Light-Water Reactor Startup/Shutdown

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Light-Water Reactor Plant Startup

— Light-water reactor plant startup refers to the process whereby all operating systems (core, primary loop, pressurizer, steam generators, turbine, condenser, etc.) are taken from a cold shutdown to a hot operating status. The attainment of criticality is a major part of this process. However, coordination of pressure and temperature for plant heatup and the initial production of steam are also of importance to plant safety.

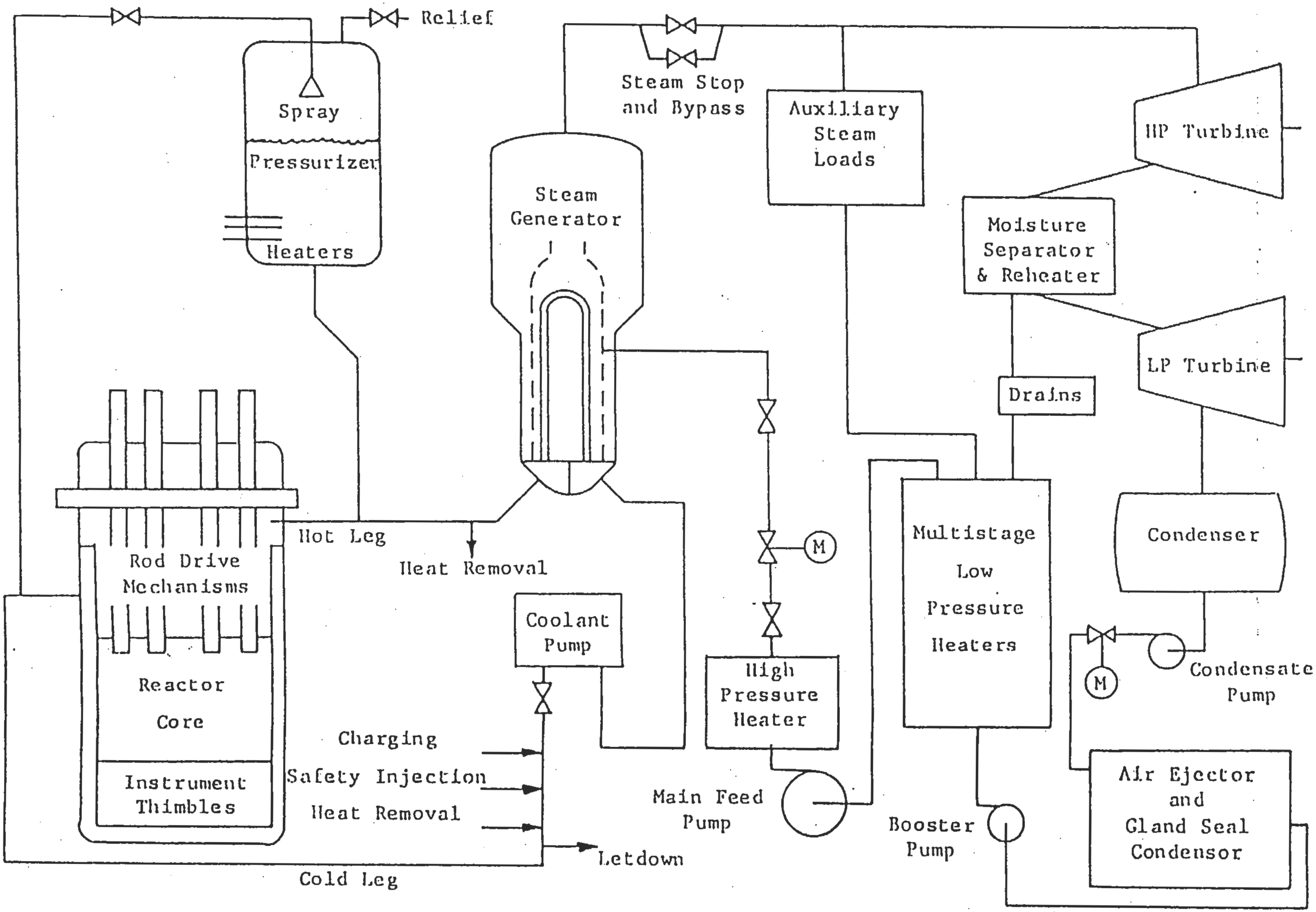
— Each plant has a slightly different startup procedure because of its unique design features. The material presented here is for a 1970 vintage Westinghouse PWR. BWR information then follows. The precise details of the startup are not important. The lesson to be learned here is the need for <u>integrated</u> control of pressure and temperature while also observing heatup limits and constraints on the attainment of criticality.

Pressurized Water Reactor

- The topics covered relative to PWRs are:

  - Systems/components for temperature, pressure, and reactivity adjustment.
  - Nuclear instrumentation.
  - Limiting conditions (pressure/temperature) for startup.
  - Startup sequence.
  - Operational and Safety Issues During Startup

Schematic Diagram of Pressurized Water Reactor



Temperature Control

— Under normal operating conditions, temperature is controlled by balancing the heat produced from the core with the energy removed by the turbine.

— During startup, temperature can be controlled by:

1. Recirculation of the pressurizer (heaters on, spray valve open).

2. Operation of the primary coolant pumps (friction losses result in a slow heatup).

3. Critical operation of the reactor core at low power.

Pressure Control

— Under normal operating conditions, pressure is produced by the pressurizer which is situated above the primary loop and core tank. Boiling occurs in the pressurizer and the resulting pressure is transmitted to the primary system which is subcooled. The pressurizer is at 680 °F; the primary at 650 °F. (Note: A common misconception is that the primary pump provides pressure. The pump's role is to provide flow energy.)

— During startup, pressure can be controlled by:

1. Coordinated manual operation of the charging and letdown systems.

2. Pressurizer heaters.

3. Pressurizer drain rate (to draw bubble).

4. Pressurizer spray (once bubble drawn).

PWR Reactivity Control

Four mechanisms are available for reactivity control. These are:

1) Movable Control Rods

   a) Full-Length Shutdown Rods: These are normally fully withdrawn at startup and kept withdrawn during operation. Their function is to protect the core against a sudden positive reactivity insertion.

   b) Full-Length Control Rods: These are to create subcritical multiplication and achieve criticality. They are also used to compensate for the reactivity associated with temperature changes when at power, to maneuver the reactor at up to 5% per minute, and to compensate for reactivity changes associated with changes of reactor power.

2. Part-Length Rods: These are used for power shaping. They are not inserted on a trip.

3. Soluble Poison (Boric Acid): This is used to control slow, long-term changes of reactivity including fuel depletion, long-lived fission product buildup, reactivity effects associated with plant heatup, xenon, etc.

PWR Reactivity Control (Cont.)

4. Negative Moderator Temperature Coefficient: This is a design feature that promotes passive safety under normal conditions. The core is under-moderated so not all neutrons are fully thermalized. If power increases, coolant temperature rises and hence its density decreases. Fewer neutrons are thermalized, negative reactivity is produced, and the power drops. The negative moderator temperature coefficient makes reactors self-regulating.

Note: PWR fuel incorporates burnable poisons. These are a design feature that reduces the need for control rods. However, burnable poisons can not be used to control the reactor.

PWR Nuclear Instrumentation

— For PWRs, all detectors are gas-filled and are located outside the core itself. There are three sets of detectors, chosen to have overlapping ranges:

a) Two source-range detectors are located on opposite sides of the reactor at the core mid-plane. These instruments are usually fission chambers that are operated in the pulse mode. The detector senses both gamma rays and neutrons. However, given that the neutron pulse height is larger than that of a gamma ray, the gamma ray component can be removed from the detector's signal by the application of a discriminating voltage. This is important because the gamma ray flux can exceed the neutron flux by a factor of one hundred in a shutdown reactor. The source-range instruments span approximately six decades, are used to take the reactor critical, and are shut off once the reactor is operating at high power. The latter action is taken to prevent damage to the electronics. The source-range instruments provide indication of both power level and reactor period as well as an automatic reactor shutdown if the period becomes too short.

PWR Nuclear Instrumentation (Cont.)

b) Two intermediate-range detectors are located in the same wells as are the source-range instruments. These instruments are usually compensated ion chambers. The gamma ray component of their signal is therefore automatically deleted. The intermediate-range detectors span approximately eight decades, are used while the reactor is critical but below the "point-of-adding-heat," and are left continuously on. They, like the source-range instruments, provide indication of both power level and reactor period as well as an automatic shutdown on short period.

c) Four power-range instruments are located on the corners of the core. Each consists of an uncompensated ion chamber which is as long as the core is high. The instrument senses both gamma rays and neutrons. This is allowable in the power-range because the gamma ray flux is proportional to the neutron flux in this range of reactor operation. The power-range instruments span approximately four decades. Their inner electrodes are divided into two equal sections thereby providing separate information on the upper and lower halves of the core. This data is used to determined the percent power being generated in the upper and lower portions of the core and to position some of the movable control rods in order to prevent excess power peaking. The power-range instruments provide indication of both power level and reactor period, automatic shutdown on short period, and automatic shutdown on high power.

Range and Location of Nuclear Instrumentation in a Pressurized Water Reactor

(Adapted from Masche)

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Limiting Conditions for Startup

— Certain limiting conditions for operation, such as linear heat generation rate and the avoidance of pellet-clad interaction, apply under all conditions. During startup, operation is further restricted by the need to observe:

- Maximum pressure to avoid brittle fracture.
- Minimum temperature for criticality safety*.
- Minimum pressure for coolant pump operation.

*This reflects the need to maintain a negative moderator temperature coefficient of reactivity.

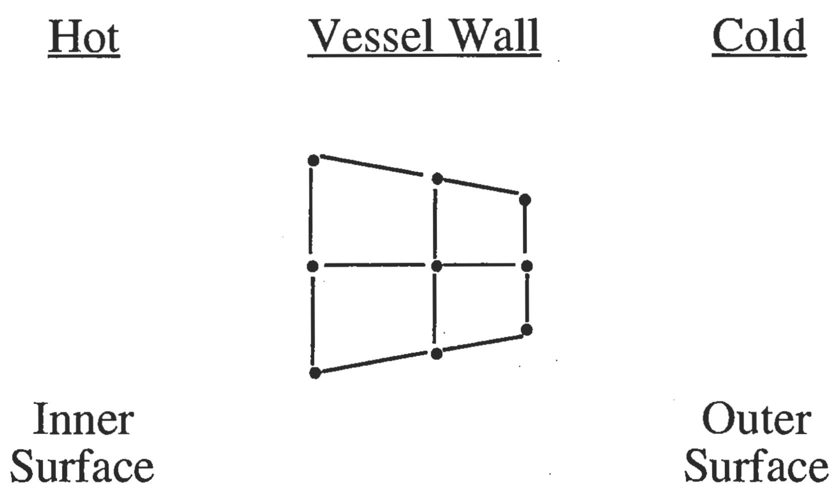
Maximum Pressure to Avoid Brittle Fracture

— Reactor pressure vessels are made of carbon steel which has a body-centered cubic lattice structure and is therefore subject to brittle fracture. That is, should the vessel temperature drop below a certain value and the vessel be subject to a high pressure, the vessel could fail in a brittle (i.e., catastrophic) rather than in a ductile manner. The temperature at which the failure mode shifts from ductile to brittle is called the nil ductility temperature or NDT. It is a function of the reactor vessel material and the integrated neutron exposure of the vessel. The NDT rises as the fluence seen by the vessel increases.

— The maximum allowed pressure of the primary system is determined by the ductility of the pressure vessel steel. As temperature decreases, so does the allowed pressure because the vessel steel is more susceptible to failure at low temperature. Stresses are different for heatup and cooldown and hence there are two separate limiting curves of pressure versus temperature.

Stress During Heatup/Cooldown

Heatup:



The dots denote atoms in the lattice of the metal. The atoms on the inner wall seek to expand. But this expansion is restrained by chemical bonds to the atoms of the rest of the wall. So, the atoms of the inner wall are in compression.

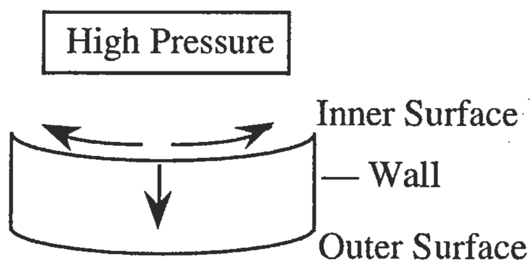
Cooldown:

The reverse occurs during cooldown. The atoms of the inner wall seek to contract. But they are restrained from so doing because of the atoms of the middle part of the wall. That part of the wall is still hot because of the time required for heat to be conducted out of it. So, those atoms are still far apart. The inner wall is therefore in tension.

Stresses on Inner Wall of Pressurized Vessel

— There are three factors that contribute to the net stress in the azimuthal direction on the inner surface of a pressure vessel wall.

a) Pressure Effect: The internal pressure tank puts the vessel wall in tension in the azimuthal direction and in compression in the radial direction.



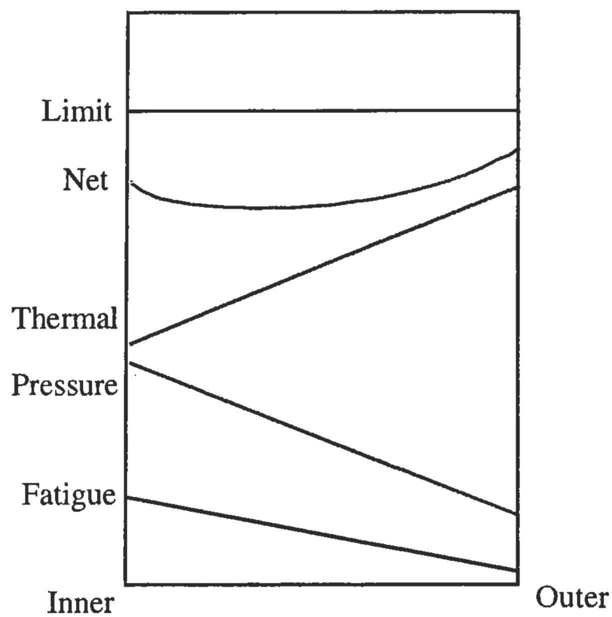
The azimuthal stress is greatest on the inner surface because the surface area is least there. (Stress is force per unit area.)

b) Thermal: Stress will be compressive for heatup; tensile for cooldown.

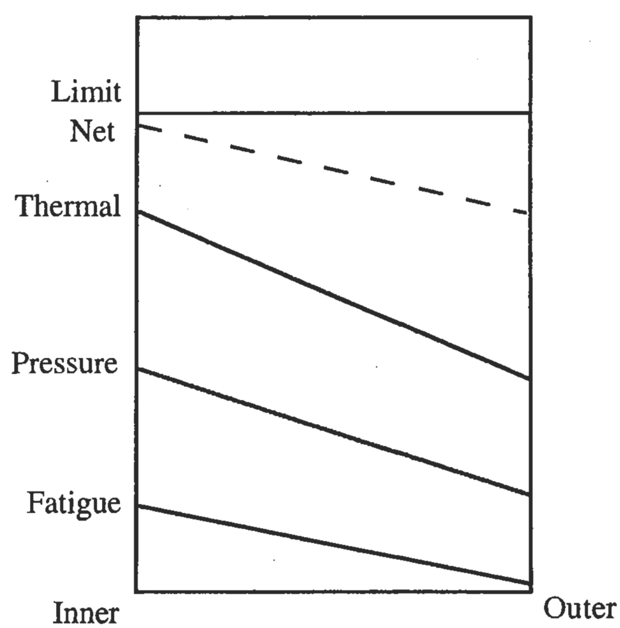
c) Accumulated Fatigue: Stress will be tensile on inner wall.

Net Stress on Inner Wall of Pressure Vessel

Heatup:



Cooldown:



Net stress is more limiting during cooldown. This is basis of the "thermal shock issue" in PWRs.

Moderator Temperature Coefficient of Reactivity

— The function of a moderator is to slow down or thermalize the fast neutrons that are produced from fission. Safety can be enhanced by deliberately under-moderating a core so as to create a negative moderator temperature coefficient of reactivity. Under such conditions, a heatup of the moderator causes negative reactivity because the moderator expands and hence thermalizes fewer neutrons. The sequence is:

$$P \uparrow T_H \uparrow P_d \rightarrow T_c \uparrow \rho \downarrow \delta k \downarrow P \downarrow$$

where P is core power, T_H and T_C are the hot and cold leg temperatures, P_d is demanded power, ρ is density, and δk is reactivity.

There are two safety restrictions placed on reactor operation as the result of temperature. First, the moderator temperature coefficient of reactivity must be either zero or negative prior to the reactor's being taken critical. Second, if the moderator temperature coefficient of reactivity is positive, then the reactor must be kept subcritical by an amount equal to or greater than the reactivity that would be inserted should there be a depressurization.

(Note: There are some exceptions to these rules. The technical specifications of some plants do permit them to be taken critical in the presence of a positive moderator coefficient provided that the magnitude of the coefficient is small. All plants must have a negative moderator temperature coefficient at their normal operating temperature.)

Minimum Temperature for Criticality

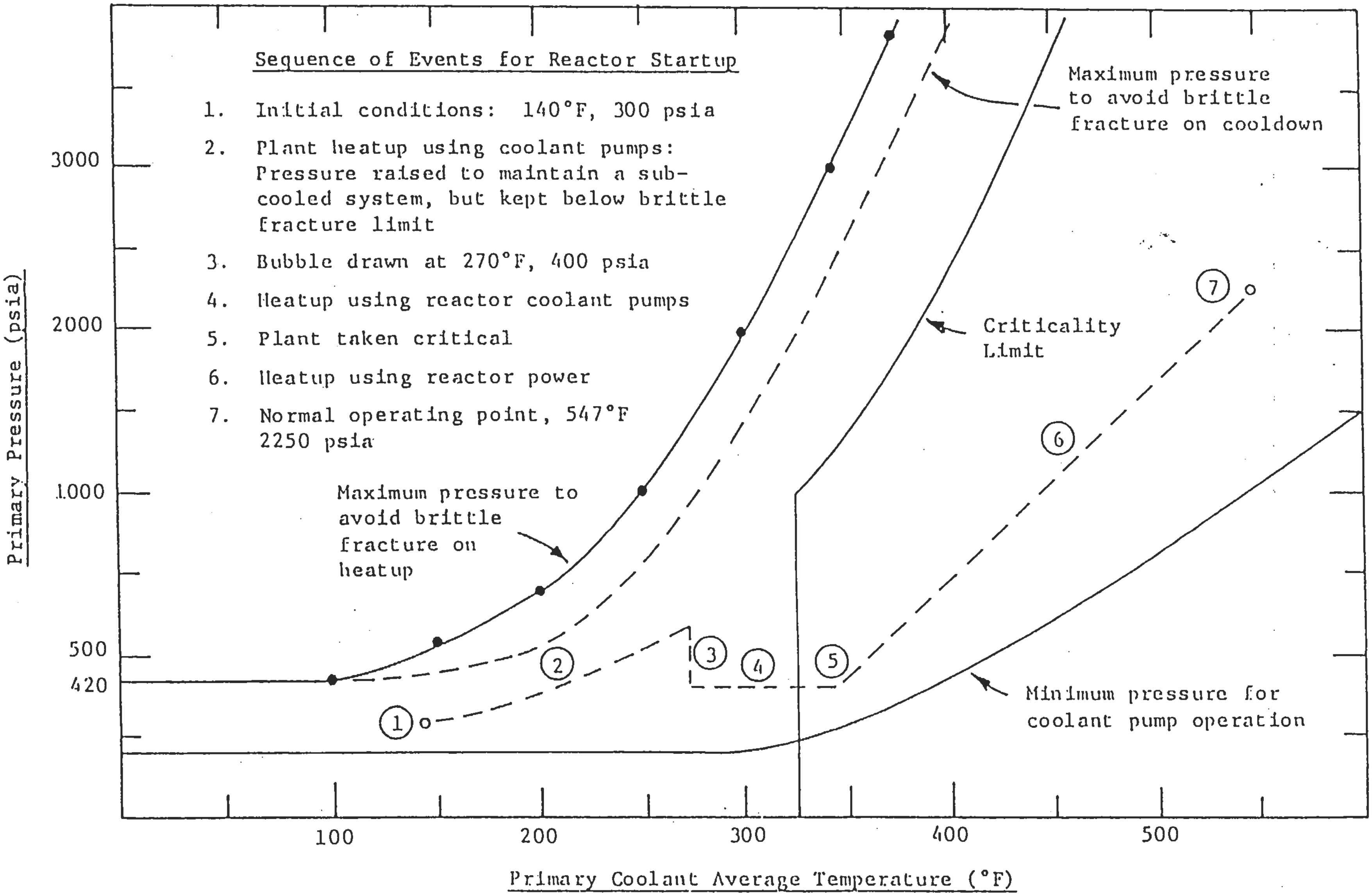
— In a PWR, the maintenance of a negative moderator temperature coefficient of reactivity is complicated by the presence of soluble poison (boric acid).

— An increase in the temperature of the moderator will cause it to expand and, as a result, there will be fewer thermalizing collisions between the high-energy neutrons produced from fission and the moderator nuclei. The result will be a decrease in reactivity. But, countering this negative effect is the fact that the moderator contains soluble boron poison. Hence, when the moderator expands due to a temperature rise, some of the dissolved boron is removed from the core. This causes a positive reactivity effect. Whenever the plant is at operating temperature (547 °F), the loss of moderation by the hydrogen nuclei in the coolant outweighs the decrease in absorption by the boron nuclei in the soluble poison. Hence, the net effect of an increase in moderator temperature will be to insert negative reactivity. However, at lower temperatures, the decrease in absorption dominates and the moderator coefficient of reactivity is positive. The crossover point is the minimum temperature for criticality. Its value, at low pressure, is about 325 °F. This temperature will be a function of pressure because there is a slight change in coolant density and therefore boron concentration with pressure. As the pressure increases, the plant must be operated at a higher temperature in order to ensure that the loss of moderation is the major effect following a temperature increase.

Minimum Pressure for Coolant Pump Operation

— Net positive suction head (NPSH) is defined as the pressure at the pump suction position less the saturation pressure the corresponds to the temperature of the fluid being pumped. It if is sufficiently positive, a centrifugal pump can operate. Otherwise, water will flash to steam in the eye of the pump and cause the pump to either cavitate or become gas-bound. Hence, there is a certain minimum pressure that must be maintained if the primary coolant pumps are being operated. This minimum pressure increases as plant temperature rises because the saturation pressure rises with temperature.

Pressure-Temperature Limits for a PWR



Explanation for Pressure and Temperature Limit Curves

— Brittle Fracture: There are two curves showing the maximum pressure to avoid brittle fracture, one for heatup and one for cooldown. The one for cooldown is the more limiting because thermal, pressure, and fatigue stresses are all tensile in this case. The primary system pressure must be kept below the appropriate curve at all times. The brittle fracture curve will shift to the right and become more restrictive as the pressure vessel incurs neutron damage.

— Criticality Safety: There is one curve for the minimum temperature for critical operation. If the reactor is critical, then the system temperature must be maintained to the right of the curve so that the moderator temperature coefficient of reactivity will be negative. As a result, the consequences of both depressurization and continuous rod withdrawal accidents will be less severe.

— Coolant Pumps: There is one curve showing the minimum pressure at which the reactor coolant pumps may be operated. Plant pressure should be kept above this curve. (Note: The curve shown is for rated flow. Less stringent curves apply if the pumps are run at reduced speed.)

PWR Startup Sequence

1. Performance of precritical checklists.
2. Adjustment of steam generator water chemistry.
3. Coordinate pressure-temperature control of primary via charging /letdown systems.
4. Energize pressurizer heaters and open spray valve to heat up pressurizer.
5. Change nitrogen gas to pressurizer to create a gas-filled volume that will dampen pressure pulses.
6. Start primary coolant pumps if above NPSH curve. Raise primary temperature using pump heat until above minimum temperature for the attainment of criticality.
7. Adjust chemistry of primary coolant.
8. Reduce concentration of soluble boron so criticality can be attained.
9. Draw a steam bubble in the pressurizer (270 °F, 400 psi).
10. Drain steam generators so level is within operating range.
11. Recirculate condenser hot well to eliminate stratification.
12. Take reactor critical by first withdrawing the full-length shutdown rods and then the full-length control rods. The latter are used to create subcritical multiplication.

PWR Startup Sequence (Cont.)

13. Position control rods for a 0.5 decade per minute startup rate. Take critical data at 10^{-8} amperes. Raise power to point-of-adding heat (1-3%).

14. Use the reactor to heat the primary at rate of 50 °F per hour. Drain coolant to compensate for density changes.

15. Once the primary temperature exceeds 400 °F, admit steam to the secondary by opening the bypasses on the main steam isolation valves. Commence warm-up of turbine.

16. Continue heatup and pressurization of primary. At 1900 psia, place high pressure injection systems on-line. At 2250 psia, place pressurizer heaters/spray in automatic. Shift nuclear instrumentation to "run" mode to correct for decrease in neutron attenuation because of hotter coolant. Raise power to 10% and open main steam stops.

17. Place steam generators/secondary feed pumps in automatic.

18. Adjust control rods/soluble boron to compensate for build-up of short-lived fission product poisons such as xenon.

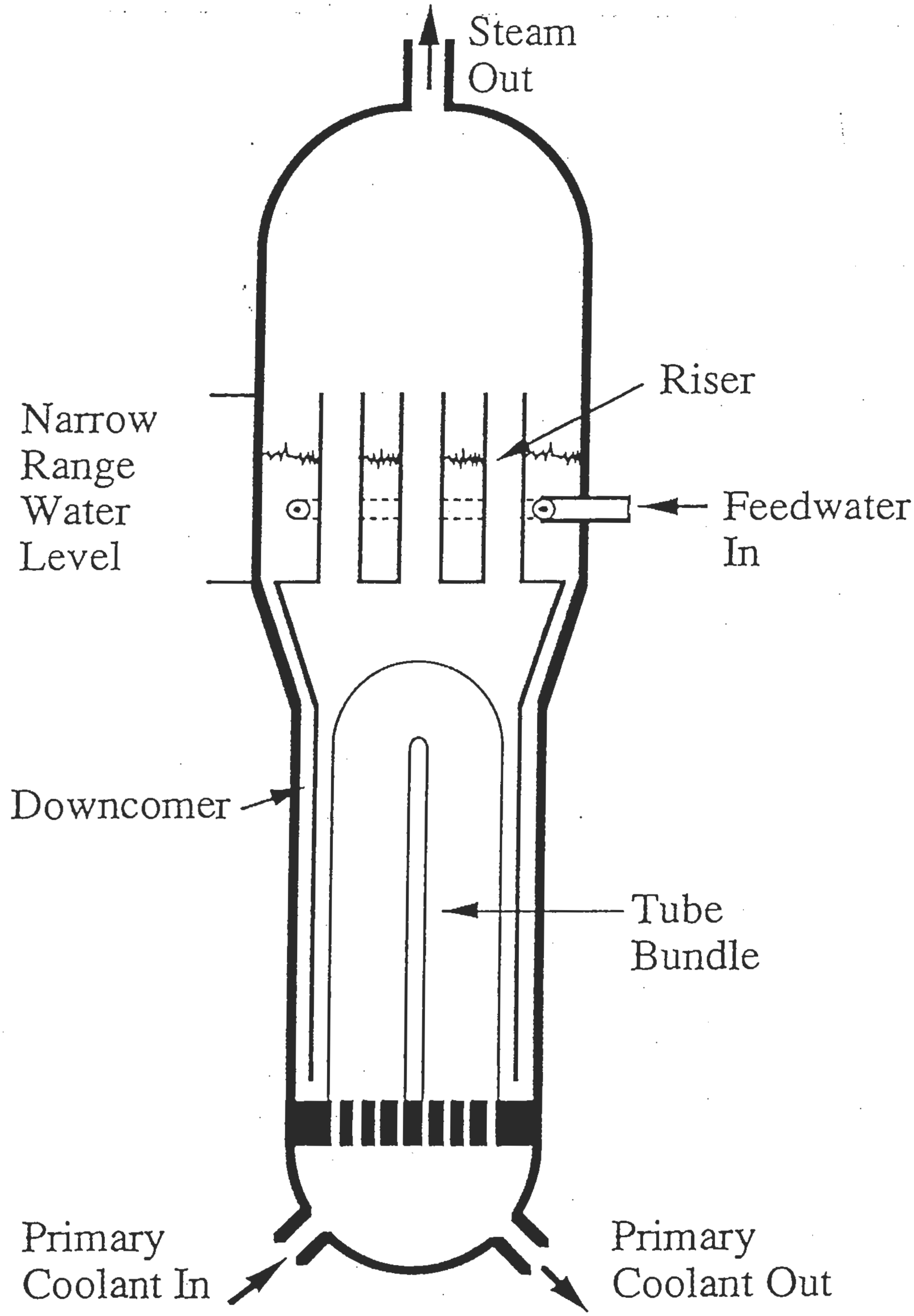
19. Adjust control rods/soluble boron to compensate for fuel depletion. The full-length control rods are left partially inserted to provide a means to alter reactor power in response to changes of load. Use part-length rods to correct for power distribution anomalies.

Operational Issues During Startup

— Nuclear Instrumentation: A PWR's nuclear instruments are located outside the core where they measure the neutron leakage flux. That flux is attenuated by the water that is in the core vessel. The degree of attenuation and hence the readings of the instruments varies with the water temperature. The instruments are calibrated for hot operation. So, unless a correction is applied, the instruments will indicate that power is lower than actual whenever the plant is cold.

— Pressurizer Level: The indicated level needs to be corrected for the change in density as the plant heats up. Also, it may not be possible to drain water from the pressurizer fast enough to allow a 50 °F per hour heat up rate.

— Steam Generator Level: Loss of steam generator level during startup has been a major cause of spurious plant shutdowns. The level controller uses three signals: steam flow, feed flow, and measured level. At low power, the steam flow signal is erratic and the level controller can not be used on automatic. Manual operation is also a problem because of the counterintuitive effects known as "shrink" and "swell." If steam flow increases, indicated level will initially rise. The operator should be increasing feed flow. But, he might reduce it by mistake because of the transient "shrink" effect.



U-TUBE STEAM GENERATOR

Shrink and Swell

— Steam generator level 'shrink and swell' refers to temporary changes in the water level in the downcomer region that occur whenever steam bubbles collapse or form in the tube bundle region. For example, on collapse of steam bubbles, the volume taken by the two-phase mixture suddenly decreases and is filled by liquid from the downcomer region. Hence, the indicated and measured levels, which are obtained from the downcomer, drop even though the mass of fluid in the steam generator has risen. This phenomenon is referred to as level shrink. Swelling is essentially the reverse effect.

— Shrink and swell are caused by changes in feedwater flow, steam flow, or primary temperature.

— Difficulties caused by shrink and swell:

- Potential uncovering of tube bundle

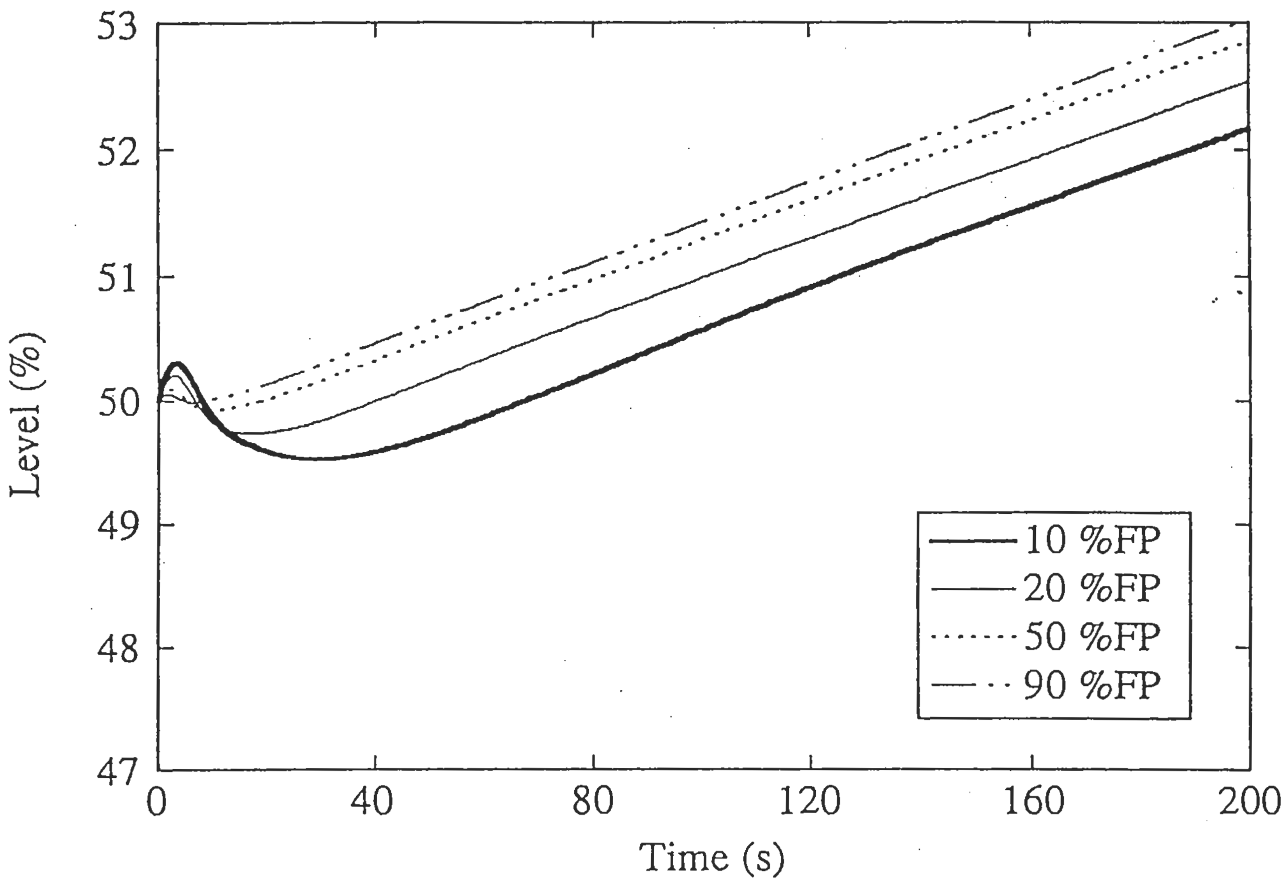
  ⇒ Loss of heat removal

- Potential carryover of water

  ⇒ Turbine damage

- Plating out of chemicals used to control water chemistry

  ⇒ Materials damage to tube bundle



STEAM GENERATOR LEVEL SHRINK EFFECT AS RESULT OF A STEP INCREASE IN FEEDWATER FLOWRATE

Safety Issues During Startup

— There are several safety issues that are unique to PWR startup.

— Loss of primary system pressure while critical during a heatup. Given that the plant must be critical in order to obtain energy necessary to accomplish the heatup, neither the low pressure scram nor the safety injection system are functional. Thus, there is no automatic protection against a loss of pressure during startup.

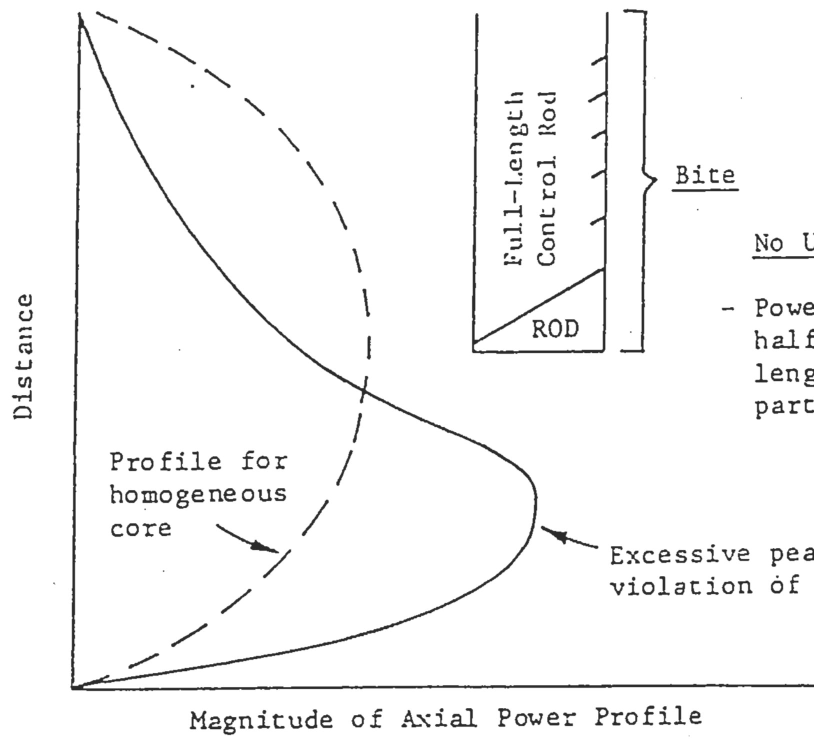
— Continuous rod withdrawals while below the "point-of-adding-heat." This is of concern because there would be no countering effect from the negative reactivity feedback because there would be no accompanying heatup of the moderator.

— A steam line rupture while above the "point-of-adding-heat" but still at low power. The sudden decrease in steam pressure on the secondary side would cause the steam flow rate to increase thereby cooling down the primary and adding positive reactivity. Power would increase rapidly. This is why steam is first brought into the secondary by opening the bypass valves and not the main steam isolation valves.

— A stuck control rod. This is of concern because failure to properly position a rod could result in an improper linear heat generation rate once the plant has attained full power.

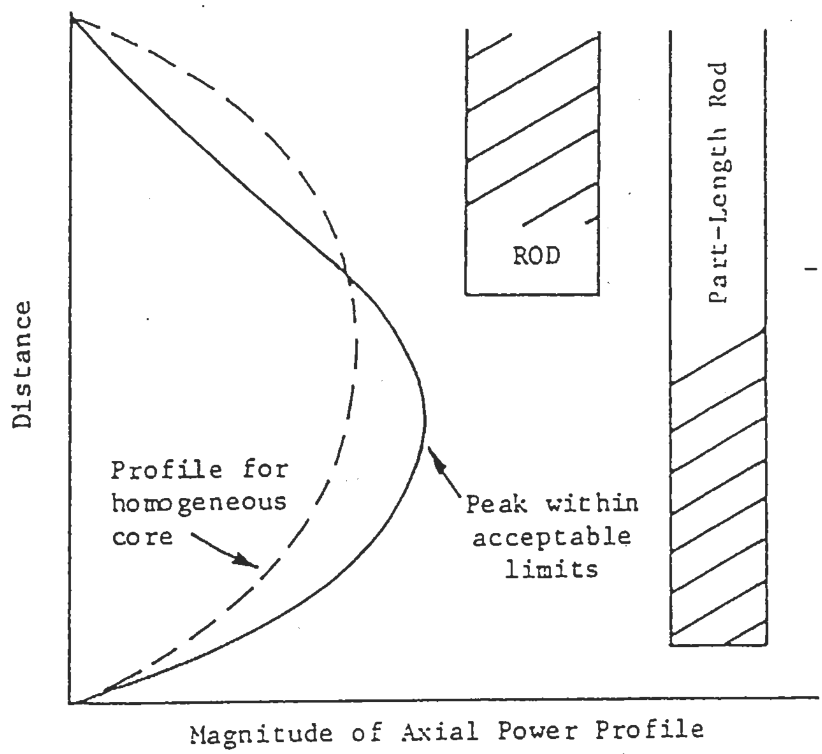
— Power shaping. See next viewgraph.

Effect of Rod Position on Power Shape



No Use of Part-Length Rod

- Power profile peaks in lower half of core since full-length control rod is partially withdrawn.



Use of Part-Length Rod

- The lower half of the part-length rod contains poison. It serves to depress the flux peak in the lower half of the core.

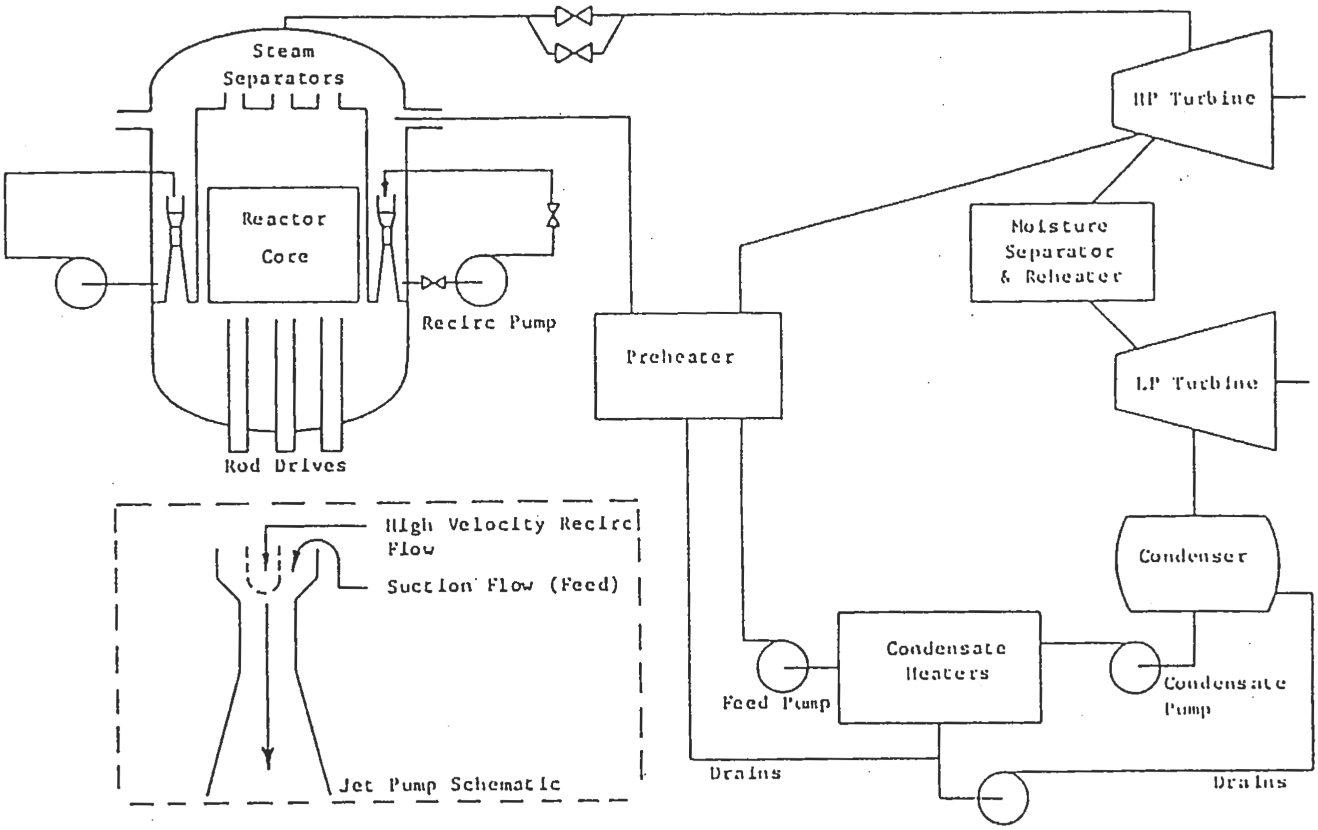
Boiling Water Reactor Startup

— The topics covered relative to BWRs are:

- Systems/components for reactivity adjustment
- Nuclear Instrumentation
- Startup Sequence
- Limiting Conditions for Startup

Note: A BWR differs from a PWR in that the source of pressure is the boiling process that occurs in the core. There isn't a separate system for pressure control.

Schematic Diagram of Boiling Water Reactor



BWR Reactivity Control

Two mechanisms are available for reactivity control. These are:

1. Movable Control Rods: One cruciform-shaped rod is present for every four fuel assemblies. These rods, which contain boron carbide, are driven upwards from the bottom of the core. Power changes in excess of 25% are accomplished with these rods.

2. Negative Moderator and Void Reactivity Coefficients: The negative moderator and void reactivity feedback coefficient is within the control of the operator. It is used to accomplish power changes of as much as 25% of rated. Increasing the recirculating pump speed increases the flow rate through the core which, in turn, increases the heat transfer and decreases the vapor fraction. Therefore, the average density of the liquid-vapor mixture that is flowing through the core increases and so does the neutron moderation. The net effect is that positive reactivity has been added to the core. The reactor power rises until increased vapor formation and fuel temperature increases reverse the process. The reactor then settles out so that it is critical but at a higher power level. Decreasing the recirculation flow will have the opposite effect.

BWR Nuclear Instrumentation

— In BWRs, all detectors are gas-filled and are located within the core. This is necessary because local power anomalies are possible as the result of the boiling process. There are three sets of detectors chosen to have overlapping ranges:

a) Four source range detectors are located within the core, one to a quadrant. These are usually pulse-mode fission counters that are operated as proportional counters. They are withdrawn from the core once the reactor is at appreciable power.

b) Eight intermediate range detectors that are also located in-core. These are fission chambers that are operated as ion chambers in the Campbelling (MSV) mode.

c) The power range contains 144-164 fission chambers that are operated as ion chambers in the current mode. These are positioned in the fuel modules so that almost every module has one.

BWR Nuclear Instrumentation (Cont.)

The power-range fission chambers are referred to as local power range monitors (LPRMs). Four such monitors, selected to provide a core-wide measure of the power, feed a single average power range monitor (APRM). The ratio of the reading from an APRM to each LPRM is the peaking factor for the fuel channel associated with the individual LPRM. This factor which includes radial, axial, and local peaking must not exceed 2.4. (Note: The average linear heat generation rate is 6.5 kW/foot, the maximum allowed is 13.6 kW/foot.) A process computer is used to compute the APRMs. Once these data (the LPRMs and APRMs) have been determined, the process computer is used to evaluate the margins to the reactor's thermal limits. In the event that further movement of a control rod might cause one of these limits to be exceeded, that rod's withdrawal is automatically blocked.

BWR Startup

— The major steps in a BWR startup are:

1) The recirculation pumps are started and set at the desired speed, 28% of rated.

2) The reactor is made critical by the withdrawal of the control rods.

3) The control rods are further withdrawn thereby placing the reactor on a positive period. Power is raised to above the "point-of-adding-heat" and the core vessel is pressurized. In accordance with the safety limit specifications power may not be increased above 25% until the plant pressure exceeds 800 psia.

4) Once the pressure rises above 800 psia, the reactor power is raised to 55% of rated. A check is then made of the estimated critical position and the rod pattern.

5) The recirculation pump speed is then increased so that total core flow rises to 100%. As it does, core power also rises due to the moderator temperature – void reactivity coefficient. Power is then leveled at the desired value.

This procedure takes approximately eighteen hours to complete. Not mentioned in the above summary are many other significant factors such as heatup limitations due to thermal stress.

BWR Operating Curves

— The figure in the next viewgraph shows the relations between thermal power and core flow that must be observed for a boiling water reactor. The initial conditions for startup are 0% power and 36% flow. (Note: The percent flow through the core exceeds the recirculation pump speed (28%) because of the effect of the jet pumps.) Once critical, the plant is coordinated so that the power and flow move along the "28% pump speed line." The recirculation pump speed is kept constant while moving along this line. The reason for the increase in flow as power rises is due to natural circulation (i.e., thermally-induced flow). The line marked 'maximum power' must not be crossed until the plant is above 800 psia. Once above the pressure, power is raised to 55% of rated. The corresponding core flow is 45%. This point is the intersection of the "28% pump speed line" and the "design flow control line." The control rods and the recirculating pump speed are then adjusted as needed to move the plant's power and flow along the "design flow control line" until conditions of 100% power and 100% flow are attained.

BWR Thermal Power and Core Flow Restrictions