BWR ENGINEERED SAFETY FEATURES

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INTRODUCTION

In a presentation at your March, 1974 seminar, you were given an introduction to the General Electric Boiling Water Reactor. For this seminar, you have requested a discussion of the GE BWR engineered safety features. This is the title of Section 6 of the Standard Safety Analysis Report, and I used this section as my guide. From my previous discussions with you, I believe that you are knowledgeble about the basic safety philosophy employed in the design of Nuclear Power Plants (such as defense-in-depth, multiple barrier protection, redudancy, etc.) and that you are up-to-date on the latest public safety concerns. Therefore I will address specific safety features that are provided to mitigate the consequences of postulated accidents. In general, these features are,

Containment Systems

Primary Containment

Secondary Containment (shield building around containment shell)

Containment Heat Removal System

Containment Isolation System (to close valves on lines to primary system and containment)

Combustible Gas Control (to control H_2 from Zr-water reactions)

Emergency Core Cooling Systems

High Pressure Core Spray (HPCS)

Automatic Depressurization System (ADS)

Low Pressure Core Spray (LPCS)

Low Pressure Coolant Injection (LPCI)

Habitability Systems – (such as control room ventilation and shielding)

Standby Gas Treatment System (to control halogens and particulates after a loss of coolant)

Other Engineered Safety Features

Overpressurization Protection

Main Steam Line Isolation Valves

Control Rod Drive Support System

Control Rod Velocity Limiter

Main Steam Line Flow Restrictor

Reactor Core Isolation Cooling (RCIC) System

For conciseness, all of these features can not be described here, but the systems I have included are representative.

CONTAINMENT SYSTEMS

Containment systems refers to the primary containment (a steel clyndrical shell), secondary containment (shield building of reinforced concrete), containment heat removal system (residual heat removal system), containment isolation system, and the combustible gas control system.

The containment systems must meet certain design bases. The containment and drywell has to withstand the peak pressures and temperatures which occur as a result of the loss of coolant accident, LOCA. The containment has to limit fission product leakage during and following the LOCA. Jet forces and missiles from any pipe rupture must not compromise the functional capability of the drywell or containment. There must be rapid isolation capability of all pipes penetrating the containment.

The Mark III containment system meets these design bases. It has been approved by the US NRC, and at least one plant has received its construction permit. Figure 1 shows the Mark III containment and shield building. The containment vessel is a free standing steel cylinder with a torispherical head. It is anchored into a reinforced, steel lined, concrete slab. Reinforced concrete is used for the shield building which serves as a secondary fission product barrier and protects the containment from external missiles. A five foot annulus gap provides a plenum for the collection and filtration of any fission product leakage from the containment after a LOCA. This annulus gap is kept at a negative pressure so that leakage is inward, and through a controlled path.

The drywell is an unlined, reinforced concrete structure. It provides shielding, a structure to support the upper pool, a path for the air-system-water mixture through the suppression pool, and protection for the steel containment from internal missiles or pipe whip.

Figure 2. Horizontal Vents with Flow.

The horizontal vents and weir wall form the path for the air-steam-water mixture following a LOCA. A build up of pressure in the drywell forces the water behind the weir wall down, and sequentially uncovers the horizontal vents. As the vents are uncovered, the LOCA mixture flows through the suppression pool wherein energy is absorbed, thereby reducing the resulting containment pressure.

Figure 3., Containment and Drywell Pressure Response, shows the pressure response of the Mark III drywell and containment. The drywell pressure peaks at about one second as the vents open. As flow from the LOCA decays, the peak drywell pressure drops and the vents reclose. After 600 seconds, the ECCS Coolant cascades into the drywell, steam is condensed and the drywell

pressure equals the containment pressure. In the long term (4 to 6 hours) the peak containment pressure of 12 psig is controlled by the residual heat removal system.

This performance is based on extensive test programs (large scale and reduced scale) and corresponding analytical models which are still underway.

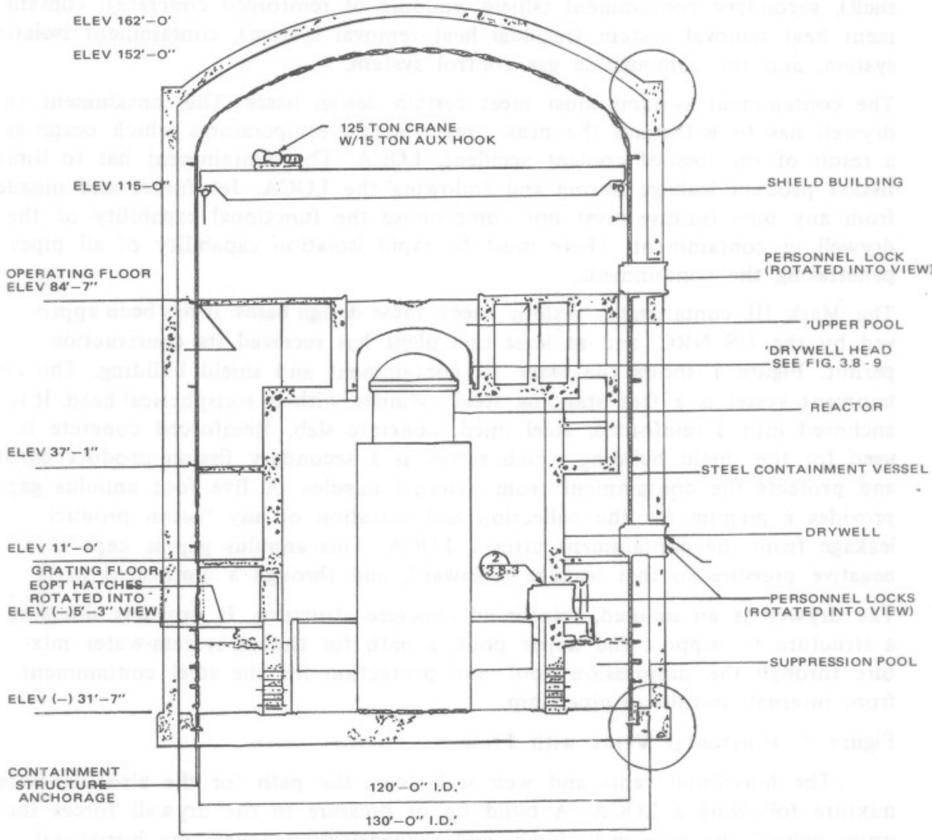


Figure 1. Mark III Containment and Shield Building

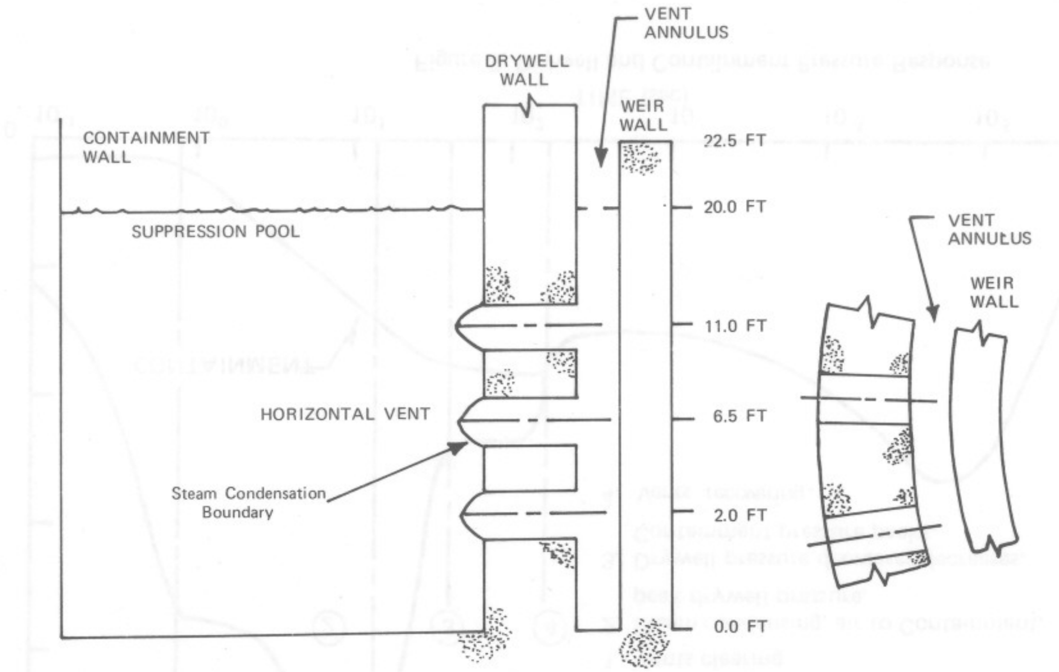


Figure 2. Horizontal Vents with Flow

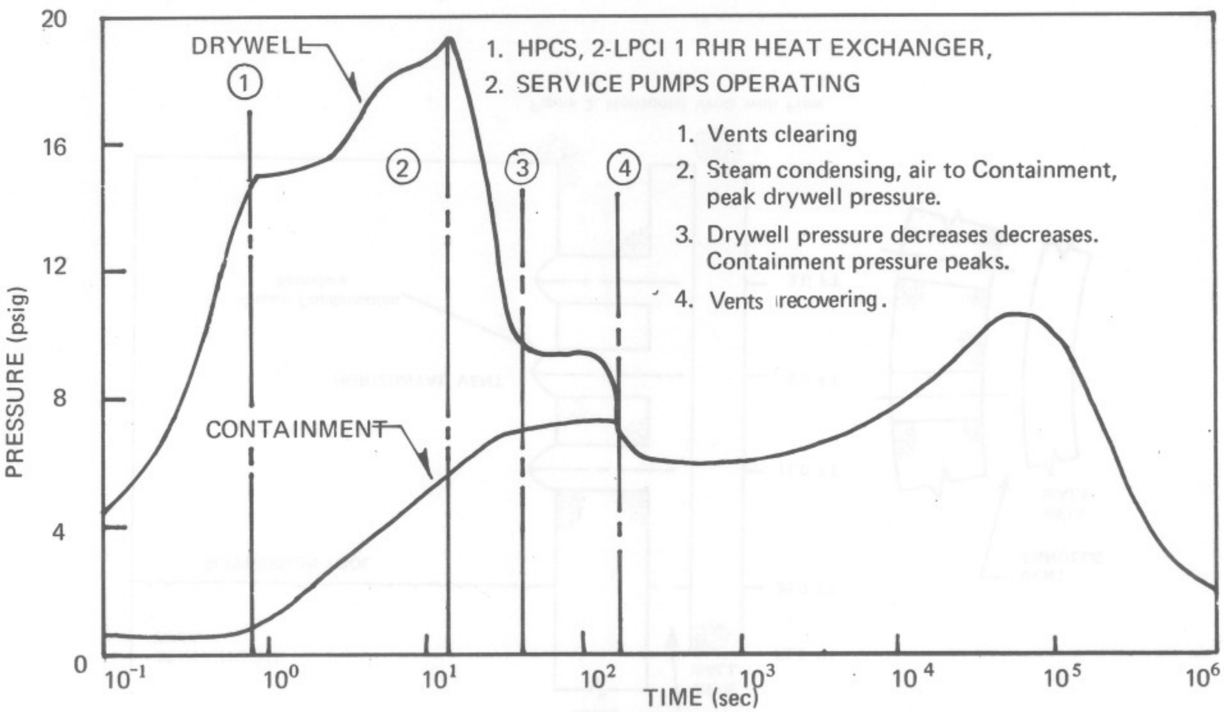


Figure 3. Drywell and Containment Pressure Response

EMERGENCY CORE COOLING SYSTEMS

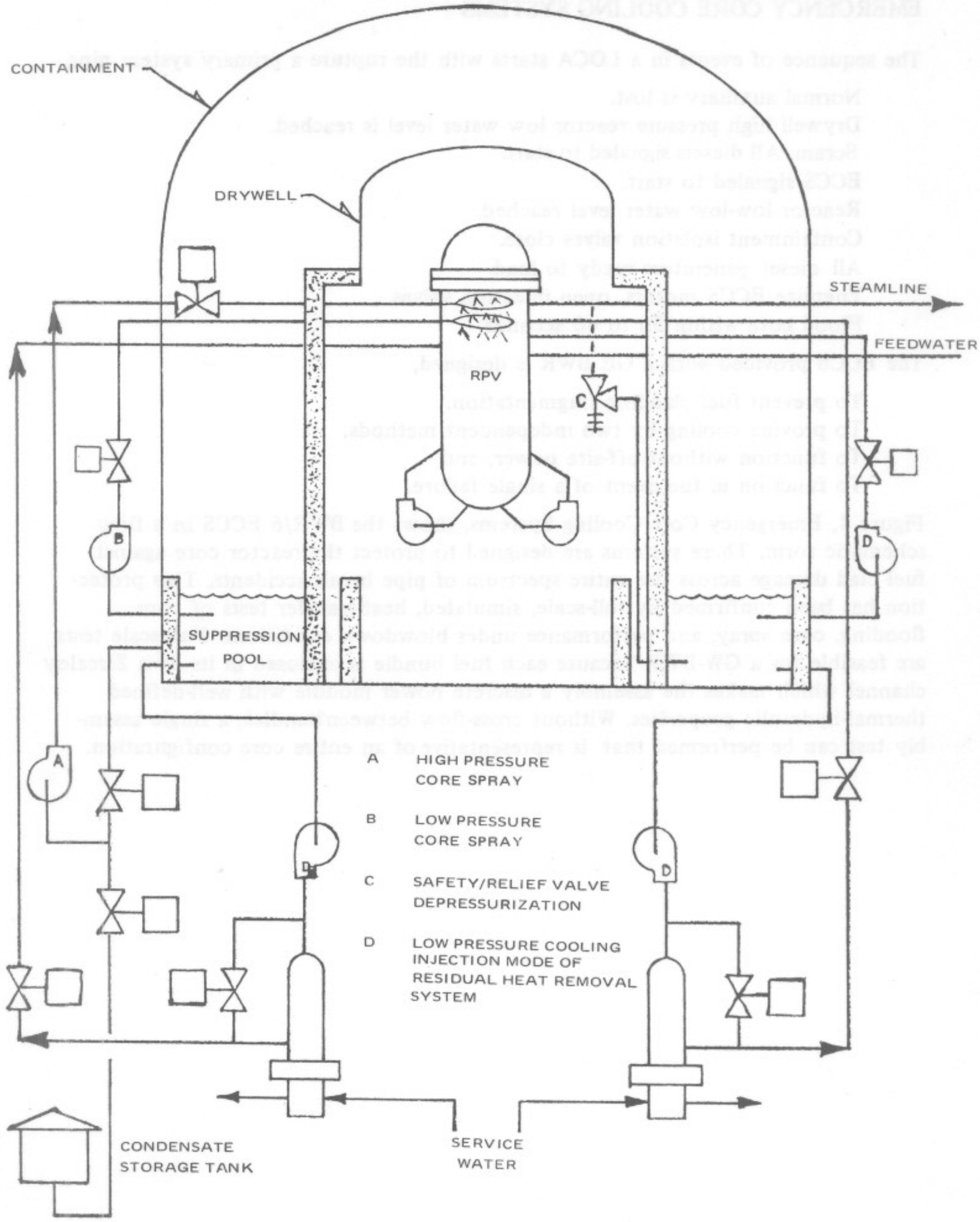
The sequence of events in a LOCA starts with the rupture a primary system pipe.

Normal auxiliary is lost.
Drywell high pressure reactor low water level is reached.
Scram. All diesels signaled to start.
ECCS signaled to start.
Reactor low-low water level reached.
Containment isolation valves close.
All diesel generators ready to load.
Energize ECCS motors, open injection valves.
Flood core within 30 to 40 seconds.

The ECCS provided with a GE BWR is designed,

To prevent fuel cladding fragmentation,
To provide cooling by two independent methods,
To function without off-site power, and
To function in the event of a single failure.

Figure 4, Emergency Core Cooling Systems, shows the BWR/6 ECCS in a flow schematic form. There systems are designed to protect the reactor core against fuel clad damage across the entire spectrum of pipe break accidents. This protection has been confirmed by full-scale, simulated, heat-transfer tests of core flooding, core spray, and performance under blowdown conditions. Full-scale tests are feasible for a GW-BWR because each fuel bundle is enclosed in its own Zircaloy channel which makes the assembly a discrete power module with well-defined thermal-hydraulic properties. Without cross-flow between bundles, a single assembly test can be performed that is representative of an entire core configuration.



EMERGENCY CORE COOLING SYSTEM

Figure 4.

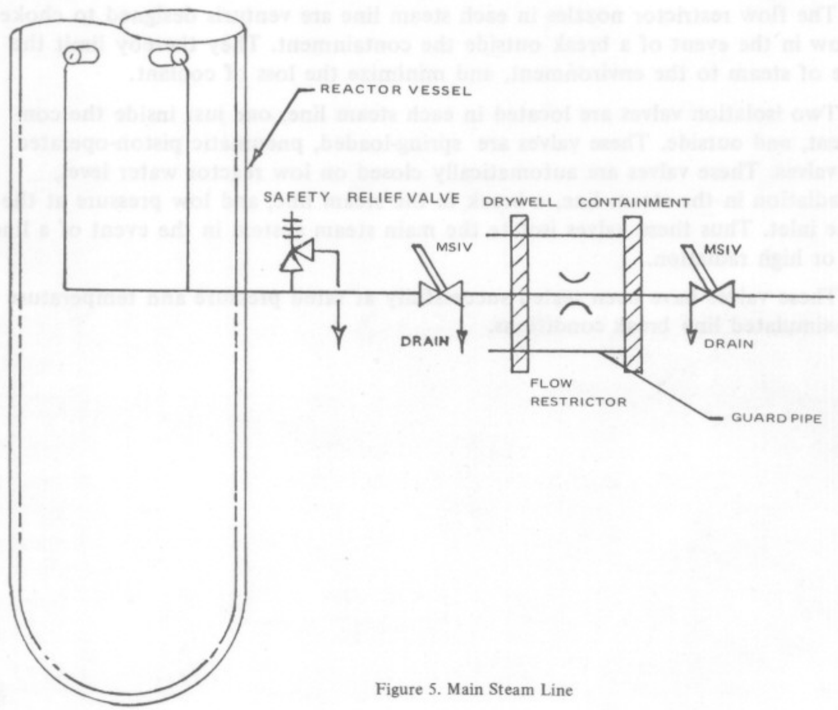


Figure 5. Main Steam Line

MAIN STEAM ISOLATION VALVES; MAIN STEAM FLOW RESTRICTOR; SAFETY RELIEF VALVES

Figures 5, Main Steam Line, and Figure 6, Main Steam Isolation Valve show three engineered safety features : the main steam isolation valves, flow restrictors, and the safety-relief valves for overpressurization protection. Steam exits from the vessel through nozzles and into four steam lines.

The safety-relief valves are dual function valves that discharge to the suppression pool. The safety function (conventional spring device) provides overpressure protection in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels. The relief function is provided by a pneumatic device that actuates the safety valve lifting arm upon a high pressure signal.

The flow restrictor nozzles in each steam line are venturis designed to choke the flow in the event of a break outside the containment. They thereby limit the release of steam to the environment, and minimize the loss of coolant.

Two isolation valves are located in each steam line, one just inside the containment, and outside. These valves are spring-loaded, pneumatic piston-operated globe valves. These valves are automatically closed on low reactor water level, high radiation in the steam line, a break in the steam line, and low pressure at the turbine inlet. Thus these valves isolate the main steam system in the event of a line-break or high radiation.

These valves have been tested successfully at rated pressure and temperature under simulated line break conditions.

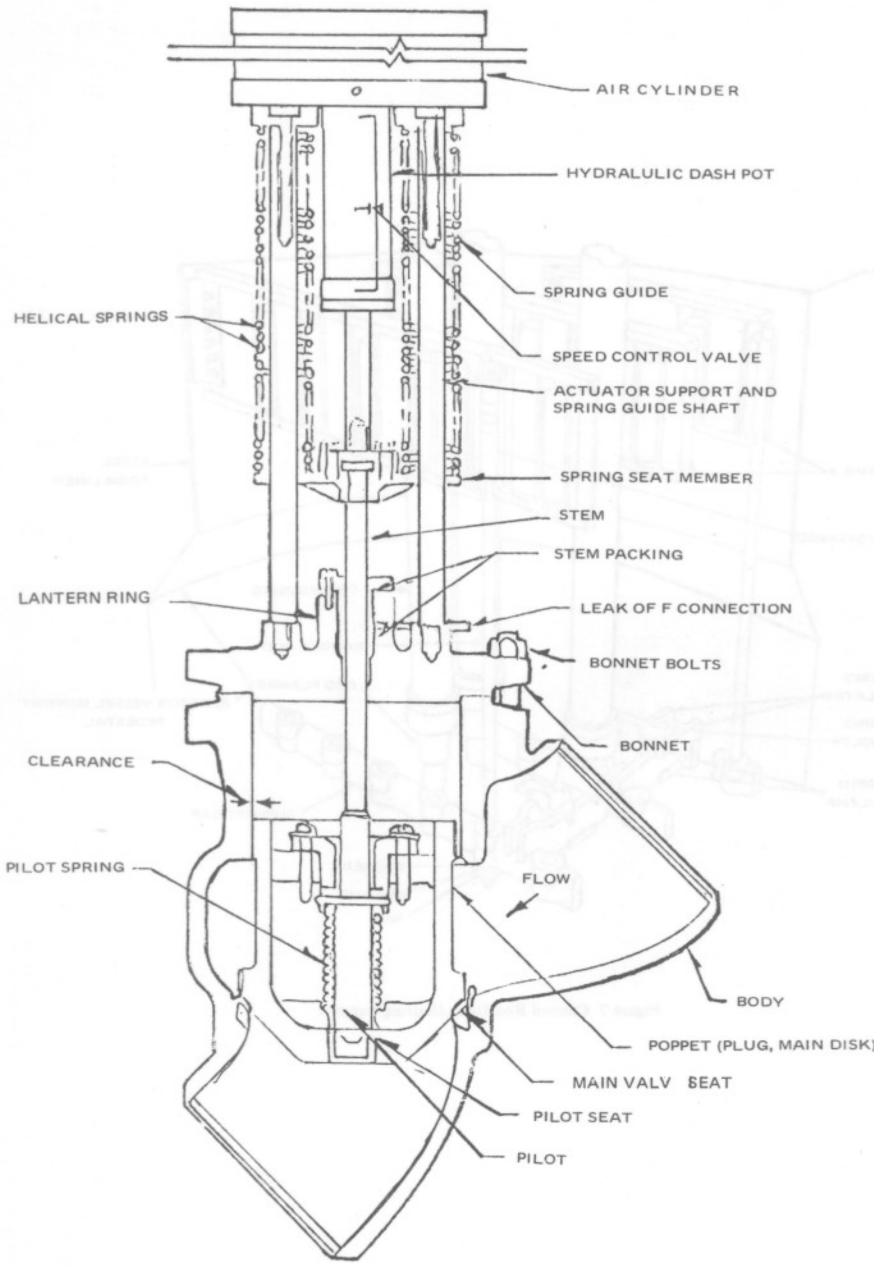


Figure 6 Main Steam Isolation Valve

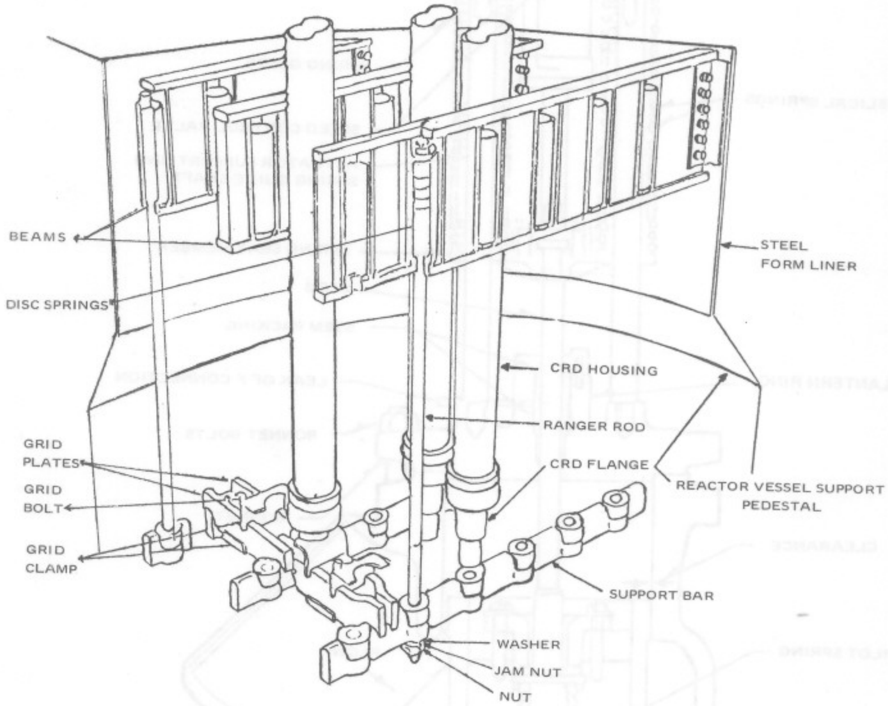


Figure 7 Control Rod Drive Housing Support

CONTROL ROD DRIVE HOUSING SUPPORTS

The control rod drive (CRD) housing supports prevent any significant reactivity addition in the event a drive housing breaks and separates from the bottom of the reactor vessel. Thus, following a postulated CRD housing failure, control rod downward motion is limited so that the resulting nuclear transient is not sufficient to cause fuel damage.

The CRD housing supports are shown in Figure 7. Horizontal beams are installed immediately below the bottom head of the reactor vessel, between the rows of CRD housings.

Hanger rods are supported from the beams on stacks of disc springs. Support bars and grids are bolted between the bottom ends of the hanger rods.

In the postulated CRD housing failure, the CRD housing supports are loaded when the lower contact surface of the CRD flange contacts the grid. The resulting load is then carried by two grid plates, two support bars, four hanger rods, their disc springs, and two adjacent beams.

Control rod movement following a housing failure is substantially limited below one drive "notch" movement (6 in.). Sudden withdrawal of any control rod through a distance of one drive notch at any position in the core does not produce a transient sufficient for any damage.

CONTROL ROD VELOCITY LIMITER

The control rod velocity limiter, Figure 8 is an integral part of the bottom assembly of each control rod. This engineered safeguard protects against a high reactivity insertion rate by limiting the control rod velocity in the event of a control-rod-drop accident. It is a one-way device in that the control rod scram velocity is not significantly affected but the control rod drop out velocity is reduced to a permissible limit.

The velocity limiter is in the form of two nearly mated conical elements that act as a large clearance piston inside the control rod guide tube.

The hydraulic drag forces on a control rod are proportional to approximately the square of the rod velocity and are negligible at normal rod withdrawal or rod insertion speeds. However, during the scram stroke the rod reaches high velocity, and the drag forces must be overcome by the drive mechanism.

To limit control rod velocity during dropout but not during scram, the velocity limiter is provided with a streamlined profile in the scram (upward) direction. Thus, when the control rod is scrammed, water flows over the smooth surface of the upper conical element into the annulus between the guide tube and limiter. In the dropout direction, however, water is trapped by the lower conical element and discharged through the annulus between the two conical sections. A severe turbulence is created, and thereby slows the descent of the control rod assembly to about 5 ft/second.

CONCLUSIONS

The BWR is designed with the purpose in mind of preventing accidents. Thus, there are inherent BWR safety features, such as the negative steam void coefficient and Doppler effect. There are the multiple barriers of fuel cladding, reactor pressure vessel, and containment. There are redundant features and systems, a high level of quality assurance, in-service inspections, and testability.

In addition, there are the Engineered Safety Features provided to mitigate the consequences of postulated, serious accidents, that have been presented here. Note that these Engineered Safety Features represent in all cases, at least a third line of defence against a hypothetical accident. Also note that we have applied these features only after extensive development and proof testing to support their design and licensing. It is this approach that has yielded the excellent safety record to date of the BWR.

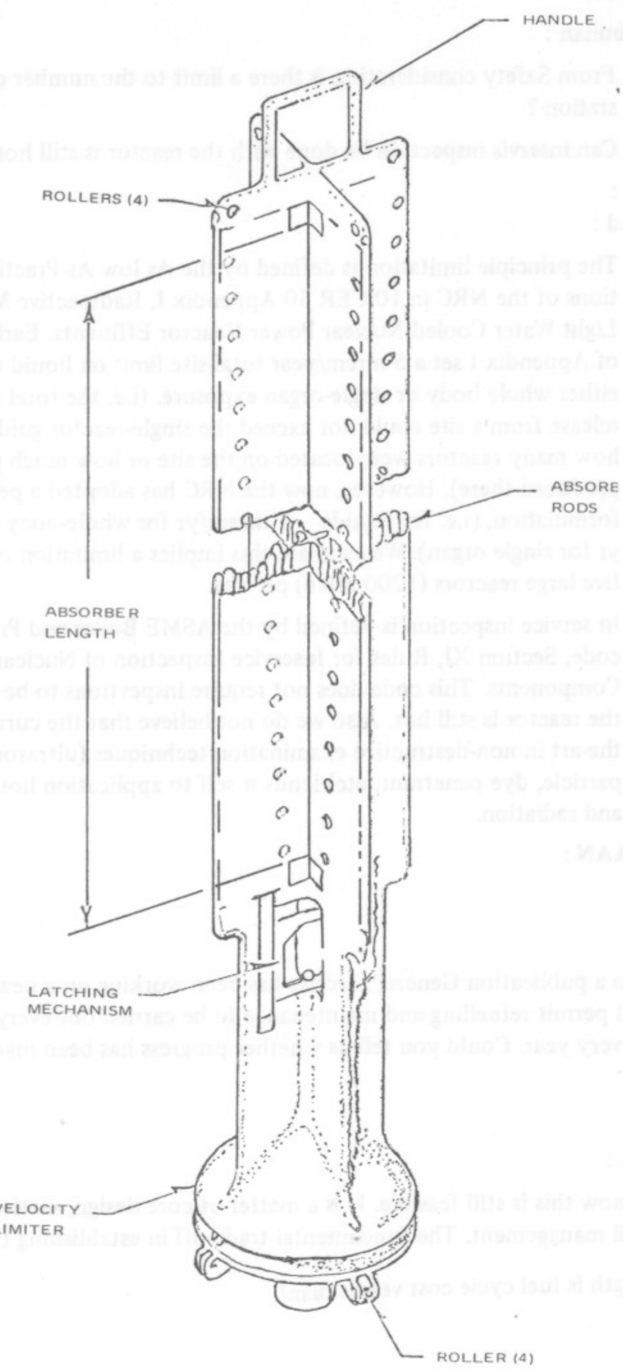


Figure 8 Typical Control Rod

DISKUSI

PERTANYAAN

Dr. F. Tambunan :

1. From Safety consideration is there a limit to the number of reactors per station ?
2. Can inservis inspection be done with the reactor is still hot ?

JAWABAN :
Manual Head :

1. The principle limitation is defined by the As low As Practicable regulations of the NRC in 10 CER 50 Appendix I, Radioactive Material in Light Water Cooled Nuclear Power Reactor Effluents. Earlier versions of Appendix I set a 5 mrem/year total-site limit on liquid release for either whole body or single-organ exposure. (i.e. the total radioactive release from a site could not exceed the single-reactor guides no matter how many reactors were located on the site or how much power was produced there). However, now the NRC has adopted a per-reactor formulation, (i.e. for liquids – 3 mrem/yr for whole-body and 10 mrem/yr for single organ). We estimate this implies a limitation of four or five large reactors (1200 MWe) per site.
2. In service inspection is defined by the ASME Boiler and Pressure Vessel code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components. This code does not require inspevtions to be done while the reactor is still hot. Also we do not believe that the current state-of-the-art in non-destructive examination techniques (ultrasonic, magnetic particle, dye penetrant, etc) lends it self to application hot-temperature and radiation.

PERTANYAAN :

Jasif Iljas :

Question :

According to a publication Generel Electric has been working on a new NSS–system, which would permit refuelling and maintenance to be carried out every 18 months instead of every year. Could you tell us whether progress has been made on this new design ?

ANSWER :

Mr. M. Head :

As far as I know this is still feasible. It is a matter of core design, method of operating, and fuel management. The fundamental trade-off in establishing the optimum fuel cyle length is fuel cycle cost versus plant

ANSWER :

Mr. M. Head :

As far as I know this is still feasible. It is a matter of core design, method of operating, and fuel management. The fundamental trade-off in establishing the optimum fuel cycle length is fuel cycle cost versus plant availability. Shorter fuel cycles decrease fuel cost but also decrease plant availability because of the more frequent hsutdowns.

PERTANYAAN :

Ir. Martias Nurdin

1. In the case of main steam break, what is the effort to minimize the fission product release (when the pipe is out side the containment).

2. Design pressure of containment system of BWR, I think is not enough to confine all the vapor formed from break of main coolant (steam) pipe.

   a. $$\frac{\textbf{excess energy per lb x total lbs inventory}}{\textbf{latent heat of vaporization per lb}} = \textbf{lbs of vapor}$$

   b. specific volume of vapor at design pressure of containment will give the volume of containment.

   c. My Question is : Is the volume of BWR's containment enough for confining the vapor formed ?

JAWABAN :

B.W.R. Presentation :

1. The fission product release is minimized by the main steam line flow restrictor and the fast closing main steam line isolation valves.

2. The design pressure of the MARK III containment is 15 psig at 185°F. Yes the containment has sufficient volume to contain the coolant mass and energy released by loose of coolant accident. This capability has been confirmed by extensive tests. Your calculation does not include the energy absorbed by the suppresion water.