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RISK BASED ASME CODE REQUIREMENTS

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RISK BASED ASME CODE REQUIREMENTS

by

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Abstract

The objective of this ASME Research Task Force is to develop and to apply a methodology for incorporating quantitative risk analysis techniques into the definition of in-service inspection (ISI) programs for a wide range of industrial applications. An additional objective, directed towards the field of nuclear power generation, is ultimately to develop a recommendation for comprehensive revisions to the ISI requirements of Section XI of the ASME Boiler and Pressure Vessel Code. This will require development of a firm technical basis for such requirements, which does not presently exist. Several years of additional research will be required before this can be accomplished.

A general methodology suitable for application to any industry has been defined and published. It has recently been refined and further developed during application to the field of nuclear power generation. In the nuclear application probabilistic risk assessment (PRA) techniques and information have been incorporated. With additional analysis, PRA information is used to determine the consequence of a component rupture (increased reactor core damage probability). This allows direct quantification of the risk associated with a rupture, as the product of expected rupture frequency times consequence, and ultimately allows risk prioritization of all pressure boundary components in a nuclear power plant. A procedure has also been recommended for using the resulting quantified risk estimates to determine target component rupture probability values to be maintained by inspection activities. Structural risk and reliability analysis (SRRA) calculations are then used to determine characteristics which an inspection strategy must posess in order to maintain component rupture probabilities below target values. The methodology, results of example applications, and plans for future work are discussed.

Introduction

The multi-disciplined ASME Research Task Force on Risk-Based Inspection Guidelines (Table 1) was formed in 1988 to address the general problem of defining a methodology for formally incorporating quantitative risk analysis techniques into the prioritization of pressure boundary and structural components for inspection. The reason for this task force effort is, that where in-service inspection (ISI) programs exist, they are based primarily on prior experience and engineering judgement, with at best an implicit consideration of risk (probability times consequences).

After review of the risk analysis field as applied to a wide variety of industries, the task force concluded that appropriate analysis methods exist for analyzing and quantifying risks associated with pressure boundary and structural failures. Subsequently, it recommended a general methodology for establishing a risk-based inspection program for any facility or structural system (ASME 1991).

During the past few years this general methodology has been further focused, and applied to address the inspection of nuclear power plant components. In particular, the use of information from probabilistic risk assessments (PRAs), which have now been produced for many nuclear power plants, has been incorporated into the methodology to improve the quantification of risks associated with component pressure boundary and structural failures (ruptures). In this process the task force has been significantly aided by research conducted at Pacific Northwest Laboratory (Vo et al., 1989; Vo et al., 1990; Vo et al., 1991) which was funded by the U.S. Nuclear Regulatory Commission (NRC). This work developed and pilot tested key methodology steps which were subsequently incorporated into the recommendations of the task force. The NRC also provided direct financial support to the task force, and was its first direct sponsor (Table 1).

The incorporation of PRA information into the general methodology has allowed the quantification of risks associated with the potential rupture of each of the pressure boundary components in the plant primary cooling system, as well as in important support systems and emergency safety systems. This allows the prioritization of these components for inspection activities on the basis of the risk associated with component rupture. A procedure has also been recommended for determining target component rupture probability values (to be maintained by inspection activities) from these quantitative risk estimates. Finally, the method for determining the characteristics which an inspection strategy must possess in order to maintain target rupture probabilities has been defined.

This paper describes the major features of this methodology, and recent progress in efforts to develop it further and to apply it. The task force report "Risk-Based Inspection - Development of Guidelines, Volume 2 - Part 1, Light Water Reactor Nuclear Power Plant Components" is presently in the publication process at ASME. That document is the second volume in a planned series of task force reports focusing on specific technologies including nuclear power, fossil power and petroleum refining and storage. It will

TABLE 1. Risk-Based Inspection Guidelines Task Force Members

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Sponsors (in order of sponsorship)

U.S. Nuclear Regulatory Commission

ASME Council on Codes and Standards

National Board of Boiler and Pressure Vessel Inspectors

PVRC Welding Research Council

American Nuclear Insurers

Industrial Risk Insurers

The Hartford Steam Boiler Inspection and Insurance Company

American Petroleum Institute

U.S. Department of Energy

National Rural Electric Cooperative Association

Oil Insurance Limited

Edison Electric Institute

present the methodology in more detail, and it will elaborate on various aspects of the required analyses and example applications.

In subsequent years, when the methodology has been fully developed and applied to produce recommendations for a complete nuclear power plant ISI program, they will be published in a "Volume 2 - Part 2" document. The ultimate objective of this effort is to produce a recommendation for comprehensive revisions to the in-service inspection requirements of Section XI of the ASME Boiler and Pressure Vessel Code. It is expected that such recommendations will be presented in that document.

In a parallel effort the task force has also begun to study the application of the general methodology to fossil fueled power plants. It is expected that this study will culminate in publication of a third volume in the "Risk-Based Inspection - Development of Guidelines" series. This application is discussed in the last section of this paper preceding the Summary and Conclusions.

Risk-Based Prioritization of Nuclear Plant Components for Inspection

The first major step in the methodology is the selection and risk-based prioritization of components for inspection. This is performed by combining information from PRAs with probabilities of pressure boundary and structural failures, using a modified Failure Modes, Effects and Criticality Analysis (FMECA) procedure. It is a fairly extensive operation, requiring both the estimation of failure probabilities and additional analysis of system fault trees and cut sets from PRAs. The risk measure used for prioritization is the same as that calculated in Level 1 PRAs, namely core damage frequency (CDF). This procedure is detailed in (Vo et al., 1989).

The reason why extensive reanalysis of PRA results is required in order to perform this risk prioritization is that published PRAs do not include sufficient details in the accident sequence and cut set information presented to allow direct calculations using standard risk importance analysis methods. PRAs focus on the dominant contributions to plant risk, namely the cut sets making the highest contribution to CDF. Cut sets having a CDF value less than a cut-off value are basically ignored. Due to the small annual component rupture probability, cut sets involving ruptures generally have CDF values below the cut-off, and are excluded. Consequently, it is necessary to use indirect methods to calculate the risk importance of component ruptures.

The risk per year of core damage resulting from rupture of any component is the product of the component's annual rupture probability, times the conditional probability of system failure given rupture, times the conditional probability of core damage given system failure. For this methodology it is recommended that the annual rupture probability be determined from an expert elicitation process similar to that developed for NUREG-1150 (USNRC 1990). This is because structural failures and ruptures are rare events, and historical data provide only a limited basis for estimating annual rupture probabilities. While structural risk and reliability analysis (SRRA) calculations could be used, they are too time consuming and expensive to perform individually for all components. The expert elicitation process requires care in enlisting a suitable panel of experts, in training of experts, in preparing the panel to provide responses to a collection of well-posed questions, and in allowing time for experts to document the rationale for their estimates. The experts need to apply a broad base of experience with structural integrity issues at operating plants, and an understanding of the response of structural materials to service environments.

At the present time this methodology is being applied at the Surry-1 plant. Two separate elicitation panel workshops have been held, due to the large amount of work required for the panelists to address all components in each system. This elicitation process required the development, communication and discussion of a considerable amount of plant-specific information on numbers of welds in piping runs, location of fixed ends, tees, reducers, hangars, and thermal and vibrational information. This information was developed from plant documentation, and subsequently verified by performing system walkdowns at the plant with the assistance of plant personnel. During these walkdowns information was also gathered on the possibility of additional consequenses of rupture (beyond the system failure information obtainable from the PRA) due to jet impingement effects on other components, and potential flooding resulting from ruptures.

The first workshop elicited estimates of annual rupture probabilities for components in: the reactor pressure vessel (RPV), the reactor coolant system (RCS), the low pressure injection system (LPI), and the auxiliary feedwater system (AFW). Results are presented in (Vo et al., 1991). The second workshop, which was held in early 1992, addressed components in: the high pressure injection system (HPI), the residual heat removal system (RHR), the service water system (SW), the component cooling water system (CCW), and the power conversion system (PCS). Analysis of the results of that workshop has not yet been completed.

The conditional probability of system failure, given component rupture, is the next factor which must be determined for use in calculating the CDF risk associated with component rupture. This conditional probability of system failure is obtained by reanalyzing the system fault tree under the assumption of the rupture. This can be done even though the fault tree does not include gates addressing component ruptures, by mimicking the effect through failure of an adjacent active component. Thus, the rupture of the piping at the discharge of a pump can be mimicked by failing the pump, since both failures yield loss of flow. (Care must be used, however, since backflow from a parallel pump out of a ruptured header could yield loss of flow from both pumps unless prevented by a check valve, which is not the same as the effect of single pump failure.)

The conditional probability of core damage, given failure of the system, is the final factor needed to calculate the CDF risk due to component rupture. This factor is approximated by the Birnbaum risk importance measure (IB) for the system. It is calculated from the dominant cut sets of the PRA by assuming a failure probability of unity for the failure probability of each component of the system in question which appears in each cut set, and summing over the cut sets involved.

The risk associated with rupture of each component is then calculated in the modified FMEA analysis as the product of the annual probability of rupture, times the consequences as determined by the product of the other factors described above. The resulting product is the additional CDF risk resulting from the possibility of rupture of the component in question. This allows not only the opportunity to rank individual components, but also to quantitatively relate the risks associated with each. This may then be used to apportion inspection efforts for each component according to its contribution to risk.

A pilot test of this methodology was performed to risk-prioritize the major piping segments in the emergency feedwater (EFW) system at Oconee Unit-3 (Vo et al., 1989). This test used fault trees and cut set information from the Oconee-3 PRA, approximate data on weld numbers and locations, and rupture probabilitities derived from historical data. The primary importance of this test was that, although it only addressed the EFW sysstem, it demonstrated the ability to calculate a quantitative value of risk due to potential rupture associated with every plant component which was addressed in the PRA. This allows the direct comparison of individual components in various systems, and answers the question, "How do you compare low importance components in high importance systems with high importance components in low importance systems?"

In addition, the pilot test demonstrated that the methodology was workable, required reasonable resource committments, and that the results were reasonable and in agreement with common-sense qualitative assessments. An unexpected result of the analysis was the identification of a single run of suction piping whose failure could fail the entire system. As a consequence, the risk priority of that piping run was determined to be in the middle of the prioritized list, instead of at the bottom as might otherwise be expected for a low-temperature, low-pressure run of suction piping (it would have been at the top of the list except for its extremely low rupture probability.

Following the pilot test, analysis efforts were directed towards determining to what extent results were generalizable, or generic, across plants. This is important to development of the methodology because the ultimate objective of this effort to develop risk based inspection guidelines for nuclear power plants is to produce a recommendation for comprehensive revisions to the in-service inspection requirements of Section XI of the ASME Boiler and Pressure Vessel Code (ASME 1989). This will require development of a firm technical basis for such recommendations, which does not presently exist. Achievement of that objective is expected to require several years of additional work applying and refining this methodology.

A system-level weld inspection importance (IW) was defined as the product of the average weld rupture probability for each system times IB, which is essentially the CDF probability given system failure. The analyses utilized the PRAs for the five plants addressed in (NRC NUREG-1150), which were performed on a common basis using similar methods, plus PRAs for three other plants. In all, six PWRs and two BWRs were analyzed (VO et Al., 1990). Although analyses of additional plant PRAs is needed to form definitive conclusions, the results indicate that it may be reasonable to group systems into categories of high, medium and low risk importance. These results also provide values of IB for systems at these plants for use in ongoing efforts to calculate and rank individual component risk importances at these plants, to determine the extent to which generic risk trends may exist on a component basis.

Component-level risk prioritization is presently being carried out for components at the Surry-1 plant. Risks associated with the rupture of individual components have been calculated for components in the four systems addressed in the first rupture probability elicitation workshop: the RPV, the RCS, the LPI, and the AFW systems.

Figure 1 identifies the most risk important components in these systems, and presents not only the risk values, but the rupture probability estimates and the conditional CDF values which comprise the risk products. The welds located within the beltline region of the reactor vessel are seen to dominate the risk, and account for almost 75% of the total CDF risk due to component ruptures. The beltline welds are followed in importance by the beltline plate material, which accounts for another 5%. The welds in the reactor vessel upper and lower heads account for another 6%, and the single AFW condensate storage tank and supply line contributes another 3%. Various welds in the LPI system (not all of which are shown on the figure) contribute another 10%, which sums to more than 99% of the total CDF risk associated with component ruptures (for the four systems analyzed to date).

Future work will complete the analysis for the Surry-1 plant by developing the risk-importances of components addressed in the second expert elicitation workshop (e.g. HPI, RHR, SW, CCW, and PCS). Subsequently, similar analyses will be done for other PWRs, and generic trends in component importances will be developed.

Target Risk and Rupture Probability Values

This part of the methodology addresses the question, "How much inspection is enough?" The philosophy upon which it is based is that inspections should ensure that the risk of core damage resulting from pressure boundary and structural failures is maintained less than a small fraction of the total core damage risk estimated in the PRA. This risk due to pressure boundary and structural failures is referred to as the "target risk," and 5% of the total PRA-estimated risk resulting from internal events has been recommended as an appropriate numerical value for it.

This total target risk is then apportioned among the components and used to determine a target rupture probability for each component, which is then to be maintained by appropriate ISI activities. It is presently recommended that this target risk be apportioned among the risk-important components in proportion to the risk which has been estimated to be associated with rupture of each component. Then, by dividing this target risk for each component by the conditional probability of core damage given component rupture, an annual target rupture probability can be calculated for each component. The object of inspections, then, is to ensure that, for each component, the probability of component rupture does not exceed the target rupture probability.

A rough value for CDF risk estimated in state-of-the-art PRAs for modern facilities is about 5.0 E-5 CDF. According to this methodology, then, an appropriate total target risk value to be maintained by ISI inspections would be 2.5 E-6 CDF, or 5% of the total PRA risk.

The risk estimates developed for the Surry-1 components in the four systems analyzed to date show that the sum of these risks is slightly greater than 2.0 E-6 CDF. Thus, the total estimated risk due to ruptures is very close to the total target risk, and it may exceed the target when the remaining systems are

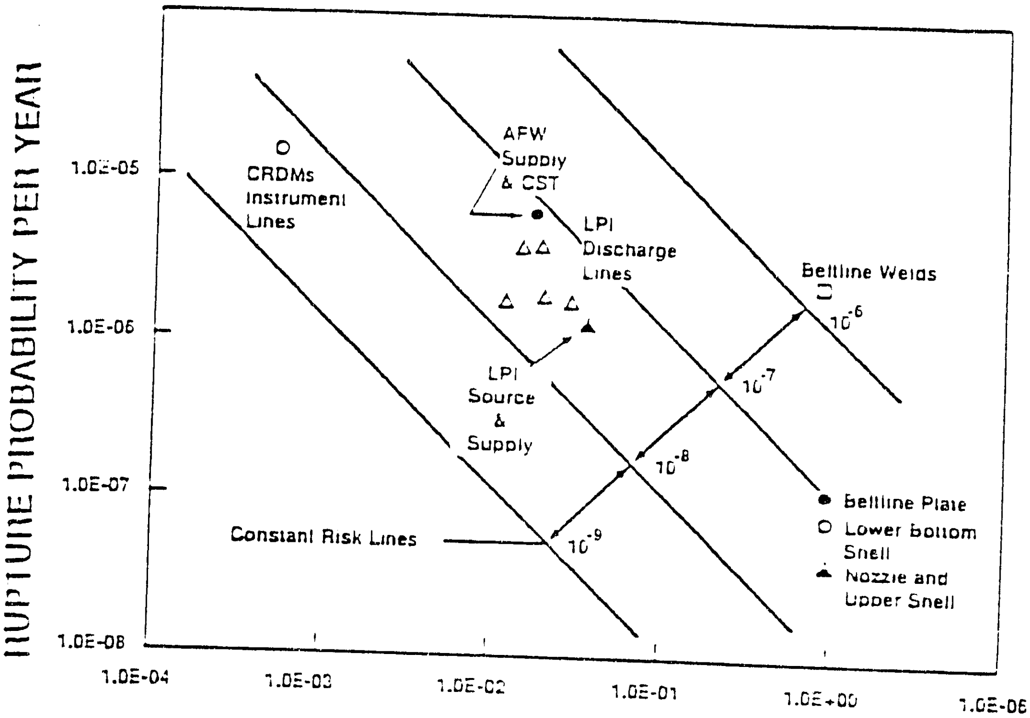

FIGURE 1. SURRY-1 Dominant Risk Results for Four Systems

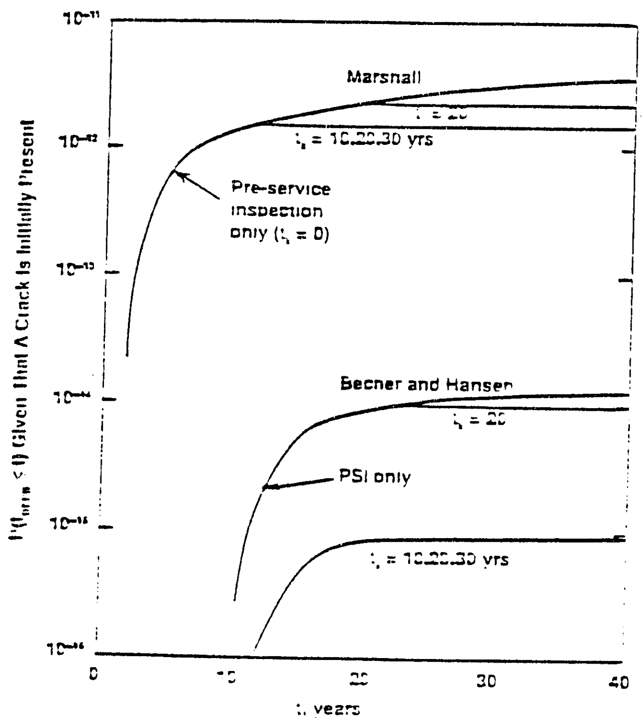

FIGURE 2. Cumulative Conditional Probability of a Double-Ended Pipe Break (DEPB) for Various Crack Depth Distributions and Inspection Schedules for an Example Case (Ref. 7)

analyzed. If so, it may be necessary to revise the recommendation for setting the target. This is an ongoing area of research at the present time.

For the Surry-1 example, however, the recommendation that the target risk be apportioned in proportion to estimated risk simply means that target risk values for individual components must be essentially the same as the estimated risk values for each component. This, in turn, means that the target rupture probability for each component must be the same as the rupture probability estimated for that component by the expert panel (since the consequences, given that a rupture occurs, are fixed by the PRA analysis). Values of these estimated rupture probabilities, which now become the target values below which inspection must hold actual rupture probabilities, may be read from the graph in Figure 1 for the most risk important components.

Examination of these target/estimated component rupture probabilities indicates that some of these targets may be more difficult to achieve than others. Future studies will also consider whether it may be appropriate to apportion the total target risk among components so as to even out the difficulties of achieving target rupture probabilities. However, until future studies addressing the practicalities of performing actual inspections are performed, the recommendation that target risk be apportioned in proportion to estimated risk seems most appropriate.

Inspection Program Development

As noted above, the object of inspections is to ensure that, for each component, the probability of component rupture does not exceed the target rupture probability. If it can be shown that, during the entire lifetime of a component, the component's expected failure probability does not exceed its target failure probability (with an acceptable degree of confidence), no inspection at all may be an acceptable strategy. Such a demonstration could be provided by a structural reliability and risk assessment (SRRA) model analysis, which would take into account specific initial conditions of component damage and thermal and mechanical conditions which the component would be expected to encounter during its lifetime. These SRRA analyses calculate the failure probability as a function of time, starting with initial information on flaw distributions and material property variations, and using stress information to evaluate the rate of crack growth.

Inspections would be required, however, if SRRA analysis indicated that the failure probability exceeded the target value before end of component lifetime. In that case, various potential inspection strategies could be postulated and evaluated using SRRA calculations. The results of specific inspections at specified time intervals would be modeled by using probability of detection (POD) information to revise crack size distributions assuming that detected cracks were repaired. The result of these calculations would again be failure probability as a function of time, but in this case inspection and repair would reduce the estimated failure probability at each repetion of the inspection cycle.

By performing analyses in which different inspection frequencies and crack detection capabilities are assumed, several potentially satisfactory inspection strategies could be developed. Alternatively, it might be determined that advanced inspection techniques would have to be developed to achieve target failure probabilities for certain components. Once successful strategies are established, decision analysis techniques could be used to select an optimum strategy.

Task force studies are only now addressing the development of specific inspection strategies to maintain component rupture probabilities below target values. A primary reason for this is that efforts to date have focused on developing the overall methodology and applying the risk prioritization portion of it to the Surry-1 example. It is clear that a considerable amount of work remains before recommendations can be made for a complete powerplant inspection plan.

Nevertheless, a variety of documented studies have demonstrated the ability of SRRA analyses to predict the effects of inservice inspections on component rupture probabilities. Such probabilistic calculations model uncertainties in stresses, fabrication quality, material properties and service environments. For example, Figure 2 shows, for two initial crack depth distributions, how the probability of as double ended pipe break increases with time under the following inspection strategies: pre-service inspection only, one additional inspection at 20 years, and three additional inspections, at 10, 20 and 30 years (Harris 1986).

In addressing nuclear components it is planned to model specific failure mechanisms such as stress corrosion cracking, thermal fatigue and erosion corrosion. In order to avoid having to perform specific calculations for each component addressed, a range of generic component types will be modeled with their associated failure mechanisms. For these componenets a range of ISI parameters (i.e. inspection frequency, detection probability, flaw sizing accuracy, and flaw acceptance criteria) will be modeled parametrically to identify the rupture probability achievable by various inspection strategies. It is anticipated that once a target rupture probability is identified for a given component, along with the expected stressors and failure mechanisms of importance, that these parametric results will allow selection of the appropriate inspection strategy without further calculations.

Once candidate inspection strategies have been determined which yield component rupture probabilities less than identified target values, decision analysis techniques can be used to select among them. Two obvious decisions are to eliminate from further consideration strategies which pose a significant hazard to inspectors or which are likely to damage (e.g. due to disassembly) the inspected component. Additional discussion of decision analysis techniques and strategies is presented in the documents prepared by the task force (ASME 1991, and the forthcoming Volume 2).

Application of Methodology to Fossil Fueled Power Plants

During the past year, this ASME Research Task Force has also been studying the application of the general methodology to fossil fueled power plants. In so doing, the advances and refinements developed in nuclear power plant studies have been evaluated, and incorporated where appropriate. The task force is presently developing a "Volume 3" document which will describe specifics of this application of the methodology to fossil fueled power plants. Document finalization and publication are expected during 1993.

For the fossil application, the situation is simultaneously simpler, and more complicated. It is simpler because fossil plants lack the "defense in depth" design concept which builds in the multiple layers of safety functions and redundant systems and components found in nuclear plants. It is the defense in depth design which results in the need for complicated, extensive PRA analyses of nuclear plants to quantify the risks associated with potential simultaneous failures of independent, operating and standby safety equipment which must take place before core damage results. In fossil plants, generally, fewer independent failures are necessary before plant damage results. However, the situation is simultaneously more complicated because the industry has not developed the risk analysis documentation which the nuclear power industry has. Consequently, although the necessary analyses are generally simpler and more straightforward for the fossil industry, they generally remain to be performed, whereas PRAs are now required for nuclear power plants.

The analytical tools for quantitative risk analysis of fossil fueled power plants are the tools of PRA. Event trees permit determination of accident sequences through which an initiating event can lead to plant damage of greater or lesser severity. As an alternative, reliability block diagrams may also be used. And fault trees provide a method for quantification of system success or failure probabilities for use in quantifying the event trees. Finally, this information is combined to identify the dominant cut sets resulting from the initiating events. In general, the process is considerably simpler for fossil plants than for nuclear plants.

There is considerably more historical data on pressure boundary failures at fossil power plants than at nuclear plants. Because the public safety consequences of failures at fossil plants are smaller than at nuclear plants, fossil plant design, maintenance and operating practices have resulted in more failures than at nuclear plants. While this results in considerably more data, it may not be readily available, analyzed, or even documented at any given plant. Plant specific data collection is recommended to supplement generic data in quantification of the risk analysis.

With these differences, then, it appears that the application of the detailed risk-based prioritization approach developed for nuclear power plants can be adapted to fossil power plant analyses. Pilot studies indicate that the methodology is workable, requires acceptable resource committments, and yields results which are reasonable and in agreemant with common-sense qualitative assessments. In addition, safety and economic factors can be integrated using decision-risk analysis and SRRA techniques to choose optimal inspection strategies for components.

Summary and Conclusions

The general methodology, previously developed by the ASME Research Task Force on Risk-Based Inspection Guidelines for application to any facility or structural system, has been further developed and applied to the inspection of nuclear power plant components. The ultimate objective of this effort is to produce a recommendation for comprehensive revisions to the in-service inspection requirements of Section XI of the ASME Boiler and Pressure Vessel Code. Development of a firm technical basis for such recommendations has been initiated, work is presently ongoing, and several years of work are expected to be required to develop this basis.

The methodology developed for nuclear power plant applications utilizes conditional risk information derived from PRAs in the quantification of risks associated with component pressure boundary and structural failures (ruptures). This methodology has been pilot tested, and is being applied to risk-prioritize components in specific power plants. Work is under way to determine whether generic trends exist in the results for different plants. A procedure has been recommended for determining target component rupture probability values, to be maintained by inspection activities, from these quantitative risk estimates. That procedure has been described and an application given herein.

Studies of the application of this methodology to fossil fueled power plants indicate that it is directly applicable, although there are far fewer documented risk analyses available. Consequently, risk analysis work may have to start from first principles in many cases. This is offset by much simpler plant designs and analysis requirements, and by greater amounts of failure data in some cases. Results of pilot studies indicate that the methodology can be adapted to fossil power plant analyses, yields results which make sense, and requires reasonable resource committments.

Finally, a general method for determining the characteristics which an inspection strategy must possess in order to maintain target rupture probabilities has been described. Ongoing task force studies are addressing the development of specific inspection strategies to maintain component rupture probabilities below target values. These studies utilize SRRA analyses to calculate rupture probability as a function of time under given initial, service and inspection conditions. They also use decision analysis techniques to eliminate strategies which pose hazards to inspectors or inspected components, and to select an optimal strategy from among potentially successful alternatives.

Clearly, much work remains before a comprehensive risk-based revision to Section XI can be proposed. Nevertheless, work is well under way toward this goal.

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