Indian Institute of Technology Madras
Present

NPTEL
NATIONAL PROGRAMME ON TECHNOLOGY ENHANCED LEARNING

NUCLEAR REACTOR AND SAFETY
AN INTRODUCTORY COURSE
Module 08 Lecture 02
Validation and Dynamic Analysis Continued

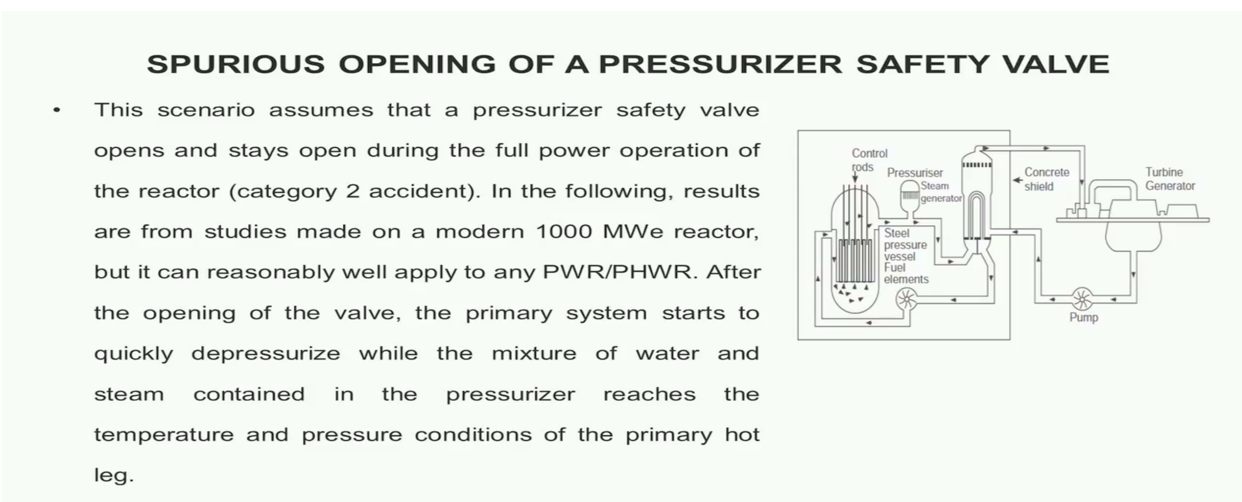
Dr. G. Vaidyanathan
School of Mechanical Engineering
SRM University

Good morning! In the earlier lecture, I had given you a glimpse of how we validated the DYNAM code which was used to analyze the parametric variations in case of events like a pump trip or a power failure, how the different temperatures vary. Finally, we have to see that the temperatures of the clad do not cross the limits the temperature of any part in the plant does not cross its limit, and also these variations are an input for the mechanical design of the plant or the different components.

So we also saw how it was validated based on literature, based on tests on similar reactors, based on tests in our own reactors, then integral tests on the plant. So that way we found that there was a good match between what was predicted and what was actually observed on the front.

In this lecture, I would like to take you to the analysis of some more events in the different reactors which are important, no doubt.

(Refer Slide: 01:55)

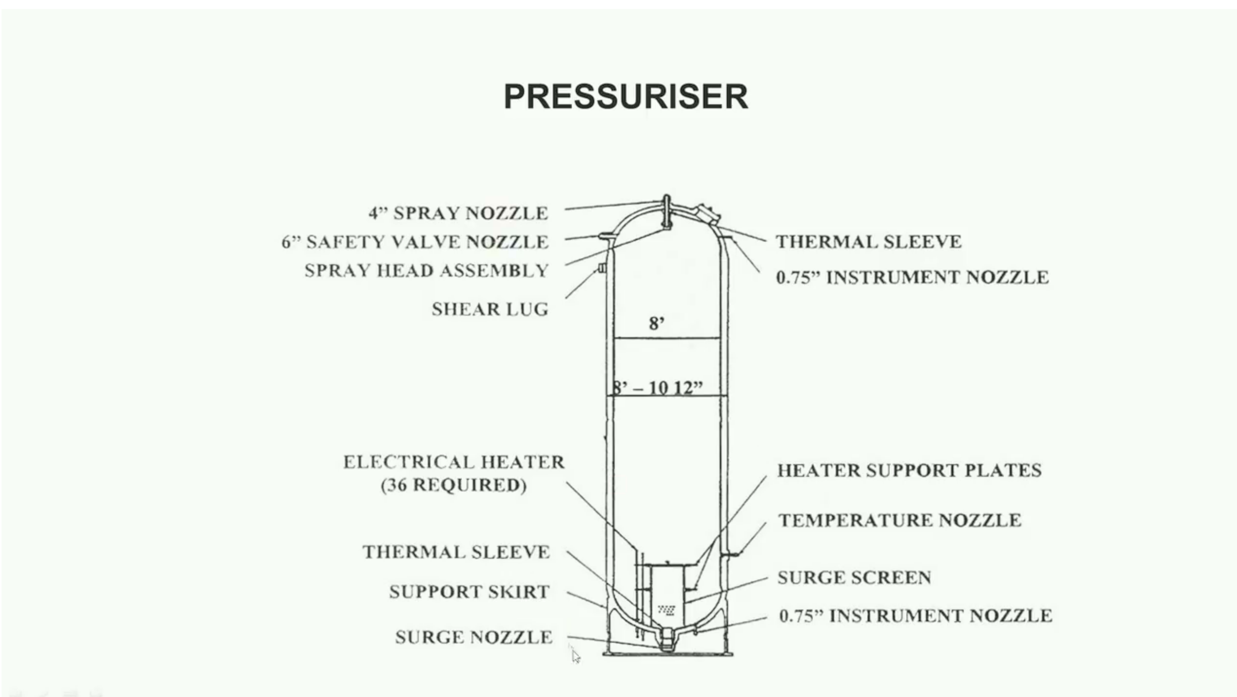


Let us take a pressurized water reactor. Just to recapitulate in the pressurized water reactor you have, this is the reactor vessel, this is the core, and these are the control rods. Water is pumped by a pump, it is light water, it picks up heat, and in order that it does not boil, we keep a high pressure and this is the pressurizer. After taking up the heat, it gives heat to the steam generator where another light water picks up, goes through to the turbine, runs the turbine and comes back.

Now what does this pressurizer do? It basically maintains the pressure of the system by two methods; one is there is a heater and there is a water spray. When it heats up, water boils and the steam space gets steamed and get pressurizes the water. In case I have to reduce the pressure I spray water into the steam area which quenches the steam and the steam volume comes down and the pressure comes down. Now in the pressurizers, there is a safety valve at the outlet and whenever there is a power failure and the pressurizer safety valve has opened, then let us look at a scenario that plant was operating at full power and this pressurizer safety valve open because of the pressure but it didn't close.

This event is exactly similar to the Three Mile Island reactor. Then, of course, the reactor was tripped and other things happen. Now how this thing went through the parameters went through let us see.

(Refer Slide: 04:17)



This is to give you a more clear picture of the pressurizer. You see the electrical heaters quite a good number, about 36 numbers maybe a megawatt power. Then you have the spray nozzle, then you have a nozzle for the safety valve, then you have got the instrumentation nozzles which measure the temperatures and the size of this pressurizer is about eight feet. So let us see how the thing goes.

(Refer Slide: 04:55)

- The reactor is shut down by the intervention of the low primary pressure signal at 10.93 MPa (abs) (109.3 bar (abs)). The normal primary pressure from which the transient starts is 15.82 MPa (abs) (158.2 bar (abs)). At a pressure of 10.93 MPa (abs), the safety injection system is automatically actuated which starts to inject water in the primary system through the high pressure pumps. Conservatively it is assumed that assumed that one high pressure injection pump only operates (single failure), the injection flow rate is initially equal to about 20 kg/ s, increasing to 45 kg /s) when the primary pressure decreases to 5 MPa (abs) (50 bar (abs)).

So the pressurizer safety valve has opened but has not closed. So when the safety valve has not closed what is going to happen. System is getting relieved so the pressure would come down. Now in the case of a pressurized water reactor when the pressure comes down to a level, let us say about 10.93 megapascals, the reactor is shut down automatically. That is how the safety logic is built in. But the normal pressure is normally around 15.82 megapascals. So once it comes to 10.93 we are tripping the reactor

Now we are losing coolant, primary coolant is getting lost through the pressurizer valve which is open, safety valve which is open. So we need to put the coolant. So at this pressure of 10.93 megapascals safety injection water starts. Here we have got two high pressure safety injection pumps but we assume that only one pump operates. This again to remind you we always consider a single failure in the whole safety chain besides the initiating event. For example, here the initiating event was the non closing of a pressurizer valve.

Next step, reactor should trip, it tripped. Ad here, in order that reactor should trip we do have multiple signals. Here we are giving the indication of a pressure signal and these are not one single channel; there are many channels of pressure measurement. So we are sure that the pressure has come down to 10.93 and we need to trip the reactor.

Then in the chain that in the high-pressure injection, we assume one injection pump has failed. So we are doing a conservative analysis and the injection flow initially will be less because the pressure is high, but as the pressure is going down, down and down, the flow increases something like 20-45 kg per second and the pressure comes down to about five megapascals.

So you know these pumps, they cannot eternally operate; they require a minimum suction head to push flow. That is called as a net positive suction head. If that is not there, this pumps would cavitate. So the high-pressure injection pumps cannot continue beyond five megapascals down. So if you just operate, the flow will not happen, they will cavitate and damage the pumps. So what we do?

(Refer Slide: 08:20)

- Subsequently, as the primary pressure continues to decrease, the safety accumulators and the low pressure injection pumps start operating. During this accident scenario, the heat transfer from the fuel rods to the water does not usually reach the threshold of nucleate boiling. In the transient described, the maximum fuel clad temperature is of the order of 843 K (570°C), well below the limit of 1477 K (1204°C) . If, as at Three Mile Island, the safety injection was shut off, the accident would continue to the start of core melt and beyond.

6

We then bring in the safety accumulators which pump in from the low pressure injection pumps which are designed for a lower pressure. So there is a high-pressure injection, now we go to the low pressure injection.

What happens in this state? Earlier because flow was not available, there was a bit of a nucleate boiling in the core, but now by the time this low pressure injection starts, it has come down, they don't go to the nucleate boiling level and the maximum clad temperature is about 570 degree centigrade whereas under the limit is about 1204 degree centigrade.

Now you might wonder, okay, we say everything is okay, the temperature should not fail, but then why the hell in Three Mile Island it failed. Remember, the safety injection was shut off by the operator. Why? He found that the level in the pressurizer is rising; he never realized that the level in the pressurizer is rising because of the steam in the core and that is pushing the water up, whereas the core itself is devoid of water. Unfortunately, because of the lack of proper instrumentation this happened, otherwise Three Mile Island incident would not have happened. So this is what is an event of pressurizer valve stuck open.

(Refer Slide: 09:57)

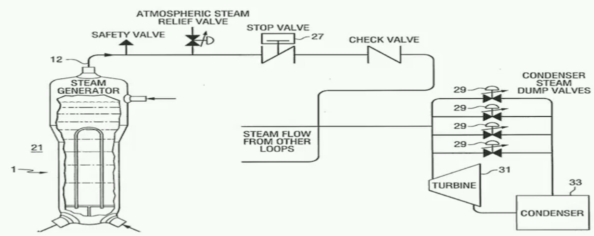
INSTANTANEOUS POWER LOSS TO ALL THE PRIMARY PUMPS

- This scenario assumes that the accident starts at full power, with a progressive slowing down of the pumps. The initiating cause may only be the instantaneous loss of all the external electric power sources.
- The fast shutdown is quick (< 2 s) actuated by the slowing down of the primary recirculation. The actuation signals vary according to design preference and they may comprise loss of pump speed, inadequacy of their electric power supply (voltage and frequency) and reduction of recirculation flow rate.

Another scenario, there is an instantaneous power loss to all the primary pumps. So what happens? The primary pumps getting no power, they do have some inertia in their drive system. It slows down, but it slows down and the power has gone, you know that the power is missing so it is an unsafe situation to run the reactor. So we trip the reactor. So less than two seconds it comes down. Then also the flow reduction also would give a signal. They may also have the pump speed or lack of voltage or the loss of voltage at the pump. So many signals would surely ensure the shutdown of the reactor.

(Refer Slide: 11:17)

- The temperature of the primary water, as well as the primary pressure, initially tend to increase and subsequently to decrease after the reactor scram has operated a few seconds from the start of the accident. The heat loss from the secondary side occurs by steam dump to the atmosphere as the turbine generator combination stops on the scram signal.



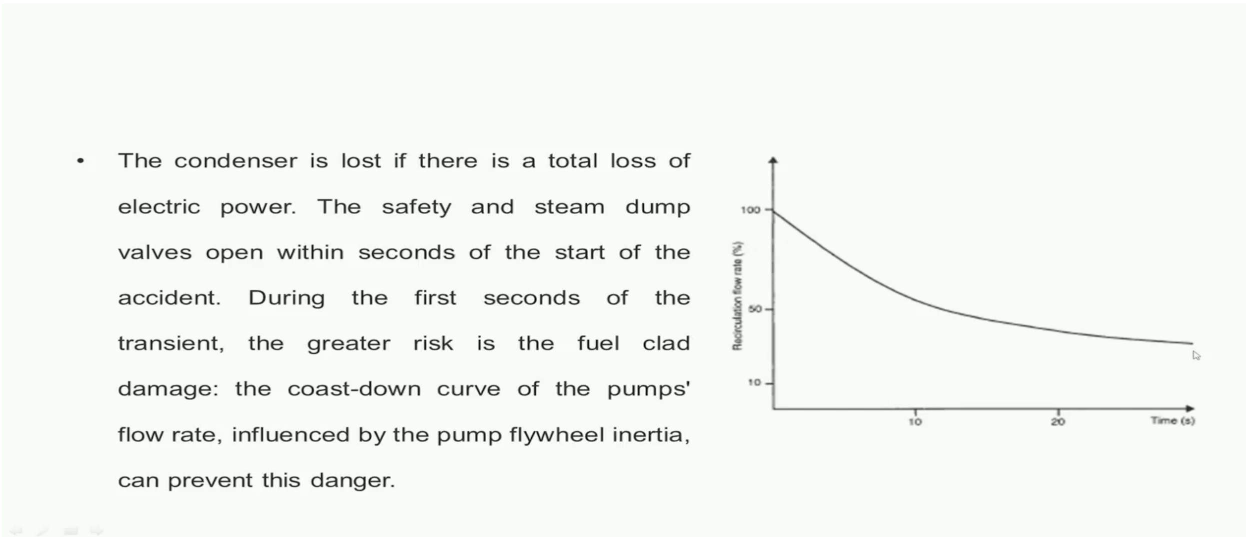
So here what happens? The temperature of the primary water rises because the scram effect takes some time for the reactivity. So initially, your temperature rises so does the pressure. But the moment your reactor has been scram, the reactor has been tripped, the control rods have come down, the power generation is stopped, so then the temperature starts coming down.

Then what happens on the secondary side of the steam generators? You have atmospheric stream relief valve by which we open this atmospheric stream relief valve and we can relieve the steam pressure and as the steam pressure is relieved, there is a flow happening in the steam generator and that in turn removes the heat from the reactor core.

So basically heat removal in this through the steam generator is the first one. There are other parts also available. You have got a bypass across the turbine by which you can directly send the steam to the condenser and from the condenser, you could have a relief valve in case deep pressure goes high.

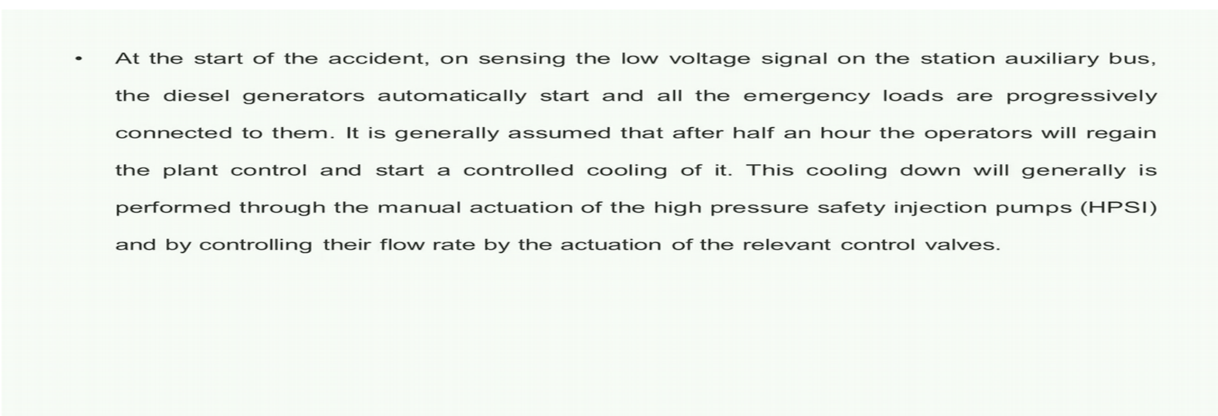
But if there is a total loss of electric power, means your condenser is not available, so if the condenser is not available, you have to wait, use only the safety and the dump valves which I mentioned. The safety valve will automatically open and in order to aid, we have a relief valve also which acts on the pressure. Normally, in all systems we have a safety valve plus a pressure operated relief valve; we do not want the safety valve which is operating on a spring to move up and close very frequently. It is like an ultimate defense in depth. So sensing the pressure, we open this control valve and relieve it. So in case this fails, this comes into picture.

(Refer Slide: 14:10)



So again this is able to relieve the pressure. So how the flows come down, this is how the flow comes down and the flow coming down is again a function of the inertia. Normally, we provide a mechanical inertia, flywheel on the pump drive system which gives a slope goes down, thereby avoiding any boiling in the initial stages.

(Refer Slide: 14:42)

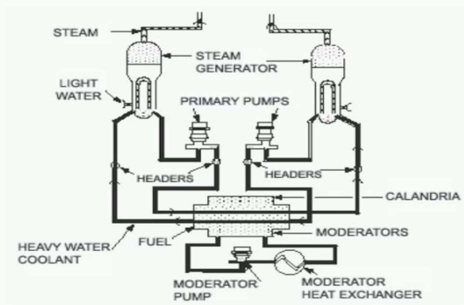


So as I mentioned, power failure is detected by a low voltage on the station bus, the diesel generators start automatically so the emergency loads which are important for the safety are given power. We assume that after about half-an-hour, operators would be in a position to take over the plant control. What I want to emphasize here in this half-an-hour after the incident, we don’t take any credit for any manual action. Operator may be able to act but we don’t take credit in the safety analysis. So this is a very important point which is done in all nuclear; it is not unique to India, every country follows that.

So once the operator has taken over, he would be in a position to take, already the plant is in a safe state; he has to maintain that safe state.

(Refer Slide: 15:49)

LARGE LOCA ANALYSIS OF INDIAN PHWR- 220 MWe



The primary heat transport (PHT) system of a standardized 220 MWe Indian PHWR consists of number of parallel channels in figure-of-eight loop configuration, with one pair of steam generators and pumps in each leg of the loop. Out of the total 306 fuel channels in the Kaiga APS, flow through 153 channels is from South to North and the flow direction in the remaining 153 channels is from North to South.

Now, let us come to the pressurized heavy water reactor which are quite a good number, about 20 of pressurized heavy water reactors we have in India, and the pressurized heavy water reactor to recapitulate, this is the reactor core, it is horizontal not vertical and there are fuel channels here. The coolant goes through, the pump pumps it through the core, comes out, goes to the steam generator, exchanges heat to light water, this is heavy water, the primary heat transport. Again pump, again back into the other opposite direction, another channel, another steam generator. This is actually called as figure-of-eight loop unique to the heavy water reactors.

And the purpose of having a horizontal configuration is because we do on-load fueling; we change the fuel on-load and that is possible only in a horizontal setup. We also have the moderator separately unlike the pressurized water reactors, they are outside in the vessel which is called as a calandria and they have a moderator system also because moderator moderates the neutrons that means some energy is given to the moderator and if you just leave the moderator stagnant, the moderated temperature will increase.

So you need to cool the moderator. So moderator is taken out to be cooled through pumps and then push back. So the moderator is continuously recirculated and this analysis is done for the Kaiga Atomic Power Station, there are about 306 channels like this, 153 in one direction, another 153 in the other direction.

(Refer Slide: 17:45)

- The inlet side of these channels is connected to the Reactor Inlet Header (RIH) and outlet side is connected to the Reactor Outlet Header (ROH) through feeder pipes. The coolant flows through the fuel channels, each of which contains twelve fuel bundles. Each pressure tube is surrounded by a calandria tube which in turn is surrounded by relatively cold moderator. The annulus space between the pressure tube and calandria tube is filled by CO_2. The reactor system consists of two U type steam generator (SG) and two primary coolant pumps (PCPs) in each leg of the loop. The pressure in the PHT system is maintained with the help of feed and bleed system. The feed is isolated on low storage tank level.

Now all these channels are connected at the inlet side and at the outlet side by the headers called as a reactor inlet header and the rear outlet header and through that header the feeder pipes are joining each channel to that and each of these channels contain 12 fuel bundles, and these are all within the pressure tubes and each pressure tube is surrounded by a calandria tube and the calandria tube is surrounded by the moderator, and the annulus between the pressure tube and the calandria tube is filled with carbon dioxide. You might ask why. See, the coolant is picking up the heat and the calandria tube is facing the moderator which is at a lower temperature. So if you don't insulate, the heat will go to the moderator.

So carbon dioxide acts like an insulation. Then the steam generators are U tube steam generators as we saw inverted U tube steam generators. So the heavy water is flowing inside the tubes and the light water is flowing outside the tubes. Then the pressure in the system in this reactor apparently is maintained by a feed-and-bleed system wherein we have a place at the primary pump inlet where we push in heavy water and bleed heavy water from another point and maintain the pressure, but in the later reactors we have used the pressurizer as in the PWRs, pressurized water reactors.

(Refer Slide: 19:45)

- Loss Of Coolant Accident (LOCA) leads to coolant expulsion in a primary heat transport system resulting in depressurization and possible core voiding. This results in deterioration of cooling conditions in reactor channels and increase in power before reactor shutdown, leading to higher fuel temperatures. Coolant expulsion rates during LOCA are dictated by critical flow conditions governed by initial plant conditions prior to the accident, break geometry, location of break, etc. The reactor is provided with an ECCS that comprises a heavy water accumulator (HWA), light water accumulator (LWA), and the long term recirculation system (LTRS). The HWA (15 Te) and LWA (each of 30 Te) are kept pressurized with nitrogen at 60 kg/cm^2 and 40 kg/cm^2 respectively. The LTRS mode comprises ECCS pumps, ECCS heat exchangers, a light water tank and suppression pool.

Now loss of coolant accident is an important one in which there is a, let us say, there is a header failure and your coolant is expelled out of the primary heat route system. With the loss of coolant there is depressurizing, the system is getting depressurized and the core could be without coolant and when the core is without coolant, we know that that there could be high temperatures and if the void loss of coolant happens in a place where there could be a positive reactivity affect, there could be a slight power increase. Then that also could add to the higher temperature, but major thing is the loss of flow which is causing the loss of coolant that is coolant loss is what is going to cause the increase in temperature in those channels.

Now the flow rate is going to be dependent very much on the place where there is a break of the pipe. If it is a small pipe, your flow out will be less, if it is a large pipe it will be more. If it is a crack, it will be less. If it is a guillotine rupture, something like just into two pieces then water can come from both sides. So we take into consideration that as if there is a guillotine rupture, worst case, so that we have a good, very conservative prediction of what is going to happen in the plan.

So it is basically conservatism is required so that we are in a safe situation as far as the reactor is concerned. Now since you are losing coolant, there needs to be an injection just as we saw in the case of the pressurized water reactor. So we have an emergency core cooling system, however, this cooling system is with heavy water whereas there it was light water because there light water was the primary coolant, here heavy water is the coolant.

No doubt, we also have a light water accumulator. The later stages should the heavy water accumulator supply will not be sufficient or should it require, we will be then go for light water. Then these tanks with heavy water and light water are kept pressurized with nitrogen. Now the system consists of pumps, heat exchangers, and then at the end of the removing heat, there will be a separation pool where the heat is given off.

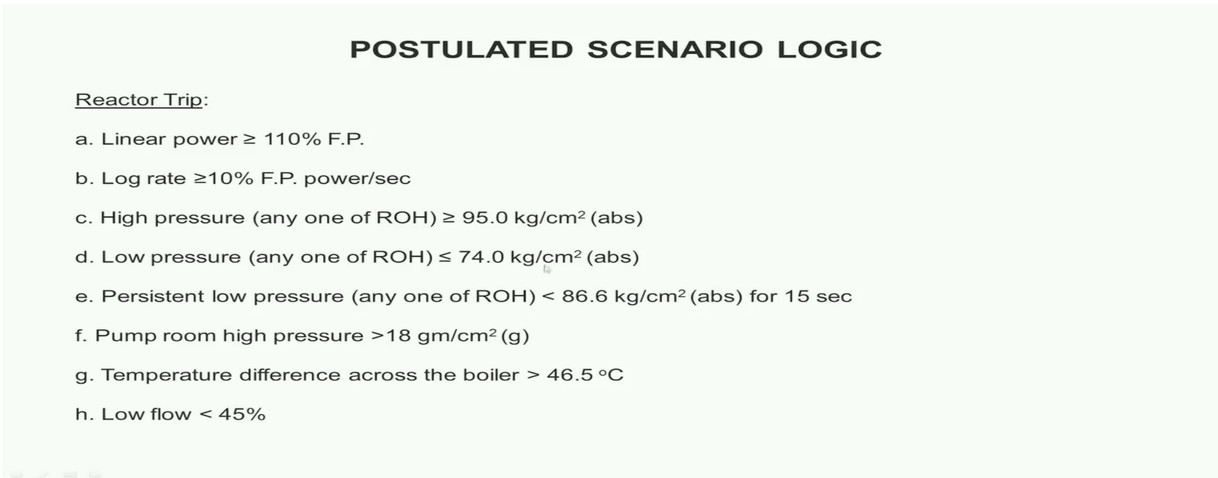
(Refer Slide: 23:26)

- LOCA is detected by monitoring the PHT system pressure with help of pressure transmitters attached to inlet (South and North bank) headers. PHT system pressure falling to 5.5 MPa is taken as LOCA signal for actuating of ECCS phase I, i.e. heavy water injection and pressure falling below 3.2 MPa is taken as LOCA signal for actuating of ECCS phase II, i.e. light water injection and phase III is invoked at 3.2 MPa i.e. water recirculation from suppression pool.

So how do you detect a LOCA? Of course, here the pressure measurement, we have got pressure indicators and transmitters to the different headers of the plant because we can’t say at one place we put, the leak might happen in another header. So all headers do have pressure transmitters and once the pressure transmitter fails to something like reading comes to about 5.5 megapascals, it is taken as a LOCA, loss of coolant accident, signal to actuate automatically the emergency core cooling phase 1. Phase 1 is heavy water injection. Then what is phase 2?

Once the pressure falls to 3.2 megapascals, phase 2 starts wherein we inject light water and then what you call as a phase 3 at about 3.2 megapascals, we take the suppression pool water and push it back to the core. So these are the three phases. So how the scenario looks based on the analysis taken from all the uncertainties into account?

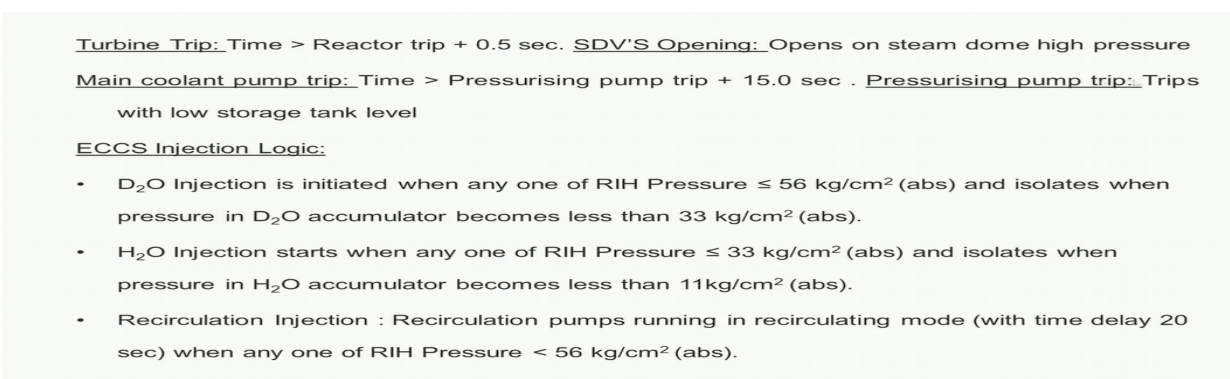
(Refer Slide: 24:46)



The power of the reactor initially due to the positive white coefficient in the channel be always considered the worst case. It has gone to about 110%; it can trip the reactor. Also, the log rate, the rate of power rise is also measured. Then high pressure which will anyway going to happen, then low pressure also in case it goes below, also we have a trip.

Then these are other trips which can come but may not be for this event. Then low flow of the primary less than 45% also could cost it. But you have enough number of signals which can trip the reactor.

(Refer Slide: 25:48)

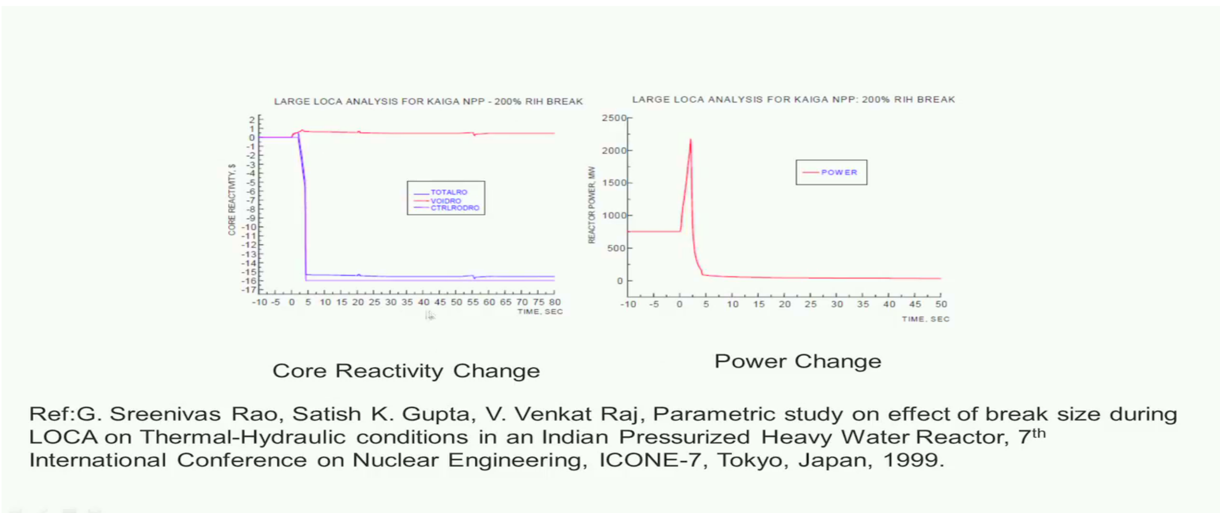


Now the turbine trip once the reactor trips, plus it takes some time for the valves to close and also the once the turbine is tripped, the dump valves open, safety dump valves open on steam high pressure.

The main pressure pump trip maybe after about some time, then the pump itself, the pressurizing pump tripping at low level once the low level goes in the storage tank then the other pump also, the high-pressure injection pump also will trip. So how the injection logic goes? High water injection is initiated as I said when any of the inlet header pressures goes less than 56 kg per centimeter square or 5.6 megapascals and once it comes down to less than 3.3, it isolates because this pump cannot pump in; it will cavitate.

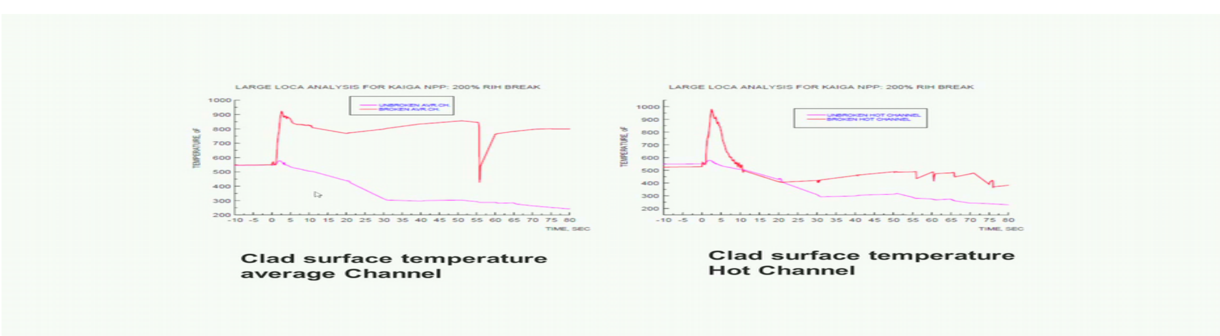
Then the light water takes over, light water injection and this starts when it is less than 3.3 megapascals and once the pressure has come down to 1.1, this also is isolated because these pumps won't be able to operate. Then we have the recirculation pumps which can operate at still lower pressure and these try to operate. In fact, they operate when the inlet header pressure comes down to less than 5.6 and because both are light water, we have no difficulty. So we do use both but even if this alone is there, it is able to cool the reactor and keep the temperature within limits.

(Refer Slide: 27:44)



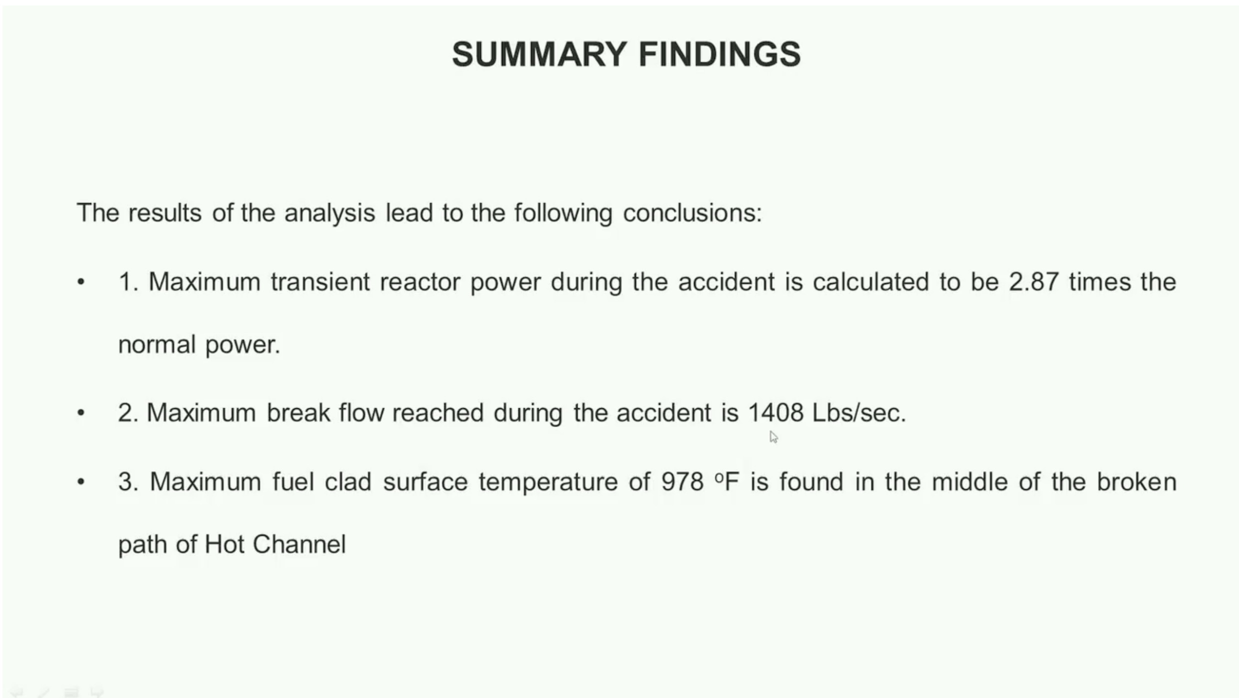
This is how the analysis goes for the reactor, what is the void reactivity, the total reactivity, how the power increases slightly and then comes back and this analysis is presented from a paper on this reactor which was presented by the ICONE-7 conference in Tokyo, you can see this for more details.

(Refer Slide: 28:16)



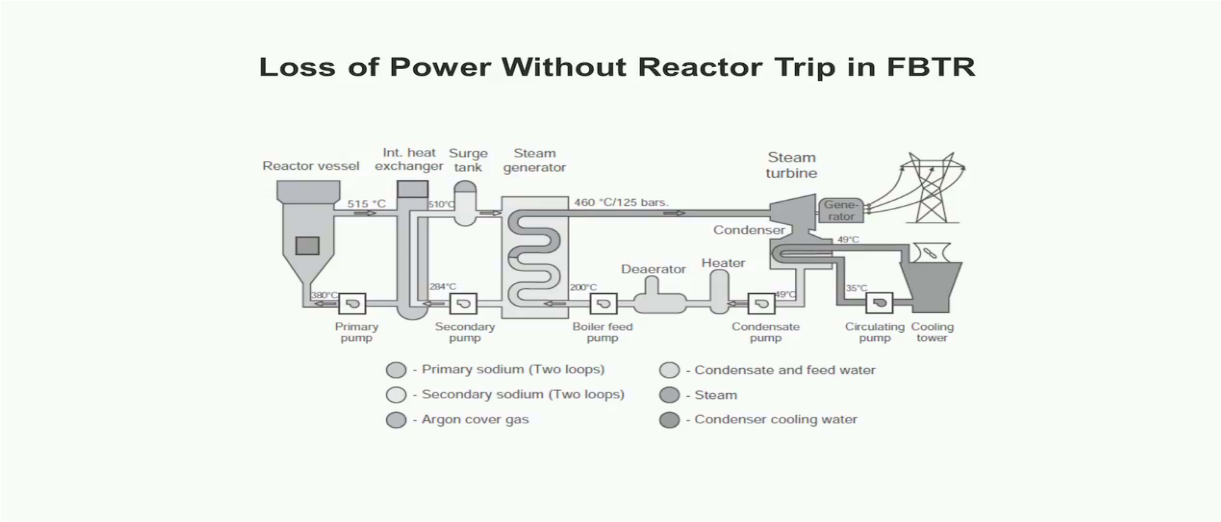
This is a clad surface temperature; we look at the broken channel and the unbroken channel both, we look at how the temperature changes. Same thing on the clad surface temperature on the broken hot channel how the temperatures go on the other channels how the changes, you see broken channel the clad temperatures are increasing. So what is the finding of this conclusion? Of course, this is accident after this has happened, you are not going to operate the reactor. At this stage, our thing is to see that we have to mitigate the consequences, no reactivity should come about; it is quite a tough accident.

(Refer Slide: 29:15)



So what we find the transient reactor power is about 2.8 times the normal power and what is the maximum break flow reached and the maximum clad temperature is this. So our assurance that the clad temperature and fuel temperature, clad temperature has gone but still they are within the limits at least as far as the calculations go. One thing I can tell you, you might find no measured temperatures here because we cannot make an accident and get the temperatures. But we do have enough conservative margins in all and these margins are such that our predictions are reasonably good and as I mentioned again, we still have the containment; should something fail still the containment is there to protect us which was there in the case of a PWR, the Three Mile Island which had a containment and the case of Chernobyl because containment was not there, there was already is outside.

(Refer Slide: 30:28)



So mind you, we are safe under the present designs of all nuclear reactors. See, I mentioned about diversity in all; diversity in signals, redundancy also we taught, redundancy of course is there. But the earlier reactors, for example, our fast breeder test reactor, we have only one shutdown system which is fixed control rods which will drop into the core. And loss of offsite power, loss of power is one common event. One advantage of this reactor is, it is a small reactor, all the temperature coefficients are negative. So if temperature increases, power will come down; there is no worry at all.

But then the main thing remains is how to remove the decayed. Nevertheless, we have many reactors abroad also which are small reactors with negative reactivity and we have the Rapsodie reactor which is similar to FBTR; the only difference is Rapsodie reactor doesn't have the steam generator; they have a sodium to air cooler which takes out the heat and pushes it out to the atmosphere. Then we had the Experimental Breeder Reactor II plant in USA; there also it is similar, we have a steam generator. In fact, we have a double valve steam generator; there is tube in tube and then they tests of having a loss of offsite power and not allowing the control rods to trip. It was a planned event so that what they want to see whether whatever we are feeling things are safe whether it will happen. Once you are planned event, something goes beyond you can always manually trip the reactor.

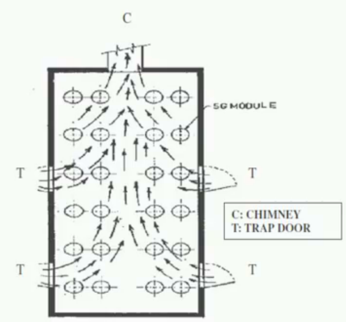
Believe it or not that it was found that the plant came down to a low temperature level and after about half-an-hour the operator could take over and it could maintain. Now, there was nothing, no anxiety, even though the control rods didn't trip. Basically decay heat removal was okay. So it showed, it gave confidence that the system designs are good and again I repeat, it applies for a small reactor. Same thing would not apply for a large reactor because some of the reactivity coefficients tend to be positive.

So there need to be a diverse shutdown system. So for PFBR, we have two systems; a main control and safety rod drive mechanism and a diverse safety rod drive mechanism.

Now, in the case of FBTR, we thought since we have a model already developed which I talk to you in the previous lecture, why not we try to see what sort of temperatures and parameters we get. I should not forget to mention, the Rapsodie reactor also tested 50% power whereas the EBR II did at 100% power, there was a difference and both reactor test showed that so once we have some available data and Rapsodie quite a close similar to FBTR, we thought why not we do an analysis.

(Refer Slide: 34:30)

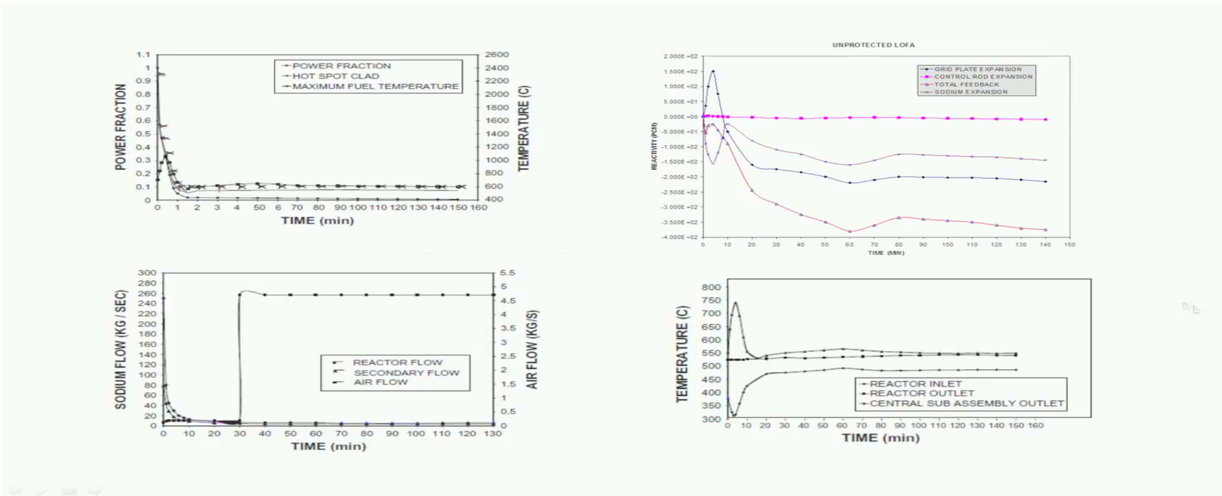
- Interest to study the response of FBTR to an event of loss of offsite and onsite power without reactor trip. This event is a low probability event as it requires simultaneous loss of offsite power and onsite power (station Diesels, Batteries) and failure of safety logic to trip the reactor. The analysis will indicate whether the design has the capability to set up natural convection for effective removal of heat, without the temperatures of clad/fuel reaching the limits. The heat sink is the steam generator casing which houses four modules of SG. Individual SGs are not insulated to allow heat removal by natural convection if required, by opening the trap doors.



So here, normally when you have a loss of offsite power, diesel should come up and mind you, we have two diesel generators; if one does not come online other will come online. Even if the single failure other thing will come. In case those two don’t come we have a battery backup for the primary pumps which can run them for about half to one hour at a low flow, all this. But the assumption made in this study is diesels are not coming, batteries are not taking over and reactor has not tripped.

So the only way as I mentioned to you of decay heat removal was the opening of the trap doors and mind you, we had already checked up what is the heat removal capability of this with sodium at 500 when we did the commissioning test of FBTR.

(Refer Slide: 35:21)



Then we thought now let us see how the temperatures are varying. The temperatures will come down, there is no control rod provided the overall reactivity is negative.

Now let us look at the power has failed, the secondary sodium flow is coming down, the primary flow is coming down, but the trap doors of the steam generators, they are open manually and if it is a manual action, we can’t take credit for it for half an hour. So there is another, for half an hour there is no heat removal through the steam generator. What happens?

You see, the secondary sodium flow rate comes down slower than the primary flow. So in the intermediate heat exchanger, if you look up, this flow would be more, this flow would be less. So apparently this inlet temperature which goes to a reactor comes down and that is what you see, the reactor inlet temperature going down. Of course, the reactor outlet temperature would go up because of the fact that the flow is coming down in the primary and of course, after some time it goes like this.

Now, this going down and at the inlet you have got the grid plate which supports. So the grid plate at a higher temperature is coming down. So there is an effect of the structures; higher the temperature it expands. When the temperature comes down, it contracts. So the grid plate contracts means the sub-assemblies coming somewhat closer that is actually means a positive reactivity.

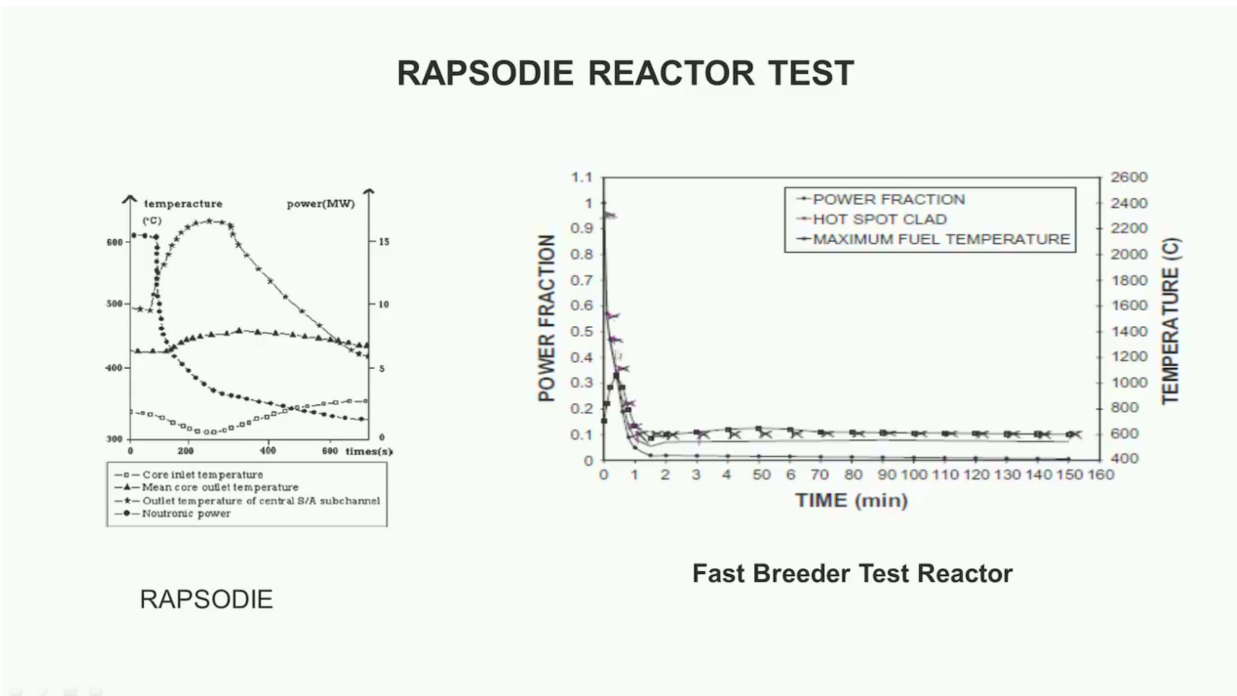
But after some time the effect of this higher temperature of the reactor outlet is felt at the inlet as you can see here. So this positive reactivity is only for a short duration and then it comes down. That means the grid plate is expanding due to high inlet temperature and negative reactivity is bringing down. All the other effects are negative. So the total feedback reactivity if you see here, this is, of course, the control rod expansion also we take which is very small. So the total feedback reactivity is negative and may be controlled by this.

So what happens? The power comes down, once the power comes down, your fuel temperature comes down, your clad temperature comes down, but mind you, initially the clad temperature goes up because coolant flow is not there and there is a slight increase in the temperatures and then this temperature falls down as the power has come down.

We saw that the overall reactivity is negative after some time so the power even though power has come down because of putting the control rods into the even -- it has come down just based on the temperatures alone because there is a negative reactivity. However, the hotspot clad temperature does show a rise because the coolant flow to the assemblies are reduced. Then nevertheless after some time as the power is coming down, the clad temperature also comes down, the fuel temperature is also coming down to a low level which is a safe level.

And now you look at here, 30 minutes, I said, only we opened the trapdoors. So the air cooling starts after 30 minutes. Immediately it picks up to a good level of over 4.5 kg per second. Here, what you see even though it is opened after half an hour and this is happening, the decay heat has come down because of power generation itself has come down. So this is a unique feature of small reactors and that is one reason why some of the proponents of fast reactors are talking about small modular reactors which can be built in large numbers instead of large fast reactors. This is a point to be kept in mind.

(Refer Slide: 40:41)



And just to compare with the Rapsodie reactor test you see how the temperature went up, how the power came down and how the mean core outlet temperature change and here you see all our things, it is on a different scale. So more or less increasing and coming down, those things are reflected. Only thing the Rapsodie results were done for 10 minutes the test, but whereas we wanted to see a long-term evolution, we really ran up to about 150 minutes and we found that things are still stable.

So this has no doubt given us a good feedback on fast reactor safety basically small reactors under event loss of offsite power, loss of offsite power, loss of onsite power, loss of batteries, even then we our reactor is able to shut down by itself.

(Refer Slide: 41:41)

SUMMARY

- This Lecture has given a flavour of safety analysis of some events in PWR, PHWR and Sodium Cooled Fast Reactors. The study of FBTR has shown how such reactors behave in a benign manner to a loss of flow event without reactor trip.

In summary, in this lecture, I have given you some flavor into the safety analysis of PWR, we looked at what happens when a pressurizer relief valve opens and doesn't close. Then we looked at a PSWR, how a loss of coolant accident in a pressurized heavy water reactor goes. Then the third one, we looked at the fast reactors, some sort of, you can look at this called as a passive safety feature that without any intervention on the reactor, the reactor is shutting down by itself that is we are invoking in the design the features and now how these features have been brought about, by engineering. In fact, it depends on how the temperatures are increasing. If the flow had come down very fast, the temperatures would have increased very high.

So what we have done in the case of our fast reactor. FBTR, we have provided enough inertia on the flywheel, inertia on the drives so that it would come slowly, the temperatures would increase slowly, the negative reactivity effects will come also slowly, everything happens slowly but steadily, then we are assured of a shutdown.

So, engineering safety of fast reactors is possible. Yes, if you ask me for a 1200 megawatt electrical fast reactor, SUPER PHENIX, they did a study to find out what the inertia should be provided so that the reactor is safe for again this similar event of loss of power. They found that it requires a really high energy because of large reactor and the flywheel itself is something like a two meter diameter flywheel and believe it or not they do have it in the reactor, they built it into the reactor.

So it is possible that we could engineer the safety in such a way that we can have safe fast reactors. There's uniqueness again I repeat of fast reactors. So it is very important that we do all these analysis and we have the capability to do this analysis which gives us confidence that our reactors are safe and under no circumstance we release radioactivity into the environment, not only to the environment, to the people who are working on the site, who are the occupational workers, everybody should be safe.

We need electricity so we need nuclear power, we have to make nuclear power safe that is the reason. Thank you.

Online Video Editing /Post Production

K.R.Mahendra Babu
Soju Francis
S.Pradeepa
S.Subash

Camera

Selvam
Robert Joseph
Karthikeyan
Ram Kumar
Ramganesh
Sathiaraj

Studio Assistants

Krishankumar
Linuselvan
Saranraj

Animations

Anushree Santhosh
Pradeep Valan .S.L

NPTEL Web &Faculty Assistance Team

Allen Jacob Dinesh
Bharathi Balaji
Deepa Venkatraman
Dianis Bertin
Gayathri
Gurumoorthi
Jason Prasad
Jayanthi
Kamala Ramakrishnan
Lakshmi Priya
Malarvizhi
Manikandasivam
Mohana Sundari

Muthu Kumaran
Naveen Kumar
Palani
Salomi
Senthil
Sridharan
Suriyakumari

Administrative Assistant

Janakiraman.K.S

Video Producers

K.R. Ravindranath
Kannan Krishnamurty

IIT Madras Production

Funded By
Department of Higher Education
Ministry of Human Resource Development
Government of India

www.nptel.ac.in