NUCLEAR REACTOR HEAT REMOVAL

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Contents

Summary

Nuclear reactors are able to produce very high volumetric heat release rates. Heat release is governed by the fissile fuel concentration and the neutron flux in the reactor core. Ultimately the heat release rate is governed, not by the fission reaction, but by the rate at which heat can be removed without the reactor fuel or structural materials reaching dangerous temperatures. Heat removal is driven by the temperature difference between the fuel and the coolant so there is an upper limit to the heat flow rates through the fuel and cladding. To some extent this determines the configuration of the fuel and the size of the reactor. A further consideration is the heat transfer from the fuel surface to the coolant. Generally liquids give better heat transfer coefficients than gases and boiling enhances these coefficients.

In water cooled reactors the coolant must be pressurized to prevent boiling at the desired operating temperature or to ensure that boiling occurs only at the required temperature. If

boiling does occur in the fuel channel it affects not only the heat transfer coefficients but also the flow conditions. The friction pressure drop increases with boiling due to the increased velocity arising from the expansion of the fluid when vapor bubbles are formed. The increased velocity also causes an acceleration pressure drop in the channel. These effects tend to retard the flow. Excessive boiling and reduced flow can lead to dry out and overheating of the fuel channel surfaces. This is a dangerous situation and must be avoided.

Calculations to determine heat transfer rates and temperatures in and around nuclear reactor fuel elements are based partially on mathematically formulated equations and partially on experimentally derived relationships. Generally the former can be used for conduction through solids while the latter must be used for the complex situations arising from convection in fluids. Without substantial experimental data the empirical equations are uncertain. This is often the case with the unusual geometries found in nuclear reactor coolant channels. Thus prediction of conditions based on these equations is an inexact science.

The problem of accurately predicting conditions in and around reactor fuel elements can be overcome in two ways. One is to build a full scale or large scale fully instrumented model of the fuel channel and to test it under full load conditions. A very large input of electric power is however required to simulate the high volumetric heat generation rate of nuclear fuel elements. The other method is to develop a sophisticated computer code based on well proven empirical equations for various parts of the system.

This chapter gives a few very general equations which can be used for simplistic calculations sufficient for a basic heat balance and estimate of temperatures and flow conditions in a typical nuclear reactor.

1. Thermodynamic Considerations

1.1. Gas Cooled Reactors

With gas cooled reactors such as the first generation graphite moderated reactors (MAGNOX), advanced gas cooled reactors (AGR) and high temperature reactors (HTR) the gas temperature is limited only by the gas circuit component material properties and steam conditions equivalent to those of fossil fuel fired power plants can be attained. It should be noted however that the AGR, being an evolution of the MAGNOX, was designed to operate at higher coolant temperatures. The CO_2 coolant however caused higher rates of corrosion on coolant circuit components than originally envisaged and coolant temperatures had to be reduced slightly to ensure adequate component life. The HTR using helium as a coolant is not restricted in this way and, by making use of ceramic clad fuel, can in fact operate at substantially higher temperatures. With both the AGR and the HTR the gas temperatures are higher than the permitted steam temperatures so the real advantage of the HTR is the ability of operating the fuel and hence the reactor core at higher temperatures. This promotes heat transfer and allows higher heat fluxes at the surface of the fuel that is in contact with the helium coolant. This in turn permits a smaller surface area and hence a more compact reactor core for a given rate of heat production. The maximum steam temperature is limited by the material of the heat exchanger tubes where the steam is superheated or reheated under pressure. With current materials this temperature is in the region of 560°C or possibly 590°C with special steel alloys. The steam is subsequently used in a steam turbine to produce electrical power with the unavailable energy being rejected to the environment.

1.2. Thermodynamic Cycle Efficiency

The thermodynamic cycle efficiency of a power plant is ultimately limited by the difference between the temperature of heat addition to the working fluid of the cycle and the temperature of heat rejection from the cycle. The efficiency determined in this way is the Carnot cycle efficiency η_{Carnot} and is defined as follows:

$$\eta_{\mathrm{Carnot}} = (T_{\mathrm{H}} - T_{\mathrm{C}}) / T_{\mathrm{H}} \tag{1}$$

Here T_{H} is the temperature at which heat is received (hot reservoir) and T_{C} the temperature at which heat is rejected (cold reservoir). For a thermodynamic cycle receiving heat at 560°C and rejecting heat at 30°C the maximum possible theoretical efficiency is thus 64 percent. This is not achievable in practice since a considerable amount of heat is received at the lower temperatures at which the steam is generated and furthermore a considerable amount of energy is lost in fluid friction as the steam passes through the turbine blading at high velocities.

A similar rough calculation done with heat received at an average temperature of 453°C which is mid-way between 560°C and 347°C (boiling temperature at a typical operating steam pressure of 16 MPa) and a turbine internal efficiency of 0.8 gives a thermodynamic cycle efficiency of 47 percent. This is consistent with the best efficiencies obtained in real steam power plants operating on optimized steam cycles.

1.3. Water Cooled Reactors

The majority of nuclear power plants operating throughout the world utilize light water or sometimes heavy water as a coolant. This includes the pressurized water reactor (PWR), boiling water reactor (BWR), pressurized heavy water reactor (PHWR or CANDU) and even the light water graphite moderated reactor (LGR or RBMK). Water is an excellent coolant with good heat transfer and heat transport properties.

It is also able to remove a large amount of heat during its phase change from water to steam. In the PWR boiling is completely suppressed while in the PHWR or CANDU limited boiling is permitted in the primary coolant loop. In both of these types the steam is generated in a separate secondary steam system. With the BWR and LGR or RBMK however the steam is generated directly in the reactor core and then separated from the recirculating coolant.

Those reactors are all limited in temperature by the steam saturation conditions. As pressure is increased water boils at higher temperatures but the relationship is not linear. In fact as temperature is increased the saturation pressure increases in an apparently exponential manner as shown in Figure 1 where saturation pressures versus temperatures are plotted.

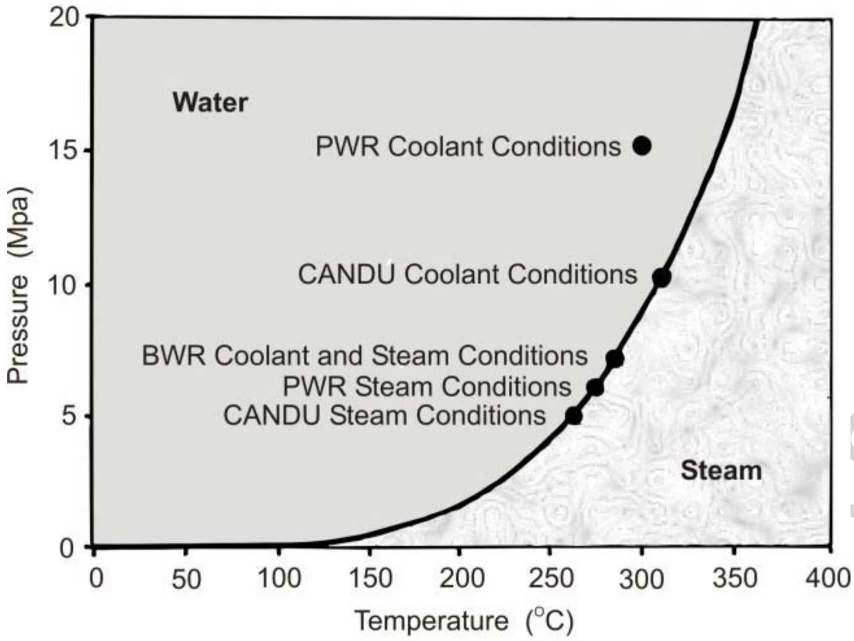

Figure 1: Variation in saturation pressure with water temperature

From this graph it is evident that, for an increase in pressure from 5 MPa to 6 MPa, there is a 12°C increase in temperature but, for an increase in pressure from 15 MPa to 16 MPa, there is only a 5°C increase in temperature. Since the primary coolant loop and secondary steam system must be pressurized to suppress boiling or to allow boiling to occur at a prescribed temperature, there is a case of diminishing returns.

For every 1 MPa increase in pressure a certain amount of structural material must be added to increase the wall thickness of the pipes and pressure vessels but there is less benefit by way of a temperature increase as pressure increases. Ultimately there is an optimum point beyond which the capital cost expenditure in making the system more robust to withstand higher pressures is greater than the operating cost savings in making the system more efficient by operating at higher temperatures.

Such considerations lead to maximum steam generation temperatures and pressures as shown in Table 1.

Reactor Type	Coolant Temperature (°C)	Coolant Pressure (MPa)	Steam Temperature (°C)	Steam Pressure (MPa)
PWR	325	15.5	273	5.8
BWR	289	7.3	289	7.2
PHWR	312	11.2	265	5.1
LGR	289	7.3	285	6.9

Table 1: Water Cooled Reactor Temperatures and Pressures

In the case of the PWR and PHWR (CANDU) heat must be transferred from the primary coolant loop to the secondary steam system so that the coolant temperature must be even higher than the steam temperature. For the PWR where boiling is totally suppressed the coolant pressure is the highest. Overall however it is the pressure of the steam that is the limiting factor since the vessel in which the steam is generated or separated must be of adequate size to accommodate the increase in volume associated with the change in phase. The size of the vessel in turn affects the wall thickness which is governed by manufacturing and transportation limitations as well as the economical considerations mentioned above.

1.4. Steam Cycle Efficiency

The efficiency of a thermodynamic cycle depends upon the difference between the temperature of heat addition T_{H} to the working fluid and the temperature of heat rejection T_{C} from the system. This is the Carnot cycle efficiency η_{Carnot} defined again as follows:

$$\eta_{\mathrm{Carnot}} = (T_{\mathrm{H}} - T_{\mathrm{C}}) / T_{\mathrm{H}} \tag{1}$$

In a water cooled reactor such as the PWR, BWR, PHWR (CANDU) or LGR (RBMK) most of the heat addition occurs during the boiling process, that is, at the saturation temperature corresponding with the saturation pressure. Some heat is of course added to raise the water temperature from slightly subcooled conditions to saturated conditions. All the heat rejection occurs in the condenser at a temperature just slightly higher than the cooling water temperature but is also constant and determined by the condenser pressure. The ideal cycle efficiency based on the above definition for a PWR operating between 273°C and 30°C is 45 percent and for a CANDU operating between 265°C and 30°C is 44 percent. This is not achievable in practice due to a considerable amount of energy being lost in fluid friction as the steam passes through the turbine blading at high velocities. Assuming a turbine internal efficiency of 0.8 the realistic steam cycle efficiencies would be in the order of 36 percent and 35 percent respectively. This is consistent with the best efficiencies obtained in real water cooled nuclear power plants operating on optimized steam cycles. There are naturally some other conversion losses and diversions of energy for plant use so the thermal efficiency of the whole plant is always somewhat less than the steam cycle efficiency.

1.5. Steam Conditions

There are certain problems associated with the generation of steam directly in the reactor core as is the case with the BWR and LGR (RBMK). The generation of steam within the fuel channel changes the following key parameters:

- Pressure drop along the channel
- Neutron moderating capabilities
- Heat transfer from the fuel

As steam is generated, the specific volume of the coolant increases. In a channel of constant cross section this results in acceleration of the mixture and an additional acceleration pressure drop due to the inertia of the fluid. An increasing pressure drop tends to reduce the flow and a reduced flow would cause an increase in the amount of steam generated. This can lead to instability in the flow and alternate surging in parallel fuel channels. This problem can be alleviated by designing the fuel channels with inlet nozzles or orifices which create a pressure drop sufficient to stabilze the flow between adjacent fuel channels.

Since both light water and heavy water have good neutron moderating properties, any void formation in the core due to boiling will affect the neutron energy and hence the fission rate. Generally increased voidage causes poorer moderation and lower probability of neutron interaction causing fission. This means that increased boiling results in lower fission power giving a desirable negative feedback effect. If however the reactor core configuration is such that the neutrons are over moderated and continue to diffuse through the moderator after thermalization, then the lesser absorption of neutrons in the moderator has an adverse effect. When voids occur due to boiling, the decreased absorption in the less dense diffusion medium causes an increased neutron flux and increased rate of fission. This creates an undesirable positive feedback effect that can lead to instability and power surges.

The third consideration is that of heat transfer from the fuel. Increasing boiling on the surface of the fuel elements causes greater agitation of the liquid so increasing the heat transfer coefficient and promoting the removal of heat. There is a limit however and, when the vapor bubbles are formed on the surface at a rate greater than that at which they are removed by the flowing coolant, a film of vapor is formed. This vapor film has poor heat transfer properties causing overheating of the heated surface. The effect is sudden and dramatic and usually results in failure. The heat transfer rate at which this occurs, known as the *critical heat flux*, changes with coolant velocity or flow rate. There is increased risk at low flow rates due to lower turbulence and less coolant to absorb the heat.

These three effects are interdependent and, to avoid instability problems, it is desirable to avoid positive feedback effects and to ensure an adequate margin below the critical heat flux while at the same time avoiding excessive changes in neutron flux in different parts of the reactor. For these reasons the degree of boiling in a reactor core is limited. The steam quality or mass fraction produced in a BWR is between about 10 percent and 15 percent and in a LGR (RBMK) between about 20 percent and 25 percent. The void fraction is naturally much greater due to the lower density of the steam. Once the steam has been separated from the circulating coolant it is dried to produce saturated steam at the prevailing pressure for use in the turbine.

In water cooled nuclear reactors having a separate primary coolant loop, steam is produced in a steam generator which is not subject to the same constraints as is the case for steam production in the reactor core. Heat is transferred from the primary coolant circuit to the secondary steam system according to the fundamental heat transfer equation:

$$\Omega = U\ A\,\theta \qquad (2)$$

Here Ω is the rate of heat transfer, U is overall heat transfer coefficient, A the surface area of the tubes and θ the temperature difference between the two fluids. A certain temperature difference is required to obtain the required flow of heat but too great a temperature difference results in poor overall thermodynamic performance.

In a typical PWR the primary coolant temperature goes from 323°C to 286°C while the steam is generated at a temperature of 273°C. This gives a θ from 50°C to 13°C giving little scope for providing superheat to the steam. At the most only about 20°C to 30°C of superheat could be provided and still maintain an adequate θ in the steam generator for effective heat transfer. To obtain such superheat a once-through steam generator is required and has been provided in certain PWR plants. It is more common however to utilize a circulating steam generator in which the steam is separated from the water and then dried to produce saturated steam at the prevailing pressure for use in the turbine.

As a result of these considerations it is seen that, except for a few PWR reactors using once through steam generators, all nuclear reactors of the water cooled type, that is PWR, BWR, PHWR (CANDU) and LGR (RBMK), produce saturated steam at moderate pressures. These pressures as explained earlier are determined largely by the saturation characteristics of steam and water and the economic balance of capital cost and thermodynamic efficiency. For all these reactors steam is produced at pressures ranging from about 5.0 MPa to about 7.5 MPa.

1.6. Nuclear Steam Cycles

Gas cooled nuclear reactors producing high temperature gas which can be used to generate steam at the same high temperatures as conventional fossil fuel fired plants can employ a conventional steam cycle and steam turbine. Water cooled nuclear reactors producing saturated steam directly, as in the BWR and LGR (RBMK), or indirectly, as in the PWR and PHWR (CANDU), however are limited in steam temperature for the reasons given in the preceding section. This means that the steam cycle is no longer conventional but has to use lower pressure and lower temperature steam. There are three important implications.

Firstly the steam entering the turbine is saturated and soon becomes wet as it expands to lower pressures and produces work within the turbine blades. The high pressure turbine must therefore be able to handle steam containing moisture. This requires the turbine to have proper drainage channels and appropriate steam path design to minimize the effect of moisture erosion. Expansion in the turbine must be limited to ensure that the moisture content does not exceed about 12 percent.

Secondly, with the lower steam pressures at the high pressure turbine inlet, the steam leaving the high pressure turbine is at a correspondingly lower pressure and the steam specific volume has already become quite large. At this point reheating of the steam is required to ensure good steam conditions at the inlet to the low pressure turbines. Due to the large specific volume of the steam at this point, the steam pipes have to be very large and reheating must be done close to the turbine to avoid long runs of very large diameter piping. Reheating is accomplished using steam directly from the nuclear reactor or steam generators. This is able to raise the temperature of the reheated steam to almost that of the original live steam. Steam conditions in the low pressure turbines are thus very nearly the same as those of conventional fossil fuel fired units and the exhaust leaves the low pressure turbine with a wetness of about 10 percent.

Thirdly the lower temperature at which the steam cycle receives heat means that the cycle efficiency is less than that of conventional fossil fuel fired units. For an equivalent power output therefore a substantially greater mass flow rate of steam is required. This necessitates a larger turbine for the same power output and, furthermore, power outputs of nuclear plants tend to be larger than those of fossil plants. The largest steam turbines are therefore to be found in large water cooled nuclear plants and are designed for these huge steam flows. One implication of this is a large exhaust flow area with very long last stage blades. These in turn are subject to high centrifugal stress and the turbines are usually designed to run at half speed that is 1500 rev min^{-1} for 50 hertz electrical power or 1800 rev min^{-1} for 60 hertz electrical power.

The large steam flow in turn requires a large condenser. One implication of a large condenser is that its hotwell can provide a significant storage capacity to absorb fluctuations of the water inventory in the nuclear reactor or steam generators so that there is a lesser need for a separate feedwater storage tank. In fact in some PWR and BWR systems with a large condenser hotwell the deaerator and associated storage tank have been eliminated. Deaeration prior to startup is done in the hotwell which also provides the required reserve feedwater storage. With such designs the nuclear reactor or steam generators must of course be able to accept feedwater at the condenser temperature without suffering thermal shock.

All nuclear steam cycles, like conventional steam cycles, incorporate regenerative feedwater heaters to preheat the feedwater returning to the nuclear reactor or steam generator. Due to the lower steam temperatures there are, however, fewer feedwater heaters with usually not more than six whereas conventional plants may have up to eight.

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Biographical Sketch

Robin Chaplin obtained a B.Sc. and M.Sc. in mechanical engineering from University of Cape Town in 1965 and 1968 respectively. Between these two periods of study he spent two years gaining experience in the operation and maintenance of coal fired power plants in South Africa. He subsequently spent a further year gaining experience on research and prototype nuclear reactors in South Africa and the United Kingdom and obtained M.Sc. in nuclear engineering from Imperial College of London University in 1971. On returning and taking up a position in the head office of Eskom he spent some twelve years initially in project management and then as head of steam turbine specialists. During this period he was involved with the construction of Ruacana Hydro Power Station in Namibia and Koeberg Nuclear Power Station in South Africa being responsible for the underground mechanical equipment and civil structures and for the mechanical balance-of-plant equipment at the respective plants. Continuing his interests in power plant modeling and simulation he obtained a Ph.D. in mechanical engineering from Queen's University in Canada in 1986 and was subsequently appointed as Chair in Power Plant Engineering at the University of New Brunswick. Here he teaches thermodynamics and fluid mechanics and specialized courses in nuclear and power plant engineering in the Department of Chemical Engineering. An important function is involvement in the plant operator and shift supervisor training programs at Point Lepreau Nuclear Generating Station. This includes the development of material and the teaching of courses in both nuclear and non-nuclear aspects of the program.