ADVANCED REACTOR INSTRUMENTATION AND CONTROL RELIABILITY AND RISK ASSESSMENT *

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Summary

Advanced nuclear power reactors will use different approaches to achieving a higher level of safety than the first generation. One approach uses the technological developments in computation and electronics in the form of digital instrumentation and control (I&C) to enhance the reliability, and accuracy of information for plant control, responding to the information, and controlling the plant and its systems under normal and upset environments in various states of degradation. Evaluating the reliability and safety of advanced I&C systems requires determining the reliability of the I&C used in the advanced reactors which involves distributed processing, data pile-up, interactive systems, the man-machine interface, various forms of automatic control, and systems interactions. From these analyses will come an understanding of the potential of the new I&C, and protection from its vulnerabilities to enhance the safe operation of the new plants.

Technological, safety, reliability, and regulatory issues associated with advanced I&C for the new reactors are discussed herein. The issues are presented followed by suggested approaches to their resolution.

Background

The basic design of U.S. nuclear power plants was developed in the late 1950's and early 1960's. Computers at this time were programmed with patch panels, used vacuum tubes and had less computing power than some modern hand calculators. Instrumentation and control (I&C) of plants at the time reflected the state of technology. Upgrades of old systems have been performed, such as microprocessor control of the feedwater control valve in a 20-year old BWR-1, but the application of these electronics advances to nuclear power is just beginning to appear.

While advanced reactors use different equipment from the older plants, the basic principles remain the same, as illustrated in Figure 1. Plant status information is collected, processed and transmitted to operators or automatic control systems from which come the control signals. This is shown in the figure for advanced I&C: The sensors, continuously monitoring the process parameters, produce analog electrical signals which undergo various types of signal conditioning before going to distributed processors for digital conversion and multiplexing onto transmission systems linking with central

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processing locations prior to operator display or use as input to automatic control systems. The distributed processors are special-purpose microcomputer systems with the necessary software for their control and decision-making. Transmission media are fiber-optics or transmission lines. As an example, the Westinghouse AP600 design uses a "Monitor Bus", as a high speed, redundant communications network to link systems important for operator control of the plant. Safety-related systems use isolation devices for ground loop isolation and for protection against failures elsewhere.

The exact way this process is executed varies with the vendor's grouping of distributed processing, the types of links, parallel processing and central processing. As there is a generic form of the I&C, there are generic and specific problems that are unique to advanced I&C. Table 1 presents some concerns associated with digital control of advanced nuclear power plants followed by suggested approaches.

Table 1. Some Regulatory Concerns Associated with Advanced I&C

- 1) Risk and Reliability Evaluation Methods
 - 2) Analyzing Information Processing Overload
 - 3) Data Sources for Reliability Analysis,
 - 4) Software Reliability Analysis,
 - 5) System Interactions/Common Cause of Failure,
 - 6) Environmental Considerations,
 - 7) Signal Validation and Disturbance Analysis,
 - 8) Test, Maintenance, Education and Training,
 - 9) Man-Machine Interface,
 - 10) Adequacy of Codes, Guides, and Standards for Evaluating Advanced I&C.
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1. RISK AND RELIABILITY EVALUATION METHODS

Discussion

Advanced I&C consists of hardware and software providing either direct feedback in the form of automated control, or feedback as an operator aid to facilitate the conduct of plant operation within acceptable limits imposed by the plant Technical Specifications. Conventional techniques for reliability and risk analyses may not be adequate for advanced I&C systems because of feedback loops and timing considerations. Fault trees find their greatest validity when used for the analysis of static situations. In dynamic situations, such as where success or failure causes reconfiguration such as occurs in adaptive, fed-back systems, it may be necessary to modify fault tree analysis (FTA) or adopt other analytical approaches. Determining the reliability of these systems may necessitate either extensions of FTA or the use of alternative methods such as Digraph and Markov.

Approach

A design for advanced reactor licensing will be carefully examined and evaluated to determine the adequacy of current methods. Adaptive, fed-back systems will be identified, evaluated for criticality to plant safety and investigated by several approaches. One method, an adaption of current fault tree practices is Phased Fault Tree Analysis (PFTA) in which time is divided into discrete segments and the system evaluated at each time step. Minimization of the number of time steps may be accomplished selecting the times for occurrence at the times of major changes. This process is closely related to finite element and mesh analysis used in many engineering computer codes. Digraph, which has inherent capabilities for adaptive analysis, will also be investigated. Markov methods because of their "state change approach" may also be useful.

2. ANALYZING INFORMATION PROCESSING OVERLOAD

Discussion

The data rate from each sensor is determined by the time response required of each sensor. The data rate is not particularly high from a given sensor but the rate from hundreds of sensors as it is presented to distributed processors is very high. Under normal operating conditions, the design should adequately accommodate this problem; under accident conditions including equipment degradation (hardware and/or software failure), the performance may not be as expected. Advanced power plants use multiple distributed processors. In one design, accident conditions were caused by delay in information processing in an overloaded distributed processor.

Approach

Phased Fault Tree Analysis (PFTA), one form of which was suggested for the analysis of feed-back, is suggested for application on a larger scale involving interacting systems. This technique consists of fault tree analysis performed on the control system at discrete, critical time steps of operation. In application, it would involve preparing fault trees of the digital control system at the necessary levels of detail at each of the critical time steps. Through a study of the time cycles, critical time periods would be identified under various degrees of upset. It is believed that, given generic trees, major changes will not be required to condition a tree system for each of the time steps. While the overall trees may not require reconstruction, the changes characteristic of a time step will be significant and identify the system aspects that are involved. Also of concern is the design adequacy for sufficient storage and display of parameters

under severe accident conditions. The PFTA will initially be used qualitatively for minimum cutset identification. Quantification may be performed if it is necessary to determine the probabilities of failure.

3. DATA SOURCES FOR RELIABILITY ANALYSIS

Discussion

Failure data are not readily available for the new electronic components that are used in advanced I&C. However, the U.S. Department of Defense has done extensive analysis of complex electronic systems and prepared data bases (e.g. MIL-HDBK-217E and GIDEP). Some of these data are for operating systems of comparable complexity in various environments. Some of the data are for use as input for modeling electronic systems from basic elements (MIL-HDBK-217E).

Approach

Advanced I&C proposed for the new reactors will be examined for data base requirements. The data bases used by applicants for modeling digital control systems will be examined for adequacy and compared with data from other sources such as military and industry data. Cases of inadequate critical data will be brought to the attention of the NRC with recommendations for testing or other forms of data acquisition. Design requirements with appropriate documentation should be standardized.

4. SOFTWARE RELIABILITY ANALYSES

Discussion

Computers consist of hardware that is controlled by software. Software failure may cause the hardware to fail, thus software must be assured to operate properly under all circumstances especially accident conditions.

There are important differences between physical devices and software that invalidate some of the common risk assessment procedures when applied to hardware controlled by software. Software does not fail from physical wear like a mechanical component, it fails due to design and coding error. Generally, such failures are not included in the PRAs. Furthermore software repairs involve changing the program thereby rendering questionable the applicability of a conventional reliability data base for software. There are three types of software tests: static, dynamic and application.

Static testing consists of such tools as data-flow analyzers to check data path consistence, path analyzers to follow the program's path and assure against "dead code", coverage analyzers to assure that all paths are tested, interface analyzers to examine variable passing, and

cross references of symbolic names. Dynamic testing tools include assertion checkers which insert probes into the code and flag violations of the programmer's "assertion", simulators that mimic hardware or missing codes, path-flow tracers to trace program logic under dynamic conditions, symbolic execution tools to execute arithmetic with symbolic variables, and mutation analyzers that place intentional errors in a program for a set of test cases. Application testing consists of operating the software with its hardware to perform key tests under simulated accident conditions. Clearly the last type of testing is the most definitive for the tests that are performed. The problem with this type of testing is assuring that the set of tests is exhaustive.

Approach

The vendor's test program will be examined to determine the structure of the code and the tests that have been performed. BNL will use various analytical models to consider the control system response, as controlled by the software, to each accident initiator from a full set of accident initiators. If the software passes these tests, the response to multiple combinations of initiators will be considered until the probability of multiple occurrences is less than some cut-off value agreed upon. Detailed analyses and review may be performed using Software Fault Tree Analyses (SFTA) to identify the software vulnerability through qualitative rather quantitative ranking. Software test strategies, as described, will be used to detect and resolve previously undiscovered errors.

An investigation will be conducted to determine minimum standards and documentation that should be provided for the licensing process.

5. SYSTEM INTERACTIONS/COMMON CAUSE OF FAILURE

Discussion

The use of large scale integration, distributed processing, and a few wide-bandwidth data links provide savings and simplicity but results in critical paths and common causes of failure which could be critical to understanding and controlling the reactor.

Approach

BNL applies tools for common cause such as sneak circuit analysis, response matrix and interactive fault tree analysis to the digital control systems of the advanced reactors to identify any "Achilles' heels" and if such are identified, determine an approximate probability as a measure of criticality.

6. ENVIRONMENTAL CONSIDERATIONS

Discussion

The extensive use of solid-state equipment raises questions of its operability under conditions of EMI (electro-magnetic interference), radiation, temperature, humidity and vibration. For example, continued operation at high temperature may cause accelerated aging not anticipated in the original design.

Approach

An exhaustive list of adverse conditions prepared and digital control systems will be analyzed under each of them. Conditions that could be detrimental to the reactor operation will be identified and importance ranked according to severity and probability according to the following subjects:

EMI: Scenarios such as in-plant electrical problems leading to arcing, will be examined regarding induced voltages that may be expected in key I&C equipment and whether such voltages approach logic levels. If such could occur, a determination will be made regarding the estimated effect. The extensive technology developed for analysis and hardening against EMP (electro-magnetic pulse) performed for the military will be used as appropriate.

Radiation: Solid-state equipment, unless hardened, is more susceptible to radiation and radiation degradation than conventional equipment. The layout of advanced reactors will be examined for the effects of all types of radiation in equipment such as the effects of F-center darkening of fiber-optic links or the possibility of Cerenkov radiation in these links.

Temperature: Solid-state equipment is more temperature sensitive than some conventional equipment and may be adversely affected by accident conditions. Accident temperature profiles will be developed and the effects on components, including aging, would be determined by Arrhenius or other acceptable modeling methods.

Humidity: At logic level voltages and small currents, sneak circuits such as may be caused by humidity could significantly affect safety systems. Humidity profiles under accident conditions will be examined to identify circuit boards that may be affected.

Vibration: The connectors used with digital equipment and the boards themselves have susceptibility to either continuous or upset conditions vibration. Such vibration levels will be identified and the equipment examined, either using results from vibration testing, or by inspection to determine vibrational susceptibility.

7. SIGNAL VALIDATION AND DISTURBANCE ANALYSIS

Discussion

Signal validation is concerned with assuring the correctness of a measurement. It is common practice to use multiple sensors and compare their signals for being within range and in comparison with other signals. Various physical models may be used in the comparison to determine the signal reasonableness. If a signal is found to be erroneous, software may substitute the expected value or disregard the information and use the other indicators. These signal validation techniques also identify failed instrument channels. The validity of the algorithm used for signal validation for critical safety systems needs investigation.

A related subject is disturbance analyses used in advanced I&C systems. Whereas signal validation only involves intercomparisons with sensors of the same parameter, disturbance analysis works with a plant model to determine if all of the instrumentation is self-consistent and by doing this provides guidance for the analysis of disturbances to the operators.

Approach

Signal validation methods will analyze the algorithm being used, its validity and weaknesses. Estimates may be calculated regarding the possibility of an erroneous indication in spite of the validation.

Disturbance analysis methods being employed by an applicant will be reviewed regarding the model being used, its simplifications, and its possibility for providing the operator with misleading information.

8. TEST, MAINTENANCE, EDUCATION AND TRAINING

Discussion

Digital control raises the level of sophistication required of the technicians maintaining the plants. They may have difficulty diagnosing and correcting failures in the advanced I&C. The customary repair approach is identification and replacement of whole PC boards followed by off-line servicing on-site or by the vendor. This is generally effective but intermittent problems are difficult to locate and difficult to assure that the repair is successful. Test procedures and frequency that take advantage of fault tolerance and self-diagnostic methods require evaluation.

Approach

The test, diagnostic, training and maintenance procedures for license applications for advanced reactors will be examined for thoroughness. Time and motion study analyses may be performed to assure in-depth test and maintenance can be performed according to schedules. The test and diagnostic procedures will be examined for completeness to assure safe operation.

9. MAN-MACHINE INTERFACE

Discussion

The trend in the advanced control rooms is toward optimized presentation of the critical plant data that must be surveyed, analyzed and comprehended by the operator crews. Reduced staffing, integration of operator interface functions for all systems to achieve uniformity of function and appearance, simplification of plant monitoring and control, incorporation of plant-wide automation, challenge of operators to assure alertness, and safe operation under accident conditions require investigation.

In the absence of direct operating experience with advanced control rooms for the advanced reactors, reliance must be placed on models of operator reaction with the new displays and controls using simulators and mock-ups.

Approach

Models will be constructed of the man-machine interface under normal and accident conditions of varying severity and under varying degrees of degradation. Particular concerns are: operation under accident conditions with reduced staffing, the ability to increase staff under emergencies, the reliability and effectiveness of an indication under accident conditions, whether the reduced information set will provide the necessary information for diagnosis of upset conditions, and decision times under this new configuration. Digital controls raise the level of sophistication above that required for the current generation of the reactors. Automated diagnostic systems and computerized procedures will play an important role in maintaining and inspecting such systems. Completeness and adequacy of these procedures needs evaluation.

10. ADEQUACY OF CODES, GUIDES, AND STANDARDS FOR EVALUATING ADVANCED I&C.

Discussion

Advances in I&C technology offer many advantages in functionality and cost. However, there are areas, both in terms of the equipment as well as the design process, where existing industry codes, standards and regulatory requirements are not directly applicable. In these areas, the standards community has initiated efforts to remedy this situation so that the benefits of modern electronics technology can be realized. EPRI, through its ALWR Steering Committee, is working with industry and regulatory participants to develop a user consensus and establish the ground rules for standards. Examples of work through standards committees are: ANS/IEEE-ANS-7-4.3.2-1982, "Application Criteria for Programmable Digital Computer Systems of Nuclear Power Generating Stations" and IEC Standard, Publication 800, "Software for Computers in the Safety Systems of Nuclear Power Stations". These and other documents are needed to establish a review plan with acceptance criteria for determining the adequacy of the submitted designs. Existing regulatory guidance, while containing the framework for determining adequacy, may require modification to incorporate the latest advances in I&C technology.

Approach

Determining the adequacy of codes, guides, and standards for evaluating advanced I&C requires a systematic approach. This begins as a large matrix with issues/concerns arrayed along one axis with the guidance documents addressing the issue/concern arrayed along the other axis. Initial matrix elements would be in the form of checks if the document address the issue/concern at all. When the matrix is complete at this level, it will provide a scoping of the codes, guides and standards. The guidance identified as relevant, will then be reanalyzed to determine the depth and adequacy of the guidance. From this, recommendations may be made as to the need for further guidance development by industry or the NRC.

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