NSP

Northern States Power Company

55

414 Nicollet Mall
Minneapolis, Minnesota 55401
Telephone (612) 337-2001

Charles E. Larson
Vice President
Nuclear Gener

July 28, 1989

Chairman Kenneth Carr
U.S. Nuclear Regulatory Commission
One White Flint North
Washington, DC 20555

Dear Chairman Carr:

ABB Atom of Sweden has submitted preliminary safety information to the NRC, requesting a review of the PIUS reactor system.

The PIUS concept has been reviewed with us by ABB representatives. I believe that the PIUS system offers a passively safe design which could be of interest to Northern States Power Company and other generating utilities when the next generation of nuclear plant designs is evaluated as a viable way to meet capacity needs.

Sincerely yours,

C E Larson

C E Larson
Vice President
Nuclear Generation

mak
c: Victor Stello

7/31...To EDO for Appropriate Action...Copies to RF...89-0694

ENCLOSURE II

PIUS Design and Verification

PIUS design description

The PIUS reactor design is based on well established LWR technology and infrastructure. It has a rearranged reactor primary system configuration compared to current LWR designs. The basic arrangement of the PIUS primary system is shown in Figures 1 through 6, and main data are provided in Table I.

The reactor core is an open, PWR type core with 213 fuel assemblies with standard PWR fuel rod diameter and a reduced height. The 2000 MWt core is located at the bottom of the reactor pool, containing a highly borated water mass enclosed by a prestressed concrete reactor vessel (PCRV). The PIUS reactor core does not use control rods, neither for reactor shutdown, nor for power shaping. Reactivity control is accomplished by means of reactor coolant boron concentration and temperature.

The core design parameters are significantly relaxed compared to current PWR practice in terms of average linear heat load, temperatures, flow rates and associated pressure drops. Reactivity compensation for burnup is accomplished by means of a burnable absorber (gadolinium) in selected fuel rods, while the moderator temperature reactivity coefficient is strongly negative throughout the operating cycle.

From the core the heated coolant - at a temperature of 290 deg C (554 F), passes up through the riser pipe, and leaves the reactor vessel through nozzles on the sides of the upper plenum. The coolant continues in the hot leg coolant pipes to the four straight tube once-through steam generators mounted beside the PCRV. The steam generators are similar to the OTSGs in a number of NPP in the US and FRG. The main coolant pumps are located below and structurally integrated with the steam generators. The pumps are sized-up versions of the glandless, wet motor design pumps that have been successfully utilized as recirculation pumps in the ABB Atom BWR plants.

The cold leg piping enters the reactor vessel through nozzles in the upper plenum at the same level as the hot leg nozzles, and the 260 deg C (500 F) return flow is directed downwards to the reactor core inlet via a downcomer. On its way down the flow is accelerated, and there are open connections between the downcomer and the pressurizer providing a siphon breaker arrangement. The siphon breaker prevents sipho-

ning off the reactor pool water inventory in the hypothetical event of a cold leg rupture. During normal operation, the siphon breaker does not affect the water circulation. There are also some open connections between the downcomer and the riser. At the bottom of the downcomer the return flow enters the reactor core inlet plenum.

A one meter (3 ft) diameter pipe that is open to the enclosing reactor pool, is located below the core inlet plenum. A tube bundle arrangement inside this pipe minimizes water mixing and ensures stable layering of hot reactor loop water on top of the colder reactor pool water. This pipe, with the bundle arrangement and the stratified water, is called the lower "density lock" and is one of the special components required to implement the PIUS principle. The position of the interface between hot and cold water is determined by temperature measurements, and this information is used for controlling the speed, and hence the flow rate, of the main coolant pumps to maintain the interface level at a constant position during normal operation. There is another "density lock" arrangement at a high location in the pool, connected to the upper riser plenum. This upper density lock has a similar arrangement of tube bundles and a number of small openings between the riser and the density lock.

This reactor system configuration with the two continuously open density locks connected to the high boron content pool, provides the basis for the PIUS principle. There is always an open natural circulation path from the pool through the lower density lock to the core via inlet pipes, through the core itself, up the riser, through the passage from the upper riser plenum to the upper density lock and back to the pool. During normal plant operation this natural circulation circuit is kept inactive by controlling the speed of the main coolant pumps, maintaining the hot/cold interface in the lower density lock and, in combination with primary loop water volume control, maintaining the hot/cold interface level in the upper density lock. The temperature measurements for the interface level in the upper lock are used for primary loop volume control purposes. The coolant flow rate is determined by the thermal conditions at the reactor core outlet relative to the reactor pool. The resulting pressure drop across the core and up through the riser must correspond to the static pressure difference between the interface levels in the upper and lower density locks. The main coolant pumps are operated to establish a pressure balance in the density locks during normal steady-state and load following operations. A sudden collapse in this pressure balance, as would occur during a severe transient or accident, would result in natural circulation of borated pool water through the core, providing both reactor shutdown and continued core cooling. The upper portion of both density locks, i.e. the volume above the hot/cold water interface, is normally filled with hot primary loop water, serving as a buffer volume to prevent ingress of pool water and spurious reactor shutdowns during minor operational disturbances.

The hot parts of the primary system are isolated from the cold reactor pool water by means of a wet thermal insulation of metallic type. This insulation consists of a number of thin, parallel stainless steel sheets with stagnant water between them.

The borated water in the reactor pool is cooled by two systems; one with forced circulation of pool water through heat exchangers and pumps outside the reactor vessel, and one passive system utilizing coolers in the reactor pool and natural circulation loops up to dry, natural draft cooling towers located on the top of the reactor building. The natural circulation system ensures the cooling of the reactor pool in accident situations and prevents boiling of the reactor pool water inventory, even with one circuit out of operation. If all pool cooling systems were to fail, the pool water ensures adequate core cooling for seven days.

The PCRV has a cavity with a diameter of about 12 m (40 ft) and a depth of about 38 m (125 ft), containing some 3,300 m^3 (870,000 gal.) of water. The concrete vessel monolith has a cross-section of about 27 m (89 ft) and a height of about 43 m (141 ft). It is anchored to the foundation mat structure by means of prestressing tendons. The pressure retaining capability of the vessel is ensured by a large number of prestressing tendons (partly horizontal tendons run around the cavity, partly vertical tendons run from the top to the bottom and by reinforcement bars).

The inside of the cavity is provided with a stainless steel liner. In addition, there is a second barrier - an embedded steel membrane about 1 m (3 ft) into the concrete - up to a level above the upper density lock to ensure that the reactor pool water volume below this level cannot be lost by liner leakage. Concrete vessel penetrations are not permitted below this level.

There is a steel vessel extension on top of the prestressed concrete vessel which is fixed by means of separate tendons anchored to the bottom of the concrete vessel. This extension contains the pipe nozzles for the hot and cold leg pipes, the forced circulation loops of the reactor pool cooling system, and some other system pipes. It also encloses the upper riser plenum, and the pressurizer.

The reactor is pressurized by means of steam supplied from an electrically heated recirculation boiler, drawing water from the pressurizer water volume. The steam volume of the pressurizer is comparatively large, and together with its volume of saturated water the reactor system can accommodate pressure and level variations that may occur during operational transients and accident situations. The pressu-

rizer is connected to the primary loop via funnels up into the steam volume, and to the pool via open passages from the pressurizer "pool".

The four steam generators are located on two sides of the concrete vessel. The two other sides are utilized for installation of equipment associated with supporting systems, the containment HVAC systems, etc.

The PCRV and the reactor system are enclosed in a large containment structure of pressure suppression type. Blowdown pipes lead from the drywell into a large condensation pool in the wetwell. All equipment containing reactor loop or reactor pool water at high pressure and high temperature is located inside the containment, which is designed to withstand a double-ended break of the largest pipe. The structure is constructed of reinforced concrete with a strength capable of resisting the impacting of an aircraft crash. The whole containment is provided with a steel liner in order to ensure leak-tightness. A steel dome closes the shaft above the reactor vessel.

During refueling operations the containment dome and the reactor vessel head are removed, and the cavity above the dome is filled with water. The reactor internals are lifted out in sections, and placed in a service pool beside the waterfilled cavity. The refueling is carried out with a conventional refueling machine from the reactor service room. Fresh fuel is brought into the cavity from a fresh fuel storage in the reactor building, and the spent fuel is removed to an adjacent spent fuel pool at the reactor service room floor level.

The steam lines from the steam generators and the feedwater lines to them are provided with isolation valves inside and outside the containment wall. The outer valves are located in a separate protected compartment. The pressure relief valves on the steam lines blow down to the condensation pool inside the containment, as do the pressure relief valves of the reactor pressure vessel.

The turbine plant is a conventional turbine plant. The 4.0 MPa (588 psi), 270 deg C (518 F) steam requires a somewhat larger size turbine than other modern LWR plants. The nominal power output of the turbine unit will be 635-665 MWe depending on the site conditions.

The reactor power is controlled by the boron content and temperature of the reactor coolant. During normal plant operation the reactor power can be controlled without adjustment of the boron content in the reactor coolant, by utilizing the strongly negative moderator temperature reactivity coefficient. A power change is accomplished by simply adjusting the rate of feedwater flow to the steam generators. An increase in

feedwater flow rate results in a reduced temperature of the return primary coolant flow to the reactor, a lowered average moderator water temperature and thus an increase in reactor power. This procedure can be applied over a 40% power range with a 20%/min rate of change in plant power. Beyond this range, adjustment of the boron content is used in order to keep the reactor core coolant outlet temperature within acceptable limits. The boron content is controlled by injecting distilled water (for power increase) or high boron content water (for power decrease) and withdrawing a compensating amount of water, corresponding to the procedures in current PWR plants. Daily load following operation between 100 percent and 50 percent can be accomplished with only a minor adjustment of the boron concentration at the start of the first day's cycle. For subsequent days no additional adjustment is needed.

The reactor core is physically protected by the encompassing containment structure and the thick-walled, strong PCRV walls. Protection against overheating and fuel damage is provided by the two loop PIUS arrangement with the core submerged in a large pool of borated water and transition to reactor shutdown and core cooling in a natural circulation mode without reliance on equipment for detection of off-normal conditions, initiation of actions, actuation of equipment, nor equipment relying on the displacement of mechanical bodies.

The PIUS plant is also provided with instrumentation systems, protection, logic, and actuation systems for reactor shutdown, residual heat removal, containment isolation, etc. in a reduced but in a similar (but reduced) way as current LWR plants. Their importance for ensuring safety is significantly reduced, however. The components of these instrumentation, monitoring, protection, and actuation systems are separated from those of other systems and located in separate, physically well protected compartments at the bottom of the reactor building. The reactor protection system - with a two-out-of-four coincidence logic - has the function of initiating power level reduction, reactor shutdown or reactor "scram" when reactor process parameters exceed set point limits.

In most cases a runback to a lower power level by means of the feedwater flow control, a further reduction to hot standby, or to hot shutdown conditions by injecting high boron content water into the primary system coolant, will be an adequate countermeasure. A reactor "scram" is initated only in a few transient situations by tripping one of the main coolant pumps. Highly borated pool water will then enter the primary system and shut down the reactor to hot, subcritical conditions. The primary loop structures will be subjected to a rapid cool - down (by some 50-60K (90-100 F)). This transient has no critical effect in terms of thermal fatigue of the components.

Compared with existing LWR designs a number of safety-grade systems have been eliminated. The control rods and the safety injection boron system are replaced by the density locks. The traditional automatic depressurization system is not required for PIUS. The auxiliary feedwater supply system for RHR is replaced by the reactor pool, while the containment heat removal and containment spray systems are replaced by the passive cooling of the reactor pool in PIUS. The safety-grade closed cooling water system, and a.c. power supply systems have been replaced by non-safety-grade systems in the PIUS design, allowing major simplification. The remaining safety-grade functions in PIUS are performed by: the reactor protection system tripping one coolant pump to achieve a reactor scram; the containment isolation system isolating the containment by closing isolation valves; the reactor vessel safety valves activated by pressure differentials; and the passive reactor pool cooling function. These functions are not needed for the short term protection of the core, however.

As a result, the plant should be simpler to operate and maintain than current LWR plants, and the elimination of severe reactor accidents as a practical concern should also contribute to simplified operation.

Transient analysis

The behavior of PIUS in severe transients and accident situations has been analyzed extensively over the years, both by experiments, and by computer simulations. For the latter purpose a very efficient computer code called RIGEL has been developed for simulation of the dynamic behavior of PIUS. The capability of the code to simulate the reactor and plant behavior with sufficient accuracy has been checked, by performing experiments in a large scale, high temperature and high pressure test rig and predicting the outcome by calculations.

A large number of transients and accident situations have been analyzed for the PIUS plant and the outcome of the simulations has always been a reactor shutdown or continued operation at a safe, limited power level. No accident sequence leading to a core uncovering and DNB (Departure from Nucleate Boiling) has been identified.

Four typical accident situations are discussed below to demonstrate the PIUS plant behavior: a loss of heat sink by loss of feedwater supply, a double-ended cold leg rupture, a reactivity insertion incident by coolant subcooling due to a steam line break, and a reactivity insertion accident by uncontrolled injection of distilled water. In all cases without scram activation.

In a loss of feedwater supply event, when the reactor scram function is postulated to fail and no other safety system is activated, the average fuel temperature increases marginally (by about 13 F) for a short period of time as the secondary side water inventory in the steam generators boils off.

In the event of a postulated double-ended cold leg rupture there will be a large outflow of hot primary loop water - through the ruptured cold leg and through the hot leg and the steam generator. The water outflow results in a flow deficit at the reactor core inlet, and reactor pool water is drawn in, shutting the reactor down. When the water level drops below the hot leg nozzles, the hot leg outflow stops. The pressure equilibrium between the pressurizer and the containment then stops the cold leg outflow by the siphon breaker arrangement.

The reactor pool water level remains well above the upper density lock and the reactor pool loop cools the core in natural circulation. From the pool the residual heat is transferred to the outer sink by forced circulation cooling, if electric power supplies are available or by natural circulation cooling, if they have failed. The average fuel temperature decreases from about 470 deg C (850 F) to about 265 deg C (510 F) during the first 20 seconds of the transient.

In a postulated "worst" case of a large double-ended steam line break close to one of the steam generators without scram or other protective action, the water inventory in the steam generators boils off rapidly, resulting in a cooling down of the return flow to the reactor core. The reactor fission power increases and reaches a maximum value of 2900 MW. The average fuel temperature increases and reaches a maximum of 517 deg C (963 F), corresponding to a maximum fuel temperature of about 800 deg C (1470 F). The thermal margins in the core are large, and thermal-hydraulic calculations have shown that the DNB margin remains high during the entire transient.

As a boundary event, an uncontrolled injection of distilled water into the primary loop during normal plant operation represents a serious reactivity insertion transient. Assuming failure of scram and other protective functions, with an injection of distilled water at the rate that is possible, the calculations show that the reactor power will oscillate between 1900 and 2200 MW. The average fuel temperature will oscillate below a maximum value of 502 dec C (936 F) and the coolant temperatures at a maximum of 300 deg C (588 F).

In terms of requirements on the containment, there are some very significant differences between a PIUS plant and current LWR plants. In the PIUS plant, the integrity of the nuclear fuel is protected by inherent, self-protective passive functions, and there

will be no core uncovery or fuel overheating following any credible accident. The radioactive matter released to the containment following pipe breaks will arise from the possible "leakers" (leaking fuel rods) in the core, prior to the accident; the accidents will not cause additional fuel damage. Only part of the hot primary loop water inventory will be released to the containment during the depressurization of the reactor.

Following the initial depressurization of the reactor system, the core will be cooled by the natural circulation reactor pool loop, transferring the reactor residual heat to the reactor pool. The natural circulation pool cooling system always ensures residual heat removal to the air via dry natural draft cooling towers without boiling of the water inside the reactor vessel.

The short term release of steam and hot water during the initial blowdown of the reactor is absorbed in the condensation pool. The containment will be pressurized to 3 bar. There is no long term release of steam, and the pressure decreases to slightly above atmospheric pressure within 2-3 hours due to condensation of steam on structures and components. No safety-grade containment cooling systems are needed.

The releases of radioactive matter to the containment will be small, and as a result of the moderate, temporary pressurization of the containment, relative to the atmosphere, the maximum credible release rates to the environment will be extremely small.

The calculated doses at the site boundary are well below the lower level Protective Action Guidelines (PAGs) of the U.S. Environmental Protection Agency (EPA), which specify maximum 1 Rem whole-body and 5 Rem thyroid doses. In relation to the guidelines set out in SECY-88-203 this should provide the basis for easing the offsite emergency planning requirements.

Design verification

The demonstration of the self-protective thermohydraulics of the reactor system in normal operation and severe transient conditions including comparison with computer code predictions represent the largest experimental effort in the PIUS program. A complete PIUS type system with an electrically heated simulated fuel assembly was built, the ATLE test rig, and a number of transients were performed with it to verify the safety characteristics of the PIUS system. The observed results were compared to predictions of the RIGEL computer code to verify the code.

The density locks have been verified in a four-year program which is now completed. The practical applicability of the density lock design has been demonstrated. The turbulence-induced boric acid transport is acceptably low. A large scale test is planned to study the optimum detailed design of the density lock.

Theoretical analyses made of the performance of the siphon breakers have confirmed its adequate function. Large scale tests are planned to optimize the detailed design.

The wet thermal insulation developed for the French gas-cooled reactors was successfully tested and verified in the 1970s in the Scandinavian Concrete Pressure Vessel Program and could be directly employed. A simpler, lower cost and more rugged insulation system has been designed. Tests of this system are planned.

Very large prestressed concrete pressure vessels represent an established technology for gascooled reactors in UK and France.

The prestressed concrete reactor vessel for LWR use was thoroughly covered in the Scandinavian PCRV development program 1967-1976. Prestressed concrete vessels are also used extensively as containment vessels for LWR plants. ABB Atom has delivered two and specified a number of such vessels that are now in service in Scandinavian nuclear plants. Studies on stress distribution under various conditions for a PIUS PCRV have been performed including seismic analysis of the reactor internals.

Conceptual design of the passive long term RHR system shows that the system can remove the residual heat indefinitely even under serious conditions like a primary pipe break. Tests of the system are planned to verify the detailed design.

Control rods are not used since their erroneous operation may lead to major potential fuel damage. The reactivity control of PIUS is based on the use of burnable absorbers in the fuel and boron in the primary coolant. The pool water boron content is ensured by continuous monitoring by redundant and diversified methods. Loss of boric acid concentration by mistake is not credible due to the large amount of boron. No credible sabotage scenario exists for removal of the boron in the pool in a reasonably short time.

Summarizing, PIUS is a reconfigured PWR employing a number of BWR features. The new features are essentially the pool loop arrangement with "density locks"

which provides the thermohydraulic self-protective features, and the prestressed concrete vessel.

It is ABB Atom's opinion that the few exceptions from proven LWR technology are verified by the simulation tests and that these, together with the verification program to be performed preceeding the lead plant, constitute a sufficient basis for constructing a full scale lead plant with full NSSS vendor warranties.

Demonstration of plant performance under transient conditions that are normally considered severe can be performed on the lead plant without harm to it. This could constitute the final proof for acceptability.

The risk of failure in these tests is virtually nonexistent and should not constitute a deterrent to commitment for further plants without awaiting results from the tests.

Table I: PIUS 600 Main Data

Core thermal power	2,000 MW
Net electric power	640 MWe
Circulating water temperature	15°C (60 F)
No. of fuel assemblies	213
Core height (active)	2.50 m (8.2 ft)
Core equivalent diameter	3.76 m (12.3 ft)
Average fuel linear heat rate	11.9 kW/m (3.9 kW/ft)
Average core power density	72.3 kW/l
Core inlet temperature	260°C (500 F)
Core outlet temperature (mixed mean)	290°C (554 F)
Operating pressure (pressurizer)	9 MPa (1280 psi)
Core coolant mass flow	13,000 kg/s (28630 lbm/s)
Average burnup	45,500 MWd/t
Equilibrium core ingoing enrichment (12 months)	3.5%
Concrete vessel cavity diameter	12 m (40 ft)
Concrete vessel cavity volume	3,300 m^3 (870,000 gal)
Concrete vessel total height	43 m (145 ft)
Concrete vessel thickness	7-10 m (23-33 ft)
No. of steam generators and coolant pumps	4

PIUS

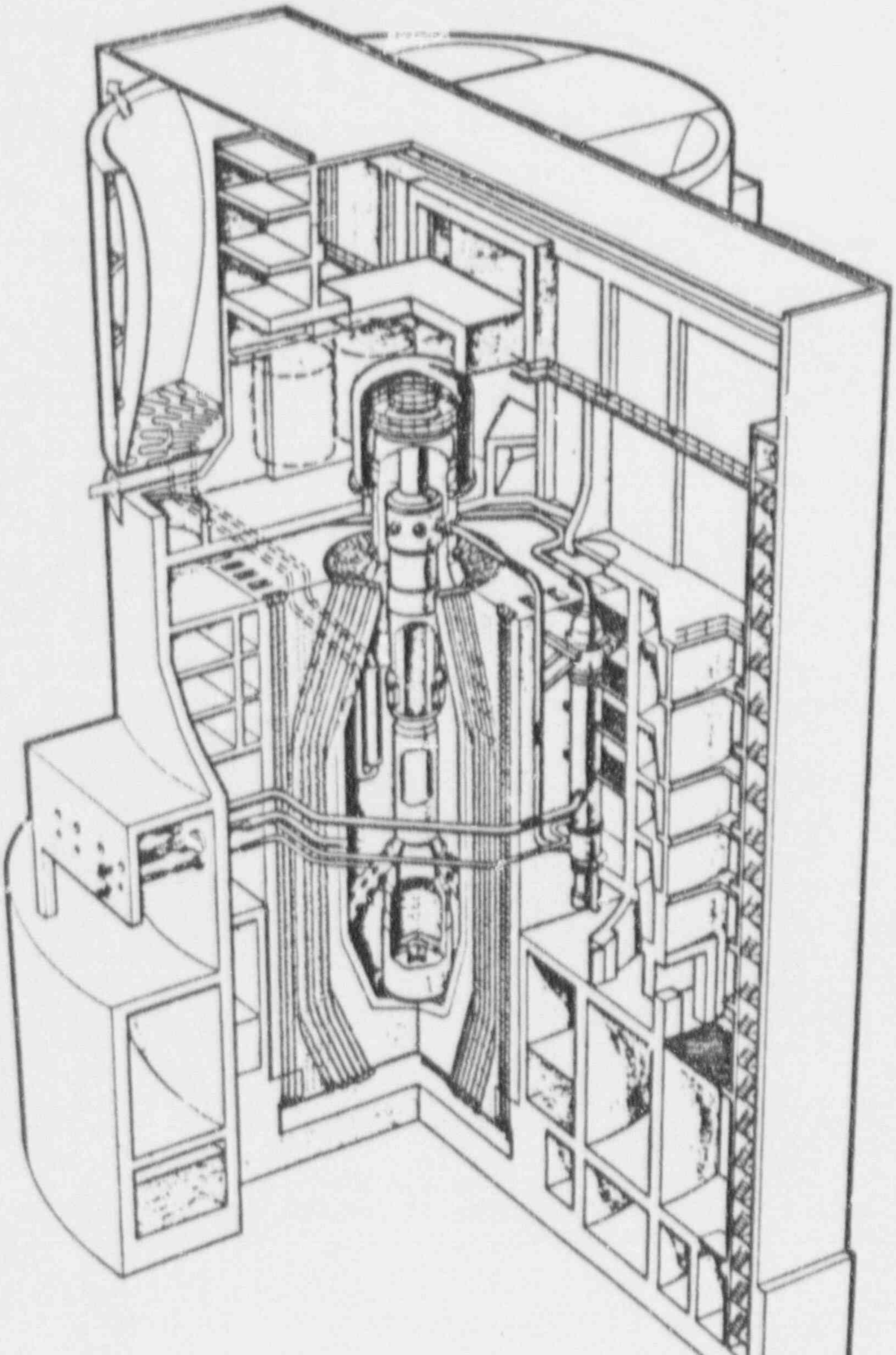


Figure 1

ABB Atom

ABB
ASEA BROWN BOVERI

PIUS

Main features of NSSS

1. Pressurizer steam volume
2. Steam generator (4)
3. Upper density lock
4. Main coolant pump (4)
5. Riser
6. Core instrumentation
7. Embedded steel membrane
8. Pool liner
9. Core
10. Lower density lock
11. Submerged pool cooler, cooled in natural circulation by ambient air.

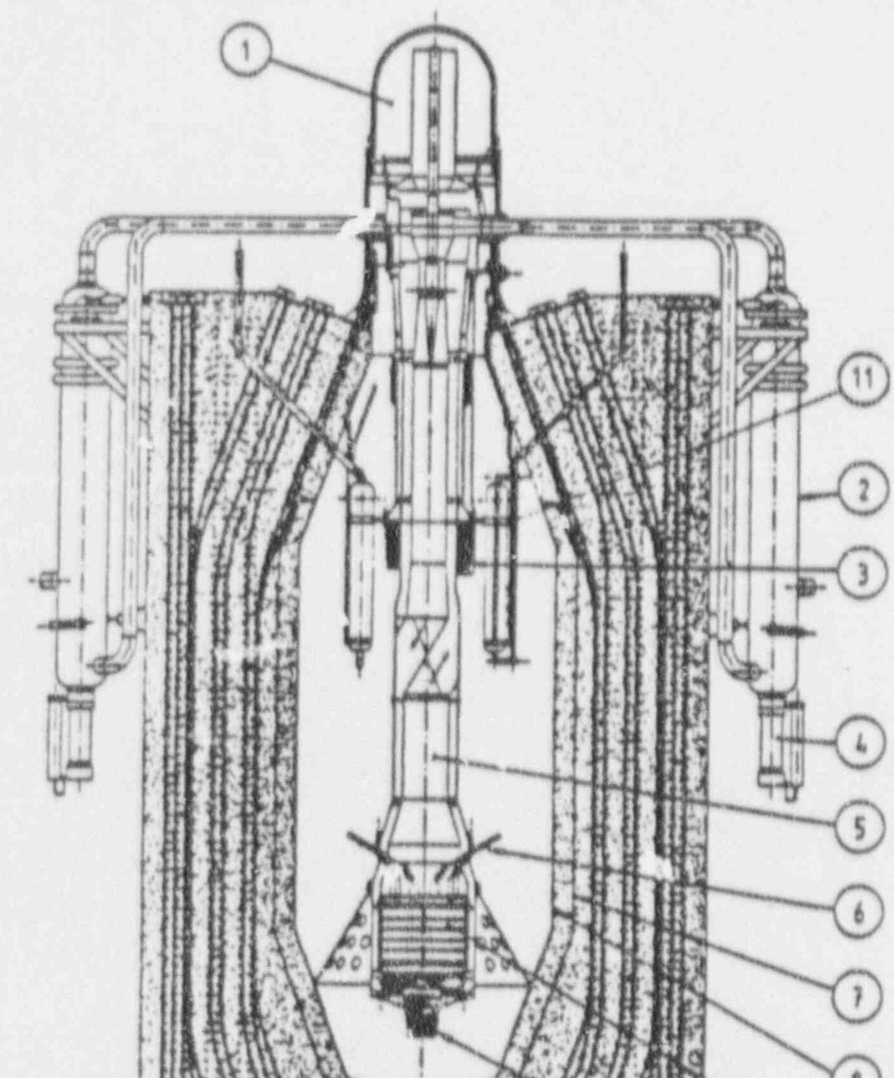


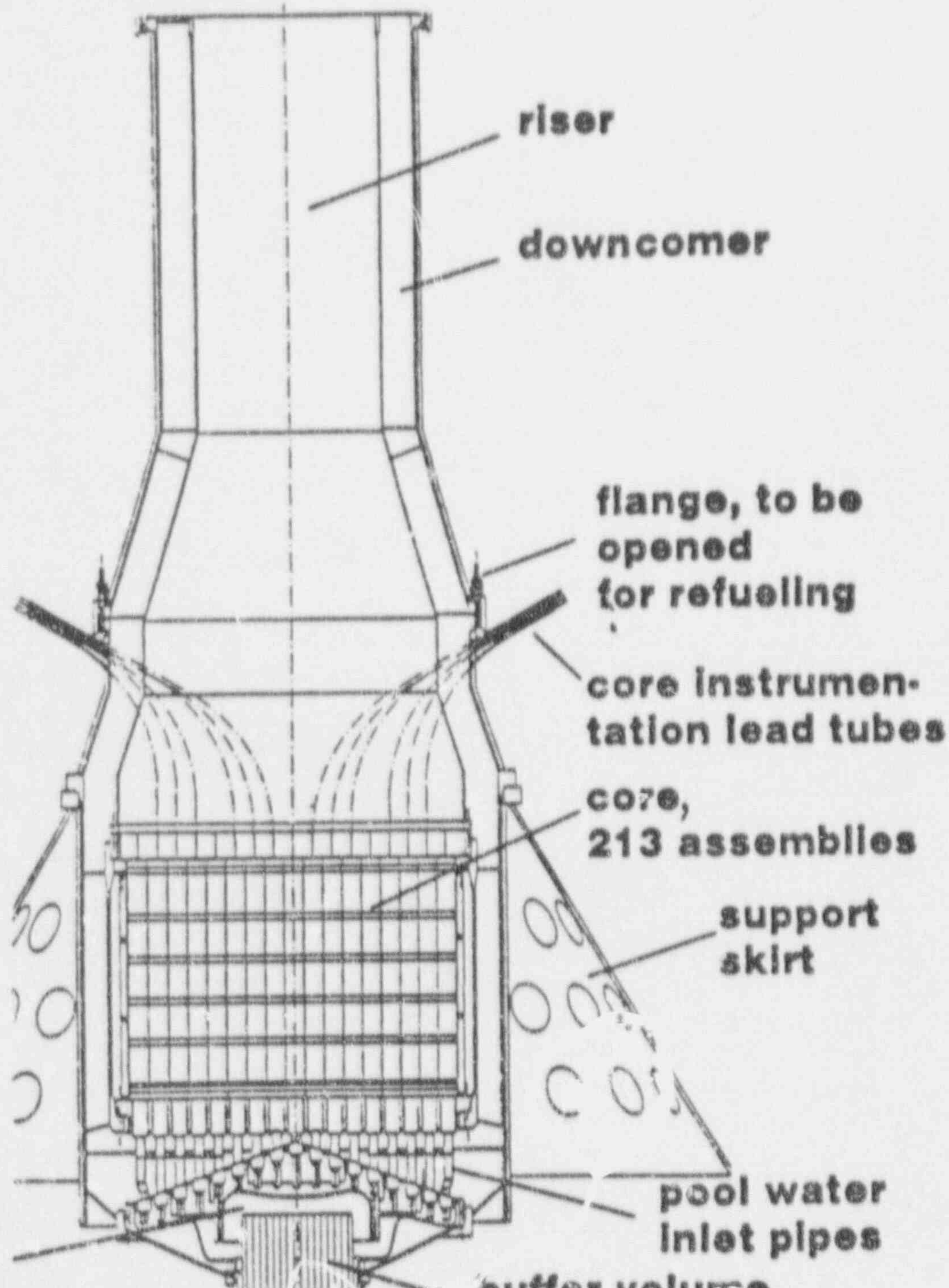
ABB Atom

Figure 2

ABB

PIUS

Lower part of vessel internals



PIUS

Upper part of vessel internals

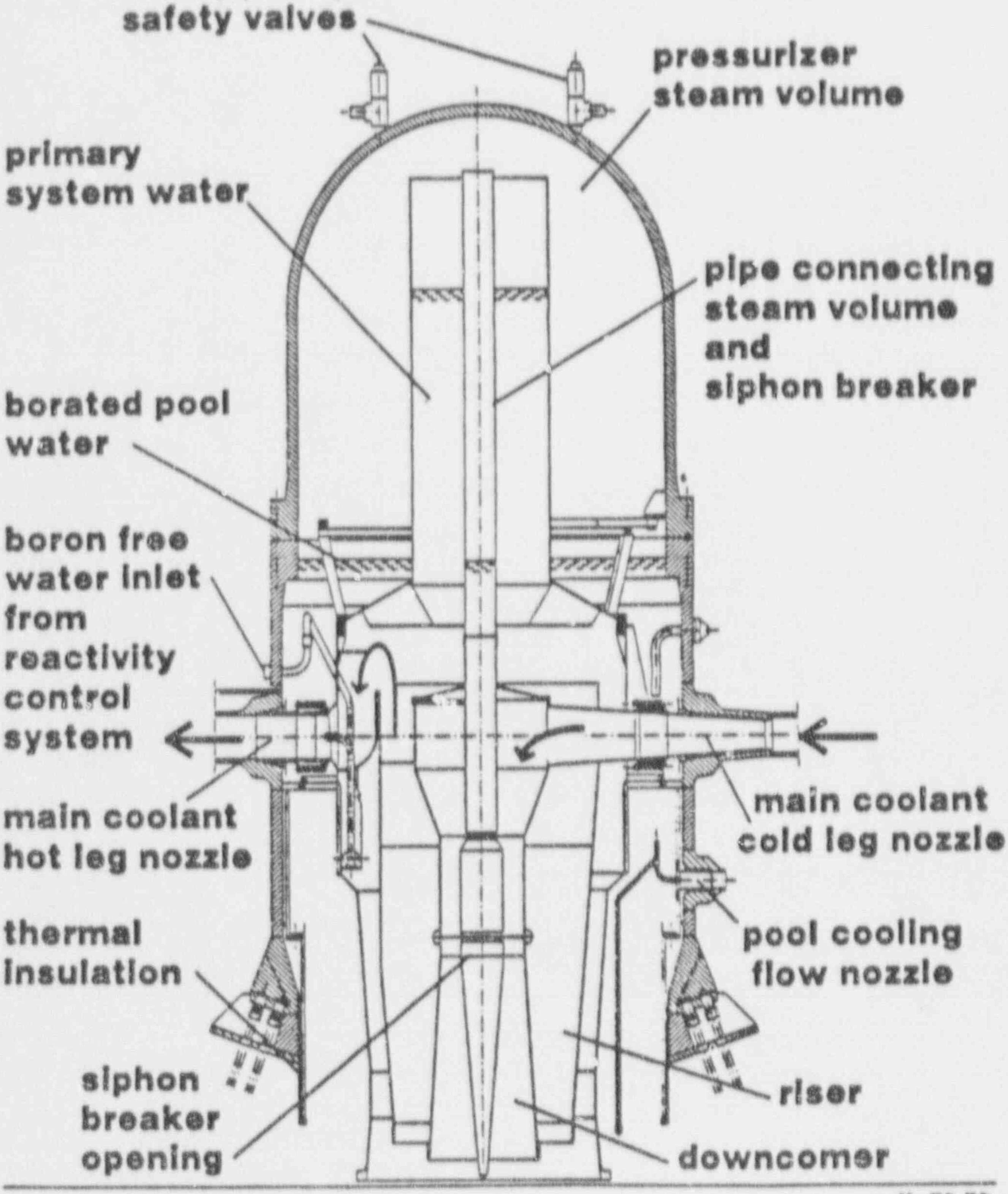


Figure 4

PIUS

Reactor building, vertical section

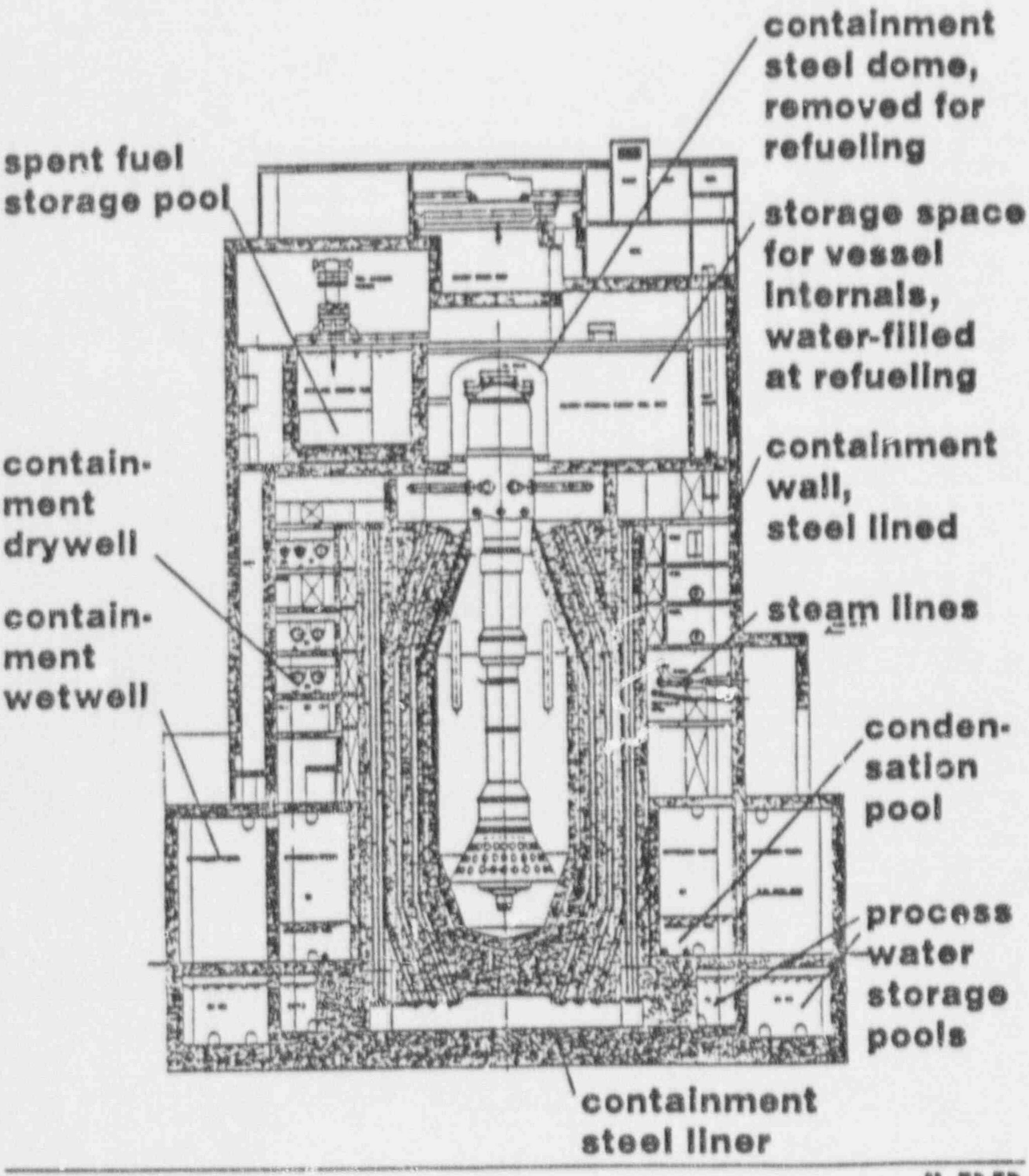


ABB Atom

Figure 5

ABB

PIUS

Reactor building, vertical section