ATOMIC ENERGY OF CANADA LIMITED
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Lecture 6 & 7

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HEAT TRANSPORT AND AUXILIARY SYSTEMS

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NUCLEAR POWER SYMPOSIUM

LECTURE NO. 6: HEAT TRANSPORT

by

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1. COOLANTS

1.1 General

Heat transport means specifically the movement of thermal energy from the fuel to the turbine. Heat is transported in many other places in a nuclear plant, but we use the term "heat transport" for the high temperature heat flow leading directly to electrical generation.

The duty that a heat transport system is supposed to perform is thus simply stated. The problems arise from the constraints put on design of the system.

1.2 Choice of Coolant

We have in Canada experience with three coolants: heavy water, ordinary water and HB-40, an organic substance similar in appearance to a light machine oil.

U. S. vendors use mainly ordinary water, although this year so far some six gas-cooled reactors have been sold in the U. S. by U. S. suppliers.

The U. K. have based their program on gas-cooling - either carbon dioxide or helium - but have built and operated reactors cooled by ordinary water and by liquid metal.

The coolant should capture relatively few neutrons in its passage through the reactor. Neutrons are expensive, and capture by coolant is a total waste, not just in loss of neutrons, but also in the radioactivity induced in the coolant, causing access problems to piping and boilers.

The most efficient liquid coolant from the standpoint of neutron economy is heavy water. The part of ordinary water that is hydrogen is deuterium in heavy water; and the deuterium atom, having a nucleus with an extra neutron, has a low affinity for another. The hydrogen in ordinary water, on the other hand, absorbs neutrons (to become deuterium) rather readily, and hence adversely affects neutron economy.

When deuterium does absorb a neutron it becomes tritium, which is a substance hazardous to health. Trace quantities are found in heavy water cooled reactors and to a lesser extent in ordinary water cooled reactors.

Ordinary water when used as a coolant requires special design of the reactor to produce an excess of neutrons to carry on the reaction. If the fuel is natural uranium - that is, not enriched in the fissile isotope - then the coolant has to boil to reduce its density and hence its propensity to capture neutrons. The Gentilly reactor in Quebec operates in this way. Such reactors are started up by inserting booster fuel enriched in fissile material until bubbles are produced in the coolant, and then the enriched fuel is withdrawn.

The other way of using ordinary water is to enrich the main fuel in fissile material. If this is done, the coolant need not boil, and the plant is a PWR (Pressurized Water Reactor). Or, alternatively, it may be boiled, and then the plant is a BWR (Boiling Water Reactor). Boiling water reactors have special nuclear control problems, as one might expect, because the amount of vapour in the core affects the power produced. This is generally solved by varying the amount of liquid water accompanying the vapour, and this is achieved by forced recirculation of the liquid using pumps variable in speed or delivering through control valves.

Reactors in which the coolant is boiled are almost always designed for direct supply to the turbine of the vapour generated in the reactor. This is very attractive; there is no heat exchanger in the path of heat flow, and not only does one save the cost of the heat exchanger but one also saves the inevitable temperature loss in it, and so the thermal efficiency is higher. It is so attractive, in fact, that one of the main U.S. vendors (G.E.) have concentrated their main effort on this type; we have built a plant (Gentilly) operating in this way, and one of our major study efforts is aimed at evaluating this type; the U.K. have built and operated one and are evaluating the type as a main contender to span the gap between the present and the advent of the large fast breeder reactors. However, as always, the advantages of boiling come inseparably wed to disadvantages. For one thing, the turbine and associated parts may become radioactive, because they are exposed to steam generated in the reactor. For another, the reactor is exposed to feedwater, which is more difficult to keep in good shape chemically than are the contents of a closed coolant circuit. And for a third, the fuel in the reactor has a generally less favourable environment, because the "interface" between all-liquid coolant and two-phase coolant - that is, the point at which boiling starts - tends to move up and down somewhat, exposing the fuel to variations in heat transfer and possibly to adverse chemical effects.

Organic coolants such as HB-40 have major advantages. The vapour pressure at any particular temperature near desirable operating values is much less than that of any form of water. Typically, heavy water at 570°F is under a pressure of 1242 psia, whereas HB-40 at 570°F is under a pressure of about 30 psia. This means that vessels may be thinner and leakage is more easily controlled. A second major advantage of organic coolant is that it is less corrosive than hot water. Hot water is not usually thought of as corrosive; but in fact it is. It dissolves the walls of pipes and vessels, not fast enough to be a hazard to the structure but fast enough to produce metallic impurities in the coolant. These impurities are, of course, carried through the reactor and are irradiated by neutrons and become radioactive. They then perversely become deposited on out-of-core components, producing radiation which restricts access. This is a major problem with every water-cooled reactor; and a negligible problem with organic coolant.

Organics, of course, present problems to balance these advantages. HB-40 is not as transparent to neutrons as heavy water, although it is better than ordinary water, and the neutron economy is less favourable than that of heavy water cooled reactors, even after credit is taken for the thinner parts allowed by the lower coolant pressure. HB-40 is inflammable, and a leak from a pipe at high temperature may produce an explosive mixture of hydrocarbon vapour - mainly benzine - in the area. Very highly reliable defences against explosion exist, and this hazard in no way rules out organics; but the precautions must be taken.

A further difficulty with organics arises from its very advantage of low vapour pressure. In the steam generator, the pressure on the steam side is greater than the pressure on the organic side, and a tube failure will produce a leak of water into the organic coolant, with very unpleasant results. Again, this difficulty is not insuperable, but perhaps it is indicative of the array of problems confronting the designer.

The situation with respect to choice of coolant in Canada today is this: a heavy water cooled CANDU reactor is not a prototype. Over 5000 MWe of plant of this type in Canada and about a further 1000 MWe abroad, is either operating or under construction. An organic cooled

CANDU reactor would have some of the aspects of a prototype. We have operated a research reactor with organic coolant, but it is not a power producer. An ordinary water cooled reactor of U.S. design need not have any prototypical features. These are proven systems. An ordinary water cooled reactor of Canadian design would have prototype aspects if it were larger than Gentilly (250 MWe) or if it employed enriched fuel. Gas cooled reactors are not a part of our program and we cannot comment on them.

The rest of this discussion will be devoted mainly to heavy water systems, with comparisons wherever possible to ordinary water systems.

2. ARRANGEMENT OF MAJOR COMPONENTS

Figure 1 shows an arrangement of the Pickering Reactor Building. The pressure tubes are horizontal, facilitating access to both ends by fuelling machines. The headers are above the reactor, providing a good place to put cooling water if the reactor should spring a leak. The boilers and pumps also are above the reactor, in an area where radiation levels after shutdown are low. Basically, the headers are above the reactor and the pumps and boilers are above the headers. This is true of all heavy water cooled power reactors since NPD (20 MWe).

In Pickering the boiler room as well as the reactor vaults have air atmospheres with the water vapour in the air partly or mostly heavy water. The vapour is recovered in dryers and upgraded as necessary. The Pickering plant is, in this respect, the last plant of its kind. All subsequent plants have or will have a separation between ordinary water carrying atmospheres and heavy water carrying atmospheres. This is achieved by placing most heavy water carrying equipment in rooms from which ordinary water is excluded. Such components as boilers, which carry both heavy and ordinary water, have the heavy water end in the heavy water room or rooms and the rest outside. The seal is made with welded steel bellows.

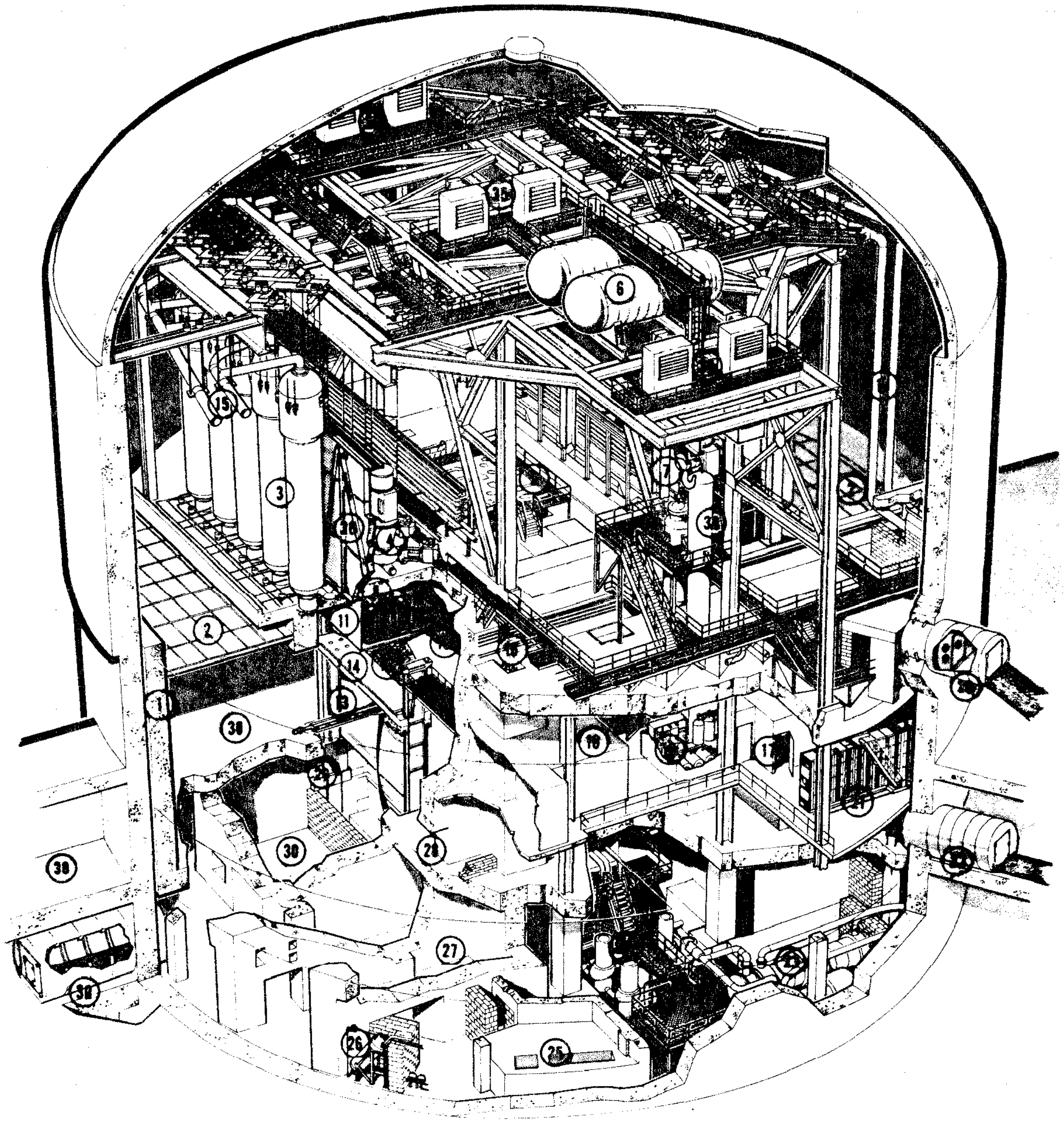
We see also in the Pickering illustration the dump tank below the reactor. This is a place to drop the heavy water moderator out of the reactor, effectively shutting it down. The moderator is held up by gas pressure, and on a signal to dump, the gas pressures below and above the moderator are equalized through large valves. The shutdown so achieved is what we call "ever-safe" - that is, the reactor cannot be brought to criticality by any manoeuvre until the moderator is pumped up again - but it is not fast enough to cope with many possible reactivity transients. For a fast shutdown, we depend upon gravity rods.

Pickering is, in respect to moderator dump, almost but not quite the last of a generation that ran through NPD and Douglas Point. I say almost because Gentilly has it. But currently we are using liquid or mechanical devices, for speed and for lower cost.

Figure 2 shows a U.S. pressurized water reactor and auxiliaries. The building here is what we call "total containment" - that is, it has no vacuum building or pressure suppression tank into which escaped coolant is released. We note that in this plant, as in most, the connections to the vessel are above the fuel. As with the headers in Pickering, a break in these pipes cannot prevent the core from being flooded. These plants are shut down once a year for refuelling, rather than being fuelled on load as our plants are. This plant produces steam at the same pressure as the 1200 MWe CANDU now in design for service in 1982.

Figure 3 shows the Gentilly reactor building. This plant has two steam drums, for storing coolant and for separating the steam formed in the core from water, but it has no boiler other than the reactor itself. The coolant is ordinary water. This is an attractive layout, easily accessible and compact.

Figure 4 shows a flowsheet for the heat transport system of a CANDU reactor.



1 PRESSURE WALLS
2 BLOWOUT PANELS
3 STEAM GENERATORS
4 PRIMARY HEAT TRANSPORT PUMPS
5 CONTROL AND SHUT-OFF RODS
6 FEED WATER RESERVE TANKS
7 BOILER ROOM CRANE
8 PRIMARY HEAT TRANSPORT REACTOR OUTLET HEADER
9 PRIMARY HEAT TRANSPORT REACTOR INLET HEADER
10 FEEDER PIPES
11 FEEDER INSULATION CABINET
12 REACTOR END FITTINGS
13 FUELLING MACHINE HEAD
14 FUELLING MACHINE BRIDGE
15 MAIN STEAM SUPPLY PIPES
16 PIPE CHASE
17 INSTRUMENTATION ROOM (WEST)
18 D_2O COLLECTION ROOM
19 ZONE CONTROL SYSTEM ROOM
20 BOILER ROOM AIRLOCK
21 REACTOR CONTROL DISTRIBUTION FRAME
22 MAIN EQUIPMENT AIRLOCK
23 MODERATOR HEAT EXCHANGERS
24 MODERATOR PUMPS
25 MODERATOR AND ION EXCHANGE COLUMNS
26 SPENT RESIN DRYING TANK
27 FUELLING MACHINE AUXILIARIES (EAST)
28 FUELLING MACHINE VAULT DOORWAY
29 FUEL TRANSFER PORT
30 FUELLING MACHINE SERVICE ROOM (EAST)
31 FUELLING MACHINE VAULT (EAST)
32 FUELLING MACHINE AIRLOCK
33 REACTOR AUXILIARIES BAY
34 BLEED CONDENSER AND BLEED COOLER
35 BOILER ROOM COOLING UNITS

Figure 1 Pickering Reactor Building - General Arrangement

1. WASTE PACKAGING AREA
2. AUXILIARY BUILDING
3. AUXILIARY BUILDING CRANE
4. NEW FUEL STORAGE AREA
5. RAILROAD DELIVERY AREA
6. WASTE HANDLING CRANE
7. SPENT FUEL PIT
8. FUEL CASK LOADING
9. HOIST FOR FUEL TRANSFER SYSTEM
10. FUEL TRANSFER CANAL - SPENT FUEL PIT GATE
11. FUEL TRANSFER CANAL VALVE
12. FUEL TRANSFER CONVEYOR UP-ENDING FRAME REACTOR 2
13. FUEL CANAL TO REACTOR 1
14. SPENT FUEL PIT BRIDGE AND HOIST
15. REACTOR BUILDING 2
16. ACCESS LADDER TO DOME
17. REACTOR BUILDING 1
18. CONTAINMENT SPRAY PIPES
19. CRANE COLLECTOR RAIL
20. STEEL CONTAINMENT VESSEL
21. ICE CONDENSER TOP DECK
22. REACTOR BUILDING POLAR CRANE
23. ICE CONDENSER SYSTEM BRIDGE CRANE
24. ICE CONDENSER SYSTEM AIR HANDLING UNITS
25. ICEBASKETS
26. ICE CONDENSER SYSTEM LOWER INLET DOORS
27. ICE CONDENSER SYSTEM FLOOR DRAINS
28. ICE MACHINES
29. ICE STORAGE BIN
30. BORAX SOLUTION MIXING TANKS
31. PACKING CHILLERS
32. CONTROL ROD DRIVE EQUIPMENT ROOM
33. EQUIPMENT HATCH - REACTOR BUILDING
34. PERSONNEL HATCH - REACTOR BUILDING
35. STEAM GENERATOR CONTAINMENT
36. MANIPULATOR CRANE
37. CONTROL ROD DRIVE MISSILE SHIELD
38. GATE TO REFUELLING CAVITY
39. STEAM GENERATORS (4)
40. MAIN STEAM PIPES
41. REACTOR COOLANT PUMPS (4)
42. PRESSURE VESSEL - UNIT 1
43. PRESSURIZER
44. PRESSURIZER RELIEF TANK
45. ACCUMULATORS (4)
46. REACTOR - STEAM GENERATOR MAIN COOLANT PIPING
47. PUMP - REACTOR MAIN COOLANT PIPING
48. STEAM GENERATOR-PUMP MAIN COOLANT PIPING
49. PRESSURIZER SURGE PIPE
50. FEEDWATER PIPES TO STEAM GENERATORS
51. VENTILATION FAN
52. ACCESS TO SUMP BENEATH REACTOR
53. RAW WATER TANKS
54. MAIN CONTROL ROOM
55. UNIT 1 CONTROL BOARDS
56. SHIFT ENGINEER'S OFFICE
57. KITCHEN AND LUNCH ROOM
58. 480 V SHUT-DOWN BOARD TRANSFORMERS
59. 480 V SHUT-DOWN BOARDS
60. AIR INTAKE HOUSING
61. FILTER UNITS
62. AUXILIARY BUILDING LIGHTING BOARD
63. MECHANICAL EQUIPMENT ROOM
64. HOLD-UP TANKS (2)
65. GAS DECAY TANKS
66. COMPONENT COOLING PUMPS
67. TURBINE BUILDING
68. FRESH AIR INTAKES
69. GLAND SEAL WATER TANK
70. PORTABLE WATER TANKS
71. TURBINE BUILDING CRANE - TURBINE 1
72. TURBINE BUILDING CRANE - TURBINE 2
73. H.P. TURBINE - UNIT 1
74. L.P. TURBINES - UNIT 1
75. GENERATOR - UNIT 1
76. TURBINE - UNIT 2
77. AUXILIARY BOILERS
78. REHEATERS - TURBINE 1
79. REHEATERS - TURBINE 2
80. HEATING AND VENTILATING EQUIPMENT
81. TURBINE OIL TANK
82. FEEDWATER CONTROL STATION - REACTOR 1
83. HEATERS - LOW PRESSURE
84. TURBINE BYPASS PIPES
85. FEEDWATER PUMP TURBINES
86. FEEDWATER PUMP TURBINE CONDENSER
87. CONDENSER
88. SERVICE BUILDING
89. SERVICE BUILDING LOADING DOCK
90. SWITCH YARD
91. HEATERS - HIGH PRESSURE
92. EXHAUST FAN HOUSING

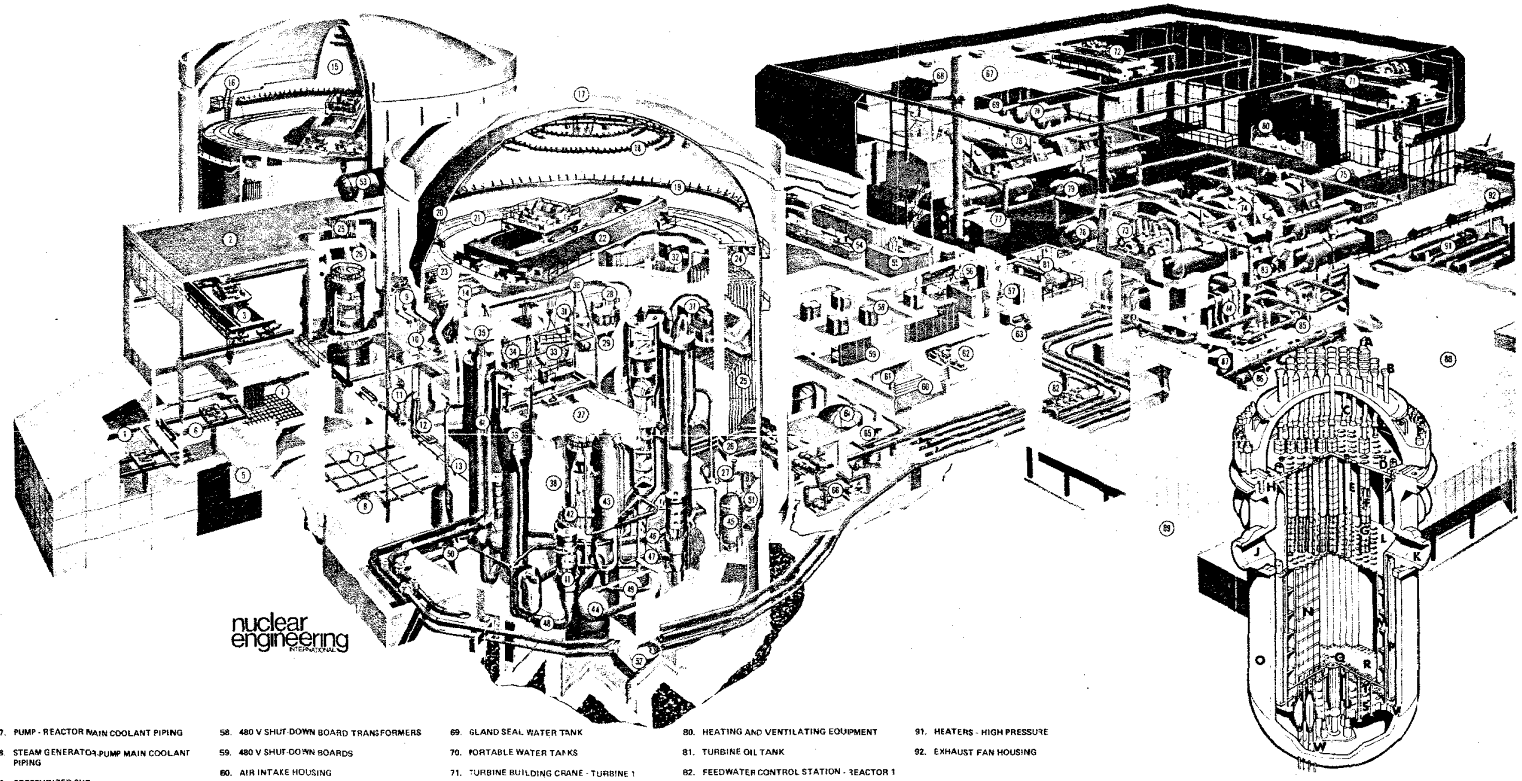
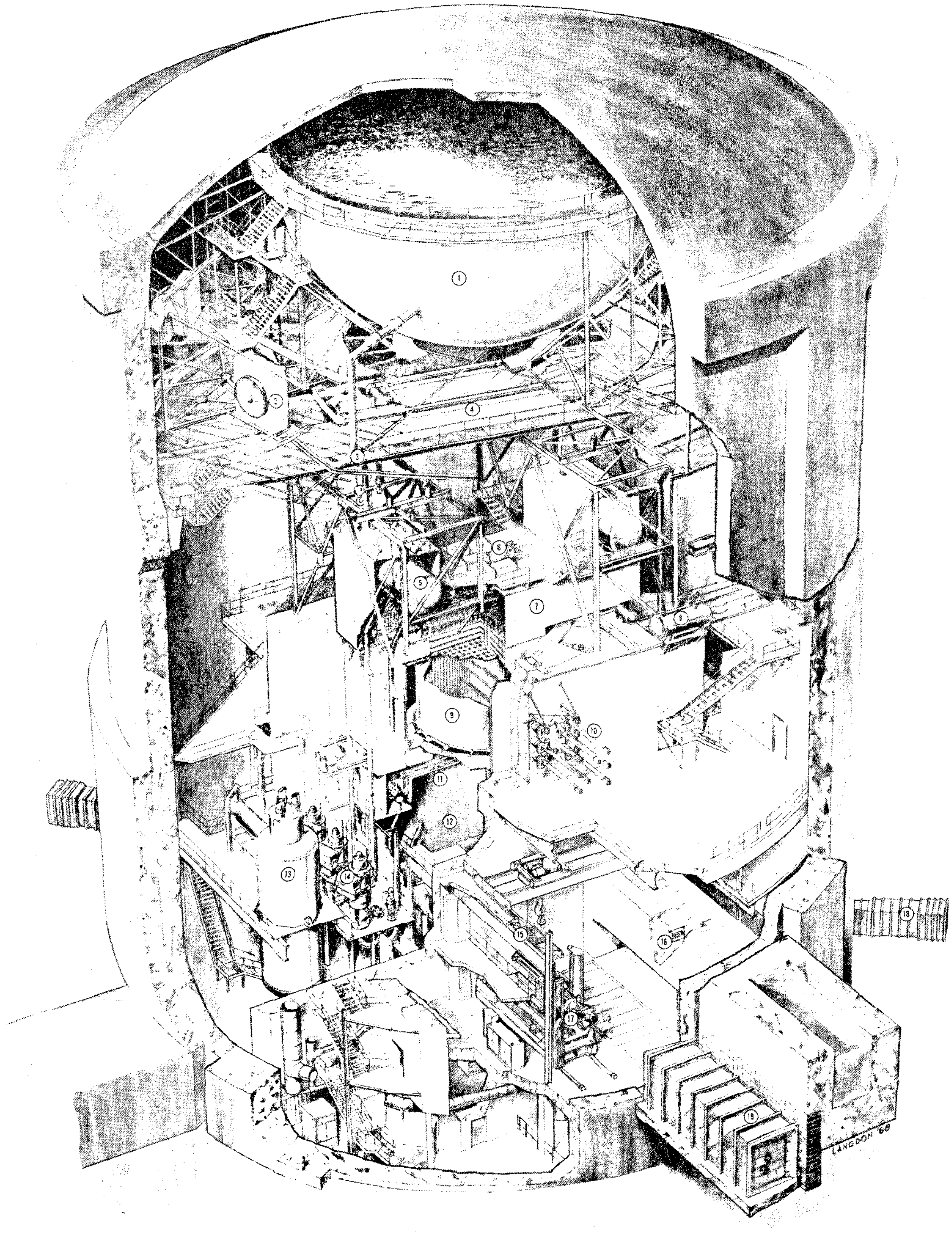


Figure 2 U.S. PWR - General Arrangement



1 DOUSING WATER TANK
2 BUILDING COOLERS
3 DOUSING PIPING
4 POLAR CRANE
5 STEAM DRUM
6 CONTROL ABSORBER MECHANISM
7 INSULATION CABINET
8 THERMAL SHIELD EXPANSION TANK
9 REACTOR
10 BOOSTER MECHANISM
11 FUEL CHANNEL END FITTINGS
12 FUELLING MACHINE VAULT
13 SUPPRESSION TANK
14 REACTOR COOLANT PUMPS
15 CATENARY
16 FUEL SHUFFLING TRENCH
17 FUELLING MACHINE
10 PERSONNEL AIRLOCK
19 EQUIPMENT AIRLOCK

Figure 3 Gentilly Reactor Building - General Arrangement

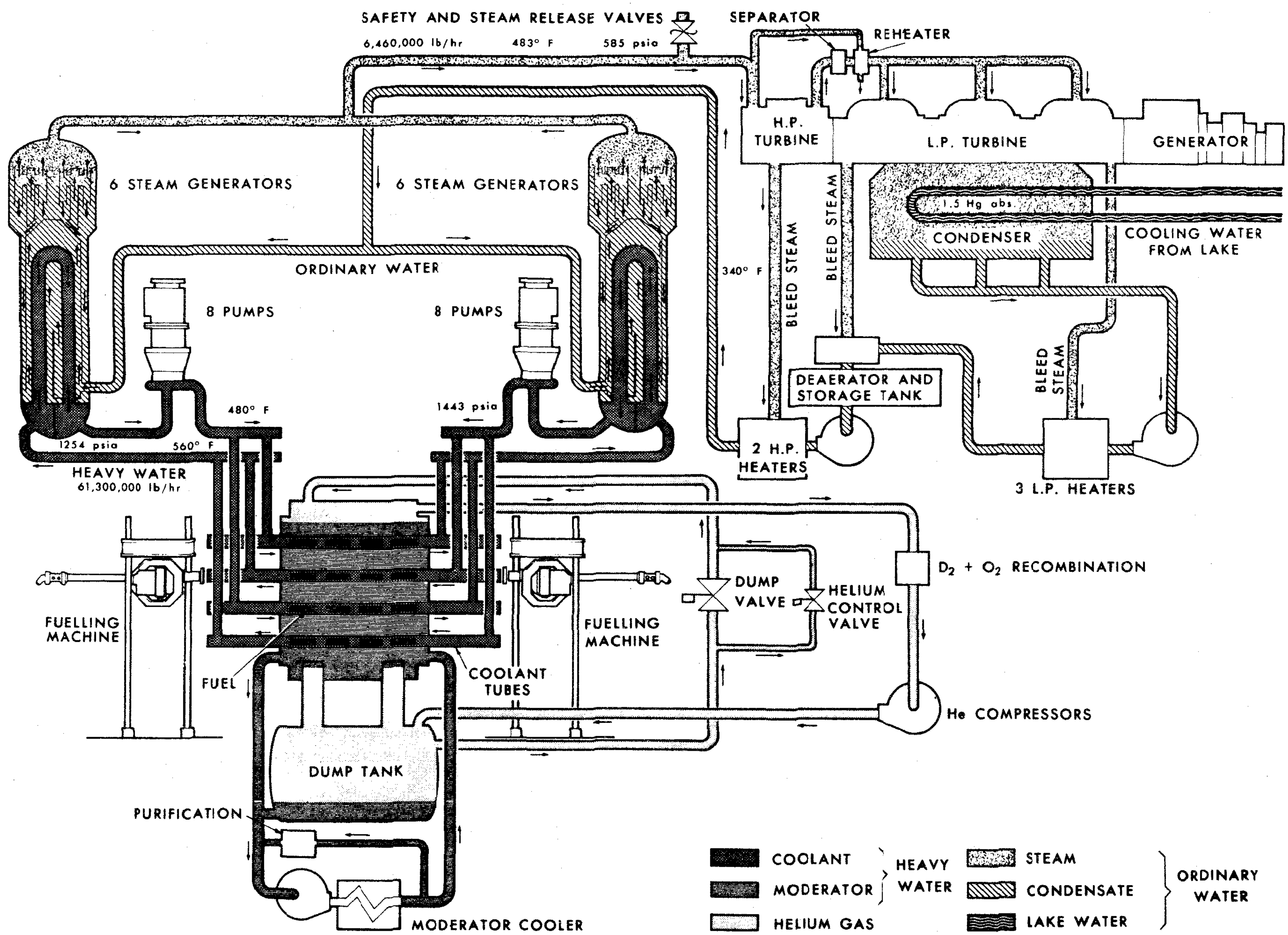


Figure 4 Flowsheet - Coolant System

3. MAJOR DESIGN PARAMETERS

3.1 Coolant Temperature, Pressure and Flow

Coolant temperature and the associated pressure is the most significant parameter. The higher the coolant temperature the higher will be the thermal efficiency of the cycle and hence the lower will be the fuel cost. But along with increasing temperature comes increasing pressure, and thicker pressure tubes (or pressure vessels) and higher corrosion rates.

Coolant temperature fixes steam temperature, as shown in Figure 5.

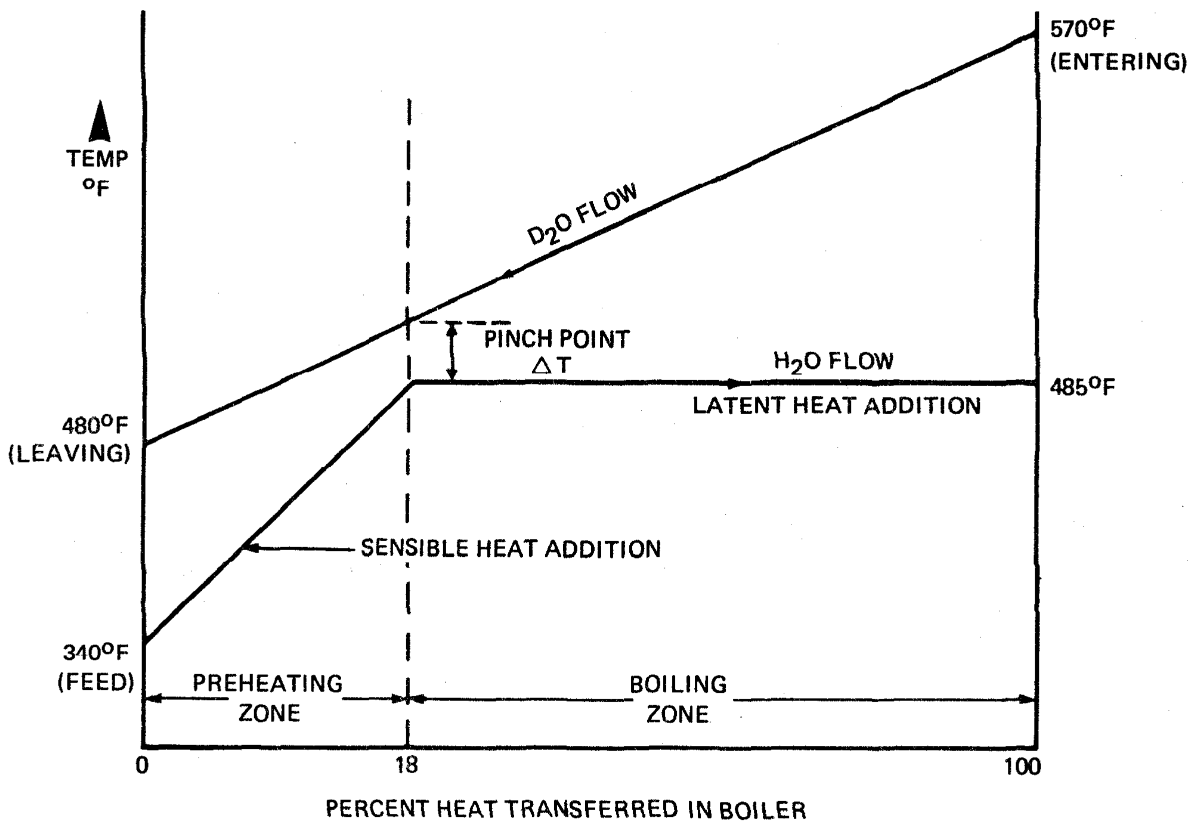


Figure 5 Relationship of Coolant Temperature to Steam Temperature

	STEAM PRESS PSIG	STEAM TEMP °F	STEAM QUALITY %	COOLANT INLET TEMP °F	COOLANT OUTLET TEMP °F
NPD (25 MWe)	415	452	99.75	485	530
DOUGLAS POINT (200 MWe)	569	482.7	99.75	480	560
GENTILLY (250 MWe)	755	514	99.75	500	514
PICKERING (500 MWe)	578.6	485	99.8	480	560
BRUCE (750 MWe)	620	492.3	99.75	485	570
1200 MWe	770	516	99.75	510	570
OCONEE (US PWR)	910	570	SH	554	604
SEQUOYAH (US PWR)	767	514	---------	545	610

Figure 6 Coolant and Steam Conditions

Figure 6 shows coolant and steam conditions for all Canadian and one U.S. power reactors.

Coolant flow is dictated by core design, almost entirely.

The heat generated in the fuel is distributed as shown in Figure 7. The flow required to remove the part appearing in the coolant can be determined by a simple energy balance, if the terminal temperatures are known. The upper limit of flow is set by pressure drop - that is, by the economics of pumps - and by fuel vibration. In CANDU reactors, using fuel elements about 20 inches long and 1/2 inch in diameter held together by metal plates at the ends, fluid velocities in excess of 30 ft/sec are of concern. The elements may chafe against each other, or may wear upon the pressure tube.

Total Fission Heat (Megawatts)	100
Heat generated in fuel assemblies	*93.9*
Heat generated in coolant tubes	0.3
Heat generated in calandria tubes	0.1
Heat generated directly in coolant	0.5
Heat generated in moderator	5.0
Heat generated in shields	0.2
Heat loss, coolant to moderator	0.07
Heat removed by moderator	5.15
Heat removed by coolant	94.6
Heat to coolant from pumps	0.68
Piping losses	0.33
Heat to turbine cycle	95.0
Electrical generation from heat	31.4

Figure 7 Distribution of Fission Heat in CANDU Fuel Channel

In connection with heat removal from fuel, both U.S. and Canadian fuel is rated in units of heat removal per unit of length. U.S. units are kilowatts per foot and Canadian units are kilowatts per metre. Canadian units are often stated in terms of $\int \lambda \, d\theta$, which has units of kW/m, but is not actual linear heat production. It must be multiplied by 4π to get heat production. The reason that fuel is rated in terms of linear heat production is that this is a measure of the temperature difference between the centre of the fuel and the fuel sheath, independent of diameter.

Pressure sets the pressure tube or pressure vessel thickness. In a pressure tube reactor, the tubes are subject to creep, and this is affected, adversely, by neutron irradiation. The material used in pressure tubes is an alloy of zirconium, relatively transparent to neutrons, but not entirely so, and neutron-induced damage contributes to creep.

The make-up of thickness of a pressure tube for an advanced reactor is shown in Figure 8.

Material:	Zirconium - 2.5% Niobium
Allowable Stress Sm:	21,000 psi at 300°C (572°F)

Internal Corrosion Allowance	0.0040 ins.
Internal Wear Allowance	0.0025 ins.
External Corrosion Allowance	0.0015 ins.
Allowance for Unknowns	0.0020 ins.
Total Allowances	0.0100 ins.

Allowance for radial creep: 3% of ID (max)

Typical Case

Nominal ID:	4.070 ins.
Max. installed ID:	4.098 ins.
Op. pressure, max.	1331 psig
Op. temperature, max.	290°C
Neutron Flux	3.39×10^{13} n/cm^2 > 1 MeV

Required minimum thickness for 2% creep:	0.192 ins.
Required minimum thickness for 3% creep:	0.132 ins.
Required minimum thickness for pressure:	0.157 ins.
Resulting creep at 0.157 ins.	2.5%

Figure 8 Pressure Tube Thickness

We are now ready to understand one of the major differences (other than coolant) between the coolant systems for U. S. pressurized water reactors and those for the CANDU cycle.

The pressure vessel reactor has fuel with flow passages relatively open to flow, and not restricted by pressure tubes. It is possible to pump a lot of water past the fuel at relatively low pressure difference. This produces a low temperature rise across the core; less temperature change between the zero power condition and the fuel power condition; and a steam temperature considerably closer to reactor outlet temperature. The flow in a CANDU fuel channel is limited by velocity and pressure drop considerations, and we must accept a higher temperature rise and a lower steam temperature, all other factors being equal.

There is a way around this problem: let the coolant boil. This is done to some extent in the Bruce reactor, and we have successfully operated a pilot plant of 20 MWe size for a long period of time in this mode. It appears that this is the way to go in the long run, even though the design problems are formidable.

3.2 Steam Conditions

Most water-cooled reactor cycles produce saturated steam. The delivered pressure varies from 600 psig to almost 1000 psig.

The boiling water reactors of U.S. design produce the highest pressure - about 950 psig.

The pressurized ordinary water cooled reactors of U.S. design currently are designed to produce steam at about 770 psig, but this is steadily being increased.

The CANDU cycle of the pressurized heavy water type - Pickering and Bruce - produces steam at about 600 psig. The ordinary water cooled CANDU as represented by Gentilly produces steam at about 700 psig.

Earlier in this discussion we showed some of the barriers to producing higher steam pressure in the PHWR type, and we showed that boiling the coolant allows a higher steam pressure for the same reactor outlet temperature and pressure. We have shown that it is economic to do this, and a current design of 1200 MWe capacity produces steam at 770 psig. One of the reasons for this is to avoid developing a turbine for a lower pressure. At this pressure we can procure turbines developed for PWR cycles.

Superheat in a CANDU is possible; but I will state without proving it that thermodynamically it is better to have pressure than superheat, if steam temperature is limiting, as it is in a CANDU cycle.

The cost barrier to superheat with heavy water coolant is the large amount of superheater surface, containing heavy water, that is required.

This barrier is less for organic coolant and for ordinary water coolant. The organic cooled cycle, in fact, is designed to produce superheated steam, by use of extended surface heated by the reactor coolant. The pressurized water cooled reactors sold by Babcock and Wilcox in the

U.S. also provide superheat, by use of a once-through steam generator.

Nuclear superheat has been considered for boiling ordinary water reactors, but has been abandoned in every instance, mainly because the material problems outweigh the relatively minor thermodynamic gain.

3.3 Materials

The coolant envelope meets much the same requirements as the feed-water system of a fossil-fuelled plant. However, it has three other major characteristics:

(a) The part within the reactor must have a low neutron capture cross-section, for neutron economy.

(b) Release of corrosion products, particularly of materials such as Cobalt-59 which become intense gamma emitters under neutron irradiation, must be kept to a minimum.

(c) Leak tightness must be extraordinarily good by ordinary power plant standards, to conserve heavy water if it is the coolant, and to minimize release of radioactivity, whatever the coolant.

The pressure tubes within the reactor are an alloy of zirconium and niobium. This material has a low propensity to capture neutrons, a high tensile strength and excellent corrosion resistance. Its cost is quite high, however. This material, or a variant of it, is used in all CANDU reactors.

The coolant piping outside the reactor is ordinary steel in CANDU systems. U.S. reactors use stainless steel, or other high-alloy materials such as Inconel. There are reasons for this difference, stemming from different chemical control necessities. We are quite content with carbon steel, except for the particular situation of piping carrying two-phase coolant, where erosion-corrosion may indicate use of a low alloy.

When we speak of ordinary steel pipe, of course, we mean pipe purchased and inspected in accordance with the requirements of Section III of the ASME Code: and we have a further requirement, that the cobalt content should be low. Most steel contains trace quantities of cobalt. It is necessary to minimize this, by using as little scrap in the melt as possible, by paying attention to furnace linings and the like. Steel producers are prepared to co-operate; and the extra cost of low cobalt steel is not high. As a matter of interest, the corrosion of the carbon steel in a Bruce coolant loop, which has no cobalt control, is expected to yield about 12 grams a year of cobalt, and this may be reduced by a factor of 10 without seriously affecting sources of supply.

Boiler tubing in the CANDU cycle is a vital barrier between coolant and ordinary water. It is discussed in Part 7.4 of this presentation.

We procure pumps with carbon steel volutes and ferritic stainless internals. The same generally applies to valves. Hard facing materials, such as are used on valve seats, are of various alloys, but we do not use the most common one, Stellite, in coolant systems because of its cobalt content.

We use virtually no austenitic stainless steels in coolant systems. Only instrument tubing and some valve stems are austenitic. The reason is that we thus gain a degree of freedom from chloride-assisted stress corrosion cracking. The source of the chloride is the ion exchange materials used in the purification systems.

4. DESIGN CODES - NUCLEAR AND CONVENTIONAL

The rules for the design, fabrication and inspection of pressure vessels, boilers and piping system are enacted in the "Boiler and Pressure Vessel Act". The authority for the administration of the Act is invested in the Chief Inspector and his appointed officers from the Boilers and Pressure Vessels Branch of the Department of Labour. In some Provinces the name of the Act and the titles can be different.

In Ontario, Clause 40 of the Act specifies that the publications of CSA, ANSI and ASME contain the rules for the design, fabrication, installation, inspection, testing, operation and use of boilers, pressure vessels and plants.

In Canada, the Provincial Act and Regulations take precedence over any Code rules. Certain paragraphs of the Code, therefore, dealing with Stamping (U or N stamp) Authorization (NDT personnel and procedures, welding qualifications and procedure, inspection personnel) and Reports (Forms issued by D. O. L. are to be used) do not apply.

Also exemptions in the Act such as 15 psig pressure, 1-1/2 cu. ft. capacity 6" ID or hydraulic service at atmospheric temperature, will have precedence over Code exemptions.

Non-Nuclear Application

Components (pressure vessels, piping systems, pumps, valves and fittings) for non-nuclear applications will follow the rules of the ASME Code Section VIII, Division 1 or 2, for Pressure Vessels, Section I for Boilers, and Section IX for Welding.

ANSI B31.1 for Power Piping System
B31.2 for Industrial Gas and Oil Piping System
B31.3 for Petroleum Refinery Piping, etc.

Other ANSI Codes are applicable to valves, pumps and fittings.

Nuclear Application

Section III of the ASME Code sets the rules for construction of nuclear power plant components.

"Construction" as used in the Code is an all-inclusive term comprising materials, design fabrication, examination, testing, inspection, certification in fabrication and installation.

A nuclear power plant consists of one or more nuclear power systems and containment systems as well as other systems not covered by the rules of Section III.

A nuclear power system as used in the Code is that system which serves the purpose of producing and controlling an output of thermal energy from nuclear fuel and those associated systems essential to the functions and overall safety of the nuclear power system.

The primary heat transport system in the CANDU reactors is a nuclear power system.

AECL as designer of the nuclear power plant has to meet the Code requirements and satisfy the authorities. Some of the designer's responsibilities are:

1) Establish Code Classification for all components (Code Classes are 1, 2, 3, or MC)

2) Prepare Design Specification for all components.
The design specification shall contain sufficient details to provide a complete basis for construction in accordance with the Code. Some of the information to be included are as follows:

(a) the function of the components

(b) the design requirements

(c) the environmental condition including radiation and earthquake.

3) Stress Report

Each Class 1 and MC component, as well as Class 2 vessels designed to Section VIII, Div. 2, requires a stress report.

The stress report can be prepared by the component manufacturer or his agent and should be verified by AECL.

For some components, such as the calandria, AECL prepares the stress report and the verification is done by different AECL Branches.

4) Materials to be used.

5) Additional NDT, if required.

6) Documentation, etc.

Section III 1971 Edition incorporates the previous Codes -

Section III 1968	-	Nuclear Vessels
ANSI B31.7 1169	-	Nuclear Power Piping
Draft 1968	-	Code for Pumps and Valves

The general requirements pertaining to components of Classes are stated in Sub-Section NA.

Sub-section NB applies to Class 1 components; for example, Article NB-2000 deals with materials for Class 1 components.

The requirements for heat treatment, impact testing, NDT examination methods and acceptance standards as well as repairs are given for the type of product such as plate, forgings, castings or tubular products.

Article NB-3000 deals with the design of Class 1 components.

Section VIII of the ASME Code covers conventional vessels. Within this Code, the designer has a choice of using Division I, which has conservative stress levels and less inspection and materials testing, or Division II, which allows the higher stresses permitted for nuclear vessels but also requires more analysis and more inspection.

Section XI of the ASME Code covers in-service inspection of nuclear vessels.

This code is not entirely appropriate for pressure tube reactors, and eventually a more suitable standard will be produced for Canadian use.

Containment vessels as used in Canadian plants are concrete, and are discussed in a Code entitled "Proposed Standard Code for Concrete Reactor Vessels and Containment", by the ACI-ASME Technical Committee on Concrete Pressure Components for Nuclear Service. I say "discussed" because this is not a final edition and is optional. We have certain reservations about it. Those in the audience interested in this topic might like to pursue it further with our specialists.

5. OPERATING CONSIDERATIONS

5.1 *Reliability and Maintainability*

Reliability is essential for nuclear power to be economic. Nuclear plant is substantially more expensive than conventional, and its advantage, besides a very low environmental impact, lies in low fuel cost. But when the plant does not run, all costs except fuel continue.

It appears currently that loss of generating capability is about equally divided between the nuclear steam supply and the conventional equipment. The boiling water reactors appear marginally less reliable than the PWR's, based on not very recent published data, but the reliability of the boiling water turbine cycle may be somewhat higher, giving a fairly uniform total capability. A typical figure for a recent plant in a typical year is 80% capability. A breakdown of reliability into NSSS and total is shown in Figure 9.

Pickering has emerged as one of the most reliable plants in the world, at least in its initial years. Incapability figures are accumulated and published by Ontario Hydro, and perhaps will be referred to by Ontario Hydro speakers.

PLANT	SIZE MWe	TYPE	PERIOD	TOTAL PLANT AVAILABILITY	NSSS AVAILABILITY	NSSS FORCED OUTAGE RATE
TARAPUR I	380	BWR	Mid 69 thru Mar 71	88.6	91	3.79
TARAPUR II	380	BWR	Mid 69 thru Mar 71	84.6	89.1	0.28
OYSTER CREEK	550	BWR	late 69 thru Mar 71	80.3	85.1	1.68
NINE MILE POINT	525	BWR	July 70 thru Mar 71	85.5	97.6	0.03
TSURUGA	357	BWR	Mar 70 thru Mar 71	82.3	86.7	0
KRB	252	BWR	May 67 thru Apr 71	73.5	89.9	0
YANKEE ROWE	185	PWR	July 71 thru Dec 70	----	84	0.6
CONN.YANKEE	590	PWR	Jan 68 thru Dec 70	80	89	1
SAN ONOFRE	450	PWR	Jan 68 thru Dec 70	66	69	2.7
ZORITA	160	PWR	Dec 60 thru Dec 70	----	95	----
BEZNAU	364	PWR	July 69 thru Dec 70	70	76	----
GINNA	450	PWR	Mar 70 thru Dec 70	63	85	----

Ref. Nucleonics Week, 22 July 1971

Figure 9 Nuclear Plant Availability

Reliability appears very high on our design priorities. Our approach is to perform a series of audits, in which the failure rate and the consequences of failure are assessed for every component in the nuclear system.

The estimates of failure rate are based on our own experience wherever possible, and where we do not have direct experience we use the data of others such as the Edison Electric Institute. When an audit is complete, those components showing a disproportionate contribution are given special attention: more redundancy, more development of components, easier accessibility for maintenance, design changes to ease the consequences of failure.

We expect to achieve 95% availability of nuclear steam on a time basis; and because substantial de-rating in our plants is expected to be rare, this means almost 95% capability on a power-produced basis. There is no serious question that we can do this; but the cost remains to be seen.

5.2 Radiation Dose

We outline here the basic problems in controlling radiation dose and our approach to solving these problems.

Units

The unit of radiation dose equivalent used in this context is the rem, or Roentgen Equivalent Man. A Roentgen is a quantity of gamma radiation which will produce so many electrostatic units of charge in air. Human tissue exposed to one Roentgen of gamma radiation is said to receive a dose of one Roentgen. A rem is that dose of any radiation which will produce the same biological effect as one Roentgen of gamma radiation. This dose received by one man is one man-rem.

The dose to which any of us may be exposed is limited by law. The maximum allowable personal whole body dose for an atomic energy worker is 5 man-rem per year, although the average dose received by operators and maintainers usually is limited to about 3 man-rem per year for administrative reasons.

Origin of Radiation

As previously described, the coolant system and some auxiliary systems become radioactive because material is transported through the neutron flux in the reactor and later appears in other areas. The principal isotope is Cobalt-60, which is made from Cobalt-59 by neutron absorption and to a lesser extent from Copper-63 by a similar process.

Further, the coolant leaving the reactor produces radiation from Nitrogen-16, which is produced by irradiation of Oxygen-16, a component of the coolant (D_2O).

To further complicate matters, a portion of the deuterium in the coolant is converted by neutron irradiation into tritium. Tritium is a weak beta emitter and may easily be shielded, but when it is released to the atmosphere it may be taken into the body. Very stringent limits are set upon internal doses of tritium.

The fuel removed from the reactor contains fission products which are gamma emitters and also contains isotopes in gaseous form which will escape to the environment if the fuel sheath should be defective.

These various sources of radioactivity affect people who must operate the station.

It is the task of the plant designer to control the sources, and to arrange the maintenance as far as possible, to keep the dose received within strict limits.

Dose Control

Early in plant design we set out a man-rem budget to allow us, the regulatory authorities and the customer to see clearly the requirements for shielding and for staff, and to indicate where improvements can be made. In such a budget, only the dose received by the operators and maintainers is considered, and these people may receive for budget purposes an annual dose of 3 man-rem. However, there is a very low background field in the plant buildings - usually a very small fraction of a rem per hour - which over a working year can cause a dose of up to half a man-rem. This leaves a dose available for specific work of about 2.5 man-rem per year.

In a typical plant with 200 operators and maintainers, the assignable total dose would then be 2.5 times 200, or 500 man-rem.

Periodic audits are made as the design progresses, and where possible the bad spots are identified and corrected. New information and new developments appear constantly and these are considered. The yardstick used is cost. The radiation dose at the levels currently permitted is in no sense a health hazard; the problem is not physical harm but having to employ more people to limit the dose that each receives.

You will have noted that nuclear plants are divided into regions - containment, confinement, service area, etc. For interest, I have calculated that about 84% of the station dose is received inside the containment structure; about 6% in the reactor building but not in the containment; and 10% in the service area.

We have made very satisfying progress in controlling radiation dose, and our plants are well within the limits set by the Regulatory Authorities. However, we anticipate reductions in the allowable levels, both to the public and to the plant staff, and we are working hard to do better. You may have read or heard that some people in the nuclear field are a little callous about this subject; this is quite untrue of us. We are fully aware of the concerns, and of course of the problems, and a significant fraction of our research budget is devoted to this single problem: lower radiation dose.

5.3 In-Service Inspection

In-service inspection is the examination of high pressure components during plant life to detect deterioration. The containment system in every plant is designed to contain the coolant released by the maximum credible accident, and hence in-service inspection is a defence "in depth" of the containment system and hence of the general public.

The ASME Code contains a section covering in-service inspection. This document is generally accepted in the U.S., but because it is written for pressure vessel reactors some parts of it are not applicable to CANDU reactors. However, the intent is met and, we feel, generally exceeded, in Canada.

The basic requirement is that an examination of certain parts of the high pressure envelope be examined at regular intervals.

To obtain a "bench-mark" for comparison of future results, an examination is made before the plant starts up. This examination differs from the normal manufacturer's examination in that it uses the techniques to be used later - ultrasonics, for example, and not radiography - and generally is done later in plant construction.

It is possible to find flaws which the manufacturing inspection missed, and cases have arisen wherein these flaws are larger than are permitted the manufacturer, but which are not detectable by the inspection techniques required of the manufacturer by current codes. This raises interesting questions about the adequacy of current codes, but the resolution of these cases also says something of in-service inspection. The rule is generally to take out the flaw, however it is found. One simply cannot ignore information, it seems, and it further seems probable that we would not be allowed to ignore it. This appears to be a mis-use of in-service inspection, guaranteed to make it unpopular, rather like the Government using income tax data for a prosecution on another matter, but generally we accept in-service inspection for its basic purpose, with which we agree.

In-service inspection is the responsibility of the owner of the plant. It may be contracted out, and firms exist which will perform the work for a fee.

6. STEAM SYSTEM

6.1 Turbine Cycle - Nuclear and Fossil

The essential differences between fossil fired and nuclear turbogenerators can best be explained by a quick review of the thermodynamic process involved, as shown on an enthalpy-entropy chart, Figure 10. The two types of machine operate in essentially different areas of the chart; both areas having their own peculiar problems. The fossil fired units have to cope with the rigors of high pressure and high temperature encountered in the 'northeast' corner of the chart, whereas the nuclear machine has to cope with the erosive potential of the steam encountered around the 'saturation line'. Relating the two types of machine to their hydraulic equivalents, the fossil fired unit is a high head, low flow unit, whilst the nuclear unit is a low head, high flow unit.

The essential comparative operating condition and machine features are given in Figure 11.

The following features should be expanded upon to give additional clarification.

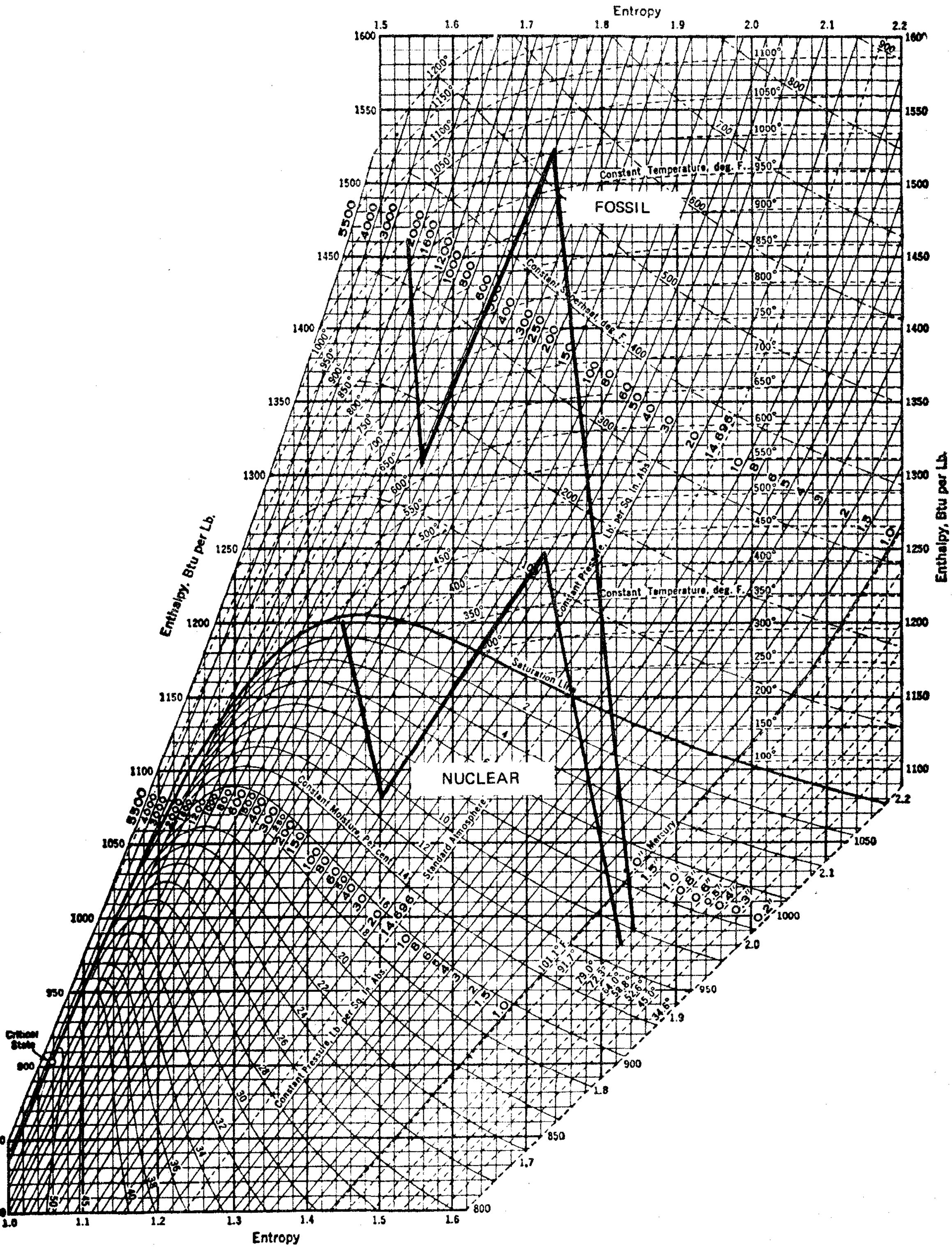


Figure 10 Expansion Lines

PARAMETER	UNITS	FOSSIL	NUCLEAR
Output	Mw	534	540
Main Steam Pressure	p.s.i.g.	2,350	570
Main Steam Temperature	^{o}F	1,000	483
Main Steam Flow	lbs/hr.	3,600,000	6,000,000
Main Steam Volume	ft^3/sec.	336	1,280
H.P. Exhaust Steam Condition	—	$100^{o}F$ Superheat	9% Wet
Reheat Pressure	p.s.i.g.	570	51
Reheat Temperature	^{o}F	1,000	435
L.P. Exhaust Area	ft^2	220	540
L.P. Exhaust Steam Volume	ft^3/sec.	290,000	630,000
L.P. Exhaust Wetness	%	10	11
L.P. Configuration	—	4 Flows 30" Blades 3600 r.p.m.	6 Flows 38" Blades 1800 r.p.m.
Turbo-Generator Length	ft	150	183
Condenser Area	ft^2	150,000	280,000

Figure 11 Comparison of Fossil and Nuclear Turbines

Reheat

With the use of very high pressures for fossil fired units, steam reheating has been employed to reduce the effect of erosion in L. P. turbines and to produce some thermodynamic gain in the cycle efficiency. The use of reheating has the same beneficial effects on the nuclear cycle and is widely used, see Figure 12. The main steam supplied to a nuclear unit is 0. 2% wet and as expansion occurs in the H. P. cylinder, the steam becomes progressively wetter. At the H. P. exhaust the steam is approximately 10% wet. In the usual nuclear cycle, this steam is passed through separators which remove 95% of the free water and then live steam reheaters before passing on to the L. P. turbines. The live steam reheaters have main steam supplied to the

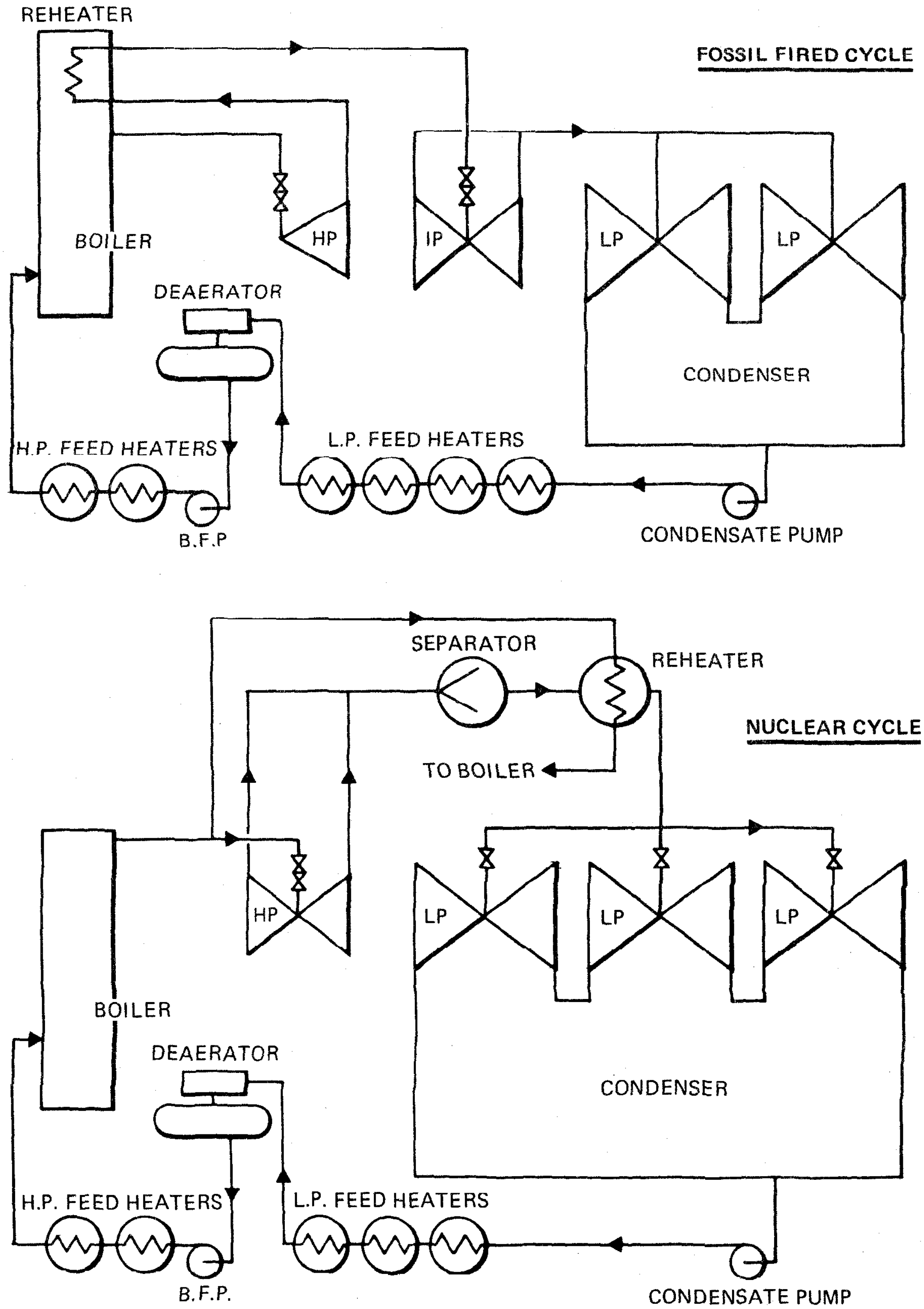


Figure 12 Nuclear and Fossil Steam Cycles

inside of the tubes which condenses, to heat the low pressure steam outside the tubes. By this means, sufficient superheat is imparted to the low pressure steam to ensure that the operating steam conditions in the L. P. cylinders is very similar to that encountered in fossil fired units. The similarity in end point for both cycles is shown on the enthalpy-entropy chart, Figure 10. There is little, if any, difference in L. P. exhaust wetness between the two systems and, therefore, no significant difference in L. P. blade erosion is encountered. If anything, the fossil fired unit is worse because of the higher rotating speed.

Speed

The problems of high pressure and high temperature design provide incentive to use 3600 rpm as the design speed for fossil fired units. In the case of nuclear units, high temperatures and pressure are not encountered and the high volume flow favours the use of slow speed machines running at 1800 rpm. The lower running speed also provides benefits in regard to H. P. erosion, bearing stability, balancing requirements and generator design.

H. P. Cylinders

The most unique feature of nuclear turbines in comparison to fossil units is the H. P. cylinder design. Fossil units are designed mainly on the basis of the creep resistance required to counteract the high temperature conditions, and materials are selected accordingly. In the case of nuclear units, the temperatures do not reach those normally associated with creep problems; however, materials must be capable of withstanding the erosive attack of high pressure wet steam. This leads to the extensive use of high chromium and stainless materials. Very careful attention must be paid to the protection of surfaces suffering direct impingement of wet steam and to all areas subject to leakage of wet steam such as diaphragm half joints, casing joints, etc. By careful design, the erosion effects can be virtually eliminated.

Operating Characteristics

The large reheat fossil fired unit is somewhat inflexible because of the thick sections and high temperature differentials involved. Loading and unloading rates are limited to limit thermal stresses. In contrast to the fossil fired cycle involving 1000°F temperatures, the nuclear turbine for a CANDU reactor such as Pickering is subjected to a maximum steam temperature of only 490°F. Consequently, the nuclear turbine is much more flexible in its operation and load changes may be achieved rapidly without danger of imposing heavy thermal stresses.

Feed Cycle

The feed heating cycles associated with the two types of unit are essentially similar, the fossil fired unit using 7 stages of feed heating and the nuclear units 5 or 6 stages of feed heating because of the lower final feed temperature.

6.2 Special Nuclear Requirements

There is a cost incentive with the CANDU system to use a final feed temperature lower than the normal thermodynamic and cost optimum. The reason is that a lower temperature reduces the size of the nuclear steam generator, or allows a higher reactor temperature difference, or both. These factors are more important, relatively, in a heavy water cooled, pressure tube reactor than in either a pressure vessel reactor, ordinary water cooled, or a fossil fuel fired design. The final feed temperature is determined by cost optimization, with the occasional glance over the shoulder at the turbine designer - mainly, to avoid departing far from what is currently available in proven form.

A further requirement, to some extent peculiar to nuclear plants, is good feedwater chemistry. Right now some European nuclear plants are experiencing serious concern about boiler tube attack, possibly caused by improper secondary side water chemistry. We do not have a problem so far, but we are watching the situation carefully.

The main requirement is a pH reagent compatible with a range of materials - Inconel or Monel in the boilers, steel or cupro-nickel in the feedheaters, brass in the condenser. Volatile reagents such as morpholine have been used, but currently we require captive alkalinity - sodium phosphate - in the boiler as well. This chemical is beneficial in treating solids originating in condenser leakage - calcium - and is non-volatile, so it remains in the boilers. It does not protect the preheaters. Nuclear preheat is largely unique to the CANDU cycle, and the problem of chemical protection of the pre-heaters thus also is unique. We seem to have solved the problem so far with Monel tubing, but we have no operating experience with Inconel-tubed preheaters. We will have when the Bruce Generating Station operates.

6.3 Conclusion

Experience is being rapidly gained on turbines for nuclear service and much of the U.S. experience on turbines receiving steam from PWR's and BWR's is directly applicable for turbo-generators being used on the CANDU system.

The problems that have been encountered with the use of wet steam have been effectively solved and because all other operating conditions are less arduous than for the fossil fired counterpart, reliable service can be expected from future nuclear turbo-generators.

Credit

I am indebted to Mr. R. J. Walters, Plant Equipment Engineer with Ontario Hydro, for the above information on turbine cycles.

7. SPECIAL PROBLEMS

7.1 Pressure Control

Coolant pressure must be regulated to cope with leakage and with thermal expansion and contraction of the coolant. The upper limit of pressure is the design pressure of the weakest component and the lower limit is set either by the amount of boiling one chooses to tolerate or by the tendency of the fuel to expand after rapid depressurization. Bruce Generating Station is controlled to be at 1242 psia at the outlet headers, and the safety devices are set at 1342 psia. The pressure is allowed to fall to about 1000 psia in certain transients.

There are two accepted methods of controlling pressure: use of a heated surge tank or pressurizer, and use of a feed-and-bleed system consisting of a continuously running high head pump and control valves. We have designed and operated plants with both types. Every U.S. PWR has a surge tank, and of course the BWR's do not require one.

The analysis of pressure, flow and temperature in transient circumstances requires computers and rather sophisticated programs. The program generally is designed to compute a modal network in finite time steps. Rare indeed is the individual or the organization which can seriously check the output of such an analysis; rather, it is accepted on faith, at least until the plant operates. The inevitability of this keeps the analyst reasonably honest.

Figure 13 shows the behaviour of a version of the Bruce heat transport system in response to a few disturbances.

7.2 Chemical Control

To get into a frame of mind to work in or indeed to appreciate the chemistry of a coolant loop, you should view it as a chemical plant carrying hot, toxic and corrosive fluid. We may have underestimated these aspects, but in the last two years we, as well as others, have moved rapidly to gain an understanding of what goes on and why.

The problem is to keep the corrosion product from settling on the fuel, where it becomes radioactive, and then rising like a flight of birds and landing somewhere else.

Corrosion rate is minimized by using a high pH - 9 to 11 - controlled by lithium hydroxide, and by maintaining very low oxygen concentration - less than 5 parts per billion.

The amount of isotopes producing high energy gamma radiation after neutron irradiation in the core is minimized by control of the amount of such isotopes in the tubes and piping. The major villain is Cobalt-59, which becomes Cobalt-60 - the isotope used in medical treatment. The next worst probably is Nickel-58, which becomes Cobalt-58, which has, fortunately, a much shorter half-life.

The corrosion products deposited on the fuel and elsewhere may be brought into suspension in the coolant by a temperature or deliberate chemical cycling. The re-deposition rate is a complex and little understood function of corrosion product concentration, chemical content and particle size. However, it is possible to compete very effectively with natural re-deposition by filtration. One of our major efforts is the development of high temperature filters. We do not have a good one yet. The best bet seems to be - of all things - carbonized coconut shells.

Oxygen is controlled by the addition of excess hydrogen or deuterium gas to the coolant. Hydrazine is not effective.

Corrosion of the walls of the system takes up oxygen, but some of the deuterium freed migrates through the vessel walls. There is a net deficit of deuterium and a production of oxygen, which must be combined with deuterium added as a gas. Radiolytic decomposition of D_2O complicates the problem; but also assists the recombination of the excess deuterium and the free oxygen.

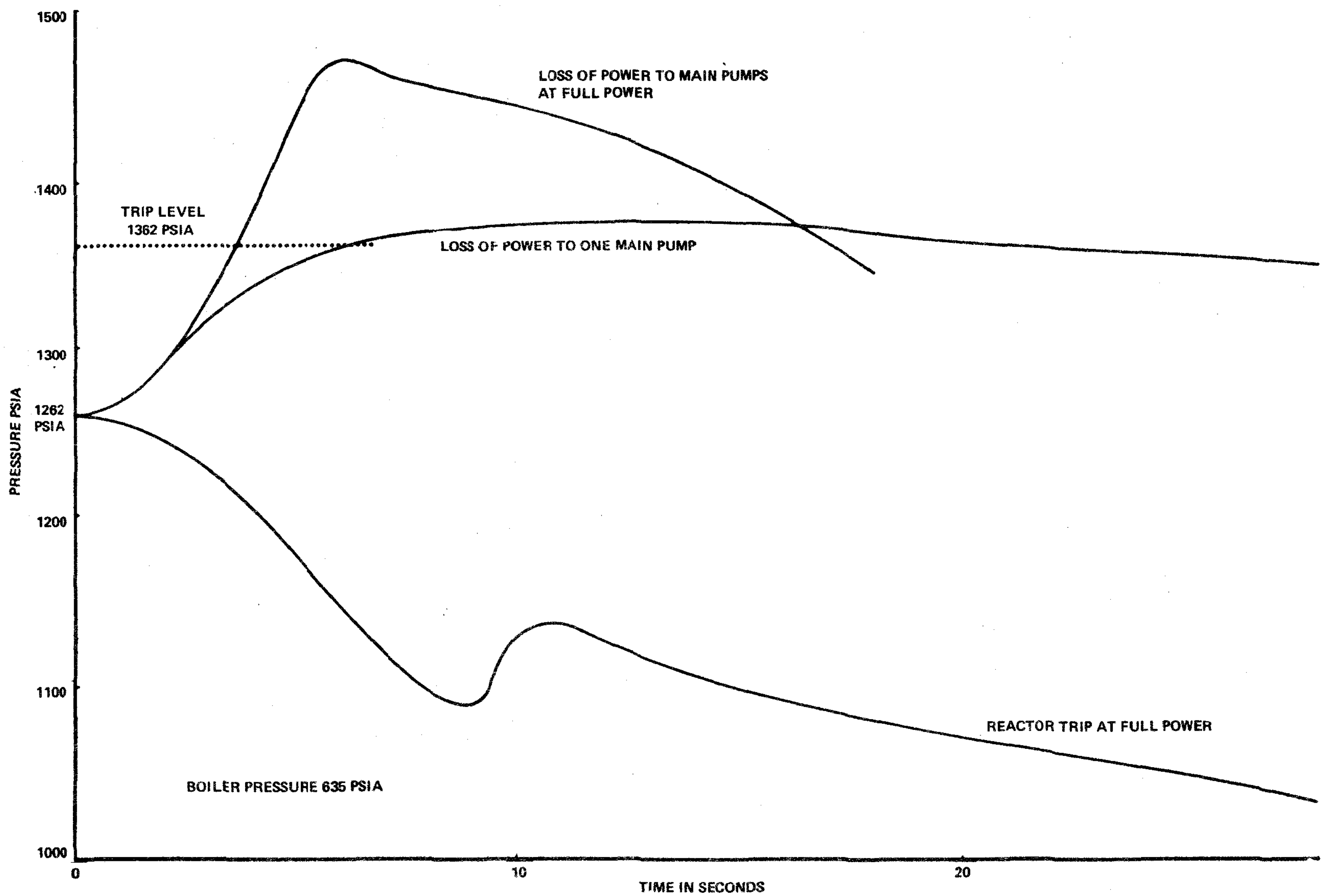


Figure 13 Response of Heat Transport System Pressure to Various Disturbances

The above is a very brief listing of the general aspects. How the various functions are performed is a special topic which I would be pleased to cover in more detail with those of you who are interested.

7.3 Pump Design

We will pass lightly over pump design in this presentation. I propose to cover all aspects in a forthcoming paper for the International Atomic Energy Agency in Vienna; and those of you who are interested might wish to write to me for a copy in the fall.

I will say, however, that pumps have caused a really incredible amount of trouble and delay in nuclear plants. In one of our plants we managed to seriously damage eight out of ten coolant circulating pumps, and the repair took about six months. The pumps were largely rebuilt, and in a small degree redesigned. Similar horror stories may be told of many other plants.

The Pickering coolant pumps - a total of 48 running - have been largely trouble-free. The first Bruce pump - the largest in horsepower of its kind so far built anywhere - has run successfully on test, and installation of others is proceeding. Both the Bruce and Pickering pumps were built in Canada. The pumps for the Rajasthan (India) and the KANUPP (Pakistan) reactors were built in Burnaby, and work pretty good. They are small, but well made, and we are very pleased with them.

The major lesson which I might pass on to you is to test the pumps and the motors thoroughly under simulated reactor conditions, and particularly under off-design conditions. Most, if not all, the troublesome pumps that I know of have run very well at the design point in the test bed, but in the plant they run somewhere else on their performance curve - for a while. And then appears the sort of trouble that is characterized, not just by cost, but by how many months it is going to take to rectify it.

We might add a word about shaft seals.

The glandless pump has the best seal, which is a bolted flange, but its low efficiency and the impossibility of installing a flywheel have retarded its development beyond about 2000 horsepower.

Everyone has his own preference in rotating seals. Our practice has been to buy pumps equipped with seal systems preferred by the pump supplier; but to buy and test a different make of seal which may be substituted later. We follow this up by carrying as spare parts one different seal for each pump manufacturer's seal which is rotating. This is not cheap - seal cartridges cost about $7000 per pump, with testing extra - but it is one way of hedging a bet.

We regard performance in seals as a saleable commodity, and we require performance guarantees. We are presently exploring forms of contract in which we will pay for hours of service obtained, without limit.

Perhaps the most important decision in buying pumps is the decision as to responsibility. There are many ways of setting up a contract. The pump supplier may take the order for the set, and buy the motor on his own. Or the utility may buy the motor and supply it to the pump maker. And so on. Each procedure is full of traps, as no doubt many of you are aware, and our preference cannot be recommended to you blindly. For what it is worth, however, it is to place the entire responsibility on the pump contractor, and then to review his every action. In particular, we ensure good communication between pump supplier and motor supplier.

7.4 Steam Generator Design

AECL at one time designed steam generators, but industry has most capably taken over this work. Two Canadian suppliers and one U.S. supplier submitted bids of their own design for the Bruce boilers. The boilers for four reactors cost about 35 million dollars including tubing. They are being erected this year.

The customer, or his consultant, is involved in setting code requirements. All boilers for water cooled reactors are designed at least in part to the ASME Code, Section III, Class 1. The head, or coolant-containing part, falls under this code. The secondary side may be designed to the same code, or at the option of the customer in consultation with the regulatory authorities it may be designed to Section VIII. Within this section Division I or Division II is designer's choice. The main issue is Section III or Section VIII.

We find that there is little or no economy to be had by using the Unfired Pressure Vessel Code (Section VIII) for secondary side design, even if permitted. The reason is that the extra expense incurred by using the Nuclear Vessel Code (Section III), mainly in inspection, is largely incurred anyway, whatever the design code, in meeting the requirements for in-service inspection (Section XI) and in meeting supplementary requirements of the regulatory authorities.

Thermal and hydraulic design of steam generators is fairly well understood now. There need be little concern about getting the size right, or the steam quality adequate. However, it is most important that all involved should understand very clearly what the vessel is supposed to do. In every case the vessel is required to operate away from its design point for some of its life, and this capability has to be built in. The vessel must be maintained - leaking tubes plugged, dryers examined, manway covers maintained leak-tight - and the clearances and radiation fields in which men must work during repair must be laid out. The range of feedwater temperatures, the rate of change of coolant temperature, the permitted range of water level in the drum, the required water storage, the rate of change of steam demand and many similar factors must first be well understood by the plant designer and then transmitted clearly to the boiler designer. There is considerable opportunity for misunderstanding and error, and the cost in engineering man-hours to carry the job through might very well appear large to those of you familiar with conventional boilers.

Every new design presents a choice of tubing materials. This really is the customer's choice. The U.S. program (for PWR's) is entirely based on the use of Inconel 600 alloy, although back-up programs for other alloys are evident in most large suppliers' plants. We used Monel 400 alloy in Douglas Point and Pickering, and Inconel in Bruce. The difference is roughly five to one in average corrosion rate, favouring Inconel. However, the corrosion rate of Monel is not unacceptable if the cobalt content is controlled to 0.002 percent or less. This is a factor of a hundred below that of stock Monel.

The Germans used Inconel 600 alloy for a while and now apparently have changed to Incoloy 800 alloy. We are not clear as to the reason. It probably is not cost, but more likely a possible higher resistance to certain forms of chemical attack.

Boiler tubing in a nuclear plant costs around five dollars per kilowatt produced, and in our plants is a vital barrier between D_2O and H_2O. In Pickering there is roughly half a square mile of boiler tube surface, and it is required to leak not at all. The cost and the stringency of the requirements justify the most detailed consideration of choice of tubing to suit the environment, and vice versa. There are leaking boilers and repaired boilers in plenty in nuclear plants to warn us of this. We have, of course, not been completely successful ourselves in avoiding tube failure. We have had two tube leaks, one in NPD and one in Douglas Point, both at least partly due to tube vibration. We consider choice of tubing and the things that go into the choice as the most important decision in design of a coolant system.

ATOMIC ENERGY OF CANADA LIMITED
Power Projects

NUCLEAR POWER SYMPOSIUM

LECTURE NO. 7: AUXILIARY SYSTEMS

We will discuss here some of the various liquid and gas systems which service the reactor.

1. MODERATOR SYSTEM

As explained elsewhere in this lecture series, the moderator in the CANDU system is heavy water at low temperature (200^{o}F maximum) and at atmospheric pressure. The pressurized heavy water cooled version of the CANDU system employs about half a kilogram of D_2O per kWe as moderator; in dollars, this is about $30 per kWe.

About 5% of the energy generated in the fuel appears in the moderator, by the slowing down of neutrons born in the fuel, by radiation heating of structural materials, and by direct conduction from the pressure tubes. On a large plant this is a significant amount of energy, but our efforts to use it have failed, so far. The energy is at too low a temperature (about 150^{o}F) to be economically useful in the turbine cycle, and because it may be cut off without notice it is not readily saleable for other uses. Heating of greenhouses has been suggested; and so has district heating.

The moderator system continuously removes heavy water from the calandria to external areas where the water is cooled and purified and then returned to the calandria. On the return path the moderator flow may be specially directed to pick up heat from metallic parts. One such part is booster fuel. This is fuel enriched in fissile material which may be inserted into the reactor temporarily and later withdrawn.

The moderator system in plants using booster fuel is a nuclear system under the ASME Code, and is built to standards similar to those for the heat transport system.

Moderator system piping is stainless steel rather than the carbon steel used in the heat transport system. This is because the chemical conditions do not allow the use of carbon steel. The pH of the moderator is lower than that of the heat transport system, and lower than is suitable for carbon steel, in order to favour the solubility of nuclear poison carried in the moderator for control purposes. The same nuclear poison, boron, is used in U.S. pressurized water reactor systems, which have a combined moderator-heat transport system; and for analogous reasons these plants use stainless steel piping in their systems.

We have been fairly successful with moderator systems. They have contributed very little to unavailability; and this in spite of the varied requirements and functions which the systems serve. The main area for improvement is in equipment cost.

2. HEAVY WATER RECOVERY DRYERS

High pressure systems leak a certain amount; for example, the water make-up to a fossil-fuelled boiler and turbine might amount to half a percent of the total system flow. CANDU heat transport systems leak, and it turns out to be necessary to recover the leakage. To make it easier to get back that part of the leakage that goes into the atmosphere, the parts of the plant containing heavy water are physically sealed and the air in these areas is passed through dryers.

A typical dryer would be a desiccant bed, passing 16,000 scfm and drying this flow to a dewpoint of zero degrees Fahrenheit.

Coolers are also provided in the heavy water areas. These normally take care of heat lost from pipes and so forth, but when the relative humidity is high, as it would be after a spill, the coolers also remove some water by condensation.

The water recovered from the dryers and the coolers is not at reactor grade; it is downgraded by ordinary water which leaks in mainly by the ingress of air and partly from leakage from ordinary water systems. The collected leakage is purified by ion exchange and then upgraded by reflux evaporation or by electrolysis before it is returned to the reactor.

3. PROCESS WATER SUPPLY

Ordinary water is pumped to various locations in the reactor building for heat removal.

Two systems, differing only in supply pressure, are normally employed. A higher pressure is required for cooling vessels at high temperature, to prevent boiling of the process water and consequent deposition.

Heavy water freezes to ice at about 40°F, and it is possible to freeze the heavy water side of a D_2O-to-H_2O heat exchanger solid by using ordinary water at 32°F to 39°F. To avoid this, process water to such vessels sometimes is tempered by recirculating a part of the effluent. Where this cannot be done, the temperature of the coolant and water is monitored and special action is taken if it drops below 40°F.

We do not employ intermediate coolant loops for cooling D_2O. This might be thought necessary to guard against loss of heavy water through leaks. It turns out, however, that possible release of radioactivity to the environment is an equally serious concern.

We have learned - and in fact we have demonstrated over many years - that we can buy heat exchangers which do not leak. We continue to monitor the process water outfall for radioactivity, of course, but it is clear that intermediate coolant loops are not generally necessary. A special case might be made if fouling were severe.

The process water systems are largely built to non-nuclear codes such as USAS B31.1. Some parts such as emergency cooling water supply are built to nuclear codes.

Materials are basic: carbon steel piping and cupro-nickel tubing.

A part of the process water systems may be supplied by the customer. It is well within the capability of most utilities to do the engineering and to arrange for supply of parts. There are some special features, for example in purification of water used to immerse spent fuel, where we would advise; but basically we would encourage utilities to participate in design of this area.