Indian Institute of Technology Madras

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NATIONAL PROGRAMME ON TECHNOLOGY ENHANCED LEARNING

NUCLEAR REACTOR AND SAFETY

AN INTRODUCTORY COURSE

Module 13 Lecture 02

Safety Regulation in India Cont...

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Good afternoon everybody. We shall continue our morning's lecture on the safety regulation in Indian nuclear power plants.

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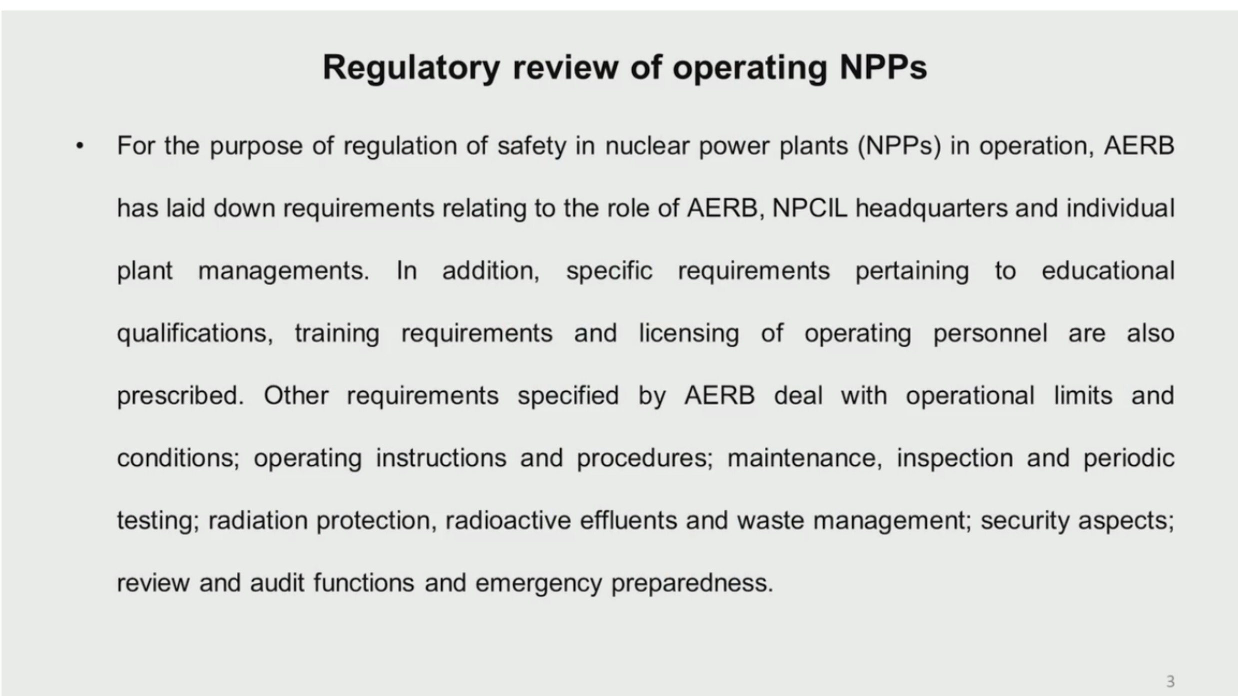

Here one of the most important jobs of the regulatory board is to have a regulatory review of all operating nuclear power plants. So here it is a clear-cut role of what NPCIL has to do, what is the regulatory authority has to do. Then first and foremost thing the utility which I call as NPCIL must have people who have the capability to do the jobs. Operation and other requirements they must meet some minimum educational standards. So here is what we call licensing of personnel. So the people who are recruited for operations are based on their educational qualifications, some are chosen as operators some are chosen for shift engineers work and we have the corresponding qualifications. But we give them a training on the different features of the plant. The training is different for an operator because his education qualification is different. The training is different for an engineer. So once they go through all these courses wherein they are taught what are the basic features of operation, what sort of operation limits which they need to see that it happens, even though automatism is provided. They should be aware that these limits should not be crossed and they are of course, given the operating instructions and manuals. Then as regards to the maintenance personnel the equipment maintenance, their inspection, and periodicity of inspection. Then your radiation protection equipment, waste management all these courses are given to them and then only they are licensed to operate.

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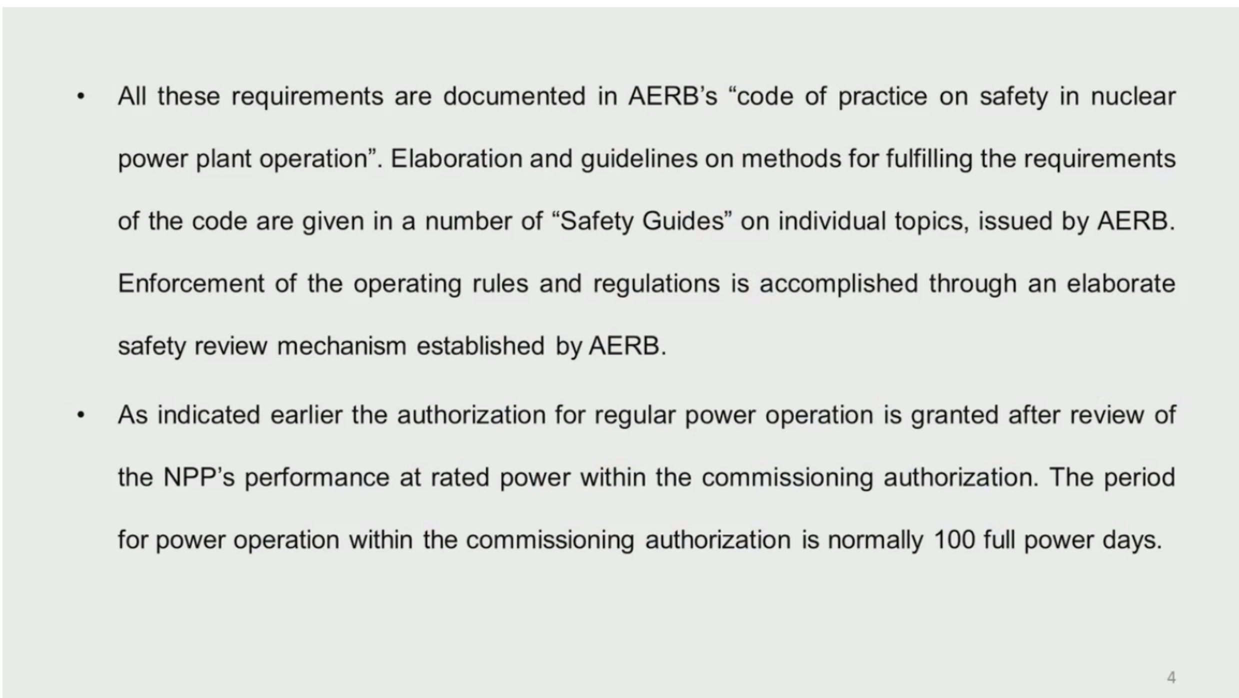

So where all these things are documented AERB has got a called as a code of practice on safety in nuclear power plant operation. So this elaborates and gives also the guidelines how to fulfill the requirements of the code. Now enforcing the operating rules is of course major responsibility lies with the utility the NPCIL but AERB does a review mechanism from time to time. So here coming to the plant proper when we give an authorization for plant operation, we do not give it for a continuous. We say after the commissioning authorization is there, you are commissioned. The period is normally about 100 full power days. So this 100 full power days is what you can do as a part of the commissioning because during that period you will get the data and there may be need to make some changes. So that should get reflected in the as-built or the final safety analysis report. So at the end of these 100 days you have understood the plant more completely. All these days it was under not a commercial operation.

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- While applying for authorization for regular operation, the applicant has to submit the Safety Analysis Report (Final) reflecting the as built design cleared by the AERB, detailed performance reports, status of and measures to resolve the pending issues (if any).
- Subsequent to grant of this authorization, the responsibility of safety review is transferred from the Advisory Committee on Project Safety Review (ACPSR) to Safety Review Committee for Operating Plants (SARCOP)/Operating Plant Safety Division (OPSD). This signifies the end of the project stage and the NPP formally comes under the purview of safety review mechanism for operating facilities. As per present regulations, the authorization for operation is valid only for a limited period and required to be renewed as per prescribed requirements.

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Now it goes for regular commercial operation. So the utility needs to submit the final safety analysis report reflecting the expected performance, the actual performance, by the plant, and where deviations are there and how issues were resolved. Suppose some issues are not resolved but they are not very important to safety. They also need to be brought out. Not only that, you need to give a time schedule in which these other issues will be resolved. Only then we get the authorization for regular operation.

Now as I mentioned earlier, once this regular power operation has started, the Advisory Committee on Project Safety Review does not come in. it is out and the SARCOP enters. So basically the project stage is over.

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- As indicated earlier, authorization for operation of a plant is issued by AERB for a specified period. During this period, the operational nuclear power plant is continuously under regulatory review. For reviews within the authorization period, the following elements are covered:

  1. Review of periodic reports submitted by the plant as per reporting criteria specified in the authorization for operation.
  2. Review of off-normal occurrences of safety significance.
  3. Radiological safety status.

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Now it is the regular operation. Now again this regular operation is given again for a prescribed limited period and then renewed subsequently. Now during this period after the authorization has been given surely the plant is under regular review. So for to be able to review the plant has to document the operation history. It has to send periodic reports to the AERB means basically the SARCOP specifying what are the observed things, what are the deviations observed during the operation and in case some off-normal occurrences have happened, what do you say the off-normal means not the expected occurrences it does not mean something bad. It does not mean unsafe. What you do not expect is off-normal event and do they have any safety significance. And of course, the radiological safety status thereby what is the activity in the containment building, what is the activity in the different operating areas. So what is the what you call in the stack release activity; all these data need to be submitted to the DAE, to the regulatory board that is SARCOP from time to time.

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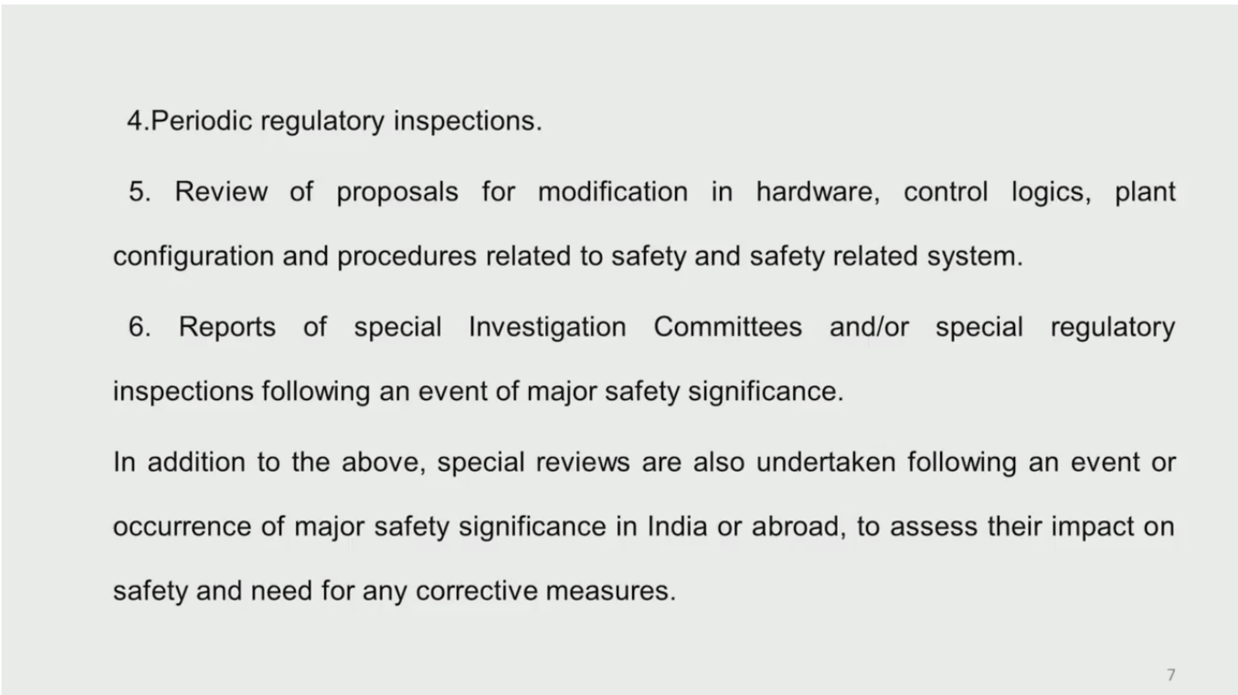

Then the periodic regulatory inspection which is done by the AERB. Mind you these inspections not only purport to the nuclear power plant per se, it goes also to the reprocessing facilities. It also goes to the R&D laboratories. Everywhere our safety implementation has no sort of a second level. It is everything we try to maintain at as highest level as possible. Here what we do all the measures which have been done one is regular operation, one could be maintenance. During maintenance some changes might have been done. Then surveillance. Surveillance means you have to go and inspect that equipment say every 15 days or every one month. All these are written down and when you inspect them you fill in okay when you have inspected and what is the health reports. So all these reports when the AERB team comes, regulatory inspection team comes, it will go through and see whether you have followed. Then if there is any safety significance of course AERB will go further deep into that and then suggest the corrective measures.

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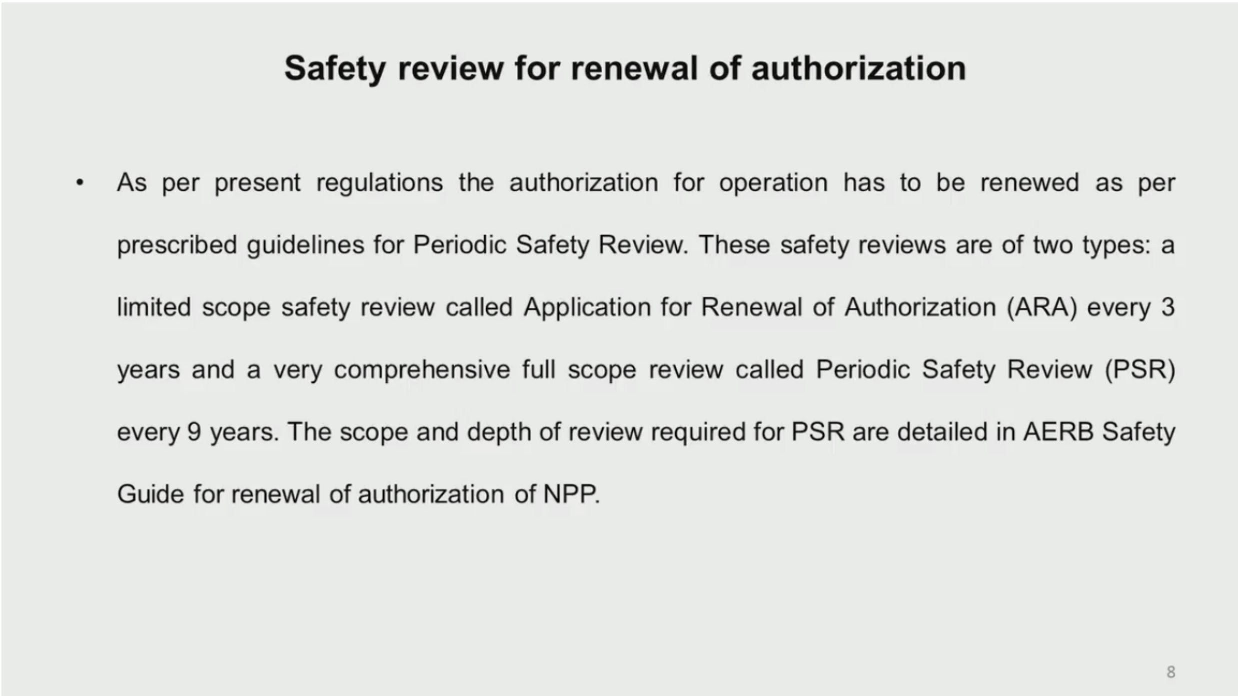

Now let us say that the first authorization for a regular operation was given and then periodic safety review has been conducted. There are two type of reviews besides a regular review. Periodic safety review then I am sorry, application for renewal of authorization every three years. So you have to apply for authorization every three years and then of course the next level is a very comprehensive and safety review every nine years. so all these requirements are elaborated in the safety guides. Now license renewal this license renewal pertains to a nuclear power plant which has supposed to have completed its life. Normally, the life of a plant we design for 30 years of full power operation. Mind you it is not time alone. It is also the power at which you operate. So then only the equipment's how much they are stressed that is a clear reflection of equivalent to a full power days, full power years. For example, the Tarapur reactor you can take. It was built in the year 1964. So after completion of about 25 years or so a thorough review was taken. The data on the plant, the number of cycles, temperature cycles the equipment has been subjected to, all these data however made available and then based on that the AERB gave clearance for subsequent operation. Similarly if you take a ABTR reactor it was made critical at Kalpakkam in 1985. A review has been conducted to look and then extension has been given for the operation. So why why it is, when you say a design has been licensed, why at all there should be again another steady required? It is looks simple but very important that during this period of 30 years some of the design rules may have changed. Of course, may be the earlier rules were very conservative based on which we estimated the life to be last only 25 to 30 years but today more knowledge is available about the material performance, material characteristics, material response, to temperatures and pressures. Our knowledge base is more. So this when this review is done with the latest data we may be able to extend the life plus should there be a safety improvement which could not be implemented in a regular way it can be implemented. So here this sort of reviews are very useful.

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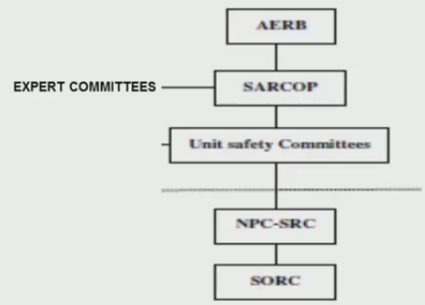
- All the above reviews are conducted through a multi-tier review mechanism. The first review is by the Station Operations Review Committee (SORC) followed by a review by a Safety Review Committee at the NPCIL's Headquarters (NPC-SRC) where design and quality assurance experts are also involved.
- In AERB, the review is done first by a plant specific Unit Safety Committee (USC) and thereafter by the Apex Committee known as Safety Review Committee for Operating Plants (SARCOP). A quarterly report to safety status of the Department of Atomic Energy units is made by SARCOP and is discussed in AERB board meetings.



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How is the review conducted? As we have seen it is a multi-tire review. All the plant operations are actually reviewed by a Station Operation Review Committee. we normally use the word SORC acronym, SORC. We never use the full word. We say SORC. So every station has got a SORC. That Kalpakkam you have got an SORC for the heavy water reactors because the commander MAPS. We have got SORC fast breeder test reactor plant. Everywhere there is a Station Operational Review Committee. Then in the case from here at the utility headquarters that is NPCIL headquarters we have this safety review committee. The SORC will give the operation inputs and the nuclear power corporation will draw the designers and the quality assurance experts also and review the whole operation and findings. Having done this it goes to AERB. So above this line is AERB. In AERB every unit has got a safety committee which is looking after the review of that plant so that the people are familiar with what has been happening in the plant. They are familiar with the history of the plant. So it is easy to make a good judgment for them to give the clearances. So the unit safety committee does this job. Having done this the unit safety committee gives the recommendations to SARCOP which is the Safety Review Committee for Operating Plants and SARCOP mind you, it will go through the things. Should it have some clarifications or it feels there is need to look into the matter in more depth it appoints expert committees. Many times this expert committees consists of people from outside the Department of Atomic Energy and from different units so that you have the expertise or the wisdom of all getting into resolution of the plants. Now SARCOP also does a quarterly progress report of all the plants and sends it to the Atomic Energy Regulatory Board which meets every quarter and discusses these reports so that it is satisfied that the all the plants are operating safely. So the AERB that way in its newsletter it gives what is happening to the different plants, what has happened, and it is quite open and transparent and available to all the public in that website. Website is www.aerb.gov.in.

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Regulatory inspection during operation

- During operation stage, regulatory inspections are carried out twice in a year to assess and verify compliance to the regulatory requirements. Besides the routine regulatory inspections, AERB also conducts special regulatory inspections with specific objectives as deemed necessary. Such inspections may be taken up after occurrence of major safety related incident or after major modifications to the plant. Authorization for resumption of operation depends upon such inspections. During inspection, the compliance of the regulatory requirements is checked by review of records and other documents maintained by NPPs.

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Regulatory inspection during operation. What we talked earlier was the renewal authorization. That is periodic safety review and application for review for renewal but here it is during operation. So here during the operation twice a year normally the units are visited. The plants are visited. We do have industrial safety experts also there besides the experts on the nuclear area, and all of them go through all the documents, all the things. They also question. They practically do the audit. The utility does everything but they do the audit whether the utility is doing as per the procedure. So sometimes when during these inspections if the regulation team find, regulatory team finds that things are not going fine it can recommend to AERB to stop the operation of the plant. So basically this regulatory inspection is to see compliance of all the regulatory requirements by the plant or the utility through review of records and other documents maintained by the utility. So that is what is the during operation.

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In particular following aspects are covered:

- Audit of operation, maintenance and quality assurance programmes.
- Adherence to the technical specifications and other licensing documents.
- Compliance to various regulatory recommendations.
- Adequacy of licensed staff at NPPs.
- Health of safety systems and safety related systems.
- Radiation safety and ALARA practices.
- Emergency preparedness.

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If you say what are the list of items in brief. Audit of operation. Maintenance and quality assurance programs. When I say maintenance I told what is the frequency of maintenance, should there, what is the preventive maintenance schedule, what is the schedule of surveillance; all those things will be there. So whether it has been done as that. Then whether the plant is being operated as per the technical specifications, whether they have not crossed the limits. If they crossed the limits have they reported to the AERB, and taken clearance or not. Then, whether the licensed staff is available in all shifts in the different areas of the plant, in control room, the local centers, whether they are available. Again this is through a documentary check. Then what about the health physics people? Are they taking data on the activity and radiation doses received by the people? Are they maintaining the records of the pocket dosimeters or the TLD badges? Whether all the theta courts are being done. Whether these badges are being sent for counting? All sorts of these how are they being followed is seen through audited by through the documents. Then what about the emergency preparedness excise, whether any drills have been conducted by the plant. Normally once in a year we are supposed to have a drill to see that all the what we call establishments in and around the plant are ready to carry out the actions needed during an emergency. So it is be something like you know training the people so whether it is being done as per the regulatory procedure.

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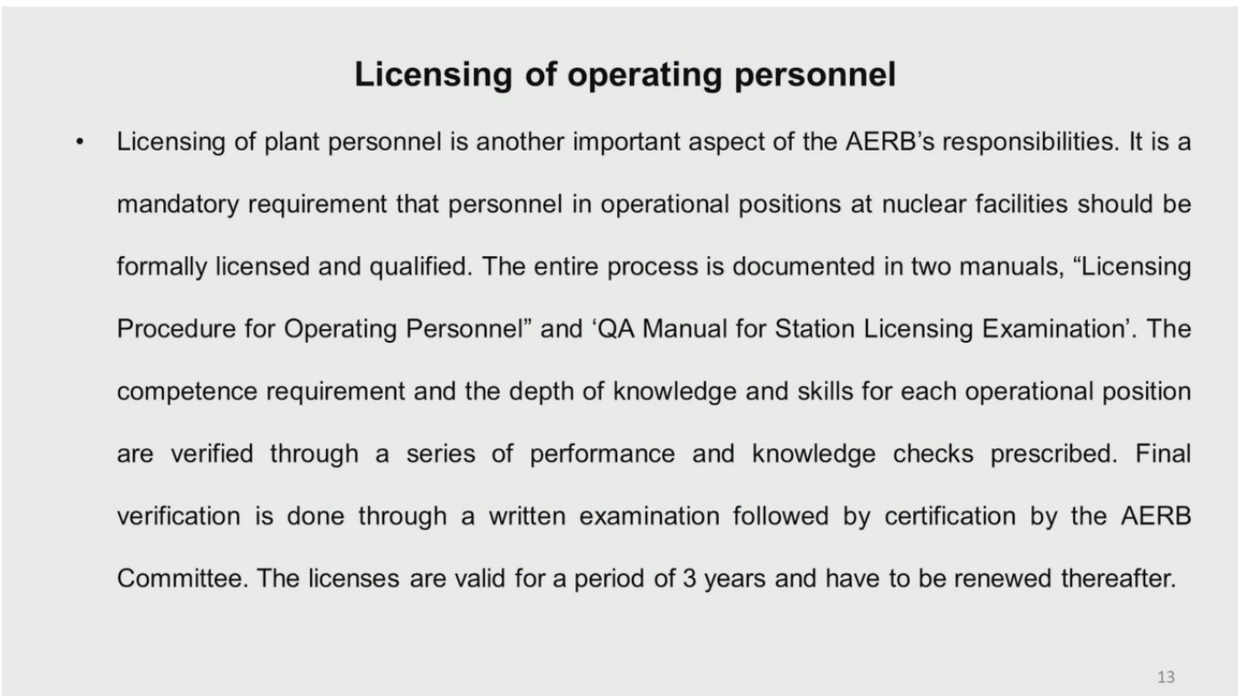

So that is one step. Then licensing of the operating personal. As I mentioned it is a very mandatory requirement that personnel in different facilities should be licensed and qualified. We have a manual for this called as the licensing procedure for operating personnel. And also a QA manual for Station Licensing Examination. Here the purpose is to check their competence, their depth of knowledge and any skills for operation positions. How fast he is able to think. No doubt all these are tested through a set of examinations, written examinations followed by interviews. One thing which I have forgotten to mention earlier today it is mandatory that all nuclear power plants have a power plant simulator. What is the power plant simulator?

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The power plant simulator is basically it is a replica of a control room but there is no plant. The plant is modeled and it is in the software but all the switches are there. So when you say pump start on the console, it will initiate the pump star and the flow will increase. All these things can be seen on the control room panels. This as I said it is a replica control room the operator is exposed to the same environment as he would be exposed to in case of a actual reactor. There is also a training console which is kept with a training supervisor and what is the role of the training supervisor, let us say the plant it should be started up. So the procedures are given. So the engineer, shift engineer starts the plant and whatever he does step-by-step is recorded and at any stage if the person is deviating from the procedure, the training supervisor will caution him, but here there is no real plant. It is only a simulation. So the operator gets a feedback through that. Similarly the training supervisor gets it also on his console. So this way the supervisor is able to guide. Then next step operator responds to different events. Let us say there is a power failure. There are a certain set of actions that he needs to take place, should be undertaken by him. So what the supervisor will do, he will have a software key by which he can simulate a power failure situation. So then the response of the shift engineer or the operator will be observed and then recorded, and then they will be assessed. So this way the training is made foolproof to the operator. Now earlier reactors that is not compulsory but of late last 10-15 years it is compulsory and we have operation training simulators. When you go to the simulator room you will feel as if you are in the control room of a actual plant. So these, after these examinations and interviews the licensers are given for a period of three years to the operating personnel and then they are renewed thereafter. Surely along with this there is a cash incentive for them by giving some special allowances.

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Safety upgradations in old plants

- All NPPs are designed and operated to meet the prescribed level of safety required by the standards and practices that existed at the time of their design. However, safety standards get revised from time to time based on operating experience, new developments in technology and improved understanding. Hence, it is necessary that all the operating plants are periodically assessed to demonstrate that required level of safety is maintained. Towards this end, AERB has stipulated Periodic Safety Reviews (PSR) for old operating NPPs. Several such reviews have been conducted by the AERB, for older plants that include RAPS, TAPS and MAPS.

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Now we talked about reviewing the safety after some 20 to 30 years of the plant life. We want to reassess whether the license can be given. But let us say during the operation of the plant sometimes there are changes needed due to some operating experience or some incident happening somewhere else. The best example is of Three Mile Island, pressurized water reactor. The moment the accident happened in the Three Mile Island, there was a review and as we saw earlier lot of changes in procedures was sought, operator training was needed to be enhanced. All such things are there and there was need for better instrumentation of the plants. So how does one go about so that means you have to do some safety upgradation. So how do we do that? So we have stipulated for all the old operating plants whenever any change happens or any incident happens we have a thorough review carried out by the utility submitted to AERB and we see whether there is any safety concern. If there is a safety concern we will do that improvement but if it is not a safety concern then we prescribe some procedures and continue to operate. Now this applies to basically three plants. The first two units at Rajasthan power station units one and two. Tarapur one and two. They are of the boiling water reactor, MAPS one and two. Subsequent to that more or less in the CANDU reactors there has not been much change.

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Rajasthan Unit-2

- It is an old generation pressurized heavy water reactor (PHWR)-200MWe, commissioned in 1981. In accordance with AERB requirements, a detailed review of Rajasthan Unit-2 was conducted and a plan for required upgrades and modifications was finalized. Rajasthan-2 was shut down in 1996 for en-masse replacement of all of its 306 coolant channels. The old coolant channels made of Zircaloy-2 pressure tubes were judged unsuitable for continued operation and were replaced by pressure tubes made of zirconium niobium alloy. A supplementary control room (SCR) was provided in a separate building to carry out important safety functions in case the main control room becomes uninhabitable due to a localized fire or damage caused by turbine missiles. An additional diesel generator of 600 kVA capacity has been provided at a high elevation.

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Now let us come to Rajasthan unit-2. We did a review for this reactor which was commissioned in 1981 and what upgrades and modifications needed. They were finalized by the utility and one of the important things compared the cause earlier they were using Zircaloy-2 for pressure tubes but based on the experience of swelling, bulging, etcetera. of these tubes we had developed Zirconium niobium alloy which was tested for a long time, tested means under the irradiation atmosphere in reactor, actual reactor, and this new alloy was formed to be suitable. In fact for that matter in the heavy water reactors which use thermal neutrons the flux levels are low but in the first reactor like FBTR the flux levels are irradiate and higher. So when we wanted to irradiate this and get its full lifetime irradiation data we irradiated the Zirconium niobium in fast breeder test reactor so that in a shorter time we could get the information of the effect of radiation and temperatures and we could come out with the Zirconium niobium alloy and these were needed to be replaced Zircloy-2 pressure tubes. So in 1996 normally every 10 to 15 years there is a need, earlier there was a need to change these pressure tubes so at that time we put Zirconium niobium alloy. One more thing we did at that time was to have the supplementary control room. Now you have a main control room where you have carry out all the important operations. But a need was felt that should there be a fire in the control room, many operations, practically you may not be able to do any operation. So most important operations which are required for safety were also wired to another control room called as the supplementary control room. Of course, today all the reactors need to have the main control room and a supplementary control room or sometimes it is also called as an emergency control room where most of the safety operations can be done from that room. And that is provided in another building so that there is not chance of a common cause effect. Also in the Rajasthan-2 we put additional diesel generators so that it could meet all the requirements.

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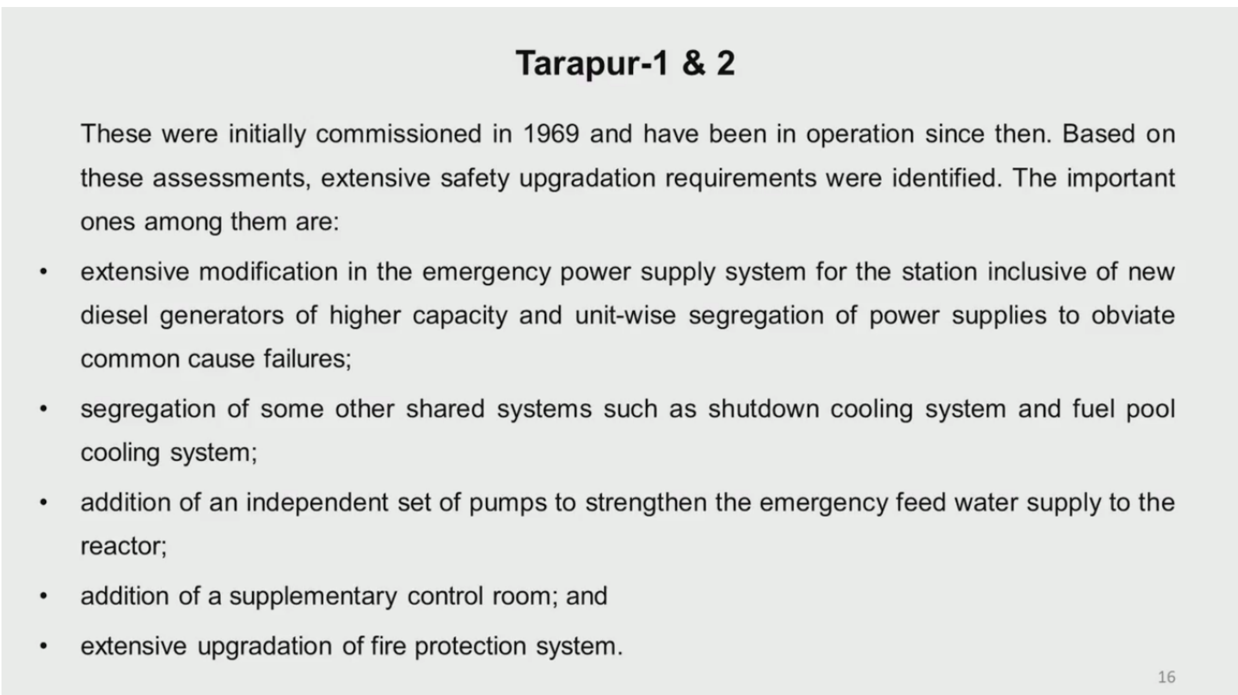

Then we come to the boiling water reactors in Tarapur. As you know the construction started in 1964 and it was commissioned in 1969. Then from time to time we had some operation problems. So we felt that there is need to improve some of the components and do some safety upgradation in some carriers. Here you know the General Electric company which has built a large number of boiling water reactors from time to time they keep on giving out the experiences of the different plants. So we not only have the experience of Tarapur, we also have the experience of other boiling water reactors. So based on this we found out that we need to do the following. we needed lot of modification in the emergency power supply system to the plant. As in the case of Rajasthan Unit-2 we needed to add more diesel generators of higher capacity, and not only that segregate these diesel generators to avoid a common cause failure. I mean common cause means common cause like flood or could be fire. Then other systems like the shutdown cooling system. Whether any of the shutdown cooling systems are shared with any other things. If wherever sharing was there we wanted to remove the sharing plus the spent fuel cooling system that also we needed some improvements. Then on the emergency feed water supply we felt the need to add independent set of pumps. Again addition of a supplementary control room and improvement of the fire protection system. Here I would tell one thing it does not mean that it is unsafe. It is better operation leading to less of troubles. But in our reactor terminology more or less we consider both alike everything is towards safety finally. A plant should be available. Plant should be operate safely without even minor issues. That is our aim. So this was all done.

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- Probabilistic safety assessment (PSA); A Level-1 PSA with internal events was done for TAPS for analyzing the existing design without considering the upgradations. The analysis indicated that the core damage frequency is around 7.0E-05/Reactor year.
- Seismic re-evaluation of structures, systems and components (SSC) of TAPS was carried out for the latest ground motion parameters derived for the site. While most of the SSCs were found to meet the requirements, a few modifications involving strengthening of support structures were found necessary.

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Then another requirement from the regulatory authority is about the probabilistic safety assessment. During my lecture on the analysis of different events I mentioned to you about the deterministic approach and the probabilistic approach. So deterministic approach is one thing but for instance which are a very very low frequency whether the margins we can have high or whether I should postulate an event whether it is really happening. So these sort of assessment based on the what you call probabilistic you know calculations is now mandatory. So we have carried out the probability of safety assessment. Basically level one what we call for all the internal events and we estimate what is the probability of a core damage and it was about 7.0E-05/reactor year. Then not only that seismic re-evaluation. That means we carry out the seismic analysis of all the systems once again, and because we now have the latest data on the ground motion parameters, we have the structure actual when earthquakes happen we got the data. So we found that in certain areas basically the auxiliary systems if we strengthen the supports things would be nice. See the auxiliary systems per se are not going to endanger safety but there are going to affect the availability of the plant. So we have gone in and strengthen those structures.

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Review following the TMI-2 accident

Some of these improvements are listed below:

- Enhancement in the reliability of emergency feed water supply to the steam generators in PHWR.
- Augmentation of feed capacity for inventory control of primary coolant system under loss of off-site power in TAPS.
- Remote operation of isolation valves of moderator heat exchangers to check spread of radioactivity to cooling water system in the event of leaks in heat exchanger tubes.
- Incorporation of high-pressure emergency core cooling system for PHWRs.
- Development of emergency operating procedures for a large number of postulated initiating events.

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Of course after that TMI-2 Three Mile Island-2 accident in the United States in 1979. Again we look into the whole designs even though we didn't have a similar reactor at that time. Now Kudankulam is similar to that. So we found that again lot enhancement of the emergency feed water system was needed. So that we implemented. Then we also augmented the capacity or you can say the inventory of the coolant system so that it could give flow for more basically in the case of the Tarapur Atomic Power Station. Then in some cases we wanted to have a remote operation of some of the valves so that without getting any activity, the operator could operate. So this is a safe operation. So there we did some modifications. Then in the case of PHWRs we also included a high-pressure emergency core cooling in addition to the existing systems so that we have a more defense in depth philosophy of providing cooling under all cases. Not only that emergency operating procedures like TMI-2 surely we didn't expect it happen. So what should be the emergency operating procedures under the new set of events. The next of course accident after Chernobyl.

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Review following the Chernobyl accident

- AERB undertook detailed review of safety of Indian NPPs following the Chernobyl accident in 1986. The review focused on (a) adequacy of safety systems and engineered safety features, (b) implications of propagating failures and (c) emergency preparedness aspects. Safety during operation of units at low power and under shutdown conditions was also reviewed in detail. The review re-emphasized the necessity for adhering to the already established principles of reactor safety design and operation. The feedback from the accident did point to the need for well-coordinated plans and organization for on-site and offsite emergencies that may arise from nuclear accidents. It also re-emphasized the need for maintaining 'safety culture' in the conduct of operations at the station, and having disciplined institutionalized procedures. Actions were taken to reinforce these aspects in operating principles and practices. Also organization and procedures for on-site and off-site emergencies were strengthened at all the power stations.

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So after Three Mile Island was Chernobyl in Russia. So again AERB undertook a detailed review and this very important. One thing was that design of that reactor or a similar reactor of design reactor is not existing in India, neither Tarapur, none. So the design itself had some fault of the Chernobyl reactor. Again it didn't have the containment as I mentioned earlier. So the major aspect of learning in this was emergency preparedness aspects. Then nevertheless, we also started examining our reactor behavior under low power operation where in the case of Chernobyl there is a positive reactivity coefficient. We again analyzed our reactors and reviewed whether under any condition it is possible to get into a positive reactivity regime. So having done that only thing we needed was to have a look again at our emergency plants both for on-site and off-site emergencies. When I say on-site emergency it means the emergency which is limited to the plant and not involving the public. Off-site means which also includes the public. So all these were in the nature of reinforcing the already existing practice with checks and balances.

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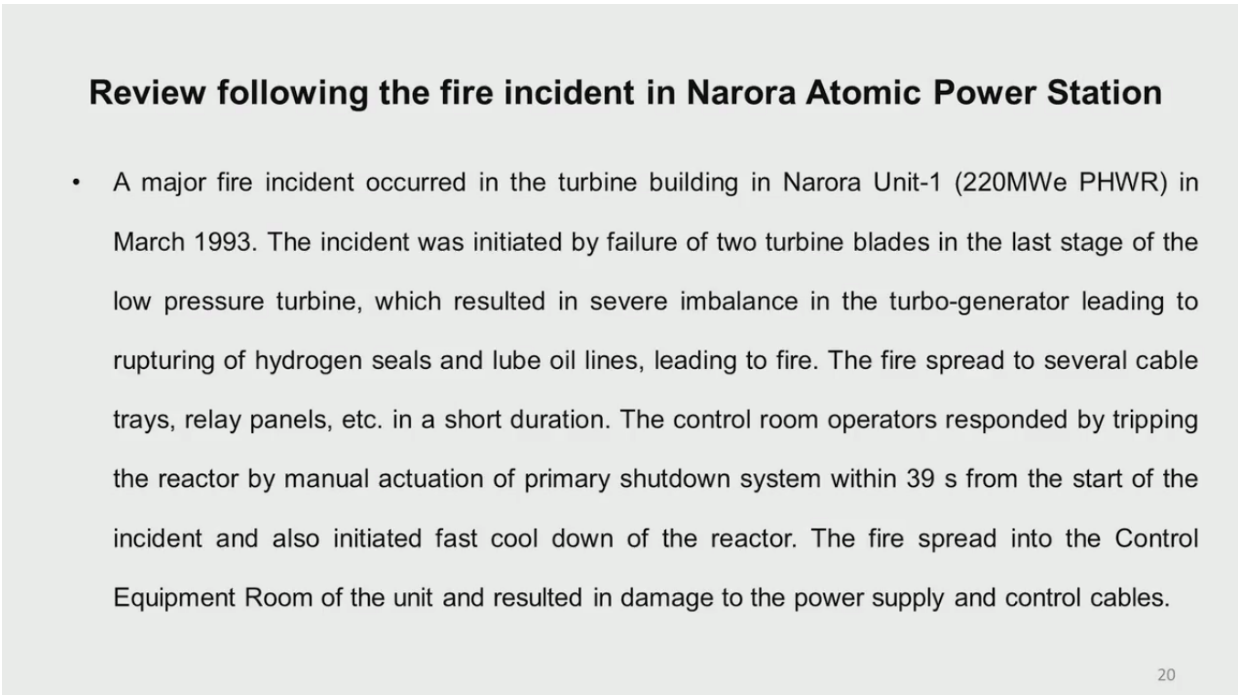

Then we had our own event of fire in the Narora Atomic Power Station. Though I have talked to you about it in my earlier lectures. Nevertheless, it is worth repeating for sake of continuity. This incident happened in the Narora turbine. The last few stages of the turbine, the blades started, you know, the clearances between the blades and the casing came down as a result of vibrations and the vibrations of the shaft were due to imbalances, severe imbalances in the thing. Maybe the blades had moved away and these blades touched the casing when there was casing was touched there are sparks, you know, 3,000 RPM blade touching a static component. There were sparks and not only that because of this imbalance the generator hydrogen seal was damaged and hydrogen came out. Hydrogen is normally used to cool the generators because the cables have got I squared R loss, heating loss as current passes we have to keep them cool so that the insulation life is long and lengthen. So we use hydrogen, and this hydrogen came out and this spark ignited the fire, ignited the hydrogen. The fire what happened? The fire spread to the cable trays, from the cable trays, relay panels, and it also went very close to the control room. Then what the operator did he tripped the reactor by manual actuation and initiated cooling down of the reactor. In the meantime the control room equipment also fire came into the control room and damaged the power supply and control cables.

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- Even though the power sources were available, power could not be supplied to the loads due to damage to the cables. A complete loss of power supply in the unit occurred at about 7 min into the incident and the resulting Station Blackout lasted for a period of 17 h. The smoke ingress into the main control room from Control Equipment Room rendered the control room uninhabitable. During the blackout, core cooling was maintained by thermo-siphoning on the primary side with the secondary side of steam generators fed by firewater to provide heat sink. The major fire was put out in about 1 h 30 min. There was no radiological impact of the incident either on the plant workers or in the public domain and no injury occurred to any individual.

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When this happens what you have the electrical power but you are not able to control your pump. You are not able to start the emergence coolant pumps which is required to cool even in a shutdown state. That is what we call as the decay heat removal. So this was needed. So effectively you can say diesel generators are available but we couldn't start the pumps. So in effect it was a loss of off-site power equivalent to your loss of off-site power and a loss of on-site power. So what you call as station blackout. And this lasted for nearly 17 hours, and also the operators couldn't work in the control room due to heavy smoke. Thanks to the training you know if you recall the engineered safety features we have got different levels of cooling and last but not the least we have fire water safety pumps. There is fire water pumps which are operated by, you know, diesel generators directly not the station diesel generators. So operators started those and maintained cooling of the plant. Of course the fire per se was put in about one hour to half hours but the cables were damaged so the operators resorted to the fire water to quench the heat in the reactor. There was no failure of fuel. There was no radiological impact of this incident and nothing happened to the public. But then this is a event which is not very good. Something like a common mode or common cause failure has happened. So this was very much in detail looked at by the Atomic Energy Laboratory Board.

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AERB carried out an in-depth investigation of the incident which resulted in several modifications and improvements covering design, operation and surveillance practices. The investigation also revealed the susceptibility of the existing design and layout of NAPS to common cause failure (CCF), mainly due to fire as the initiating event.

- Rerouting of power and control cables of safety related loads to minimise the vulnerabilities to common cause failures.
- Provision of additional fire walls at identified areas in turbine building for limiting spread of fire.
- Strengthening of fire barriers at cable penetrations in floors and walls in the turbine building as well as at their entry points into the control equipment room.

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We found that everywhere independence was there of the signals, the shutdown systems, everything. But then when these cables were entering into the reactor building, they were through a common penetration. So the common penetration was the cause of the fire happening. So they re-routed all the power and control cables and independence. The true independence was maintained and also not only that providing additional firewalls so that the fire should not spread like fire barriers at the penetrations wherever and whether fire retardant cables wherever possible these were introduced.

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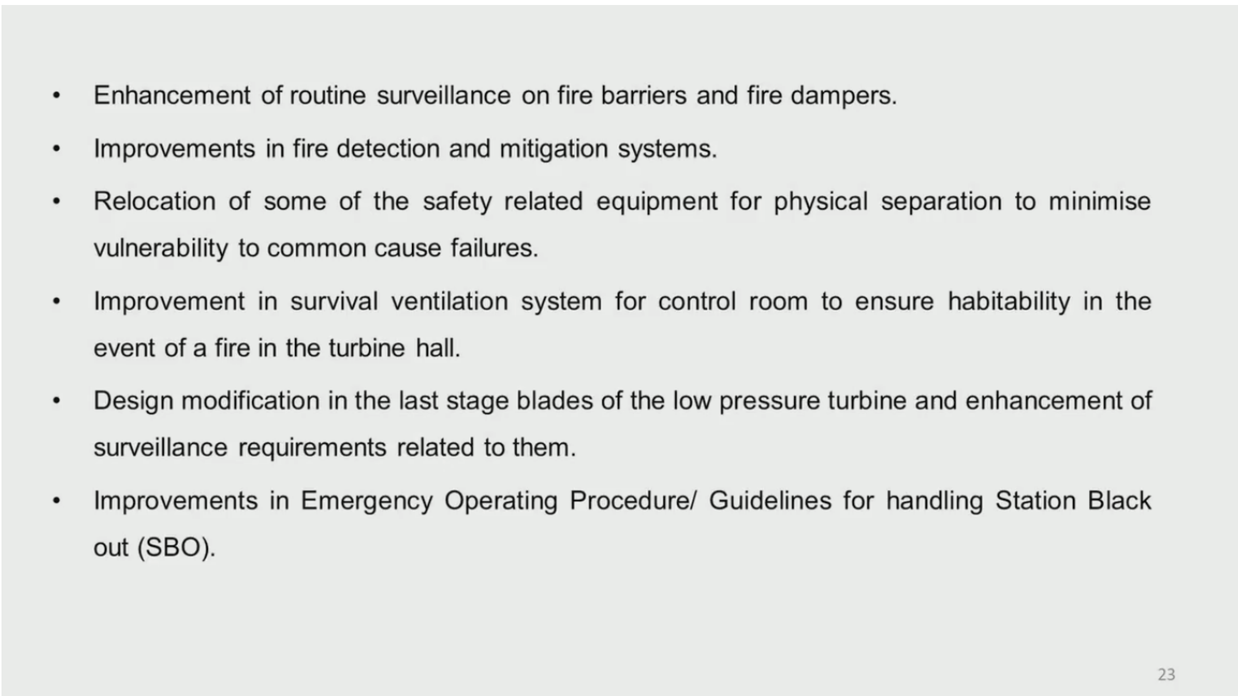

Thereby the chance of a fire affecting all the systems was minimized. Then of course having done this continued surveillance was improved. Then the fire detection systems they were also some improvements were done. Then the ventilation system of the control room we found that control room could not be habituated due to the smoke so what about the ventilation system for the control room. Then last but not the least the design modification to the last stage turbine blades. Here you see these operate on saturated vapor. It is not super-heated steam. So when the saturated vapor you use the last few stages do have moisture and moisture is the one which can cause erosion of the blades. Thereby causing you know these vibrations and imbalances. So this is also an area where material improvement and other things were necessary and we have strengthened these blade designs. Last again, all the emergency operating procedures in case of a station blackout we have again improved.

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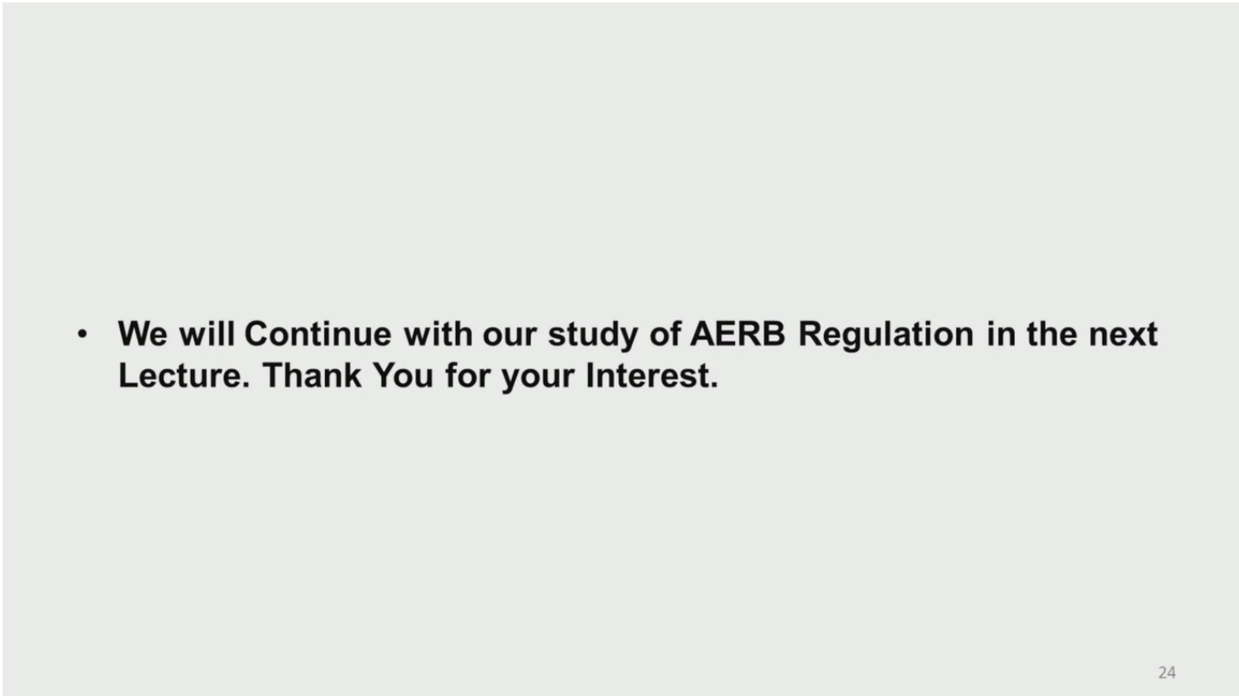

Now we will continue of further regulation of AERB in the next lecture. Thank you.

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