Research and Development Activities in Level-1 Probabilistic Safety Assessment and Aging Studies

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Abstract

Research and development activities that are in progress in Reactor Safety Division in the areas namely Level-1 Probabilistic Safety Assessment (PSA) and ageing of control and instrumentation components and cables are presented. The results of AHWR level-1 PSA studies carried out are presented. PSA studies have applications in many areas. A software “Risk Monitor” was developed that can be used to continuously monitor the status of the Nuclear Power Plant (NPP) with regard to the functioning or non-functioning of its different subsystems and assess the associated risk emanating from the facility on the basis of this information. Probabilistic precursor analysis studies have been initiated for assesssing risk arising out of the events that occurred during the operation of the plant. A case study has been carried out for a typical NPP. All the PSA studies are dependent on failure parameters that enter into various component models. A case study has been carried out, along with NPCIL and AERB for assessing more applicable failure parameters for class IV failure frequencies using Bayesian methods. Fire PSA studies have been carried out for MAPS, as a case study, to identify the vulnerable areas. Pilot studies on Risk informed In-Service-Inspection studies have been carried out, for a typical NPP and a section of Heavy Water Plant, for categorizing the components for In-Service inspection based on their contribution to risk. These two pilot studies indicate that RI-ISI has a very high potential in reducing inspection requirements compared to the requirements as per the currently adopted standards. A neural network based diagnostic system for identification of accident scenarios in 220 MWe Indian pressurized heavy water reactors (PHWRs) has been developed for operator support and accident management. Various facilities have been set up and a few were updated for carrying out the Thermal & Radiation Ageing and Loss Of Coolant Accident (LOCA) Qualification Studies of Control & Instrumentation (C&I) components and Equipments. Instrumentation systems for carrying out these studies are also being up dated. Nowadays digital systems are increasingly being used in instrumentation and control. Studies based on physics of failure are in progress to address the issues related to their failure, which are needed in PSA studies. Even though software that goes into these systems undergoes extensive validation and verification, quantification of the failure rates of these software are needed in PSA studies. Work on software reliability is also under way.

Introduction

Probabilistic Safety Assessment (PSA) is an analytical technique for assessing the risk by integrating diverse aspects of design and operation of a Nuclear Power Plant (NPP). In the context of Nuclear Power Plant, PSA can be carried out in three levels. Level-1 PSA estimates the frequency of accidents that cause damage to the nuclear reactor core. This metric is commonly called core damage frequency (CDF). Level-2 PSA starts with the output of Level 1 studies and estimates the frequency of accidents that release radioactivity from the nuclear power...
One of the early PSA for a Nuclear Power Plant is presented in the report WASH-1400, popularly known as the Rasmussen Study, showed that the risks associated with Nuclear Power Plant (NPP) operation are far less compared to the risks due to various other means of power generation. This study formulated a systematic procedure to carry out PSA. This activity gained momentum only after the Three Mile Island (TMI) incident (which was considered in WASH-1400) and further studies were conducted on various NPPs. In India, PSA was initiated with reliability analysis studies that were conducted on various systems of Dhruva and MAPS. However, the first PSA study was conducted was on Narora Atomic Power Plant.

In this paper, Level-1 PSA activities that are in progress in Reactor Safety Division are highlighted, with emphasis on applications to plant operation, maintenance, regulatory, accident management and ageing issues.

Probabilistic Safety Assessment Studies

PSA studies require
(a) Identification of initiating events and estimation of their frequencies.
(b) Construction of event trees for these initiating events and identification of accident sequences (combination of initiating event and subsequent failure/operation of mitigation system(s) necessary for this initiating event).
(c) Construction of fault trees for these mitigation systems and generation of minimal cut sets (combination of minimal component failures that result in system failure).
(d) Quantification of fault trees and event trees, using data on component failure, human errors, external events, etc.

AHWR PSA

A level-1 PSA has been carried out for AHWR considering only internal IEs, full power operation state with reactor core as the source of radioactivity release. In this study, consequence of event trees have been categorized into four states as defined below:

(i) Core Damage State: The Core Damage State is defined as the accident condition which results in peak clad temperature beyond 1473 K.
(ii) Core Degradation State: The Core Degradation State is defined as the accident condition which results in peak clad temperature beyond 1073 K, and within 1473 K.
(iii) Deviation from Safe State: The deviation from “Safe State” is defined as the accident condition which results in the peak clad temperature beyond 673 K, and below 1073 K, which is the fuel failure criterion.
(iv) Success State: The Success State is defined as the safe condition, wherein fuel temperature is less than peak clad temperature 673 K.

The accident sequences have been binned to one of the above four states. The accident sequences resulting in Core Damage State have been considered in Core Damage Frequency (CDF) estimation. The Core Damage Frequency is found to be \(-5.46e-8/yr\). The frequency for Core Degradation is found to be \(2.56e-7/yr\).

Applications of Probabilistic Safety Assessment Studies

Probabilistic Safety Assessment analysis provides insights into plant processes and mechanisms and possible interaction between plant systems, both for existing plants with operating histories and for
plants still in the design stage. In addition to providing an estimate of base line risk, these studies do provide the information on plant vulnerabilities. Hence, these studies can be utilized in adopting optimal configuration control strategies, operational event analysis, etc. Some of these applications are realized in Reactor Safety Division and these are elaborated in subsequent sections.

Risk Monitor Development and Applications

Nuclear Power Plant configuration undergoes changes due to changes in component status and/or operating/maintenance procedures. Some components are randomly down and/or others are planned for test, maintenance and repair. These configuration changes result in a variation of CDF over operating time (called risk profile). Risk Monitoring can be defined as the process whereby a complex technical facility is continuously monitored with regard to the functioning or non-functioning of its different subsystems (configuration changes) and the associated risk emanating from the facility on the basis of this information is evaluated. This can be regarded as being part of the risk management of a plant.

As a result of the availability of level-1 PSA studies, there is a desire to use them to enhance plant safety and thereby reduce risk in the operation of NPPs in a most efficient manner. Towards this, Risk Monitor4, a PC based and user friendly tool, which computes the real time safety level in terms of Core Damage Frequency (CDF), has been developed. Risk Monitor assesses the CDF of the plant in the “as it is” configuration and is useful in optimization of test and maintenance activities to reduce risk in the operation of Nuclear Power Plants. It also supports in deciding the effect of various operation and on maintenance strategies on risk.

In addition to the data requirement mentioned above, Risk Monitor needs detailed:

(a) Information on component data which include type of model (Tested, repairable, non-repairable, mission time etc.,) and their corresponding parameters.

(b) Information on Common Cause Failures (CCF). This includes different number of CCF groups, basic events of each group and their corresponding factors (β factor, α factors etc.)

Data Flow Diagram of Risk Monitor is shown in Fig. 1.

Once the inputs are specified, Risk Monitor will calculate the risk coming from the plant (in terms of core damage frequency in case of nuclear power plants). This is called the base line risk which is based on the as designed/recently updated configuration of the systems and components in the plant. Risks arising out of changes in configuration of systems/components are also evaluated as a function of time as shown in Fig. 2. The figure has different colour bands. If the current risk is considered as completely acceptable if the risk is in the green band where as it is unacceptable if it falls in the red band.

The bands are usually selected based on industry experience or guidelines set by regulatory bodies. ‘What if’ feature in Risk Monitor helps to analyse various planned strategies and is useful in arriving at optimal solutions to configuration control.

Probabilistic Precursor Analysis Studies

Incidents occur during the operation life of a complex industrial facility and sometimes, these can act as indicators (precursor) of impending serious situations (for e.g. core damage). In Probabilistic Precursor Analysis studies, PSA results are used to assess whether the incident that occurred in a plant is a precursor or not. The metric, Conditional Core Damage Probability (CCDP), is used for analyzing precursors. The deciding value for precursor, based on CCDP, is dependent on national regulations (e.g. 10^-6).
Fig. 1: Data flow diagram of Risk Monitor

Fig. 2: Main summary of the Risk Monitor
Basically there are two types of precursor events:

(i) Transient which interrupts the normal operation of the plant. In this case the event can be easily related to an initiating event (IE) of the PSA model (if modeled). The accident scenarios affected by the event are those depicted in the event tree corresponding to this initiating event. CCDP for this type of events is calculated as

\[
CCDP = \frac{f_{IE}}{\lambda_{IE}},
\]

Where \(f_{IE}\) is the sum of the frequencies of the accident scenarios affected by the event and \(\lambda_{IE}\) the frequency of occurrence of IE.

(ii) Unavailability or a degradation of equipment or systems without any immediate impact on plant operation. If the event is related to one (or several) safety functions, a systematic survey of the principal scenarios which the event impacts needs to be done. First, all the initiators which require the affected safety function(s) need to be identified. In the event scenarios or sequences developing from these initiating events, only the scenarios which entail the precursor event are retained. In this case, the CCDP is evaluated as:

\[
CCDP = T_{\text{event}} \times \left(\frac{CDF_{\text{event}} - CDF_{\text{base}}}{A}\right)
\]

Where,

- \(A\) is the fractional the duration of power operation per year,
- \(T_{\text{event}}\) the duration of the operational event (yr),
- \(CDF_{\text{event}}\) the core melt frequency during the event (1/yr), and
- \(CDF_{\text{base}}\) the base value of core damage frequency during power operation (1/yr).

Another metric, Risk index, is a measure of unacceptable consequences, typically either core damage frequency or beyond design basis frequency. It is obtained by multiplying the conditional probability of the precursor event with the frequency (one event within the observation time), and summing up all precursor events within the observation time in reactor years.

\[
\lambda = \sum_{i} \frac{CCDP_i}{\text{Observation time}}
\]

Based on the available event information, during the period of 2002-2006, from a NPP, precursor analysis has been carried out. For the plant under consideration, there were 9 events observed, out of which 7 events identified as IEs (type I above) that are amenable for PSA analysis where as the remaining two events were identified as not amenable for PSA analysis (these events have not been considered in the PSA analysis). This identified the need to update the PSA model to take care of the observed operational events.

In Table 1 the events are presented along with the grading assigned to them based on International Nuclear Event Scale (INES) and the CCDP values. The INES grading is in the order of increasing levels of severity. These levels are: ‘anomaly’, ‘incident’, ‘serious incident’, ‘accident with local consequences’, ‘accident with wider consequences’, ‘serious accident’ and ‘major accident’. The aim in designing the scale was that the severity of an event would increase by about an order of magnitude for each increase in level on the scale (i.e. the scale is logarithmic). For example, the 1986 accident at the Chernobyl nuclear power plant, which had widespread impact on people and the environment, is rated at Level 7 on INES.

The results of precursor analysis are shown in the Figs. 3 & 4. Fig. 3 shows the CCDP values of events. It can be noticed from this figure that all the events are having the CCDP value well below \(10^{-6}\). Hence, no event is of ‘precursor’ type and all are of ‘no precursor’ type. Fig. 3 shows the total number of
events of a given type of category (important precursor, precursor and no precursor based on CCDP value) year wise. From this figure, it is found out that there is only one event of ‘no precursor type’ in the years 2002 and 2003. There are 3 events in the year 2004 and two events in the year 2006. There is no event in the year 2005. All these events are of ‘no precursor type’.

From Table 1, it can be seen that for the first two events the INES values assigned are ‘0’, and this means these are not significant events. The corresponding CCDP values (5.64E-10) obtained is also low for these events. INES value for the third event is 1 and the CCDP value is 9.00E-09. But, for the fourth event even though the CCDP value is 5.91E-09, which is less than that for the 3rd event, the INES value is given as 2. Similarly, CCDP value for the fifth event is 8.24E-09, which is in between the above mentioned two events (3 and 4), but the INES value is given as zero. For the 8th event INES value is given as zero, but the CCDP value is 3.21E-07 which is greater than the CCDP value of 3 and 4 events for which INES values have been given as 1 and 2 respectively. From the above discussion it is evident that INES values for some of the events are underestimated.

Risk index calculated for this plant (as per equation 3) is 8.78E-08/year.

The major advantages of this approach are the strong potential for augmenting event analysis which is currently carried out purely on deterministic basis. From the observations it is found that there is slight discrepancy between CCDP values and INES scale associated to an event. Also, the risk index gives an indication about the safety culture followed in plant and can be used as a metric for comparing between various plants.

Table 1: Events observed, their INES level and the CCDPs

<table>
<thead>
<tr>
<th>Sl. No.</th>
<th>Event description</th>
<th>INES level</th>
<th>CCDP</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>SG tube leak– IE</td>
<td>0</td>
<td>5.64E-10</td>
</tr>
<tr>
<td>2</td>
<td>SG tube leak– IE</td>
<td>0</td>
<td>5.64E-10</td>
</tr>
<tr>
<td>3</td>
<td>refueling-IE</td>
<td>1</td>
<td>9.00E-09</td>
</tr>
<tr>
<td>4</td>
<td>LORA- IE</td>
<td>2</td>
<td>5.91E-09</td>
</tr>
<tr>
<td>5</td>
<td>PCP trip – IE</td>
<td>0</td>
<td>8.24E-09</td>
</tr>
<tr>
<td>8</td>
<td>IE – Partial loss of Class IV supply</td>
<td>0</td>
<td>3.21E-07</td>
</tr>
<tr>
<td>9</td>
<td>IE – IRV open</td>
<td>1</td>
<td>6.27E-09</td>
</tr>
</tbody>
</table>
Fig. 3: Graph between CCDP Vs Event for NPP-1

Fig. 4: Graph between number of Events Vs Year for NPP-1
Studies Related to Failure Data Base

The results of PSA are as reliable as the component failure and repair data that goes in to the evaluations. Components, used in NPPs, undergo strict quality control and hence their failure rates are, in general, very low. Use of generic component failure data, that is available, raises questions regarding the applicability of the results to our situations. To generate a component data base, data collection over a large operating time (component hours) is needed. Until enough data from our plants is collected for generating our own failure database, updating of data based on Bayesian analysis studies can be undertaken. These studies combine the generic failure data and the limited operating experience available at our plants to arrive at more appropriate data that can be applied in the PSA studies of our plants. Presently Bayesian analysis studies are being carried out along with NPCIL and AERB to develop generic prior failure rate distributions for important components by pooling the failure information from all the NPCIL plants. Using this generic prior and the failure information for a particular plant, applicable posterior distribution of failure data can be generated for that plant.

A case study was carried out for the case of class IV power failure frequency. The data used for this corresponds to the failure records of class IV power failure at our power plants. These studies provide a sound basis for the failure data for use in the PSA studies of our NPPs. In addition, these help in estimating the uncertainty in risk indices due to epistemic reasons.

Fire PSA

Fire PSA is conducted by identifying fire scenarios that may affect the safe operation of the plant (through impacts on equipment and human actions), and estimating the frequency of occurrence of those scenarios. The primary output of a fire PSA is typically the estimated frequency core damage initiated as a result of fire. The various steps involved are as detailed below and shown in Fig. 5.

1. Plant walkdown - start with listing out of fire zones. For each fire zone, fire hazards, combustible materials present and presence of safety related equipments are noted.
2. Screening analysis - depending on fire loads and safety related equipments present, screening analysis shortlists the fire zones for detailed analysis.

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**Fig. 5: Essential elements of fire PSA**
Risk Informed In-Service Inspection

Risk Informed In-Service Inspection (RI-ISI) aims at categorizing the components for In-Service inspection based on their contribution to Risk. Probabilities and consequence of component failures need to be computed for defining their contribution to risk. The failure probabilities are computed by understanding the possible degradation mechanisms that can be present. Statistical models, structural reliability models and expert judgement are normally employed for failure probability estimation. For nuclear applications consequence modeling is done using PSA whereas for non-nuclear application the consequences depend upon the considerations in their risk analysis such as toxic health effects. Depending on the severity levels of failure probabilities and consequences, various risk categories can be identified. The in-service inspection requirements of equipment are defined on the basis of these risk categories (Fig. 7).

RI-ISI has been applied for Primary Heat Transport System of Indian Pressurised Heavy Water Reactor (PHWR) and CT1-HT1 tower of Hydrogen Sulphide (H₂S) based Heavy Water Plant (HWP).
During the last several years, nuclear industry has recognized that Probabilistic Safety Assessment (PSA) has evolved to be more useful in supplementing traditional engineering approaches in reactor regulation. In RI-ISI, consequence of failure of a component is expressed in terms of Conditional Core Damage Probability (CCDP). EPRI has designed a Risk matrix with different inspection categories, depending on the CDF/CCDP values and degradation mechanism for determining the inspection interval.

The inspection currently followed at Indian PHWR is partly based on CAN standards, ASME Section IX and expert judgment derived from operating experience. Performance of RI-ISI over the current inspection has been studied through a risk impact analysis. CAN standards has categorized Steam generator and pump in high category which demands a 5 year inspection period. From RI-ISI consideration, they can be put in low and medium category, which allows for a relaxation in the inspection interval. Similar reduction in inspection activities were observed for other components.
(ii) **RI-ISI of HWP**

The production of Heavy Water in Hydrogen Sulphide (H$_2$S) based Heavy Water Plants (HWPs) involves handling of large quantities of H$_2$S, which is toxic and highly inflammable. The In-Service Inspection (ISI) practices in these plants are based on engineering judgement and operating experience. Recently, an ISI plan has been formulated based on operating experience, ASME Sec XI guidelines, Atomic Energy (Factories) Rules and API 510 guidelines. Consequence quantification for flammable, explosive, toxic release causing environmental impact and economic loss has been provided in Base Resource Document by American Petroleum Institute, API 581 as simple factors, which were not available in ASME code. In fact, API 581 builds upon the ASME efforts to develop usable tools that can provide the benefit of Risk Based Inspection (RBI) with a reasonable expenditure. API 581 deals ISI requirement for petroleum refineries and petrochemical industries. However, this can be safely applied to heavy water plants with certain degree of accuracy based on statistical input from past experience of plant operation and a few technical judgements.

Risk impact analysis has been conducted to study the benefit of applying Risk Based Inspection methodology. Current inspection plan puts almost 50% of its equipments considered in pilot studies in high category. But RBI puts on 4% of equipments in high category at the end of 15 years and 20% of equipments in high category at the end of 25 years. It has been found that large quantum of inspection is reduced by placing the equipments in inspection category based on RBI methodology (Table 2).

These two pilot studies indicate that RI-ISI has very high potential in reducing inspection requirements compared to the requirements as per the adopted standards.

**Neural Networks Based Diagnostic System for Identification of Accident Scenarios**

Nuclear power plants are highly complex systems that are operated by human operators. When faced with abnormal operating conditions, such as a plant accident scenario, equipment failure or an external disturbance to the system, the operator has to carry out a series of decisions based on the information available.

<table>
<thead>
<tr>
<th>Items</th>
<th>Quantity</th>
<th>Consequence category</th>
<th>Failure Probability category</th>
<th>Risk</th>
<th>Current ISI</th>
</tr>
</thead>
<tbody>
<tr>
<td>Towers</td>
<td>2 Nos.</td>
<td>H</td>
<td>H</td>
<td>H</td>
<td>H</td>
</tr>
<tr>
<td>Process HX</td>
<td>4 Nos.</td>
<td>M</td>
<td>VH</td>
<td>VH</td>
<td>M</td>
</tr>
<tr>
<td>Coolers, Chillers, Steam Heater</td>
<td>5 Nos.</td>
<td>M</td>
<td>VH</td>
<td>VH</td>
<td>M</td>
</tr>
<tr>
<td>Gas Lines</td>
<td>6 Nos.</td>
<td>H</td>
<td>M</td>
<td>M</td>
<td>H</td>
</tr>
<tr>
<td>Liquid Lines</td>
<td>30 Nos.</td>
<td>2H 28M</td>
<td>4H 24M</td>
<td>27 VH</td>
<td>1H 2M</td>
</tr>
</tbody>
</table>

Table 2: Comparison of risk categories based on current ISI and Risk Informed-ISI

H- High, VH- Very High, M- Medium, L- Low
out diagnostic and corrective actions. The conditions that arise as a consequence of the disturbance may subject the operator to various types of stresses and these stresses may contribute to inappropriate and untimely actions that aggravate the situations. This necessitates the need for developing an operator support system to assist the operator to identify such accident scenarios during the early stages of their development. Early detection will help in minimizing or even mitigating the consequences of such transients.

A neural network based diagnostic system for identification of accident scenarios in 220 MWe Indian pressurized heavy water reactors (PHWRs) has been developed for operator support and accident management. The objective of one such system, the plant diagnostic system, is to give the plant operators appropriate inputs to formulate, conform, initiate and perform the corrective actions in any potentially unsafe scenario that may arise in the plant. The aim of the diagnostic system is to identify the plant condition from the process parameters, using intelligent tools. As a pilot study, large break LOCA in reactor inlet and outlet headers, and with and without the availability of ECCS has been analyzed.

The significant parameters are shown with red background for quick attention of the operator. The trend of important parameters and appropriate operator actions are also displayed under abnormal conditions.

Ageing Studies

Today we find that there are a significant number of nuclear power plants all over the world which have seen operation over 25-30 years. This means that the life of some of the components is approaching the end of their design life. Hence, it is very much essential to ensure that the components perform as intended during the remaining period of their life. In addition, life extension of NPPs is also being contemplated. In the light this Ageing studies play...
a significant role. R&D activities related to the ageing of control and instrumentation components and cables are taken up as these constitute a significant part of replaceable components in NPPs.

**Thermal & Radiation Ageing and Loss Of Coolant Accident (LOCA) Qualification Studies of Control & Instrumentation (C&I) components and Equipment**

C&I components and equipment in various systems in NPPs may be subjected to harsh environmental conditions like high humidity, high temperature and high radiation during their designed life. Hence, it is essential to ensure reliable operation of these components during their designed life. Towards this objective, qualification approval and ageing studies on hardware systems/components/materials, used in NPPs, are needed to provide reasonable assurance regarding their survival capability under simulated environment even at the end of specified service life. Various facilities have been designed, developed and are in operation to carry out (i) thermal ageing (under accelerated conditions), (ii) radiation ageing and (iii) LOCA environment qualification studies of C&I components, equipment and cables.

Accelerated thermal and radiation ageing studies are carried out to estimate the life-spans of these items at operating temperature, in reactors, using Arrhenius Model. Accelerated temperature level should be selected in such way that no new failure mechanism, which is not prevailing during the natural ageing, is introduced during the course of ageing studies. Thermal ageing is done in thermal chambers whose temperature can be controlled according to the need.

The cumulative radiation dose the component is likely to receive during its life span is considered in radiation ageing studies. Radiation ageing is carried out using the ISOMED facility of BRIT available in Trombay. Usually C&I components, equipment and cables are subjected to an integrated gamma dose of 100 M rads to take care of the 40 years life of cable and occurrence of LOCA at the end of 40 years. It may be noted that many of the C&I components/equipment do not get exposed to such a high radiation dose. In addition to the dose, dose rate is also controlled to be around 0.1 M rad/hour. Thermal and radiation ageing are done sequentially. However, in a real environment, the components undergo both thermal and radiation ageing simultaneously in a humid environment.

In order to study interaction effects of combined environments, prevailing simultaneously in NPP containment, synergism simulator has been set up. This facility consists of temperature & humidity chamber, gamma radiation source along with a provision for applying electrical stresses. It is possible to vary magnitudes of these stresses as per design of experiment. Temperature can be varied from room ambient to 80°C with relative humidity up to 95±5%. However, temperature can be varied from room ambient to 150°C when used as temperature chamber alone. Dose rates can be varied from 2 to 30krad/hr. It is also possible to study dose rate effects. Provision has been made for on-line measurement of performance parameters. Dose rate outside the synergism simulator can be varied from 1 to 900krad/hr depending upon the distance of test items with respect to source.
In LOCA environment qualification studies, C&I components/equipment and cables are subjected to high humidity and high temperature that simulates the environment the components are likely to encounter, if at all LOCA occurs. LOCA qualification studies are carried out in LOCA simulation test facility that is designed by Reactor Safety Division. The facility is very useful to improve quality and reliability of critical items. Desired LOCA temperature/pressure test profile is achieved by automatic control through Programmable Logic Controller (PLC) using Proportional Control Valves (PCVs) to control the steam flow. The desired LOCA test profile is generated, by NPCIL, from thermal hydraulic calculations of LOCA in the reactor. A typical
test profile is shown in Fig.12. Maximum steam temperature and pressure achievable in the LOCA chamber are 150°C and 3.4kgf/g (50 psig) respectively. Provision has been made for on-line measurement of performance parameters of the items being tested inside the LOCA chamber. Humidity Generation and control inside the LOCA simulation test facility is also possible to expose the items to the controlled humidity level inside the LOCA chamber. LOCA test facility is useful for testing small components (volume of the chamber is 1 m³). For testing larger components like Primary Heat Transport pump motors, a larger facility has been set up at Electrical Research & Development Association (ERDA), Vadodara under BRNS Project. These facilities are constantly being used by various users mainly DAE units. Services are being provided to upcoming and operating Nuclear Power Stations regularly in the areas such as (i) Standardisation of new engineering hardware and their procurement, (ii) Estimation of residual life, (iii) Failure analysis and reliability improvement and (iv) Import substitution.

**Augmentation of Test and Measurement Facilities for Ageing Studies**

Measurement of performance parameters of components and equipments is also very important activity associated in ageing research studies to judge whether item is ready to perform its intended function under prevailing conditions.

Nuclear Power Plants (NPPs) contain myriads of electrical cables (insulated with some form of polymeric insulation) of various sizes and voltage ratings. Failure of cables is primarily due to the hardening and embrittlement of the insulation resulting in the formation of micro cracks, loss of dielectric strength and high leakage currents. Percentage elongations-at-break (E-at-B) are derived from measurements by tensile tests on cable insulation materials to establish the qualified life. Monitoring cable degradation by measuring E-at-B is in many cases not feasible. Correlation study of the E-at-B with the physical/chemical deterioration of the insulation and jacket materials is an important element and it helps in predicting the cables life. E-at B is generally correlated with Oxidation Induction Time (OIT), weight loss in thermogravimetric analysis and Indenter Modulus (IM).

As explained above, E-at-B and Tensile Strength are basic parameters for prediction of life of cables. Tensile test is performed in accordance with ASTM-D2633-82, using a Universal Testing Machine (UTM) equipped with pneumatic grips and having an extensometer clamped to the sample. Special tensile specimens (dumbbells for larger cables or cylinders for smaller cables) of the insulation or jacket materials without the copper conductor are used for these tests. These tests are destructive, and therefore, many samples are required if tests are conducted regularly.

Thermal analysis on microsamples of cables is mainly non-destructive and is linked to the level of antioxidants present in the polymeric materials. Ageing of cable materials is evaluated by measuring the period of time before a small sample of insulation experiences rapid oxidation when subjected to a constant elevated temperature in an oxygen atmosphere. OIT measurements on artificially aged
specimens are useful to formulate life estimation criteria based correlation with E-at-B values. OIT is carried out using Differential Scanning Calorimeter (DSC) shown in Fig.14. A correlation of E-at-B with OIT has been established for some I&C cables

Ageing affects both the steady state (calibration) and dynamic (response time) performance of sensors (i.e. Resistance Temperature Detectors (RTD), Thermocouples and Pressure Transmitters, solenoid valves, Pressure Switches etc.). Multifunction calibrator (Fig.16), designed by RSD, is being used for calibrating and monitoring the performance parameters of various process instruments during thermal & radiation ageing studies and LOCA simulation testing of the equipments. One of the important features of this setup is to simulate transmitter input and measure transmitter output.

Apart from above test facilities, various test and measuring instruments are also available in Ageing Research Laboratory in RSD, Hall No. 3. Some of them are; Fully Automatic Capacitance and Dissipation Factor (C&TanDelta) Measuring instrument, Fully Automatic AC Dielectric Breakdown Tester, Insulation Resistance Tester,

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Fig. 13: Universal Testing Machine

Fig. 14: Differential scanning calorimeter

Fig. 15: DSC thermogram and illustration of DIT Measure

Fig. 16: Multifunction calibrator pneumatic & hydraulic test pumps and pressure gauges regulators and Indicators along with Mimic Panel
In addition, instruments like Time Domain Reflectometer for diagnosing cable faults are being procured under XI plan.

Challenges Ahead

In recent years, Nuclear Power Plants have seen a sudden increase in the deployment of digital systems for instrumentation & control functions. This is seen in new NPPs as well as in older plants (replacement of previous analog systems). This analog-to-digital preference is largely driven by the fact that (1) digital systems offer better performance and additional features compared to analog systems, and (2) analog replacement parts are becoming increasingly difficult to obtain. While digital technology has the capability to improve operational performance, the introduction of this technology opened up new challenges for the treatment of these systems in Probabilistic Safety Assessment (PSA). In particular, these challenges include (1) rapid changes in technology used in digital systems (2) new failure modes associated with digital technology; (3) modeling of software and (4) handling the interaction of hardware and software failures.

Traditionally, hardware failures are evaluated using MILSTD 217 F (N2) and its upgrades. But the rapid change in technology and improvements force technologists to look for alternate techniques in the absence of operating experience data. Recently Physics of Failure models are extended to these semiconductor devices, focusing much on the degradation mechanisms and precipitating them through suitable tests. Physics of Failure models looks into modeling the degradation mechanisms possible depending on material of construction, use environment, etc. and using this information in estimating the failure rate. Research is in progress, in Reactor Safety Division, to develop the Physics of Failure models for various degradation mechanisms.

Similarly, modeling of software reliability is a crucial task, and reliability community has still not come to consensus on the standardization of approach to be used. Extensive verification and validation is carried out on software, which qualitatively assures its quality. However, PSA needs a quantitative estimate of software reliability. Applying reliability growth models is one of the techniques for software reliability quantification, which was in practice for software having some failure experience information. However, this approach cannot be extended to new software and more ever it doesn’t give any credit to the software development process. Recent trend is to account for software development process, quality standards used, software engineering principles adopted etc. in reliability prediction. This has resulted in harnessing the strength of software engineering metrics for reliability prediction. Research is underway in identifying the critical quality metrics that have bearing on reliability prediction and employing them for estimating reliability. This is also pursued under XIth plan project at Reactor Safety Division.

References


