Book of Abstracts

IAEA Technical Meeting on Integrating Analog and Digital Instrumentation and Control Systems in Hybrid Main Control Rooms at Nuclear Power Plants

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Keynote address

Digital Instrumentation, Controls, and Human-Machine Interface (ICHMI) Technologies: Issues and Current Research

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Abstract

Instrumentation, controls, and human-machine interfaces are essential enabling technologies that strongly influence nuclear power plant performance and operational costs. The nuclear power industry is currently engaged in a transition from traditional analog-based instrumentation, controls, and human-machine interface (ICHMI) systems to implementations employing digital technologies. This transition has primarily occurred in an ad hoc fashion through individual system upgrades at existing plants and has been constrained by licenseability concerns. Although international implementation of evolutionary nuclear power plants and the progression toward new plants in the United States have spurred design of more fully digital plantwide ICHMI systems, the experience base in the nuclear power application domain is limited. As a result, there are challenges that need to be addressed to enable the nuclear power industry to effectively and efficiently complete the transition to safe and comprehensive use of digital technology.

To respond to technology challenges, roadmaps for research, development, and demonstration (RD&D) are being developed. These roadmapping efforts address technology gaps, technology maturity, and technology experience by establishing a comprehensive, systematic approach to meet high-priority technological needs. The first RD&D objective is to identify and eliminate technology gaps that may constrain measurement, monitoring, control, or protection. The second RD&D objective is to ensure technology maturity so that needed methods, tools, equipment, or other products are available with a sound infrastructure. The third RD&D objective is to demonstrate performance and resolve licensing and usage uncertainty.

This presentation summarizes the key elements of an ICHMI technology roadmap and discusses current research activities.

* The Oak Ridge National Laboratory (ORNL) is managed for the U.S. Department of Energy by UT-Battelle, LLC, under contract DE-AC-05-00OR22725.

Presentation 1

Design Approaches for Field Programmable Gate Arrays in Safety-Related Applications

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Abstract

Oak Ridge National Laboratory (ORNL) is conducting research for the U.S. Nuclear Regulatory Commission regarding design approaches for field programmable gate arrays (FPGAs) in high-integrity applications. As part of this study, ORNL has evaluated experience and existing guidance and the experience base for the use of FPGA technology in high-assurance applications. This presentation describes findings from this investigation, including an overview of programmable digital logic technology and a summary of effective implementation practices.

* The Oak Ridge National Laboratory (ORNL) is managed for the U.S. Department of Energy by UT-Battelle, LLC, under contract DE-AC-05-00OR22725. This work is sponsored by the U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research. Opinions and conclusions expressed by the author do not necessarily represent positions endorsed by NRC

Presentation 2

New Digital Technologies for Improving Field Operation

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EDF R&D
France

Abstract

Most of nuclear utilities are willing to extend the lifetime of their power plants, and they are wondering which modernization strategy would be the most efficient for their plant, among the two following ones:

- The "Defensive" approach, i.e. only guided by the constraints (evolution of industrial or licensing environment, increase in I&C maintenance costs, I&C obsolescence or ageing, loss of knowledge on existing systems ...) and the control of the risks associated with the evolution of these constraints. This approach is usually favored: "do nothing except if you are obliged";

- The "Offensive" approach, i.e. guided by the improvement of certain indicators of value added by the I&C system in a direct or indirect way: functional or performance improvements (power output…), O&M costs reduction...). Of course, this "offensive" strategy can (or must) also be considered together with a "Defensive" approach to improve the ROI: reduction of the cost impact of the necessary I&C equipment modernization.

Whatever the strategy, we have also to consider the risks introduced by the modernization project such as the environment of the installation, competencies to be set up, the availability (knowledge) of the data of the existing I&C system, the new I&C system characteristics (performance, lifetime…), licensing risks and effort. Besides these risks, the cost of the project (costs of the new system, production unavailability, training….) and the impact in the Man-Machine Organization are to be evaluated also.

At EDF, both approaches are considered, in order to extend the lifetime of the nuclear production, at a lower cost, while reducing the operation and maintenance costs.

Four (three) main projects are currently undergoing at EDF R&D to investigate new digital technologies such as wireless, RFID tags, panel PCs, PDA… that make it possible to improve the operation of a power plant while minimizing the impact on the control rooms and the existing I&C systems and, therefore, reducing the risks and costs.

- The SIAMOI project (Industrial Supervision and Multimodal Architecture for Operators in Interactive Situation) that will contribute to proposition for hybrid control room retrofit, based upon new interaction needs between operation in and outside of the main control room;
- The UTILE project, that focuses on the identification of prospective wireless usages for field operation, in interaction with main control room operation. One of the main aspects is to secure local operation with RFID tags in order to insure that the right component is being operate;
- The "m-EDF" project, that aims to build an opportunistic modular architecture with current components of wireless and web technologies, such DSL, ZigBee, Wifi in opportunistic networks configurations;
- The "Radioprotection Control Station" projects that attempt to adapt this wireless technology to radioprotection purpose, integrated in the US concept of Outage Center.

Beside these projects that improve field operation by bringing additional operator aids, EDF is also interested in the retrofit of its I&C systems with digital technologies, even though the current strategy is to store spare parts to face the obsolescence issue. Previous analogue designs are going to be replaced by digital systems, providing the opportunity to use smarter algorithms to operate power plants more efficiently by reducing margins or to choose a different level of automation and opening new perspectives. Automation should not be restricted to the replacement of human tasks by automated mechanisms. It should include cooperation between man and machine, for example when the computer provides diagnosis assistance. Sharing tasks and responsibilities between man and machine and managing these new complex installations has become an acute issue. Technical decisions must be driven by sound safety analysis evaluating their risks and benefits. A European project, leaded by EDF, is going to start regarding the impact (risk and benefit) of the I&C retrofit on the Man-Machine-Organization and Safety aspects. This project –

MMOTION - will focus on several specific objectives that are necessary to reach a better understanding of human factors at the European community level:

- To better understand the Man-Machine-Organisation interface in the plant installation;
- To share experience between different European partners on safety culture and skills;
- To transcend the boundaries between disciplines : systems behaviour, human factors, work organization, management, making it possible to homogenise formalisms used for the functional specifications that are necessary to the control room design;
- To capture the needs in information provided by control systems and used by operation and maintenance staffs, and to devise ways to make this information available;
- To identify the MMO performance shaping factors concerning safety improvement or risk reduction: automation level, plant-level operation knowledge and supervision, verification of operational activities, human organisation, share of responsibility, management issues;
- To propose methods to evaluate the safety of an installation, better taking into account its MMO characteristics;
- To propose, as a pilot exercise, a prototype to analyse Man-Machine-Organisation aspects in plant operation and to qualify possible remedial actions;
- To disseminate a roadmap towards the community of MMO actors in Europe.

Presentation 3

Probabilistic Design Verification of Instrumented Protection Functions

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Abstract

IEC 61508 supports performance based instrumentation design using probabilistic analysis of critical failure modes. This risk based approach has been used quite extensively in several industries and is applicable to nuclear designs as well. The IEC 61513 group of nuclear instrumentation standards mentions the subject but does not discuss detail. Probabilistic analysis of instrumentation requires inclusion of many important variables including the failure rates as a function of failure mode for all instruments, proof testing schedules and effectiveness, and common cause for all redundant components. This presentation will discuss methods and tools used to probabilistic failure analysis of instrumentation.

Presentation 4

Innovative Utilization of SMART Instrumentation at a Nuclear Power Plant

Eddie Saab

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Canada

Abstract

With 30 MW of lost power generation for up to two weeks after a reactor start-up and pipe vibration from water hammer leading to questions of piping integrity, a high visibility opportunity for improvement in the High Pressure Feedwater Heater System existed. Bruce Nuclear turned to Lakeside Process Controls to provide an integrated digital controls solution. This presentation describes the problem, the solution and the quantified technical and business results.

Presentation 5

Increase of NPP safety by modernization of the traditional analog I&C systems with the use of digital equipment

Ievgenii Bakhmach and Volodymyr Bezsalyi

Closed Joint Stock Company "Radiy"
Ukraine

Volodymyr Sklyar

Ukrainian State Scientific Technical Centre on Nuclear and Radiation Safety
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Abstract

1. Modernization of the traditional analog I&C systems which is performed by the Company "Radiy" (Ukraine) with use of digital equipment directs to the following goal achievement:

- increasing of quality and effectiveness of reactor units operating modes;
- increasing of accuracy and speed of control and checking processes;
- increasing of equipment dependability, availability and reliability;
- increasing of equipment external impact tolerance;
- improvement of realization of operator information support as well as visualization of technological processes parameters;

- improvement of realization of prognosis calculation for control functions;
- enlargement of diagnostics functions for I&C systems and technological equipment of reactor units;
- implementation of effective means of I&C systems administration;
- creation of long-term archive with history of reactor units operation;
- ensuring of I&C systems upgrades and modifications in the future;
- ensuring of compliance with requirements of national and international safety codes, guides and standards.

2. The Company "Radiy" has designed, produced and delivered for NPPs' units about thirty digital I&C systems since 2003. There are the following main types of I&C systems of the Company "Radiy":

    - Reactor Protection Systems (the main and the diverse sets);
    - Reactor Power Control and Limitation Systems;
    - Control Safety System;
    - Control Rod Actuation System.

    Information about above system is given in the report.

3. Proposed and approbated by the Company "Radiy" strategy of I&C systems modernization includes the following points:

    - an experience of integration and commission of digital equipment of logical systems which joint on NPP site with sensors, actuators and adjacent I&C systems including analog equipment;
    - use FPGAs (Field Programmable Gates Arrays) for performing of safety-related functions;
    - use multi-components distributed structure with heterogenic networks;
    - software verification approach bases on multi-components V-shape life cycle;
    - equipment permits to perform testing and to confirm compliance with functional and qualification requirements on producer site (although final acceptance testing is performed only on NPP site);
    - use of commercial off-the-shelf (COTS) products (for example bundling of workstations);
    - use of maximal possible quantity of own pre-developed hardware and software components with positive service history for design of new I&C systems;
    - diversity implementation in reactor protection systems;
    - possibilities for large I&C systems upgrades and modifications in the future;
    - compliance with requirements of national and international safety codes, guides and standards.

4. There are some technology and design risks of new digital technologies which are identified and taken into account by the Company "Radiy" specialists. The following measures are assumed for risks reducing of use of equipment produced by the Company "Radiy":

    - FPGAs use for control algorithms realization, complete verification on all life cycle stages, diversity in Reactor Protection Systems – for reducing of software failures risks including common-cause failures risks;

- redundancy of network components and their complete diagnostics – for risks reducing of network components failures;
- complete validation and qualification testing of COTS for all types of external impact with equipment of I&C systems – for risks reducing of COTS products failures;
- configuration management – for reducing of maintainability risks;
- development of technologies independent from specific electronic components – for risks reducing of electronic components obsolescence during I&C systems life cycle.

5. Conclusion:

At the present time digital equipment produced by the Company “Radiy” is successfully used for modernization of existing I&C systems. Taking into account flexibility and generality of equipment produced by the Company “Radiy” it is possible to use that equipment for reactors of the new generation.

Presentation 6

Implementation Strategy for Separating Display and Control in CANDU Stations

H. Storey, T. Rector, L. Yu, J. Carmody, R. Doucet, and D. Trask

Atomic Energy of Canada Limited
Canada

Abstract

Digital Computer Controllers (DCCs) have been used in CANDU® plants for over 25 years to perform both control and HMI display/annunciation functions. AECL is now developing new technology to separate display and control in order to take advantage of modern plant display systems such as AECL’s Advanced Control Centre Information System (ACCIS) product for existing and new-build CANDU plants.

This presentation presents an overview of recent control and display system interface development work performed at AECL to connect such disparate technologies. It also provides strategies for implementing such an upgrade in an operational CANDU station.

Presentation 7

Model for Obsolete Control System Replacement Economics Under Assumptions of Risk

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Canada

Abstract

Traditional Benefit Cost Analysis (BCA) methods are not adequate to support the analysis of a complex mission critical system (MCS) equipment replacement decision such as in the case of a nuclear plant control system. BCA methods can be adapted and extended by incorporating life-cycle costing and risk modelling methods. A working example of the approach is provided using the hypothetical analysis of a control system replacement decision. The method incorporates Monte Carlo simulation and random variables to model parameter uncertainty. The approach is combined with equipment replacement timing assessment and graphical visualization techniques. The methodology permits analysis of complex I&C system replacement decisions and is particularly suited to problems with low-probability and high cost-consequence events. The approach supports simultaneous assessment of uncertainties and traditional sensitivity analysis. Timing and financial risk of the replacement decision are key decision parameters.

Presentation 8

A Long Term Maintenance Strategy for the EDF's NPP Atos Origin

Georges Garcia and Jean-Louis Deimerly

Atos Origin Integration
France

Abstract

Atos Origin is providing Electricite de France (EDF) with renewable 20-years-long Maintenance of the Instrumentation & Control Systems of the 58 units of the 900, 1300 and 1450 Mw series.

Following an initial period dedicated to studies and means preparation, this long-term commitment has been instanced into a multi-phased contract ensuring EDF with a renewable long term maintenance support for their 20 units of the 1300MW serie, 34 units of the 900MW serie and 4 units of the 1450MW serie including a total of 278

control, monitoring, calculation, data configuration and LAN systems most of them delivered during the 80s.

This "MOC" multi-phased contracts are the contractual consequences of these analyses, representing the best Long Term Maintenance Strategy between EDF & Atos Origin, aiming at outsourcing the maintenance of the whole I&C of the 58 French nuclear units and relevant simulators.

Looking forward to propose the most efficient methodology at the best possible conditions along the contract duration, Atos Origin put in place a dedicated organization for this activity, offering a unique system approach, common means and a complete set of standard services.

EDF wanted a team that could respond to both high reactivity tasks applied to any components of the systems (hardware, software and documents) and identification, analyses and proposed management of the up-coming obsolescences.

The profile of the current team, requiring permanent full set of competencies and commitment on techniques which are turning to be unattractive for young engineers nowadays, has therefore to be managed with specific care.

In order to succeed in setting up the MOC, enabling works were performed: setting up back-to-back long term agreement with main manufacturers, a Knowledge Databases, preliminary actions and supplementary studies,

The maintenance team, consisting of both generalist and specialist competencies, will be able to provide the MOC services:

- The generalists competencies provide the operational support and the first level analysis of the site events with an high reactivity;
- The specialists' competencies provide advanced analysis and the production of corrective solutions. This includes the configuration management maintenance,
- Human knowledge and competencies management. Training strategy.
- Investigation and focus on the state-of-art technologies potentially re-usable to maintain the systems functions ("system approach"). Equipment's ageing analyses.
- Inventory and definition of the strategies of refreshing all the identified obsolescences covering the total set of competencies, eventually proposed for systemic upgrades.
- Set-up of the necessary platforms, processes and web-based tools; (maintenance knowledge server)

Presentation 9

Challenges in Digital Upgrade at OPG

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Canada

Abstract

To address obsolescence of analog controllers, OPG has been replacing these with digital devices. This got underway in a serious fashion in 2003 and is on-going. Replacements are usually one-to-one replacements (i.e., replace one analog controller with a functionally equivalent digital controller). In some cases a one-to-many replacement (e.g., 13 analog deaerator level control devices replaced by one digital controller) is needed. OPG has encountered a number of challenges in our controller replacement program.

- The power-up configuration must be established and there are several approaches not all of which are appropriate in specific circumstances.
- Digital device tolerances and responses to power-interruptions demand careful consideration especially with respect to things such as bus transfers.
- Even with relatively little operating experience some display failures have been noted.
- Digital devices are more sensitive to static interference.
- Even with relatively new devices memory capacity can be a limiting factor when combining multiple analog devices into one digital device.
- Software based systems often require more extensive testing than was customary for analog devices. These tests are not covered by the traditional Factory Acceptance Tests
- Training and documentation updates are more extensive due to differences between the analog and digital devices.
- Providing controllers compatible with the existing control panel cut-outs often requires hardware modifications.
- Operations have over the years adapted to the analog device. Providing the equivalent visible and legible from all necessary distances and angles, and in the existing workplace illumination is difficult.
- Providing adequate display design with the limited size, resolution and flexibility of the graphical display area requires significant design effort.
- Digital devices allow for complex applications but these often need special formats in addition to standard ones.

Presentation 10

Nuclear Power Plant Control Computer Aging Management Strategy, Implementation and Results

Richard John Hohendorf

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Abstract

The digital control computers (DCCs) at Ontario Power Generation (OPG) nuclear plants at Pickering A and B and Darlington are more than thirty years old. No technical support is available from the original equipment manufacturers of the DCCs for Pickering A (IBM 1800), Pickering B (Varian V72) or Darlington (DEC 11/70). Yet the DCCs are vital to plant operations since they provide reactor control, main control room annunciation and process displays, fuel handling controls, and control of several other important process systems. The DCCs are dual-redundant to achieve the necessary system reliability and if at least one channel of the dual DCC system is not available, the Generating Unit must be shut down.

The DCC aging management strategy was initiated at Pickering A since it had the oldest DCCs. The initial focus was to acquire and stockpile as many IBM 1800 spare systems as possible in order to ensure the long-term availability of necessary replacement parts. This was driven by the understanding of the inherent value of retaining the proven and stable control software running on the Pickering A DCCs and of the inherent risk in trying to migrate this software to a different hardware platform. The Pickering A control programs, like all the other DCC control programs at OPG, are written in assembler language running under a custom real-time executive. Process control time constants are embedded within this combination of hardware and software. Also embedded therein are the self-checks and fault responses which are essential to reliable and fail-safe DCC operation. A fundamental element of the DCC aging management strategy has been to retain the integrity of the time-proven control programs and the infrastructure in which they operate.

For Pickering A, it became obvious in the early 1990's that acquiring spare parts alone would not be sufficient to sustain the reliable life of the IBM 1800 DCCs. The Pickering NGS A Plant Computer System Lifetime Strategic Assessment was issued in 1992. The approach selected was to seek out and qualify a hardware emulator that would faithfully execute the Pickering A DCC software with no or minimal change. Fortunately, the U.S. military also used IBM 1800 computers and had previously initiated development of a high fidelity emulator, the ES-1800. After extensive supplementary testing by OPG personnel, a project was undertaken to replace the IBM 1800 DCCs at Pickering A with the ES-1800, which is built up using much more modern technology than the IBM 1800. Under this project, ES-1800 computers were successfully installed on Units 1 and 4 at Pickering A. Since the 2002 replacement, the Pickering A DCC software has run without any emulator attributable faults on these units and the system health has improved dramatically from "RED" to "WHITE". The replacement of the DCCs on Units 2 and 3 was suspended due to the decision to put these Units into a Safe Store state. (The concerns about the continuing viability of the IBM 1800s have been validated based on the difficulty encountered in maintaining the Unit 2 and 3 IBM 1800 DCCs to perform de-fuelling activities).

Presentation 11

DCC Control Program Reverse Engineering - Process, Tools and Results

Gregory Rolf Moum

Ontario Power Generation
Canada

Abstract

The control programs in the Digital Control Computers at the Ontario Power Generation's Pickering NGSs perform significant control functions including reactor regulation, zone thermal power calculations and boiler pressure control. These control computers are now over thirty years old.

While the design basis documentation for the DCC control programs at the Pickering A and B nuclear generating stations exists and was prepared in accordance to contemporary standards, it is not sufficiently detailed and verified for long term DCC operation and maintenance. This issue has been mitigated by the proven performance of the software logic and by the support available from experienced individuals. Changes to the software have been and will be necessary to accommodate obsolescence of components and station aging. A good understanding of control program behaviour in support of the analysis of plant upsets is also necessary. With the impending departure of experienced and knowledgeable staff, this issue has become more urgent.

The goal of the DCC control program reverse engineering is to provide the necessary elements to enable implementation of DCC control program logic changes at low risk and timely operability analysis in response to plant upsets.

This paper describes the reverse engineering process, the tools to support that process and the results achieved so far. The reverse engineering process produces Software Requirement Specifications (SRSs) describing the behaviour of the control programs from the existing assembler code. A set of tools create a model of this behaviour from the SRS and use that model to predict expected results for test cases. The control programs are then run in an emulated environment against these test cases with expected results to confirm that the SRSs accurately and completely reflect the behaviour of the control programs. The tools provide metrics to determine test coverage of both the SRS and the assembler code. To date, several SRSs have been prepared and tested. These SRSs have also been validated by process control engineers. Feedback from the software engineers responsible for the control programs has been positive.

Presentation 12

Implementation of a new digital system in Atucha-II NPP

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Abstract

Atucha-II is a delayed nuclear power plant project in Argentina.

The Safety and Operational I&C were designed using modular electronic systems based on analog technologies of 1980 that presently are out of fabrication and no more available in the market.

When restarting the project in 2006, it was found that some of the operational I&C modules were not supplied in sufficient quantity to complete the erection and the quantity of spares was not enough to assure the plant operation.

To solve these problems it was decided to begin a modernization process of the central I&C in steps.

The first step, which is described here and will be performed before the commissioning of the plant, consists in the change of part of the operational cubicles and the Supervisory Computer System to a new programmable digital system.

The original design of the central I&C based in Functional Complexes makes the hardware migration easy to implement.

Each functional complex is a group of cubicles in which all the I&C functions are located which are related to the monitoring and control of a given process system.

By means of the replacing of some functional complexes to digital I&C, electronic modules can be obtained to complete the installation or to be used as spares.

Also the supervisory computer will be changed to a new digital HIS (Human System Interface), which will assume all the annunciation and data logging functions of the old computer system and will also allow the monitoring and control functions of the new digital operational I&C.

The new HSI system also can be connected by bus to digital electronic cubicles of the Safety I&C, which will replace the present analog ones in a future step.

With this basic first step of modernization future partial I&C replacements during plant outages with the plant in operation will be easier.

Presentation 13

Analog to Digital Control System Replacement Projects

Randy Long

Bruce Power
Canada

Abstract

Problem Statement:
Numerous Control Systems within Operating Nuclear Units are nearing or at the end of their useful life. These systems are typically Analog in design, installed and commissioned during original construction dates. Problems associated with these aging system which are plaging Nuclear Operators are Obsolescence, Inadequate Spares, Complexity in Troubleshooting large hardwired systems, Eroding Experience and Device Failures leading to safety and economic impacts.

There are a number of factors to consider when attempting to justify a complete system replacement versus a component by component replacement. This presentation will look at some of these systems which have been chosen for replacement and discuss the rationale used to drive this digital upgrade decision.

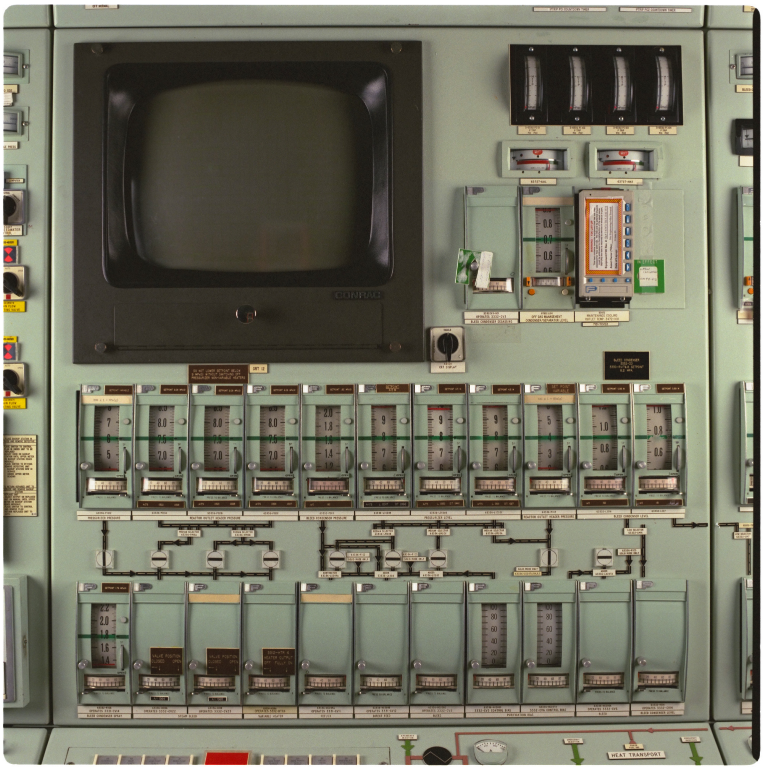
Specific project related concerns due to the digital technology such as Software Qualification shall be addressed. Also specific to hybrid main control rooms is the Human Factors requirements when replacing analog controllers and meters with Touch Screen based HMI.

One aspect of the Nuclear Power Plants which traditionally differs from other industry is the requirement for Full Scope Simulator for training purposes. This presention will identify the importance of the full scope simulator and the role it plays in digital system replacements.

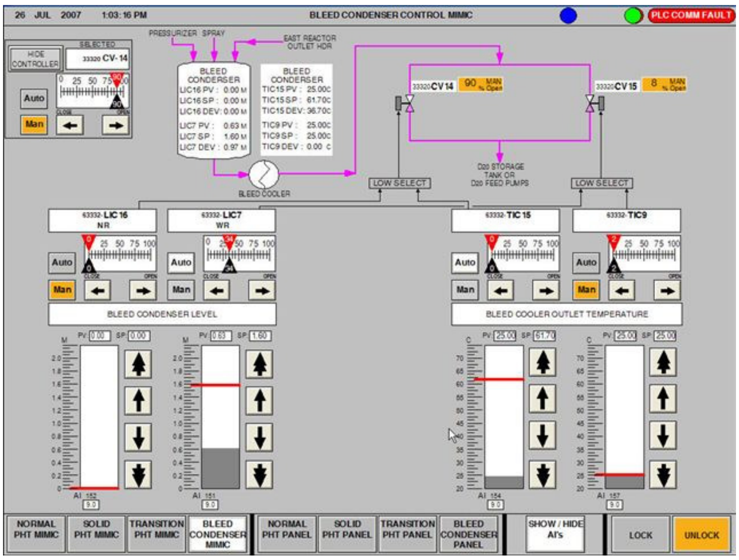
Other topics to be discussed in proposed presentation include:

- Probablistic Risk Assessment
- Stakeholder Involvement
- New Technology Benefits
- New Technology “Traps”
- Security
- Phased Acceptance Testing

The proposed presentation shall be in a PowerPoint format.



Existing Feed Bleed Relief Control Panel 4 MCR



Typical HMI Graphic to replace Bleed Consensdor Controllers.

Presentation 14

Computer Based Plant Display and Digital Control System of Wolsong NPP Tritium Removal Facility

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**Atomic Energy of Canada Limited*
Canada.

J. Ahn**

***Korea Hydro Nuclear Power Co., TRF, Wolsong NPP,*
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Abstract

The Wolsong Tritium Removal Facility (WTRF) is an AECL-designed, first-of-a-kind facility that removes tritium from the heavy water that is used in systems of the CANDU reactors in operation at the Wolsong Nuclear Power Plant in South Korea.

The Plant Display and Control System (PDCS) provides digital plant monitoring and control for the WTRF and offers the advantages of state-of-the-art digital control system technologies for operations and maintenance. The overall features of the PDCS will be described and some of the specific approaches taken on the project to save construction time and costs, to reduce in-service life-cycle costs and to improve quality will be presented.

The PDCS consists of two separate computer sub-systems: the Digital Control System (DCS) and the Plant Display System (PDS). The PDS provides the computer-based Human Machine Interface (HMI) for operators, and permits efficient supervisory or device level monitoring and control. A System Maintenance Console (SMC) is included in the PDS for the purpose of software and hardware configuration and on-line maintenance. A Historical Data System (HDS) is also included in the PDS as a data-server that continuously captures and logs process data and events for long-term storage and on-demand selective retrieval.

The PDCS of WTRF has been designed and implemented based on an off-the-self PDS/DCS product combination, the Delta-V System from Emerson. The design includes fully redundant Ethernet network communications, controllers, power supplies and redundancy on selected I/O modules. The DCS provides field bus communications to interface with 3rd party controllers supplied on specialized skids, and supports HART communication with field transmitters.

The DCS control logic was configured using a modular and graphical approach. The control strategies are primarily device control modules implemented as autonomous control loops, and implemented using IEC 61131-3 Function Block Diagram (FBD) and Structured Text (ST) languages for easy reviewing, documentation, testing, commissioning, and maintenance.

As the project was a design collaboration involving a multi-national team of design organizations led by AECL, the control functional requirements for the PCDS come in

the form of design input documents from several design authorities. To ensure smooth and efficient integration and consolidation of all the functional requirements for control and monitoring AECL provided two key design guide documents: a design guide for the preparation of design input documents for PDCS, and a plant control philosophy guide. Another innovation on the project included an AECL supplied and in-house developed tool (IntEC) to define and manage I/O signals and cabinet wirings by process system designers. To accommodate the skid-based design process taken for the plant and to save site construction time, AECL introduced pre-engineered control and field I/O cable marshalling and cross-connection cabinets. This approach reduced scheduling sequence inter-dependencies, permitted flexibility in design, and accommodated late design changes often encountered in a first-of-a-kind plant design. Finally, to decouple hardware delivery from the software (i.e., application logic) development, AECL developed the PDCS software configuration test-bed. This allowed both activities to proceed in parallel to ensure the schedule was met.

Presentation 15

Integration of Different HSI Equipments in the MCR During I&C Systems Modernization

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Abstract

One of the problems during modernization of NPPs is integration new and obsolete HSI equipments from one side and different types of new HSI equipments from another side. HSI equipments of following NPP I&C systems are mainly installed in the MCR:

- safety systems: reactor protection systems, safety actuation systems and safety system support features;
- safety related systems: control systems, process information systems;
- emergency communication systems and radiation monitoring systems.

Approaches to the integration of HSI equipments in the MCR are defined by initiative requirements and key points, which are defined by initiative requirements accordingly (based on the experience of I&C systems modernization at the WWER-440 NPPs):

Initiative requirements:

1. Requirements for aims, volumes and terms of modernization.
2. National and International codes, standards requirements for the MCR.
3. Requirements are defined by existing HSI equipments in the MCR:
   - human factors engineering;
   - technical parameters;
   - layout of equipments.

Key points are defined by initiative requirements:

1. Time schedule of modernization steps.
2. Scope of modernized I&C systems on each step.
3. Scope of modernized HSI equipments, including cabling for each I&C system on each step.
4. Human factor engineering and technical parameters for each new HSI equipment.
5. Scope of new HSI equipments for each I&C system.

The functional types of modernized equipments in the MCR can be categorized as following:

1. Process information equipments: display meters, strip chart recorders, VDUs of CRT type.
2. Spatially distributed manual switches.
3. Mimic diagrams with hard-wired lights and indicators.
4. Alarm windows equipments.
5. Cabling of HSI equipments.

Presentation 16

Development of the Advanced CANDU Reactor Control Centre

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Atomic Energy of Canada Limited
Canada

Abstract

Atomic Energy of Canada Limited (AECL) has adapted the successful features of CANDU®* reactors to establish Generation III+ Advanced CANDU Reactor®** (ACR®**) technology. Generation III+ designs are being developed to satisfy the following key design goals:

- further enhanced safety,
- lower cost/better economics, and
- ease of operability/operations and maintenance

The ACR-1000®**, AECL's newest flagship reactor, is an evolutionary product, starting with the strong base of CANDU reactor technology, coupled with thoroughly demonstrated innovative features to enhance safety, economics, operability and maintainability. The enhanced ACR-1000 Control Centre is a key feature of the design and has been an area where feedback has been a significant component of the design evolution. This evolutionary design includes the long proven functionality at several existing CANDU control centres such as the 4-unit station at Darlington, with advanced features made possible by new control and display technology. The control centre design is based upon the recent Qinshan control room with further upgrades to meet customer needs with respect to the key design goals discussed above. Additionally, ACR control centres address characteristics resulting from Human Factors Engineering (HFE) analysis of control centre operations in order to further enhance personnel awareness of system and plant status.

The Control Centre concept/vision has focused on the following topics to address the key design goals:

- Control Room Workplace
- Plant Controls and Process Automation
- Displays for Process and Equipment Information
- Operation Support Applications and Resources
- Equipment Testing

The meeting presentation will present the evolution of the ACR-1000 control centre and how the current design addresses the key design goals and topics.

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* CANDU® is a registered trademark of Atomic Energy of Canada Limited (AECL).
** Advanced CANDU Reactor®, ACR®, ACR-1000®

Presentation 17

A Modern Control Room for Indian Advanced Heavy Water Reactor

Clement C. Verghese, Jose Joseph, B.B. Biswas and R.K. Patil

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India

Abstract

Advanced Heavy Water Reactor (AHWR) is a next generation nuclear power plant being developed by Bhabha Atomic Research Centre, India. AHWR is a vertical, pressure tube type, heavy-water-moderated, boiling light-water-cooled, innovative reactor, relying on natural circulation for core cooling in all operating and accident conditions. In addition, it incorporates various passive systems for decay heat removal, containment cooling and isolation.

In addition to the many passive safety features, AHWR has state of the art I&C architecture based on extensive use of computers and networking. In tune with the many advanced features of the reactor, a centralized modern control room has been conceived for operation and monitoring of the plant. The I&C architecture enables the implementation of a fully computerised operator friendly control room with soft Human Machine Interfaces (HMI). While doing so, safety has been given due consideration. The control & monitoring of AHWR systems are carried out from the fully computer-based operator interfaces, except safety systems, for which only monitoring is provided from soft HMI. The control of the safety systems is performed from dedicated hardwired safety system panels.

Soft HMI reduces the number of individual control devices and improves their reliability. The paper briefly describes the I&C architecture adopted for the AHWR plant along with the interfaces to the main and backup control rooms. There are many issues involved while introducing soft HMI based operator interfaces for Nuclear Power Plants (NPP) compared to the conventional plants. Besides discussing the implementation issues, the paper elaborates the design considerations that have undergone in the design of various components in the main control room especially operator workstations, shift supervisor console, safety system panels and large display panels. Mainly task based displays have been adopted for the routine operator interactions of the plant which is complemented with functions and systems based displays in order to aid the operators during unforeseen circumstances.

The design process of the control room and its components have been carried out by following relevant national and international standards. Due care is given to the Human Factor Engineering (HFE) principles in the design of the control room and its components. Considerable experience gained from the design, implementation and operation of a computerised hybrid control room for a research reactor employing soft console based HMI, large display panel and computer based operator information system has also helped in evolving the current design for control room of AHWR.

A scaled down prototype of the control room is setup in order to evaluate the design aspects of the soft control interfaces for the operator interaction. The setup consists of the operator workstation for the reactor side along with the large display panel. Track ball and touch screen based interfaces are incorporated for the HMI at the operator work station. Touch screen interfaces display the organization of the display network hierarchy which helps the operators to quickly select the required display page as well as to identify the location of the current display.

Presentation 18

The Design of the Computerized Main Control Room of Lingao Phase II NPP

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China

Abstract

For most of the newly built NPPs, the computerized main control rooms (MCR) are adopted. LingAo Phase II NPP is under construction. In terms of reactor type, it is the duplication of LingAo NPP (Phase I). One of the main difference is the I&C system and the MCR. LingAo Phase II NPP adopts digital control system and computerized MCR while the LingAo Phase I has a conventional MCR. This paper will present the main features of this MCR and its design process, and will address the main difficulties in the following aspects: 1) to implant the functions of conventional I&C into computerized human machine interfaces(HMI), 2) to develop a reduced conventional I&C as the back-up control means which can accommodate accident situations, 3)to introduce computerized State-oriented accidental operating procedures (SOP) , 4) to build a platform to validate the HMI.

Presentation 19

Modernizing of Oskarshamn NPP with focus on Control Room issues

Thomas Gunnarsson

Oskarshamn Nuclear Power Plant
Sweden

Abstract

In Oskarshamn NPP Sweden a relative large upgrading and modernizing projects has been performed and are under planning. The projects are results of new safety requirements, and old equipment that has to be changed and modernized as well as need of more power and better efficiency.

Oskarshamn 1 BWR has been in operation since 1972, and is Sweden's oldest running unit. The original control room was quite compact, and provided a fairly good overview of the process. New requirements for design changes in the process systems and instrumentation, as well as existing shortcomings in the control room due to many previously performed modifications, forced a renewal of the control room. It was decided, in addition to several plant modifications, to build a new modern screen-based control room located in the same space as the old one, and with the same number of operators and to do the upgrade in one big step. Extensive human performance measures were acquired to validate that the operator performance in the new control room was at least equally good as in the old control room.

The modernized plant was operational in January 2003, and the new control room can be regarded as a hybrid solution.

Oskarshamn 2 BWR has been in operation since 1975, and is Sweden's second oldest running unit, has a little bit different approach to modernizing. It's decided due to several reasons to do the modernizing in some steps. For the first big step is actually under implementation and commission. In this first step all I&C and control equipment for the Turbine is changed to digital equipment. The result in the control room is that nearly the half of the control room equipment is changed to modern technology as operator stations for information and control and large screens for overview information.

Second step of upgrading of Oskarshamn 2 is to improve the safety in the plant and change the I&C and control room equipment for these functions. In the same outage also a power upgrade is decided. The electrical power will increase with 25 % from 620 to 840 MWe. The impact on control room and operator interface will be that more or less all of the rest of the control room and I&C will be changed to new modern equipment.

Oskarshamn 3BWR has been in operation since 1985 and is the newest plant in Sweden. In 2008 a power upgrade will take place in the plant. The power will increase from 1200 Mw to 1450Mwe. On the I&C side it's decided to keep the existing technology and control room concept.

Such extensive modernization and upgrade projects requires large resources that are not always available. It has proven to be of high importance to have an accepted and traceable design process. It's of importance to use simulator also for V&V activities, more focus on defining the operator tasks, roles and operational strategy, higher vendor involvement, and more precise specifications of the system requirements.

The full paper will provide more details on the project, including results and lessons learned from the modernization program in OKG as input to other modernization programs.

Presentation 20

Integration of Analog and Digital Systems in the Paks Main Control Rooms

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Hungary

Abstract

Since the beginning of the ‘90s, the Paks Nuclear Power Plant in Hungary has pioneered in the introduction of state-of-the-art digital I&C systems. The most significant area of these modernization steps was the entire replacement of the old safety I&C system in all four nuclear units.

Additional, formerly analog I&C systems have also been replaced due to obsolescence, disappearing spare parts, and poor functionality. All the replacement systems are now based on digital solutions. A good example of these latter activities is the total replacement of the turbine controllers.

These installations had major effects on the main control room. Some of the conventional panels and consoles had to be removed and new structures needed to be installed in place of them. In the design phase of these changes, it was a major task to establish and discuss the new HSI with the plant operators, some of whom had operated the analogue equipment for 15+ years.

In the safety I&C area, the following, formerly individual and separate subsystems were integrated in the new, digital equipment:

- Reactor Trip System
- Ex-core Neutron Monitoring System
- Emergency Core Cooling System
- Reactor Protection Central Cabinets
- Diesel Generator Load Sequencer
- Reactor Power Limitation System
- Steam Generator Protection System
- Loss of External Power Supply Automation

As the digital safety I&C system did not have its own, computerized human-machine interface, a one-way connection was established to the new process computers to provide information processing and presentation means for the safety system signals. In addition, parts of the MCR were entirely replaced with new panels and consoles. The same is true for the additional system replacements, such as the turbine governor change.

The lessons learned and experiences gathered in these modernization projects will be presented in the IAEA technical meeting.

Presentation 21

Validation Plans for a Modernized Main Control Room

Andrew Lang

Westinghouse Electric
USA

Abstract

An instrumentation and control (I&C) modernization project for the Main Control Room (MCR) of Ringhals Unit 2, located on the west coast of Sweden, is nearing completion. This modernization project replaces all control boards and desks, as well as all indicators and controls associated with the boards and desks located in the MCR. In many cases, physical indications and controls are replaced by interfaces that are computer-based. Subsequently, new training programs and procedures are being developed to support the operation of these redesigned indications and controls.

Because of the large number of changes to the MCR human-system interface (HSI), an Integrated HSI Validation is required to ensure that the newly designed control room operates as well as or better than the current control room. Since the MCR HSI, the training program, and the procedures all significantly affect plant operation, they together are viewed as the 'MCR Ensemble'. Therefore, it is the MCR Ensemble that will be the subject of the Integrated HSI Validation.

The Integrated HSI Validation test is scheduled to take place on the Ringhals Unit 2 simulator once the operators have completed a training program. The Integrated HSI Validation compares the performance of the new MCR Ensemble to benchmark values of performance measured in the current MCR, which were obtained from an assessment of performance using the current MCR Ensemble design. Criteria for the acceptance of the new MCR Ensemble have been established and documented in the Integrated HSI Validation test plan.

This presentation provides a description of the methods used to assess and confirm acceptability of the new Ringhals Unit 2 MCR Ensemble. It covers the scope, objectives and requirements of validation, the measures that will be used to evaluate performance, the criteria for acceptance of the design, and other aspects such as scenario selection, participants, facility, and administrators.

Presentation 22

Procedures to Demonstrate the Required Safety in Implementing Software-Based I&C and Modern Main Control Room Equipment in German Nuclear Power Plants

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Germany

Abstract

For German nuclear power plants an evolutionary concept has been selected to modernize the instrumentation and control as well as the control room equipment. According to this concept software-based I&C equipment is applied for instrumentation and control functions which are important, but not critical to safety. An analogue-digital hybrid solution is applied while screen-based equipment is partially implemented in the control room and the reliance on implemented software-based components is growing step by step. This modernization concept allows to collect operating experience from new equipment without changing totally and immediately the traditional plant operating principles and procedures, for which the control room staff and the equipment maintenance personnel is well trained.

With respect to a further application of software-based equipment for safety critical instrumentation and control functions and a broader use of screen-based control room equipment different safety aspects have to be addressed, like selection, qualification, and maintenance of the new I&C as well as the reliability of the new control room equipment respectively. Thus the German regulatory authority and its technical support organisations are developing appropriate procedures to proof an adequate safety, based on an amended nuclear regulatory framework.

The paper will address the related parts of the German regulatory framework and those aspects needed to demonstrate the safety of software-based systems important to safety.

Basic ideas/issues of the paper (may be not all of them):

- Updated regulatory framework and ongoing developments in Germany: BMU guidelines with a separate module dedicated to digital I&C equipment: KTA (3501 in revision, 3904 new version); DIN IEC Standards (digital I&C and CR); annex to BMU guideline: recommendation of procedures to perform the safety demonstration for digital I&C important to safety
- New PSA guidelines (chapter on models and data to perform CR/HF analysis)
- Acceptance criteria of operating experience for digital equipment in nuclear safety applications
- For the safety demonstration of safety critical I&C most credit can be taken from functional diversity
- There are three main directions of screen-based control room equipment qualification: digital incidence instrumentation (persistency against environmental conditions), software qualification, qualification of the composition and layout of screen-based signal and information presentation

- Different types of evidence to the safety demonstration for pre-developed software: analytic qualification (V&V documentation on testing and analysis); vendor's production process (documentation); vendor's quality assurance system (certification); source code; operating experience; constructive qualification measures like compilation of additional functional properties, not implemented within the original equipment, but engineered on the target system level
- Consideration of digital equipment in PSA: Comparison of HF related models; method to determine software complexity
- Diversity

Presentation 23

Application-Specific Qualification of Digital I&C Products in a Safety-Related Nuclear Context

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Canada

Abstract

Procured pre-existing commercial off-the-shelf (COTS) digital I&C equipment (i.e. products or components) can offer reliable and cost-effective alternatives to custom-designed (i.e. bespoke) systems or nuclear-grade products, provided they can be shown to meet the quality assurance, functional safety, environmental, and reliability requirements of a particular application. Referred to as "programmable electronic system" (PES) equipment, such equipment typically contains both complex logic that is vulnerable to systematic design faults, and low voltage electronics hardware that may be susceptible to external environmental conditions and is also subject to random faults. The process of confirming PES equipment is suitable and can be made safe in a given context of life-cycle use is referred to as application-specific product qualification (ASPQ) and can be challenging and costly. Nuclear power plant (NPP) designers and operators are increasingly faced with the need to perform ASPQ for procured COTS PES equipment (including sub-components or systems) for use in safety or safety-related applications. An overview will be provided of the approach developed at Atomic Energy Canada Limited (AECL) and successfully applied to digital I&C equipment intended for use in domestic CANDU® 6 nuclear power plants, the Advanced CANDU® Reactor (ACR-1000®), and special purpose reactors at AECL's Chalk River Laboratories. The approach was developed over the past ten years and has recently been adapted to be consistent with, and take advantage of new International Electrotechnical Commission (IEC) standards that are applicable to nuclear safety-related I&C systems. The approach enables the use of recognized 3rd-party safety-certifications (of PES equipment) to IEC standards to reduce overall effort.

Presentation 24

Enhancing the licensing of safety systems in response to the new digital technologies

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USA

Abstract

In January 2007, the NRC formed a digital instrumentation and controls (I&C) steering committee to provide management focus on the NRC regulatory activities in progress across several offices, to interface with the industry on key issues, and to facilitate consistent approaches to resolving technical and regulatory challenges. This committee formed task working groups to focus on key areas of concern. Each task working group was to develop specific project plans to address both short and long term solutions, if feasible. Essentially the plans are to identify and resolve technical issues that will result in more efficient licensing of digital I&C systems for new reactor applications and for retrofits at operating reactors/facilities. The specific short-term objective of this project plan is to identify digital I&C technical and regulatory issues for which Interim Staff Guidance (ISG) can be developed in time to support the review of the anticipated licensing actions. This is in addition to existing regulations and guidance which are being updated and revised for long term evolution and integration of the new digital technologies.

Presentation 25

Improvement of the Reliability of Shutdown Systems with Analytic Redundancy

Sungwhan Cho and Jin Jiang

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Abstract

Nuclear power plants are designed to tolerant various types of accidents. They are often known as “design base accidents”. These accidents include internal events caused by system component failures as well as external events such as seismic activities. In situations when an accident occurs, the first step to ensure plant safety is to terminate the fission chain reaction by inserting sufficient amount of negative reactivity into the reactor and keeping it in a sub-critical condition. The systems to carry out such actions are known as shutdown systems (SDSs). The reactor

protection system (RPS) in the Westinghouse (WH) designed pressurized water reactor (PWR) and the plant protection system (PPS) in the Combustion Engineering (CE) designed PWR can all be categorized as shutdown systems. There are two independent shutdown systems in Canadian Deuterium Uranium (CANDU) nuclear power plants (NPPs). They are referred to as shutdown system number 1 and 2 (SDS1, SDS2).

For safe operation of NPP and guard the reactor against potential accidents, the SDSs are constantly monitoring several key process parameters throughout the plant, such as temperature, pressure, level, and neutron flux, and to take actions whenever necessary. This is because, if the reactor is not shut down promptly in the event of a severe accident, core damage may occur. This could result in release of radioactive materials. The probability of such consequence can be reduced, if the reliability of the SDSs can be improved.

In a NPP, the outputs of the sensors used in the SDSs normally fluctuate due to process and measurement noises. These random fluctuations exist in measurements of flux, heat transfer, turbulence, vibration, and other mechanical and thermal hydraulic quantities. Unfortunately, such fluctuations can lead the trip signal get beyond the set-point to result in a spurious trip, which is very costly. Efforts should be made to reduce the chance of spurious trips whenever possible. However, the dilemma is that if one increases the trip set-point in an effort to reduce the probability of spurious trips caused by the process and measurement noise, he/she has inevitably increased the probability of severe accidents.

To increase the reliability of the shutdown system and the robustness against spurious trips, very high reliable components (1 failure/100000year) are used in the SDSs. Furthermore, multi-channel voting logics, diversity, and redundancy design for performing the same function with different measurements and hardware are also adopted. Another approach to improve the system reliability is through on-line surveillance tests and overhauling components during the reactor outage.

Due to high temperature, pressure, and hazardous radioactive environments within a nuclear power plant, it is undesirable to push to install more sensors in the pressure boundary, because installation of additional sensors can affect the integrity of the pressure boundary, and can potentially cause severe events such as loss of coolant. The cost of redundant sensors and instruments for each measurement point is also high. Redundant systems will also increase the cost of maintenance through the life-time of the system. Clearly, it is highly desirable to develop ways to improve the reliability of the shutdown system without resorting to installation of additional sensors and instruments.

This study investigates the use of an analytic redundancy in shutdown systems to improve their reliability without installation of redundant sensors in the pressure boundary of nuclear reactor systems. The behavior of process variables can be predicted based on fundamental conservation laws, such as mass, momentum, and energy balance. The predicted quantities are then used as redundant measurements in decision-making. Many techniques to improve reliability and credibility based on analytical redundancy have been studied for nuclear power plants over the last two decades. A limitation of these methods is that they may produce false decisions when operating conditions have changed. During system operation, it is possible that changes in the system be made based on operation procedures. When this happens, the static relationships among measured variables can also change. If these conditions are not considered in the analytical redundancy generation, a false decision will be generated.

Current study investigates the use of analytical redundancy in improving the reliability of SDSs. As an application example, the pressurizer low level trip parameter of SDS1 is selected and the mathematical model is used to capture the pressurizer level behavior. The detailed behavior of the pressurizer level, pressure and temperature under different perturbations are considered. Finally, the faults detection algorithm which can substitute physical sensors and instruments is demonstrated using the pressurizer model, and the reliability improvement of SDS1 has been examined.

Presentation 26

Evaluation of Distributed Control Systems and Network-based Controllers for Nuclear Power Plants

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Canada

Abstract

It is evident that distributed control systems (DCS) and network-based controllers will play an important role both in refurbishment of existing nuclear power plants and new builds. Before widespread adoption of such systems within nuclear power plant environments, comprehensive evaluation of such systems has to be carried out to ensure that they meet both national safety and international standards (such as IEC61513 for safety systems).

To evaluate DCS and network-based control for their potential applications in NPP a test bench has been constructed. The test bench consists of DeltaV DCS, Tricon V9 Safety PLC, Honeywell C300 DCS, HFC ECS06 DCS, and Siemens PCS7 DCS control systems. The system is also capable of connecting Ethernet, Foundation Fieldbus (FF) H1, Profibus, DeviceNet, Asi, Modbus, and HART industrial networks. For evaluation purposes, the system has been connected to (a) a NPP simulator provided by the Ontario Power Generation (OPG), (b) a physical steam generator model, and (c) a physical shutdown system (SDS) input signal generation panel. The measurement and analysis tools include: (i) National Instruments (NI) data acquisition system, (ii) FF H1 network analyzers, (iii) Matlab based testing workstations, and other general testing equipment, such as signal generators, scopes, and power supplies. The overview of this system is shown in Fig. 1.

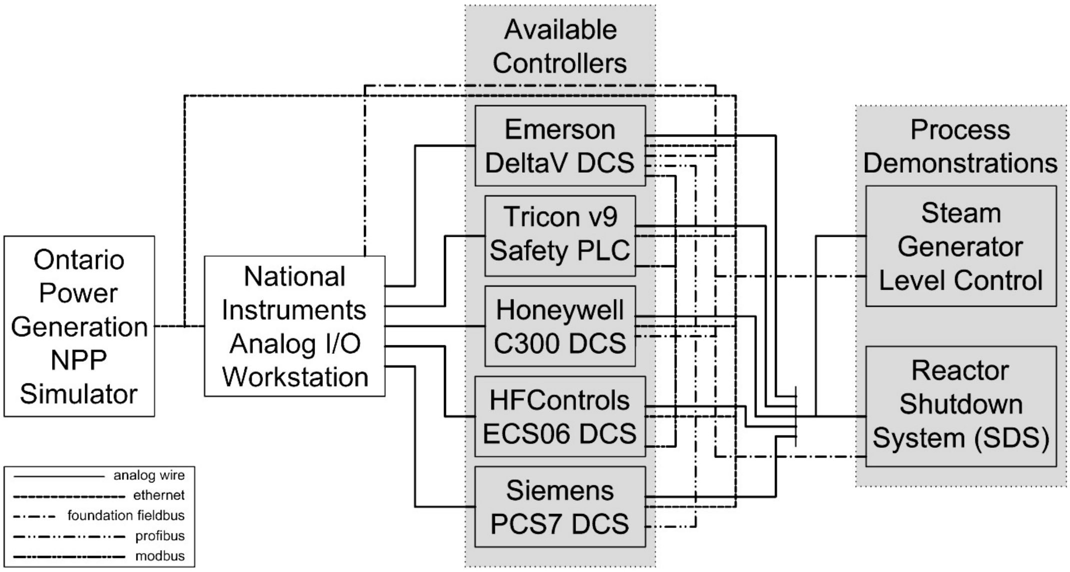

Fig. 1 Platform for evaluating DCS and network-based control systems.

This presentation mainly focuses on the study of the effects of a FF H1 network in network-based control systems (NCS). It is found that only a portion of the total delays with the use the FF H1 network is due to communication. Furthermore, the relationship between the network-induced delays and the NCS parameters (control program scan interval, FF H1 macrocycle, and control architecture) are investigated, analytical models to predict such delays are developed. In addition, tests are conducted to evaluate the performance of the NCS using the physical steam generator demo. The results reveal that the introduction of the FF H1 network may degrade the NCS control performance, when the delay is in the same magnitude of system dynamic range.

Presentation 27

An Experimental Testbed for Programmable Logic Controller Implementation of CANDU Nuclear Power Plant Shutdown Systems

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Abstract

Implementation of trip logic functions associated with the shutdown systems of CANDU nuclear power plants is the focus of this paper. The experimental aspects of this work includes development of the logic structure embedded in the shutdown systems of CANDU based NPPs.
Using independent I/Os and processors, this controller is compatible with CANDU shutdown systems that operate separately and independently from each other and from process systems as a main regulatory requirement for safety systems.

Physical test environment constructed as a part of this project is designed to simulate the measurements of in-core flux detector (ICFD) and ion chamber (I/C) signals.

The functionality and timeline of trip mechanism is tested in this method and compared against regulatory requirements.

The programmable logic and process controller used in this experimentation provides Triple Modular Redundant (TMR) architecture. For this purpose, a voting mechanism is used upon execution of shutdown logic algorithm on each independent channel.

The implemented system is capable of transient and steady-state error detection through triple redundant capability of the programmable logic controller.

In order to evaluate the designed shutdown system, various plant conditions and scenarios are experimented against the trip logic design using a broad range of parameter specification.

The main purpose of the experimental project is to analyze the respond characteristics and reliability of shutdown systems implemented in hardware controller and to compare the result against current implementations and regulation requirements.

Presentation 28

Historical Data Systems and Enhanced Functionality

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Abstract

Historical data systems (HDS) are one of the most significant components of unit process computer systems and overall plant process information systems. HDS collects and stores large volumes of process data from many sources, makes it accessible for analysis and reporting and provides an environment for applications which add new beneficial capabilities.
The paper describes the architecture, functions and user experience in operation of the overall plant process information system at NPP Dukovany. The system was put into operation in 2006 and presents an open technological infrastructure built up on commercial off-the-shelf product called Wonderware Factory Suite and in house developed a set of services which support common software infrastructure and functionality.

The system was originally developed for the technical support center and connects and integrates separate and heterogeneous information and control systems over-all plant each with its own set of components and internal databases. Some of those systems have been working since the NPP Dukovany started its operation in 1985 and were more or less upgraded some of those systems are up-to-date.

Beyond standard functions as data displaying, reporting, trending and linking the plant floor several applications of advanced data processing and decision support making have been developed. At the present time there are two main applications - the leakage detection and thermal efficiency monitoring and evaluation both of them based on analytical models described certain system behaviours being modelled. The system serves for technical staff, safety engineers, performance engineers, main control room operators and managers. Terminals of the system for accessing historical information and displaying results of advanced data processing are situated around the plant and in control rooms as well.

A combination of process data from independent data sources and advanced data processing is a concept leading to more safety, efficient and flexible operation. The system improves the plant-wide information flow, increases information understanding and cross-communication between different professional teams. Some obtained practical results are presented.

Presentation 29

Operator Information Systems in Indian Nuclear Power Plants

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Abstract

NPCIL has constructed and is operating sixteen nuclear power plants and has also increased its pace of establishing new power plants. The operator information systems used in these plants has reflected the state of art in indigenous technologies at the time of commissioning of these plants or when technology upgrade has been carried out. Computers have been introduced steadily for several functions for control, protection, operator information as well as test and monitoring and for Centralized Operating Plant Information system. This paper will discuss about existing Operator Support Systems used in various nuclear power plants in India and the features of the Operator support Systems in the upcoming plants. All these systems are designed and developed in-house.

All nuclear power plants in India are having centralized control room. Operator is supported with help of conventional instruments located on dedicated panels of each process control systems, alarm annunciator system common for all the systems, data loggers for periodic monitoring of other parameters with facility of on demand printing and selected parameter displaying, Event Sequence Recorders and high speed recorders for important analog signals for helping analysis of plant disturbance is available right from the first nuclear power plant, newer power plants are having higher capacity and rich feature for better analysis.

In earlier nuclear power plants, operator information system was acquiring data by tapping the signal from the current loop of control system. Due to advances in technology and success of Ethernet and computer networks in industrial environment, now most of the process control systems are sending processed data in binary format over Ethernet to the main computerized operator information system through gateway computers. Hence operator is having information of all parameters of any system on screen. In addition, the relevant Computer Systems at the operating stations communicate information to Centralized Operator Plant Information System (COPIS) over a VSAT network to NPCIL Corporate Office at Mumbai and this is available in an Emergency Centre.

Computerization of Operator Information Systems and data communication over Ethernet has reduced cabling, system maintenance and effort considerably. Number of annunciation windows in main control room are reduced drastically hence this reduces operators burden. High speed analog and digital recorders for more number of signal and with user friendly graphical user interface helped operating staff to find the root cause of disturbance easily and which in turn improved performance of operating stations.

Development work has been commenced for Computer Based C&I systems and Operator Support System for the next set of nuclear power plants under construction and the entire process is being standardized in terms of system architecture, hardware

and software. The new operator support system is also having computerized "Symptom Based Operator Support System" for handling emergency situations.

Presentation 30

Plant Information Systems Architecture for Integration into Control Rooms

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Abstract

Over the years, the need for additional monitoring in the Main Control Room (MCR) has been identified. In addition to the requirement for replacement of the obsolete Digital Control Computer (DCC) Computers, there is a corresponding need to decide on the best way to deal with the Human Machine Interface (HMI) replacement and/or extension. This paper proposes an approach to satisfy additional monitoring requirements while replacing the HMI.

The primary HMI for CANDU stations are Ramtek based display systems used for:

- Historical data storage/retrieval
- Annunciation
- Display (including schematics, trending and bar charts)
- Keyboard entry (including setting of control set points)
- Logging/report requests

The approaches available for extending the HMI monitoring functionality include the following:

1. Add the functions to the DCCs
2. Connect common process signals in parallel to the DCC and a supplementary computer
3. Extract DCC data into a supplementary computer
4. Add a standalone computer to supplement the DCCs if common process signals are not needed

In general, approach 1 (adding the functions to the DCC) has not been used due to software qualification costs, limitations of the DCC development environment, DCC in capacity and overall complexity risk. Approach 2 has typically been avoided due to the cost of wiring and ensuring appropriate isolation. Where common process signals are involved, extracting the data from the DCC into the supplementary computer, (approach 3) has generally been the most cost-effective approach. All the CANDU stations have installed DCC gateway computers to extract the DCC data and export it to supplementary computer systems. Where common process signals are not required, approach 4 is the simplest and most cost effective approach. Discussion in this paper focuses on approach 3 along with illustrative examples of new display functions and capabilities. With a well-designed architectural framework, this approach has the added advantage of being capable of easing further integration of process data extracted from other plant digital instrumentation and control systems.

Presentation 31

Alarm Presentation System (APS) at Ringhals Nuclear Power Plant Unit 2 – Sweden

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USA

Thomas Andersson

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Sweden

Abstract

The Ringhals new Alarm Presentation System (APS) is an advanced VDU based alarm system that replaces the existing hardwired alarm system and also the existing VDU standard list-based alarm systems.

- The existing hardwired alarm system consists of approximately 1600 small alarm windows in the consoles (Reactor, Turbine 21, Turbine 22 and Electrical).
- The existing VDU standard list-based alarm supervision systems are used for example for: Plant Computer, 6 kV Motor-winding Temperature, Generator and Transformer protection.

Ringhals had a quite detailed plan to replace their old alarm system located on each operator's console, vertical panels and in different areas as VDU standard list-based alarm supervision systems with a common VDU based alarm system. The new APS would preserve the good features from the hardwired alarm system of parallel presentation and pattern recognition and also provide the additional capability such as access to all relevant alarm information with links to: alarm response procedures, related process displays, background information, etc. from each alarm system work area.

In the new design, six (6) 18" flat panel displays (FPDs) are arranged in a 2x 3 matrix to create a virtual screen of 5120 pixels by 2048 pixels. Four (4) of the FPDs are dedicated to show fixed alarm tiles similar to the existing hardwired system, thus providing the parallel display of information so valued by the control room operators. With the four FPDs it is possible to display a maximum of 336 alarm tiles with the size of 24 x 48 mm.

The two remaining FPDs are used to display the individual alarms in a group alarm, display of various alarm lists (acknowledged, unacknowledged, reset, bad quality, suppressed alarms, etc), process and control displays, trends, historical data and procedures.

Each alarm tile can be driven by an analog or digital point. For analog points, the alarm tile message is dynamic thus is able to show multiple alarm levels.

The operator can click on each alarm tile to acknowledge/reset it, or with a right click to bring up a context menu with choices as for example:

1. Display alarm response procedure in HTML format via a browser
2. Call up the process display where the alarm is shown within the correct plant-operating context.
3. Call up the display that shows the alarm logic highlighting the active part of the logic.
4. Display detailed information about the alarm (alarm set points, actual values, etc)
5. Disable or enable the alarm.
6. Add an operator note to any of the alarms.

By double clicking on a group alarm, the operator can display the individual alarms in the group in a matrix or a list format and access the related information.

The layout, alarm text, alarm grouping and process point information is configured with a simple visual interface and the data is stored in a database making it easy for the users to layout the alarm tiles, group alarms and specify configuration data. The APS program reads the data from the database and automatically generates the layouts and groupings for the runtime environment.

Presentation 32

Instrumentation upgrade at the Training Reactor of Budapest University

Bálint Szabó

Institute of Nuclear Techniques, Budapest University of Technology and Economics
Hungary

Abstract

After more than 30 years of operation, parts of the instrumentation of the Training Reactor of Budapest University is still working with original circuit boards, components from the 70's. Nowadays, with the convenience of using integrated circuits and computers, it is expectable to change from the old technology to a newer, more up-to-date measurement system, with higher reliability and with services that cannot be made with current instrumentation.

Our radiation control system will be soon upgraded to a new, self-developed high-tech data acquisition system. Its heart is a microcontroller based, standalone circuit. With this change reliability will dramatically improve due to small number of components, to modular firmware witten in assembly, and to the simple power supply unit.

We solved the problem of remote data access (data representation), data archivation, and on-line measurements using ethernet network connection, a linux based database system with automated data upload softwares, and common, off the shelf web browsers.
Software reliability was maximized by using only commercial, proven software applications which was tested by million of people.
Data security is made by storing collected data on more computers, in more formats (raw text file, database file), updating them minute by minute. Using only local network addresses, encrypted data transfers, and secure connections very good access security is achieved.

This way of measurement system development helps us to save time, and to make cheap systems with high flexibility.

In my presentation I will show these systems in detail, with all the novel and useful features we made and are using currently at our training reactor.

Presentation 33

I&C Systems Covering the Entire Plant Life Cycle – Hitachi's Latest Achievements

Hideo Harada

Hitachi Ltd
Japan

Abstract

On a global scale, as the importance of harmonization between environmental and economic activities (energy consumption) increases, so does the importance of the stable management of maintenance for existing and of the construction of new nuclear power plants (NPPs), an earth-friendly and significant energy source. As an essential element of any NPP, I&C systems management spanning the entire NPP life cycle is becoming a crucial issue.

Based on its continuous involvement in NPP design, fabrication, construction and maintenance since the 1970s, Hitachi has accumulated vast experience of NPP I&C systems in all the phases of the NPP life cycle:

(i) New Build Projects (from Planning to Commissioning)

Hitachi's latest achievement is the Advanced Boiling Water Reactor (ABWR), applying its integrated I&C system, NUCAMM-90e, consisting of HIACS-7000 DCS and Flat Panel Display-based HMIs for the major plant scope items including Safety, Non-Safety NSP and BOP systems. By incorporating new project management methods based on its lessons–learned from past projects and using its entirely in-house DCS platform, Hitachi has provided overall system solutions at each phase of construction.

(ii) Refurbishment Projects

Hitachi took the lead in refurbishment from the planning to the implementation based on its thorough understanding of the NPP requirements and specifications. To counteract obsolescence issues and to enhance system value, Hitachi's I&C systems modernization and upgrading solutions focused on the dedicated DCS equipment and underlying networks as well as the HMI devices in the operator's console:

1. accommodating the latest technical trends (e.g. replacing central computing with distributed computing); and
2. reflecting customer needs for improved operability and maintainability (e.g. expansion of plant automation scope and reduced plant start-up / shutdown times).

(iii) Tackling Obsolescence

To provide stable I&C systems operations and maintenance in existing NPPs, properly addressing the issue of device and equipment obsolescence is rapidly

becoming crucial, as rapidly, in fact, as the (accelerating) pace of progress in the electronics industry. There are 3 major factors affecting electronics device life:

1. obsolescence of the functions which the device provides;
2. obsolescence (non-availability) of maintenance support; and
3. obsolescence due to device physical limitations

Through its 30+ years of experience in maintenance work in existing BWR/ABWR NPPs, Hitachi has achieved the following capabilities:

- Development of technologies such as derating and screening of parts, monitoring and diagnosing, and design methods for the accommodating of possible future renewal.
- Solution plans to support customers for the maintenance including annual inspection and the refurbishment.

This paper addresses Hitachi's representative experiences and latest approaches to provide the solutions to supply I&C system for NPP covering all the plant life cycle.

Abbreviations:

- BOP: Balance of Plant
- DCS: Distributed Control System
- HIACS: Hitachi Integrated Autonomous Control System
- HMI: Human Machine Interface
- I&C: Instrumentation & Control
- NSP: Nuclear Steam Plant
- NUCAMM: Nuclear Power Plant Control Complex with Advanced Human-Machine Interfaces

Presentation 34

Automated Simulation Integration Supporting Digital Control System Upgrade Projects

Keith Bradshaw and Harry Snowden

VASPAC Inc
Canada

Blaine Leslie and Alex Chan

Bruce Power
Canada

Abstract

This paper describes how modelling and simulation methods were used to uncover the root cause of a control stability problem in a digital governor for an 800 MW nuclear powered steam turbine which were detected during the original commissioning process.

When a design review was launched into the turbine governor upgrade project, the following tasks were defined for the review team:

- Determine the root cause for the stability problem
- Determine how software verification and validation didn't detect the stability issues prior to installation
- Determine how the digital governor software could be integrated into the plant's full-scope and desktop training simulators

The first part of the paper focuses on determining the root cause for the stability problem. The problem was reproduced using simulation. The control system software was connected to the standard IEEE turbine model used for testing the system prior to installation. It was not possible to accurately reproduce the stability problems. The standard IEEE model does not include the valve stroke rates, and most importantly, does not model LP valves between the reheater and the low pressure turbines. A more representative turbine model was developed where all turbine valves were modeled complete with stroke rate lags. With the improved turbine simulation model, the stability problems were accurately reproduced and the root cause of the stability problem was identified. Corrective design was verified, using the new model, prior to installation and final commissioning.

The second part of the paper describes how the control software was then integrated into the plant's desktop and full-scope training simulators. Manually upgrading the control system emulation was considered to be a huge task due to the program's immense size. Further, it was recognized that enhancements to the digital governor would be forthcoming in the near future. The effort to maintain digital control system software emulation could be a huge drain on simulator maintenance resources if left to be done manually.

A unique proposal was put forward to write a code generator that could automatically translate the control system configuration into standard simulator code modules. The code generator has been developed and the Bruce Power simulator team has commissioned the new auto-generated turbine governor control emulation models.

Using this automated simulator integration approach, the Bruce Power simulator team plans to integrate subsequent digital control system software prior to field installation to reduce the risk of unforeseen commissioning and operational problems.

Presentation 35

A Digital Regional Overpower Protection System for CANDU reactors

Victor Mihaylov

AECL Chalk River Laboratories
Canada

Abstract

The Regional Overpower Protection (ROP) system of a CANDU reactor protects against overpower in the fuel that could arise from reactor core flux shape distortion, e.g., during a loss of reactivity control transient, by shutting down the reactor when signals from an array of in-core flux detectors exceed a pre-calculated trip set point. The ROP detector trip set points, for each of two independent shutdown systems, are calculated to ensure prevention of fuel dryout for a design basis set of nearly 1000 accident flux shapes. The actual detector readings are adjusted frequently to account for normal perturbations in the operating flux shape caused by fuel burn-up and on-line re-fueling. These adjustments are based on detailed flux shape and coolant flow calculations performed off-line and are implemented by manual adjustment of the gains of the flux detector amplifiers.

A computerized implementation of the ROP system will provide superior reliability, maintainability, and flexibility by means of on-line signal validation, automated tracking and adjustment of detector signal processing parameters, and continuous on-line self testing, all contributing to increased safety of operation. A computerized implementation also offers the opportunity to significantly increase the ROP margin-to-trip through on-line tracking of the reactor flux shape.

This presentation explores the issues, benefits, and risks of a digital implementation of the ROP system from the perspective of retrofit and upgrade projects for CANDU-6 reactors.

Presentation 36

Development of a 1E Qualified VDU System (Tentative)

Dave Trask

AECL
Canada

Abstract

(not available)

Presentation 37

The Application of Fieldbus and Other Networking Technologies in NPPs: an Overview

Lawrence Yu

Atomic Energy of Canada Limited
Canada

Abstract

Communication networking technologies such as fieldbus and wireless have become increasingly prevalent in non-nuclear industrial applications. Although they are currently not an integral part of the CANDU I&C architecture, it is expected that future CANDU system retrofit and new build implementations will leverage such technologies to reduce installation and maintenance costs, and to improve reliability and availability. First, an overview of the benefits and the current state-of-the-art of the main process control fieldbus technologies (Foundation Fieldbus, PROFIBUS, HART) is provided. Current and potential nuclear-specific applications are described, with the focus on the applicability of the technology to upgrading existing NPPs. Finally, technology adoption issues are reviewed, with particular emphasis on qualification for potential use in nuclear safety-related applications.

Presentation 38

Implementation Of Digital Technology In A Standby Generator Upgrade

Steven Poyner and David Calkin

Ontario Power Generation
Canada

Abstract

Pickering B is a four unit 500MW CANDU facility operated by OPG and located approx 40km east of Toronto on the shores of lake Ontario, Canada. Standby generation capability provides one of the cornerstones of a nuclear safety system, an awareness heightened by the Aug 2003 major grid blackout. OPG's decision to embark on an ambitious major controls upgrade of the standby generators at Pickering B was taken only after careful consideration of equipment obsolescence, failures and end of life projections outweighing the risks involved, eventually becoming one of the major planks in OPG's rehabilitation program.

Advances in technology has enabled the use of programmable logic controllers (PLC's) to perform the functions of the antiquated analogue governor and relay driven generator control package previously installed on six 7 MW gas turbine generators. Using off the shelf technology and levering systems developed for general industrial use enabled the project team to develop a cost effective tailor-made solution that will be supportable for many years to come.

This paper deals with the challenges of developing solutions based on standard industrial platforms in a nuclear safety support environment; where equipment reliability, performance and predictability are paramount. Engaging vendors traditionally involved in the gas turbine retrofit marketplace made the software development quality assurance (SQA) challenges especially acute, however the tradeoff of proven turbine technology over considerations of nuclear experience resulted in a very positive outcome.

The authors have been associated with the development and installation of the referenced control system over a number of years and bring together strong project management and control system design experience.

Presentation 39

The Future Impact of Safety-Fieldbus

Joseph Bodnar

Siemens Canada Limited
Canda

Abstract

Digital communications within a safety system is not new technology, as safety bus communications have been in use with several programmable safety systems from as early as the 1980s. Safety-fieldbus technology is growing, but how quickly this technology will be accepted in the process industry remains to be seen. Continued support from the major process fieldbus organizations (Profibus and Foundation Fieldbus), and process automation suppliers will ensure this technology is propelled into the future.

Many end-users see the advantages of a fieldbus technology within their process control systems, and are looking to see how these advantages could be implemented into their safety systems. Reduction in installation and commissioning time, simplified wiring, comprehensive diagnostics, improved diagnostics for field devices, communication ties to asset management systems, savings in wiring and terminations, simplified maintenance/modification over the lifecycle of the system, all would be considered major areas where fieldbus technologies could provide a valuable addition to the current mindset of instrumentation and control.

The ability to collect the information available from intelligent Safety Integrity System (SIS) components will enable analysis of the safety performance of the SIS, helping users avoid spurious trips. This new approach to safety engineering has been accepted in the manufacturing industries and is under review within the process industries. Even with current safety-fieldbus protocols TÜV Certified or in compliance to IEC 61508 Safety Integrity Level (SIL) 3 (PROFIsafe, FF-SIS) , acceptance that a safety-field bus can be as safe and reliable as traditional 4-20 mA systems is one of the first major hurdles to be addressed.

This paper will further discuss the technology behind the safety-fieldbus, as well as the current status and future direction of the safety-fieldbus technology.

Keywords:

safety-fieldbus, process safety, PROFIsafe, FF-SIS, SIS

Presentation 40

Replacing Analog Meters with Digital Solutions

Craig S. Irish

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USA

Abstract

The nuclear industry is constantly faced with the challenge of maintaining the reliability of old analog instrumentation and controls. The industry has done an excellent job of maintaining 1960's vintage equipment especially in the face of dwindling OEM support, shrinking internal plant expertise and growing parts obsolescence. An excellent example of this situation is the on-going effort to maintain reliable analog meters. This presentation will provide the highlights of a case history to replace 700 analog bargraph meters at a major southeastern nuclear utility within the United States.

The presentation will begin with the need to replace the existing Versatile Model 2000 meters due to increasing failures, compounded by parts obsolescence and expensive refurbishments. The existing meter uses a fluorescent vacuum display, the life expectancy of which is usually considered to be 5 – 8 years on average. A replacement solution was selected using the OTEK digital bargraph meter designed to be a one for one replacement with the existing meters. A new meter was designed to work within the existing Versatile housings. This allowed the utility to maintain the existing wiring, terminations and not make any cuts in the control room panels. This approach also helped minimize the changes necessary to plant drawings, training manuals and technical specifications greatly reducing the cost of the design change. Great effort was put into the design and production of the displays to meet the strict requirements of the plant operators.

Approximately 700 bargraph meters were supplied, consisting of single, dual, vertical and horizontal styles. The number of custom scale plates exceeded 150.

The meters were supplied with qualification in accordance with IEEE Std. 344 (Seismic), IEEE Std. 323 (Mild Environment), IEEE Std. 7-4.3.2 (Software V&V) and EPRI TR-102323 (EMI/RFI).

The immediate benefits to the client include:

- Improved HMI
  - Automatic tri-color bargraph
  - Bright LED display
  - Excellent visibility under virtually all light conditions, greater distance

- Improved MMI
  - Multiple options for Serial Communication

- State-of-the-art technology
  - Dual micro-processor = faster response time

- A2D failure indication
  - Alarm set points

- Main processor failure detection

- Higher accuracy (4 digits vs. 3)

- Greater Reliability
  - Life Time Warranty

- Upgrade possibility to DCS/SCADA interface

Each meter is fully tested and calibrated prior to delivery to the client. The meters were supplied with new documentation consisting of Qualification Reports, factory acceptance test results, and Instruction Manuals.

The presentation will include pictures of the new meters including pictures of the control room with the meters installed.

The case study is one example of our on going efforts to provide digital solutions to aging analog meters within the nuclear industry.

Presentation 41

High Speed Digital Data Logging for On-Line Monitoring and Diagnostics in CANDU Nuclear Power Plants

Bhaskar Sur, Harry Storey, Vinicius Anghel, Tony Hinds (retired)

Atomic Energy of Canada Ltd.
Canada

Dean Taylor, Evan Young (retired)

Point Lepreau Generating Station
Canada

Abstract

High speed, high fidelity digital data from sensors and systems are highly effective for monitoring the health of nuclear power plant components and instruments, and for quantitatively verifying dynamic response parameters that play a crucial role in the plant safety analysis. All Canadian nuclear power plants use high speed data logging systems to monitor safety system sensors and components, such as in-core flux detectors, pressure and flow transmitters, shut-off rods, poison injection nozzles, etc., via data acquired both during steady-state operation, and during manually induced reactor trip tests. Most stations acquire high speed data only during campaigns going into a planned outage with stand-alone systems, such as AECL's Noise Analysis System (NAS), temporarily connected to one or more safety system instrument channels. Point Lepreau Generating Station has, since 1996, operated the High Speed Data Logger (HSDL), a high-speed, high fidelity data acquisition system that is permanently connected via isolation amplifiers to a subset of safety system instruments, including all safety system in-core flux detectors. The HSDL automatically collects 16-bit data at 20 ms intervals, for five minutes preceding and five minutes following any reactor transient, such as a reactor trip. The HSDL can also be triggered manually to collect data during steady state operation.

This presentation will provide an overview of high speed data acquisition systems and the techniques and results of high speed data analysis for CANDU reactor components and instruments, with emphasis on Point-Lepreau's permanently installed HSDL system. The issues and merits of on-line monitoring and diagnostics via future permanently installed data logging systems will also be discussed.

Presentation 42

On-line Condition Monitoring in Nuclear Power Plants

H.M. Hashemian

Analysis and Measurement Services Corporation
USA

Abstract

Signals from existing sensors in nuclear power plants can be monitored while the plant is operating to verify the performance of the sensors and associated instrumentation and to diagnose process anomalies. This paper provides some examples of analytical tools that have been developed and successfully used in nuclear power plants for this purpose. The paper will also present a review of computer-aided maintenance technologies as well as active methods for employing test signals to measure sensor performance and to identify problems in their cables and connectors.

Two international activities involving on-line monitoring technologies for aging management in nuclear power plants are underway. One is sponsored by the International Atomic Energy Agency, and the other by the International Electrotechnical Commission. These activities will be described in the paper.

Presentation 43

Integration of a modern I&C monitoring system to operate Beloyarsk Nuclear Power Plant (Russia)

Georges Garcia and Jean-Louis Deimerly

Atos Origin Integration
France

Abstract

In the frame of the TACIS program, the European Community funds former USSR countries to improve the safety of nuclear power plant. For the Beloyarsk plant (nuclear fast-breeder, located in the Urals, Russia), the EC awarded to Atos Origin the refurbishment of the computerized information system in 2000, for a system to be installed and run on site in 2004.

The design, integration, delivery and validation of the system has been performed by Atos Origin with a close cooperation of the Russian End User, Russian metrological

institute and German sub-contractor, monitored by Italian Technical and Procurement Agent acting under behalf of the EC.

On site installation has been performed by a joint team including Russian End Users, Russian and German sub-contractors and members of the French Atos Origin team.

The main goal of the Process Monitoring System delivered to Beloyarsk NPP was to concentrate and present on reliable operator desks' workstations, offering up-to-date HMI, all available information about the state of the nuclear reactor and utility systems (reactor protection, turbine supervision, fuel management), so as to allow the safe operation of the fast breeder. Consistently, the scope of the contract also included the improvement of the existing training simulator.

The project involved more than 15 suppliers and subcontractors. The main challenge has been to integrate the modern technology and cultural use of the project in the site environment with mandatory limited impact on the operation of the fast breeder.

In close co-operation with the Russian parties and the EC representatives, the model used to perform this contract has been to integrate and test (FAT) the whole project, including associated services, in the Atos Origin premises in France and to deliver, install and qualify on site (SAT) a validated and consistent project.

Site works (Installation, Acceptance Tests and operational switch from the former to the new system) have been given a particular attention and performed according to a safety and open-oriented process, involving progressive checks and parallel running techniques of the 2 systems so that the operators gain confidence in the new system and the switch-over could be performed during a standard outage…

The issues which have been faced and overcome, occurred during the complete duration of the project and addressed very various areas of it: technical interface to existing "exotic" sensors, management of Cyrillic configuration database, end user's requests regarding HMI, supplies modifications in quality and nature, qualification process, delivery process, on-site installation, parallel running of the 2 systems

As a final result, beyond rules and regulations, the success of such project closely depends on a tight multicultural co-operation of the involved parties and relevant confidence in each other's professionalism, whatever their cultural different approach to technical questions.

Presentation 44

On-Line I&C Monitoring At Nuclear Power Plants

Ahmed Osgouee

Ontario Power Generation
Canada

Abstract

The safe, economical, reliable and efficient operation and control of a complicated industrial system such as a Nuclear Power Plant requires knowledge of the state of power plant. This knowledge is obtained by measuring critical plant parameters with sensors providing information about Process State.

Periodic calibration of certain safety-related instrument channels is required to help ensure safe, efficient, and economical operation of nuclear power plants. Many calibrations are required to be performed at a frequency prescribed by the plant's Technical Specifications to provide assurance that the instruments are performing within their specified limits. This type of traditional calibration approach is costly and does not make optimum use of the data collection and also increases plant operational costs through increased I&C maintenance, the potential impact on instrument availability, increases personnel radiation exposures, and also increases potential for damage to equipment.

Fault or miscalibrated instrumentation channels might lead to the problems such as reduced plant performance and efficiency and process uncontrollability and instability that they have always negative economical consequences due to forced shutdowns and losses of efficiency.
Continuous transmitter accuracy monitoring can minimize out-of-calibration conditions while also reducing the frequency of calibration.

On-line monitoring of I&C helps to improve the economic performance of nuclear power plants and maintaining high levels of safety. On line monitoring can be used to identify instrument channels that are not functioning properly and that might require adjustment or corrective maintenance.

Many fault detection techniques have been studied for use in Nuclear Power Plants including expert systems, model-based techniques, state estimation techniques, artificial neural networks, neuro-fuzzy and hybrid combinations of these techniques. The Multivariable State Estimation Technique (MSET) Surveillance System that has been used in a variety of applications will be discussed in this paper.

Presentation 45

Digital Chart Recorder Replacement Main Control Room Bruce A NGS

Randy Long

Bruce Power
Canada

Abstract

Problem Statement:
Maintenance on existing pen chart recorders is a highly labour intensive activity. In addition obsolecense of these devices is making maintenance virtually impossible. Not having pen recorders for monitoring system parameters is promoting unfavourable Operational environment

The driver for the Bruce A Digital Chart Recorder replacement project was Obsolescence. Maintenance spent time and effort in trying to seek parts or exact replacement from industry suppliers. This effort did not result in favorable results. Reverese Engineering option possible but these costs exceed full out replacmentent with new digital device. In addition, the advantages present with new digital chart recorder clearly reveal why an upgrade to new technology was the correct business decision.

The requirements for nuclear applications can be considered unique to the industry. The proposed presentation will describe these requirements and how the manufacture was able to successfully comply. Early involement with the Vendor and OEM representatives will be highlighted. This involvment with our Engineering and Operations staff made the transition from electromechanical to digital relatively seamless. Flexibility of new digital parameters aided greatly in making improvements in Operations monitoring capabilities.

As an overview the proposed presentation shall discuss the following:

- Bruce Power
- The Project Scope
- Software Qualification
- Human Factors
- Stakeholder and Vendor Participation
- Device Configuration and Archival capablities

The proposed presentation shall be in a PowerPoint format.

OLD VS NEW

SAME

Vertical Trend

Scales on Top

IMPROVED

Engineering Units

Time & Date

Presentation 46

Replacement of the NRU Sequence of Events Recorder

M. O'Kane, J. Arnold, D. Trask and R. Didsbury

AECL Chalk River Laboratories
Canada

Abstract

NRU is a 135MW(th) reactor owned and operated by Atomic Energy of Canada Limited. In operation since 1957 it is capable of producing neutron fluxes in the order of 2.5 x 10^{14} neutrons/cm^{2}s. It serves as both a valuable scientific instrument as well as the producer of a significant quantity of the world's medical isotopes, most notably molybdenum-99.

NRU's Sequential Events Recorder (SER) is a early 1980's vintage computer system that monitors the reactor's trip and alarm system. It is a supplemental alarm annunciation system that operates in tandem with the main alarm panel.

The SER operates by scanning approximately 1024 contacts to determine if these contacts have changed state (i.e., Closed to Open or vice versa). Once a contact has changed state the SER then determines if the event is an alarm or a clearing of an alarm. The SER then prints this information on the printer as well as on the Video Display Unit (VDU) with a timestamp. Early in 2006 NRU's SER started to show signs of deteriorated performance. This resulted in a project aimed at replacing the SER with a modern digital solution

The solution revolves around the use of AECL's Advanced Control Centre Information System (ACCIS) as the system-machine-interface, commercial off-the-self computer hardware components and RTP corporation's RTP-2500 controller. This presentation will describe the details of the solution architecture as well as the design, installation and commissioning activities leading up to placing the solution in-services. Lessons learned from the project as well as opportunities for expansion and extended use of the system will also be discussed.

Presentation 47

Bruce Display and Printer System Replacement

Tim Rector

AECL Chalk River Laboratories
Canada

Abstract

Units 1 and 2 at the Bruce A NPS use Data Disc 6500 display generator units to render operator displays. These display units date from the early 1970s and are no longer practical to maintain. A modern replacement for this display system is currently being developing for Bruce Power by AECL.

The replacement design introduces commercial, off-the-shelf hardware and a client/server software architecture despite the continued reliance upon the DCC for display generation. The hardware consists of LCD panel and desktop displays, fibre optic connections for electrical isolation, FPGA cards for communication with DCC ports, and other modern equipment. The replacement display system does not rely upon any changes to the DCC software while continuing to provide all of the existing display functionality plus the capability for enhanced displays (e.g. colour).

The performance of the replacement hardware and the flexibility of the system architecture provides a future capability towards migrating display generation from the DCC itself to a separate plant display system.

The presentation outlines the replacement display system, describes some of the issues faced during the implementation of the design, and the future system potential using the current replacement hardware as its basis.

Presentation 48

Alarm Management for Humans - Dealing with Alarm Inflation and Overload

Garry R. Mitchel

Atomic Energy of Canada Limited
Canada

Abstract

The advance of digital technology in nuclear control rooms has fostered exponential growth in the quantity of operating data of potential interest to the humans responsible for the surveillance and navigation of plant operation. Unfortunately, human cognitive limits have not evolved at the same speed, and thus the data have to be distilled into actionable information if the operators are to avoid being overwhelmed.

Nowhere is this need more acute than in dealing with alarms, which by their nature are intended to interrupt and draw attention to conditions unforeseen. Atomic Energy of Canada Limited (AECL) has pioneered the use of the CANDU®[*] Alarm Message List System (CAMLS) which sorts, prioritizes and condenses a chronological alarm stream (or flood, during an upset) into a risk-prioritized, actionable list, with less urgent information stored and accessible.

This presentation will first outline the philosophy and design of CAMLS, designed as a retrofit to existing plants with a pre-existing set of alarms, thresholds and associated message 'lists'. Then we describe how to build on that alarm-processing experience for new plant designs (e.g. the Advanced CANDU Reactor®[**] (ACR®[**]).

Processing is most efficient if input is digestible. The CAMLS experience has underscored the value of up-front alarm design, with consistent guidelines, syntax, prioritization criteria, and links to variable definition ('points', 'tags') plant-wide. Alarm inflation must be contained - increasing digitization has led to an increase in the number of different alarms in a plant, from hundreds to tens of thousands. Not all are appropriate on the main operators' console, and so should be re-directed according to end-use (e.g. MCR, fuel handling, monitoring computer diagnostics). Systems designers are human, too, and so will naturally prescribe alarms with different 'rigour' from system to system, although all will defend the critical importance of their designed systems. Thus there is a need to review alarm design for consistency from system to system.

[*] CANDU® is a registered trademark of Atomic Energy of Canada Limited (AECL).
[**] Advanced CANDU Reactor®, ACR®, ACR-1000®

Presentation 49

Human Factors Integration Considerations in Technical Upgrades

Kira Berntson

Bruce Power
Canada

Abstract

The Canadian Nuclear Safety Commission (CNSC) guidance documents related to human factors do not specify design implementation as a specific aspect requiring review, as is done in NUREG-0711. However, even without being separately identified, it is clearly understood by any Human Factors specialist that issues arising from the implementation of a design need to be carefully reviewed and addressed. One of the key areas where implementation issues become a concern is the area of technological changes. This can be of particular concern in multi-unit control rooms, such as are used on many CANDU stations, as periods will exist where different designs and technologies are in use for the same purpose in the same control room.

This paper will look at the techniques used by the Bruce Power Human Factors section to ensure that the existence of multiple technologies does not have a negative performance impact for the personnel operating and maintaining the station. This will be done using examples from past Bruce Power experience. Lessons learned will also be discussed.

Presentation 50

Modernization Expectations

Bryan K. Patterson

Human Factors Practical Inc.
Canada

Abstract

Key words pop up in discussions on modernizing the control room of a Nuclear Power Plant. The author presents these modernization watch words and gives at least one practical and pragmatic principle that should apply to each of their designs and their implementation. Workstations and Workstation Displays, Overview Display Panels, Annunciation, Operator Aids, Mental Models, Soft Controls, Soft Procedures will be addressed.

Bryan Patterson was the Electrical Commissioning Engineer and the Controls Group Superintendent at the Point Lepreau Generating Station between 1975 and 1993. Responsibilities included Control Computers, Instrumentation and Safety Systems.

Presentation 51

Control Centre Change - Operator Support Experience, Principles and Challenges

Eric Davey

Crew Systems Solutions
Canada

Abstract

Operational experience worldwide has demonstrated that the effectiveness and efficiency with which tasks are performed in plant control rooms are primary enabling factors in attaining effective plant operations. To achieve and sustain top production performance, and regulatory and peer performance ratings, utilities require control rooms whose workspace layouts, functions, support tools, and resources effectively support shift staff in working together to meet station operation objectives.

Beginning with initial implementation and commissioning, and extending through operations and refurbishments, nuclear generation facilities undergo continual change in response to evolving commercial, technological, regulatory and societal factors. This ongoing change leads to hybrid facilities with a variety of equipment capabilities, technologies, ages, support needs, and operational practices. While the introduction of new equipment and technologies can bring advantages in individual capability improvements, the evolving mixture of old and new equipment and technologies can increase operating and maintenance complexity and constraints.

This paper will discuss some of the impacts on control room operations of working within hybrid control room facilities using examples drawn from CANDU plant experience.

The paper will briefly review the evolution of CANDU control room capabilities, discuss some of the common impacts of operating within a continually evolving control room work environment from an operator perspective, describe principles for guiding equipment change implementation and introduction to better support control room operators in their operating duties, and outline some generic challenges facing control room change implementation and operator support.

Presentation 52

A Method to Display Massive Tile-Alarm Information on a Space-Limited Display

S.M. Seo, J.Y. Keum, G.O. Park, Y.S. Suh and H.Y. Park

Korea Atomic Energy Research Institute
Republic of Korea

Abstract

In a conventional control room of a Nuclear Power Plant, a great number of tiled alarms are generated when it is under emergency conditions. As it evolves into a hybrid one, the annunciator-based tile display for indicating alarm status is required to be replaced by a computer-based tile display. Where this happens, it places an additional task burden on control room operators, because they have to navigate and acknowledge those tiled alarms generated on a space-limited display, generally in a hierarchical manner.

In this paper, we present a method, called ETD(elastic tile display), to display effectively large quantity of the tiled alarm information on a limited displaying space like LCD display unit. If it is to be helpful to the operators of hybrid control room, it should support their navigation tasks accessing the alarm information with less cognitive effort. To validate this, we used CPM-GOMS(cognitive-perceptual-motor GOMS) technique which is widely recognized as a tool for evaluating usability of HSI(human-system interface). We found out the ETD method to be a promising one that can increase operators' performance without imposing a significant additional burden on them.

Presentation 53

Halden Project activities relating to hybrid control room automation systems

Svein Nilsen

OECD Halden Reactor Project
Norway

Abstract

This paper is a high-level presentation of OECD Halden Reactor projects activities that relate to hybrid control room design, implementation and V&V. The Halden Project has a long tradition working on the human factors related aspects influencing the usability of any (hybrid) control room.

Yet, the paper start out with a review of experiences on a much broader scale that takes into account many other contributing factors involved in a control room upgrade projects, e.g. change management principles, work procedures and methods, communication and crew interaction as well as crew training. The reported review was implemented back in 2001 but it is deemed of current value to many planned and ongoing control room upgrade projects. The reported project distributed a questionnaire to obtain input from as many industry contracts and members of the HPG as possible. The review is thus a synthesis of experiences and opinions of the persons and organizations participating.

Next, the paper presents a selection of human factors issues that should be taken into account when doing control room design and upgrades. Moreover, some methods to implement V&V related to these issues are presented – some of which are based on experiments in a simulator environment. These V&V methods are exemplified by a few real plant upgrades in which HRP staff acted as consultants.

The paper ends with a description of an ongoing project that uses virtual reality (VR) technology to enable the study of human factors issues at a very early stage during the design process. Applying VR for this purpose has obvious potential advantages as it can be used to identify costly errors within the design. It has already been proven that VR can be used to improve the communication within the design team, establishing a common reference model that can be understood by all members of the team. It is hoped that functional- and job analysis directly supported by the visualization of planned control room modifications opens up for visual scenarios of work processes. Such scenarios can be used as a basis for walk-through and talk-through verification techniques helping the design team to understand the consequences of a given design.

Presentation 54

A Flexible Simulation and Design Assist Tool for the MAPLE Reactor

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Canada

Abstract

In recent years, high fidelity simulators that can be used for operator training, or to assist in plant design or as a tool for control strategy evaluation have become more practical and less expensive. This paper describes an AECL-developed software simulation package named MMIRSIM. MMIRSIM has the ability to simulate the dynamics of the MAPLE reactor, its coolant system, and its instrumentation and control, under a wide variety of conditions. MMIRSIM development used an approach that permits the integration of multi-paradigm models and heterogeneous simulations, and includes mutual design and validation interactions. One of the key benefits of MMIRSIM is that in addition to being an analytic simulation tool for model-based design purposes, the same tool can be used as a desktop simulator to support informal training or even as a commissioning aid to predict and/or explain expected or observed reactor start-up tests. The use of such a simulator results in earlier design problem identification and resolution, safer plant operations, minimization of spurious trips, and improved plant performance. MMIRSIM was built according to the exact plant design specifications and verified with actual commissioning data. It is used to apply a simulation-based approach for the verification, validation, and improvement of plant design, primarily with respect to the plant control and instrumentation, however, has also aided in diagnosing and resolving issues related to process functions, plant commissioning, and plant operating procedures.

Presentation 55

DCS Implementation challenges for Control Room Simulators

Robert Boire

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Abstract

The presentation will review current techniques for the implementation of a DCS on a full scope control room simulator (FSS) and the advantages and disadvantages of each approach from the perspective of the simulation vendor and the end user. The use of a simulator as a DCS V&V tool will be discussed. The presentation will also review MAPPS's experience on selected current projects as they relate to the implementation of DCS projects on new builds.

Presentation 56

Simulator Upgrades and New Simulators at the Bruce

Blaine Leslie

Bruce Power
Canada

Abstract

(Not available)

Presentation 57

Plant Specific Simulators – A Test Bed for NPP upgrades and modernizations

Burkhard Holl

Kraftwerks-Simulator-Gesellschaft
Germany

Abstract

(Not available)

Presentation 58

Licensing Processes for Safety-Related Instrumentation and Controls at United States Nuclear Power Plants

Richard T. Wood, David E. Holcomb, Kofi Korsah

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Abstract

This presentation summarizes the best practices employed within the U.S. nuclear power industry to implement and license safety-related digital I&C systems. This summary includes an overview of the current regulatory process, presentation of relevant requirements (e.g., hardware, software, system), discussion of the lifecycle approach, and identification of key quality assurance practices.

[*] The Oak Ridge National Laboratory (ORNL) is managed for the U.S. Department of Energy by UT-Battelle, LLC, under contract DE-AC-05-00OR22725.

Presentation 59

Resolution of Control Room and Human Factors Technical and Regulatory Issues for New Plants and for Modernization of Operating Plants

Joseph Naser

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Abstract

There are several technical and regulatory issues in the areas of instrumentation and control, human factors, and control rooms identified as needing generic resolution. If they are not generically resolved, they can contribute to protracted regulatory reviews for operating plants and substantial delays and increased costs for new plant COL approvals. Therefore a coordinated, proactive program is needed to resolve the key issues. Both Industry and the NRC have roles in resolving these key issues and addressing them in future design efforts and regulatory reviews.

The Industry initiative began with a workshop sponsored by EPRI and NEI on March 28-29, 2006, which led to the creation of the NEI Digital I&C and Human Factors Working Group. Three task forces and two focus groups have been established under the NEI working Group. The working group has identified issues, determined priorities, and established resolution plans from the industry perspective. EPRI is providing technical input and resolution leadership for some of the issues.

On November 8, 2006 there was a briefing to the NRC Commissioners on digital I&C by the NRC Staff and industry. After this briefing, the NRC established six task working groups to work on the resolution of NRC identified issues. These task working groups and the NEI task forces and focus groups are working together on the definition of common problems and problem statements, development of resolution plans, and identification of which organization has the lead on the resolution actions.

The NEI Human Factors Task Force and the NRC Highly Integrated Control Room – Human Factors Task Working Group have been working together in the areas of control rooms and human factors for new plants and operating plants. The issues that have been identified include: 1) minimum inventory for HSIs, 2) computer-based procedures, 3) graded approach to HFE activities, 4) crediting manual operator action, and 5) use of non-safety HSIs to control safety systems. For the last two, this Task Working Group is supporting two of the other Task Working Groups.

Presentation 60

Introduction to the IEC Nuclear family of standards

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Abstract

The International Electrotechnical Commission (IEC) has created a series of functional safety standards specifically for analog and digital instrumentation. The foundation standard is IEC 61508. From this basic safety standard, industry specific standards have been written including IEC 61513 for the Nuclear industries. The IEC 61508 standard will be explained along with the industry specific version IEC 61513. This presentation will discuss the principles, objectives and several years of experience in using the standards in several industries.

Presentation 61

Application of microprocessor based technology in CANDU Stations

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President & CEO
COG
Canada

Ujjal Mondal
Project Manager
COG
Canada

Abstract

In early 80s, absence of more efficient, cost effective, reliable control hardware was experienced by the nuclear industry. Usage of microprocessor based technology was not prevalent in the nuclear industry due to lack of experience and infancy of the new technology at that time.

In 1985, microprocessor based technology, "Chameleon micro DCI Controllers" manufactured by Fischer & Porter was the 1st application of such technology in any safety related applications in a CANDU Station. The application was for In-Core LOCA logic in Emergency Coolant Injection System (ECIS) modification in Pickering A Nuclear Station. This was followed by use of the same hardware for modification of "Dump-Arrest" logic for shut-down system in Pickering A in 1986. Other applications were "Heat Transport Pump Trip Logic" in Bruce A in 1989. The hardware performed very well and far exceeded its predicted availability during the last 22 years of operation.

In early 90's Power House Emergency Venting system used 22 micro DCI controllers for Pickering A and B Stations. The use of microprocessor based technology helped meeting the timing requirements of initiating Powerhouse Emergency Venting within 3 seconds of a steam break accident in reactor auxiliary bay.

Development of digital trip meter was undertaken in 1991 to solve human-factors related issue of interpretation of HT system saturation trip margin as the analog indicator lacked accuracy and resolution. This was a significant milestone in use of digital technology in safety related applications in any CANDU Station. The process offered an opportunity of trial use of software standards developed by OPG and AECL for category 1 safety critical software systems. This application generated significant learning experience for OPG (then Ontario Hydro) and the industry.

OPG was ahead of other nuclear utilities in implementation of microprocessor based technology in early 80s and 90s. OPG initiatives generated lot of design, maintenance and operating experience for application of software based technology in the nuclear industry. The paper will discuss the process followed, regulatory implications and lessons learned.

Presentation 62

Digital Communication Assessment for Highly Integrated Control Rooms

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Abstract

Oak Ridge National Laboratory (ORNL) is conducting research for the U.S. Nuclear Regulatory Commission regarding high integrity digital communication to support evaluation of highly integrated control rooms (HICRs). The principal features of the HICR are extensive use of digital network communications and compact operations consoles. These features provide operations flexibility and potentially increase operations and maintenance efficiency. However, new failure modes are possible depending on design methodology that must be considered.

The research approach into high-integrity digital communication focuses on determination of relevant operating experience and lessons learned, identification of accepted consensus practices, and analysis of possible failure mechanisms. The first research element draws principally from international nuclear power plants experience. Industry standards from organizations including Institute of Electronic and Electrical Engineers (IEEE) and International Electrotechnical Commission (IEC) are considered in the second research element to determine accepted consensus practices. For the third research element, digital network communication failures are studied. In particular, architectures that apply to nuclear safety systems are examined and a taxonomy of error types, message types, and failure mechanisms is established.

This presentation includes findings from each of the research elements of this investigation.

[*] The Oak Ridge National Laboratory (ORNL) is managed for the U.S. Department of Energy by UT-Battelle, LLC, under contract DE-AC-05-00OR22725. This work is sponsored by the U.S. Nuclear Regulatory Commission (NRC) Office of Nuclear Regulatory Research. Opinions and conclusions expressed by the author do not necessarily represent positions endorsed by NRC.

Presentation 63

Safety State Monitoring: Past, Present and Future

Richard Brown

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Canada

Abstract

Many of the control rooms in operation today were designed prior to the accident at Three Mile Island (TMI) – a time when human factors and operation during and following accidents were rarely formally considered in design.

Standards and guidance have developed significantly since that time and as the standards and guidance evolved so too did the safety state monitoring of control rooms. Modifications were made and new technologies adopted to address emerging concerns and new requirements. The result for many control rooms after all these years is a patchwork of changes, each individually representing an improvement but often lacking the cohesion expected of an well-integrated overall design.

With the large potential for new builds, and a growing maturity of digital technology, designers now have an opportunity to take a step back, consider the needs as a whole, and develop more optimal solutions.

This presentation will briefly review the evolution of safety state monitoring in general from a Canadian / CANDU®* perspective and will provide an overview of its implementation in the Advanced CANDU Reactor®* (ACR®*).

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* CANDU®, Advanced CANDU Reactor® and ACR® are registered trademarks of Atomic Energy of Canada Limited (AECL).

Presentation 64

The Plant Display and Control System Support Environment for the MAPLE Reactors

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Abstract

AECL is commissioning two new MAPLE reactors (MAPLE 1 and 2) and a processing facility at AECL's Chalk River Laboratories site, which will be the world's first nuclear reactors dedicated exclusively to medical isotope production. The MAPLE reactors are controlled and monitored by the Reactor Control Computer System (RCCS). RCCS consists of four major components: a dual redundant digital plant control system (PCS), a non-redundant monitoring system, a redundant plant display system (PDS) and a maintenance system. To support RCCS hardware/software maintenance activities and upgrades, AECL has developed the MAPLE Maintenance Test Facility (MTF). The MTF is comprised of two functional parts: 1) RCCS Testbed (duplication of RCCS including the operator HMI); 2) A real-time Stimulator to provide simulated plant inputs to RCCS Testbed. This presentation will describe AECL's approach to manage software/hardware upgrades, technology obsolescence and plant operational changes with MTF. Benefits of the approach taken as well as lessons learned from the MTF experience will be discussed.