inch in diameter. The penetration is secured to the interior surface of the head with a partial penetration weld. This penetration is capped but does provide the tap for the reactor vessel head venting system.
3. Reactor Vessel Head Venting System (RVHVS): This system can be used to remove non-condensible gases or steam from the reactor vessel head under certain conditions when these gases could disrupt natural circulation core cooling (Figure 3.1-8). This system was installed as a result of the accident at Three Mile Island. When the RVHVS is operating, flow from either vent line will enter the containment atmosphere. An orifice is installed to limit the flow of hydrogen from the reactor coolant system to permit a reasonable venting period without exceeding containment atmosphere combustible limits. In addition, the orifice is designed to limit flow to within the capacity of one centrifugal charging pump in the event of a ruptured vent line or an inadvertent opening of the vent valves. The solenoid valves in each vent line are powered from redundant vital buses. These valves are normally closed and are energized to open. They fail closed on a loss of electrical power. The RVHVS is operated from the main control board. There are no interlocks associated with this system, and all venting operations require manual operator action. In addition to the solenoid valves, there is a manual globe valve in the common piping upstream of each branch vent line. This valve is open during normal plant operations but may be closed during refueling or system maintenance.

Reactor Vessel Supports

Generally all RCS supports are designed to permit unrestrained thermal growth but restrain vertical, lateral, and rotational movement resulting from seismic and pipe break loadings. The reactor vessel is supported by steel pads integral with four of the coolant nozzles as shown in Figure 3.1-9.

These pads rest upon steel base plates on a support structure mounted to the concrete foundation wall. Thermal expansion and contraction of the vessel are accommodated by sliding surfaces between the support pads and the base plates. Side stops on these plates keep the vessel centered and resist lateral loads, including all pipe loads. The support shoes and support structure are cooled by the reactor cavity cooling system.

3.1.3.2 Reactor Internals

The reactor internals consist of the lower core support structure, the upper core support structure and the incore instrumentation support structure. The internals are designed to support, align and guide the core components, direct the flow to and from core components and guide and support the incore instrumentation.

Any operational accident or seismic loads imposed on the fuel assemblies are transmitted to the upper and lower support structures and ultimately to the reactor vessel flange (Figures 3.1-1 and 3.1-2) or to the lower radial support at the bottom of the lower support structure. The internals also provide a form-fitting-baffle around the fuel assemblies to direct most of the coolant flow through the fuel assemblies.

During initial assembly and refueling the upper core support structure is removed and installed as a unit. The lower core support structure can also be removed as a unit after all fuel has been removed. The lower core support structure is normally removed only for reactor vessel surveillance.

Lower Core Support Structure

The major support assembly of the reactor internals is the lower core support structure, shown in Figure 3.1-10 and 3.1-11. This support structure consists of the core barrel, the core baffles, the lower core plate and support columns, the neutron shield pads, specimen holders and the lower core support plate.

The core barrel supports and contains the fuel assemblies and directs the coolant flow. The core barrel is a cylindrical shell 147.25 inches in diameter and 330.75 inches long. The barrel hangs from a ledge in the reactor vessel flange and is aligned by four flat sided pins which are press fitted into the barrel at 90 degree intervals. Four main coolant outlet nozzles penetrate the shell in the upper region. The lower end of the core barrel is restrained from transverse movement by a radial support system. The radial support system consists of six (6) equally spaced key and keyway joints around the reactor vessel inner wall. At each point an Inconel clevis block is welded to the reactor vessel. An Inconel insert block is bolted to each clevis block forming the keyway. Welded to the lower end of the core barrel are six (6) keys. These keys engage the keylocks when the core barrel is lowered into the reactor vessel. Radial and axial expansions of the core barrel are accommodated but transverse movement of the core barrel is restricted.

Located on the barrel flange are sixteen reactor vessel head cooling flow nozzles that direct 0.13% of the reactor coolant flow upward to cool the head. The flow passes down minor holes in the control rod guide tubes and into the main outlet flow stream.

An internals hold-down spring is positioned in a groove in the core barrel main flange beneath the upper internals main support flange. The spring separates the upper internals and lower internals flanges and maintains a force to resist any uplift of the lower internals.

There are four (4) neutron shield pads attached to the core barrel. The neutron shield pads attenuate fast neutrons that would otherwise excessively irradiate and embrittle the vessel walls. In addition, gamma radiation is attenuated by the shield pads to alleviate thermal stresses due to uneven gamma heating of the vessel. The four pads are 147 inches high, 48 inches wide and 2.7 to 2.8 inches thick. The pads are made of 304 stainless steel and are attached to the core barrel by spacer blocks. The spacer blocks allow cooling flow between the pads and the reactor vessel.

The reactor vessel specimen holders are attached to the outside of the neutron shield pads. The specimens are used to evaluate the irradiation effects on the vessel. One sample is evaluated after the first refueling and subsequent samples are checked at 10, 20, and 30 years.

The lower core support plate is 18 inches thick and is welded to the bottom of the core barrel. The support plate carries the weight of the fuel assemblies and distributes the coolant flow to the fuel assemblies. The secondary support structure, tie plates, instrument guide columns, and support columns hang below the lower core support plate (Figure 3.1-11).

The secondary support structure is in place in the event of a postulated downward vertical displacement of the lower internals (or portion thereof) and the reactor core. As the secondary support base falls against the vessel bottom head, the energy absorption features will stretch and, by strain energy dissipation, will limit the force applied against the vessel head to an acceptable fraction of the yield strength of the pressure vessel. The support structure also ensures that downward displacement of the lower internals is limited so that the upper core plate remains in engagement with the fuel assemblies. This prevents a misalignment ensuring that the controls rods remain capable of inserting.

The secondary core support has four (4) energy absorbing columns that extend up to the lower core support plate and are connected at the bottom by a sole plate. The sole plate is contoured to the shape of the lower head. A 1/2 inch clearance is maintained when the RCS is hot. The energy absorbing devices allow a 3/4 inch displacement for a total displacement of 1-1/4 inches.

The lower core plate is a two (2) inch thick plate located above the lower core support plate through which flow distribution holes for each fuel assembly are machined (four holes per assembly). Additional holes are machined in the plate to allow for passage of the incore instruments and to allow for cooling of the core baffles and formers. Fuel assembly locating pins (two per assembly) are also attached to this plate. The lower core plate is supported by a circumferential ring that is welded to the core barrel. Support columns are also installed between the lower core plate and the lower core support plate of the core barrel in order to provide stiffness and to transmit the load to the lower core support plate.

The boundary of the fuel region of the core is established by the core baffle assembly. This assembly extends from just above the lower core plate to above the fuel assemblies, and guides the coolant flow through the core. Since the fuel assemblies are square and the core barrel is round, gaps would exist around the periphery of the core and a large portion of the cooling water flow would bypass the core. This baffle assembly consists of a number of vertical plates (baffles) and horizontal stiffeners (formers). The formers are bolted to circular grooves in the core barrel and the baffle plates are bolted to the formers to provide alignment and structural rigidity. Small flow holes are drilled in the formers to allow flow mixing to eliminate stagnant areas. This small amount of flow bypasses the fuel region and is considered as a core bypass flow path in the design.

The lower core support structure serves to provide support for the core and guide the flow of reactor coolant. Reactor coolant flow from the vessel inlet nozzles proceeds down the annulus between the core barrel and the vessel wall, flows on both sides of the neutron shield pads and then into a plenum at the bottom of the vessel. It then turns and flows up through the lower core support plate and passes through the lower core plate. The flow holes in the lower core plate are arranged to provide a uniform entrance flow distribution to the core. After passing through the core, the coolant enters the area of the upper support structure and then flows radially to the core barrel outlet nozzles and through the reactor vessel outlet nozzles.

The upper core barrel contains nozzles to direct the outlet flow from the fuel assemblies out of the upper core support structure. These nozzles have a close but non-contacting fit with the reactor vessel outlet nozzles. Since there is no positive seal at this point, a small amount of coolant inlet flow bypasses the core and internals and goes directly to the vessel outlet. This flow is considered a core bypass flow in the design.

Upper Core Support Structure (Upper Internals)

The upper core support structure and its components are shown in Figures 3.1-12, 3.1-13, and 3.1-14. The upper core support structure provides structural support for the fuel assemblies, RCCAs, and incore instrumentation. The structure consist of an upper support assembly, upper support columns, RCCA guide columns, thermocouple columns and the upper core plate.

The upper core support plate is five (5) inches thick and 172 inches in diameter. The support plate rests on the lower internals holddown spring and transmits the upper internals assembly weight to the hold down spring, and ultimately to the reactor vessel flange ledge. Holes machined in the upper support plate accommodate the RCCAs, the five (5) thermocouple columns, and some covered spare penetrations.

The 48 upper support columns establish the spacing between the upper support assembly and the upper core plate and are fastened at top and bottom to these plates. These support columns serve to transmit the weight of the upper core plate and the spring force of the fuel assembly top nozzles to the upper core support plate.

The upper core plate is the lowest major component of the upper internals assembly. The 147.25 inch diameter, three (3) inch thick plate serves to align and locate the fuel assemblies. The plate transmits the uplift forces of core flow and spring forces (of fuel assembly top nozzles) to the support columns and hence to the upper core support plate. The upper core plate also has machined holes to accommodate the same equipment as the upper support plate.

Control rod guide tube assemblies (Figures 3.1-12 and 3.1-13) shield, guide, and support the RCCAs. They are fastened to the upper support plate and the upper core plate for proper orientation and support. Additional vertical support for the absorber rods is provided by the control rod guide tube extension which is attached to the upper support plate.

The upper core support structure, which is removed as a unit during refueling operation, is oriented properly with respect to the lower core support structure by the slots in the upper core plate (Figure 3.1-14) which engage flat sided upper core plate alignment pins welded into the core barrel. At an elevation in the core barrel where the upper core plate is positioned, the flat-sided pins are located at angular positions of 90° from each other. As the upper core support structure is lowered into the lower internals. the slots in the plate engage the flat-sided pins. Lateral displacement of the plate and of the upper support assembly is restricted by this design.

As in the lower core plate, fuel assembly locating pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper core support structure is lowered into place. Proper alignment of the lower core support structure, the upper core support structure, the fuel assemblies and control rods are thereby assured by this system of locating pins and guidance arrangement.

Vertical loads from weight, earthquake acceleration, hydraulic loads and fuel assembly pre-load are transmitted through the upper core plate via the support columns to the upper support plate and then to the reactor vessel flange. Transverse loads from coolant cross flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the upper support and upper core plate. The upper support plate is particularly stiff to minimize deflection.

In-Core Instrumentation Support Structure

The incore instrumentation support structure (Figure 3.1-15), consists of an upper system to convey and support incore temperature monitors (thermocouples) penetrating the vessel through the head and a lower system to convey and support incore nuclear instrumentation flux thimbles penetrating the vessel through the bottom.

There are 65 thermocouples attached to the upper core plate at selected fuel assembly outlets. The leads from these thermocouples are enclosed in sealed stainless steel tubes called thermocouple conduits. The conduits are routed along and supported by the support columns of the upper support structure. The thermocouple conduits are grouped and routed inside five (5) instrumentation port columns which penetrate the reactor vessel head.

A mechanical sealing device is provided at the head penetration to prevent leakage during operation, while allowing disconnection of the thermocouple leads when the head must be removed for refueling. Details of the incore temperature monitoring system may be found in Chapter 9.2.

The lower system is provided to align and support the guide thimbles for the incore nuclear instrumentation system. These guide thimbles are dry, sealed, stainless steel tubes which are inserted into 58 fuel assemblies. A movable miniature fission chamber can be inserted into this dry thimble during reactor power operations, for monitoring of core power distribution.

To each of the 58 penetrations in the reactor vessel bottom head there is welded a stainless steel conduit of larger diameter than the incore guide thimble. These conduits extend from the bottom of the reactor vessel, through a concrete shield area (this opening is also used for access under the reactor vessel when shutdown) to a seal table located above the height of the core.

The closed, dry guide thimble is then inserted through the conduit and extended into the fuel assembly instrumentation thimble (Figure 3.1-17). The dry guide thimbles are closed at the fuel assembly end and serve as a pressure boundary during operation. Since the conduit welded to the reactor vessel is open to vessel pressure, the space between the guide thimble and conduit is subjected to full reactor vessel pressure. Mechanical seals between the guide thimble and conduit are installed at the seal table.

During normal operation, the dry guide thimbles are fixed inside the conduit and the movable detectors are moved in and out of the core through the dry guide thimbles. For refueling operations, the mechanical seals are broken and the dry guide thimbles are mechanically withdrawn from the fuel assembly instrumentation thimbles to allow fuel movement. This can produce high radiation levels under the reactor vessel during refueling since these dry guide thimbles, which have been in the fuel during operation, are now in the conduit outside the reactor vessel. Since these dry guide thimbles are of small diameter, flexible, and have little structural strength, they must be supported and aligned constantly from the penetration to their entrance into the fuel assembly.

Figures 3.1-2 and 3.1-10 show components called instrumentation thimble guides. Either these devices or the hollow lower core support columns will slip over the vessel penetration and enclose the guide thimble to protect it from flow induced vibration. The instrumentation thimble guides are tubes with cross-shaped (cruciform) blades which are inserted through flow holes in the lower core support thus providing alignment and support while not restricting flow.

The incore instrumentation support structure is designed for adequate support of instrumentation during reactor operation and is rugged enough to resist damage or distortion under the conditions imposed by handling during the refueling sequence. Additional information may be found in Chapter 9.2.

3.1.3.3 Reactor Core Components

The reactor core components consist of the fuel assemblies and all components which can be inserted into a fuel assembly to affect reactor power, power distribution or flow distribution.

Fuel Assemblies

All fuel assemblies are mechanically identical, open cage assemblies (Figure 3.1-16). Fuel assembly design parameters are listed in Table 3.1-2. Each fuel assembly consists of: two hundred and sixty-four fuel rods, twenty-four guide thimble tubes and one instrumentation guide thimble supported and aligned by grid assemblies and top and bottom nozzles in a 17 x 17 fuel rod array. The instrumentation guide thimble is located in the center position (Figure 3.1-17) and provides a channel for insertion of an incore neutron detector dry guide thimble if the fuel assembly is located in an instrumented core position. The instrumentation guide sheath is open at the bottom and closed at the top to prevent core bypass flow.

The absorber rod guide thimbles provide channels for insertion of either a RCCA, a neutron source assembly, a burnable poison assembly or a thimble plug assembly, depending on the position of the particular fuel assembly in the core. Figure 3.1-16 shows a fuel assembly with a full length rod fully inserted. The fuel rods are loaded into the fuel assembly structure so that there is clearance between the fuel rod ends and the top and bottom nozzles to allow for expansion without causing bowing of the rods. Fuel rods provide no structural function for the fuel assembly. All structural strength is supplied by the top and bottom nozzles, the grid assemblies and the absorber rod guide thimbles.

Each fuel assembly is installed vertically in the reactor vessel and stands upright on the lower core plate, which is fitted with alignment pins to locate and orient the assembly. After all fuel assemblies are set in place, the upper support structure is installed. Alignment pins, built into the upper core plate, engage and locate the upper ends of the fuel assemblies. The upper core plate then bears downward against hold-down springs in the fuel assembly top nozzle Figures 3.1-18 and 3.1-19, to hold the fuel assemblies in place against any flow induced lifting forces while still allowing differential expansion between the internals and the fuel assemblies.

Fuel Rods

Each fuel rod consists of uranium dioxide ceramic pellets which are placed inside a slightly cold worked zircaloy-4 tube. The tube is then plugged and seal welded at the both ends to encapsulate the fuel. Design parameters for the Fuel Rods are listed in Table 3.1-3.

A schematic of the fuel rod is shown in Figure 3.1-20. The fuel pellets are right circular cylinders consisting of slightly enriched uranium-dioxide powder which has been compacted by cold pressing and then sintered to the required density. (Sintering is high-temperature fusing of metal particles). The ends of each pellet are dished slightly to allow greater axial expansion at the center of the pellets.

To avoid over-stressing of the cladding or seal welds, void volume and clearances are provided within the rods to accommodate fission gases released from the fuel, differential thermal expansion between the cladding and the fuel, and fuel density changes during burn-up. Shifting of the fuel within the cladding during handling or shipping prior to core loading is prevented by a stainless steel helical spring which bears on top of the fuel. At assembly the pellets are stacked in the cladding to the required fuel height. The spring is then inserted into the top end of the fuel tube and the end plugs pressed into the ends of the tube and welded.

All fuel rods are internally pressurized with helium during the welding process in order to minimize compressive clad stresses and prevent cladding flattening due to coolant operating pressures. The helium pre-pressurization may be different for each fuel region. Fuel rod pressurization is dependent on the planned fuel burn-up as well as other fuel design parameters and fuel characteristics (particularly densification potential). The fuel rods are designed such that the internal gas pressure will not exceed the nominal system coolant pressure even during anticipated transients, and clad flattening will not occur during the fuel core life.

All fuel rods contain approximately six (6) inches of natural uranium at each end of the rod. This design reduces axial leakage, which is desirable, but increases the heat flux hot channel factor (F_q).

Bottom Nozzle

The bottom nozzle is a box-like structure which serves as the bottom structural element of the fuel assembly and directs the coolant flow to the assembly. The square nozzle is fabricated from type 304 stainless steel and consists of a perforated plate and four angle legs with bearing plates as shown in Figures 3.1-16 and 3.1-18. The legs form a plenum for the inlet coolant flow to the fuel assembly. The holes in the plate are smaller than the fuel rods and prevent a downward displacement of the fuel rods from their fuel assembly. The bottom nozzle is fastened to the fuel assembly control rod guide tubes by weld-locked screws which penetrate through the nozzle and mate with a threaded plug in each guide tube.

Coolant flow through the fuel assembly is directed from the plenum in the bottom nozzle upward through the holes in the plate to the spaces between the fuel rods. The penetrations in the plate are positioned between the rows of the fuel rods.

Axial loads (hold-down) imposed on the fuel assembly and the weight of the fuel assembly are transmitted through the bottom nozzle to the lower core plate. Indexing and positioning of the fuel assembly is by alignment holes in two diagonally opposite bearing plates which mate with locating pins in the lower core plate. Any lateral loads on the fuel assembly are transmitted to the lower core plate through the locating pins.

Top Nozzle

The top nozzle assembly functions as the upper structural element of the fuel assembly. In addition, the top nozzles provide a partial protective housing for the rod cluster control assembly or other inserts. It consists of an adapter plate, enclosure, top plate, hold down springs, clamps, and pads as shown in Figures 3.1-18 and 3.1-19. The springs and bolts are made of inconel 718, whereas other components are made of type 304 stainless steel.

The square adapter plate is provided with round and oblong penetrations to permit the flow of coolant upward through the top nozzle. Other round holes are provided to accept sleeves which are welded to the adapter plate and mechanically attached to the thimble tubes. The holes in the plate are of smaller diameter than the fuel rods and prevent their upward ejection from the fuel assembly. The enclosure is simply a sheet metal shroud between the adapter plate and the top plate. The top plate has a large square hole in the center to accept control rods and the control rod spiders when the rods are inserted. Hold-down springs are mounted on the top plate and are fastened in place by bolts and clamps located at two diagonally opposite corners. On the other two corners are located the holes which accept the alignment pins of the upper core plate.

A serial number is stamped in the top nozzle of each fuel assembly for quality assurance tracking purposes. The assembly can thus be tracked from construction, shipment, core loading and shuffling during refueling. These serial numbers are normally scanned by a television camera located on the refueling crane to verify proper core loading.

Absorber Rod Guide and Instrument Thimbles

The guide thimbles are the major structural member of the fuel assembly Figures 3.1-16 and 3.1-17. They also provide channels for neutron absorber rods, burnable poison rods, neutron source, or thimble plug assemblies. Each RCCA guide thimble is fabricated from zircaloy-4 tubing of two different diameters. The top of the thimble is of a larger diameter to produce less restriction during reactor trip and allow faster rod insertion times.

At the bottom of the thimble, there is a transition to a smaller diameter tube. This produces a dashpot effect and slows the RCCA after a trip. Four small flow holes are located just above the dashpot region to reduce rod drop time, provide a path for water displaced from the dashpot by the RCCA and allow some flow through the guide thimble to cool the control rod (this coolant is considered a design core bypass flow).

The dashpot is closed at the bottom by means of an end plug which is provided with a small flow port to avoid fluid stagnation in the dashpot volume during normal operation. The top end of the guide thimble is fastened to a tubular sleeve by three expansion swages. The sleeve fits into and is welded to the top nozzle adapter plate. The lower end of the guide thimble is fitted with an end plug which is then fastened into the bottom nozzle by a weld-locked hollow screw.

The central instrumentation thimble (also called instrument guide sheath) of each fuel assembly is not rigidly attached to either the top or bottom nozzles, but the thimble is constrained by its seating in counterbores of each nozzle. The thimble internal diameter does not vary. In-core neutron detector guide thimbles are installed through the bottom nozzle's large counterbore into the instrument guide sheath.

Grid Assemblies

The fuel rods are supported laterally at intervals along their length by spring grid clip assemblies (Figures 3.1-21 and 3.1-22), which maintain the lateral spacing between the rods throughout the design life of the assembly. Each fuel rod is afforded lateral support at six contact points within each grid by a combination of support dimples and spring fingers. The magnitude of the grid restraining force on the fuel rod is set high enough to minimize possible fretting, without over-stressing the cladding at the points of contact between the grids and fuel rods. The grid assemblies also allow axial thermal expansion of the fuel rods without imposing restraint sufficient to develop buckling or distortion of the fuel rods.

Two types of grid assemblies are used in each fuel assembly. Both types consist of individual slotted straps interlocked in an "egg-crate" arrangement. The straps contain spring fingers, support dimples, and mixing vanes. One type, used in the high flux region of the fuel assemblies, consists of zircaloy straps permanently joined by welding at their points of intersection. This material is chosen primarily for its low neutron absorption properties. The internal straps include mixing vanes which project into the coolant flow and promote mixing of the coolant.

The other grid type, located at the ends of the fuel assemblies, does not include mixing vanes on the internal straps. The material of these grid assemblies is inconel-718, chosen because of its corrosion resistance and high strength. Joining of the individual straps is achieved by brazing at the points of intersection.

The outside straps on all grids contain mixing vanes which, in addition to their mixing function, aid in guiding the grids and fuel assemblies past projecting surfaces during handling or during loading and unloading of the core.

Handling of Fuel Assemblies

Due to its method of construction (long, square array supported by the long, small diameter RCCA guide thimbles) the fuel assembly exhibits excellent structural strength in the axial direction but cannot accept large mechanical loads in the radial direction. Fuel assemblies are, therefore, handled by grasping the top nozzle with a special tool and moving them while in the vertical position. If it is required to position the assembly horizontally, it must be supported at the grid locations to prevent bowing.

Other Fuel Designs

The description of the fuel assembly above is of the optimized fuel design. There are some other fuel designs that provide some improvements to the basic optimized fuel assembly.

One change is a debris filter bottom nozzle. The examination of leaking fuel over the years has shown that about 75% of the leakers were due to debris-induced fretting. The debris filter bottom nozzle helps mitigate the effects of debris in the coolant system by trapping it at the entrance of the fuel assembly. The bottom nozzle has a pattern of many small holes which allows reactor coolant to pass through while minimizing the passage of debris large enough to cause wear or fretting.

Although the holes are smaller than the earlier design bottom nozzles, the hydraulic performance is the same and there is no difference in coolant flow or heat transfer.

Another change is the addition of intermediate flow mixing grids. These grids are non-structural grids located in the three uppermost spans between the zircaloy mixing vane structural grids. They incorporate a similar mixing vane array with a prime function of mixing in the hottest fuel assembly spans. The increased mixing increases the heat transfer in the upper part of the core. Therefore, the margin to departure from nuclear boiling is increased. The intermediate flow mixing grids are physically shorter than the structural grids, and therefore, the advantages of the mixing are accomplished with a minimal pressure drop.

A change has also been made to the material of choice for the guide thimbles, instrument tube, fuel cladding, and some of the grids. This new material is an improved zirconium alloy which contains reduced tin content and now contains niobium. This new material is more resistant to corrosion but has the strength and performance of zircaloy-4.

The top nozzle of the fuel assembly has also undergone some design changes. The material of construction is now a low cobalt stainless steel, which helps reduce radiation levels due to the activation of cobalt. Also, the top nozzle is manufactured to be removable. This allows the removal of leaking fuel rods from a fuel assembly and then replacing them with other fuel rods, stainless steel rods, or water holes. This allows the burnup of fuel rods that are not leaking, which results in fuel savings for the plant.

One of the driving forces behind these and other fuel design changes is longer operating cycles. The desire to burn as much of the fuel as possible is much less expensive than going to higher enrichment fuel. However, the longer operating cycle would also mean more fission products produced than with the lower burnup fuel. Therefore, changes must be made to allow the fission product gases to be collected inside the cladding without having excessive stresses. The thickness of the top and bottom nozzles has been reduced. This allows the fuel rod to be slightly longer. However, since the fuel length has not changed, the plenum area is now longer. The design of the plenum spring has also been changed to allow more space for the gases.

Full Length Rod Cluster Control Assemblies (RCCA)

The RCCAs (also called control rods) provide a rapid means for reactivity control during both normal operating and accident conditions. RCCA Design Parameters are listed in Table 3.1-4.

The rod cluster control assemblies are divided operationally into two categories: control banks and shutdown banks. The control banks may be inserted or withdrawn to compensate for various reactivity changes during operation of the reactor and can trip (scram) to provide shutdown capability. The shutdown banks are reserved for shutdown use only and are always fully withdrawn from the core when the reactor is critical. They are inserted only when a reactor trip occurs.

Two criteria are employed for design of the RCCAs. First the total reactivity worth must be adequate to meet the nuclear requirements of the reactor. Second, since some of these rods may be partially inserted at power operation, the total power peaking factor should be low enough to ensure that full power capability can be met. The control and shutdown banks provide adequate shutdown margin. Shutdown margin is defined as the amount of reactivity by which the core would be subcritical if all RCCAs are tripped assuming that the highest worth RCCA remains fully withdrawn.

An RCCA consists of a group of individual neutron absorber rods fastened at the top end to a common spider assembly, as illustrated in Figure 3.1-23. A cutaway of a fuel assembly with RCCA inserted is shown on Figure 3.1-16.

The absorber material used in the control rods is silver-indium-cadmium alloy (Ag-In-Cd). The silver-indium-cadmium alloy is in the form of extruded rods which are sealed in stainless steel tubes to prevent the poisoning material from coming in direct contact with the coolant. The stainless steel tubing is sealed at the bottom and the top by welded end plugs. Sufficient clearance is provided to accommodate thermal expansion. The bottom plugs are made bullet-nosed to reduce the hydraulic drag during reactor trip and to guide smoothly into the dashpot section of the fuel assembly guide thimbles. The upper plug is threaded for assembly to the spider and has a reduced end section to make the joint more flexible.

The spider assembly is in the form of a central hub with radial vanes containing cylindrical fingers from which the absorber rods are suspended. Handling detents and detents for connection to the drive rod assembly are machined into the upper end of the hub. A coil spring inside the spider body absorbs the impact energy at the end of a trip insertion. The radial vanes are joined to the hub by welding and brazing, and the fingers are joined to the vanes by brazing. All components of the spider assembly are made from types 304 and 308 stainless steel except for the spring retainer which is 17-4 pH stainless steel and the springs which are inconel 718 alloy.

The absorber rods, also called rodlets, are fastened securely to the spider to assure trouble free service. The rods are first threaded into the spider fingers and then pinned to maintain joint tightness, after which the pins are welded in place. The end plug below the pin position is designed with a reduced section to permit flexing of the rods to correct for small operating or assembly misalignments. The overall length is such that when the assembly is fully withdrawn, the tips of the absorber rods remain engaged in the guide thimbles so that alignment between rods and thimbles is always maintained. Since the rods are long and slender, they are flexible enough to conform to any small misalignments with the guide thimble. A hollow long, grooved drive shaft (Figure 3.1-24) is attached to the spider hub of each RCCA by a split coupling.

When the RCCA is inserted, the drive shaft extends through the upper core support structure, through a reactor head penetration to the control rod drive mechanism. A disconnect rod, used during core disassembly, is located inside this shaft which, when fully inserted into the drive shaft, will expand the split coupling and lock it into place in the hub. Figure 3.1-24 shows the relative positions of the RCCA and associated equipment in the reactor. Since the poison rods are long and slender, they must be supported and aligned as they are withdrawn from the fuel assembly. The control rod guide tubes of the upper core support structure shown in Figure 3.1-2, and 3.1-13 perform this function.

Burnable Poisons

Burnable poisons are added to the core to provide a fixed discrete poison when needed due to nuclear considerations, such as controlling power peaking and moderator temperature coefficient. These discrete poisons may be in the form or burnable poison rod assemblies and/or a poison coating on the fuel pellets.

Figure 3.1-26 shows a burnable poison rod assembly (BPRA). BPRA design parameters are listed in Table 3.1-5.

The poison rods of the burnable poison assemblies are suspended from a flat spider plate and are inserted into the RCCA guide thimbles of selected fuel assemblies at selected unrodded locations. The spider plate fits within the fuel assembly top nozzle and rests on the top adapter plate. A T-bar and spring assembly is attached to the spider. As the upper core support structure is installed, the upper core plate contacts the T-bar and compresses the spring holding the burnable poison assembly in place.

There are two designs of the poison rods which could be used in the burnable poison assemblies. The poison rods in the older design, Figure 3.1-27, consist of borosilicate glass tubes contained within type 304 stainless steel tubular cladding which is plugged and seal welded at the ends to encapsulate the glass. The glass is also supported along the length of its inside diameter by a thin wall 304 stainless steel tubular inner liner. The top end of the liner is open to permit the diffused helium to pass into the void volume and the liner extends beyond the glass. The liner is flanged at the bottom end to maintain the position of the liner with the glass.

The rods are designed in accordance with the standard fuel rod design criteria; that is, the cladding is free standing at reactor operating pressures and temperatures and sufficient cold void volume is provided within the rods to limit internal pressures to less than the reactor operating pressure assuming total release of all helium generated in the glass as a result of the B10 (n,a) reaction. The large void volume required for the helium is obtained through the use of glass in tubular form which provides a central void along the length of the rods.

Based on available data on properties of borosilicate glass and on nuclear and thermal calculations for the rods, gross swelling or cracking of the glass tubing is not expected during operation. Some minor creep of the glass at the hot spot on the inner surface of the tube is expected to occur but continues only until the glass comes into contact with the inner liner.

The inner liner is provided to maintain the central void along the length of the glass and to prevent the glass from slumping or creeping into the void as a result of softening at the hot spot. The wall thickness of the inner liner is sized to provide adequate support in the event of slumping but to collapse locally before rupture of the exterior cladding if large volume changes due to swelling or cracking should occur. The top end of the inner liner is open to receive the helium which diffuses out of the glass.

The second burnable poison design is called a wet annular burnable absorber, Figure 3.1-28. This design uses aluminum oxide/boron carbide as the absorber. The poison is in the form of a stack of hollow pellets wrapped inside and out with stainless steel cladding and pressurized with helium. Water is allowed to flow in the central passage. This allows for an increase in moderation in the center of the poison rod, which allows for better absorption of neutrons and a more complete burnout of the poison material.

Zirconium boride is used as the poison that is coated on the fuel pellets. Fuel assemblies which have coated fuel pellets are called integral fuel burnable absorbers (IFBA). One of the advantages of using integral absorbers is that burnable absorbers can be placed in fuel assemblies that also have control rods inserted. The flux shape can be more controlled by the placement of the absorber material and by controlling the number of fuel rods with absorber material as dictated by individual plant needs.

Neutron Source Assemblies

Prior to initial reactor operation and after long shutdown periods, core neutron level may be too low to be detected by the installed excore nuclear instrumentation system. To insure adequate indication for the operator during long term reactor shutdown and during reactor startup, neutron source assemblies are installed in the core. These sources will, in conjunction with subcritical multiplication, produce a neutron level high enough to be monitored by the source range nuclear instrumentation channels (2 cps required). It would be unacceptable for the operator to achieve criticality with no indication of reactor power or rate of change of power. The source is also necessary to provide a measurable neutron flux during fuel loading. A source is not required to enable the reactor to physically achieve criticality.

There are two types of source assemblies employed in the Westinghouse design: primary source assemblies (Figure 3.1-29) and secondary source assemblies (Figure 3.1-30).

The primary source spontaneously produces neutrons for indication for initial reactor start-up while the secondary source (which is activated only after being exposed to neutron bombardment) provides neutrons for subsequent start-ups.

Both types of source assemblies are constructed as burnable poison assemblies with several of the poison rods replaced by source rods. The primary source rods for the Trojan plant were capsules of californium. The decay mechanism of californium is by spontaneous fission which produces neutrons directly.

The two (2) secondary sources contain a symmetrical grouping of four (4) secondary source rods and twenty thimble plugs. The secondary sources are constructed of antimony beryllium (SbBe), which is activated only after exposure to a neutron flux during reactor operation. The secondary source produces neutrons by the following reactions:

$$_0n^1 + {}_{51}Sb^{123} \quad {}_{51}Sb^{124*} \quad {}_{52}Te^{124}$$

$$+ \quad + \qquad \text{(60 days)}$$

and:

$$+ {}_4Be^9 \quad {}_4Be^8 + {}_0n^1$$

Neutron source assemblies are located at diametrically opposite sides of the core at locations close to the source range nuclear instrumentation detectors. These assemblies are inserted into the rod cluster control guide thimbles in fuel assemblies at selected unrodded locations.

Thimble Plugging Assemblies (Thimble Plugs)

In order to limit core bypass flow through the absorber rod guide thimbles in the fuel assemblies not containing control rods, source assemblies, or burnable poison rods, the fuel assemblies at these locations are fitted with thimble plugging assemblies (Figure 3.1-31).

The thimble plugs consist of a flat spider plate with short rods suspended from the bottom surface, and a spring pack assembly and mixing device attached to the top surface. When installed in the core, the plugging devices fit within the fuel assembly top nozzles and rest on the adaptor plate.

The short rods project into the upper ends of the guide thimbles to reduce the bypass flow area. The spring pack is compressed by the upper core plate contacting the T-bar when the upper internals package is lowered into place.

When the core is fully assembled, all fuel assembly absorber rod guide tubes will have either a control rod, burnable poison rod, source rod, or thimble plug inserted into them. All components in the plugging device, except for the springs, are constructed from Type 304 stainless steel. The springs (one per plugging device) are wound from an age hardened nickel base alloy to obtain higher strength.

3.1.3.4 Control Rod Drive Mechanisms (CRDM)

The control rod drive mechanisms are electro-mechanical devices (magnetic jacks) used to position the rod cluster control assemblies (control rods) in the reactor core. The CRDM is capable of withdrawing or inserting the control rod in discrete increments (steps) or holding it at a constant position. Tripping (scramming) is accomplished by simply de-energizing the mechanisms and allowing the control rods to fall by gravity into the core. Each RCCA is operated by its own individual CRDM.

The complete drive mechanism shown in Figures 3.1-32 and 3.1-33 consists of an internal latch assembly, the drive shaft assembly, the pressure vessel (including the rod travel housing), the operating coil stack, and the individual rod position indication coil stack. Reactor coolant at full system pressure fills the pressure housing and rod travel housing. All moving parts are immersed in reactor coolant. This housing forms the RCS pressure boundary.

Each assembly is an independent unit which can be dismantled or assembled separately. The mechanism pressure housing is threaded onto an adaptor on top of the reactor pressure vessel head and seal welded. The operating drive assembly is connected to the control rod (directly below) by means of a grooved drive shaft. The upper section of the drive shaft is suspended from the working components of the drive mechanism. The drive shaft and control rod remain connected during reactor operation, including tripping of the rods.

Three magnetic coils, which surround the rod drive pressure housing, induce magnetic flux through the pressure housing wall to operate the working components. They move two sets of latches which lift, lower or hold the grooved drive shaft. To move the control rod, the three magnets are turned on and off in a fixed sequence by the solid state full length rod control system (Chapter 8.1). The sequencing of the magnets produces discrete steps (5/8") of rod motion over the 144 inches of normal control rod travel.

The mechanism develops a lifting force approximately two times the static lifting load. Therefore, extra lift capacity is available for overcoming mechanical friction between the moving and the stationary parts. Gravity provides the driving force for rod insertion and the weight of the whole rod assembly is available to overcome any resistance.

The mechanisms are designed to operate in water at 650°F and 2485 psig. The temperature at the mechanism head adaptor will be much less than 650°F because it is located in a region where there is limited flow of water from the reactor core, while the pressure is the same as in the reactor pressure vessel.

Latch Assembly

The latch assembly consists of the components which engage and support the grooved drive shaft during insertion, withdrawal and holding operations. All latch components are located inside the pressure housing.

The primary components are the two latches which engage the drive shaft: the stationary gripper latch; and the movable gripper latch. A number of pole pieces, springs, pivot pins and linkages are provided to support and align the latches during movement or holding operations.

The upper set of latches (movable grippers) are engaged when the movable gripper coil is energized and can move upward 5/8 inch when the lift coil is energized. The lower set of latches (stationary grippers) are engaged when the stationary gripper coil is energized and can move only 1/16 inch axially to allow load transfer between movable and stationary grippers during rod movement. The stationary gripper latches are normally engaged during periods of no rod motion (holding operations). An insertion and withdrawal sequence is discussed in Chapter 8.1.

Pressure Vessel

The pressure vessel consists of the pressure housing and rod travel housing. The pressure housing is the lower portion of the vessel and contains the latch assembly. The rod travel housing is the upper portion of the vessel. It provides space for the drive shaft during its upward movement as the control rod is withdrawn from the core.

Operating Coil Stack

The operating coil stack is an independent unit which is installed on the drive mechanism by sliding it over the outside of the pressure housing. It rests on a pressure housing flange without any mechanical attachment and can be removed and installed while the reactor is pressurized. Forced air cooling from the CRDM cooling fans flows along the outside of the coil stack to maintain a coil casing temperature of approximately 248°F or lower. These fans must be operating whenever RCS temperature is greater than 350°F.

Drive Shaft Assembly

The main function of the drive shaft, as shown in Figure 3.1-24, is to connect the control rod to the control rod drive mechanism. Grooves for engagement and lifting by the latches are located throughout the 144 in. of control rod travel. The grooves are spaced 5/8 inch apart to coincide with the mechanism step length and have 45° angle sides. The drive shaft is attached to the control rod by a split coupling. The coupling engages the grooves in the spider assembly hub.

A 1/4 inch diameter disconnect rod runs down the inside of the hollow drive shaft. It utilizes a locking button at its lower end to expand and lock the coupling to the control rod hub. At its upper end, there is a disconnect assembly. For remote disconnection of the drive shaft assembly from the control rod, a button at the top of the drive rod actuates the connect/disconnect assembly. The drive shaft assembly can be attached and removed from the control rod only when the reactor vessel head is removed.

During refueling, the drive shafts are uncoupled from the control rod spider and are removed as the upper internals (upper core support structure) are removed. The control rods remain in the fuel assemblies.

Individual Rod Position Indication (IRPI) Coil Stack

The IRPI coil stack is installed around the rod travel housing. It detects control rod position by means of cylindrically wound differential transformers which span the normal length of rod travel (144 inches). As the drive shaft moves inside the rod travel housing, it changes the coupling between the primary and secondary coils and produces a signal proportional to control rod height.

A more detailed description of the individual rod position indication system can be found in Chapters 8.3 and 8.4.

Materials of Construction

All components exposed to reactor coolant, such as the pressure vessel, latch assembly and drive rod, are made of or clad with materials which resist the corrosive action of the water and boric acid. Three types of metals are used exclusively: stainless steels, Inconel, and cobalt based alloys. Wherever magnetic flux is carried by parts exposed to the main coolant, 400 series stainless steel is used.

Cobalt based alloys are used for the pins, latch tips, and bearing surfaces. Inconel X is used for the springs of both latch assemblies and 304 stainless steel is used for all pressure containment. Hard chrome plating provides wear surfaces on the sliding parts and prevents galling between mating parts (such as threads) during assembly. Outside of the pressure vessel, where the metals are exposed only to the reactor plant containment environment and cannot contaminate the reactor coolant, carbon and stainless steels are used. Carbon steel, because of its high permeability, is used for flux return paths around the operating coils. It is zinc-plated 0.001 inch thick to prevent corrosion.

3.1.4 System Interrelationships

3.1.4.1 Reactivity Control

Reactivity control is accomplished by control rods and soluble neutron absorber (boric acid) in the moderator. Boric acid concentration is varied to control long-term reactivity changes such as:

1. Fuel depletion and fission product buildup.
2. Cold to hot zero power, reactivity change.
3. Reactivity change produced by intermediate term fission products such as xenon and samarium.
4. Burnable poison depletion.

Control rod position is varied to control more rapid reactivity changes due to:

1. Shutdown
2. Reactivity changes due to coolant temperature changes in the power range.
3. Reactivity changes associated with the power coefficient of reactivity.
4. Reactivity changes due to void formation.

Due to operating restrictions some of the reactivity changes designed to be compensated by control rod motion are now accomplished by boron concentration changes. These are the changes associated with planned power changes during operation.

3.1.4.2 Vessel Flowpath

As shown in Figure 3.1-34, reactor coolant enters the reactor vessel through the four inlet nozzles to impinge against the upper core barrel and flows down the annulus between the core barrel and the vessel wall. In passing the neutron shield pads, some flow goes between the neutron shield pad and the vessel. The flow passage continues between the radial support members into the lower vessel head plenum among the tie plates and instrument guide columns where the flow is directed upward.

The coolant then continues through the lower core plate and into the core to remove the heat generated in the fuel assemblies. Flow continues upward through the fuel assemblies (cross-flow between assemblies is also possible) and exits the core through the upper core plate. The coolant then flows laterally past the support columns and control rod guide tubes of the upper core support structure to the outlet nozzles of the core barrel.

These outlet nozzles have flanges which direct the flow. The flanges do not contact the outlet nozzles but maintain some small clearance. Ninety-three and one half percent (93.5%) of the total coolant flow is available for core heat removal. The remainder (6.5%) bypasses the core. Core bypass flows and their values consist of:

1. Nozzle bypass flow from the reactor vessel inlet directly to the outlet nozzles through the small clearance between the core barrel outlet nozzles and reactor vessel outlet nozzles accounts for 0.68%.
2. RCCA guide thimble bypass flow through the small flow holes at the dashpot section of the guide thimble. This flow provides cooling for any inserted control rods and the burnable poison assemblies and accounts for 3.85%.
3. Core baffle bypass flow between the core barrel and the core baffle through holes machined in the former plates for cooling. This flow accounts for 1.84%. Included in this value is 1.51% actual design flow, 0.23% outer fuel assembly to baffle gap leakage and 0.10% leakage through the plugged barrel flow holes.
4. Head cooling bypass flow through holes drilled in the core barrel flange to corresponding holes in the upper support plate flange to allow some reactor inlet flow into the head area for cooling. This flow passes down holes in the upper guide tubes to rejoin coolant outlet flow. This flow accounts for 0.13% of the core bypass flow.

3.1.5 Summary

The reactor vessel and internals support and align the reactor core and its associated components. Additionally, the vessel and internals provide a flowpath to ensure adequate heat removal capability from the fuel assemblies.

The cylindrical reactor vessel has a removable head with penetrations for the control rod drive mechanisms and incore temperature monitoring instrumentation. Incore nuclear instrumentation enters through penetrations in the bottom of the reactor vessel. The vessel is supported from pads welded to its nozzles.

The lower internals (lower core support structure) are hung from the reactor vessel support ledge and align and contain the fuel assemblies. The core barrel is aligned and supported radially by several keys at its bottom which mate with keyways attached to the reactor vessel wall. Neutron shield pads are attached to the core barrel at core level to reduce radiation damage (primarily neutron embrittlement) to the reactor vessel. Retrievable specimens of reactor vessel materials are attached to the core barrel to monitor radiation induced changes in the reactor vessel. A core baffle surrounds the core to guide most of the coolant flow through the fuel assemblies.

The upper internals are also supported by a flange at the reactor vessel support ledge. Control rod guide tubes support and align the RCCAs when they are withdrawn from the fuel assemblies. The support columns provide structural strength to transfer normal and accident loads to the reactor vessel support ledge and support the incore temperature detection system.

The fuel assemblies are constructed in 17x17 arrays, consisting of long thin zircaloy tubes containing slightly enriched uranium dioxide in the form of ceramic pellets. Structural strength for the fuel assemblies is provided by the control rod guide thimbles, grid straps and top and bottom nozzles. The fuel rods do not provide any structural support for the fuel assembly. The fuel assemblies are aligned by pins in the upper and lower core plate which mate with holes in the top and bottom nozzles. All control rod guide thimbles are filled with either control rods, burnable poison assemblies, source assemblies or thimble plugging assemblies.

Full length control rod drive mechanisms are magnetic jack assemblies which move control rods in discrete steps. Tripping is accomplished by de-energizing the mechanism. The control rods are designed to respond to fast reactivity changes while slow changes such as fuel burnup are compensated by boron concentration changes.

Nearly all reactor coolant flow is available for core cooling with a few percent bypassing the core at the core barrel to vessel gap, head cooling flow holes, control rod guide thimble cooling and core baffle flowpaths.

Table 3.1-1
Reactor Vessel Design Parameters

Hydrostatic test pressure	3,107 psig
Design pressure	2,485 psig
Design temperature	650°F
Reactor vessel height	525.8"
Number of closure head studs	54
Closure head stud - diameter	7.0"
Vessel flange - width	32.5"
Vessel shell - inside diameter	173"
Inlet Nozzle - inside diameter	27.2"
Outlet Nozzle - inside diameter	28.8"
Closure head - thickness	6.5"
Beltline - thickness	8.5"
Lower hemispherical head - thickness	5.5"
Vessel cladding - thickness	0.156"
Stainless steel insulation - thickness	3.0"

Table 3.1-2
Fuel Assembly Design Parameters

Number of fuel assemblies	193
Number of fuel rods per assembly	264
Number of guide thimbles per assembly	24
Number of grid straps per assembly	8
Rod array	17 x 17
Fuel (UO_2) - weight	222,739 lbs
Cladding (Zr_4)- weight	50,913lbs
Grid straps (Inconel-718) - weight	2,324
Guide thimble - outside diameter	0.482"
Guide thimble - inside diameter	0.450"

Table 3.1-3
Fuel Rod Design Parameters

Number of fuel rods (total)	50,952
Fuel rod - outside diameter	0.374"
Gap - diameter	0.0065"
Fuel rod cladding - thickness	0.0225"
Fuel rod cladding material	Zr_4
Fuel pellet - diameter	0.3225"
Fuel pellet - length	0.530"
Fuel pellet - density	95% theoretical
Fuel pellet material	UO_2

Table 3.1-4
RCCA Design Parameters

Neutron absorber	Ag-In-Cd
Neutron absorber composition	80%-15%-5%
Number of RCCAs	53
Number of rodlets per RCCA	24
RCCA rodlet - cladding material	304 SS
Cladding - thickness	0.0185"
RCCA - weight	157 lbs

Table 3.1-5
BPRA Design Parameters

Number of burnable poison rods	1,440
Burnable poison rod material	Borosilicate glass
BPRA rodlet - outside diameter	0.381"
BPRA rodlet - cladding material	304 SS
BPRA rodlet - boron loading (B_2O_2)	12.5% weight
Reactivity worth - hot	7,630 pcm
Reactivity worth - cold	5,500 pcm

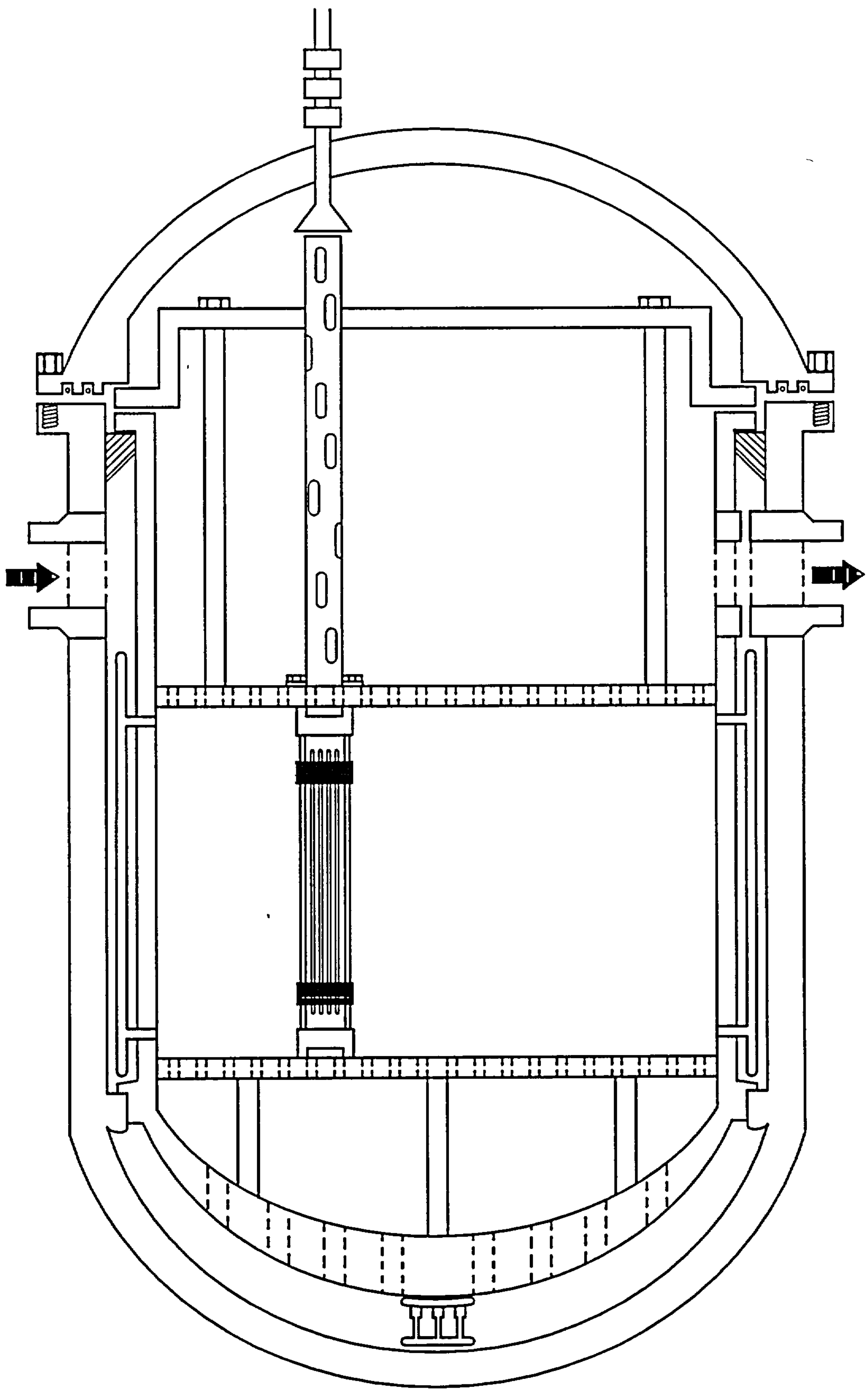


Figure 3.1-1 Reactor Vessel Cutaway

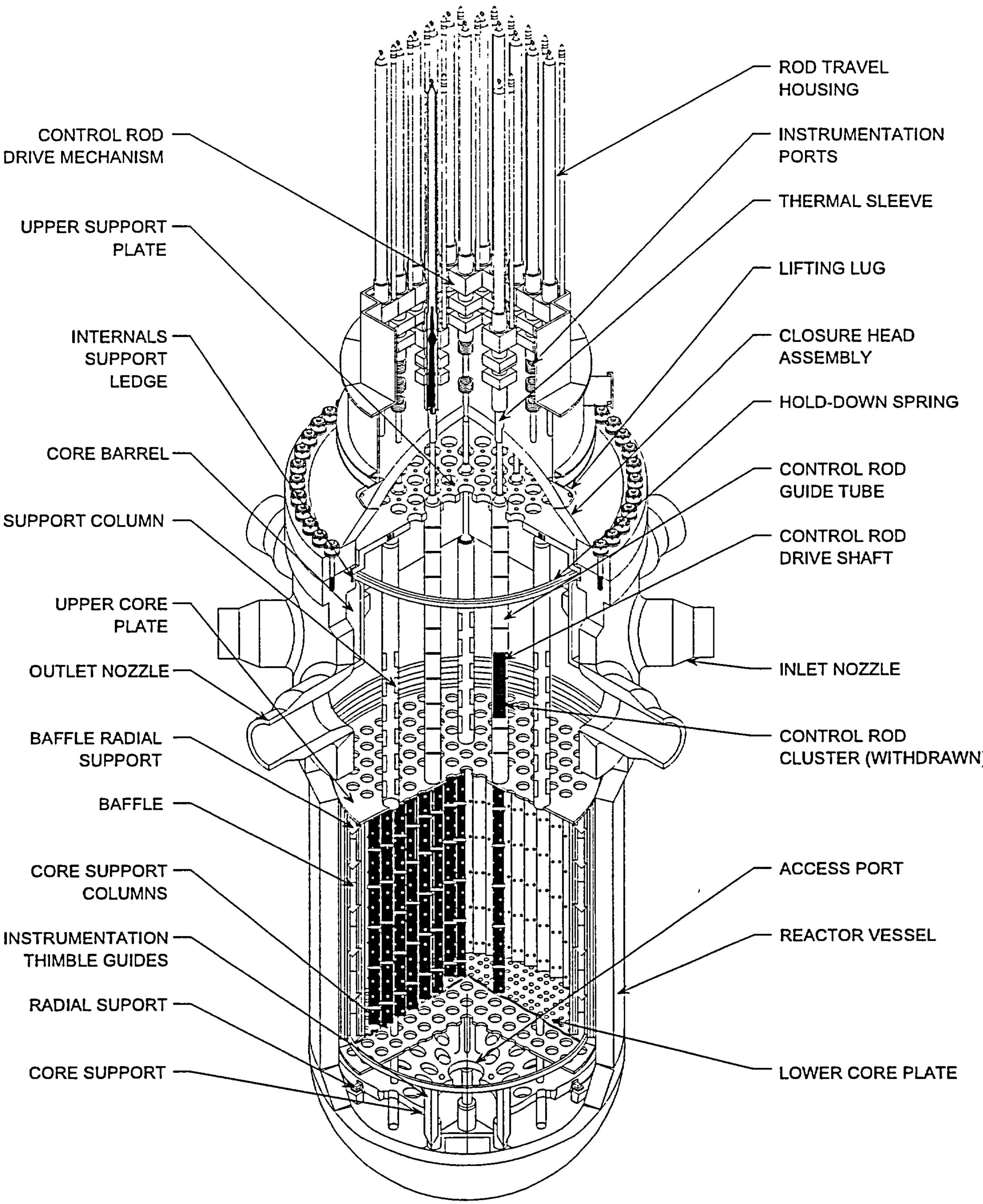


Figure 3.1-2 Reactor Vessel Internals

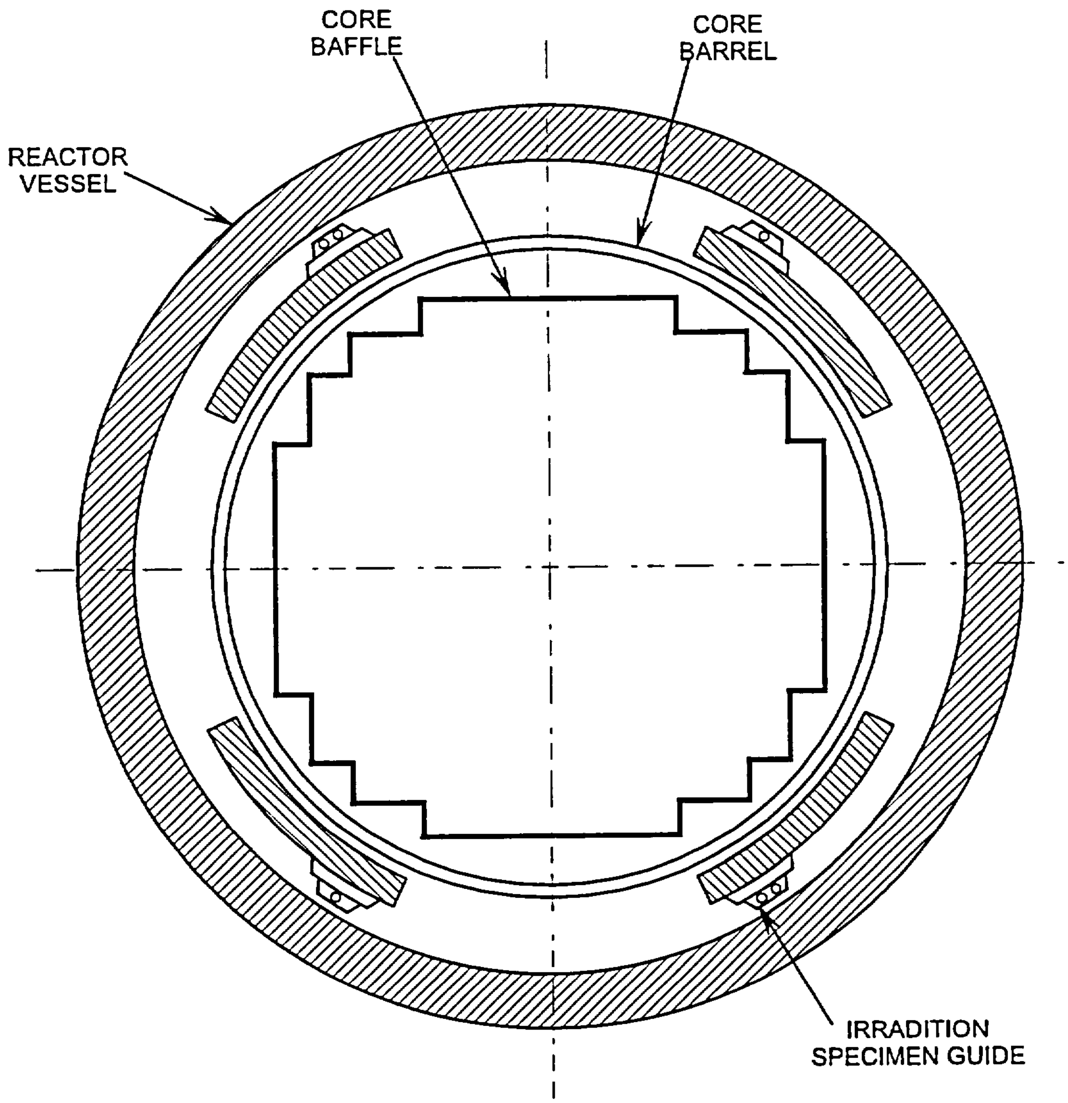


Figure 3.1-3 Reactor Vessel Cross Section

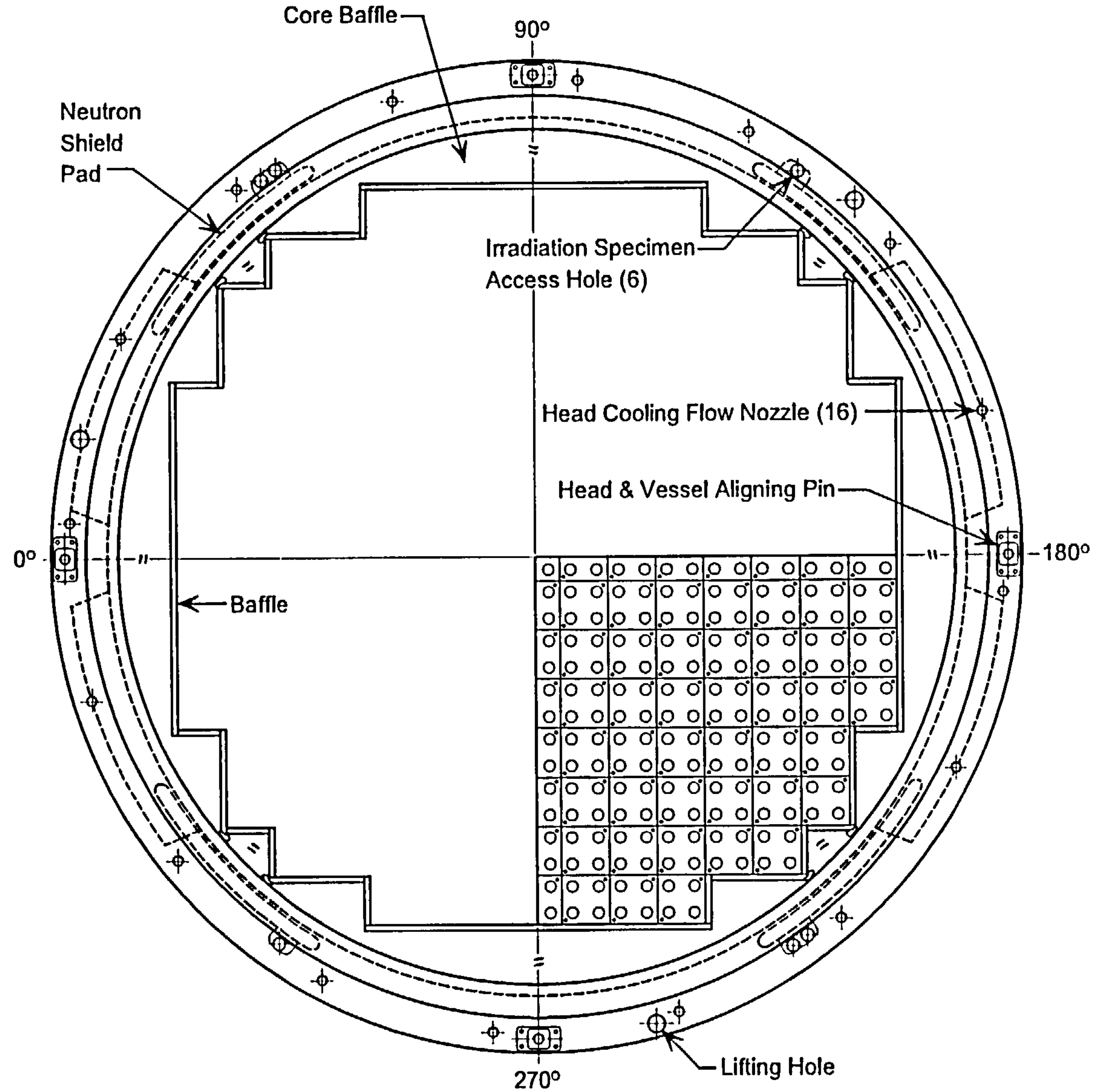


Figure 3.1-4 Core Barrel Flange

0196

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					N02 K-05	L13 K-15 Cy12	N19 E-04	R33	N23 J-06	N29 K-11	N08 L-04				
2			N14 D-14	R17	R14	R30	R32	N04 J-10	R07	R25	R18	R35	N18 M-14		
3		N06 B-12	D38R K-10	R37	M17 N-05	M45 G-15	L11 A-06 Cy12	P06 G-12	M14 L-03	M48 F-15	M20 K-15	R43	F11 F-10	N13 P-12	
4		R16	R38	P28 J-12	P18 H-12	P17 D-10	Q18 E-02	P38 G-14	Q26 L-02	P34 M-10	P08 J-05	P07 M-07	R41	R10	
5	N12 M-11	R12	M19 A-10	P21 L-09	Q04 N-03	Q17 C-04	P04 G-09	P40 J-14	P19 L-11	Q21 N-04	Q01 C-03	P09 M-08	M07 E-03	R20	N15 E-06
6	N32 E-10	R28	M43 A-06	P27 F-12	Q11 M-13	F04R H-09	Q27 K-14	F07R H-10	Q07 F-14	F56R J-08	Q24 D-13	P11 K-12	M26 R-09	R29	L12 R-06 Cy12
7	N22 K-09	R09	M15 N-11	Q20 P-11	P13 E-11	Q08 B-06	P29 G-05	N34 H-14	P25 E-09	Q15 P-06	P20 J-09	Q23 B-11	L23 F-15 Cy12	R24	N24 D11
8	R04	N25 F-09	P35 D-07	P42 B-07	P41 B-09	F18R F-08	N36 B-08	D12 H-13 Cy 4	N33 P-08	F05R K-08	P37 P-07	P39 P-09	P16 M-09	N21 K-07	R03
9	N11 M-05	R36	L09 K-01 Cy12	Q28 P-05	P03 G-07	Q22 B-10	P33 L-07	N35 H-02	P02 J-11	Q14 P-10	P24 L-05	Q19 B-05	M36 C-05	R21	N26 F-07
10	L20 A-10 Cy12	R01	M47 A-07	P23 F-04	Q09 M-03	F03R G-08	Q13 K-02	F02R H-06	Q06 F-02	F21 H-07	Q10 D-03	P15 K-04	M18 R-10	R27	N27 L-06
11	N20 L-10	R15	M05 L-13	P01 D-08	Q03 N-13	Q05 C-12	P22 E-05	P44 G-02	P05 J-07	Q12 N-12	Q02 C-13	P32 E-07	M38 R-06	R08	N09 D-05
12		R11	R40	P26 D-09	P30 G-11	P36 D-06	Q16 E-14	P43 J-02	Q25 L-14	P12 M-06	P14 H-04	P10 G-04	R39	R19	
13		N10 B-04	F30 F-06	R44	M30 F-01	M32 K-01	M46 E-13	P31 J-04	L30 R-10 Cy12	M04 J-01	M24 C-11	R42	F19R K-06	N05 P-04	
14			N16 D-02	R05	R23	R26	R02	N28 G-06	R22	R13	R31	R06	N07 M-02		
15					N03 E-12	N31 F-05	N30 G-10	R34	N17 L-12	L01 F-01 Cy12	N01 F-11				

Legend	
XXX	Assembly Identity
XXX	Position in Previous Cycle
XXX	Discharge Cycle of Reinserts

Figure 3.1-5 Core Loading Arrangement

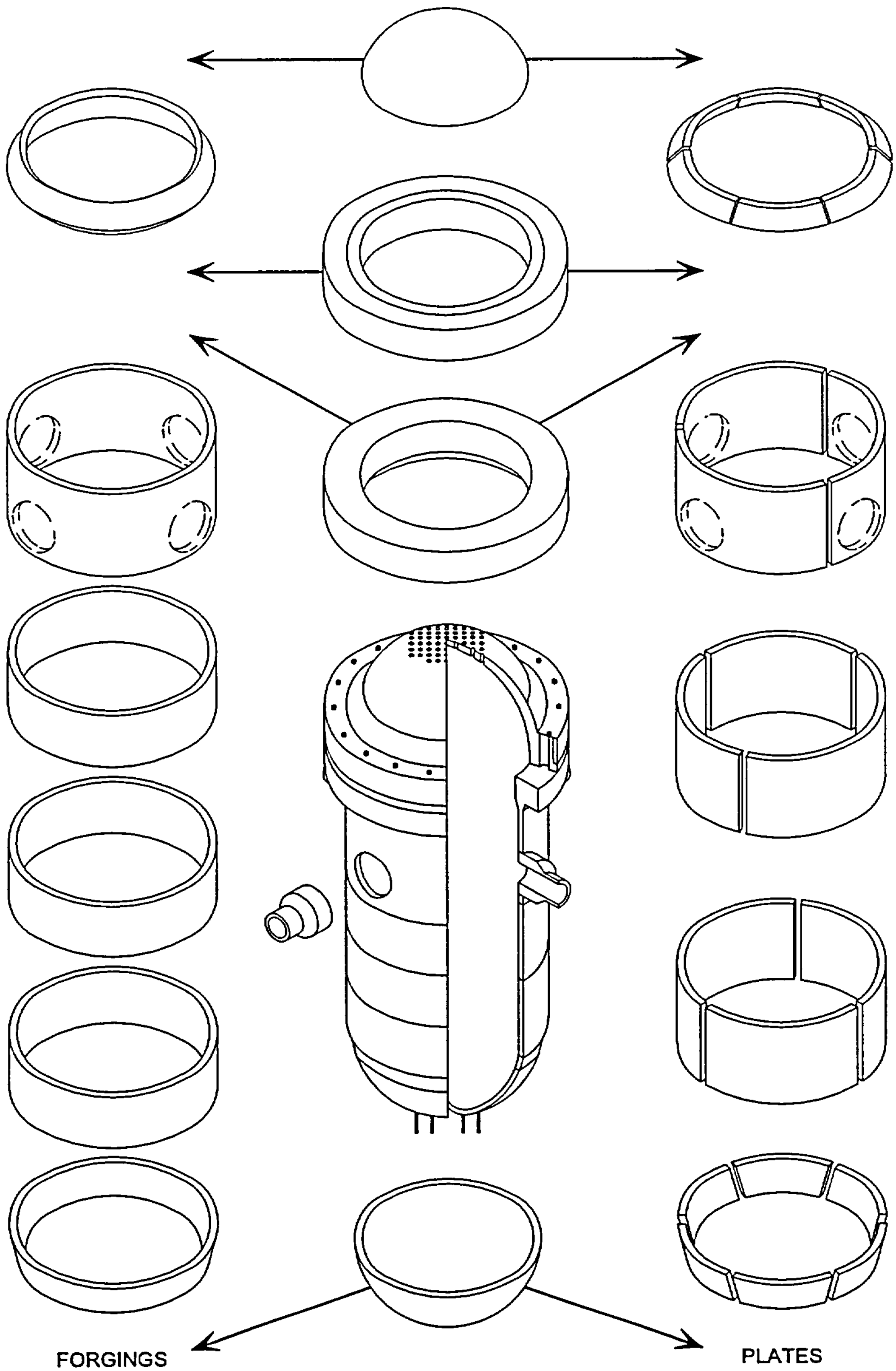


Figure 3.1-6 Reactor Vessel Construction

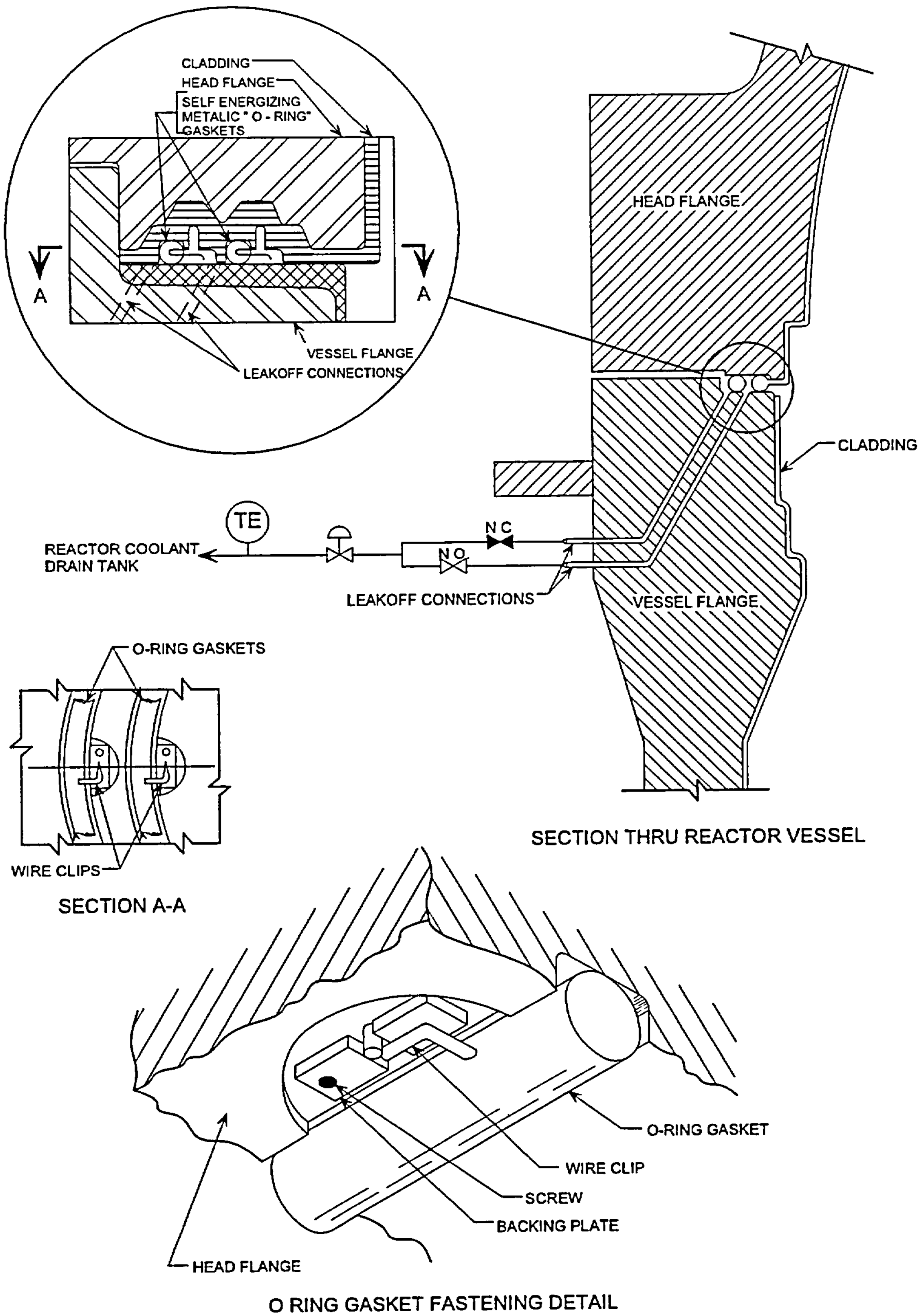


Figure 3.1-7 Reactor Vessel Seal

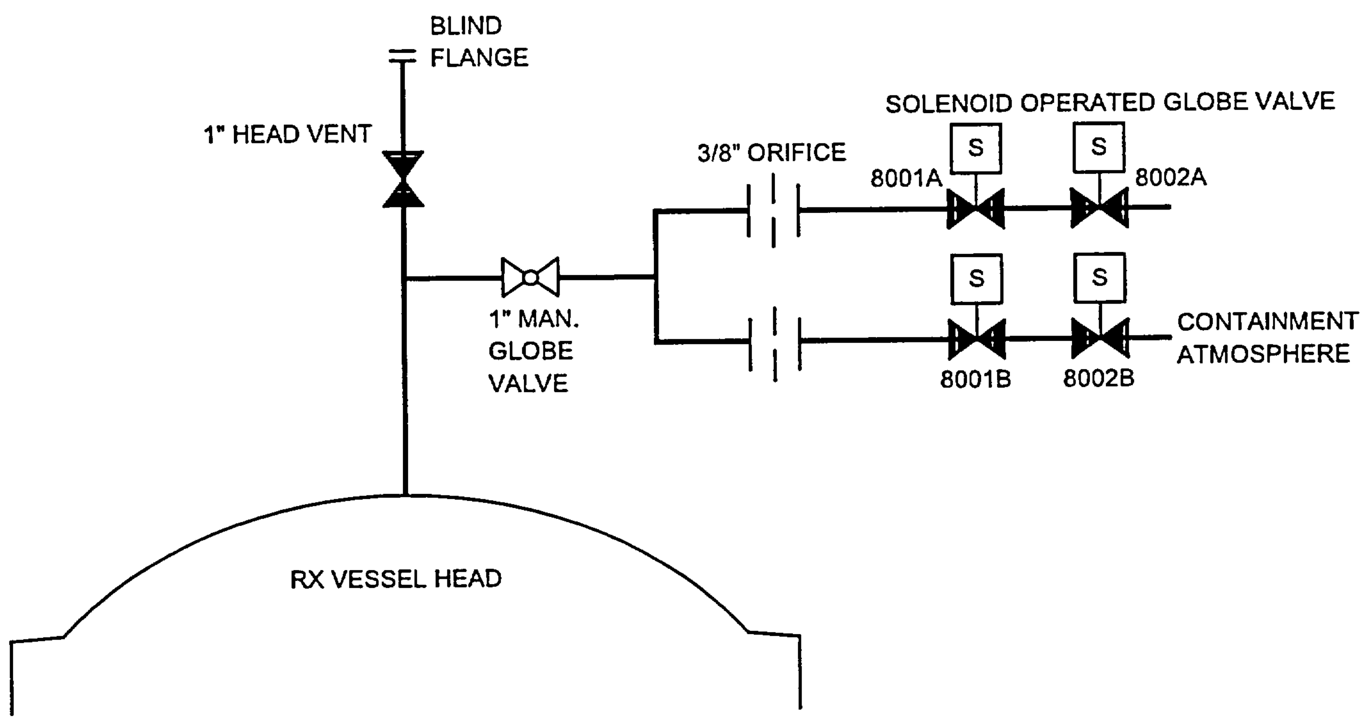


Figure 3.1-8 Reactor Vessel Head Venting System

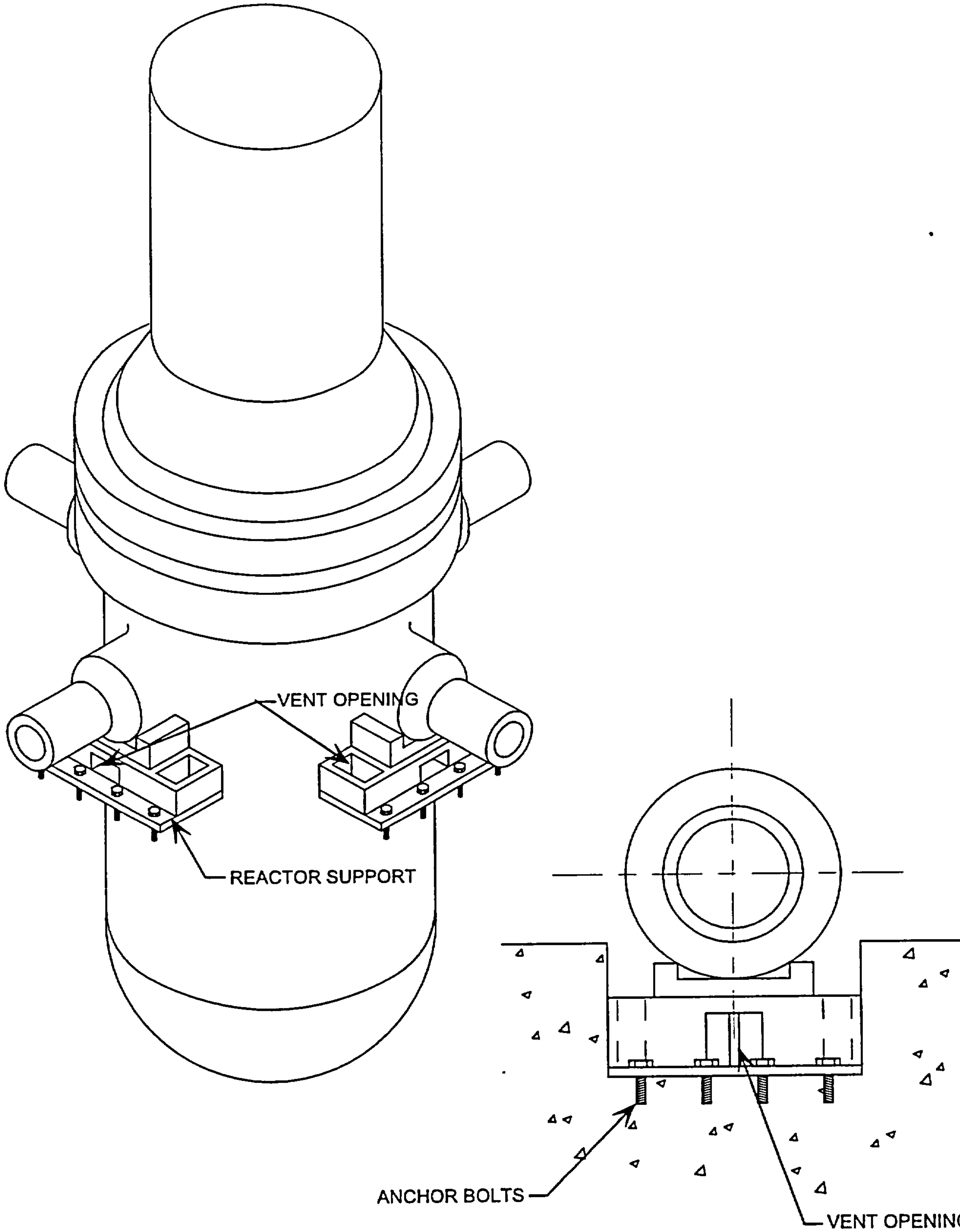


Figure 3.1-9 Reactor Vessel Supports

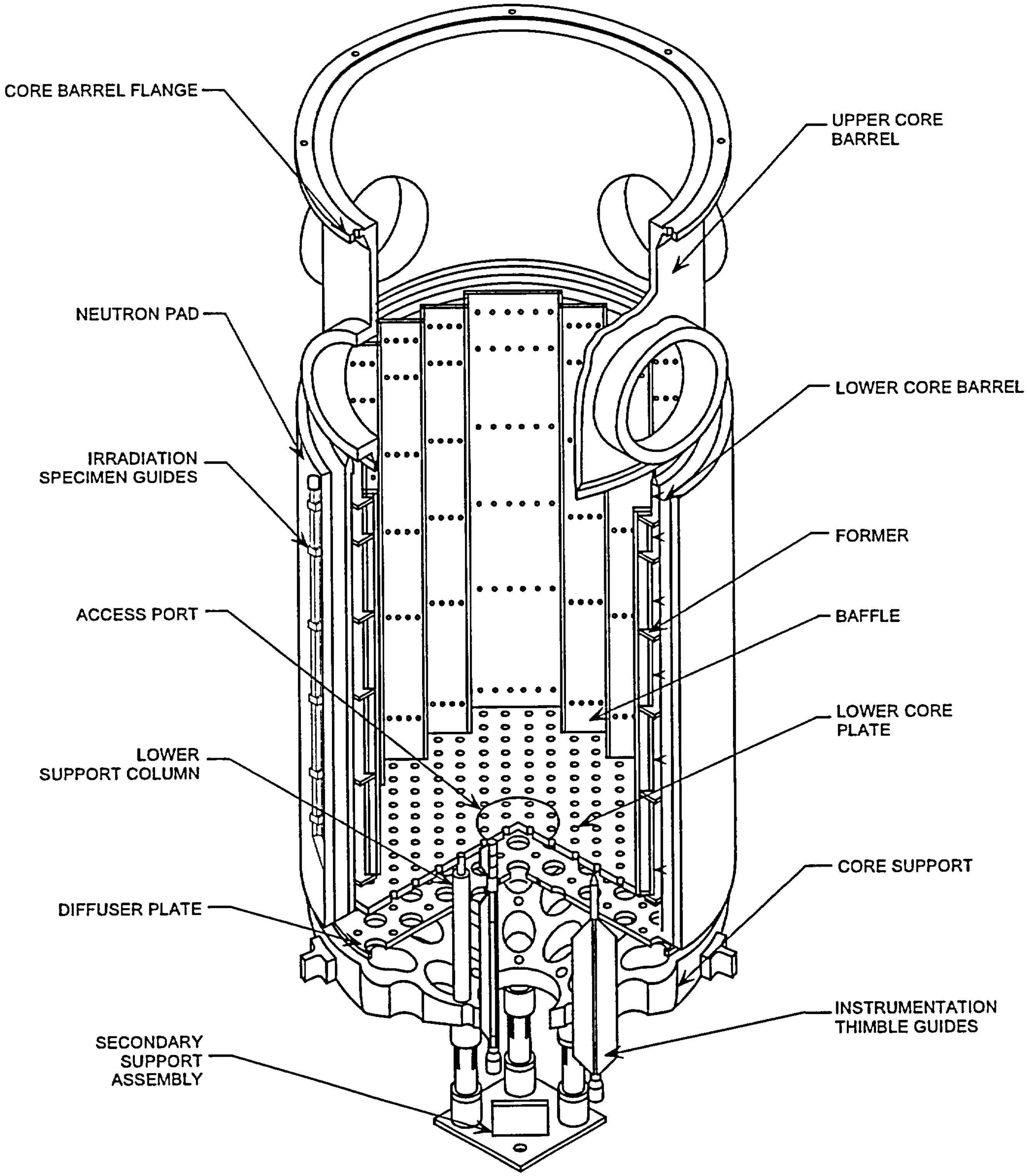


Figure 3.1-10 Lower Core Support Structure

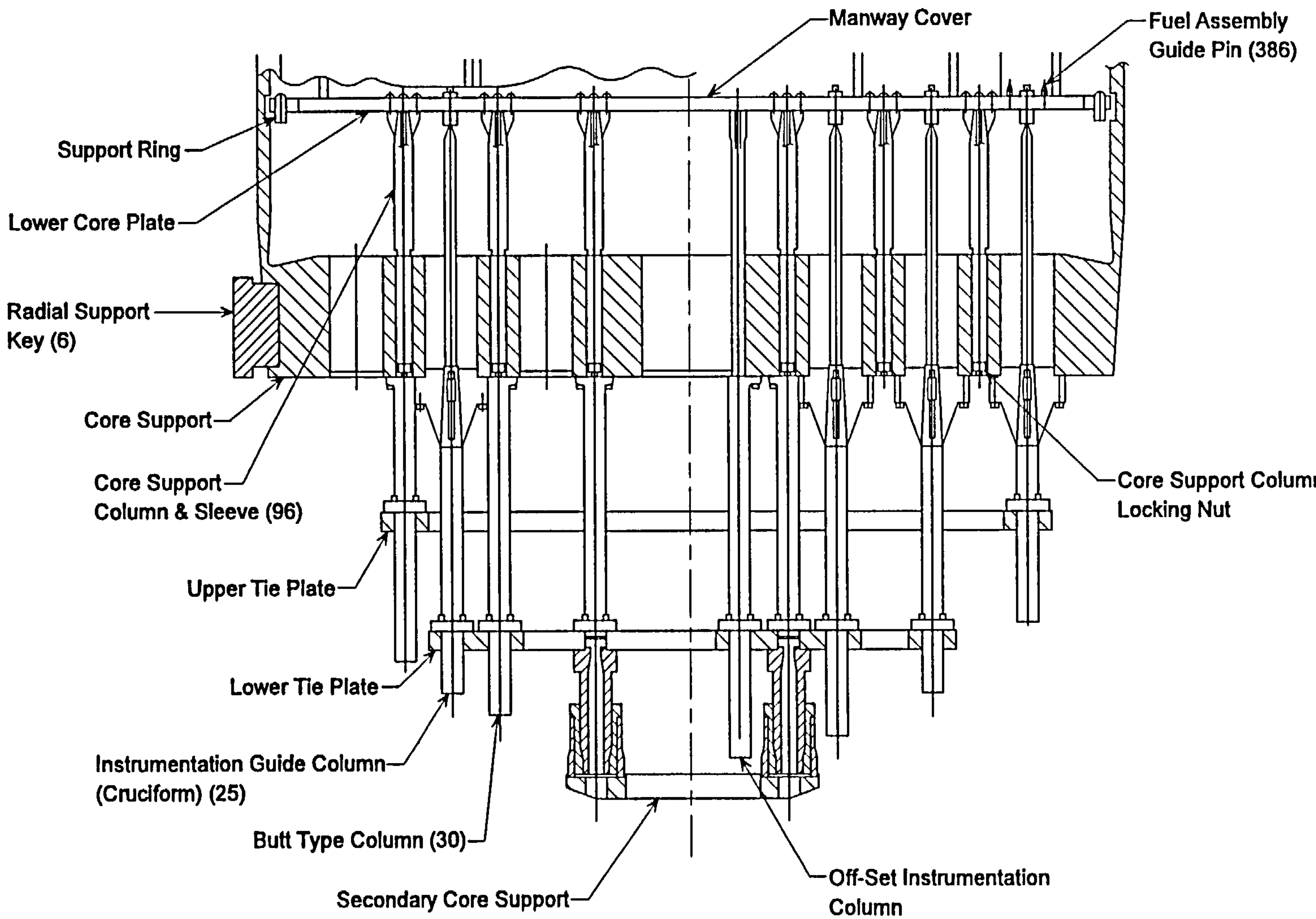


Figure 3.1-11 Instrument Guide and Secondary Core Support Assembly

0199

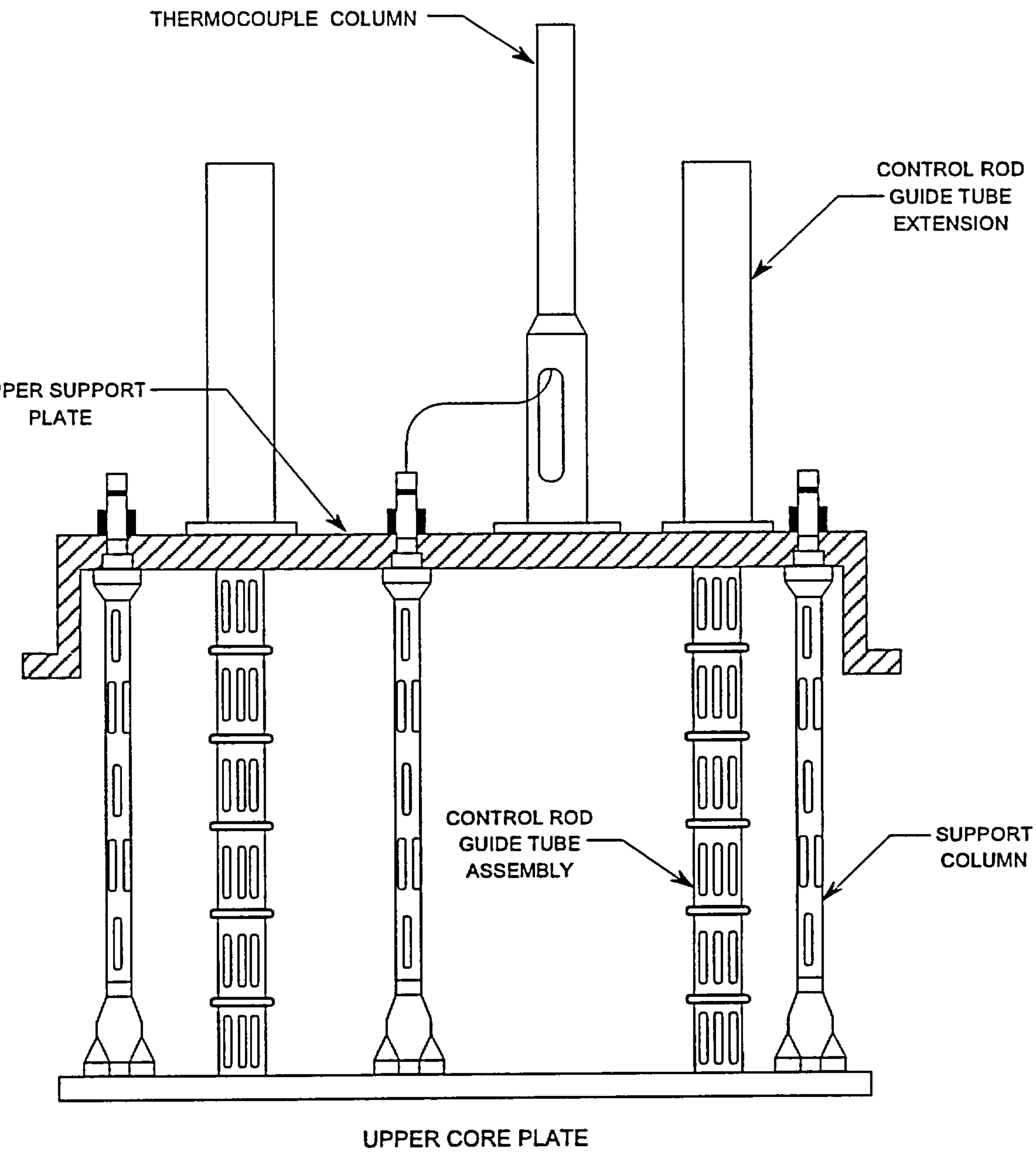


Figure 3.1-12 Upper Core Support Structure

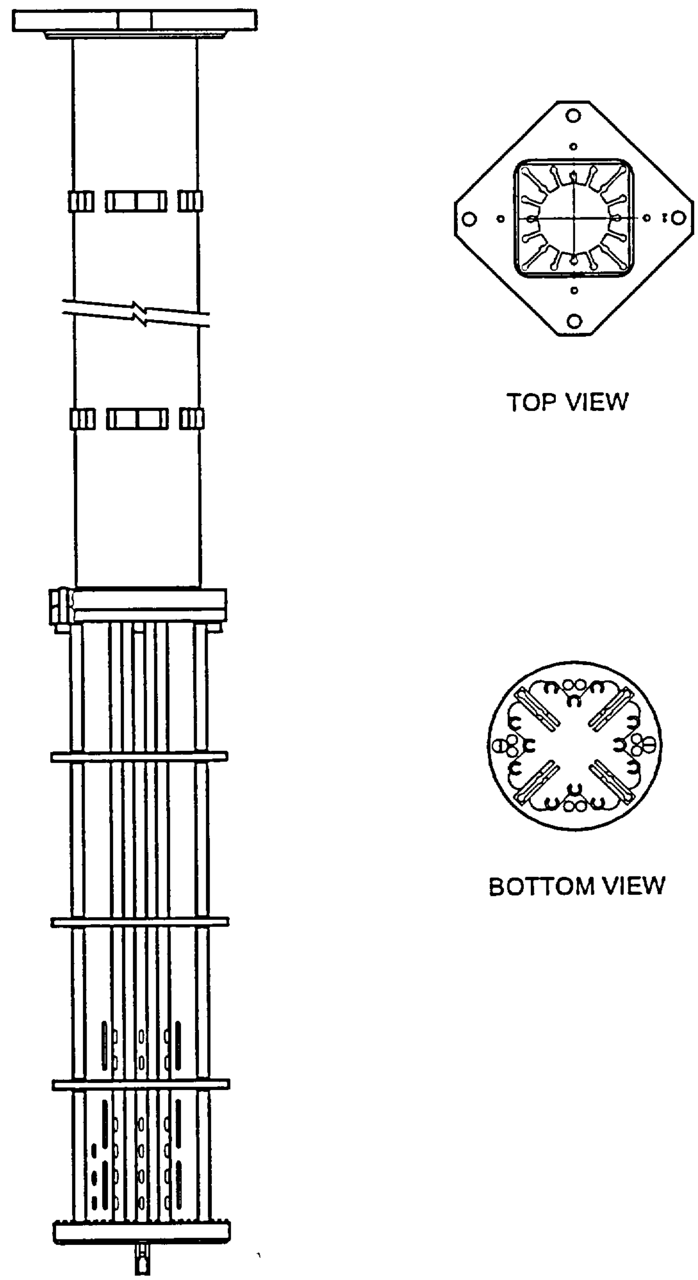


Figure 3.1-13 Control Rod Guide Tube Assembly

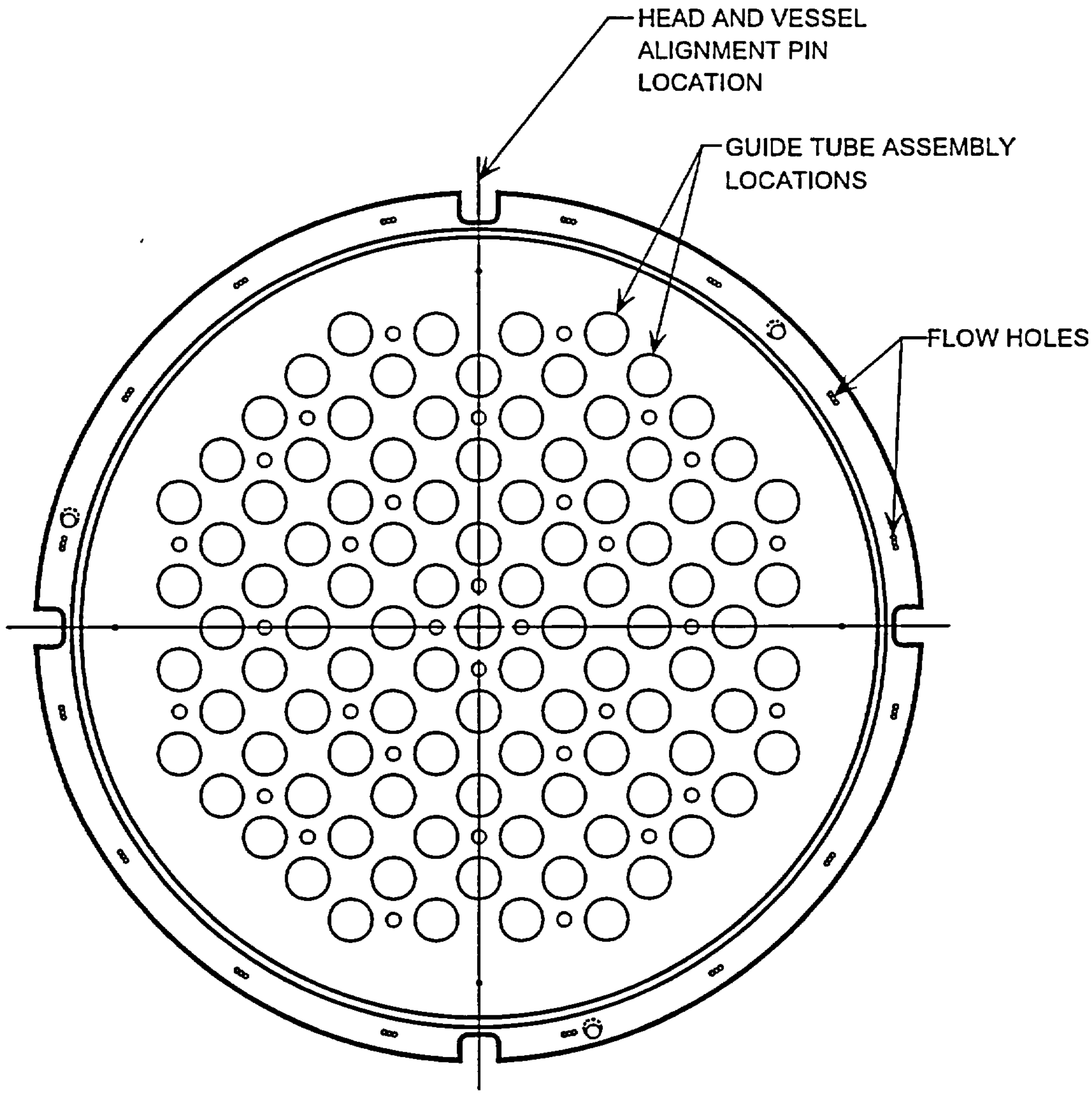


Figure 3.1-14 Upper Core Support Plate

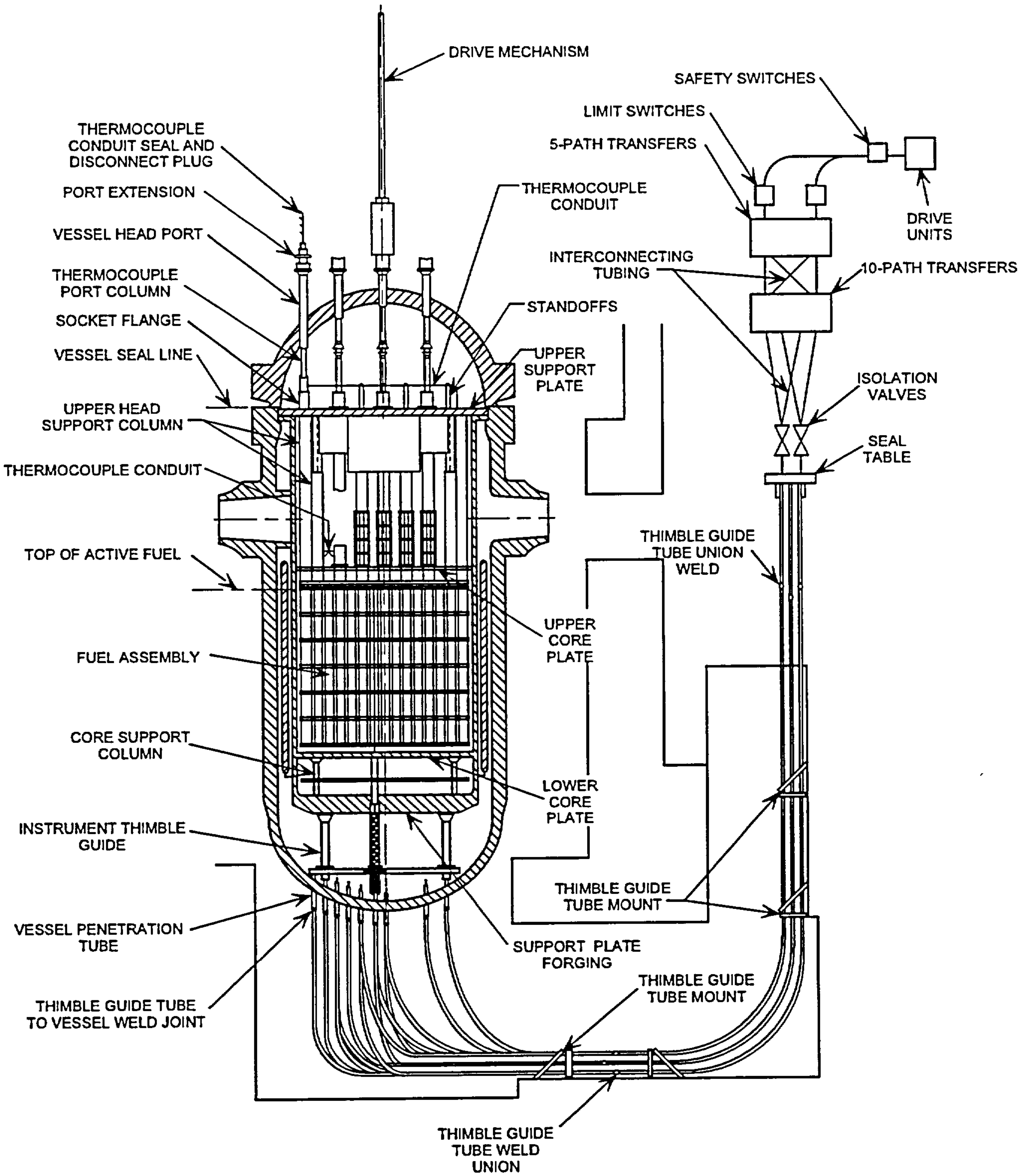


Figure 3.1-15 In-Core Instrumentation

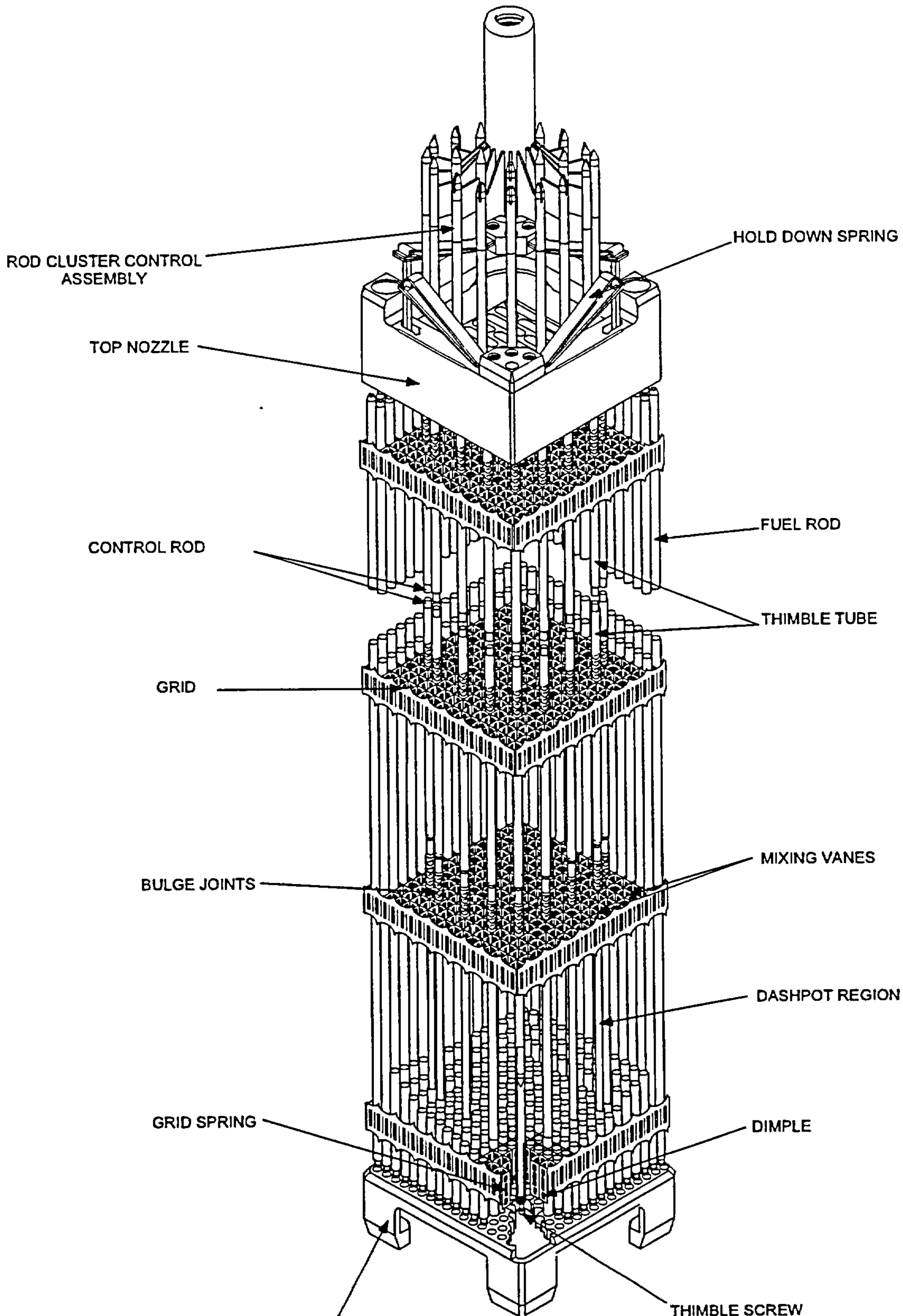
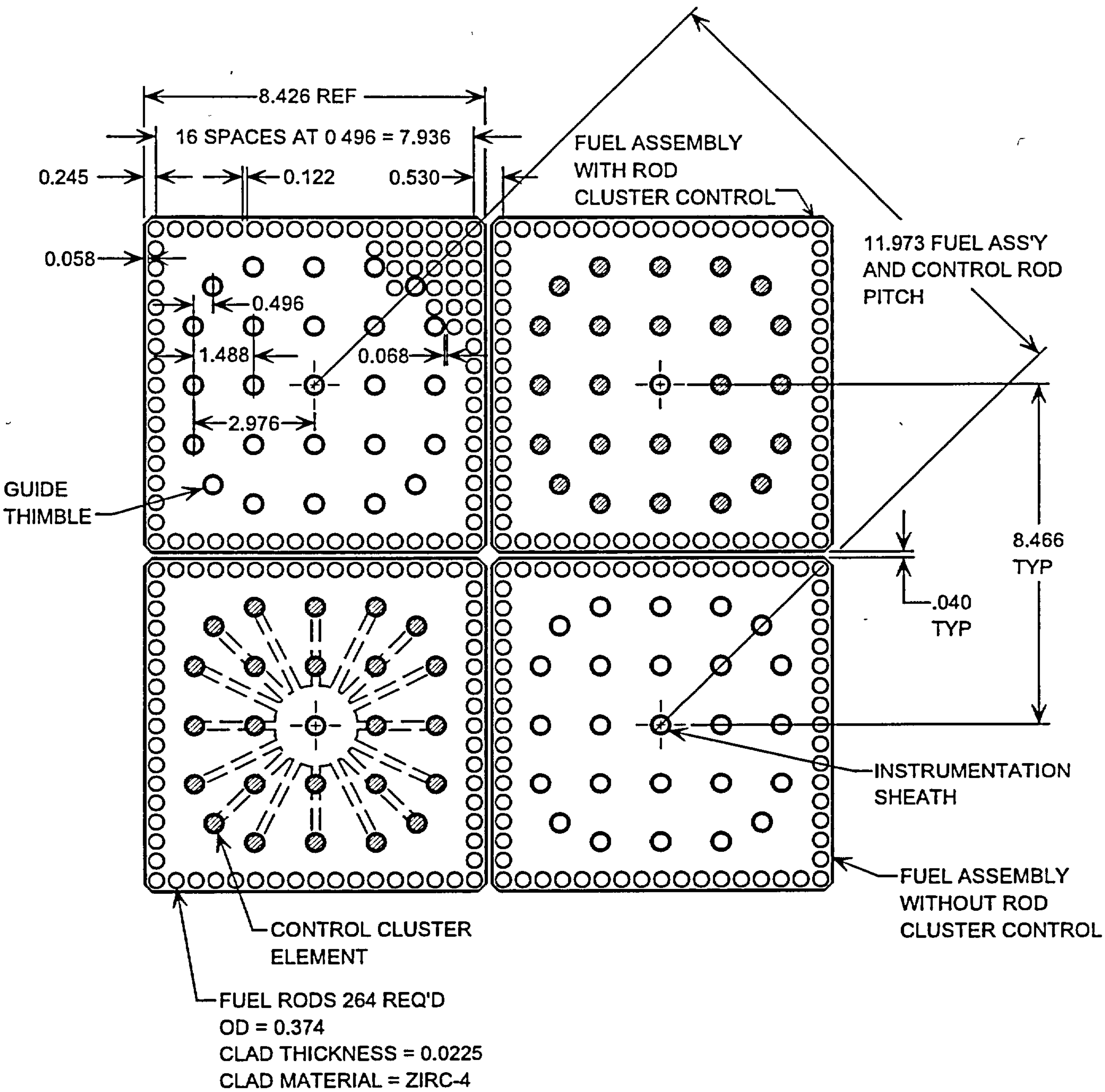
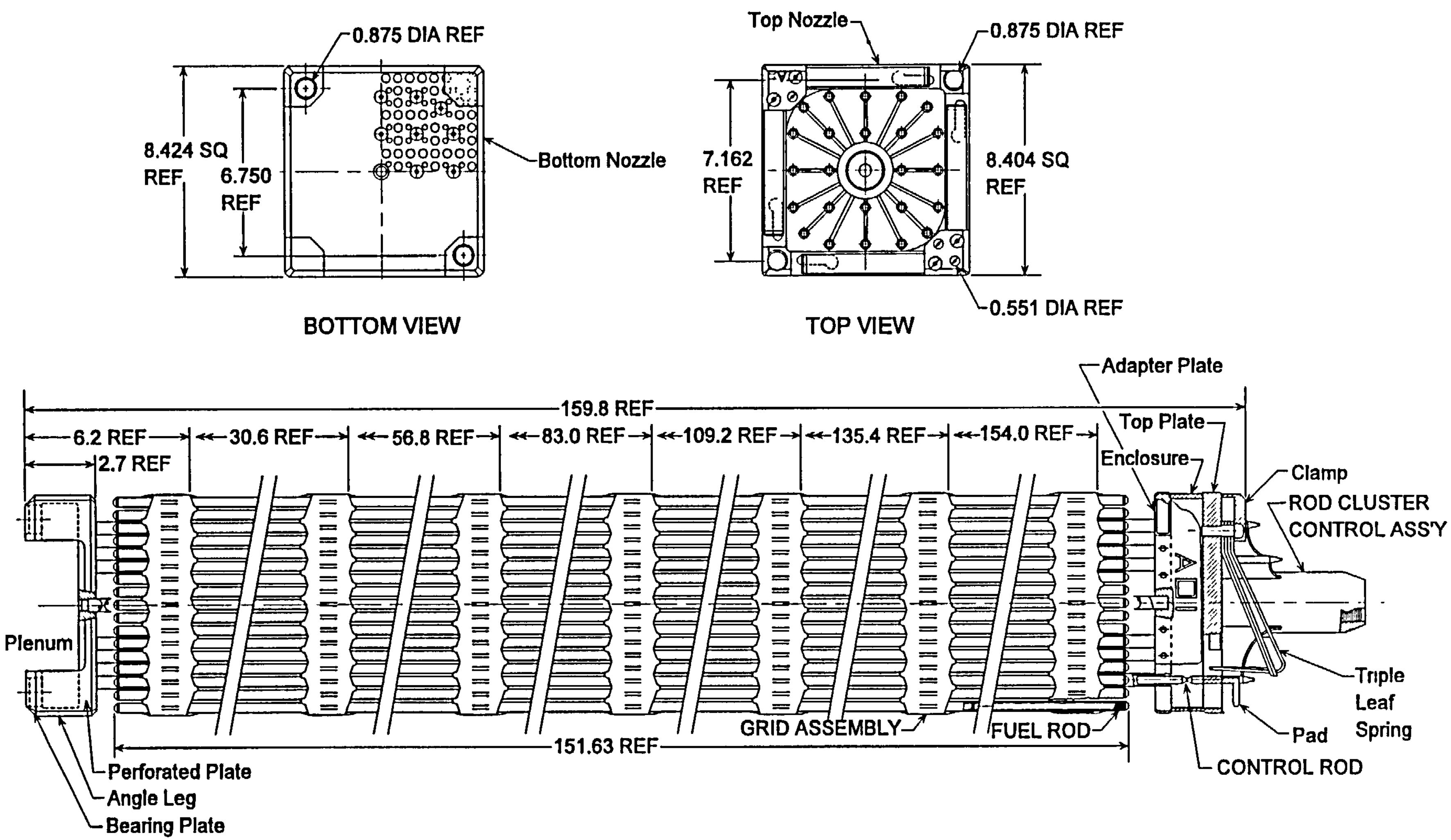


Figure 3.1-16 Fuel Assembly and RCCA Cutaway



NOTE ALL DIMENSIONS ARE IN INCHES.

Figure 3.1-17 17 X 17 Fuel Assembly Cross Section



NOTE: ALL DIMENSIONS ARE IN INCHES.

Figure 3.1-18 17X17 Fuel Assembly Outline

0199-3

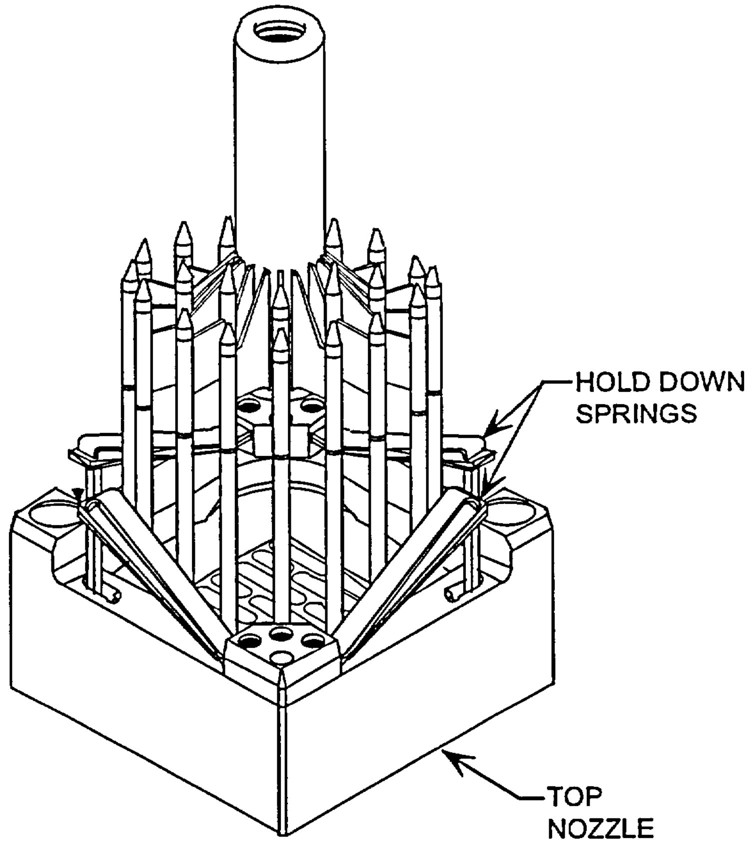
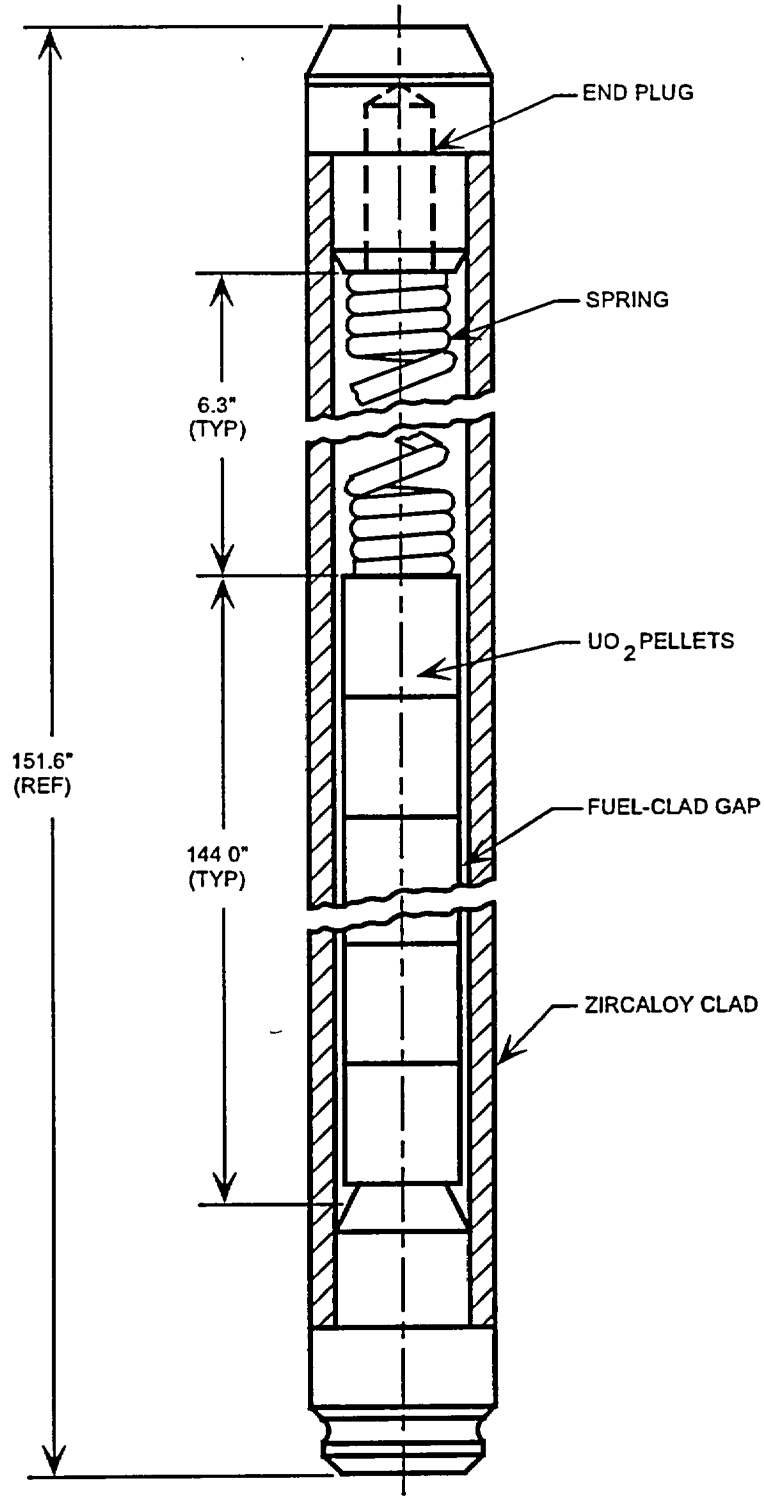


Figure 3.1-19 Upper Fuel Assembly and RCCA Spider



SPECIFIC DIMENSIONS DEPEND ON DESIGN VARIABLES SUCH AS PRE-PRESSURIZATION, POWER HISTORY, AND DISCHARGE BURNUP

Figure 3.1-20 Fuel Rod

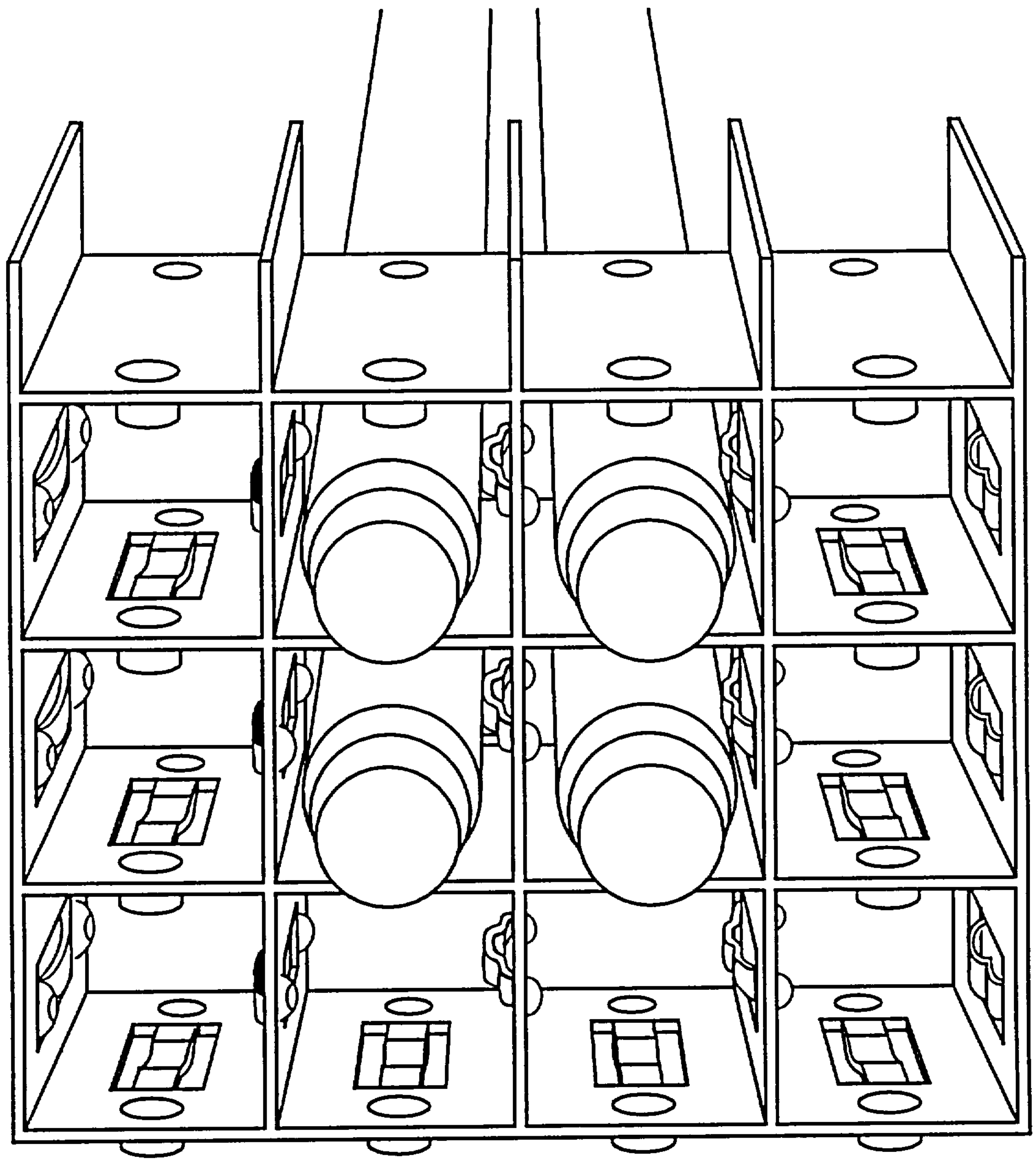


Figure 3.1-21 Spring Grid Clip Assembly

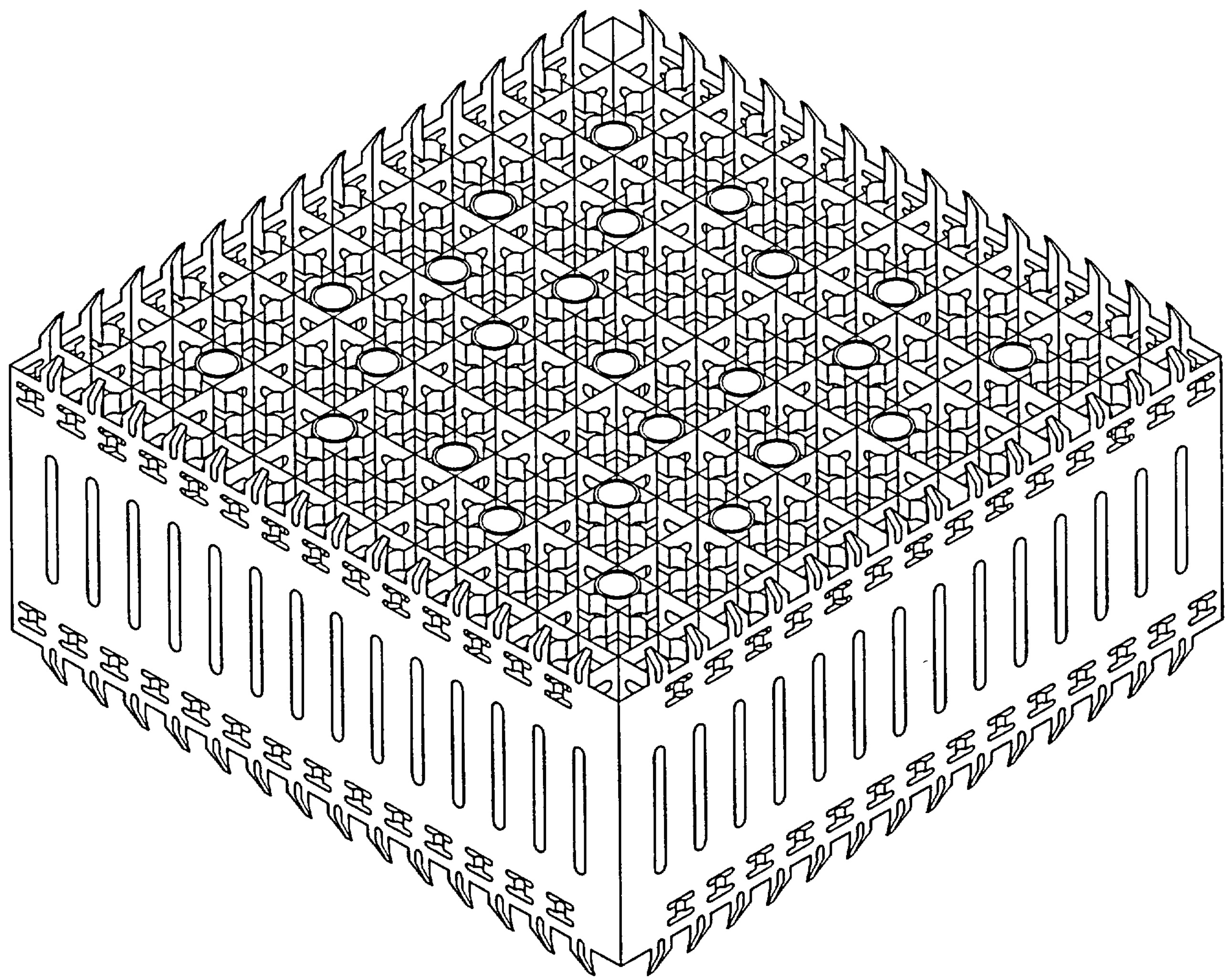
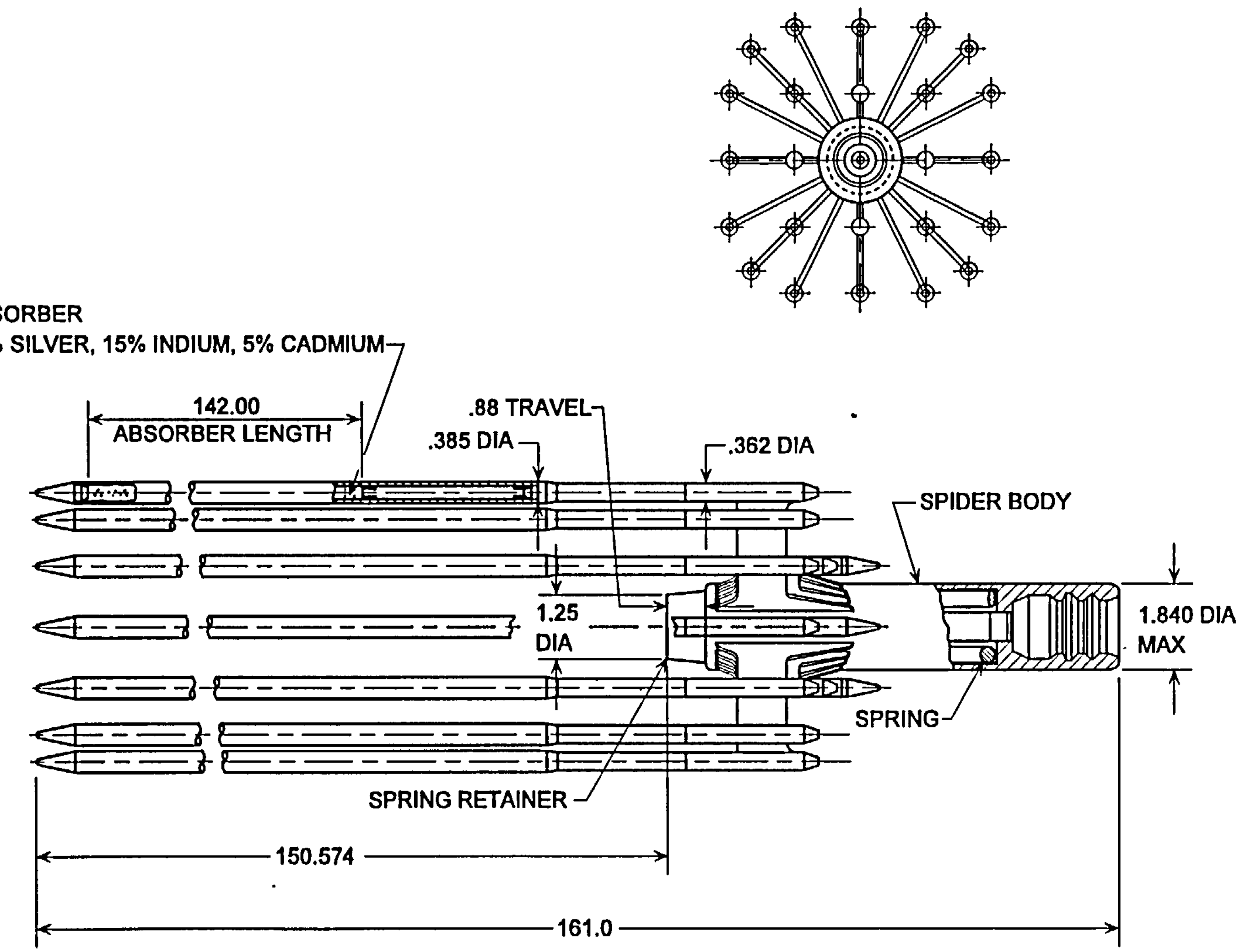


Figure 3.1-22 Spring Grid Assembly



NOTE. ALL DIMENSIONS ARE IN INCHES.

Figure 3.1-23 Rod Cluster Control Assembly

0199-3

0199-3

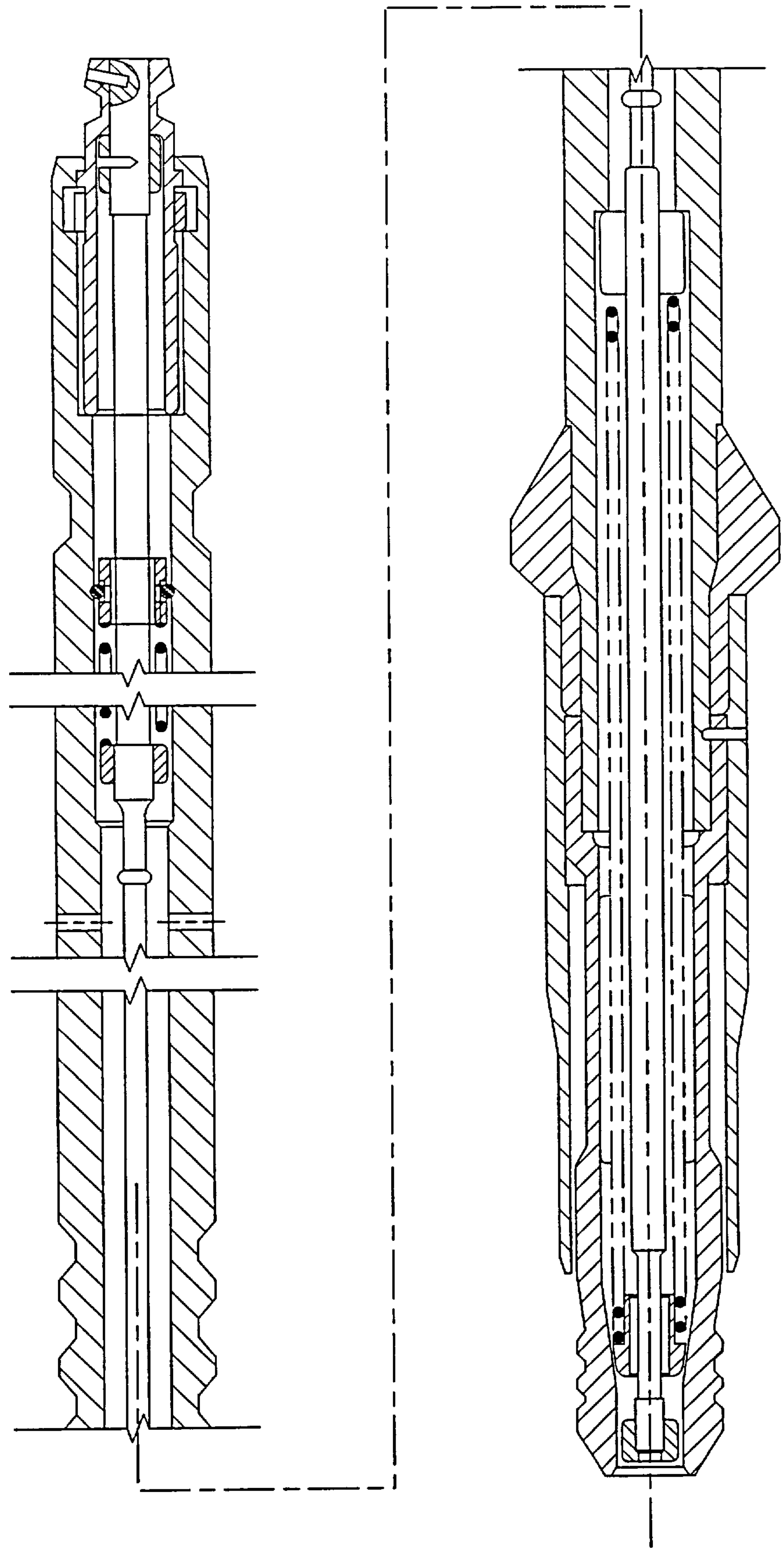


Figure 3.1-24 RCCA Drive Shaft

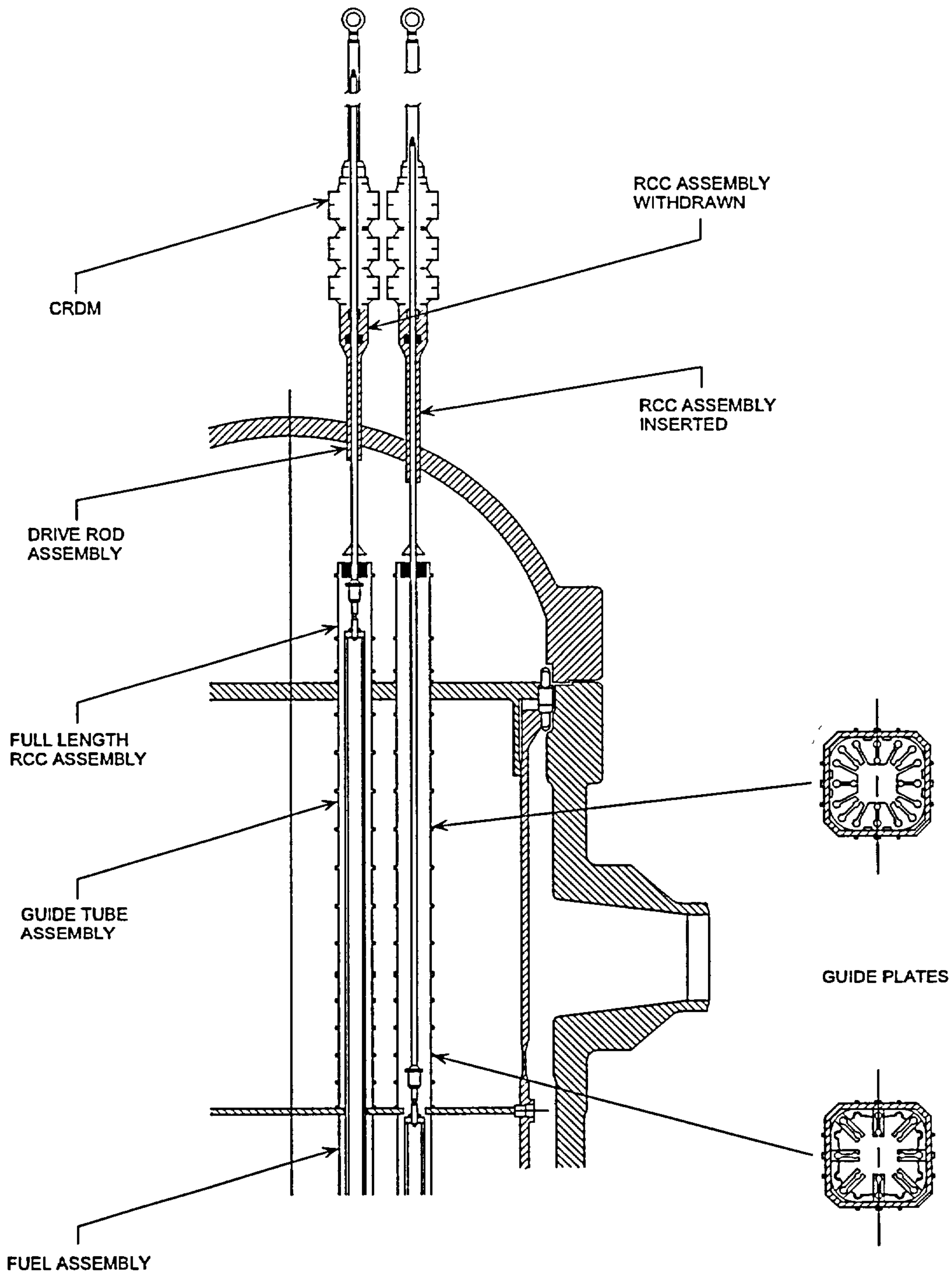


Figure 3.1-25 Full Length Rod with Interfacing Components

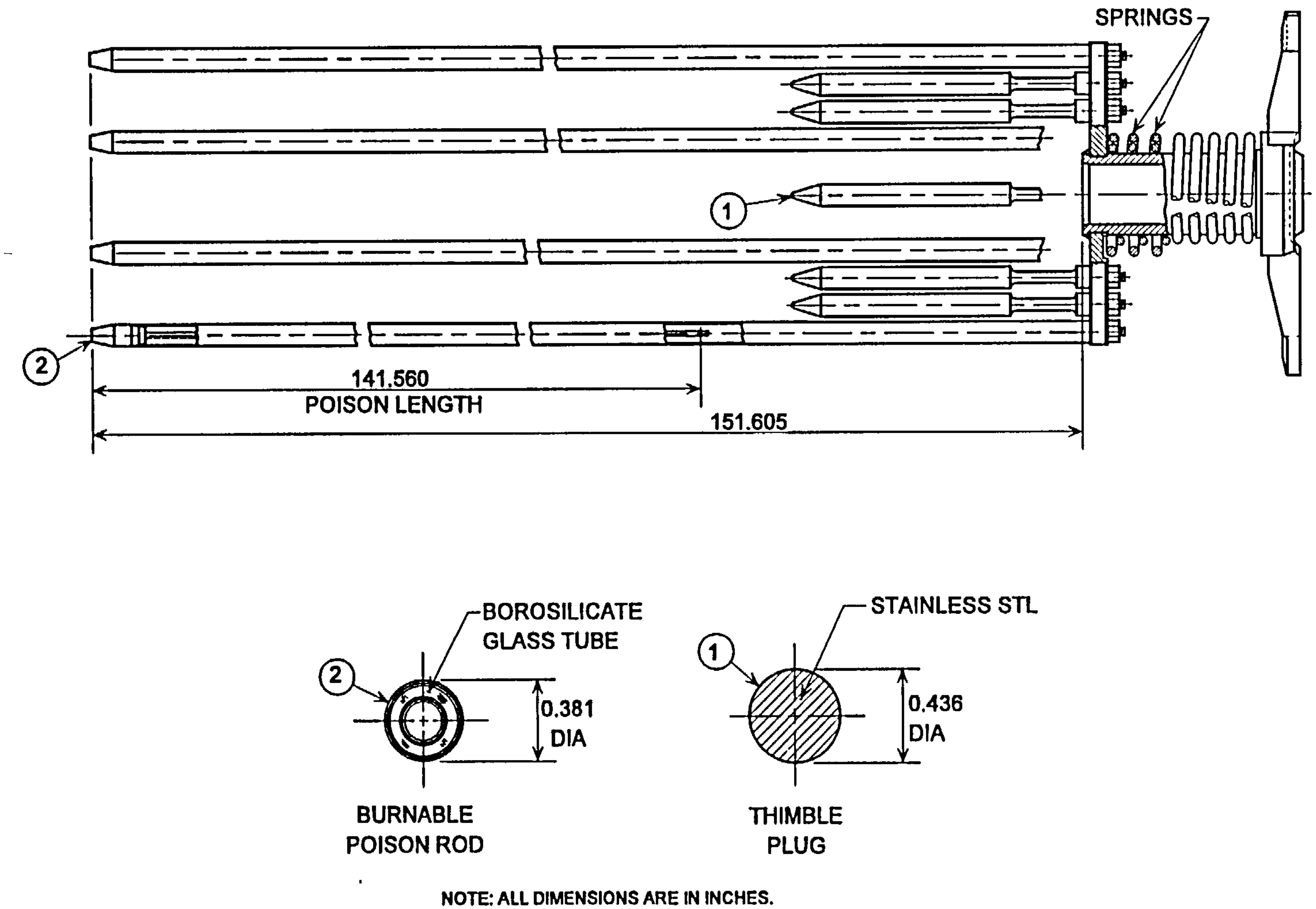
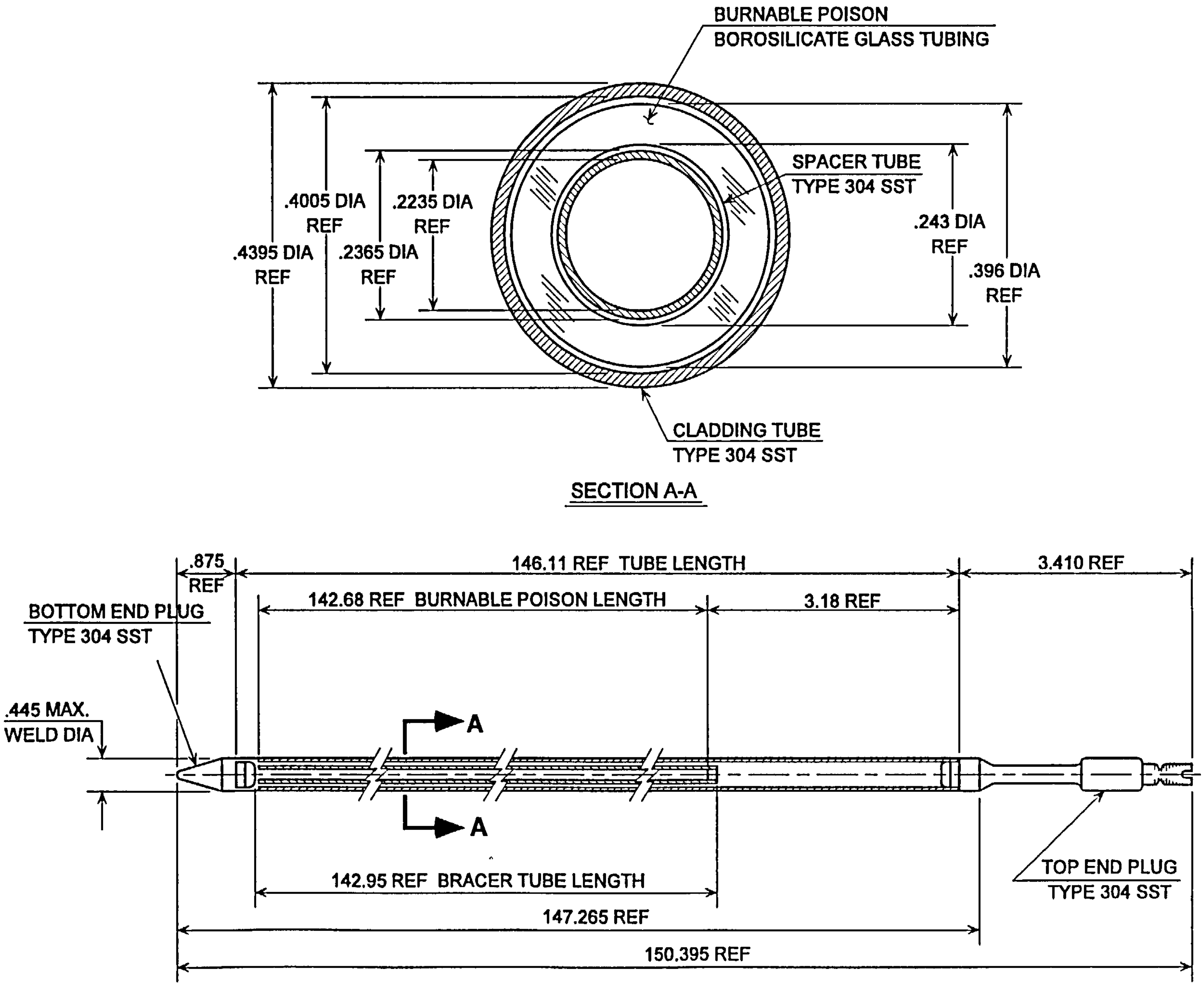


Figure 3.1-26 Burnable Poison Rod Assembly

0199-3



NOTE ALL DIMENSIONS ARE IN INCHES

0199-3

Figure 3.1-27 Burnable Poison Rod

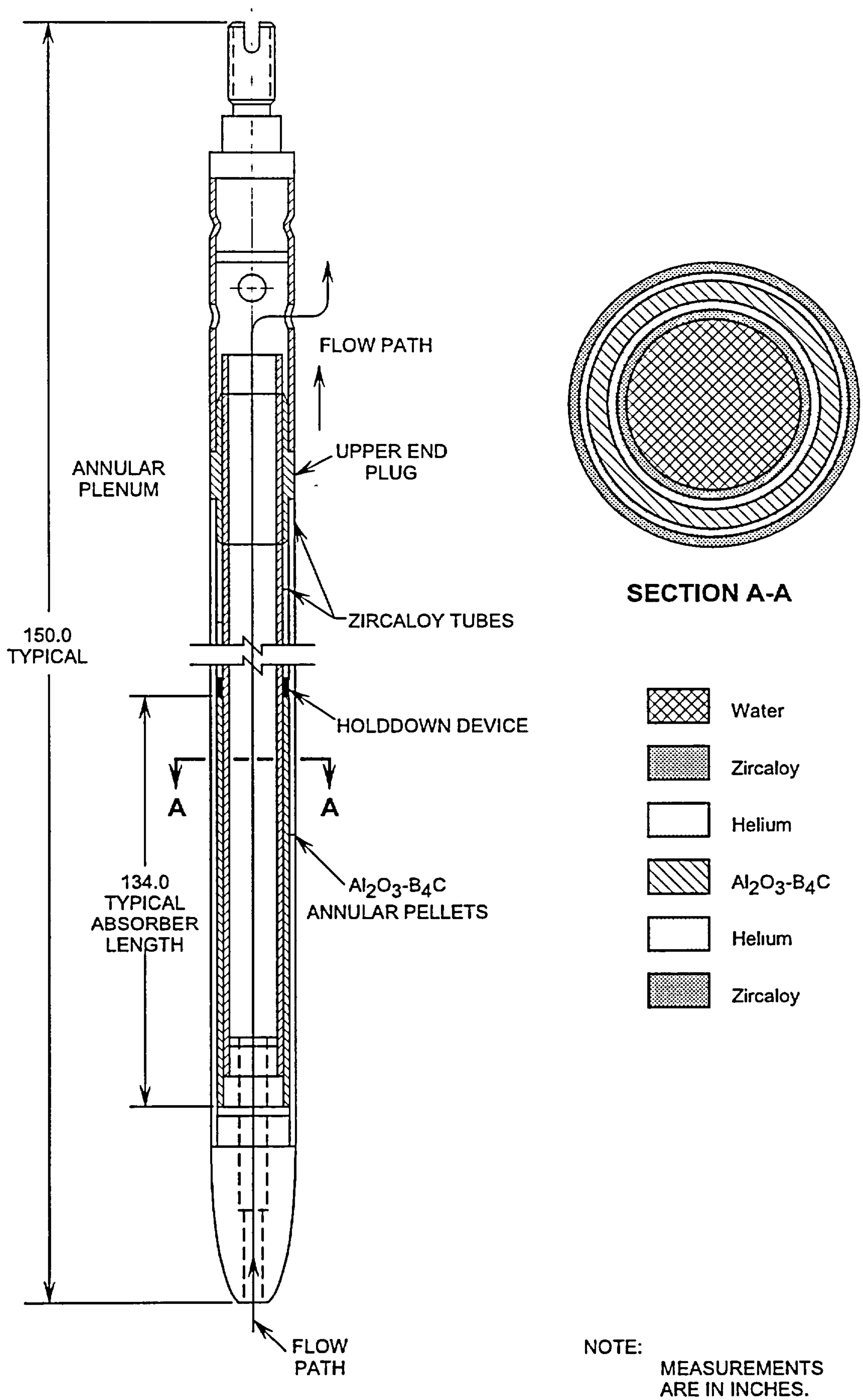


Figure 3.1-28 Wet Annular Burnable Absorber

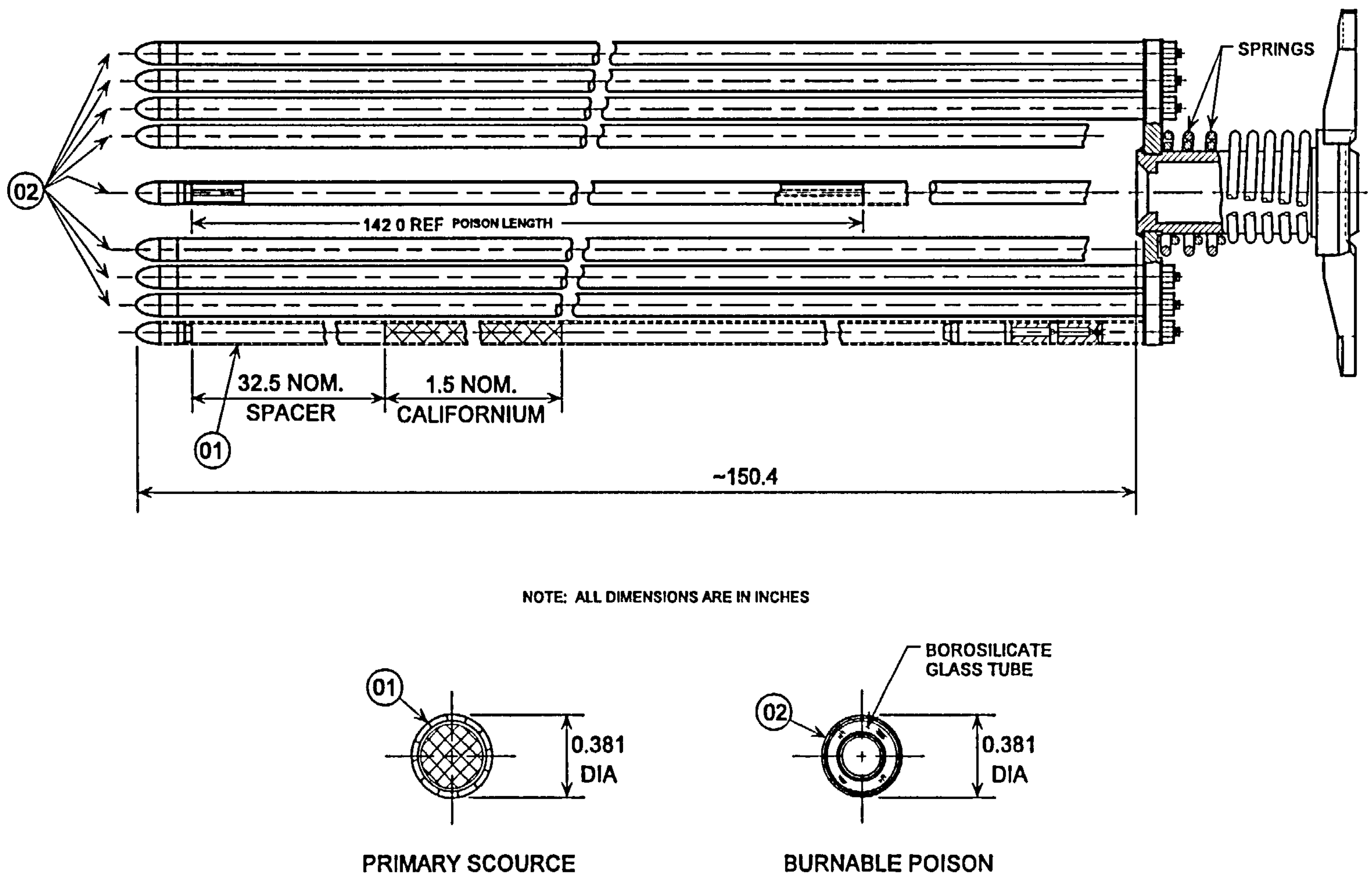


Figure 3.1-29 Primary Source Assembly

0199-3

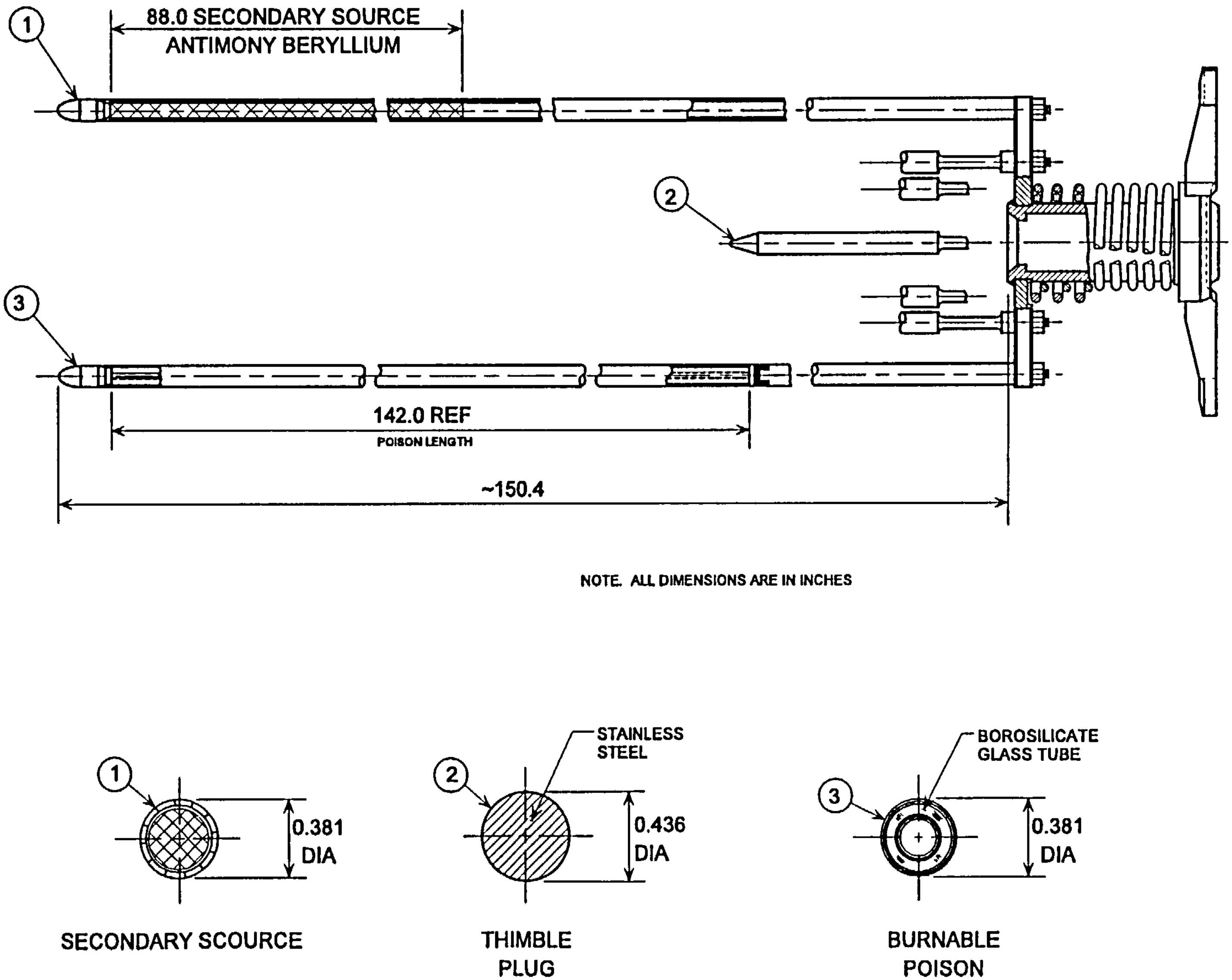
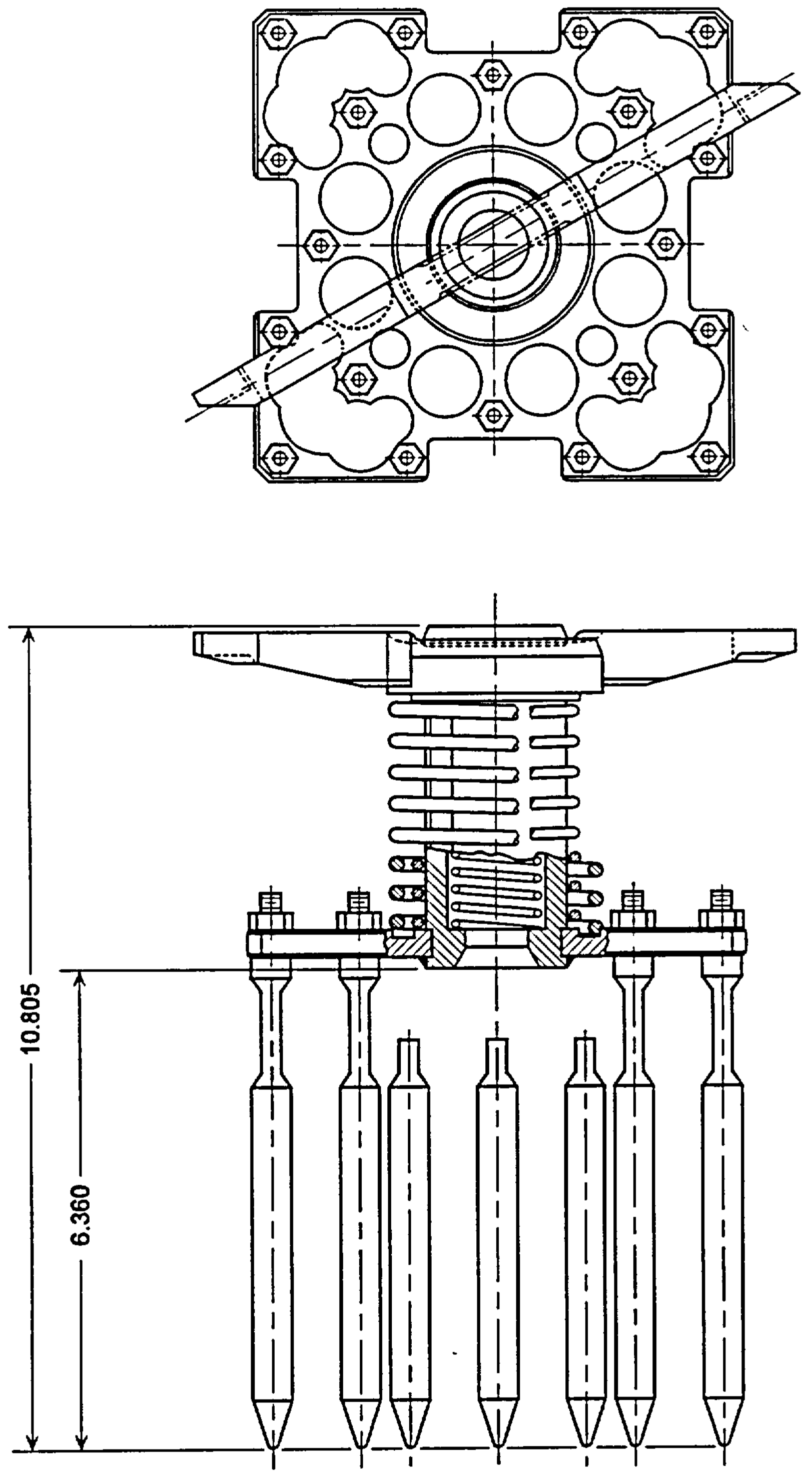


Figure 3.1-30 Secondary Source Assembly

0199-3



NOTE: ALL DIMENSIONS ARE IN INCHES

Figure 3.1-31 Thimble Plug Assembly

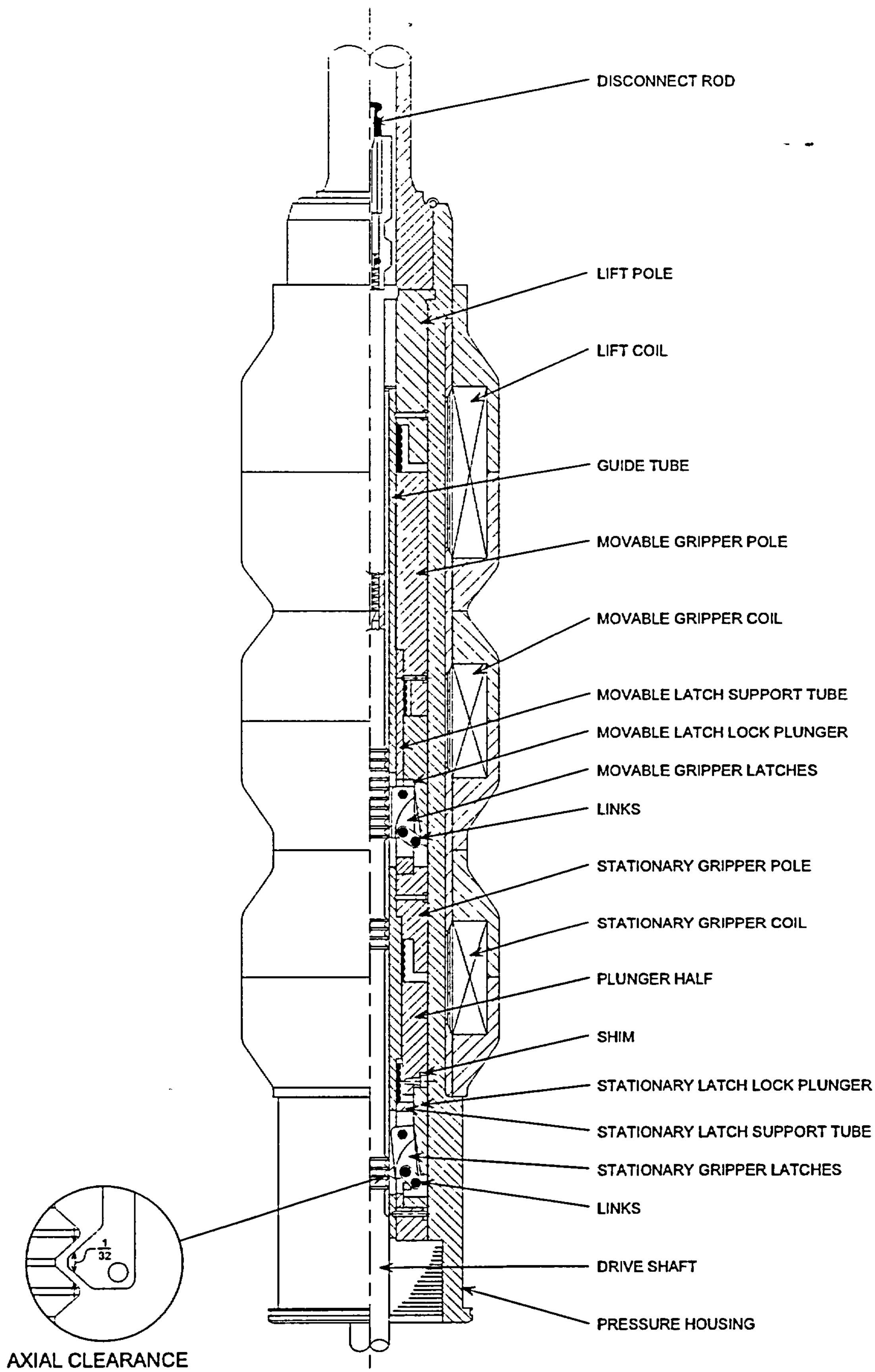
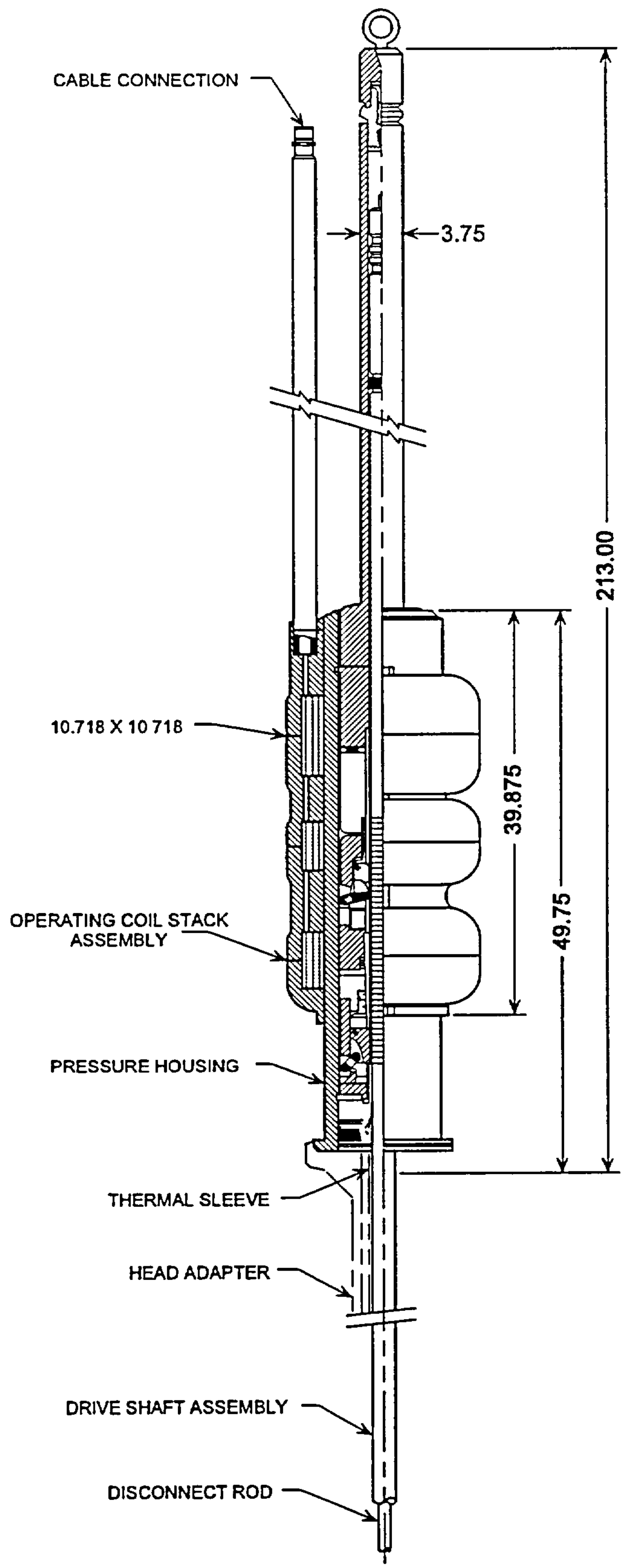


Figure 3.1-32 Control Rod Drive Mechanism



NOTE: ALL DIMENSIONS ARE IN INCHES.

Figure 3.1-33 Control Rod Drive Mechanism Assembly

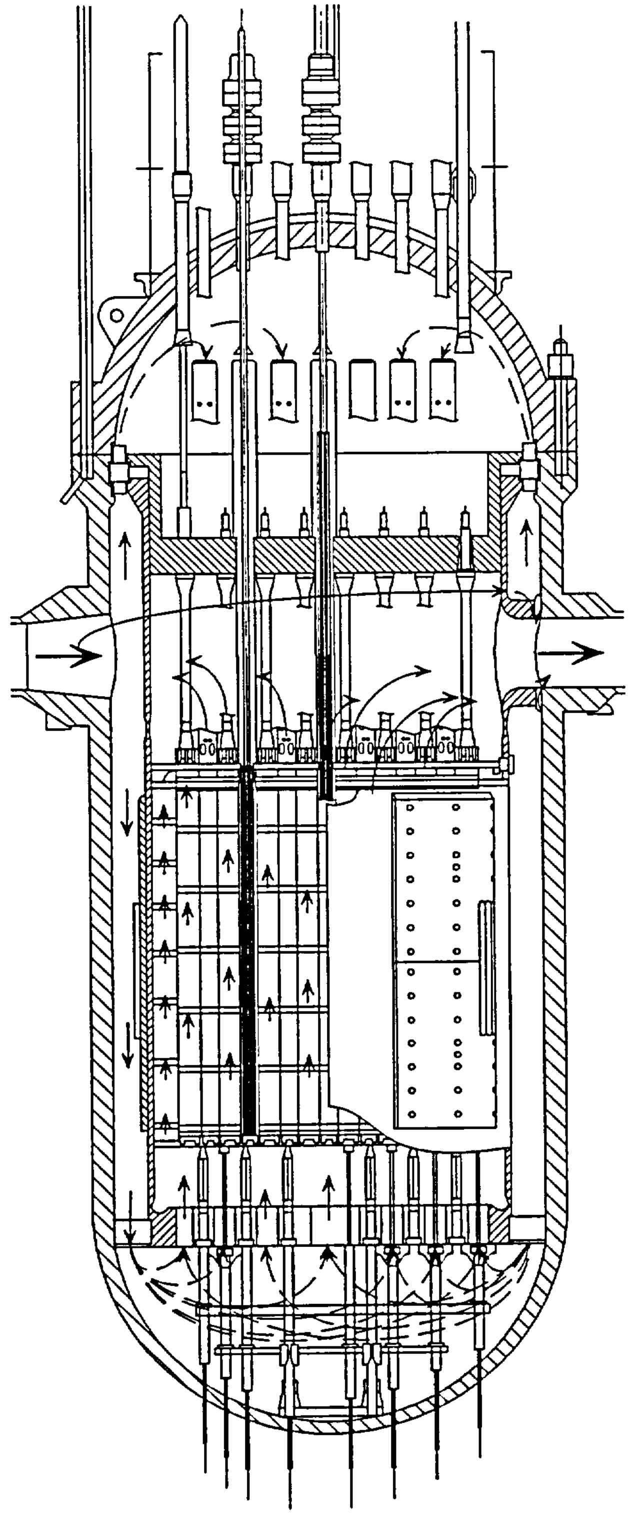


Figure 3.1-34 Core Flow Paths

Westinghouse Technology Systems Manual

Section 3.2

Reactor Coolant System

TABLE OF CONTENTS

3.2 REACTOR COOLANT SYSTEM 3.2-1

3.2.1 Introduction 3.2-1

3.2.2 System Description 3.2-1

3.2.3 Component Description 3.2-2
- 3.2.3.1 Reactor Coolant Piping 3.2-2
- 3.2.3.2 Instrumentation RCS Temperature 3.2-3
- 3.2.3.3 Pressurizer 3.2-5
- 3.2.3.4 Steam Generators 3.2-10
- 3.2.3.5 Reactor Coolant Pump 3.2-13

3.2.4 Leakage Detection 3.2-19

3.2.5 Vibration and Loose Parts Monitoring 3.2-20

3.2.6 System Interrelationships 3.2-20
- 3.2.6.1 Materials of Construction 3.2-20
- 3.2.6.2 Heat up and Cool down Limits 3.2-21
- 3.2.6.3 Natural Circulation 3.2-21

3.2.7 PRA Insights 3.2-22
- 3.2.7.1 Reactor Coolant Pump Seals 3.2-22
- 3.2.7.2 Pressurizer Power Operated Relief Valves 3.2-22

3.2.8 Summary 3.2-22

LIST OF TABLES

3.2-1 Reactor Coolant Piping Design Parameters 3.2-25
3.2-2 Pressurizer Design Parameters 3.2-25
3.2-3 Spray Control Valve Design Parameters 3.2-26
3.2-4 Code Safety Valve Design Parameters 3.2-26
3.2-5 Pressurizer Power Operated Relief Valve Design Parameters 3.2-26
3.2-6 Pressurizer Relief Tank Design Parameters 3.2-27
3.2-7 Steam Generator Design Parameters 3.2-27
3.2-8 Reactor Coolant Pump Design Parameters 3.2-28
3.2-9 Reactor Coolant Pump Motor Design Parameters 3.2-28
3.2-10 Thermal and Loading Cycle Design Parameters 3.2-29

LIST OF FIGURES

3.2-1 .... Reactor Coolant Loop Penetrations
3.2-2 .... Reactor Coolant System
3.2-3 .... Pressurizer Spray Scoop
3.2-4 .... Sample Connection Scoop
3.2-5 .... Hot Leg RTD Tap
3.2-6 .... Reactor Coolant Flow Taps
3.2-7 .... Pressurizer
3.2-8 .... Pressurizer Manway Cover and Spray Nozzle
3.2-9 .... Model 51 Steam Generator
3.2-10 .... Steam Generator Flow Paths
3.2-11 .... Feed Ring Assemblies
3.2-12 .... Feedring and Moisture Separators
3.2-13 .... Steam Generator Moisture Separators
3.2-14 .... Tube Support Plate
3.2-15 .... Steam Generator Shrink and Swell
3.2-16 .... Reactor Coolant Pump
3.2-17 .... Seal Water Injection and Leakoff
3.2-18 .... Shaft Seal Arrangement
3.2-19 .... Seal Flow Diagram
3.2-20 .... Controlled Leakage Shaft Seal
3.2-21 .... RCP Flywheel and Anti-reverse Rotation Device
3.2-22 .... Vibration and Loose Parts Monitoring Transducer Locations
3.2-23 .... Vibration and Loose Parts Monitoring System
3.2-24 .... Vibration and Loose Parts Monitoring Cabinet Arrangement
3.2-25 .... Vibration and Loose Parts Monitoring Indicator Assembly
3.2-26 .... RCS Pressure - Temperature Limits (Heatup)
3.2-27 .... RCS Pressure - Temperature Limits (Cooldown)

3.2 REACTOR COOLANT SYSTEM

Learning Objectives:

1. State the purpose of the Reactor Coolant System (RCS).

2. List and state the purpose of the following RCS penetrations:

   a. Hot Leg (T_h)
      1. Pressurizer surge line
      2. Residual Heat Removal (RHR) suction
      3. Sample line
      4. RHR recirculation/Safety Injection (SI)

   b. Intermediate Leg
      1. Elbow flow taps
      2. Chemical and Volume Control System (CVCS) letdown
      3. Loop drain

   c. Cold Leg (T_c)
      1. Pressurizer spray line
      2. CVCS charging
      3. Common injection penetration for RHR, SI, and an Accumulator
      4. High head injection
      5. Excess letdown

3. Describe the primary and secondary flow paths in the steam generator.

4. State the purpose of the following components of the reactor coolant pump.

   a. Thermal barrier heat exchanger
   b. Seal package
   c. Flywheel
   d. Anti-reverse rotation device
   e. Number 1 seal bypass valve
   f. Number 1 seal leak off valve
   g. Seal stand-pipe

5. Explain why seal injection flow is supplied to the reactor coolant pumps.

6. State the purpose of the following:

   a. Pressurizer
   b. Code safety valves
   c. Power operated relief valves (PORV)
   d. PORV block valves
   e. Pressurizer relief tank (PRT)
   f. Pressurizer spray valves
   g. Pressurizer heaters

7. Describe the methods for determining pressurizer relief and safety valve position and/or leakage.

8. Explain the following:

   a. Pressurizer spray driving force
   b. Purpose of pressurizer spray bypass

9. Explain how failure of the following components could lead to core damage.

   a. Reactor coolant pump seals
   b. Power operated relief valves

3.2.1 Introduction

The purposes of the RCS are as follows:

1. Transfer the heat produced in the reactor to the steam and power conversion systems.
2. Provide a barrier to limit the escape of radioactivity to the containment.

3.2.2 System Description

The reactor coolant piping and components are generally referred to as the reactor coolant system, even though the RCS also includes the reactor vessel and the components directly attached to the reactor vessel. The RCS consists of four similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a steam generator, a reactor coolant pump, penetrations for connection with auxiliary and emergency systems, and appropriate instrumentation. In addition, a pressurizer is connected by a surge line to one loop for pressure control. A pressurizer relief tank is provided to collect any release from the pressurizer relief and code safety valves. All RCS components are located inside the containment. A typical RCS loop is shown on Figure 3.2-1.

During normal operations, the RCS transfers the heat produced in the reactor to the steam generators where steam is produced to supply the turbine generator and other secondary system components. Hot reactor coolant exits the reactor vessel and flows through the hot leg piping to the steam generator where energy is removed to make steam. Coolant exits the steam generator and flows through the intermediate leg piping to the suction of the reactor coolant pump. The pump discharge is directed through the cold leg piping to the reactor vessel inlet nozzles to complete the cycle.

The RCS pressure boundary (described in section 3.0) acts as the second of the three barriers or "lines of defense" against fission product release to the environment. The first and third barriers are the fuel cladding and the reactor containment building respectively. Seismic restraints and supports are provided as required to ensure continued integrity of the RCS during seismic events. All RCS components are designated Seismic Category 1.

3.2.3 Component Description

3.2.3.1 Reactor Coolant Piping

The reactor coolant piping and fittings are austenitic stainless steel. The loop piping is sized to limit the maximum flow velocity for erosion or vibration considerations. The crossover line between the steam generator and reactor coolant pump is larger due to pump suction considerations.

All smaller piping which is part of the RCS boundary, such as the pressurizer surge line, spray and relief line, loop drains and connecting lines to other systems are also austenitic stainless steel. The nitrogen supply line for the pressurizer relief tank is carbon steel. All joints and connections are welded, except for the pressurizer relief code safety valves, where flanged joints are used.

Penetrations (Figures 3.2-1 and 3.2-2) which are common to all reactor coolant loops include the following:

1. Hot leg injection/recirculation from the Emergency Core Cooling Systems (ECCS).
2. Hot leg wide range, well mounted, RTD.
3. Elbow flow detector high and low pressure taps (intermediate leg).
4. Drain connection to the Reactor Coolant Drain Tank (RCDT).
5. Cold leg wide range, well mounted, RTD
6. Cold leg injection from the ECCS safety injection pumps, residual heat removal pumps and the accumulators
7. Cold leg injection from ECCS boron injection tank (high head injection)
8. Hot and cold leg taps for the RTD bypass manifolds.

Additional penetrations specific to individual loops and their typical locations are as follows:

1. Pressurizer surge line from loop 2 hot leg
2. Supply to the RHR System from loop 4 hot legs
3. CVCS letdown from loop 3 crossover leg
4. CVCS charging to loops 1 and 4 cold legs
5. Pressurizer spray from loops 2 and 3 cold legs.
6. Excess letdown to CVCS from loop 3 cold leg.
7. Sample line from loop 1 and 3 hot legs.

Thermal sleeves are installed at points in the system where high thermal stresses could develop due to rapid changes in fluid temperature during normal operational transients. These points include:

1. Charging connections from the CVCS.
2. Pressurizer end of the surge line.
3. Pressurizer spray line connection to the pressurizer.

Piping connections from process systems are made above the horizontal center-line of the reactor coolant piping, with the exception of:

1. RHR suction, which is 45° down from the horizontal centerline. This enables the water level in the RCS to be lower in the reactor coolant pipe while continuing to operate the RHR system, should this be required for maintenance.
2. Loop drain lines and the connection for temporary level measurement of water in the RCS during refueling and maintenance operation.
3. The differential pressure taps for flow measurement which are downstream of the steam generators on the 90° elbow.

The following penetrations extend into the coolant flowpath:

1. The spray line inlet connections which extend into the cold-leg piping in the form of a scoop so that the velocity head of the reactor coolant loop flow adds to the spray driving force (Figure 3.2-3).
2. The reactor coolant sample system taps are inserted into the main stream to obtain a representative sample of the reactor coolant (Figure 3.2-4).
3. The RTD hot leg taps which extend into the reactor coolant are used to obtain a representative temperature sample from each hot leg (Figure 3.2-5).
4. The wide range temperature detectors are located in RTD wells that extend into the reactor coolant pipes.

All valves in the RCS which are in contact with the coolant are constructed primarily of stainless steel. Other materials in contact with the coolant are manufactured of special materials that are non-corrosive.

All RCS valves which are three inches or larger, and that will normally be operated at a temperature above 212°F, are provided with double-packed stuffing boxes with stem lantern gland leak off connections. All throttling control valves, regardless of size, are provided with double-packed stuffing boxes and stem leak off connections. All leak off connections are piped to the PRT. Leakage to the containment is essentially zero for these valves.

3.2.3.2 Instrumentation RCS Temperature

There are two ranges of RCS temperature monitors. Both ranges use RTDs. Temperature of the hot leg and of the cold leg of each loop are monitored for indication only by wide range (0-700°F) detectors. These detectors are mounted in wells which penetrate the coolant piping and are part of the pressure boundary. These RTD's do not come in contact with the reactor coolant.

Separate bypass manifolds for each RCS hot leg and cold leg provide individual narrow range temperature signals.

The narrow range T_h and T_c temperature detectors are used for reactor control and protection and are mounted in individual bypass manifolds around each steam generator. The narrow range (530°F-650°F)T_h RTDs obtain a representative sample of the hot leg temperature by mixing the flow from three hot leg taps which extend into the flow stream 120 degrees apart. The narrow range (510°F-650°F)T_c RTD's are also mounted in the bypass manifolds and have one tap directly downstream of each reactor coolant pump. Only one connection per cold leg is required to get a representative temperature indication due to the mixing of the reactor coolant caused by the reactor coolant pump.

The information derived from these hot and cold leg RTDs (Tavg and ΔT), from all four loops, is indicated in the control room. Additional information about these temperature indicators may be found in Chapter 10.0, Primary Systems Control and Instrumentation.

RCS Flow

Elbow taps are used in the RCS to indicate the status of the reactor coolant flow (Figure 3.2-6). The function of this device is to provide information as to whether or not a reduction in flow has occurred. An advantage of this type of flow detection system is that no components need to be inserted into the coolant flowpath. Components in the flowpath produce pressure drops and either reduce flow or require increased pumping power. The elbow flow instrument measures the differential pressure between the inner and outer radius of the intermediate leg piping elbow. The outer radius will experience a slightly higher pressure due to the dynamic effects of coolant flow.

The correlation between changes in flow and elbow tap indication has been well established and is described by the following equation:

$$\frac{\Delta P}{\Delta P_o} = \frac{\varpi^2}{\varpi_0^2}$$

Where:

ΔP is the differential pressure corresponding to the present measured flow rate.

ΔP_o is the differential pressure corresponding to the initial reference flow rate, and ϖ is the flow rate

The full flow reference point (ΔP_o) is established during initial plant start-up. The low flow reactor trip set point is then established by extrapolating data from a curve which correlates ΔP to flow. This technique has been well established in providing core protection against low coolant flow in Westinghouse PWR plants. The expected absolute accuracy of the channel is within $\pm 10\%$ and field results have shown the repeatability of the trip point to be within $\pm 1\%$. The accident analysis for a loss of flow transient assumes a instrumentation error of $\pm 3\%$.

Pressurizer Pressure

Redundant pressure detectors are provided to monitor the pressure in the steam space of the pressurizer. Information from these detectors is used for indication, control, and protection. For additional information see Chapter 10.0, Primary Systems Control and Instrumentation.

Pressurizer Level

Redundant level instrumentation is provided to monitor the water level in the pressurizer. Differential pressure detectors monitor the difference between the height of a known column of water (reference leg) and the height of the water in the pressurizer.

This instrumentation is used for indication, control, and protection. A separate channel calibrated for cold plant conditions is provided for shutdown, start-up and refueling operations. Additional information may be found in Chapter 10.0, Primary Systems Control and Instrumentation.

Safety and Relief Valve Monitors

Temperature detectors and acoustic monitors on the discharge piping from the pressurizer relief and safety valves indicate leakage or opening of these valves. Each power-operated relief valve also has a stem-mounted position switch with indication in the control room.

Loop Pressure

Three pressure detectors are installed at the hot leg suction of the residual heat removal system. These instruments are wide range transmitters (0-3000 psig) which provide pressure indication over the full operating range.

One of the three pressure transmitters provides indication to the remote shutdown station only. The remaining two pressure transmitters provide indication in the main control room, a PORV actuation signal in conjunction with the low temperature over pressure mitigation system, the permissive signals for the RHR loop isolation valves interlock circuits to protect the low pressure piping of the RHR System and input to the subcooled margin monitor.

Two local pressure indicators are also provided for operator reference during shutdown conditions. One taps the loop 3 sample line and the other senses the pressure in the loop 4 to RHR suction line.

3.2.3.3 Pressurizer

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads, constructed of manganese-molybdenum alloy steel, clad with austenitic stainless steel on all surfaces exposed to reactor coolant water. Electrical heaters (78) are installed through the bottom head of the vessel, while the spray nozzle, relief, and safety valve connections are located in the top of the vessel.

The pressurizer has four basic functions:

1. Pressurizing the RCS during plant heat up,
2. Maintaining normal RCS pressure during steady state operations,
3. Limiting pressure changes during RCS transients to within allowable values and,
4. Preventing the reactor coolant system pressure from exceeding its design pressure value of 2500 psig.

During normal operations, the pressurizer is maintained at saturation conditions by electrical heaters (Figure 3.2-7). Under saturation conditions, each temperature has a corresponding saturation pressure. At nominal full power conditions, approximately sixty (60) percent of the pressurizer volume is saturated water with the remaining forty (40) percent saturated steam.

The pressurizer is connected to a loop hot leg by the surge line. Since the RCS (except for the pressurizer) is a hydraulically solid system, the pressure in the pressurizer will be maintained throughout the system. Design parameters for the pressurizer are listed in Table 3.2-2.

If the RCS were operated completely full of water any change in temperature of the reactor coolant would produce unacceptably large changes in pressure due to the density change of the coolant. Therefore, the pressurizer is attached to the reactor coolant system to provide a surge and makeup volume to accommodate these density changes.

To understand how the pressurizer maintains RCS pressure, the following explanation is provided:

> At normal system operating temperatures, water is approximately six times as dense as steam. Therefore, as the water is heated by the pressurizer electrical heaters to produce steam, a factor of six volume change occurs. For example, boiling one cubic foot of water would produce six cubic feet of steam. Since the volume in the pressurizer which is available for steam is fixed by the pressurizer level control system, the density of steam must increase. This increase in density produces a pressure increase in the RCS. Conversely if steam is condensed, a factor of six density reduction occurs, which results in a reduction in pressure. The steam inside the pressurizer responds in a manner similar to an ideal gas (pressure is proportional to density).

During steady-state operations a small number of heaters are energized to make up for ambient heat losses. The small amount of subcooled liquid which is continuously circulated from the RCS via the spray line bypass valves promotes mixing and aids in chemistry control.

Most transients affecting pressure are caused by changes in the RCS temperature. This temperature change produces changes in RCS density which forces water into (insurge) or out of (outsurge) the pressurizer.

In the case of an increase in RCS temperature, the expansion of the coolant produces an insurge into the pressurizer. This insurge is accommodated for, by compressing the steam "bubble" which results in an increase in density and temperature producing a corresponding increase in pressure. If pressure increases by a predetermined amount, the pressurizer pressure control system will modulate the spray valves to admit relatively cool water to the steam space to condense some steam. This reduces the density of the steam and limits the pressure increase.

If the RCS temperature decreases, the contraction of the coolant produces an outsurge from the pressurizer. This is accommodated for by an expansion of the steam "bubble" and a decrease in its density. This will cause a corresponding decrease in pressure. As the pressure decreases some of the saturated water in the pressurizer will flash to steam to help maintain system pressure. If pressure decreases to a predetermined value, the pressurizer pressure control system will energize additional electrical heaters to boil more water and return pressure to normal.

If the RCS pressure increases towards the design limit, power operated relief and self-actuating code safety valves open. These valves release steam from the steam space of the pressurizer and thereby limit the over pressure condition.

The volume of the pressurizer (1800 ft^3) is greater than, or equal to, the minimum volume of steam and water combination which satisfies all of the following requirements:

1. The combined saturated water volume and steam expansion volume is sufficient to provide the desired pressure response to programmed system volume changes.
2. The water volume is sufficient to prevent the heaters from being uncovered during a step load increase of ten percent of full power.
3. The steam volume is large enough to

accommodate the insurge resulting from a 50 percent reduction of full load with automatic rod control and full steam dump capacity without the water level reaching the high level reactor trip set point.

4. The pressurizer does not empty following a reactor trip with a turbine trip.
5. The steam volume is large enough to prevent water relief through the safety valves following a loss of load with the high pressurizer water level initiating a reactor trip.
6. A safety injection signal will not be activated following a reactor trip and turbine trip.

Pressurizer Heaters

The electrical heaters installed in the pressurizer are replaceable, direct immersion, tubular sheath type, and hermetically sealed. They are located in the lower portion of the pressurizer vessel and maintain the steam and water contents at equilibrium conditions. There are 78 heaters installed for a total capacity of 1794 kW. The heaters are broken down into two groups, the proportional heater group consists of 18 heaters for an output of 414 kW and the backup heater group consists of 60 heaters for an output of 1380 kW. The heaters are capable of raising the temperature of the pressurizer and its contents at approximately 55°F/hr.

The hermetically sealed terminals connected to the external pressurizer vessel wall are capable of retaining full system pressure should the immersed tubular heater sheath fail. Ventilation of the heater connections is accomplished through holes drilled in the pressurizer support skirt.

Pressurizer Spray

Spray water is injected into the steam volume of the pressurizer through a spray nozzle located in the top of the vessel (Figure 3.2-8). Automatically controlled, air-operated valves with remote manual overrides are used to control pressurizer spray from two loop cold legs. In parallel with each spray valve is a manual throttle valve which permits a small continuous flow to bypass the spray valves. This small flow (1 gpm) is provided to reduce the thermal stresses and/or thermal shock to the pressurizer spray penetration and spray nozzle inside the pressurizer when the spray valves open. In addition, this flow helps maintain uniform water chemistry in the pressurizer to that of the reactor coolant system. Design parameters for the spray valves are listed in Table 3.2-3.

Temperature sensors with low temperature alarms are provided in each spray line to alert the operator of insufficient bypass flow. The layout of the spray line piping to the pressurizer forms a water seal which prevents steam buildup back to the spray valves. The maximum spray flow rate (840 gpm) is selected to prevent the pressure in the pressurizer from reaching the operating set point of the PORVs, following a 10 percent step decrease in power.

The pressurizer spray lines and valves are sized and located to provide adequate spray using the differential pressure between the surge line connection in the hot leg and the spray line connection in the cold leg as the driving force. The spray line inlet connections (Figure 3.2-3) extend into the cold leg piping in the form of a scoop so that the velocity head of the reactor coolant loop flow adds to the spray driving force. The spray valves and spray line connections are redundant so that the spray flow can be admitted to the steam space of the pressurizer even when one of the associated RCP's is not operating. In addition to their pressure control function, the sprays may be manually operated to circulate coolant from the loop to the pressurizer for boron concentration equalization.

A flow path from the CVCS to the pressurizer spray line is also provided. This connection provides auxiliary spray to the vapor space of the pressurizer during cool down if the reactor coolant pumps are not operating. The thermal sleeve on the pressurizer spray connection is designed to withstand the thermal stresses resulting from the introduction of this relatively cold auxiliary spray water.

Surge Line

The surge line connects the bottom of the pressurizer to one of the reactor coolant system hot legs. It is sized to limit the pressure drop between the pressurizer and the RCS during the maximum anticipated insurge. It is designed in this fashion to ensure that the highest pressure point in the RCS does not exceed the design pressure with the code safety valves discharging at their maximum allowable accumulation. The surge line also contains a thermal sleeve at the pressurizer end which is designed to withstand the thermal stresses resulting from surges of relatively hotter or colder water which will occur when the temperature of the RCS changes.

A temperature sensor with remote indication and a low temperature alarm is provided on the surge line. This indication along with the spray line temperature indication monitors performance of the continuous spray bypass flow.

Code Safety Valves

The three pressurizer code safety valves are totally enclosed pop-open-type valves. The valves are spring-loaded, self-actuating and have back-pressure compensation designed to prevent the reactor coolant system pressure from exceeding the design pressure by more than 10 percent. This meets the requirements of the ASME Boiler and Pressure Code, Section III. The set pressure of the safety valves is 2485 psig. Design parameters for the pressurizer code safety valves are listed in Table 3.2-4.

A water seal is maintained below each code safety valve seat to minimize leakage. The 6" pipes connecting the pressurizer nozzles to their respective code safety valves are shaped in the form of a loop seal. Condensate, as a result of normal heat losses to ambient, accumulates in the loop and floods the valve seat. This water seal prevents leakage of steam or hydrogen gas to pass by the safety valve seats. If the pressure inside the pressurizer exceeds the set point of the code safety valves, they will lift and the water from the loop seal will discharge during the accumulation period.

Due to the high pipe and pipe support loads caused by these "water slugs", catch pots were designed and placed immediately downstream of the relief and safety valves. A total of four of these Slug Diversion Devices (SDDs) are installed (one for each of the code safeties and one for the PORV combined discharge). The SDDs are located at the change in pipe direction so that the water slugs flow into the devices and are trapped. These devices are totally passive and ensure that the piping system is not subjected to stresses or loads beyond code allowable.

A temperature indicator in the safety valve discharge manifold alerts the operator to the passage of steam due to leakage or valves lifting. Acoustic monitors are also provided for each valve to provide a positive indication of leakage or code safety valve operation.

Power Operated Relief Valves

The pressurizer is normally equipped with two PORV's which limit pressure in the reactor coolant system and minimize the probability of actuation of the high-pressure reactor trip. The operation of the PORVs also limits the operation of the fixed high-pressure code safety valves. The PORVs are air operated and can be opened or closed automatically or by remote manual control. Should the air supply to the PORVs fail, a backup air supply system is provided to maintain the PORVs operable for 10 minutes following a loss of instrument air. Remotely operated block valves are provided to isolate the PORVs if excessive leakage occurs. Design parameters for the power operated relief valves are listed in Table 3.2-5.

The PORVs are designed to limit the pressure in the pressurizer to a value below the high pressure reactor trip set point for design transients up to and including a 50 percent step load decrease with full steam dump actuation. Acoustic monitors are also installed at the outlet of the PORVs to detect leakage and/or relief opening.

The PORVs, with additional actuation logic, are also utilized to mitigate potential RCS cold over pressurization transients during cold shutdown conditions.

Pressurizer Relief Tank

The PRT collects, condenses and cools the steam discharge from the pressurizer code safety and power-operated relief valves. The tank is operated approximately three quarters full of water with a nitrogen cover gas. Steam is discharged through a sparger at the bottom of the tank below the water level. This condenses and cools the steam by mixing it with water that is near containment ambient temperature. Discharge from a number of smaller relief valves located in systems inside or outside of the containment are also piped to the pressurizer relief tank.

The design of the PRT is based on the requirement to condense and cool a discharge of pressurizer steam equal to 110 percent of the volume above the zero power pressurizer water level set point (25%). This tank is not designed to accept a continuous discharge from the pressurizer. The volume of water in the PRT is capable of absorbing the heat from the assumed discharge from the pressurizer code safety valves, with an initial water temperature of 120°F and increasing to a final temperature of 200°F. If the temperature inside the PRT rises above 114°F during plant operation, the tank is cooled by spraying in cool primary makeup water and draining the warm mixture to the reactor coolant drain tank. Design parameters for the pressurizer relief tank are listed in Table 3.2-6. The spray rate inside the PRT is designed to cool the tank from 200°F to 120°F in approximately one hour following the design discharge of pressurizer steam. The nitrogen gas over pressure inside this tank is selected to limit the maximum pressure following a design discharge to 50 psig.

The PRT is constructed of austenitic stainless steel. A flanged nozzle is provided for connection of the pressurizer relief and safety valve discharge line. The discharge piping and the sparger are constructed of austenitic stainless steel. A vacuum breaker hole is located on the discharge line inside the PRT which will allow the PRT and the discharge line pressure to equalize, preventing water in the PRT from being forced up into the discharge line. Two rupture discs are provided to prevent exceeding the design pressure of the PRT. The rupture discs have a relief capacity equal to the combined capacity of all three pressurizer safety valves. The design pressure of the PRT (and the rupture discs setting) is twice the calculated pressure produced by the design relief volume described above. This margin is provided to prevent deformation of the rupture disc.

The PRT and its connections are designed to operate under a vacuum to prevent the collapse of the tank if the contents are cooled following a discharge without adding additional cover gas. Piping penetrations are provided so that the cover gas may be sampled and analyzed for hydrogen and oxygen. A vent to the Gaseous Waste Disposal System allows venting of excess pressure or removal of any non-condensable or fission product gases which might be transferred from the pressurizer to the relief tank. A nitrogen supply connection allows initial supply and makeup of the cover gas.

3.2.3.4 Steam Generators

The steam generators are vertical shell and U-tube heat exchangers where energy from the hot pressurized reactor coolant is transferred to the secondary coolant to produce dry, saturated steam. The steam generator provides the boundary between the radioactive primary system and the non-radioactive secondary system.

Each steam generator is fitted with a ring on the inside of the hot and cold leg nozzles for the fitting of a nozzle dam. These nozzle dams are fitted robotically and allow RCS level to be raised to refueling levels while the steam generator primary side is drained. This allows more flexibility in scheduling the time consuming steam generator tube inspections. In addition it reduces the time where the RCS must be operated in a reduced inventory status with RCS water level at the loop mid point.

A cutaway diagram showing the construction of a Westinghouse model 51 steam generator may be found on Figure 3.2-9 and will be used for explanation purposes. The design data for the Steam Generators is listed in Table 3.2-7 found on page 3.2-12.

The steam generator is designed to produce saturated steam with less than .25% percent moisture by weight under the following conditions:

1. Steady-state operation at up to 100% of full load steam flow assuming that the water level in the steam generator is at program.
2. Ramp load changes at a maximum rate of 5% per minute in the range from 15 to 100% full power steam flow.
3. Step load changes of 10% of full power between 15 and 100% full power steam flow.

The steam generator is constructed of carbon steel with all surfaces in contact with reactor coolant made from or clad with appropriate corrosion resistant material. Construction and operation of both the primary (reactor coolant) and secondary (steam) sides of the steam generator are described in the following paragraphs. Primary and secondary flow paths are shown on Figure 3.2-10.

Primary (Reactor Coolant) Side

Reactor coolant enters and leaves the steam generator through nozzles in the bottom hemispherical head. The head is divided into inlet and outlet chambers by a vertical partition (divider) plate which extends from the bottom of the tube sheet to the bottom hemispherical head. Bolted, gasketed manways are provided for access to both the inlet and outlet sides of the bottom hemispherical head.

The tube sheet and the heat transfer tubes form the boundary between the primary and secondary systems. The tubes and the divider plate are manufactured of Inconel. The primary side of the tube sheet is clad with Inconel while the interior surfaces of the hemispherical head and the nozzles are clad with austenitic stainless steel. After the tube ends are seal welded to the tube sheet, they are roller expanded for the full depth of the tube sheet cladding. This is done to prevent the leakage of the high pressure reactor coolant from the primary side to the lower pressure secondary side.

Secondary (Steam) Side

The secondary side of the steam generator consists of the feed and steam nozzles, the tube bundle and supports, the tube bundle wrapper, primary and secondary moisture separators, appropriate instrumentation and blow down penetrations. The steam generator shell and its internals are constructed of carbon steel.

At 100% thermal power feedwater at a temperature of approximately four hundred thirty degrees Fahrenheit (430°F) enters the steam generator through the feedwater inlet nozzle. As the feedwater leaves the feed ring it is distributed to the annulus between the tube bundle wrapper and the steam generator shell where it mixes with the hot recirculation water from the moisture separation equipment. This annulus is called the downcomer and is where steam generator level is measured for indication, control and protection. The feed water is distributed through an upper feed ring (Figure 3.2-11) into the annulus.

The feedwater-recirculation flow mixture then flows between the bottom of the tube wrapper (shroud) and the tube sheet into the tube bundle region. In the tube bundle region, heat is transferred to the secondary coolant producing a steam-water mixture. This mixture flows upward to the primary or swirl-vane moisture separators (Figure 3.2-12). These separators consist of a number of stationary vanes which impart a swirling motion to the steam water mixture. The steam, being less dense than the water can change direction easily and passes through the swirl vanes. The water is slung to the outside and flows to the downcomer where it mixes with incoming feedwater. The water in the downcomer is maintained at a level that is equal to a height in the boiling section of the steam generator that is approximately at the bottom of the swirl vanes.

Although the swirl vanes remove most of the moisture from the steam, a second stage of moisture separation is necessary to meet design requirements. The steam passes through chevron separators (Figure 3.2-13) and then leaves the generator through the outlet nozzle. These chevron separators force the steam to take a torturous path. Again the steam can change directions easily as it passes through while the more dense water cannot. The separated moisture is collected and drained to the downcomer. Steam exiting the steam generator has a minimum quality of 99.75% (less than .25% moisture).

The steam generator operates as a natural circulation boiler. The flow from the downcomer to the tube bundle is caused by the difference in density between the slightly subcooled liquid in the downcomer and the steam-water mixture in the tube bundle. At full power, the flow in the downcomer is three (3) to five (5) times that of the incoming feedwater. This is caused by the recirculation flow from the moisture separators and is necessary to ensure proper thermal-hydraulic performance of the steam generator. The higher flow velocities also help prevent the collection of impurities at the tube to tube sheet area of the steam generator.

In order to support and align the steam generator tubes and prevent flow induced movement, tube support plates are provided (See Figure 3.2-14). Most steam generators use drilled support plates where the tubes pass through holes which are slightly larger than the diameter of the tube. Additional flow holes are drilled to provide passage for steam flow. During cold plant conditions, the gap between the tube and support plate increases due to differential expansion. Corrosion products deposit in the hole in the support plate. When the steam generator is heated, the differential expansion closes the gap. Since the hole is now smaller due to the corrosion products, the tube can actually be dented which may damage the integrity of the tube. This is known as tube denting.

To help maintain the steam generator relatively free of dissolved solids and remove corrosion products which tend to concentrate at the tube sheet, a penetration is provided to drain a portion of the steam generator water for processing. This drain is called steam generator blowdown. It removes water from just above the tube sheet and diverts it to the steam generator blow down system. This path is monitored for radiation which would indicate a steam generator primary to secondary leak. Steam generator blowdown is in service continuously during plant operations.

Steam Generator Level Instrumentation

There are two types of steam generator level indication - wide range and narrow range. Both indicators measure the water level in the downcomer. The wide range level instrument (one per steam generator) provides indication only, and monitors the level from the top of the tube sheet to a height of approximately forty-eight (48) feet.

The narrow range level instruments monitor the upper twelve feet of the wide range span. Therefore, if the narrow range instrument indicates 0% level, the wide range should indicate 75% level. The narrow range indicators are used for indication, control, and protection.

There are three narrow range channels and each provides indication in the control room. The steam generator water level control system utilizes a dedicated level channel in the automatic feed water control system. For protection, a two out of three logic provides a low-low steam generator level reactor trip for loss of heat sink protection, and a high-high steam generator level turbine trip to protect the turbine against excessive moisture carryover.

Steam Generator Operational Characteristics

It is a characteristic of U-tube steam generators that steam pressure decreases with increasing load or steam flow. The rate of heat transfer across the tubes is approximated by the following equation:

$$Q = UA\Delta T$$

Where:

Q = The rate of heat transfer in BTU/hr.
U = Heat transfer coefficient
A = Total heat transfer area
ΔT = Temperature difference across the tubes which is approximately equal to the average temperature of the reactor coolant (Tavg) minus the saturation temperature of the steam (T_{sat}) inside the steam generator.

The rate of heat transfer (Q) is determined by the plant load. The heat transfer coefficient (U) can be assumed to be constant since it is a function of the materials of construction. The heat transfer surface (A) is also a constant given that the tubes remain covered at all power levels. Therefore, in order to transfer more energy across the tubes, the ΔT must increase. It would be desirable to accomplish this increase in $\Delta T$by raising Tavg but there are reactor thermal limits which are limiting. Therefore, Tavg is increased by the rod control system to gain part of the required ΔT increase, but the additional increase in ΔT must come from a reduction in T_{sat}. As T_{sat} decreases it causes a corresponding decrease in steam pressure because the steam generator operates as a saturated system. Steam pressure decreases approximately one hundred fifty (150) psi from no-load to full- load.

As the secondary load is increased, the number and size of the steam bubbles in the steam generator increases. This reduces the density and increases the specific volume of the mass in the steam generator. If mass were held constant, steam generator level would increase unacceptably. The steam generator water level control system maintains a programmed level during power operation (see Chapter 11.1).

In order to maintain level constant with increasing load, the mass in the generator must decrease. Due to this reduction in mass at high power, a plant trip produces a large reduction in level as the steam bubbles collapse and the specific volume of the steam generator secondary fluid decreases. As described above it should be understood that there is more mass of secondary coolant in the steam generator at no load than at full power. Therefore, when the plant is at no load it will be the worst case load for steam line break analyses.

Shrink and Swell

Another characteristic of a U-tube steam generator is the phenomenon of "shrink and swell." Referring to Figure 3.2-15, at time 01 steam flow is decreased. This results in a rapid increase in steam pressure. This causes the steam bubbles in the tube bundle region to collapse and the saturation conditions to change. The boiling rate in this region decreases because of the pressure increase. Both affects result in the inability of the mass in the tube bundle region to hold up the level in the downcomer, and a drop in the measured level occurs. The continued heat input from the primary and the drop in heat removal from the generator combine to raise the temperature in the generator to a new saturation condition and boiling increases. The increase in bubble formation results in the ability of the mass in the tube bundle region to hold up the level in the downcomer and level starts to return to setpoint. The response of the feed water system causes level to oscillate, but eventually returns to setpoint.

With an increase in steam flow, at time 02, and resultant decrease in steam pressure the opposite effect, swell, will occur. Again the saturation conditions in the generator are affected. The more rapid the change in steam flow, the more pronounced will be the affect of shrink and swell on generator level.

At low power levels, a similar effect is produced by rapid changes in feed flow. The heat input rate from the primary is low compared to that at high power levels. This limits the ability of the steam generator to compensate for rapid changes on the secondary side. For example, a rapid increase in feed water flow will put more relatively cold water in the steam generator than the primary can heat up in a short time period. This results in a drop in the temperature of the steam generator and a reduction of the boiling rate in the tube bundle region. Again the ability of the mass in the tube bundle region to hold up the level in the downcomer is affected and measured level drops.

As the water is heated and boiling begins to occur, there is an expansion of the additional mass in the steam generator, the water level in the downcomer will rise to some higher value.

3.2.3.5 Reactor Coolant Pump

The reactor coolant pump provides sufficient forced circulation flow through the core, to ensure adequate heat transfer to maintain a departure from nucleate boiling ratio (DNBR) of greater than 1.3. The required net positive suction head (NPSH) for the reactor coolant pumps is always less than that available by system design and operating limits. Design parameters for the reactor coolant pumps are listed table 3.2-8.

To ensure adequate core cooling after a loss of electrical power to the RCPs, each pump is designed with a flywheel that is attached to the top of the pump motor. If the reactor trips on a loss of flow due to a loss of power to the reactor coolant pumps, the flywheels will extend the coast-down time to maintain adequate heat transfer capability and help establish natural circulation flow. Power to the reactor coolant pump motors is from the non-vital station service power and cannot be supplied from the emergency diesel generators. The RCPs are capable of operation without mechanical damage at speeds of up to one hundred twenty-five percent (125%) of normal speed. Periodic surveillances are performed on the flywheel to ensure its integrity.

The reactor coolant pump is a vertical, single-stage, centrifugal pump designed to pump large volumes of reactor coolant at high temperatures and pressures. A cutaway of a typical reactor coolant pump is shown on Figure 3.2-16. The pump consists of three sections from bottom to top:

1. The hydraulic section consists of the inlet and outlet nozzles, casing, flange, impeller, diffuser, pump shaft, pump bearing, thermal barrier and thermal barrier heat exchanger.
2. The shaft seal section consists of the number one, controlled leakage seal and the numbers two and three rubbing face seals. These seals are located within the main flange and seal housing.
3. The motor section consists of a vertical, squirrel cage, induction motor with an oil lubricated double Kingsbury thrust bearing, two oil lubricated radial bearings, and a flywheel with an anti-reverse rotation device and appropriate support equipment.

Hydraulic Section

Casing - The pump casing contains and supports the hydraulic section of the pump and is part of the reactor coolant pressure boundary. The casing is a 304 stainless steel casting whose nozzles are welded to the RCS piping. The entire weight of the pump and motor is supported by pads attached to the casing. This support system is designed to allow for thermal expansion of the RCS.

Impeller - The impeller is designed to impart energy to the reactor coolant. This energy is added in the form of kinetic energy (velocity head). The impeller is a stainless steel casting which is shrunk and keyed to the lower end of the pump shaft. An impeller nut is then threaded and locked to the shaft. The impeller is designed for counter clockwise rotation as viewed from the top of the pump.

Diffuser and Turning Vanes - The diffuser, which is located above the impeller, converts the velocity head generated by the impeller to a static head by reducing the fluid velocity in the expanding flow channels between the diffuser vanes. The flow is then directed to the outlet nozzle by the turning vanes. The diffuser and turning vanes are constructed of stainless steel.

Radial Bearing Assembly - The pump bearing is a self-aligning, spherical, graphitar-coated, journal bearing. The bearing provides radial support and alignment for the pump shaft and is water cooled and lubricated. It is essential that the water circulating through the bearing be kept cool. High temperatures will damage the graphitar coating and cause bearing failure. This cool water is normally supplied by seal injection from the CVCS as described in Chapter 4.0.

Thermal Barrier and Heat Exchanger - The thermal barrier assembly consists of the thermal barrier and thermal barrier heat exchanger. The thermal barrier is designed to reduce the rate of heat transfer from the hot reactor coolant to the pump radial bearing and thermal barrier heat exchanger. It consists of a number of concentric stainless steel cylinders extending vertically from the top of the impeller to the thermal barrier flange, and a number of stacked horizontal plates at the flange. The barrier to heat transfer is provided by the gap between the cylinders and plates. The thermal barrier heat exchanger is located at the bottom of the thermal barrier assembly below the pump radial bearing. The function of this heat exchanger is to cool any reactor coolant leaking up the shaft to protect the radial bearing and shaft seals.

Seal injection water is normally supplied to the reactor coolant pump from the CVCS (Figure 3.2-17). This water is injected into the pump at a point between the radial bearing and the thermal barrier heat exchanger. Of the eight (8) gallons per minute total injection flow to each RCP, three (3) gpm flows upward through the radial bearing and pump seals and five (5) gpm flows downward through the heat exchanger and into the RCS. This downward flow acts as a buffer to prevent the hot reactor coolant from entering the bearing and seal area.

The reactor coolant pump is designed to operate with either seal injection or thermal barrier heat exchanger cooling. However, it is desirable to maintain bearing and seal cooling from the purified and filtered seal injection water rather than the contaminated unfiltered reactor coolant leaking up the shaft from the RCS. The thermal barrier heat exchanger is used as a backup if a loss of seal injection flow were to occur. Under this condition approximately three (3) gpm (the normal shaft seal leakage) flows from the RCS through the heat exchanger to the pump radial bearing and the controlled leakage seal package. Operation of the reactor coolant pump under these conditions is permitted only for a short period of time due to the fact that the unfiltered coolant flowing through the seal package could damage the seals. Labyrinth seals between the shaft and heat exchanger force most of this water through the heat exchanger. Component cooling water is the cooling medium for the thermal barrier heat exchanger.

Coupling/Spool Piece - A spool piece connects the pump and motor shafts. The spool piece can be removed to make the shaft seals accessible for maintenance without removing the motor.

Shaft Seal Section - The function of the shaft seal assembly is to provide essentially zero leakage from the RCS along the pump shaft to the containment atmosphere during normal operating conditions. The assembly consists of three seals, two of which are full design pressure seals and a third which is simply a leakage diversion seal. Figure 3.2-18 shows the relative position of the three seals. The seal assembly is located concentric to the pump shaft as it passes through the main flange. The seals are contained in a seal housing which is bolted to the top side of the main flange.

Number One Seal - The number one seal (Figure 3.2-18) is the main seal of the pump. It is a controlled-leakage, film riding seal whose sealing surfaces do not contact each other. Its primary components are a runner which rotates with the shaft and a non-rotating seal ring. The seal ring and runner are faced with an aluminum oxide coating. If the two surfaces come in contact during operation, the seal will be damaged and excessive leakage will result. The number one seal produces a 2200 psi pressure drop.

During normal operation, cool injection water at a pressure greater than RCS pressure, enters the pump through a connection on the thermal barrier flange at a rate of about 8 gpm (Figure 3.2-19). About 5 gpm of this injection water flows downward through the thermal barrier/heat exchanger and into the RCS.

This downward flow of water prevents the primary coolant from entering the seal area of the pump. The remaining 3 gpm of the injected water passes through the pump radial bearing and number one seal. The seal is termed a "controlled-leakage" seal because the leakage through the seal is controlled to a design value and is maintained by floating the seal ring so that the gap between the non-rotating seal ring and the rotating seal runner is always held to a constant value (Figure 3.2-20).

To understand the concept of why the gap between the seal ring and the runner stays constant, it is necessary to examine the hydrostatic forces on the seal ring by dividing them into "closing forces" (those forces tending to close the gap) and "opening forces" (those forces tending to open the gap). A constant closing force proportional to the pressure differential across the seal is imposed on the upper surface of the ring. This is shown on Figure 3.2-20, as a rectangle on the force balance curve.

At equilibrium conditions, an equal and opposite opening force acts on the bottom of the ring. The non-uniform shape of the opening force area is due to the taper on the underside of the ring. The taper causes the rate of change of the pressure drop, and thus the associated force, to be different from those in the parallel section of the ring. If the gap closes the seal ring moves downward and the percentage reduction of flow area in the parallel section will be greater than that in the tapered section. This will cause the resistance to flow in the parallel section to increase more rapidly than it does in the tapered section. This, in turn, will distort the force diagram and give a slight increase in the opening force. The increased opening force will push the seal ring up which causes the gap to widen until equilibrium conditions are again established. A similar discussion will show if the gap increases, the opening force will decrease. The closing force (being greater) will then push the seal ring down and close the gap. Again, the seal ring will be restored to its equilibrium position.

If the pressure in the primary system is decreased, the shape of the force balance diagram will not change. However, the actual value of the forces will decrease. If the pressure in the RCS continues to decrease, the weight of the seal ring becomes a large part of the "closing" force.

At pressure differentials below about 200 psid, the hydrostatic lifting forces become insufficient to float the seal ring, and contact between the seal ring and its runner may occur causing damage to both rings. Therefore to prevent damage to the number one seal, it is not permitted to operate the pump with the number one seal differential pressures less than 275 psid. The minimum required differential pressure of 275 psid should be obtained when the RCS pressure is 400 psig.

The flow rate through the seals at lower RCS pressures decreases to a value less than that required to cool the pump radial bearing. A penetration is provided to bypass some flow around the number one seal when the pressure in the RCS is less than 1500 psig (Figure 3.2-19). This ensures adequate radial bearing cooling flow.

Number Two and Three Seals - The number two seal is a rubbing face type seal consisting of a graphitar faced seal ring which rubs on an aluminum oxide coated stainless steel runner. During normal operation, the number two seal directs the leakage from the number one seal to the CVCS (Figures 3.2-18 and 3.2-19).

The function of the number two seal is to act as a backup in case of number one seal failure. The number two seal has full operating pressure capability. If the number one seal fails, it will pass greater amounts of water. This is sensed by leak off flow detectors which are indicated and alarmed in the control room. The operator should then shut the number one seal leak off flow control valve. This directs all number one seal leakage through the number two seal placing it into service as the primary seal. The plant should then be shut down using normal procedures to replace the failed seal. Normal leakage through the number two seal (number one seal not failed) is three (3) gph.

The number three seal is a rubbing face seal similar to the number two seal except that it is not designed for full RCS pressure. It is provided to divert the leakage from the number two seal to the RCDT. The number two seal leak off is directed to a stand-pipe which maintains sufficient back pressure to ensure flow through the number three seal for cooling purposes. The leak off from the number two seal is then piped from the top of the stand-pipe to the reactor coolant drain tank. High and low level alarms on the stand-pipe alert the operator to malfunctions of the number two and three seals. Number three seal leak off is also routed to the RCDT. Normal leakage through the number three seal is 100 cc/hr.

The primary components of the number three seal are a 304 stainless steel rotating runner with a chrome-carbide coated rubbing face and a Graphitar 114 stationary ring which is fit to a 304 stainless steel holding ring. The operation of the seal package provides a near zero leakage from the RCS at the reactor coolant pump shaft.

Instrumentation - Temperature detectors are provided to monitor seal water inlet temperature to the pump bearing and the number one seal outlet temperature. These are indicated and annunciated in the control room.

Differential pressure across the number one seal is also indicated and annunciated in the control room to ensure minimum ΔP for pump operation (Figure 3.2-21). This ensures sufficient gap between the number one seal ring and its runner. Each RCP seal supply has a flow transmitter and flow indicator followed by a seal injection throttle valve, all are located outside containment.

The number one seal leak off flow is monitored, recorded and annunciated in the control room. A high leak off flow indicates a failed number one seal and alerts the operator to close the number one seal leak off valve to place the number two seal in service. A low flow is usually produced by low RCS pressure and indicates that insufficient seal leak off exists to ensure proper cooling for the pump bearing. The operator should then open the number one seal bypass valve (a common valve for all pumps) to increase the leak off and provide sufficient cooling.

Motor Section - The motor (Figure 3.2-16) is a vertical, six-pole, squirrel-cage induction motor of drip-proof construction. It is equipped with upper and lower radial bearings, a double Kingsbury type thrust bearing, flywheel, anti-reverse rotation device, oil coolers, and appropriate instrumentation and support equipment. The power supply for reactor coolant pump motors is from the non-vital plant electrical distribution system. Design parameters for the RCP motors are listed in Table 3.2-9.

Flywheel - In the case of loss of power to the reactor coolant pumps, it is necessary to maintain a relatively high flow through the reactor core for a short while after reactor shutdown. Each reactor coolant pump has a 13,200 lb. flywheel keyed to the top of its shaft above the motor upper bearings. The stored energy in the flywheel increases the total inertia extending the pump coast-down period by 22 to 30 seconds. This, in conjunction with the reactor protection system, assures adequate heat removal during a plant trip and loss of power to the RCPs. The flow coast-down provided by the flywheel also assists in initiating natural circulation flow. The flywheel is shown on Figures 3.2-16 and 3.2-21.

Anti-Reverse Rotation Device - If one reactor coolant pump is de-energized while the remaining pumps continue to operate, reverse flow will occur in the idle loop (the loop with the RCP that is off). This reverse flow is not available for core cooling because it bypasses the core. The reverse flow also would cause the pump to turn backwards. Although this would produce no mechanical damage, an attempt to start the pump would produce excessively high starting currents which could overheat or otherwise damage the motor.

To prevent the pump from turning backwards, an anti-reverse rotation device is provided (Figure 3.2-21). The anti-reverse mechanism consists of eleven ratchet pawls mounted in the bottom of the flywheel at its outside diameter, a serrated ratchet plate that is mounted to the motor frame, return springs and two shock absorbers for the ratchet plate are also mounted to the motor frame. After the pump has stopped, one pawl will engage the ratchet plate. As the motor begins to rotate in the reverse direction, the ratchet plate will also rotate slightly until stopped by the shock absorbers. When the motor is started, the return springs return the ratchet plate to its original position.

As the motor speed increases, the pawls drag over the ratchet plate until the motor reaches approximately 80 RPM. At this time, centrifugal force and friction keep the pawls in an elevated position.

Thrust Bearing, Upper Radial Bearing, and Oil Lift System - The upper bearing is a combination double Kingsbury type thrust bearing (suitable for up or down thrust) and a radial guide bearing (Figure 3.2-16). The babbitt-on-steel thrust bearing shoes are mounted on equalizing pads, which distribute the thrust load to all shoes equally.

The radial bearing is of conventional, babbitt-on-steel design. Both the thrust and radial bearings are located in the upper oil pot and are submerged in oil. During operation, the bearings are self-lubricating and require no external oil pump. Oil is circulated from the bearings to an external oil cooler by means of drilled passages in the thrust runner which act as a centrifugal pump. Component cooling water provides the heat sink for the bearing cooler.

The thrust bearing oil lift system is provided for use during pump starts to reduce starting currents and prevent damage to the thrust bearing (the thrust bearing is only self-lubricating at relatively high pump speeds). A small, high-pressure pump provides an oil film to lift the thrust shoes away from the thrust runner. An interlock prevents starting the pump unless proper oil lift pressure has been present for a preset time. The oil lift pump is secured after the pump has been running for about a minute and is not required for pump shutdown.

During operation, the thrust is carried by the upper thrust bearing. This load is due to both RCS pressure and pump dynamic forces. The only time the lower thrust shoes carry load is when the RCS pressure is at a pressure less than the minimum allowable for pump operation or when the motor is run uncoupled from the pump.

Motor Lower Radial Bearing - The motor lower radial bearing is also a conventional, babbitt-on-steel design and is immersed in oil in the lower oil pot. A cooling coil supplied by component cooling water is provided in the oil pot.

Motor Air Cooling - The motor windings are air-cooled. Integral vanes on each end of the rotor force air through cooling slots in the motor frame.

It is then discharged to the containment atmosphere. Some designs are supplied with air/water heat exchangers for additional cooling capacity.

Instrumentation - The motor is provided with various instruments which may provide indication in the control room and in most cases will provide annunciation. The following list includes some of these instruments:

1. Temperature detectors are provided for stator temperature and bearing temperatures.
2. Oil level switches are provided for both the upper and lower oil pots and alarm in the control room on low level.
3. Vibration monitors are installed to detect pump misalignment or mechanical problems.
4. Ammeters are provided for each pump to monitor motor starting and running currents.

Cooling Water - It is essential that cooling water be supplied to the motor bearing coolers and thermal barrier heat exchanger during pump operation. Although it is possible to operate the pump without damage with no cooling flow to the thermal barrier heat exchanger, operation under these conditions should be minimized. If seal injection were lost while thermal barrier cooling was not available, hot reactor coolant would leak up the shaft into the bearing and seal area and damage these components.

The component cooling water system supplies the reactor coolant pump heat exchangers. The piping to the thermal barrier heat exchanger is designed to withstand full system pressure in case of a leak in the heat exchanger. The remainder of the system is low pressure piping.

In the event of a leak from the RCS into the thermal barrier heat exchanger, a high flow is sensed in the component cooling return line. This initiates an alarm and automatically isolates the return. Isolation of the return stops the leak flow and the high pressure piping of the component cooling system becomes part of the RCS pressure boundary. Component cooling water to the reactor coolant pumps is automatically isolated only by a containment isolation phase B signal.

If component cooling water is unavailable, the reactor coolant pumps must be secured within approximately two minutes. Additional information concerning the component cooling water system may be found in Chapter 14.0, Cooling Water Systems.

3.2.4 Leakage Detection

Detection of leakage from the RCS is accomplished by a number of diverse systems and methods. Several of these are described below:

1. The most sensitive indication of reactor coolant system leakage is the containment air monitoring system. Experience has shown that the particulate activity in the containment atmosphere responds very rapidly to increased leakage. Systems are provided to monitor particulate and gaseous activity from the areas enclosing the reactor coolant system components so that any leakage from them will be easily detected.
2. Any leakage will cause an increase in the amount of makeup water required to maintain a normal level in the pressurizer. The makeup rate is monitored and any unexplained increase indicates a leak. The magnitude of the leak can be calculated by measuring the increased makeup requirements.
3. Leakage through the head to vessel closure joint will result in flow to the leak-off between the double gaskets of the closure and produce a high temperature alarm.
4. Other methods of detecting leakage in the

containment are containment pressure, temperature, humidity, containment condensate measurement, visual inspection, ultrasonic inspection and the containment sump water level. Primary-to-secondary system leakage is detected by the air ejector and steam generator blow down monitors as well as by chemical analysis of secondary water samples.

3.2.5 Vibration and Loose Parts Monitoring

The purpose of the Vibration and Loose Parts Monitoring System (VLPMS) is to detect the presence of unexpected impacts within the boundary of the RCS and to actuate alarms to warn plant operators of the abnormal condition. The VLPMS equipment is installed to guard against inadvertent operation of the RCS with inadequately secured parts or drifting metallic parts within the coolant system.

The major components of the VLPMS are accelerometer sensors, preamplifiers, indicator assembly, audio monitor, FM cassette tape recorder, X-Y plotter and a chart recorder with selector switches.

Accelerometer sensors - There are 12 sensors installed at various locations (Figure 3.2-22) on the external surfaces of RCS components and piping. The purpose of these sensors is to monitor the sonic outputs of the components and piping(Figures 3.2-23, 24, 25).

Preamplifiers - A preamplifier is installed in the circuitry from each accelerometer sensor to boost the sensor output signal before its transmission to signal processing and alarm instrumentation in the control room.

Indicator assembly - The indicator assembly receives amplified signals from the preamplifiers. Performing the amplification, signal conditioning, and detection, assures that only valid metallic impact signals will provide alarm indications.

Audio monitor - The audio monitor is an amplifier and speaker assembly that provides an audible means of detecting vibration or loose parts at each of the 12 sensor locations.

Cassette tape recorder - The FM tape recorder provides on-line recording of signals from any of the 12 sensors.

Spectrum analyzer - The spectral analyzer operates in the low frequency or audio range and measures the amplitude and frequencies of the spectral components of an input signal which allows a more detailed analysis of sensor signals.

X-Y plotter - The plotter may be used to produce graphic tracings showing the relationship between two variable functions.

Chart recorder - The recorder is a dual pen recorder capable of recording an appropriate voltage that represents any type of variable data.

3.2.6 System Interrelationships

3.2.6.1 Materials of Construction

Each of the materials used in the RCS is selected for the expected environment and service conditions. All RCS materials which are exposed to the coolant are corrosion-resistant. They consist of stainless steels and Inconel, and they are chosen for specific purposes at various locations within the system for their compatibility with the reactor coolant.

The phenomena of stress-corrosion cracking and corrosion fatigue are not generally encountered unless a specific combination of conditions is present. The necessary conditions are a susceptible alloy, an aggressive environment, stress, and time. Therefore, to minimize or prevent stress-corrosion of the RCS components, strict chemistry specifications must be maintained to control the aggressive environment of the reactor coolant.

3.2.6.2 Heat up and Cool down Limits

The maximum heat up and cool down limits are based upon the total stress and temperature of the reactor vessel. There is a temperature below which the metal of the reactor vessel may experience brittle fracture. Brittle fracture is the catastrophic failure (cracking) of a material with little or no plastic deformation. Conditions necessary to produce brittle fracture are a susceptible material, an existing flaw (stress riser), low temperature, and stress.

The nil-ductility transition (NDT) temperature of the reactor vessel material opposite the core is established at a Charpy V-notch test value of 30 ft-lb or greater. The material is tested to verify conformity to specified requirements and to determine the actual NDT temperature value. In addition, this material is 100 percent volumetrically inspected by ultrasonic test using both straight beam and angle beam methods.

The transition temperature is increased by exposure to radiation and this must be accounted for in setting operating limits. Stresses on the reactor vessel wall are produced by reactor coolant system pressure and the differential temperature between the inner and outer walls which exists during heat up and cool down. The pressure stresses upon the reactor vessel are tension stresses. The stresses due to heat up cause a compression stress on the internal wall and tension stress on the outer wall.

Therefore, the heat up induced stresses subtract from the pressure stresses on the internal wall of the vessel. This results in a lower overall stress on the inner wall. The stresses due to cool down cause a tension stress on the inner wall and compression stress on the outer wall. Therefore, the cool down induced stresses add to the pressure stresses on the inner wall of the vessel. This results in an overall stress which is larger than the pressure-only stress on the inner wall.

To limit the maximum stress on the vessel to within allowable limits, pressure and cooldown rates must both be limited. Typical pressurization and cooldown rate limit curves are shown on Figures 3.2-26 and 3.2-37. The curves are a plot of allowable pressure vs. temperature and heatup or cooldown rate. Permissible operation is below and to the right of the heatup and cooldown limits. The criticality limit on Figure 3.2-26 defines a minimum temperature at which criticality may be achieved in order to limit pressure excursions which could occur with a critical reactor.

In actual operation, other limits prevent the reactor from being made critical until operating temperatures are achieved. Typical design thermal and loading cycles for the Reactor Coolant System are shown on Table 3.2-10 found on page 3.2-25.

3.2.6.3 Natural Circulation

It is essential to ensure sufficient flow to remove reactor decay heat even when reactor coolant pumps are not operating. The steam generators are located higher than the reactor vessel which will produce a thermal driving head to establish and maintain flow in the RCS when heat is removed from the steam generators by dumping steam. Natural circulation flow is sufficient only for decay heat removal of a shutdown reactor, not for power operation.

3.2.7 PRA Insights

3.2.7.1 Reactor Coolant Pump Seals

The purpose of the reactor coolant pump seals is to control the leakage of the reactor coolant pumps to basically zero leakage to the containment. A failure of the reactor coolant pump seals is a small break loss of coolant accident, which requires the high-head injection system to be operable to supply core cooling, and later the recirculation system to provide the long-term core cooling.

For PRA accident sequences the failure of the reactor coolant pump seals is a result of the loss of component cooling water initiator and the loss of all ac power initiator. Both of these initiators result in a loss of the pump seals, followed within a certain amount of time by core damage due to the unavailability of the high-head injection and/or recirculation system. The time to core damage is very dependent upon the assumed size of the seal LOCA, and the time between the seal LOCA occurring and the time of recovery of the high pressure injection flow. The causes of a loss of ac power and the loss of component cooling water are discussed in the appropriate chapters.

3.2.7.2 Pressurizer Power Operated Relief Valves

The pressurizer power operated relief valves are used to limit the primary pressure on transients in order to avoid a reactor trip on high RCS pressure and lifting of the code safety valves. The pressurizer PORVs are also used to remove the heat from the core if no other methods of heat removal are available.

The failure of the pressurizer power operated relief valves are present in several accident sequences which lead to core damage. There are two general failure modes for the relief valves. First, the failure of the power operated relief(s) to shut when required leads to the need for recirculation cooling of the reactor, and the subsequent failure of the recirculation mode of the emergency core cooling system results in core damage. The second failure is the failure to open when required for the purpose of initiating feed and bleed cooling for the reactor. This failure of heat removal results in core damage. At Surry, the failure of the PORV's to shut and, therefore, requiring recirculation cooling, are present in the accident sequences which are 21.2% of the core damage frequency. The failure of the PORV's to open which limits the bleed and feed capability are 4.4% at Surry and 3.2% at Sequoyah. The values for bleed and feed capability will be affected by whether the plant requires one or two relief valves to open to provide sufficient flow for bleed and feed.

Probable causes of a loss of the power operated relief valves are :

1. Failure of the PORVs to open on demand
2. Failure of the block valve to shut to isolate a stuck open relief valve, or
3. Failure of the power supply to the PORVs.

NUREG-1150 studies on Importance Measures have shown that the power operated relief valves are not a major contributor to Risk Achievement or Risk Reduction.

3.2.8 Summary

The reactor coolant piping and components circulate the reactor coolant to transfer the energy produced in the reactor to the steam system to produce electrical output from the turbine generator. The reactor coolant piping is part of the primary pressure boundary and is considered the second line of defense to the release of fission products to the environment. The first line of defense is the fuel cladding and the third is the reactor containment building. Appropriate supports and seismic restraints are provided to ensure integrity of the RCS pressure boundary during normal and accident operations including the effects of the design basis earthquake.

The pressurizer, with heaters, sprays, relief and safety valves is used to pressurize the RCS during plant start-up; maintain normal RCS pressure during steady state operation; limit pressure changes during RCS transients; and prevent RCS pressure from exceeding design values. Pressurizer pressure, level and temperature instrumentation are provided for indication, control and protection.

The steam generators are vertical, shell and U-tube heat exchangers which transfer energy from the radioactive reactor coolant to the non-radioactive steam system. The dry, saturated steam produced is used to operate the main turbine and auxiliaries to produce electricity. The steam generator U-tubes provide the boundary between the primary and secondary systems. Steam generator level, steam flow, feed flow and steam pressure instrumentation is provided for indication, control and protection.

The reactor coolant pumps provide sufficient flow to ensure adequate heat transfer from the RCS to the steam generators. A flywheel extends flow coast-down after a loss of power to maintain flow through the core and help establish natural circulation. An anti-reverse rotation device prevents the pump from turning backwards which would increase core bypass flow and pump starting current. Shaft sealing is accomplished by a film-riding, controlled-leakage seal with a backup, rubbing-face seal.

Seal injection from the CVCS, or flow from the RCS through the thermal barrier heat exchanger, cools and lubricates the seals and pump bearing. Cooling for the motor bearings and thermal barrier heat exchanger is from the component cooling water system. Electrical power to the pump motors is from a non-vital supply. Reactor coolant piping is constructed of stainless steel and is sized to limit maximum flow velocities and provide proper pump suction characteristics. Penetrations which could experience large thermal stresses have thermal sleeves.

Reactor coolant system temperature instrumentation consists of well-mounted and direct immersion RTDs. The direct-immersion detectors are located in the RTD bypass manifolds. These instruments are used to supply indication of Tavg and ΔT.

Reactor coolant flow is measured by elbow flow monitors which do not introduce a pressure drop into the RCS. This flow signal is used for indication and protection. Leakage detection is accomplished by: containment activity monitors; observation of RCS makeup requirements; containment sump level, pressure, temperature and humidity; and steam generator activity. Heatup and cooldown limits maintain the reactor vessel total stresses to within limits, especially during low temperature operation, when the vessel might be susceptible to brittle fracture. The transition temperature from brittle to ductile fracture increases with radiation dose and limits become more restrictive as the plant ages.

The orientation of the major components in the RCS is designed so that a thermal driving head will exist to produce natural circulation flow if the RCPs are not operating. Natural circulation flow is sufficient for removal of reactor decay heat.

Table 3.2-1
Reactor Coolant Piping Design Parameters

Inlet piping - inside diameter	27.5"
Inlet piping - nominal thickness	2.69"
Outlet piping - inside diameter	29.0"
Outlet piping - nominal thickness	2.84"
RCP suction piping - inside diameter	31.0"
RCP suction piping - nominal thickness	2.99"
Design pressure	2,485 psig
Design temperature	650°F

Table 3.2-2
Pressurizer Design Parameters

Design pressure	2,485 psig
Hydrostatic test pressure	3,107 psig
Design Temperature	680°F
Volume	1,800 ft^3
Full power water volume	1,080 ft^3
Full power steam volume	720 ft^3
Shell - inside diameter	84.0"
Surge line - inside diameter	14.0"
Surge line - wall thickness	1.40"
Surge line design temperature	680°F
Heater capacity	1,794 kW

Table 3.2-3
Spray Control Valve Design Parameters

Number of spray valves	2
Design pressure	2,485 psig
Design temperature	650°F
Design flow rate per valve	420 gpm

Table 3.2-4
Code Safety Valve Design Parameters

Number of code safety valves	3
Set pressure	2,485 psig
Design temperature	680°F
Relieving capacity per safety valve	420,000 lb/hr
Normal backpressure	3 psig
Maximum backpressure during discharge	500 psig
Accumulation	3%
Blowdown	5%

Table 3.2-5
Pressurizer Power Operated Relief Valve Design Parameters

Number of power operated relief valves	2
Design pressure	2,485 psig
Design temperature	680°F
Relieving capacity per relief valve	210,000 lb/hr

Table 3.2-6
Pressurizer Relief Tank Design Parameters

Volume	1,800 ft^3
Design pressure	100 psig
Design temperature	340°F
Rupture disc release pressure	100 psig
Rupture disc relieving capacity	$1.6x10^6$ lb/hr

Table 3.2-7
Steam Generator Design Parameters

Number of steam generators	4
Design pressure RCS side	2,485 psig
Design pressure steam side	1,185 psig
Design temperature RCS side	650°F
Design temperature steam side	600°F
Height	67.75'
Shell - outside diameter	175.75"
U-Tubes - number	3,388
U-Tubes - number	0.875"
U-Tubes - number	0.050"
Steam generator conditions at full load	
Steam flow	$3.77x10^6$ lb/hr
Steam pressure	895 psig
Steam temperature	533.3°F
Moisture carryover - weight	0.25%

Table 3.2-8
Reactor Coolant Pump Design Parameters

Number of reactor coolant pumps	4
Design pressure	2,485 psig
Design temperature	650°F
RCP capacity per pump	88,500 gpm
Speed rating	1,200 rpm
Discharge head	277 ft
Minimum required NPSH	170 ft
Overall height	28.5 ft
Overall weight	188,900 lb
Suction nozzle - inside diameter	31.0"
Discharge nozzle -inside diameter	27.5"

Table 3.2-9
Reactor Coolant Pump Motor Design Parameters

Horsepower rating	6,000
Voltage	12,500Vac
Phase	3
Frequency	60 hz
Power (cold RCS)	5,997 kW
Power (hot RCS)	4,540 kW
Starting current	3,000 amps
AC induction motor, single speed, air cooled. Insulation class B, thermoplastic epoxy	

Table 3.2-10
Thermal and Loading Cycle Design Parameters

	Design Cycles
Heatup at <100°F/hr	200
Cooldown at <100°F/hr	200
Loss of load without an immediate reactor trip	80
Loss of offsite power	40
Loss of RCS flow (one pump only)	80
Reactor trip from 100% power	400
Auxiliary spray >320°F differential	10
Main RCS pipe break	1
Safe shutdown earthquake	1
Leak test at >2335 psig	50
RCS hydrostatic test >3107	5
Steam line break >6" in diameter	1
Steam generator hydrostatic test >1485 psig	5
Turbine rolls on pump heat with >100°F/hr cooldown	10

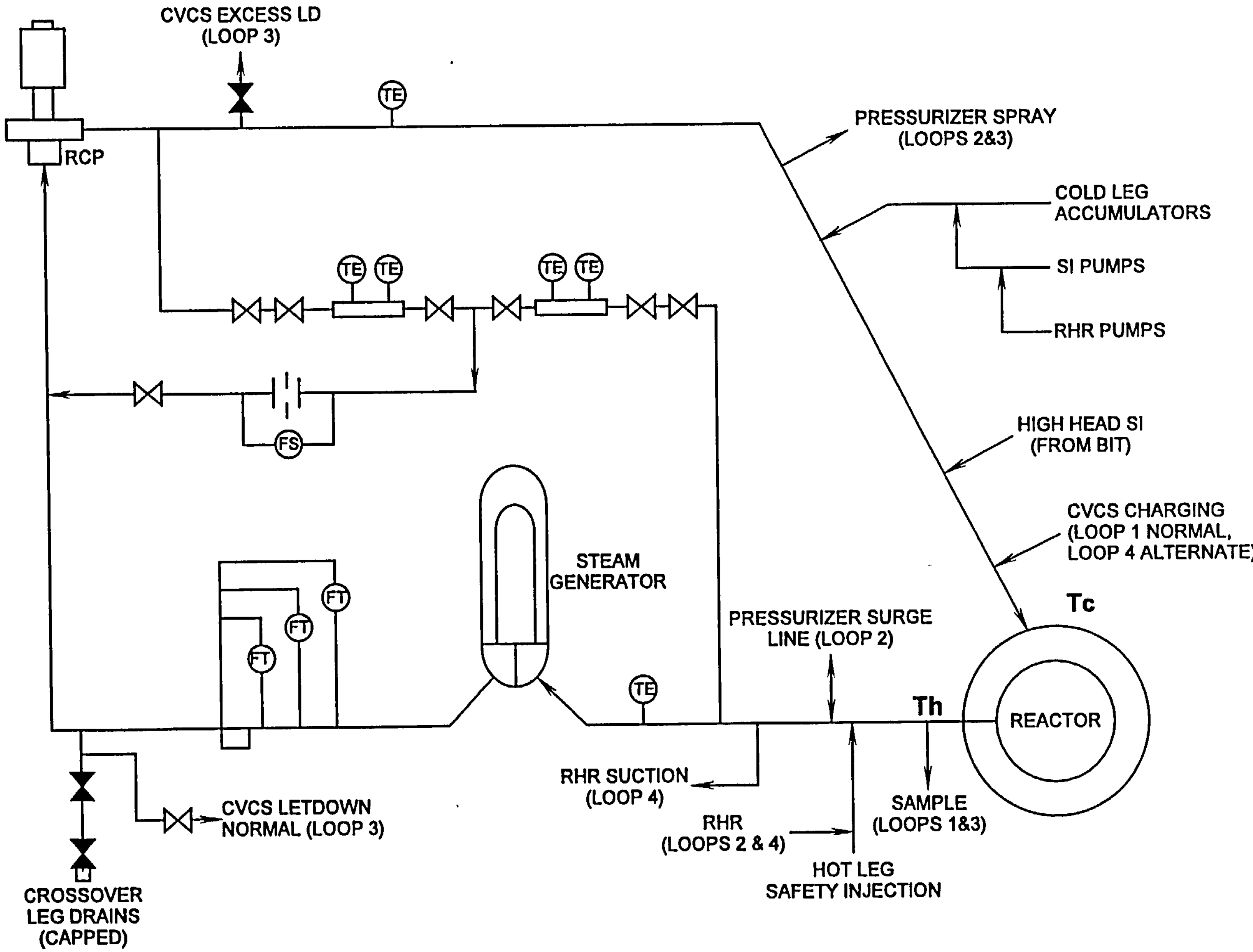


FIGURE 3.2-1 Reactor Coolant Loop Penetrations

0995-32
3.2-7

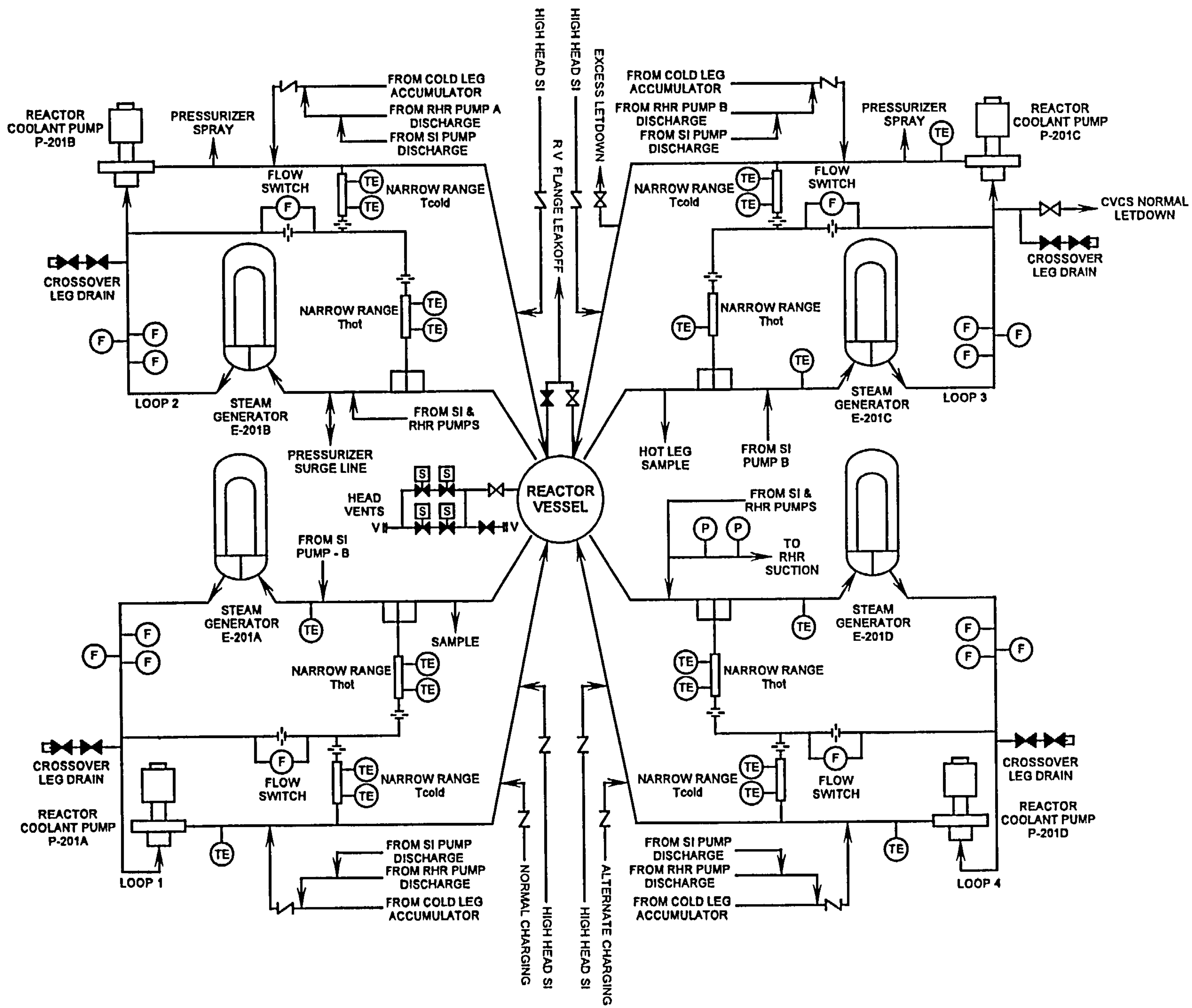


Figure 3.2-2 Reactor Coolant System

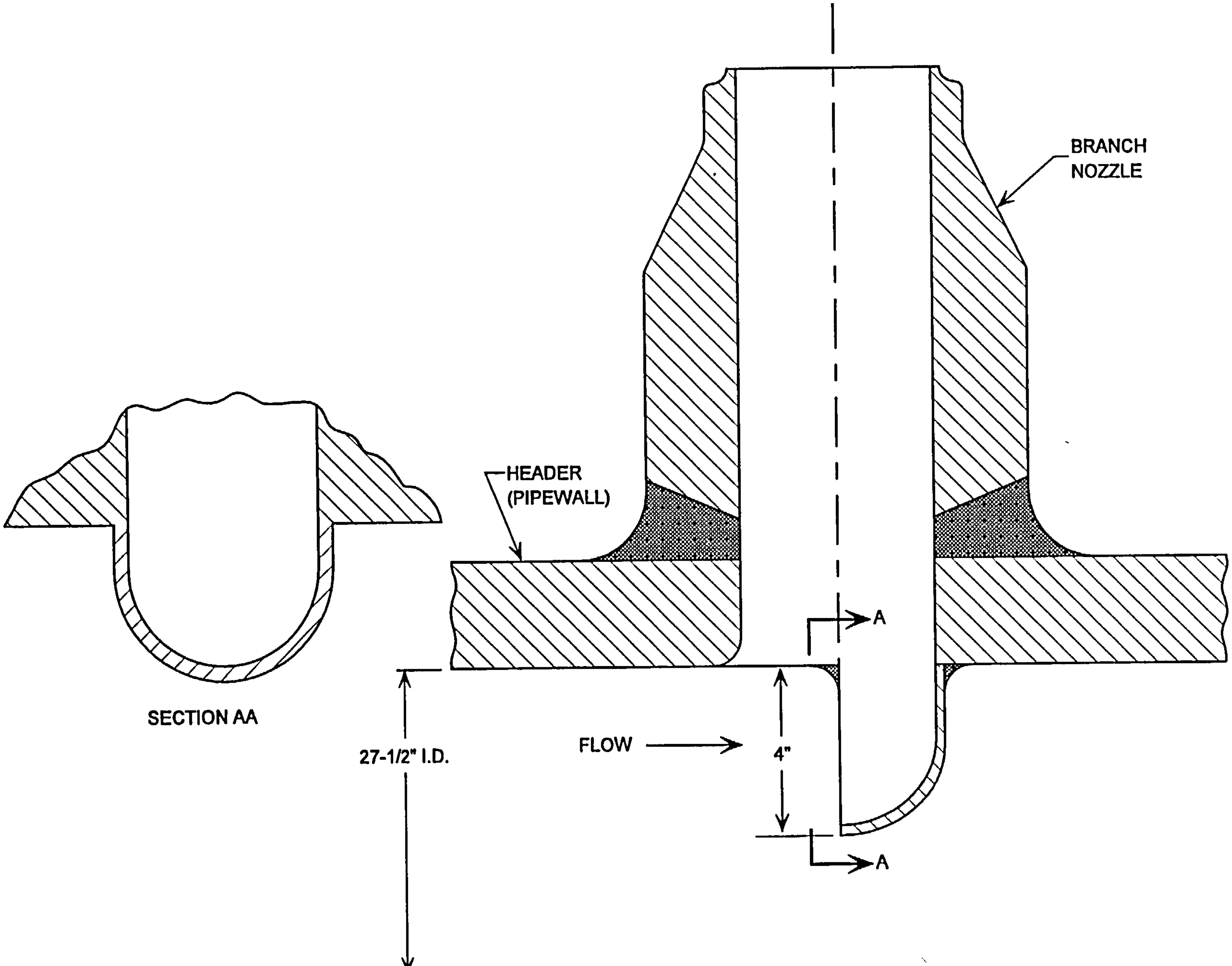


Figure 3.2-3 Pressurizer Spray Scoop

0199-3

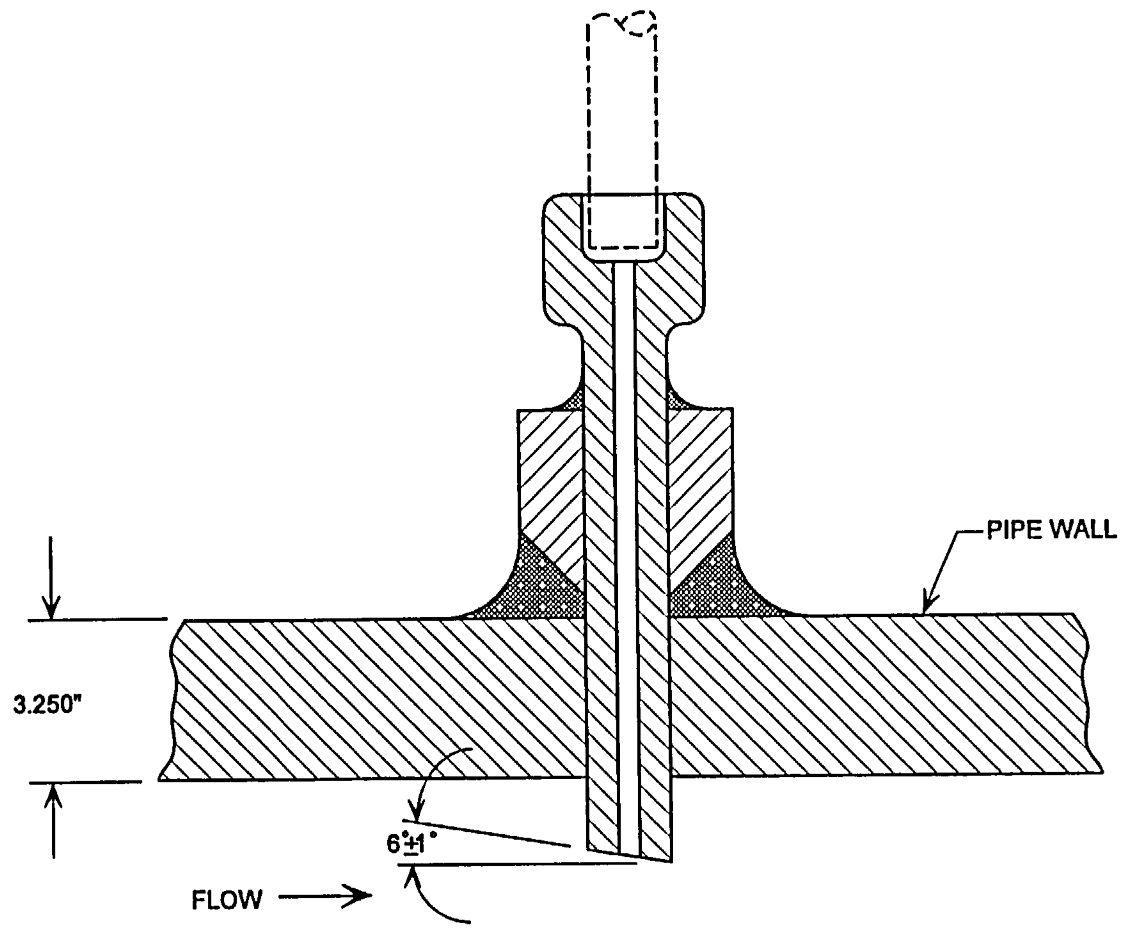


Figure 3.2-4 Sample Connection Scoop

0199-3

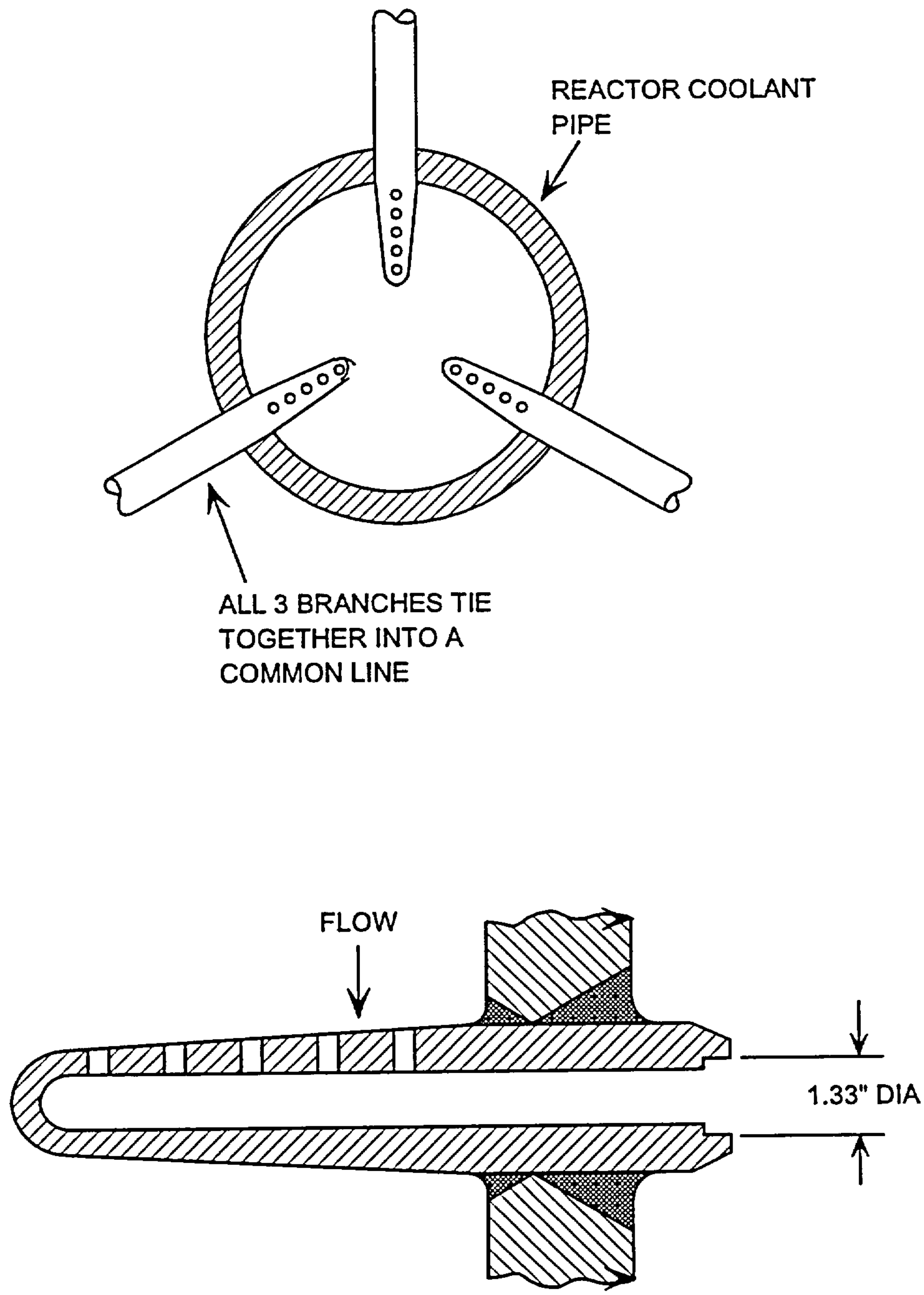
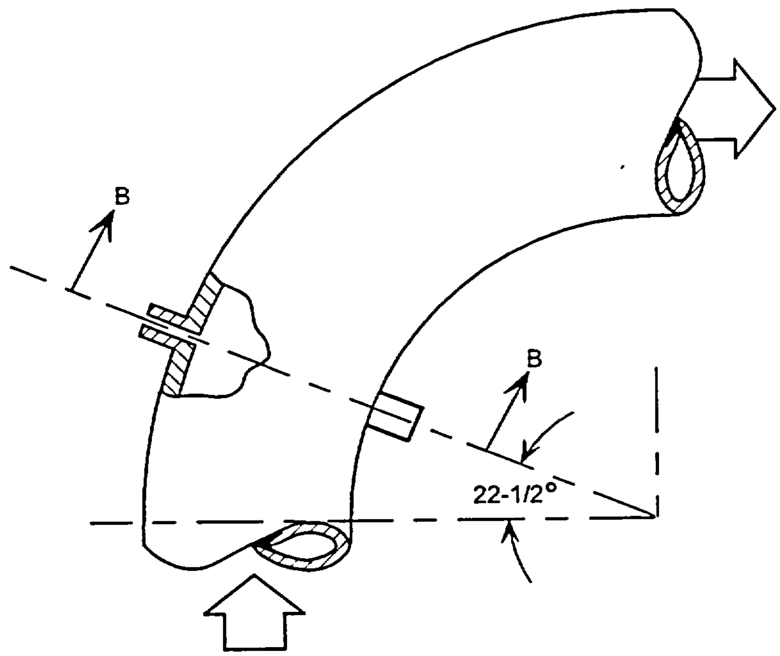
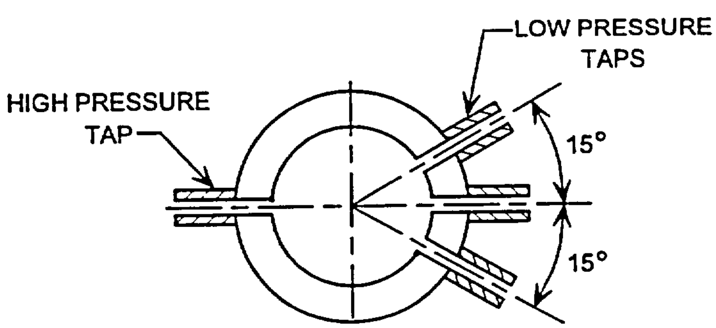


Figure 3.2-5 Hot Leg RTD Tap





SECTION BB

Figure 3.2-6 Reactor Coolant Flow Taps

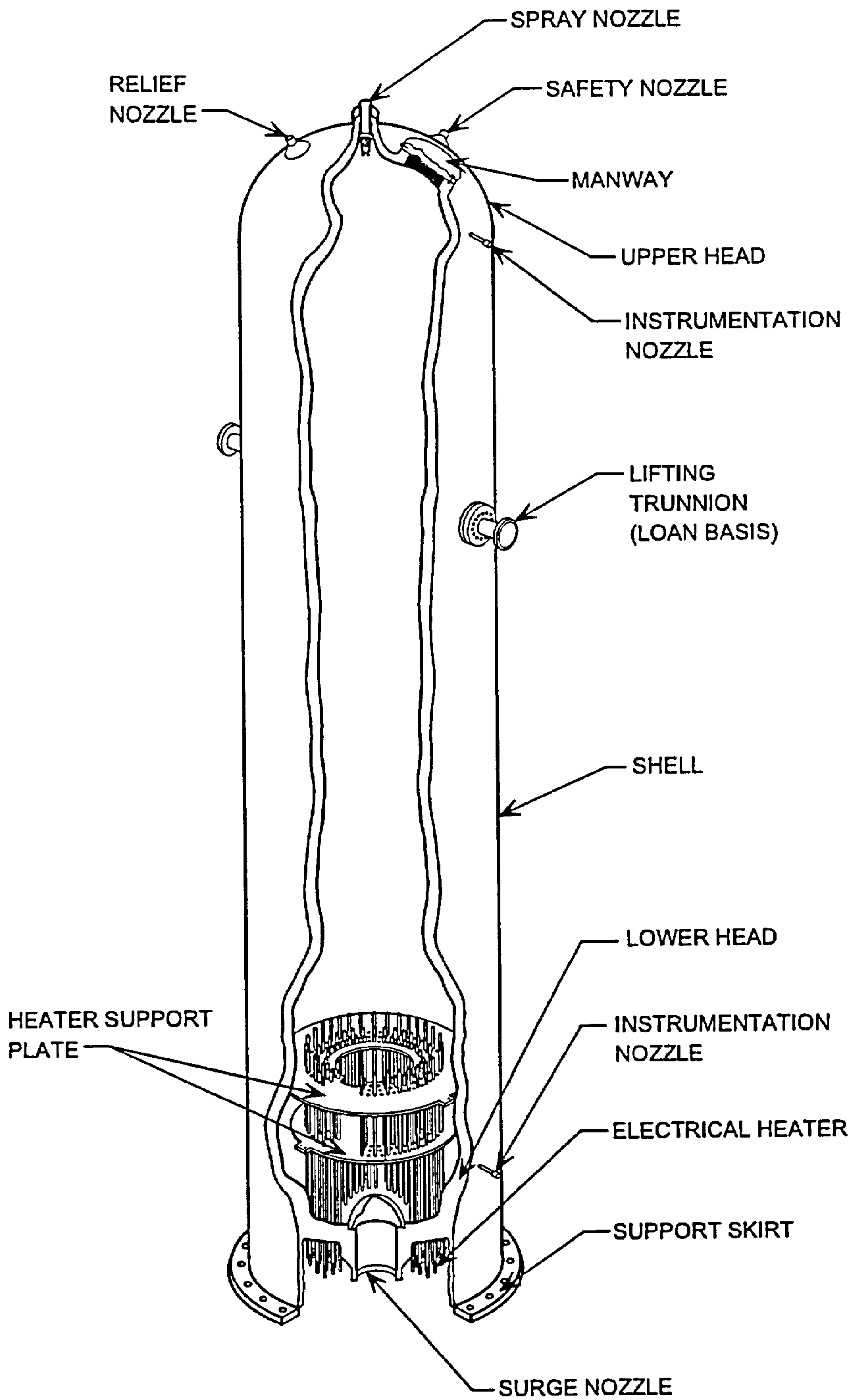
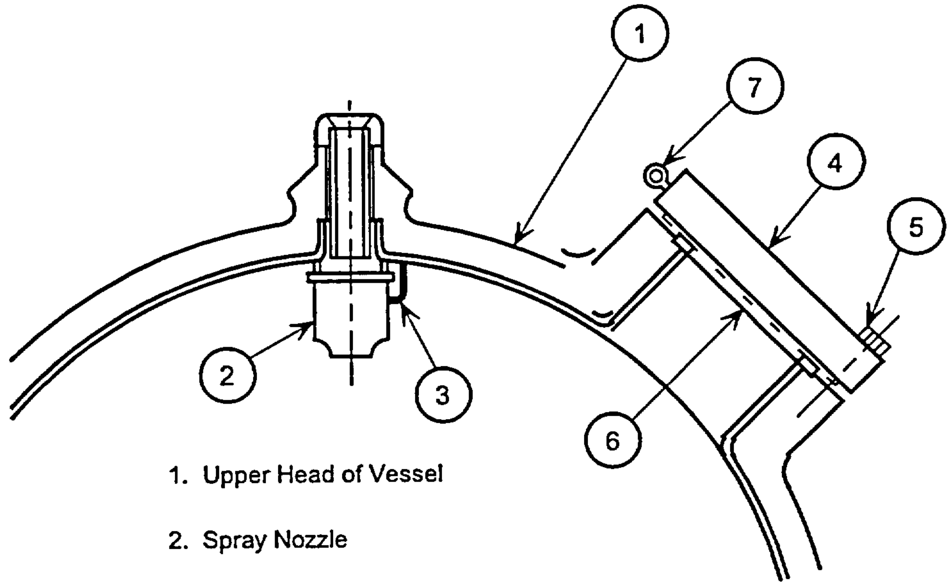
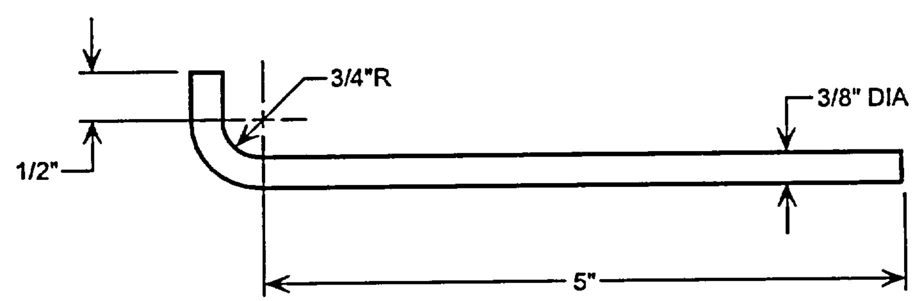


Figure 3.2-7 Pressurizer



1. Upper Head of Vessel
2. Spray Nozzle
3. Spray Nozzle Locking Bar
4. Manway Cover
5. Manway Cover Bolt
6. Manway Cover Insert Assembly
7. Lifting Eye Bolt Location



Matl to Stainless Steel, 18-8 Type 304

Spray Nozzle Locking Bar Detail

Figure 3.2-8 Manway Cover and Spray Nozzle

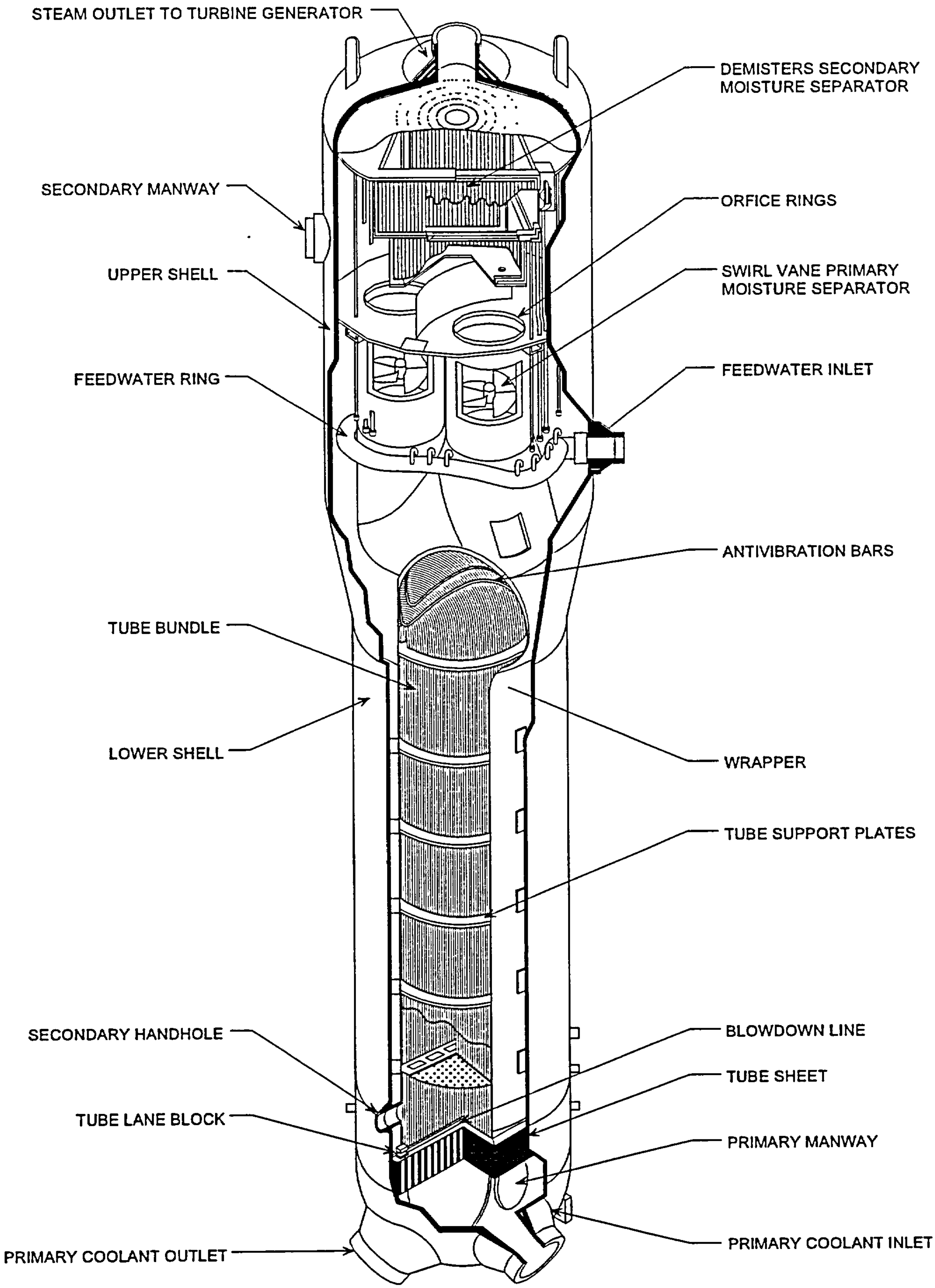


Figure 3.2-9 Model 51 Steam generator

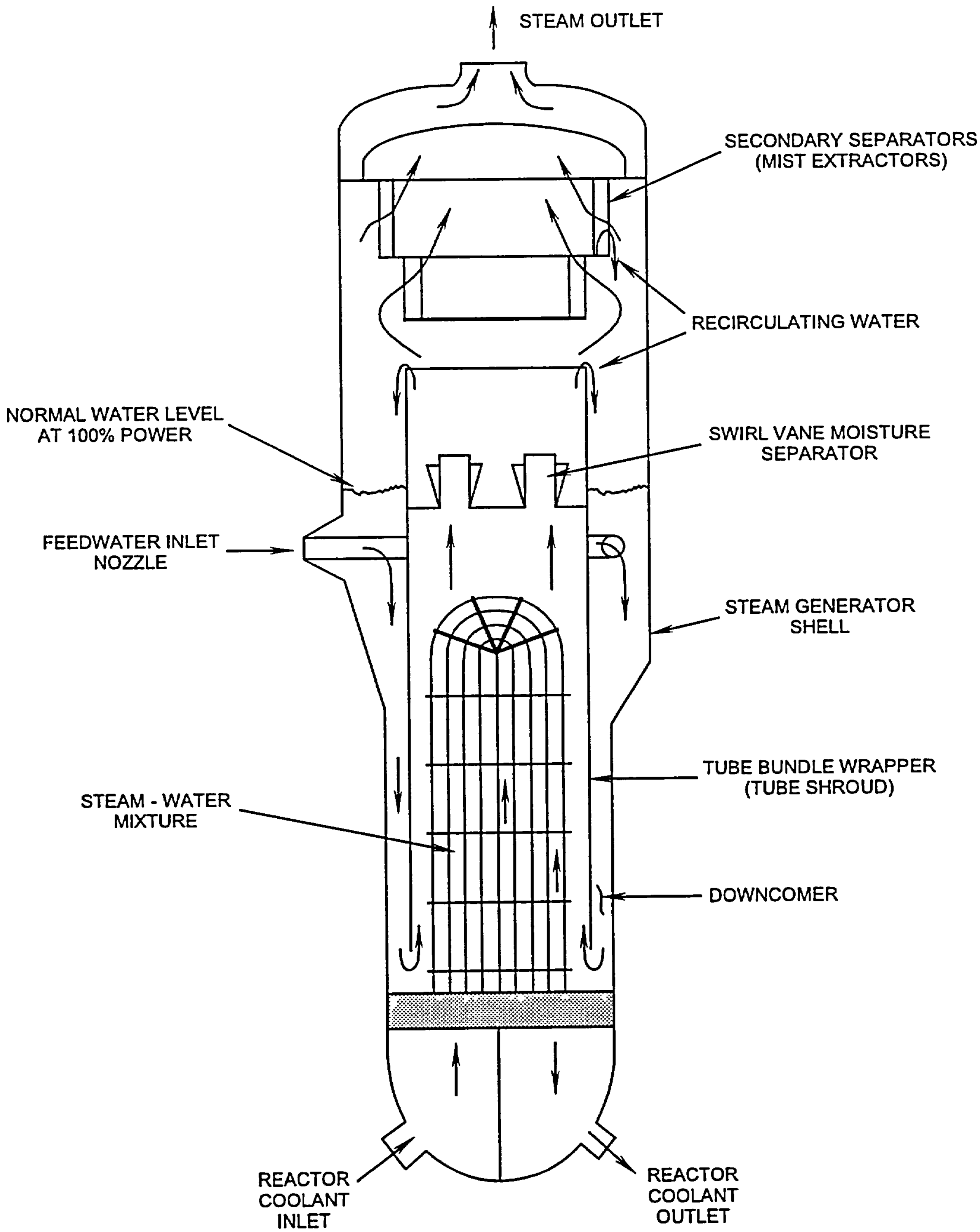
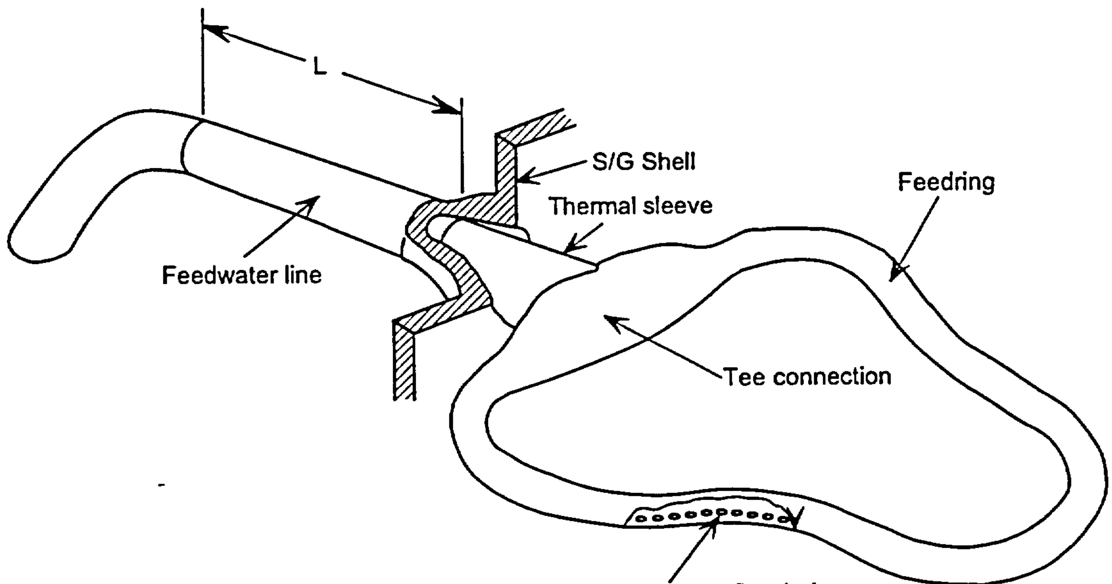


FIGURE 3.2-10 Steam Generator Flow Paths

FEEDRING ASSEMBLY
FEEDRING TYPE STEAM GENERATOR



J-TUBE CONFIGURATION
FEEDRING STEAM GENERATOR

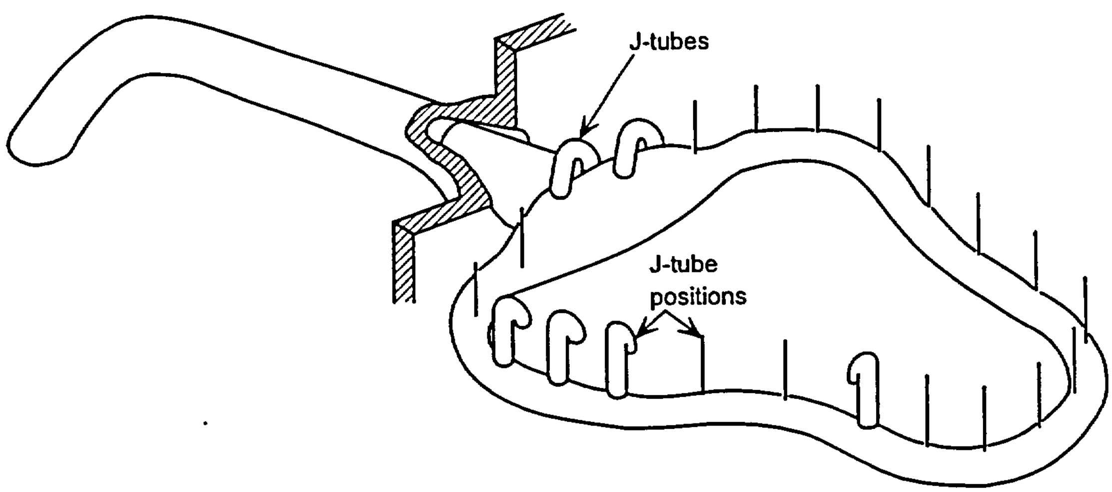


Figure 3.2-11 Feed Ring Assemblies

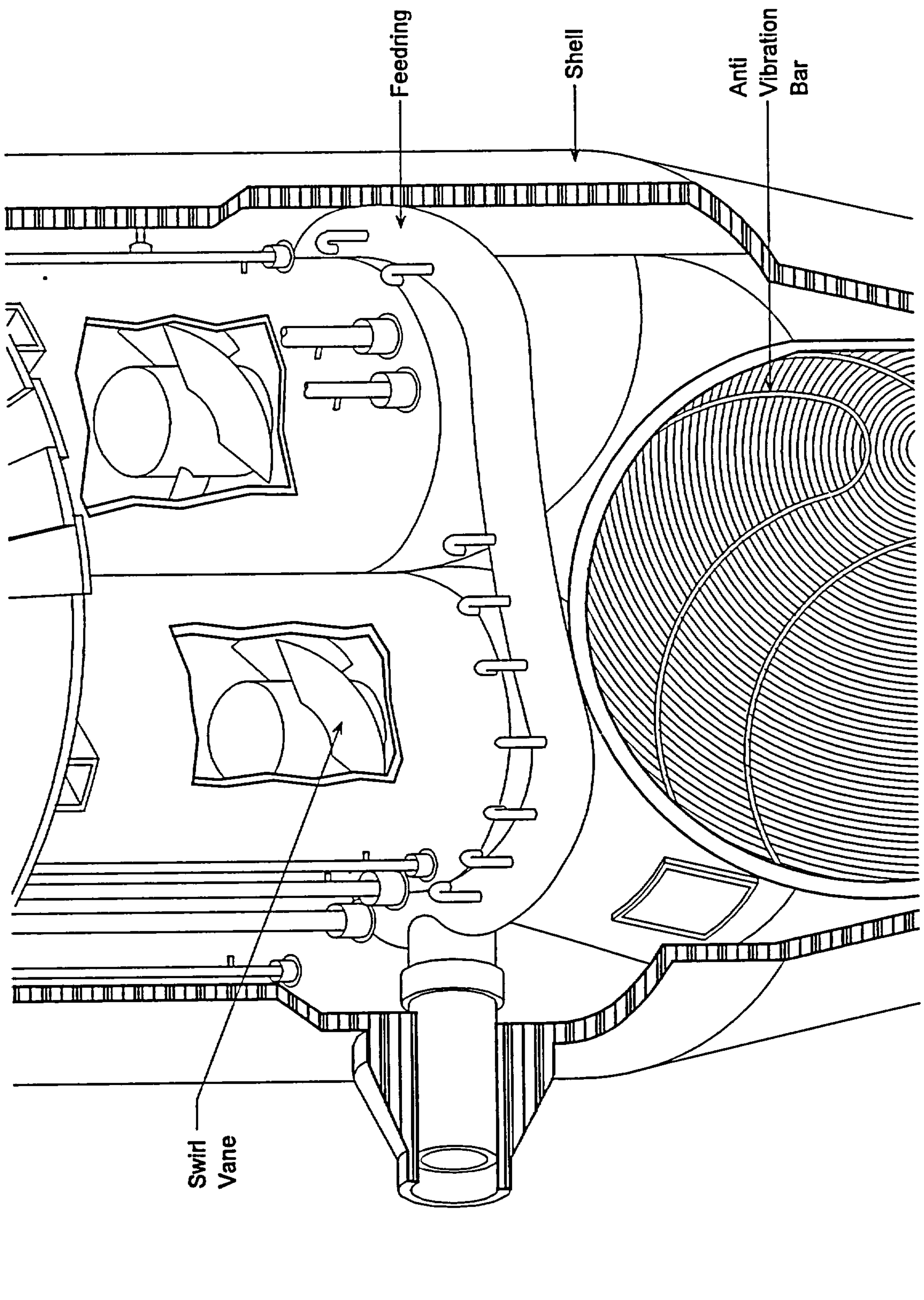


Figure 3.2-12 Feedring and Moisture Seperators

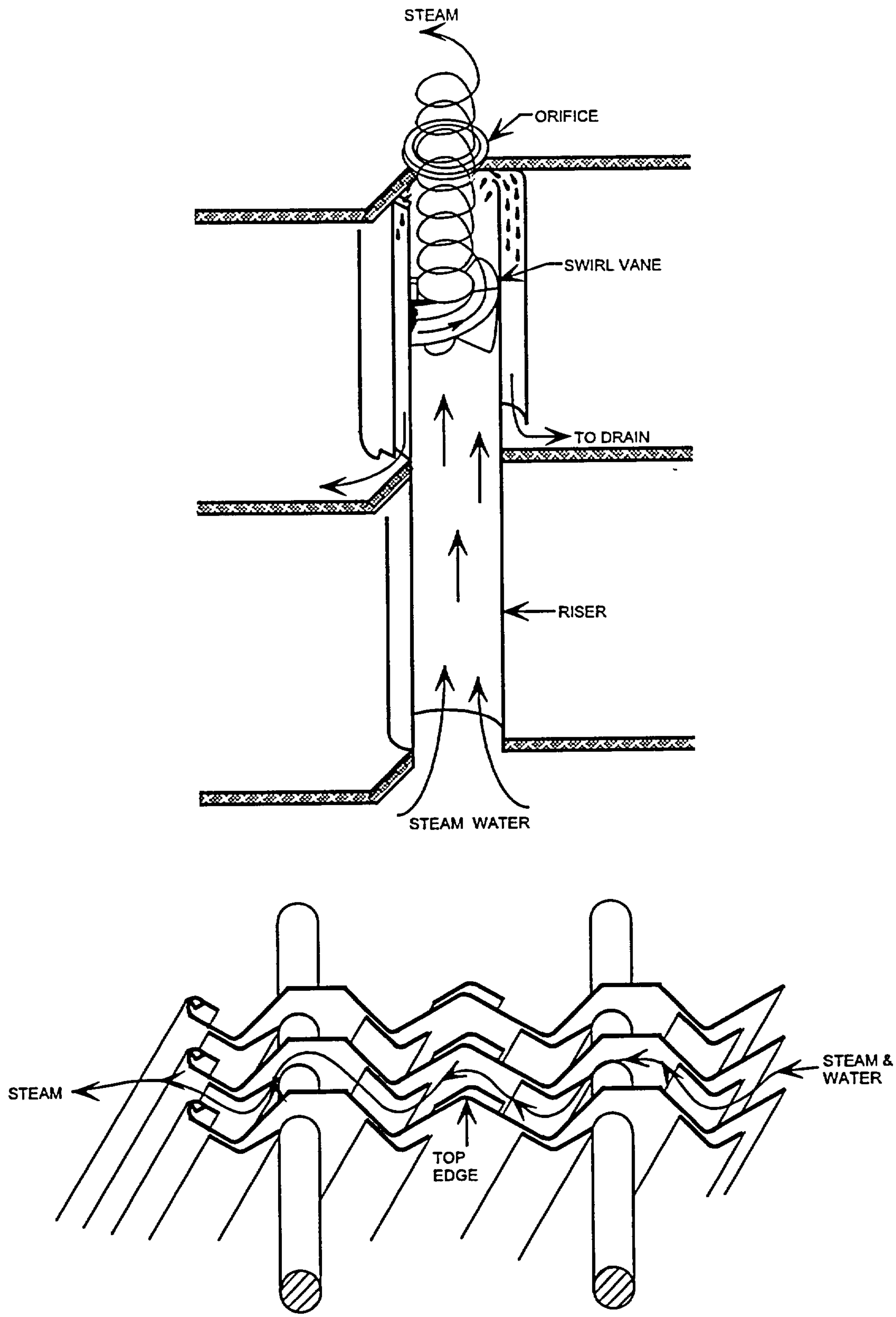
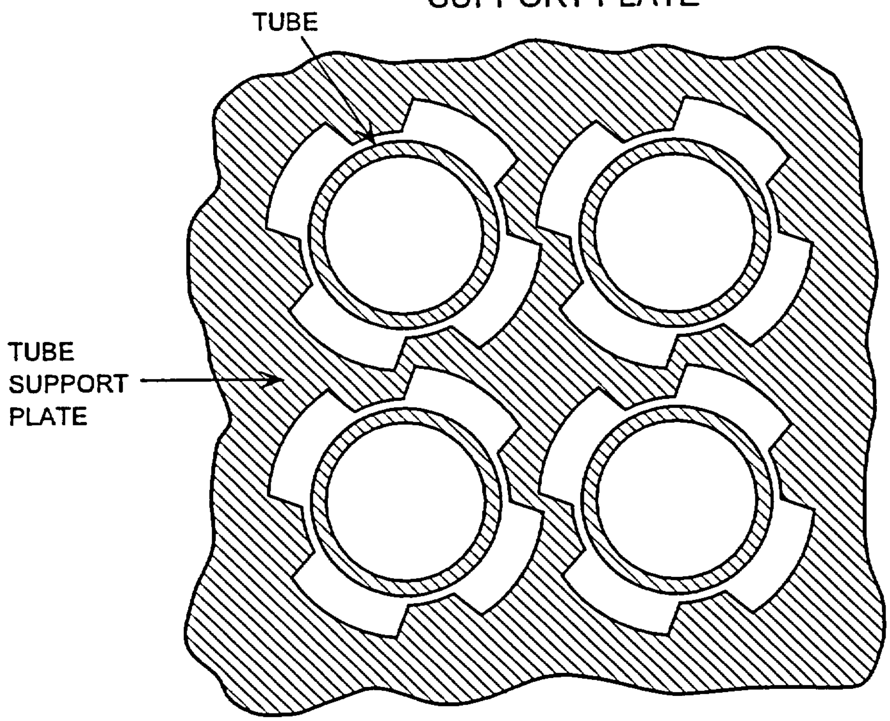


Figure 3.2-13 Steam Generator Moisture Seperators

QUATREFOIL TUBE
SUPPORT PLATE



DRILLED TUBE
SUPPORT PLATE

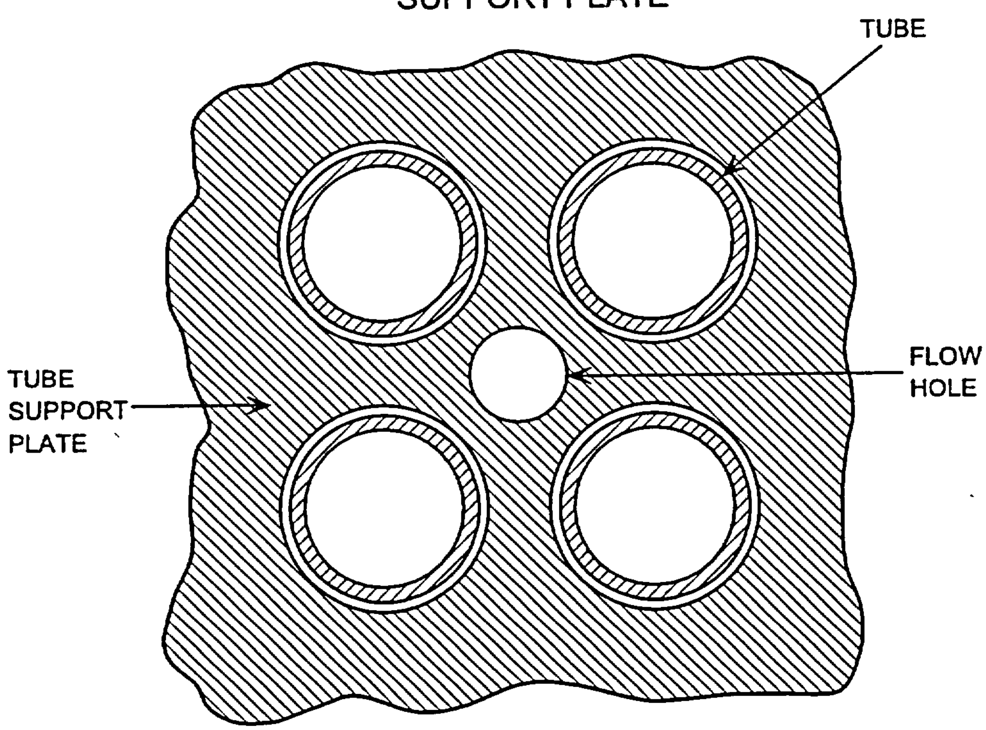


Figure 3.2-14 Tube Support Plate

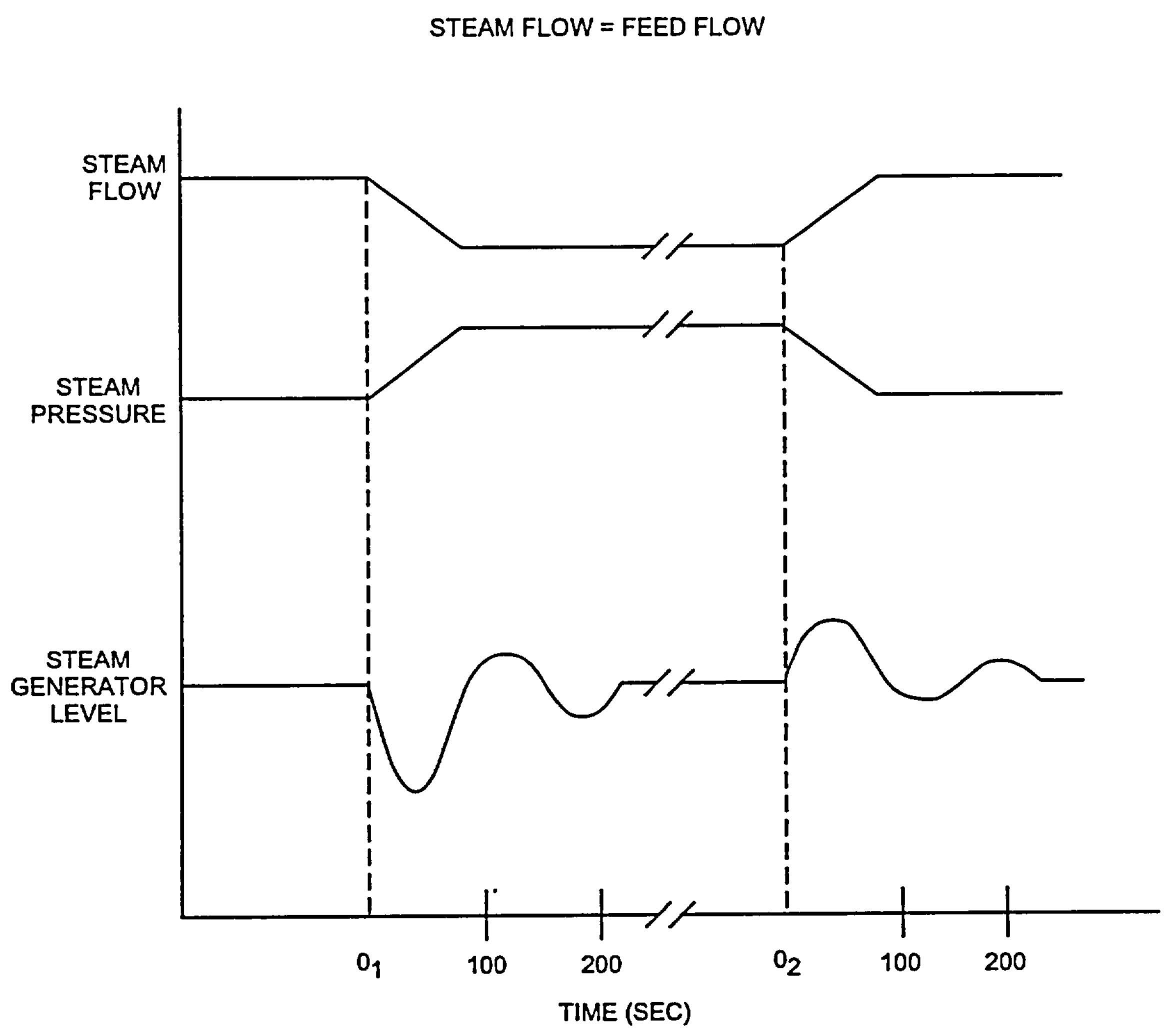


FIGURE 3.2-15 Steam Generator Shrink and Swell

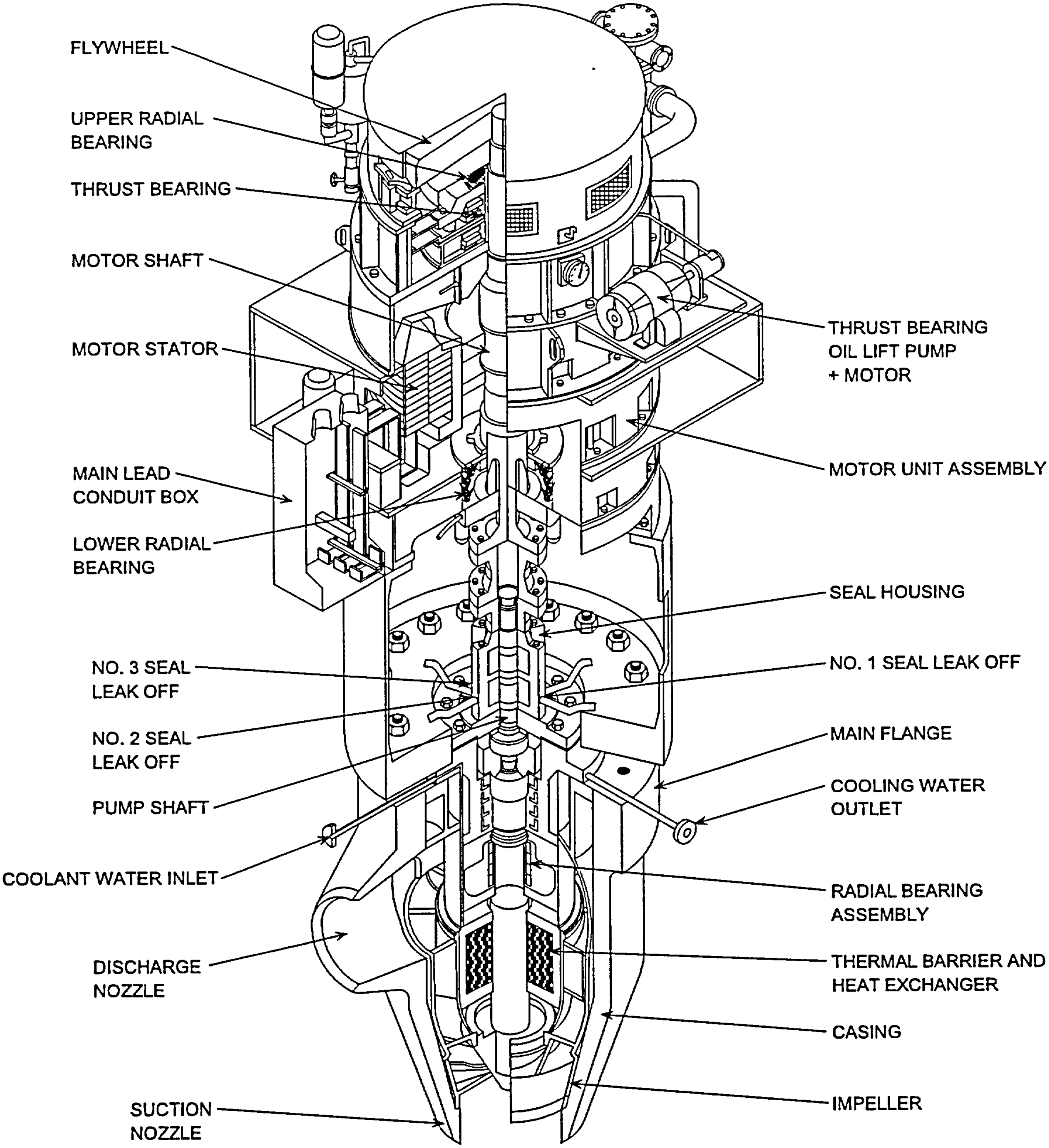


Figure 3.2-16 Reactor Coolant Pump

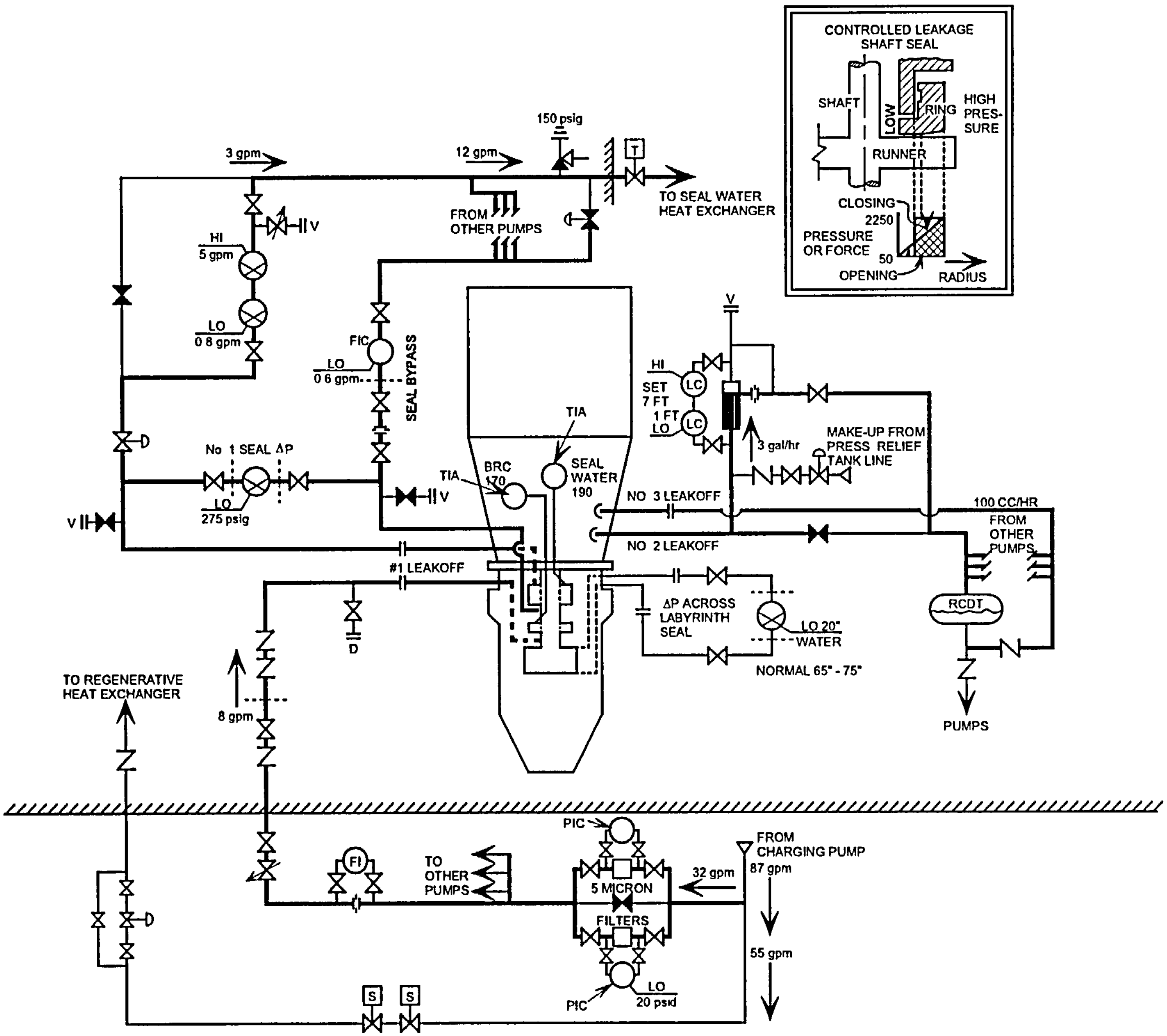


Figure 3.2-17 Seal Water Injection and Leakoff

0199-3

PRELIMINARY OPERATING PARAMETERS		
SEAL	INLET PRESSURE	FLOW RATE
NO 1	2250	3 GPM
NO 2	50	3 GPH
NO 3	6	100 CC/HR

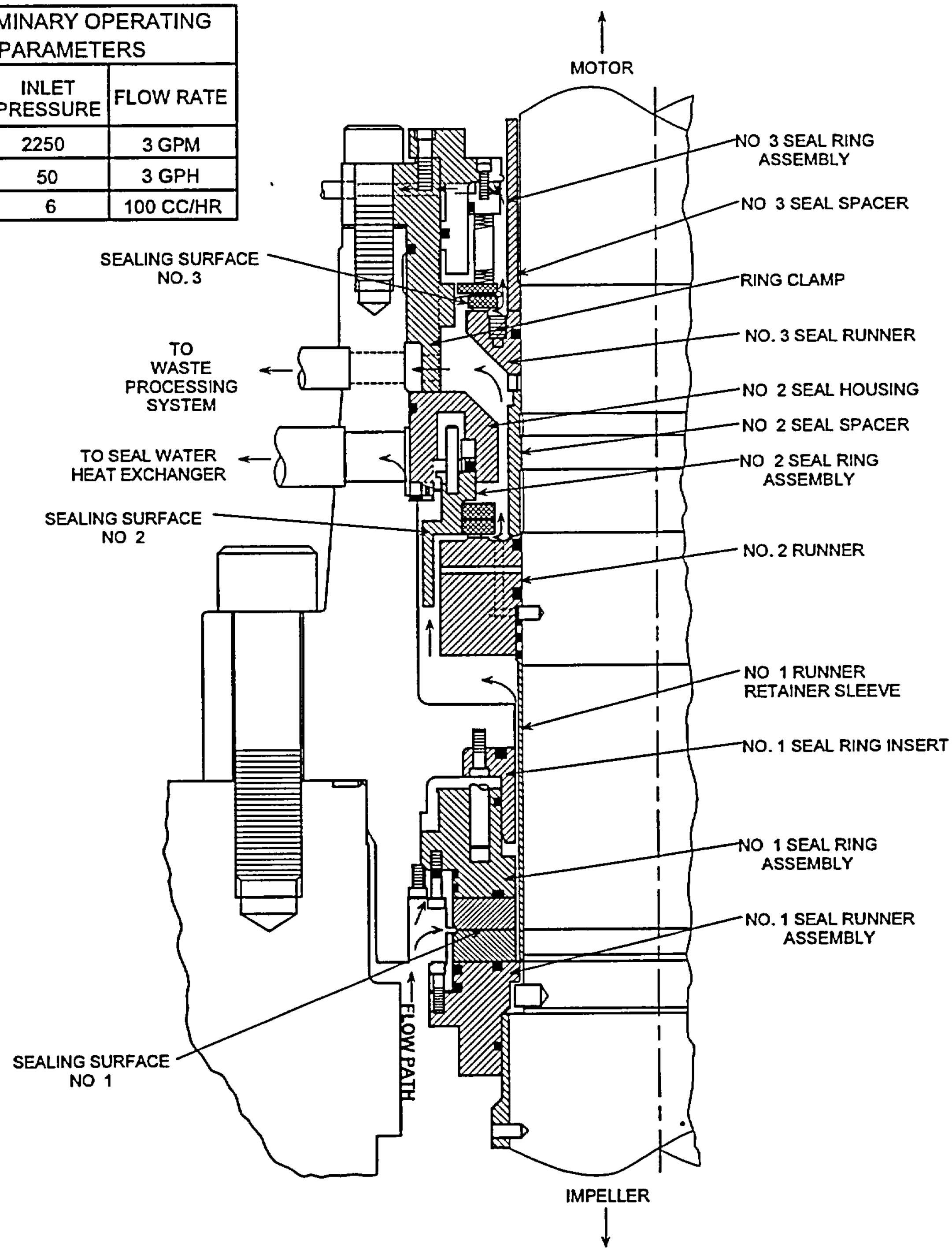


Figure 3.2-18 Shaft Seal Arrangement

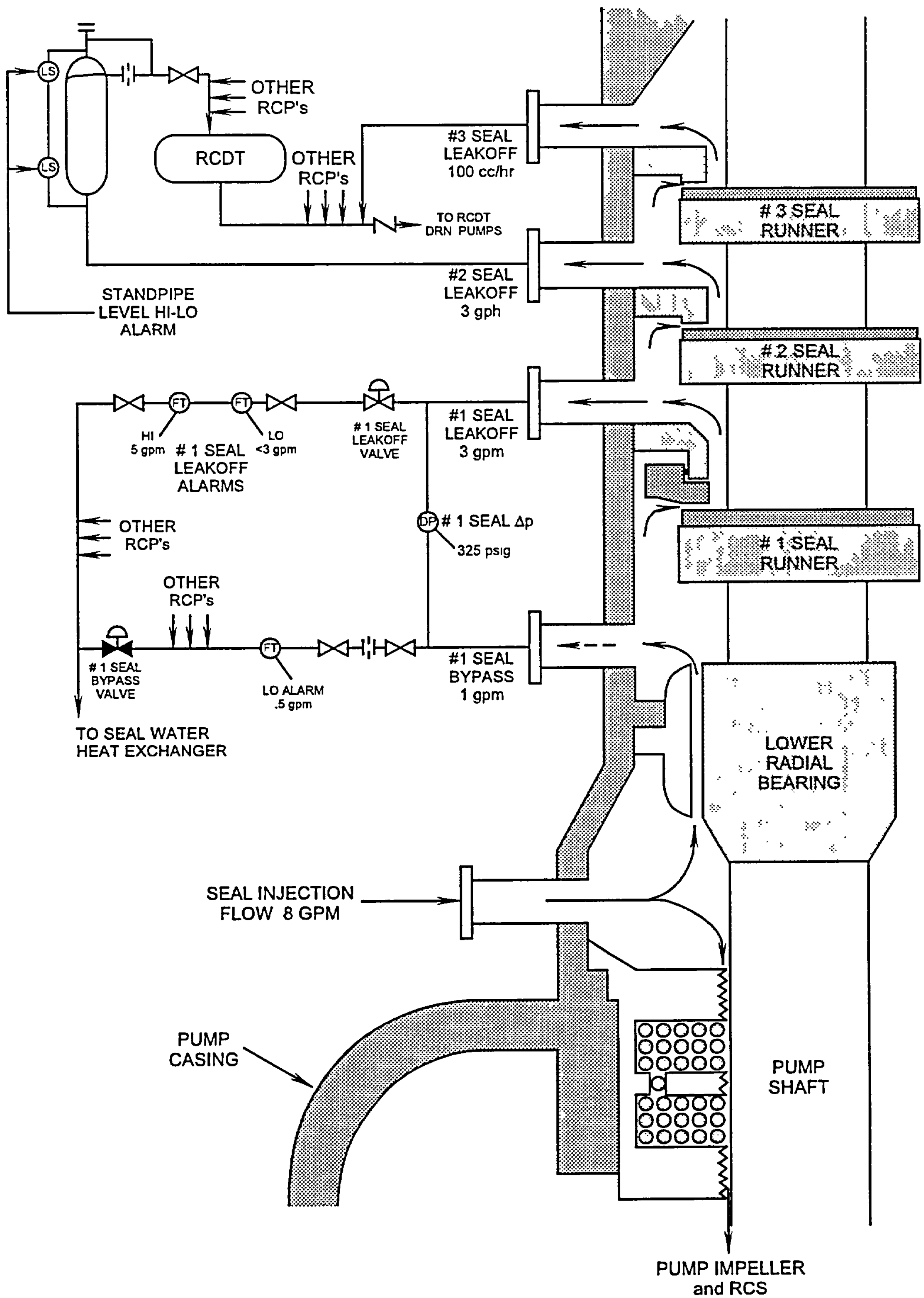


FIGURE 3.2-19 Seal Flow Diagram

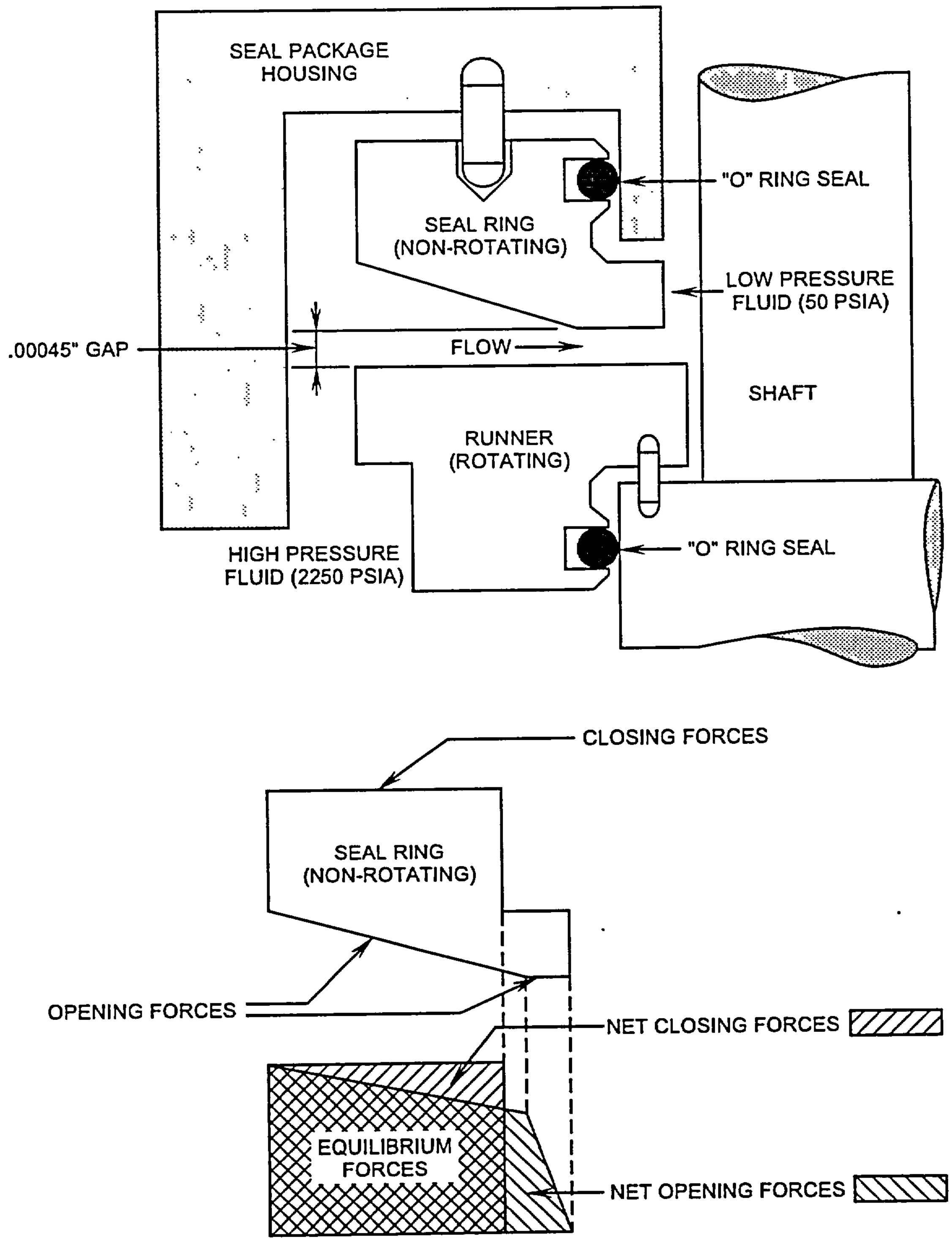


Figure 3.2-20 Controlled Leakage Shaft Seal

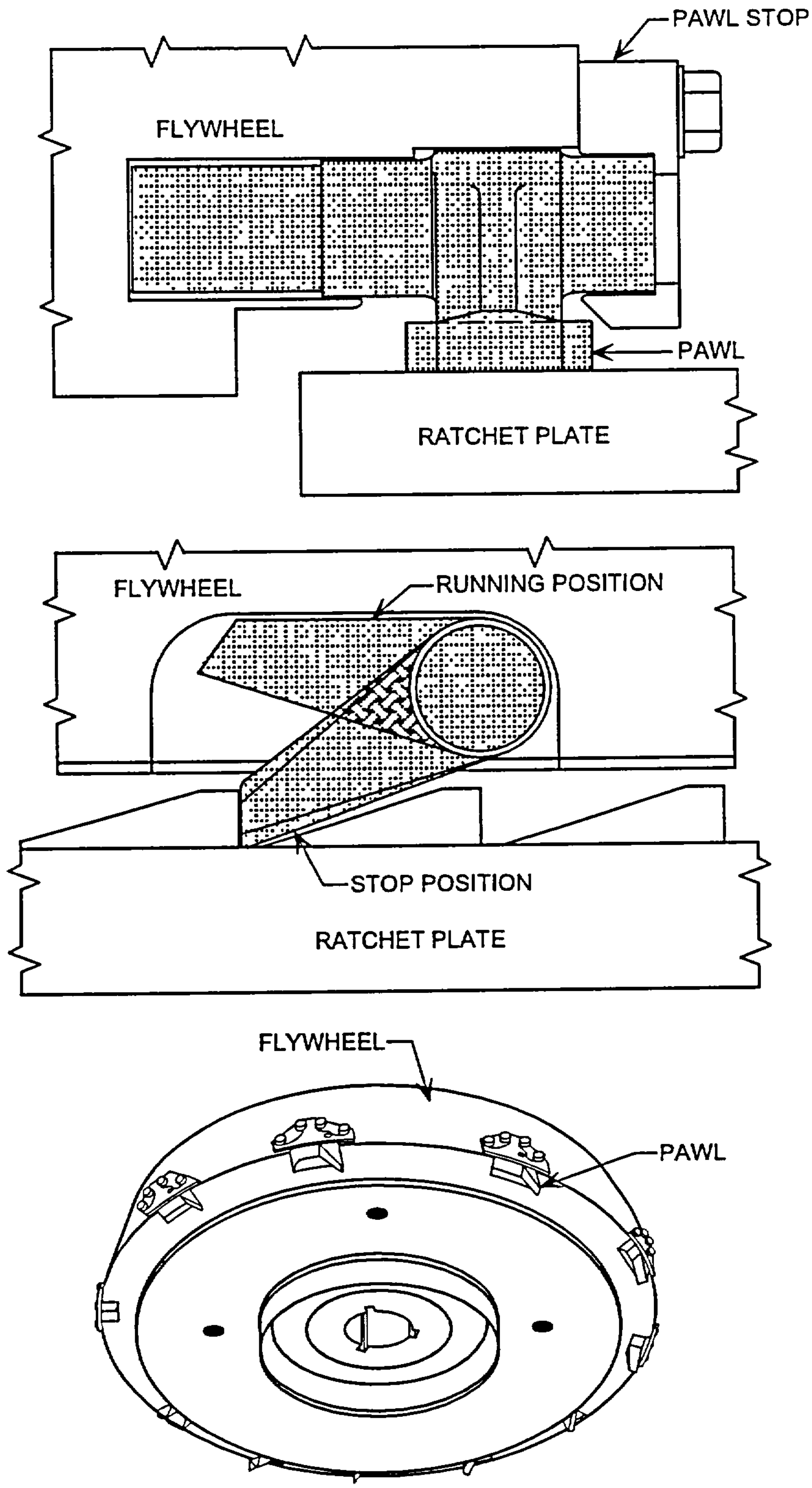
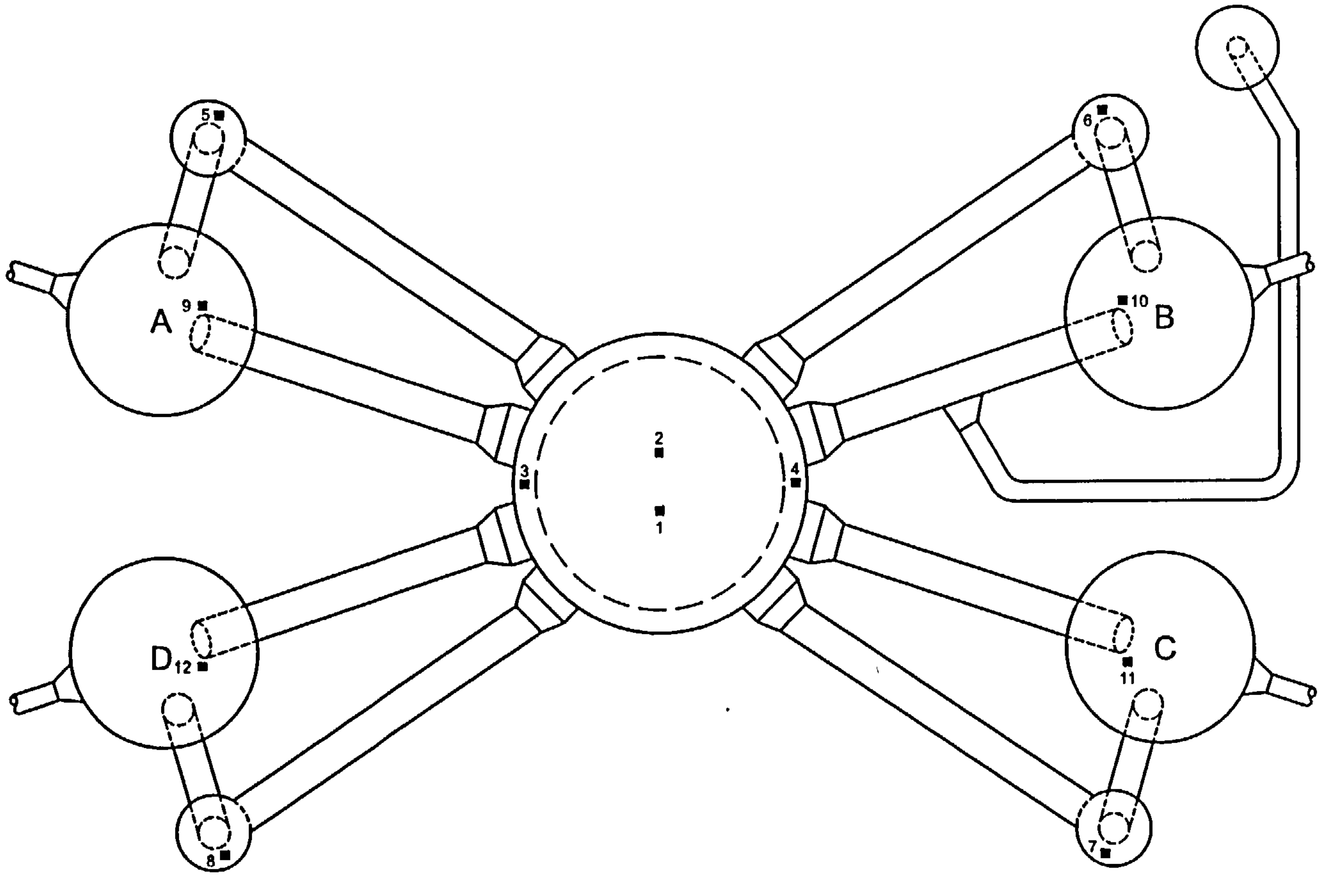


Figure 3.2-21 RCP Flywheel and Anti-reverse Rotation Device



CHANNEL	LOCATION	CHANNEL	LOCATION
1	LOWER VESSEL (WEST)	7	RC PUMP C
2	LOWER VESSEL (EAST)	8	RC PUMP D
3	UPPER VESSEL (NORTH)	9	STEAM GENERATOR A
4	UPPER VESSEL (SOUTH)	10	STEAM GENERATOR B
5	RC PUMP A	11	STEAM GENERATOR C
6	RC PUMP B	12	STEAM GENERATOR D

Figure 3.2-22 Vibration and Loose Part Monitoring Transducer Locations

0199-3

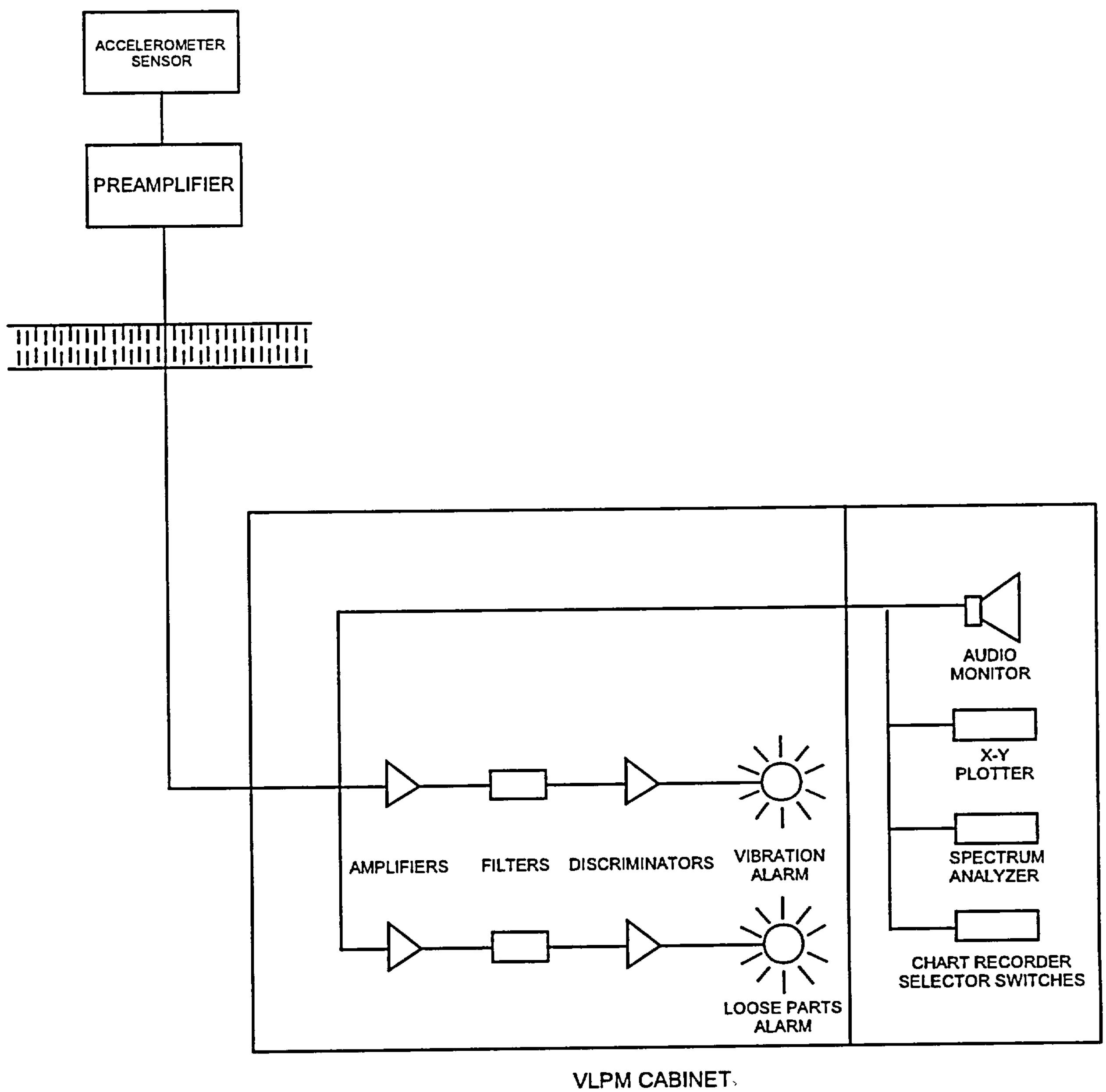


Figure 3.2-23 Vibration and Loose Parts Monitoring System

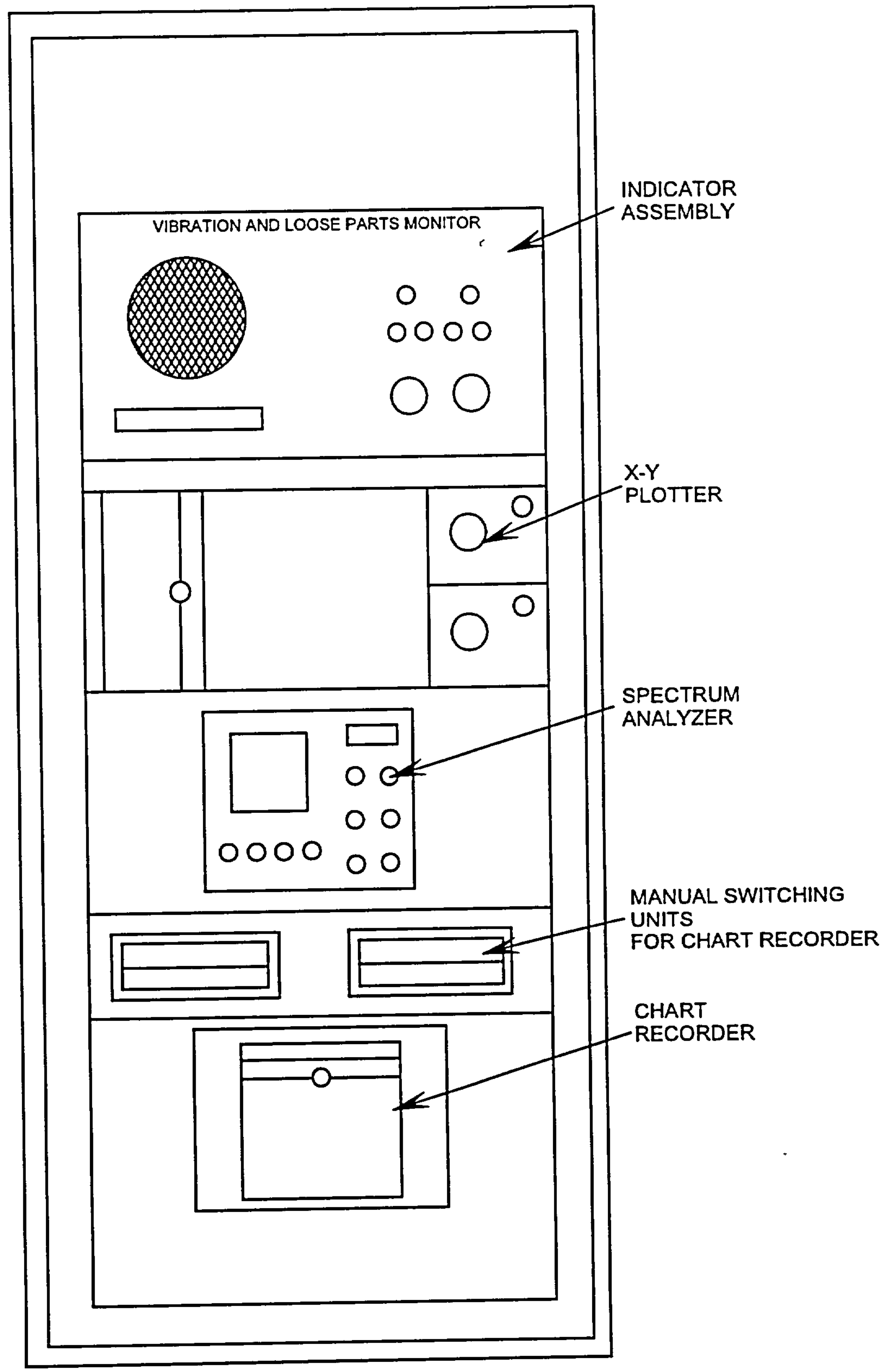


Figure 3.2-24 Vibration and Loose Parts Monitoring Cabinet Arrangement

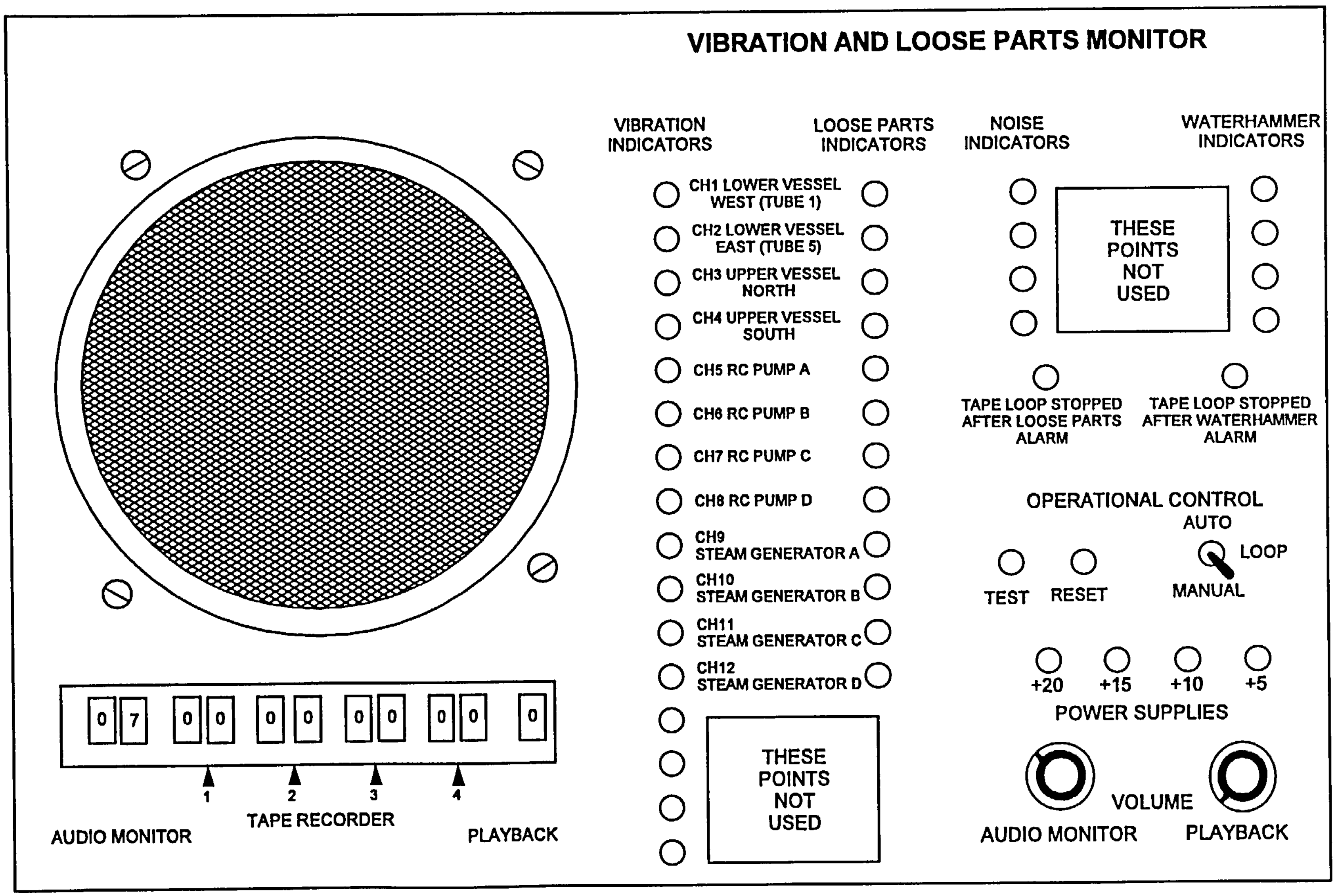


Figure 3.2-25 Vibration and Loose Parts Monitoring Indicator Assembly

0199-3

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL· REACTOR VESSELL LOWER SHELL

COPPER CONTENT. 0 16 WT%

PHOSPHORUS CONTENT. 0.012 WT%

INITIAL RT_{NDT}: 10°F

RT_{NDT} AFTER 10 EFPY 1/4T 111°F

3/4 T 55°F

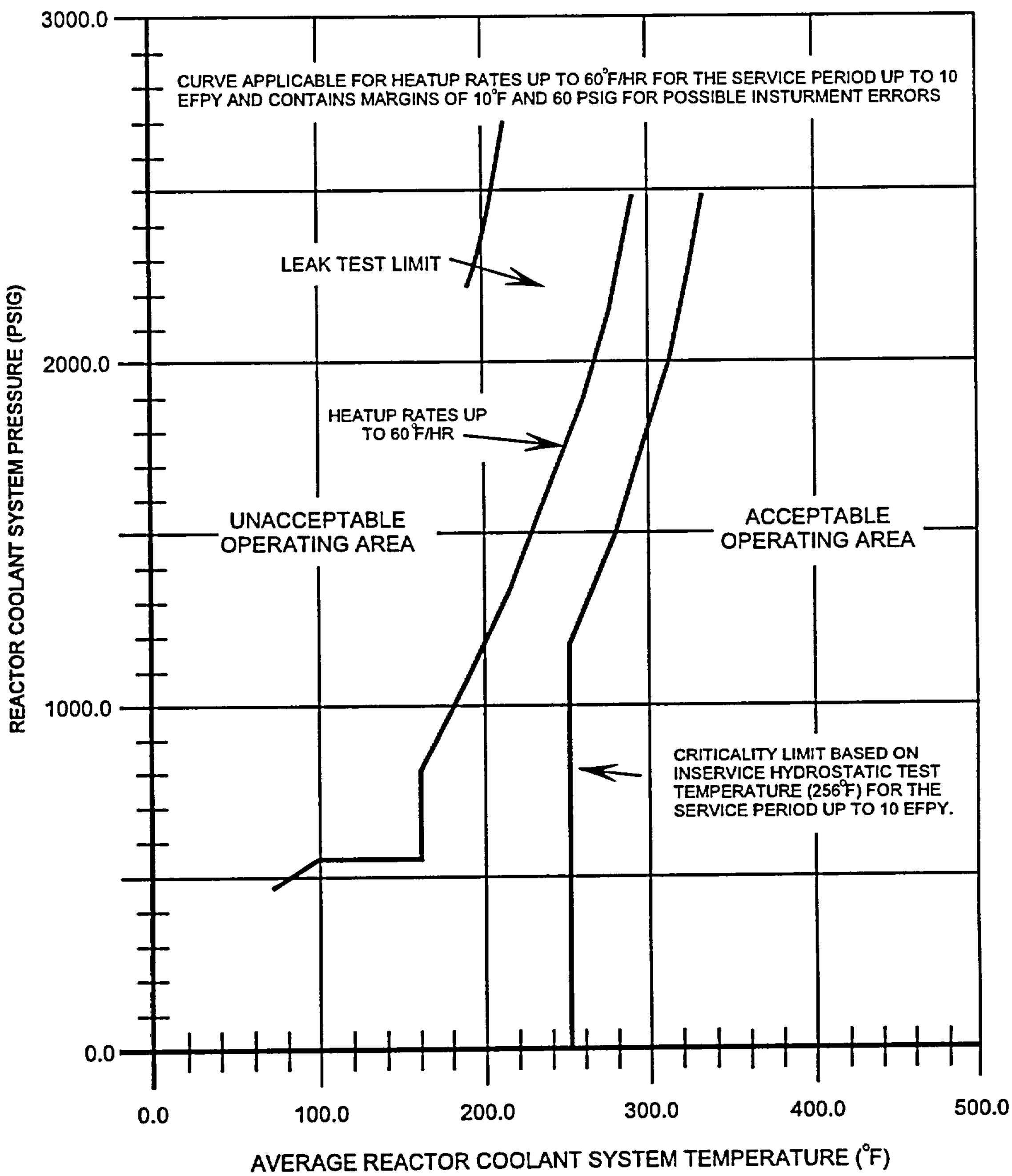


Figure 3.2-26 Reactor Coolant System Pressure - Temperature Limits (Heatup)

Material Property Basis:

Controlling Material: Reactor Vessel Lower Shell
Copper Content: 0.16 WT%
Phosphorus Content: 0.012 WT%

Initial RT_{NDT}: 10°F
RT_{NDT} after 10 EFPY 1/4 T 111°F
3/4 T 55°F

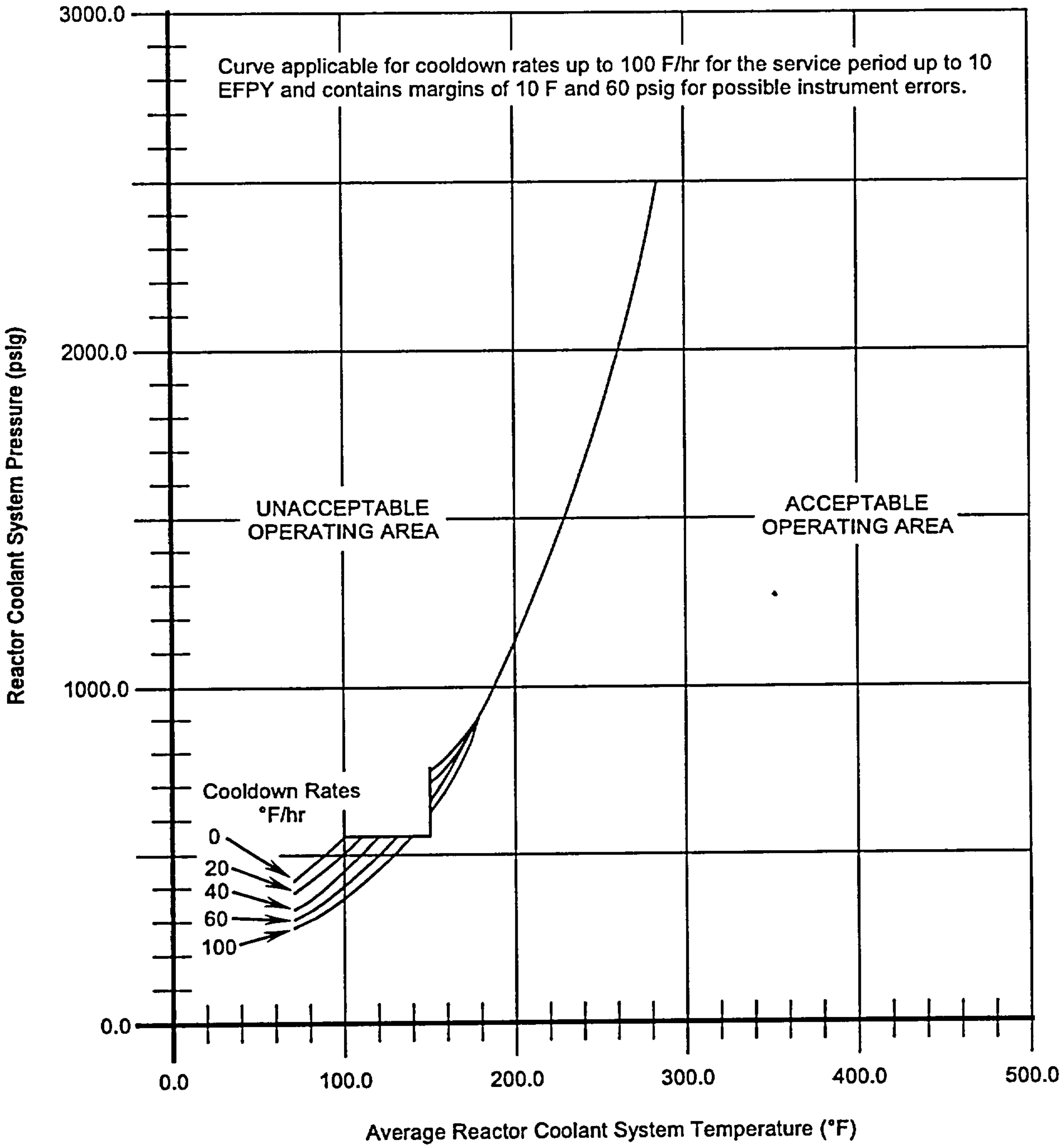


Figure 3.2-27 Reactor Coolant System Pressure - Temperature Limits (Cooldown)