STUDIES ON APPLYING RISK INFORMED IN-SERVICE INSPECTION FOR INDIAN NUCLEAR POWER PLANT AND HEAVY WATER PLANT

G. Vinod
Reactor Safety Division

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What is risk informed IN-SERVICE INSPECTION?

Risk Informed In-Service Inspection (RI-ISI) focuses the utilization of risk quantification in formulating an ISI plan, thereby emphasizing the importance of surveillance and maintenance activities on plant risk. This methodology has been successfully adopted by Oil & Gas industry and Chemical plants, to that extent, they were able to establish an effective structural integrity management programme, which reduces plant down time, industry and regulatory burdens and continues to maintain plant safety. Risk informed methodology has started catching up in nuclear industry as well wherein Probabilistic Safety Assessment (PSA) technique is widely used, for quantifying risk to a plant. PSA results were able to provide insights into myriads of decision making issues such as optimizing Surveillance Test Interval, Allowable Outage Time etc. Since these parameters are directly involved in analyzing the unavailability of safety systems, solutions are provided in a fairly direct manner. But the problem becomes slightly complicated while suggesting an optimum inspection interval for critical components, involved in In-Service Inspection. This paper attempts to traverse through these issues and provides a suitable solution.

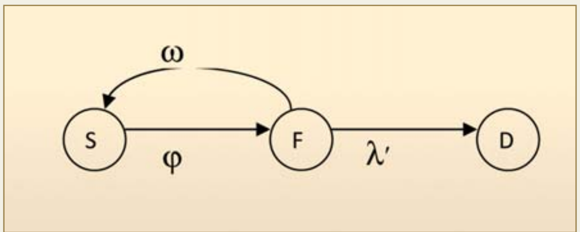
Risk can be defined as the likelihood frequency of an event multiplied by the consequence of the event. Extending this definition of risk in RI-ISI [1,2], two factors, viz., failure frequency of the component and its failure consequence are required, for applying this methodology. The In-Service Inspection plan has direct impact on failure frequency of component. After estimating the failure frequency, the next issue involves the selection of systems / components to include the RI-ISI programme. In Nuclear Power Plants (NPPs), consequence is quantified using PSA methodology, whereby the Core Damage Frequency (CDF) is computed. Electric Power Research Institute (EPRI) has suggested a Risk Matrix for RI-ISI, which includes the failure frequency of the components versus its conditional core damage probability. The estimation of Conditional Core Damage Probability (CCDP) is dealt in detail in section 4. Risk Matrix has suggested 7 inspection categories, depending on risk level, which determines the inspection interval, scope of inspection and type of inspection. Many components can fall in the same category, which specifies a period of inspection interval. Proper optimization techniques should be adopted, in deciding the components to be taken up in each year, with respect to cost of inspection, risk level of plant and permissible radiation exposure. These issues and an appropriate solution are summarized below:

1. How to select the component for inspection? (Find a suitable importance measure)
2. How ISI affects component? (Use Markov model)

3. What to do when many components fall in same category? (Formulate optimization problem).

How In-Service Inspection affects the risk of plant?

In-Service Inspection checks for the structural integrity of piping components, using various Non Destructive Testing (NDT) techniques, to detect the presence of flaws or cracks in the component, which can lead to undesirable situations like leak or rupture. Failure of the piping can be termed as the probability that the component reaches these undesirable states. Even if the piping reaches the leak state, repair activity can be carried out during in-service inspection, so that, it can be made functional. In other words, in-service inspection decides the time-dependent failure probability of the component. The NDT technique used during ISI, also decides the probability of detection of flaw that can lead to undesirable state. These aspects have been efficiently handled using Markov model [3,4], so that a realistic estimate of piping failure probability can be used.



Markov model for single failure state

Piping system state

S = Success state

F = Flaw State

D = Degraded (Leak or Rupture)

State Transitions

φ = Occurrence of flaw

ω = Inspect and Repair Flaw

λ' = Occurrence of degraded state = $\lambda_L + \lambda_R$

λ_L = Frequency of leakage

λ_R = Frequency of rupture

φ, the frequency for initiation of flaw can be chosen based on the operating experience or expert judgement. Inspect and repair flaw rate, ω is defined as

$$\omega = P_{f1} P_{FD} / (T_I + T_R) \qquad (1)$$

P_{f1} = probability that piping element with a flaw will be inspected per inspection interval. The value will be 1 if it is in the inspection programme, else it will be 0.

P_{FD} = probability that a flaw will be detected given this element is inspected. This is the reliability of inspection programme and equivalent to Probability of Detection. For most Non Destructive Examinations, the value is between 0.84 and 0.95.

T_I = Mean time between inspections for flaw, is typically 10 years for nuclear power plants

T_R = Mean time to repair once detected, is in order of days, 200 hrs.

Which System to select?

Importance measures have established their credibility in decision making issues. But the conventional importance measures are designed for applying in components from safety systems, which appear in the form of unavailability in PSA analysis. Since RI-ISI principally deals with the components from process systems, which appear as failure frequency in PSA analysis, these conventional importance measures fail to provide suitable solution. Borgonovo [5] has suggested a new measure, Differential importance measure, principally based on sensitivity methods, which has been found suitable for RI-ISI.

Differential Importance Measure (DIM) can be defined as the fraction of total change in Risk that is due to a change in parameter xi.

$$DIM_{x_i} = \frac{\delta R_{x_i}}{\delta R}$$
$$\delta R_{x_i} = R(x_i + \delta x_i) - R_0 \qquad (2)$$
$$\delta R = \sum_i \delta R_i$$

where R is the risk or Core Damage Frequency from PSA model x_i is the component appearing in the PSA model.

A detailed discussion of this measure with its application in RI-ISI can be found in reference [6].

Consequence Analysis and Risk Matrix Categorization

The risk information provides an estimate of consequence of failure of a component, using PSA models. Consequence can be quantified through the estimation of Conditional Core Damage Probability (CCDP). The risk from failure of a component on failure of a specific joint i (eg. a particular weld) is expressed in terms of $CCDP_i$. The consequence evaluation group is organized into two basic impact groups: **(i) Initiating Event and (ii) Loss of Mitigating Ability [1,7].**

In *Initiating Event (IE) impact Group*, $CCDP_i$ can be directly obtained from the PSA results, by dividing the CDF due to the specific IE by the frequency of that IE.

$$CCDP_i = \frac{CDF_{\text{due to IE}}}{IE_{\text{frequency}}} \qquad (3)$$

In the loss of mitigating ability group, the event involves the component failures in safety system. Safety system can be in two configurations, standby and demand. In standby, $CCDP_i$ is

$$CCDP_i = [CDF_{(qi = 1)} - CDF_{(BASE)}] * T_E \qquad (4)$$

and in demand

$$CCDP_i = [CDF_{(qi = 1)} - CDF_{(BASE)}] * T_t \qquad (5)$$

Where,

$CDF_{(qi = 1)}$ = CDF given the component failure from joint i

$CDF_{(BASE)}$ = Base or reference CDF

q_i = component failure/unavailability due to failure from joint i.

T_E = Exposure Time (Detection time + AOT)

T_t = Mean time between tests or demands.

Risk matrix is defined as the decision matrix that is used to categorize the components based on degradation mechanism and consequence of its failure (Fig. 1). From international experience [3], a basis has been established, for ranking components rupture

LIKELIHOOD FREQUENCY (/yr)	CONSEQUENCE CATEGORY (CCDP)			
	None $<10^{-8}$	Low $10^{-8} < CCDP < 10^{-6}$	Medium $10^{-6} < CCDP < 10^{-4}$	High $>10^{-4}$
High ($>10^{-4}$)	Low 7	Medium 5	High -3	High –1
Medium ($10^{-7} < F < 10^{-4}$)	Low 7	Low 6	Medium 5	High – 2
Low ($<10^{-7}$)	Low 7	Low 7	Low 6	Medium 4

Fig. 1 : EPRI risk matrix

potential as High, Medium or Low, simply by understanding the type of degradation mechanism present. Each component is assigned the appropriate category depending on its ΔCDF and degradation mechanism. Each category specifies inspection programme strategy, which will include prioritisation of components for inspection, inspection interval, inspection method and scope/volume of inspection.

How to optimize?

Many components may fall in the same inspection category. An optimum plan should be devised subjected to constraints such as risk to plant, cost of inspection and radiation exposure to workers, if the component is in radioactive area. A typical optimization problem can be defined as

$$\lambda_{system} = \sum_{i=1}^{n} \lambda_i \quad (6)$$

Subject to constraints such as

$$C_j = \quad (7)$$

$y_i = 1$ for $j = x_i$, where x_i is the year of allocation of the feeder. Else, $y_i = 0$

$$E_j = \sum_{i=1}^{n} e_i y_i \leq E_{max} \quad (8)$$

$y_i = 1$ for $j = x_i$, where x_i is the year of allocation of the feeder. Else, $y_i = 0$
Where,

n is the number of the components in ISI programme.

C_j and E_j are constraints for cost and exposure time respectively.

The objective function itself is an implicit constraint

Case Studies in RI-ISI

(i) Heavy Water Plant- Kota

A pilot study has been carried out for first pair (CT1-HT1) of first stage of exchange towers including gas and liquid loops. Failure data collected from HWP (K) have been statistically analyzed by Bayesian updating technique and generic values are taken from API 581 [8]. ANSI /ASME B31G model has been used, to estimate the remaining strength of pipeline containing corrosion defects. First Order Reliability Method (FORM) has been used for reliability analysis of pipelines [9]. Detailed sensitivity analysis was also carried out, for identifying various critical parameters. For consequence analysis, sample quantitative factors from guidelines of API 581 have been used for consequences, due to toxic release and fire.

The study demonstrates that even at 25 years of plant life, all the Process Heat Exchangers, Coolers, Chillers, Steam Heaters and the liquid pipe line (barring one) poses Medium Risk mainly due to medium consequences. Study also demonstrates that the Towers and large diameter gas lines, fall in the High Risk Category. Prima-facie the study indicates that there is scope for optimization in the existing ISI Plan. It would be proper to put a cautionary note that the pilot study has considered certain failure models, damage mechanisms, likelihood and consequence factor, analytical technique etc. that may be improved upon, in the detailed study.

With this study, it is strongly felt that by amalgamation of ASME and API approach, statutory requirements, past experience of HWB and generic data available for similar plants, the ISI can be optimized.

(ii) Indian PHWR

In the systems considered, any failure in the components present in PHT will be an initiating event, viz., Loss of Coolant Accident (LOCA). Depending upon the area of leak or rupture, they are classified

Table 1: Results of the pilot study on RBI of Heavy Water Plant

Items	Number of items	Consequence category	Failure probability category[1]		Risk		Current ISI [10]
			15 Yrs	25 Yrs	15 Yrs	25 yrs	
Towers	2 Nos	H	H	VH	H	H	H
Process HX	4 Nos	M	VH	VH	M	M	H
Coolers, Chillers Steam Heater	5 Nos	M	VH	VH	M	M	H
Gas Lines	6 Nos	H	M	H	M	H	4H 2M
Liquid Lines	30 Nos	2H 28M	4H 24M 2L	27 VH 1H 2M	28M 2L	1H 29M	10H 20M

either as small or large LOCA events. In the case of Small LOCA, CCDP comes from the accident sequence involving failure of Reactor Protective system, where as in the case of Large LOCA, CCDP comes from accident sequence involving failure in injection and recirculation of Emergency Core Cooling System (ECCS).

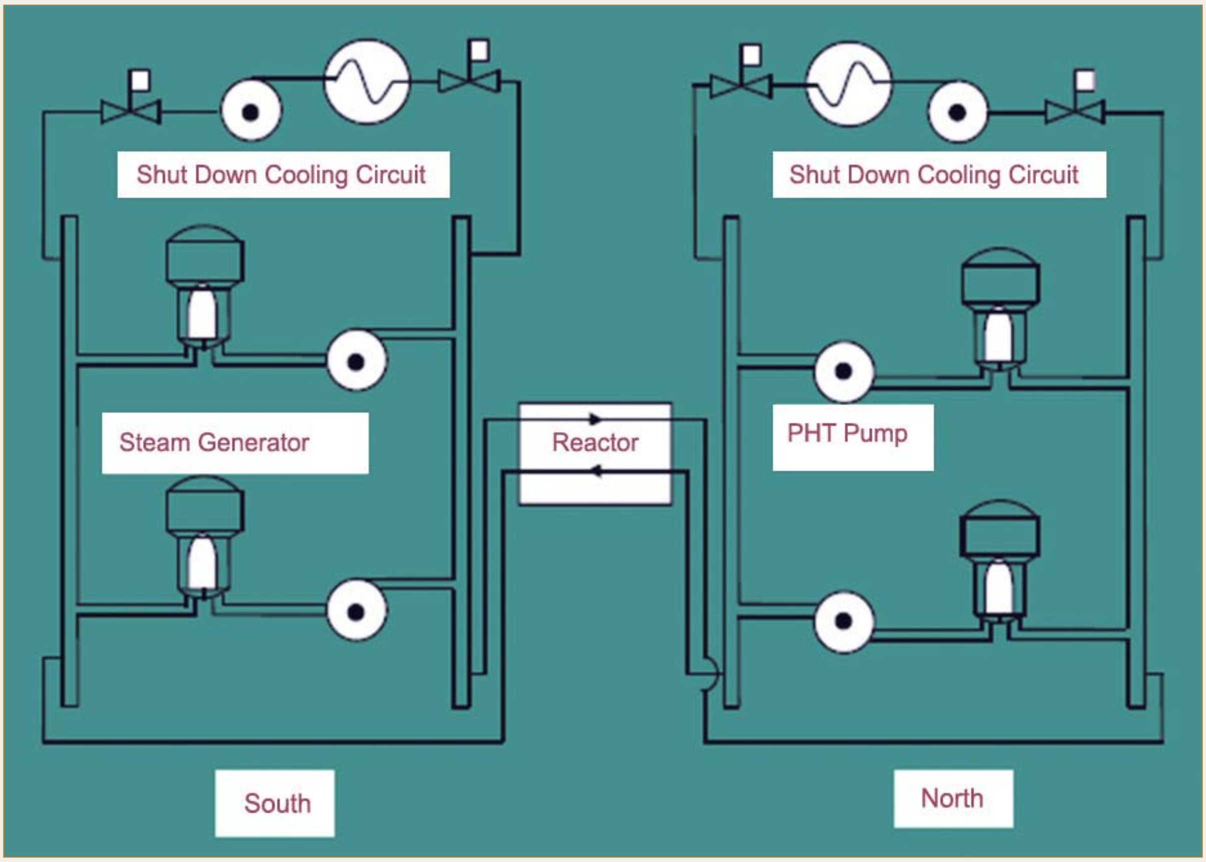


Fig. 2 : Schematic of PHTS and SDCS

For the shut down cooling system, pipelines upto the first isolation valves (MV1, MV2, MV3 and MV4) are connected with PHTS. So any leak or rupture in these pipelines will be equivalent to a LOCA event, and treated likewise. After these isolation valves, components will be in an on-demand configuration. Here, mean time between the tests / inspection interval or demand is to be considered.

Risk impact analysis essentially carries out a comparison between the existing inspection programme and risk-based inspection programme. For the components considered in the case study, these aspects were compared and summarized in Table 2.

Conclusions

Risk Informed In-Service Inspection is in itself an efficient methodology to reduce the inspection resources and plant down time. In the case of nuclear power plant, radiation exposure is also another limiting factor. By employing RI-ISI, the inspection can be focused on the high risk significant systems / components, thereby avoiding unnecessary radiation exposure during the inspection of low risk significant components. Safety is a prime concern in the nuclear industry, which can be suitably contained by the ISI programme suggested using RI-ISI methodology. There are lot of inherent uncertainties involved in PSA analysis, from the point of view of data and modeling, which needs to be properly addressed for successful implementation of RI-ISI.

Table 2 : Risk impact analysis

Sl. No:	COMPONENT DESCRIPTION	CURRENT CATEGORY	RISK CATEGORY
1	Piping from Outlet Header – Steam Generator	B	5
2	Piping from Steam Generator – Primary Coolant Pump	B	5
3	Piping from Primary Coolant Pump Valve	B	6
4	Piping from Primary Coolant Pump – Inlet Header	B	5
5	Steam Generator	A	6
6	Primary Coolant Pump	A	5
7	Reactor headers (RIH/ROH)	A	3

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About the Author

Dr. (Ms.) Gopika Vinod, a graduate of 37th batch of BARC Training School, joined Reactor Safety Division, BARC in 1994.

She has made significant contributions in the area of Probabilistic Safety Assessment (PSA). She has actively participated in the PSA Studies of AHWR and research reactors, Fire PSA, Risk Informed In-Service Inspection methods for piping systems in DAE facilities and in the development of Risk Monitor.

She has conducted extensive research on applicability of Risk Informed In-Service Inspection methods which have resulted in formulating a complete framework. She has successfully translated the research methods into the application framework. The most important contributions are the introduction of a new importance measure, incorporating the effects of In-service inspection through markov model and using genetic algorithm for optimizing inspection. She could later extend her research for chalking out the RI-ISI programme for nuclear power plants as well as in heavy water plants. She has co-authored a standard for applying the risk-based inspection and maintenance procedures, for European industry.