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Safety Preventive Maintenance In Nuclear Power Plant

R.Mohan, N.Partheeban

Abstract: - Safety analysis of a nuclear power plant for postulated initiating events (PIEs) is an essential part of the design process, both as a regulatory requirement and also to generate performance requirement of safety system. A wide variety of computer codes have been developed in-house at Nuclear Power Corporation of India Limited (NPCIL) for safety analysis of Indian Pressurized Heavy Water Reactors (PHWRs). The applications and validation of these codes are discussed in this article. In addition to the conventional 'deterministic' safety analysis approaches, probabilistic safety assessment (PSA) techniques are also being applied in Indian PHWRs to gain additional insights Passing of steam trap becomes a major concern for all power plants due to high-enthalpy energy loss as well as wastage of costly DM water. On the other hand, if steam trap does not do its intended function, then it may lead to the damage of turbine and pipelines. Unfortunately, most power plants do not have a proper condition monitoring or preventive maintenance programmed for steam trap to know its healthiness.

Index Terms—Nuclear Energy, Thermal Hydraulic, Stream Trap. Thermo dynamic, Radioactive Waste.

I. INTRODUCTION

Safety considerations get the highest priority in anything to do with nuclear power plants (NPPs), be it their siting (site selection), design, construction or operation. The overriding objective is to ensure safety of operating personnel, public and the environment. Public concern regarding the safety of nuclear power hinges mainly on three issues: firstly, the possibility of accidents in nuclear power stations leading to release of large amounts of radioactivity; secondly, pollution of the environment from routine discharge of radioactivity from the station; and finally, the question of long-term storage and disposal of radioactive wastes generated from operation of nuclear reactors. The focus of this article is on the first issue, possibility of accidents, and considerations and measures that go into ensuring that risks from these are minimized. The specifics of the discussion here apply to the Pressurized Heavy Water Reactor (PHWR), which is the reactor type currently adopted in most of India's NPPs; the broad principles are, however, representative of all current generation NPPs worldwide.

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* Correspondence Author
R.Mohan*, Dept. of PG Computer Science & Engg, S.A.Engineering College, Chennai-77, Tamil Nadu, India
Dr. N.Partheeban, Associate Professor, Dept. of IT, S.A.Engineering College, Chennai-77, Tamil Nadu, India.

The chief aim of reactor safety is to ensure that the radioactive fission products generated in the reactor are contained under all circumstances. A series of barriers are provided in a nuclear reactor to achieve this containment function. The radioactive fission products are generated within the uranium-dioxide fuel matrix, and most of them are contained within this matrix. Escape of radioactivity from the fuel will therefore require a breach in four barriers, namely, fuel, fuel cladding, primary heat transport (PHT) system and containment building.

A. Safety Design Principles

There are well-recognized safety design principles, which are routinely applied in nuclear reactor technology to ensure that the above safety functions will be performed with a very high degree of reliability. These include the following: A defense-in-depth approach is adopted. This approach is at the heart of the safety philosophy, where there are several lines of defense, one backing another. More than one mean are provided for performing a safety function. Failure of one barrier or level of defense still leaves others to perform the safety function. In applying defense-in-depth to these levels, emphasis is placed on prevention and on minimizing challenges to higher levels. The safety design principles include Requirement of physical and functional separation between component and systems, redundancy, including requirement to meet 'single-failure criterion', fail-safe features, testability and environmental qualification of equipment as required. An important part of the design process is safety analysis, which involves Postulation of initiating events, and, detailed calculations to predict the consequences of postulated accident scenarios (including highly unlikely ones). This is further detailed in next section.

B. Nuclear Power Generation

Nuclear Power Stations use a fuel called uranium, a relatively common material. Energy is released from uranium when an atom is split by a neutron. The uranium atom is split into two and as this happens energy is released in the form of radiation and heat. This nuclear reaction is called the fission process. In a nuclear power station the uranium is first formed into pellets and then into long rods. The uranium rods are kept cool by submerging them in water. When they are removed from the water a nuclear reaction takes place causing heat. The amount of heat required is controlled by raising and lowering the rods. If more heat is required the rods are raised further out of the water and if less is needed they lower further into it.

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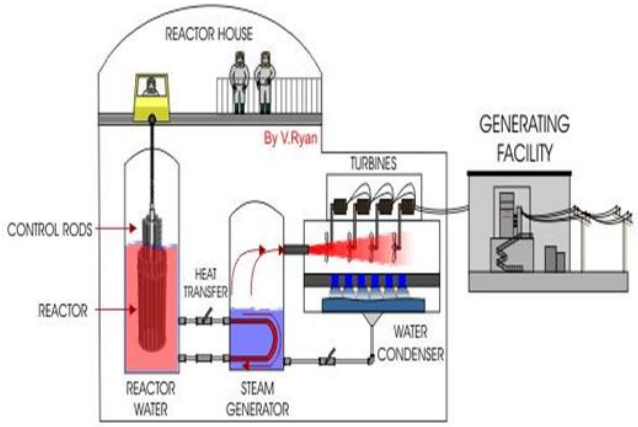

Fig.1. ***Nuclear Power Generation***

Advantages:

- The amount of electricity produced in a nuclear power station is equivalent to that produced by a fossil fuelled power station.
- Nuclear power stations do not burn fossil fuels to produce electricity and consequently they do not produce damaging, polluting gases.
- Many supporters of nuclear power production say that this type of power is environmentally friendly and clean. In a world that faces global warming they suggest that increasing the use of nuclear power is the only way of protecting the environment and preventing catastrophic climate change.
- Many developed countries such as the USA and the UK no longer want to rely on oil and gas imported from the Middle East, a politically unstable part of the world.
- Countries such as France produce approximately 90 percent of their electricity from nuclear power and lead the world in nuclear power generating technology - proving that nuclear power is an economic alternative to fossil fuel power stations.
- Nuclear reactors can be manufactured small enough to power ships and submarines. If this was extended beyond military vessels, the number of oil burning vessels would be reduced and consequently pollution.

C. Radioactive Waste Produced By Nuclear Power Stations

Low Level Waste

Low level waste includes materials that are used to handle nuclear material such as radiation suits and laboratory equipment. They are normally stored for up to 15 years in secure storage and then, after careful packaging they can be disposed of as normal waste. However, there is disagreement over the way the waste is disposed. For example, The British and Irish Governments do not agree on the disposal of low level radioactive material in the Irish Sea.

Intermediate Level Waste

These are much bulkier materials and are characterized by low heat emission. They contain metal fuel cladding, chemical sludge's and other radioactive wastes. The waste is first encased in resin or concrete and sealed in steel drums. The drums are then packed into concrete casks and placed in concrete trenches up to 18 meters deep. When completely filled the trenches are covered with a concrete slab, a layer of compacted clay and a reinforced concrete intrusion shield and a final layer of clay. Deep disposal of intermediate wastes also takes place, storing the wastes in a suitable geological formation at a depth of at least 100 meters.

High Level Waste

High level waste is extremely radioactive and remains in this state for thousands of years. Safe and stable storage of this type of waste is of great concern. Modern storage methods include the use of glass verification. This involves combining the radioactive liquid waste with glass to form a solid compound. Because of the solid nature of the waste it is much less likely to contaminate the surrounding area. Unlike liquid waste, it cannot leak into the ground if the stainless steel container it is in becomes faulty. In theory, the highly radioactive waste can be stored indefinitely in deep stable formation such as caves and caverns Given the demographic shift in many countries toward elderly populations and the subsequent increase of chronic health condition and disability.

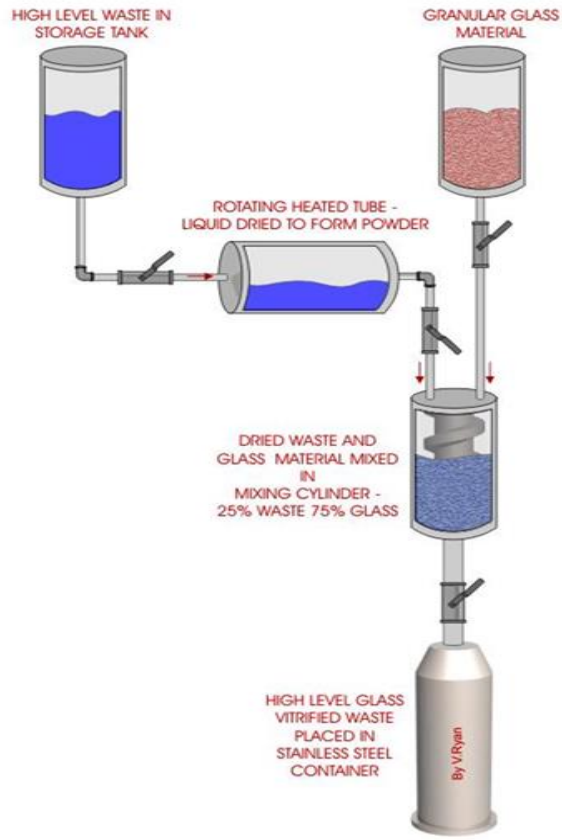

Fig.2.Radioactive Waste Produced By Nuclear Power Station

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D. Safety Analysis Maintenance of a Nuclear Power Plant

Extensive preventive maintenance and testing (surveillance) programs exist to ensure that nuclear safety significant equipment will function when it is supposed to. Diesel generators, pumps, motor operated valves and air operated control valves are typically operated every one to three months. When you drive a car, you depend a lot on the sounds, the feel of the steering wheel and the gauges to determine if the car is running correctly. Similarly with operating equipment at a power plant - if sounds or vibration of the equipment or the gauges and test equipment indicate a problem or degradation, actions are taken to correct the deficiency. If the equipment fails to start or run, more immediate actions are taken. In some cases, regulations called technical specifications may require the plant to be shut down if the equipment is not corrected within a certain period of time. The length of time depends on the safety significance of the equipment.

Every year to two (2) years, the power plant may be shut down for an outage. The outage may last 30 to 60 days and depends on the amount of major maintenance to be done. Outages are used to perform activities that cannot be done when the plant is operating: Refueling the reactor and other preparations (removal of reactor head, upper internals, and reactor refueling)Preventive maintenance on equipment that must run all the time, e.g. turbine-generator must be inspected every 5 years or so, transformers may be checked out each outage; Modifications or replacements of major equipment, as a steam generator, that cannot be shut down. The maintenance personnel who maintain the equipment at the power plant must go through craft-specific training to qualify to perform the plant maintenance. Training programs are inspected and certified by the accrediting board of the National Nuclear Training Academy. Engineers at the power plant are often responsible for specific systems at the plant and manage the work done (preventive maintenance, repairs, and modifications) on their system. Similarly, engineer training programs are inspected and certified by the accrediting board of the National Nuclear Training Academy.

II. BACKGROUND

A. Literature Review

trip, off-site power failure, trip of a feed-water pump), it is required to demonstrate that the operational limits and conditions are satisfied, and that the plant either survives the transient or trips safely and characterization of postulated initiating events (PIEs) that are appropriate for plant design and its location. These may range from events of infrequent nature (with frequency of occurrence up to 10-2 per year) such as small-break loss-of-coolant accident (LOCA) to limiting design basis events (frequency of occurrence down to 10-6per year) such as large-break loss-of coolant accident, and multiple failures, e.g., LOCA coincident with failure of a mitigating safety system. For each PIE, consequent event sequences are required to be worked out, identifying the mitigating actions and systems that will cut in, either automatically or with operator intervention (if adequate time and warning signals are available). Finally, for each of these event sequences, it is not only required to evaluate the consequences, primarily in terms of radioactivity release/dose to the environment.

Consideration is also required to be given to events and event sequences beyond those considered in design basis. These events would typically have frequencies of occurrence of lower than 10-6 per year, and would involve multiple failures, e.g., an accident sequence leading to failure of containment integrity.

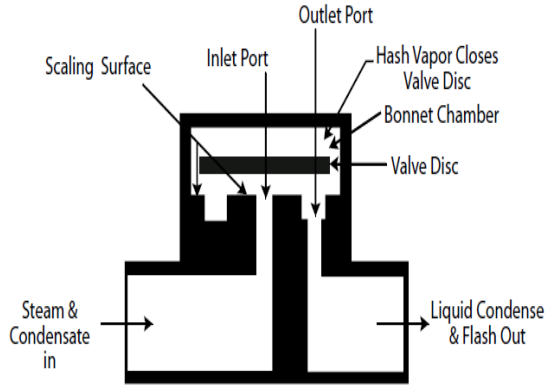

Fig.3.Thermodynamic Disc Steam Trap

Thermodynamic traps use the difference in kinetic energy (velocity) between condensate and live steam to operate a valve. The disc trap is the most common type of thermodynamic trap, but piston or impulse traps are sometimes used. Disc-type thermodynamic steam traps are most commonly used steam traps. This type of thermodynamic traps uses the position of a flat disc to control steam and condensate flow. When the condensate flows through the trap, the disc is raised thereby causing the trap to open. Steam and condensate rise through the center hole, turn around under the disc, and bleed off through the surrounding space. The cap over the disc forms a chamber in which the disc is enclosed. At the top is an insulating cover, which improves strap performance. For infrequent anticipated operation occurrence and transients, the requirements are that the primary coolant pressure should be within the design overpressure limit of the system and there shall be no fuel damage as demonstrated from prediction of no departure from nucleate boiling and no fuel centerline melting. Also there should be minimum challenges to protection and safety systems, i.e., most of the corrective actions should take place by actuation of normal regulating and control systems. For accident conditions considered in design, fuel failures may occur but the calculated radiological consequences to the environment must be shown to be within prescribed reference dose limits.

III. COMPUTER CODES AND VALIDATION

A. thermal hydraulic of the reactor system

Over the years, well-laid-out regulatory Requirements have developed for performing safety analysis.

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While some of the requirements are specific to particular reactor type, most of the requirements are pretty universal. The analyses require the use of complex computer codes which model the physics of various physical phenomenon's relevant to the accident scenario, such as the thermal hydraulic of the reactor system, core neutrons, fuel and core component heat-up effects, etc. These codes have to be qualified by validation usually against several sets of experimental data. The inputs and assumptions in the analysis are chosen such that there is a deliberate conservatism in the results. The objective is to have inbuilt margins to cater for uncertainties as well as to implicitly provide for accident scenarios which have not been explicitly considered within design basis. The scenarios thus evaluated for the prescribed postulated initiating events and event sequences are stylized enveloping scenarios with the objective that the actual accident sequences, should they occur, would have consequences within those predicted. Accordingly, some of the rules of the safety analysis include the following:All input parameters are chosen at the highest or lowest end of their normal range so as to yield worst scenario results. The off-site grid power supply is assumed to have failed/not available so that all power requirements are met by starting on-site emergency diesel generators (DGs).The mitigating safety systems are assumed to be available with their most effective single active components failed, e.g., the reactor shutdown system's availability is considered with its most effective shutoff rod not available. In PHWRs where two independent reactor shutdown systems are provided, the one which is more effective, is assumed failed While considering automatic trip of reactor on protective system actuation, the first trip parameter is usually ignored, for each of the two shutdown systems in PHWRs In PHWRs, the design also considers failure of one of the mitigating safety systems following the occurrence of the postulated initiating event, e.g., loss of- coolant accident followed by failure of emergency core cooling system (ECCS).

The results of the safety analyses are required to predict the sequence of events on a time scale starting from the initiating event to the final stabilized safe conditions, covering, e.g., timings of reactor trip, primary system pressure reaching safety relief valve set point, safety relief valve operation, emergency core cooling actuation, containment isolation signal initiation, etc. Any operator action credited in the sequence has to be shown to have availability of unambiguous signal as well as adequate time for the operator to act. Important results for assessment of safety include timely shutdown of the reactor, adequacy of core cooling, fuel temperatures and its integrity, primary coolant system pressure and integrity of the primary system boundary, integrity of the secondary system, continued decay heat removal in the stabilized long-term basis, performance of containment and other parameters, and finally, assessment of reactivity release to the environment. The acceptance criteria for different scenarios depend on the category of scenarios in terms of frequency of their occurrence. For normal operations and operational transients, the acceptance criteria would be that the operation limits and conditions should be satisfied and reactor should survive safely without tripping

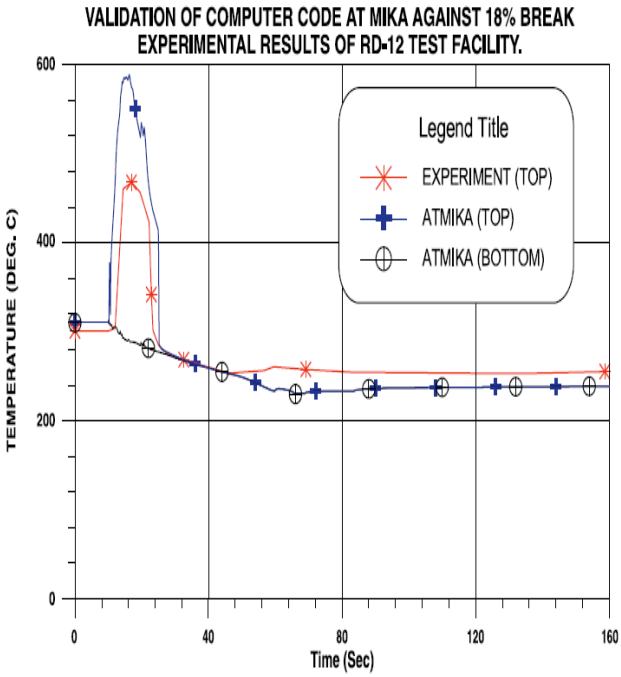

Fig.4. Variation of sheath Temperature in Heated Section

B. Probabilistic Safety Assessment (PSA)

The above approach based on 'deterministic' safety analysis of specified design basis events/accidents with conservative margins together with defense-in-depth and safety design principles has been traditionally used for safety of NPPs. In recent years, this deterministic approach is being complemented by 'probabilistic' safety assessment techniques, which have added a new dimension to our understanding of overall safety of our NPPs. A PSA is a structured tool to represent an integrated model of the safety of the plant encompassing design, operational safety practices, component reliabilities, dependencies (that is, 'common-cause failure' vulnerabilities) and human reliabilities. This is done by establishing logical relationships between combinations of component failures and operator response ('basic events) on one hand, and plant response on the other, through the use of fault-trees and event-trees, and considering the probability, progression and consequence of the basic events. This enables determination of numerical estimates of a consistent measure of safety, e.g., `core-damage frequency', or `large release frequency'. Such an integrated model can be used to identify dominant contributors to possible severe accidents and thus aid in safety decision-making in design and operation of NPPs.

An important difference between deterministic and probabilistic approaches is that while the former studies specified events in detail and adds margins in order to envelope possible real situations and to address more severe situation implicitly, the probabilistic approach aims to address all scenarios, including severe accidents explicitly; also 'best-estimate' approach is aimed for. Depending upon the extent of the study, PSAs have come to be defined in terms of three levels: Level-1 PSA addresses the identification of failures in the plant leading to core damage and their frequencies of occurrence.

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Level-2 addresses the assessment of containment response leading, together with Level-1 results, to determination of containment release frequencies, and qualification of releases.

Finally, Level-3 is the assessment of off-site consequences leading, together with results of Level-2 analysis, to estimates of public risk. Among these, Level-1 PSA is the most commonly performed, and is recognized to provide the biggest gains in terms of understanding the importance of various design/operation features to safety of plant with regard to potential accidents. In India, like elsewhere, while safety assessments and safety decisions have mostly been based on deterministic approaches, probabilistic safety assessments are entering the scene progressively, as a complementary approach, to aid 'risk-informed' decision-making. One of the most important elements in a steam system in terms of performance and safety is the steam trap. A steam trap is essentially a self-actuated valve that is used to remove condensate and non-condensable gases from a steam system like pumps, heat exchangers, pressurizer and associated control systems and logics. The computer code has been used to demonstrate. Disc traps commonly have an intermittent discharge and when Disc traps commonly have an intermittent discharge and when they fail, they normally fail open. As a result of any further heat loss, the temperature of the condensate will fall. A thermostatic trap will pass condensate when this lower temperature is sensed. As steam reaches the trap, the temperature increase.

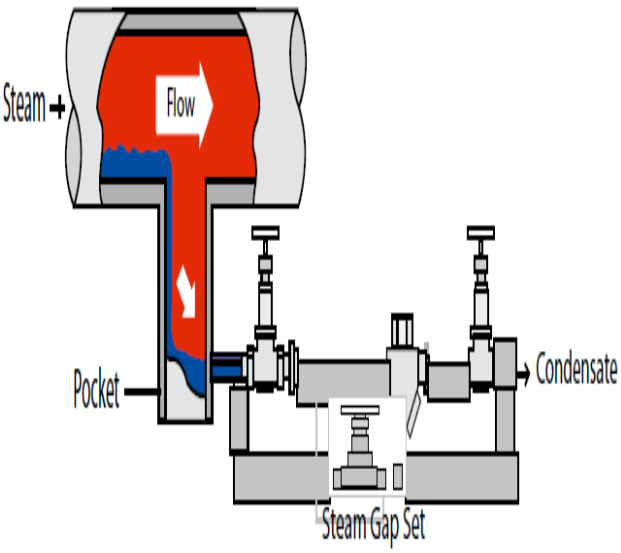

Fig.5. Typical Layout of a Stream Trap

When operating properly, the trap will open in the presence of Condensate, regardless of temperature, and allow the condensate to pass out of the system, but close before steam escapes. Pressurized Heavy Water Reactor (PHWR) type nuclear power stations in India essentially use dry saturated steam with 0.26% wetness factor. Due to dry saturated steam, operation of steam traps are vital as underperformance of traps will lead to the increase of moisture content, thus damaging turbine blades and also it can cause water hammering in steam. On the other side, if steam traps passes, then it leads to the huge steam losses, thus increasing DM water losses and reducing output of the plant. 2.0 Various Types of Steam Traps There are three basic types of steam trap into which all variations fall; all three are classified by International Standard 2.1 Thermostatic (operated by changes in fluid temperature) The temperature of saturated steam is determined by its pressure. In the steam space, steam gives up its enthalpy of evaporation (heat), producing condensate at steam temperature.

IV. PROJECT RESEARCH METHODOLOGY

Safety analysis is required for plant conditions ranging from normal operation/operational transients through anticipated operational occurrences/ transients, low-frequency events to limiting design basis events and beyond. Some specific areas of this analysis. Station transient analysis System thermal-hydraulic analysis for accidents. Fuel behavior under accident. Conditions. Thermo-mechanical behavior of pressure tube under accident condition. Containment analysis; describing containment behavior during accident conditions including Fission Product transport and release. Analysis of 'Beyond Design Basis Severe Accident' in PHWRs each of these areas is elaborated below. As mentioned before, safety analysis usually requires use of complex computer codes modeling the physics of the various phenomena involved.

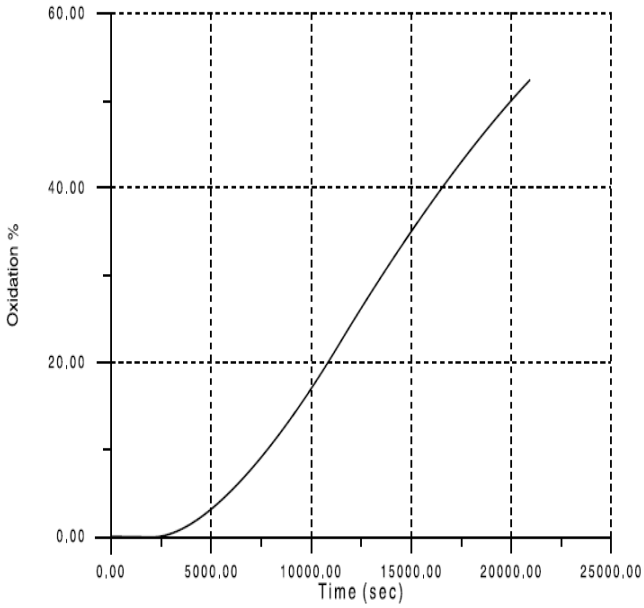

Fig.6.Percentage Oxidation in case of Severe Accident

Safety analysts worldwide have access to a number of standard computer codes available off-the-shelf commercially or otherwise. In the Indian context, most such codes have had to be developed in-house by the utility, i.e., NPCIL. Development of such codes is a specialized activity. As steam and air pass through the trap the disc moves downward.

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The force that causes the disc to move downward is generated by the difference in pressure between the low-velocity steam above the disc and the high-velocity steam that flows through the narrow gap beneath the disc. These computer codes are put to rigorous benchmarking and testing and validated against either experimental results or comparison with predictions from other validated codes. Two kinds of experimental data are used: i) Separate Effects Tests, which are designed to validate the specific Modeling aspects. For example, critical flow assessment, critical heat flux assessment, two-phase flow instabilities in a set of 28 parallel channels, etc., and ii) Integral Loop Tests, which are used to verify the overall code predictions applicable for the plant, viz. thermo syphoning, loop flow instabilities, etc. Data from plant commissioning tests and operational transients is another source for validating some codes. One such computer code 'ATMIKA' [3] developed and validated in-house in NPCIL is a system thermal-hydraulic neutrons computer code, which is used for simulation of postulated accidents such as LOCA and other transients. The code incorporates appropriate models for simulating the core neutron kinetics, system fluid dynamics, flow quality, wall heat transfer within fuel to coolant, heat transfer, critical discharge, single and two-phase friction pressure drops, heat conduction in fuel, component model.

V. DESIGN IMPLEMENTATION

A. *Specific Areas of Analysis*

Safety analysis is required for plant conditions ranging from normal operation/operational transients through anticipated operational occurrences/ transients, low-frequency events to limiting design basis events and beyond. Some specific areas of this analysis are: Station transient analysis, System thermal-hydraulic analysis for accidents, Fuel behavior under accident conditions, Thermo-mechanical behavior of pressure tube under accident condition, Containment analysis; describing. Containment behavior during accident conditions including Fission Product transport and release.

B. Station Transient Analysis

These are disturbances arising from both primary and secondary sides causing deviation from normal operation. Transient analysis is carried out as part of design as well as regulatory requirement to study the dynamics of the nuclear power plant (NPP). The basic aim is to safely override these transients with the help of the control system and minimize the number of reactor trips. In NPCIL, transient analysis is carried out using in-house developed system thermal hydraulic-neutronic computer code. Thermal-hydraulic modeling of both primary and secondary systems is available in this code. These models, along with model of neutronics and controls, are used to study the integrated plant response.These codes are validated with the appropriate plant data for every NPP during commissioning tests.

C System Thermal-hydraulic Analysis – Accidents

The system behavior is analyzed for various postulated initiating events (PIEs), which governs the design of safety system. Important PIEs considered are loss-of-coolant accident (LOCA), main steam line break (MSLB), primary coolant pump seizure, and station blackout (SBO). One of the important PIEs for which emergency core cooling is designed in Indian PHWRs is LOCA. A LOCA is caused by a break in the pressure retaining boundary of the reactor primary coolant system, resulting in the discharge of high-enthalpy primary coolant into the containment building.

D. Fuel Behaviour under Accident Conditions

Fuel, in case of PHWR, has collapsible cladding. Under normal operation, the cladding internal pressure due to fission gas is lesser than the external temperatures to determine, oxygen embrittlement. Oxygen embrittlement criterion is used in predicting the performance of fuel. During LOCA in PHWR, there exists a power ramp for a short period before reactor trip. Under such a situation, fast energy deposition rate may lead to failure of fuel because of pellet-clad interaction. Hence fuel-enthalpy criterion is used .

E. Containment Analysis

In a nuclear power reactor, containment system is one of the most important safety systems. The response of containment system is analyzed during both normal and abnormal conditions. It is the final barrier between radioactive fission products and the outside atmosphere. Therefore, containment system has to be designed to withstand pressure and temperature transients prevailing under accident conditions and its integrity and functional requirements should not get jeopardized under such conditions.

VI. CONCLUSION AND FUTURE WORK

Safety analysis of Indian PHWR is predominately based on deterministic approach. Nowadays, probabilistic safety analysis (PSA) techniques are progressively coming in vogue as complementary approach, to provide added insight in overall safety of nuclear power plants. Capability to perform comprehensive safety analysis of NPPs requires the availability of a wide variety of well-validated computer codes as well as experienced experts in using these codes. At NPCIL, development of computer codes for safety analysis is a continuing activity aimed at further refinements and additional validation in existing codes as well as development of newer codes for areas of analysis still not adequately covered by existing codes. Additional experimental setups including an integral loop test facility are planned within NPCIL to acquire required data to validate such codes.

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AUTHOR PROFILE

R.MOHAN received the bachelor's degree in 2009 in computer science & engineering from Anna University in Meenakshi College of Engineering. He is currently doing Master's degree in computer science & engineering from Anna University in S.A Engineering College. His research interests include Data mining & Data Warehousing, Mobile Computing wireless networks, network security. He is published more than international journals and national conference and national conference Topics include coverage problems in mobile Authentication, Secure Multimodal Authentication. He is a member of the IEEE.

N.PARTHEEBAN received the bachelor's degree in 1998 in Computer Science & Engineering from Madras University. Master's degree in 2005 in Software Engineering from Bharath University. He is currently doing Ph.D. in Information &Communication Engineering from Anna University of Technology and also Associate professor, Head of the Department in Information Technology in S.A Engineering College. He is published more than international journals and national conference and national conference. His research interests include Data mining & Data Warehousing, web services & E-Learning, Network Security. He is a membership of the ISTE, IAENG.

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