Indian Institute of Technology Madras
Present

NPTEL
NATIONAL PROGRAMME ON TECHNOLOGY ENHANCED LEARNING

NUCLEAR REACTOR AND SAFETY
AN INTRODUCTORY COURSE
Module 08 Lecture 01
Validation and Dynamic Analysis

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Good morning, everybody! In the last few lectures, we touched upon the safety principles that we follow in citing a plant, designing a plant, commissioning a plant, operating a plant, and so on. We also gave a idea about the safety approaches which we do in all the above areas, but now in this talk, I am going to give you a flavor of how we assure a good design that is a quality assured in the design, how is it we make a robust design which is able to predict very well what is going to happen in the plant. Unless we have that predictive capability to know what will happen, how then can we take precautions? So a modeling of the plant which gives you all the variations in the different parameters is a must.

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So today I am going to give you some glimpses of how we validated the dynamic model related to the fast breeder test reactor which went critical in 1985 at Kalpakkam. One advantage was that we had a collaboration with France for the design of FBTR which was similar to the Rapsodie reactor in France. So we did have access to the results of their commissioning tests.

So as a first step what we did, we made the mathematical model of the different components, we developed the solution techniques, then we put it into a computer code called as DYNAM code. Here it was required the background knowledge of differential equations, good solutions, and good modeling. This was just a one-dimensional model

So based on this model, we first try to see what happens when the power fails to the primary pumps that is power has failed so the pump will coast down due to the inertia.

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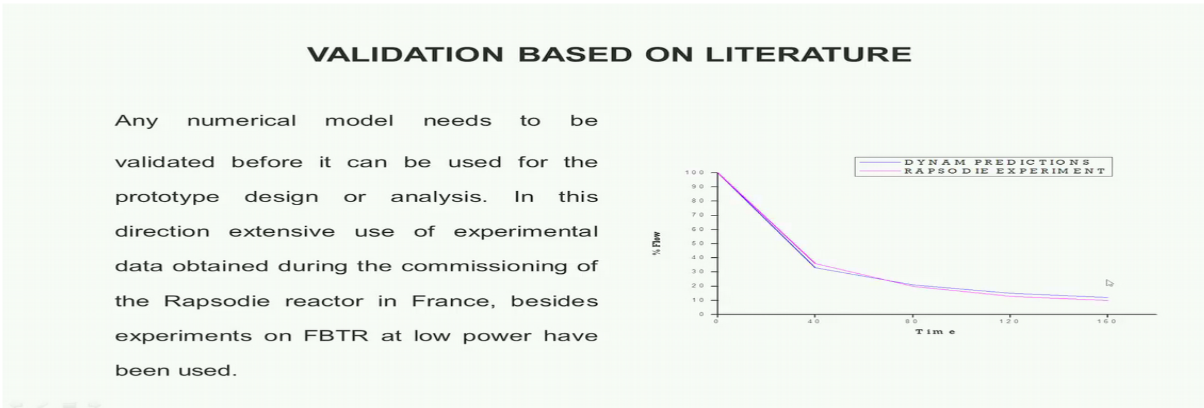

And here you can see, this is a flow versus time graph and the pink line represents what was predicted in the Rapsodie experiments and the blue one shows what we got in our calculations. Of course, this was one of the first few validations that we did of our code because flow is a very important thing. Depending on the flow change your temperature change will be good. Flow is a very important aspect. So that is why we thought we should just check whether the flow determination is accurate that means our hydraulic modeling, our modeling of the hydraulic system including the pressure drops in the core ,etcetera, the levels, everything is reasonably quite accurate.

So this was the first validation we did. Then the question is, now this is a force flow; pump running on a force flow. Many times when the whole power is lost, let us say, the power which we are getting from the grid has failed what we called as the offsite power and normally your onsite power should start that is our diesel generator should start and supply the power to the pumps so that they can be run at low flows.

Bt then there is another situation wherein we presume that the diesel generators don't come up. Then in that case what will happen? The driving force will be basically the buoyancy forces created because of the temperature differences in the hot leg and the cold leg.

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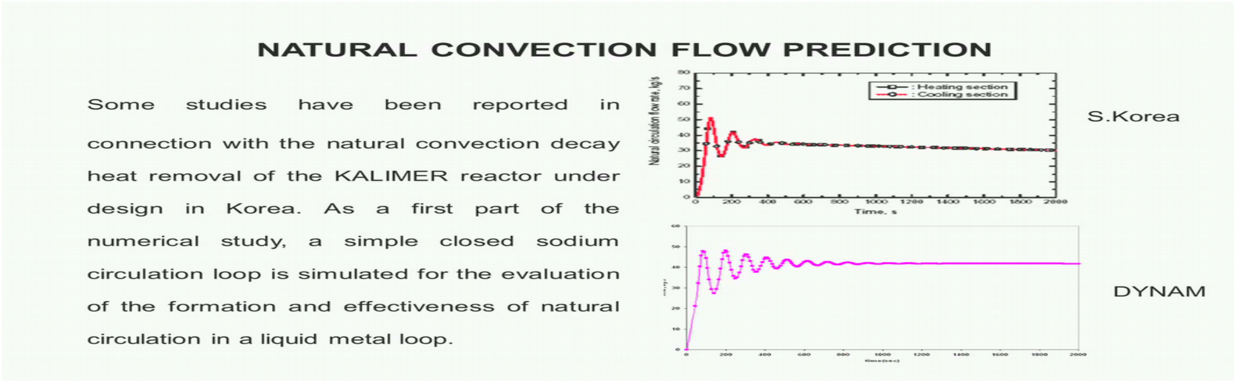

So here we came across the predictions for a small natural convection loop with regard to the KALIMER reactor in South Korea. So we did have the data of the loop in the literature. So we modeled it and tried to find out. So this is how their predictions are, goes up to about 50 kg per second and then comes down, goes up, goes down, and there are co-heating section and the cooling section; flows are nearly -- there is not much phase lag between them.

Here below you find. So here you see in the beginning stages it is quite close but in the later stages you find that the flow predictions are not similar. This put us to investigate this and we realized that the pressure drop coefficients when the flows are very low will be different than what they will be when it is high. So we understood that we need to put a pressure drop correlation which is in tune with the Reynolds number at that. So this was one input which we got and we improved this model.

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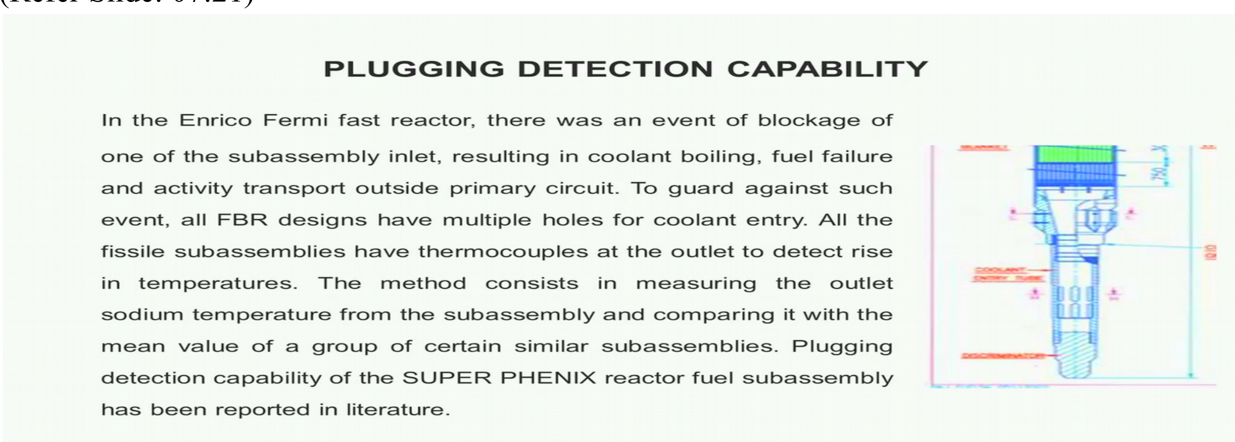

Next, you know whenever we are looking at the dynamics, we are most worried about the core and here we talk something like a plugging detection capability. Why? One of the first few reactors which was built, fast reactors which was built in USA called as the Enrico Fermi fast reactor. I have talked about this reactor in my earlier lectures where I brought out the incidents in different reactors, but recapitulate in brief what happened. They had the fuel assembly, there was just one entrance coolant was entering from the bottom and unfortunately, a plate which had detached from somewhere had come and blocked, maybe not fully.

So flow in the assembly was not there, very minimal, and the flow was not there, the sodium started boiling and that sodium boiling gave rise to your negative reactivity, but apparently, there Was still some flow, not totally plugged. So the question was, there was a little bit and the power was, reactivity came down, the operator adjusted it. Again it went down slightly, operator adjusted it. Finally, the fuel had melted and the reactivity was known through the cover gas argon circuit, increasing the activity.

Now after that reactor one thing was sure, we can't say that any hole will not be blocked or plugged. So in all our reactors subsequent to the Enrico Fermi reactors, our reactors means the whole world reactors, the sub-assembly gets not only from the bottom hole, it also gets from the side holes. There are multiple holes for the entry of the coolant and not only that, at the outlet of the sub-assembly, all the fuel sub-assemblies, we have got thermocouples which monitor the temperature and in case, there is a temperature increase, they would be able to detect it and forewarn that that assembly is not getting enough flow.

So we thought that this must be established, of course, we didn't have much of the data of the Enrico Fermi reactor but then we had some data regarding the SUPER PHENIX reactor where they had to convince their safety authorities is what is the plugging detection capability of that design. So we thought why not we use that data.

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BLOCKAGE PREDICTION CAPABILITY

Reduction in Sodium Flow to Subassembly		Limiting Conditions	
		Clad hot spot 800 °C	Max. Sod. Temp. 900 °C
Max. step reduction in flow (%)	Calculated	32.0 (37.5)*	71.0 (77.5)
	Reported for SPX-1	27.0 (30.0)	70.0
Max. rate of reduction in flow for 96% flow Reduction (% / s)	Calculated	5.0 (15.0)	16.5 (44.0)
	Reported for SPX-1	3.0 (13.75)	9.2

*The values given within brackets are for nil response time for measurement and scanning system.

So here, there are two things, one is a step reduction in flow and the other one is a slow reduction in flow. In both cases the objective was to calculate how much step flow it can sustain, reduction it can sustain if I were to reach a hot spot clad temperature of 800 degree centigrade limit.

Similarly, suppose I put a limit for maximum sodium temperature as 900 degree centigrade, what would be the maximum step reduction that the design can allow. So here you see what we calculated was about 32% reduction it would go to 800, in their case they reported it as 27.

You also see two another figures; 37 in the brackets. Now when we do these calculations, we also take into consideration the time constant of the thermocouples because there is a delay; you come to know only later. So that also needs to be considered. But suppose I take a zero time constant that is the immediate response then this shows in our case about 37.5 which showed us about 30.

For the other case of maximum sodium temperature where our prediction was about 71 their prediction was about 70%.
Coming to the other case here you see for a slow reduction of 96% flow, our rate was 5% per second was what it could accept to give 800 degree centigrade whereas they reported as about 3% per second. For the other case, 16.5 is what we got an end point. Now if you look up, this just tell that there are some differences because you don't know the exact; there could be some differences in the data which we have got because literature does not give you the complete data.

Nevertheless, you just see quite a good sort of agreement that the trends are good and we are not very far from what is happening. So this again gave us the confidence that we are able to predict the core conditions well.

Okay, now let us move from the core to the steam generator. Our steam generator is a once-through steam generator in which water enters as sub-cooled water as a liquid, it goes through the tube, picks up the heat from the sodium, and finally comes out as superheated steam. So we try to compare our design correlations which we have used. With that we try to design a steam generator for the PHENIX reactor.

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Steam Generator Heat Transfer Model

OUTPUT DATA	PHENIX-Serpentine		SNR300-Straight	
	Code	Design	Code	Design
Surface Area Sq.m	35.7	37.42	209.9	221.1
Pressure drop ,Bars	7.13	8.0	2.33	2.6
Economiser,m	25.14	23.76	9.52	10.0
Evaporator, m	27.20	31.89	8.89	9.4
Superheater, m	5.49	5.11	-	-
Total Length	57.83	60.76	18.41	19.4

Similarly, our SNR300 reactor of Germany, PHENIX is in France. This was operated right from

1974 and was recently in 2009, it was started decommissioning. It has got a serpentine type of steam generator as we have for FBTR. So we just did the calculations. Our core gave a length of about 57.83 and their lengths actual were about 60. 76. Here we felt maybe some margins they would have given in the design. So it gave us a feeling that if I give, add some, for example, just about three in 50, maybe 5-6% if I give I would be very close.

But then for a different design, a straight tube design which we were to go for PFBR when we compared we got a length of about 18.41 meters whereas actual design was 19.4. So apparently, it is clear that there have been some safety margins given and since this has operated and this has been validated based on test facilities in Germany, we felt to our calculations if I am able to add something like 5-6% I would be able to get a reasonably good design which I can confidently go ahead.

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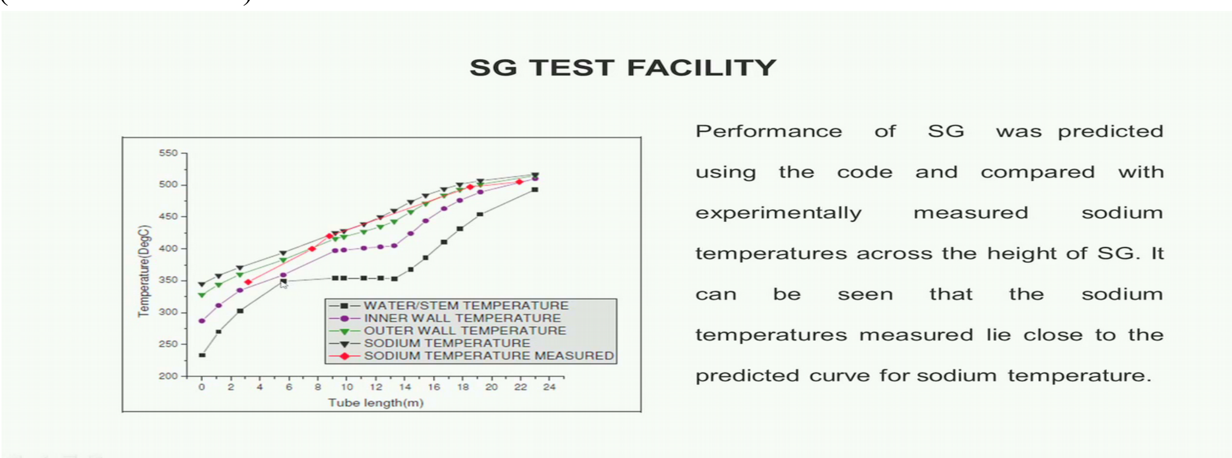

That is on the part of the steam generator area. Now, if I look at the heat transfer what sort of regimes are there. As I said water enters, it picks up the heat, becomes saturated liquid, then you have the nucleate boiling, then you have the filum boiling, then you have the superheat. For every region we apply different correlations depending on the conditions. So as we had set up a test facility for testing steam generators. So this is a five-megawatt steam generator test facility. So we instrumented the steam generator by providing thermocouples in all the steam generator tubes; the red one shows the measured temperatures that is what we could measure and the green ones, dark green ones show you what is the actual calculated temperatures.

So we found that the temperatures are reasonably close, thereby giving us a confidence that every region of predictions the heat transfer calculated is quite close to the reality, our own results, our own experimental results. So this also has given us the confidence.

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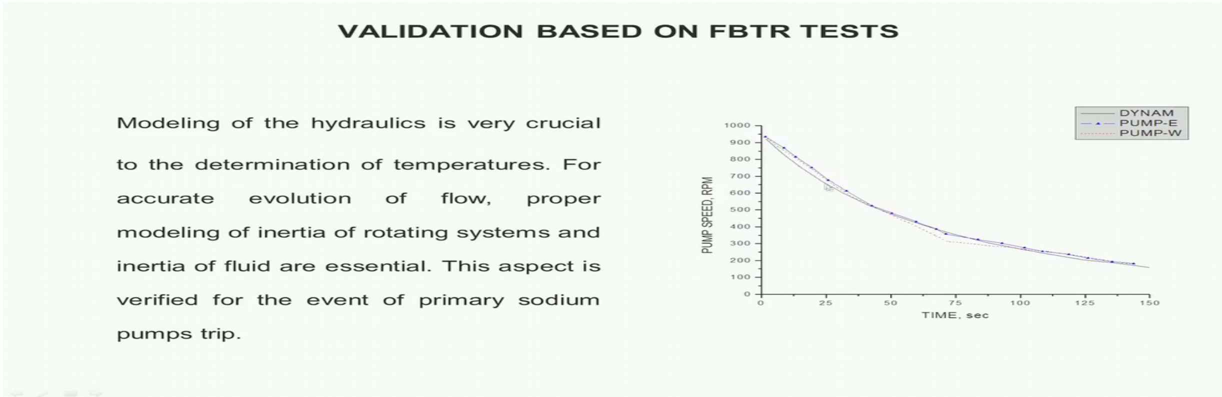

Then when we started commissioning FBTR reactor, earlier I mentioned to you we compared our results with what we had from France but now we started comparing results with what was there in FBTR. Now you see again the pump speed versus time, E means East loop, W means west loop and DYNAM is our core, you can see the predictions and the actual findings are quite close and that means as I mentioned earlier, there is a proper modeling of the inertia of the rotating systems, the inertia of the fluid, all are well. So this was again gave us a confidence that our modeling is good.

So at every level whenever we develop the confidence we know that we are close to the reality. Then validating the core model; even though we did validate it against some of the SUPER PHENIX test results for a sudden plugging, we just thought why not we do something about the measurement capability and see whether we are really getting.

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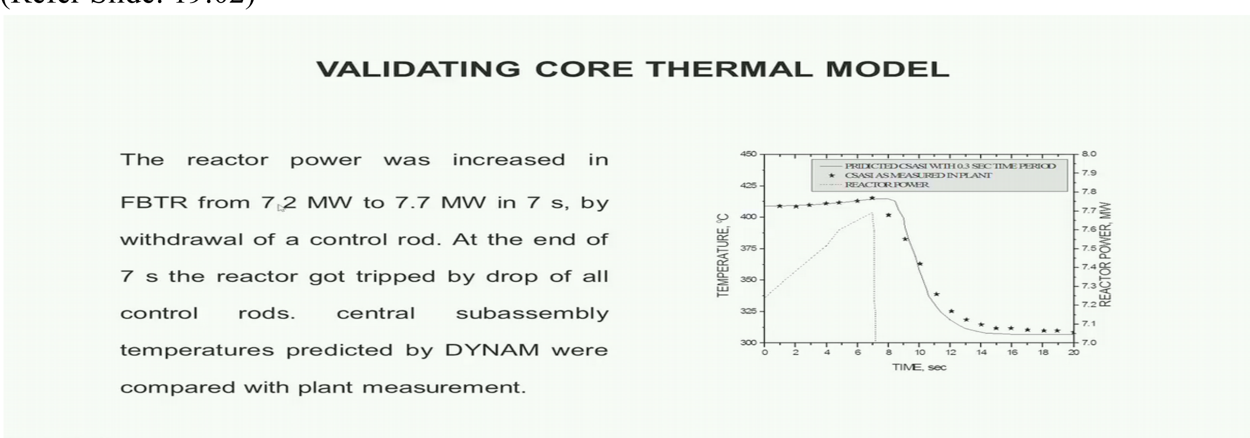

So what we did, we raised the power of the reactor from 7.2 megawatts to 7.7 megawatts in seven seconds by withdrawing the controller rod. At that time the reactor tipped. So this data we had and the central sub-assembly, the centermost sub-assembly, the monitoring of those temperatures were also recorded. So we said okay come on, let us now, why not we use our model to look at this transient event and let us predict. Now this star ones show you what is the measurement and the straight line shows you what was predicted and the predictions were with a 0.3 second time constant. See, the thermocouple time constant, if it is more, it will delay the actual relate to the actual conditions. Even that should be matched if you are interested in proper predictions. So we had done some experiments outside where we found that it could be something like 0.3 to 0.5 seconds time constant. So since majority of thermocouples had 0.3 we just used that and you can see it is a reasonable match between the two. That is our own plan measurements with our own computer code. So this was a good sort of happiness and confidence boosting for us.

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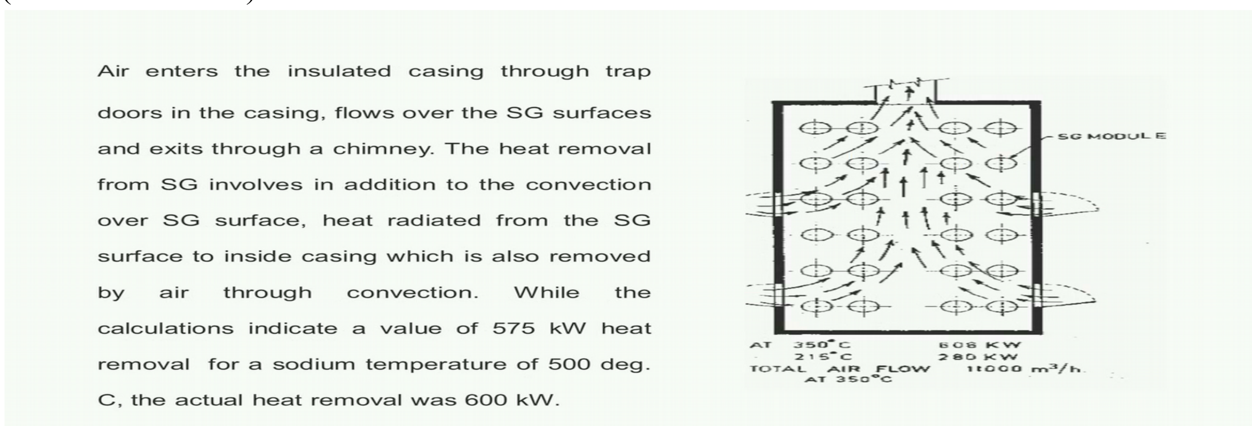

I mentioned to you something about natural convection that is when your offsite power is not there, your onset diesels don't start, what to do. So we looked at the natural convection pattern in the sodium. Now you know, in any reactor there is a decay heat which is in the core, even though you've shut down the reactor means the fission reaction has stopped, chain reaction is stopped but the fission products which have been produced in the fission reactions are still decaying and when they decay they produce heat.

If we don't remove that heat, it could go to heat up the fuel clad. If the clack could fail, the fuel could fail. So very important is that you must remove this decay heat and in the FBTR, fast breeder test reactor at Kalpakkam, we have the four steam generator modules put in a casing a casing, and there are four trapdoors; two in the middle and two at the bottom. They are closed and at the outlet, there is a chimney which goes out. The purpose of this is whenever there is a loss of offsite power and the diesel generators don't come up, you have to remove the decay heat, you just open these four trapdoors.

What happens, air comes in, flows over the steam generated tubes, you see this is one module, this is another module, this is the third module, this is the fourth module. It flows over the thing, picks up the heat by convection and then the heat goes out. Here there are two processes happening, heat transfer processes; one is this air flows over the shell of the steam generator which contains sodium and removes heat by convection. But this shell is at a high sodium temperature also radiates heat to the casing and whatever heat is radiated to the casing again is picked up by the air which is coming in.

Once the air picks up the heat it becomes light, it goes out, fresh air enters and there is a natural convection of the air setup and this is called as a natural convection cooling of the steam generator. So once the heat is removed in the steam generator, it sets up natural convection in the secondary sodium. When the secondary sodium natural convection has been set up, it sets up in the primary and thereby the core gets a continuous cooling and the heat is removed through the steam generator.

Here, our calculations indicated that for a sodium temperature of about 500 degree centigrade, we could remove about 575 kilowatts of heat whereas in the experiments, actual heat removed was the order of 608, very close. So again, on the safer side, conservative side, so this also gave us a good confidence that we are in a position to really find out natural convection in air also quite comfortably.

So you see we are trying to validate part by part. Then once integrated, it has to behave in the same manner. So this is called validation by parts.

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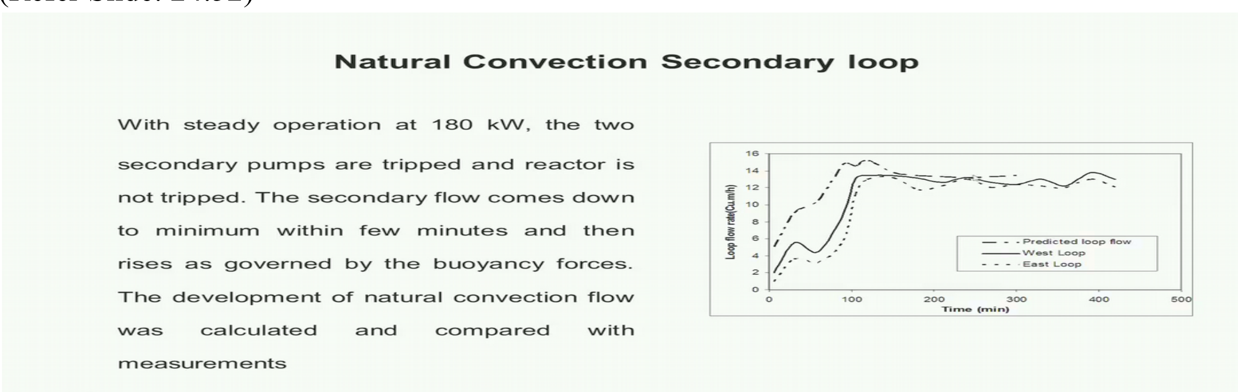

So coming to the natural convection in the secondary system, we trip the pump and we operated the reactor about 180 kilowatts and heat was getting removed through the losses in the pipe. The decay it was not much so we didn’t open the trap doors. So then we analyzed how the flows are coming in, the two secondary loops. Here you see what we predicted was something like this and what we got was something like this. East loop, West loop and what is our predictions. If you look at the numerical values, we got a maximum of about 12 meter cube per hour and our predictions were somewhere close to 15, but the steady state is coming very close.

We then looked at why this. We realized that the secondary loop we had treated this as a single pipe, but there are two tanks, a surge tank and the expansion tank and their level changes. So basically there is a difference in the scheme that instead of treating the whole pressure drop a single, we should treat it as in different parts and then maybe and when we did that, we became very closer to the realities, but still this prediction per se is not bad.

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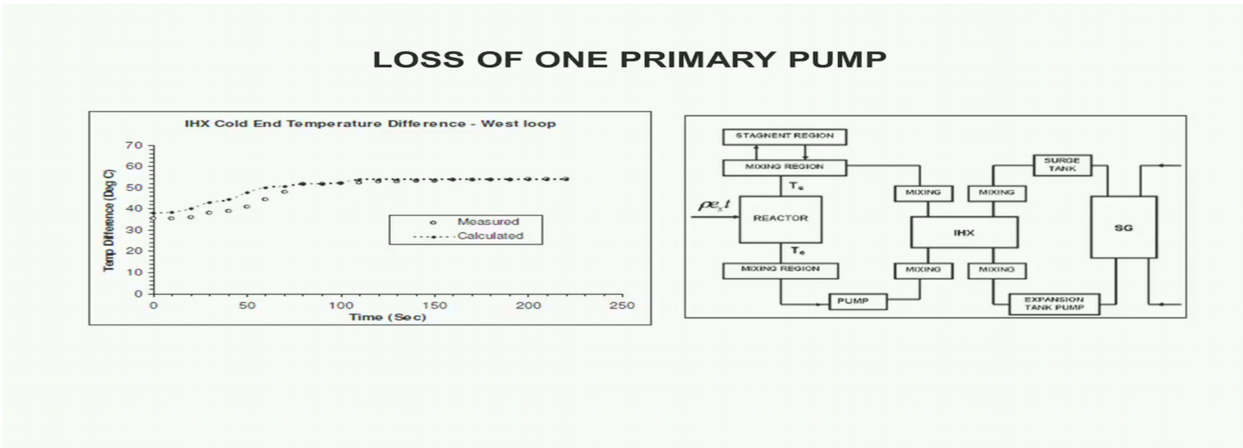

Of course, based on this confidence, we raised the power, we went to about 10 megawatts. I am sorry with these experiments we conducted about 8 to 8.5 five megawatts thermal. We tripped one primary pump and you see here it gives you how we modeled the reactor, there is a schematic. Then whatever comes at the outlet, we represented it by a mixing region and whatever is above the outlet pipe, we put it like a stagnant region which exchanges heat with this mixing region. And then this goes to IHX and whatever is not in the heat transfer region we treated it like a mixing, same thing at the outlet; IHX was represented by a thermal model. I am not talking about it here. Then again the pump then again in that mixing region.

On the secondary sodium side, outlet mixing, surge tank was treated as a mixing, then the steam generator, then the pump again and expansion time of mixing, then the inlet and this is a steam generator.

So here, when it is tripped, what happens, this primary pump is tripped. So that means primary pump which means heat coming in is going to be less but the heat removal capability remains same. So what happens, this ΔT, temperature difference between the two will start increasing. So this is how it happened. The measure the measured things are shown by this and this is the calculated temperatures and you can see here that there is a reasonable match between the predictions and the calculations, means predictions and the measurements.

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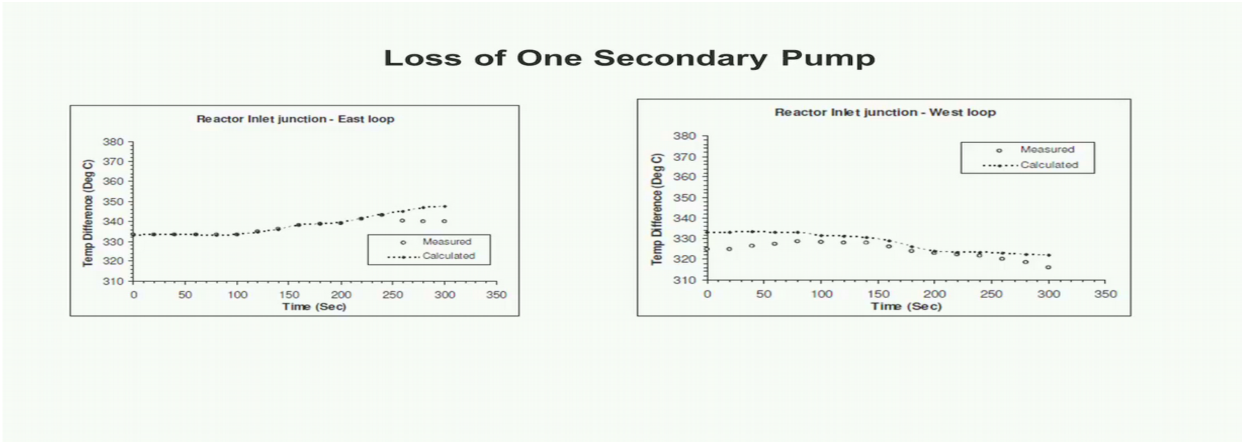

Then we did a tripping of the secondary pump. When the secondary pump is tripped, what happens, your heat removal comes down. When the heat removal comes down, your primary outlet temperature increases. That reflects on the reactor inlet temperature and that is how you see it is happening, but at later times, we see some difference. So this we reconciled later, we apparently found that the two loops were not being very identically, so we had needed to change the data which we have put into the computer code and then we could get a good match,

Similarly, East loop and West loop, here on the West loop what happens, there are not much change in the flow, there is no change; only the East loop was tripped. So here more or less, it is going with the same, does not change.

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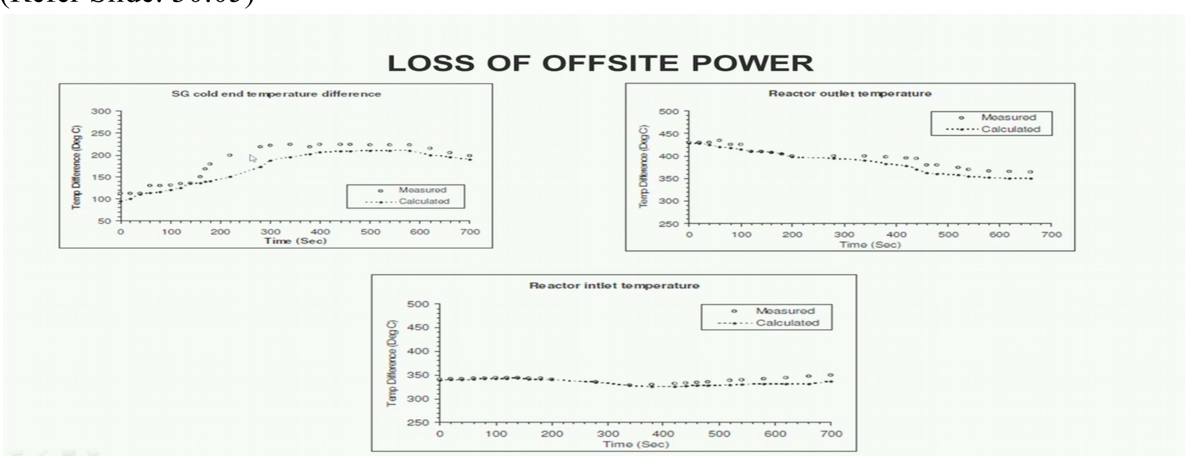

Then the loss of offsite power, there is no power coming from the grid, but your diesels are operating. Then what happens; this is the reactor inlet temperature, you can see, of course, how things, they are quite close. Reactor outlet temperature, they are not very bad, the quite close gives us a good sort of confidence that we are able to predict. At the steam generator cold end surely because the power water pump water supply is not there, there is increase in the temperatures, how it goes.

Here also you find the trend is okay, the final temperatures are okay in between. This apparently we attributed to the process modeling, there's a bit difference, but if you take the overall ΔT change which is important for our design as an input for the mechanical design, the overall change is important. So we found that this is able to do a good prediction.

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DETERMINATION OF SAFETY SET POINTS IN FBTR

- Safety limits are essential on different process parameters to protect the integrity of each of the barriers. The maximum fuel centerline temperature should not cross the fuel melting point(2594 deg.C) and the maximum hot spot clad temperature should not cross 700 deg. C. The automatic safety action should be such that it will trip the reactor before the safety limit is reached. Taking reactor power as an example, the safety limit (SL) would be that power at which fuel temperature has reached the melting point, while the Limiting safety system setting (LSSS) would be the power level at which scram is to be initiated. The settings need to consider the measurement and instrumentation uncertainties linked to the process variables.

Okay, now when we design a plant, we have to satisfy what are the requirements of the different components, what is the limits on the different components. If we take the fuel which is the most important, we need to see that fuel melting does not happen. In the case of FBTR, our melting point was something like 2594 degree centigrade and for the clad which is made of stainless steel, we should not cross 700 degree centigrade.

So in the case of any event happening, I should have a safety threshold such that under no conditions, this temperature should cross the melting point of fuel or the clad temperature should not cross 700 degree centigrade. What I mean the hotspot clad, considering all the uncertainties in the properties of the fuel, you have a maximum prediction, the conservation prediction should not cross.

Okay, so how I should set my limit. Now, let us say we put a temperature limit. We don't measure the fuel temperature, we do not measure the clad temperature, difficult to put thermocouples on the fuel and the assembly. So we use a surrogate variable, surrogate parameter and that is the temperature of sodium at the outlet of the reactor, means at the outlet of each sub assembly we have and we measure this and we have to correlate the fuel temperature to inlet temperature, outlet temperature and the different uncertainties with that we link. Under that condition we put a limiting safety system setting.

For example, safety limit is fuel should not melt. So 2594 degrees centigrade would be the thing, but our calculations will have uncertainty. So considering this, I would put that good margin at which my reactor should trip and once the reactor trips in the time when I generate the signal and by the time your control rod drops that could still be a rise. So these parameters I consider in my calculations and then set up what is called as the limiting safety system setting or the depth setting at which I must initiate the tripping of the reactor.

So as I mentioned, we need to consider measurement uncertainties, time response uncertainties. For example, let us say, I measure a temperature, it could be plus/minus five degrees, it is not error free. So I must consider that there could be a negative error. So I should consider that.

Similarly the time constant of the thermocouple, there is a variation, there is a band. If I say six seconds time constant, it could be six plus/minus two or three. We make these assessments before the reactor is commissioned by testing these thermocouples individually and establishing their time constants. So we know this data before we put them on the plant.

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It is essential that each conceivable event should have at least two safety parameters on which automatic trip could be ordered. Studies have indicated that the following design basis events govern the thresholds on the safety parameters of the core, viz..(a) Accidental withdrawal of one Control Rod, (b) Failure of one primary pump and (c) total power failure.

Incident	Parameter	Threshold	Hot Spot Clad Temp., deg.C	Max. Fuel Temp., deg C
Accidental Withdrawal of one control rod	Reactivity	10pcm	648.5	2321
	P/Po	1.1	660.0	2470
	Central S/A Temp.	10.0	668.0	2470
Failure of one Primary pump	Central S/A Temp.	10.0	680.0	2180
	Power /Flow	1.15	675.0	2200
Total Power Failure	Central S/A Temp.	10.0	712.0	2096
	Power/flow	1.15	675.0	2200

Now, very important factor, we talked about redundancy, we talked about diversity. So we don’t go by one thermocouple; we have two thermocouples for every sub-assembly, every fuel assembly outlet is measured by two thermocouples so that one of them would give a correct signal. Even if one fails, one would give us a signal.

Then there is another requirement of a diversity that is on a different principle you must be able to detect the same event. For example, I will take a total power failure in which I have my power supply is lost that is both. Then what will happen? My flow will come down. Once flow comes down, we have a process parameter called as power by flow which is calculated and we have a limit. Then central sub-assembly also gives me threshold. So we have got two diverse parameters which can trip the reactor before the melting point is crossed.

Here you find the hotspot clad crosses by about 10 degree centigrade. So here apparently we need to reduce the threshold, but then if you go to the other case of failure of one primary pump again power by flow and central sub-assembly temperature come into picture. Then we look at over power that is let us say, we are withdrawing a control rod to raise the power, but we didn’t stop, we continuously raised it. What happens? It is not a good safe situation so the reactor must be tripped.

So we find that reactivity is able to trip, overpower by 10% is able to trip and all these things are able to trip so here you have three signals. So essentially, we have convinced ourselves that we have diverse parameters for the different incidents, just to give you a flavor, we have unleashed all incidents which we have foreseen.

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Negative Reactivity Threshold

- The negative reactivity in a reactor always reduces the power and is in safe direction. In the initial design, reactivity trip was given a threshold of only +10 pcm. Detailed analysis of a large blockage was carried out based on the Enrico Fermi reactor incident, involving blocking of one of the fuel subassembly inlets by a plate, resulting in temperature rise in sodium. There was a negative reactivity resulting in small fall in power. This was compensated by the operator by rise of control rods. This continued and finally the subassembly subassembly melted and failed. Studies showed that temperatures at the exit region of fuel could be in boiling but downstream after the blanket and subassembly head, it might be much lower, due to the thermal capacities. Hence negative reactivity has to be considered as an input to safety logic.

So this again has given us a very good confidence. Then on the reactivity, I would like to make a comment, when it is a positive reactivity, only the power increases, but when there's a negative reactivity, anyway it is safe, the power will come down. So there could be a feeling that why if it is a negative reactivity why I should trip the reactor. Here again, I would point out to the incident which happened in the Enrico Fermi reactor where there was a rise in the temperature of sodium due to flow blockage, there is a plugging of the assembly at the inlet and the flow reduced and the temperature started increasing.

When the temperature started increasing there was a negative reactivity in that reactor and it was compensated by raising the control rod many two, three times or four times. But since there was no temperature measurement, there was no idea about what is the temperature, it was just one sub-assembly and all other sub-assemblies are all in proper shape. It melted and that activity came down. So we should not just like that say okay, negative reactivity we can do. Even any change from the normal critical condition, reactivity is $\Delta K/K$, how much it is away from criticality.

Positive side or negative side we must investigate so we must have a trip on the negative reactivity also and which we have put in FBTR in the safety logic so that even if there is a negative reactivity of about 10 PCM, there is a trip of the reactor.

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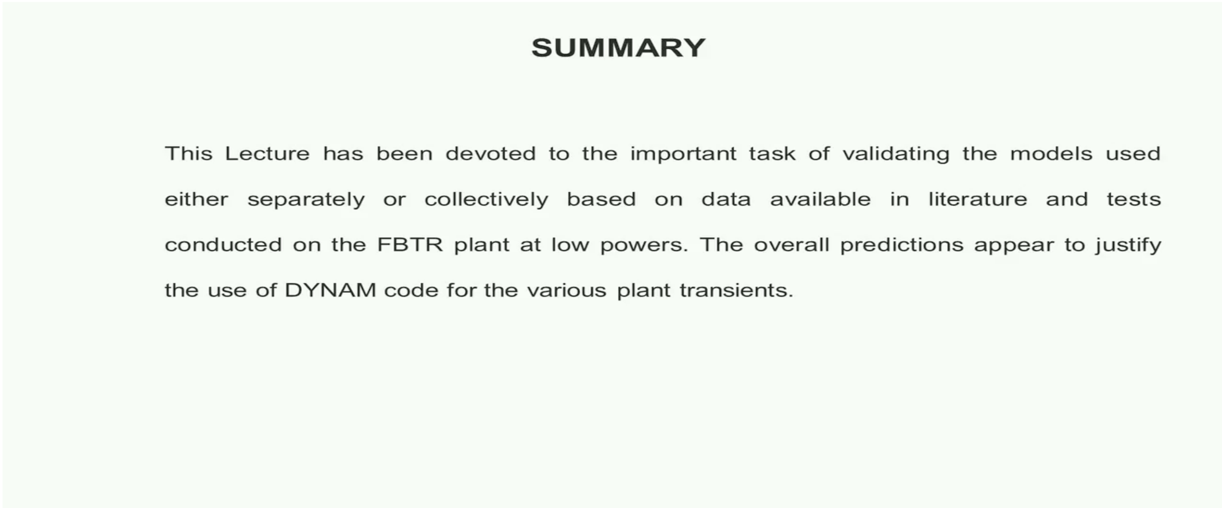

Now, I would like to summarize this lecture. We have looked at validating the different models in a separate that is validation in parts based on literature, based on tests conducted in other reactors, based on tests conducted in our own reactor at lower flows and the overall predictions appear to justify that this DYNAM code can be used with a high degree of confidence for assessing the plant transients. Thank you.

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