ELECTRIC POWER RESEARCH INSTITUTE

Review of US and Japanese Methods to Reduce Personnel Dose and Control Radiation Fields

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Review of US and Japanese Methods to Reduce Personnel Dose and Control Radiation Fields

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REPORT SUMMARY

This report summarizes aspects of U.S. nuclear industry's performance in several areas related to radiation exposure management. The document supports efforts of the Japan Nuclear Energy Safety Organization (JNES) to find ways to reduce radiation exposure in Japanese plants.

Background

For over 20 years, the EPRI Radiation Exposure Management Program has been the performing research and development in the United States on controlling worker radiation exposures. This program develops advanced technology for radiation protection and radiation field control and produces application and good practice guidelines that are used throughout the industry. The program also provides expert guidance on a plant-specific basis in supplemental-funded projects.

JNES is evaluating operating programs in the United States and elsewhere to identify opportunities for exposure reduction that can be implemented without compromising Japan's standards for reactor safety. This document comprises a summary of several reports over a three-year period that highlighted U.S. radiation protection programs. EPRI has supported JNES by identifying the fundamental differences between U.S. and Japan's occupational exposure trends and by providing U.S. experience in several key areas. This work has been focused on regulatory impact and technology deployment.

Objectives

The primary objective of this work is to identify the highlights of U.S. industry performance in the following areas related to radiation exposure management:

- Task and Area Based Dose Evaluation
- The Impacts of Regulation on Dose
- Remote Monitoring Technology
- Source Term Reduction
- Groundwater Contamination
- Main Control Room Habitability

Approach

The project team analyzed industry experience data for its applicability to the project. This process involved conducting several site visits and capturing data using industry surveys and personal contact.

Results

The report is divided into six sections:

- The Task and Area Based Dose Evaluation section of the report summarizes monitoring that shows that scaffolding, refueling, and other routine maintenance activities were the largest contributors to dose for both PWR and BWR plants.
- The Impacts of Regulation on Dose section highlights the dose acquired during inspections and alternative applications performed by plants to reduce dose.
- The Remote Monitoring Technology (RMT) section provides details about the remote monitoring programs used in Vogtle, Arkansas Nuclear One and also provides input about how Duke Energy and Electricité de France (EDF) apply RMT. The results show that RMT is not only technology dependent, but requires the active sponsorship of senior management and inclusion of the technology into the station ALARA program.
- The Source Term Reduction section compares the relative performance of chemical and volume control systems (CVCSs) with reactor water cleanup (RWCU) systems for activity removal and discusses the impacts of cleanup systems on overall source term.
- The Groundwater Contamination section reviews the potential costs of groundwater contamination, and reviews requirements to ensure minimal propagation of the plume.
- The Control Room Habitability section discusses post accident scenarios. The section reviews NRC regulatory requirements including associated testing methods.

EPRI Perspective

EPRI seeks to communicate the current state of technology to its customers. Exposure reduction has been accomplished in the United States through an integrated development of source term reduction technologies and improved ALARA performance. The successful elements of the United States nuclear industry with regards to regulation, technology implementation, and ALARA program development are applicable to the entire international nuclear community.

Keywords

ALARA
Radiation exposure control
Source term reduction
Chemistry

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1 TASK AND AREA BASED DOSE EVALUATIONS

A comparison of task-based cumulative exposure was a key concern in the collaborative project between JNES and EPRI. This section outlines the task based dose obtained from US utilities.

Approach

Using data from five PWR plants and five BWR plants, the 20 highest exposure tasks (CRD replacement, RPV head replacement) were defined. These data were plotted in a summary histogram where each graph bar displays the results by task for the primary task. Where data were available, the total dose was reported by subtasks. For example, for a valve repair, the following tasks are to be considered:

- Scaffolding
- Disassembly
- Repair
- Inspect
- Reassemble
- Test
- Cleanup

Similarly, using data from five PWR plants and five BWR plants, the 20 highest dose rate areas (e.g., steam generator channel head, cleanup room, pipe penetration area, etc.) were identified and that data was plotted in a dose rate graph.

Data Sources

Data were collected from the stations in Table 1-1.

Table 1-1
Plants surveyed for Phase II study

Content deleted - EPRI Proprietary Information.

Results

All data were collated in one spreadsheet. Individual worksheets were developed for BWR Task Dose Data, PWR Task Dose Data, and one for Area Dose Rates (combined BWR and PWR). Additionally several graphs were plotted using the collective Task data.

Task Data

Figure 1-1 and Figure 1-2 were generated using data for BWR Combined Minimum – Maximum – Average and PWR Combined Minimum – Maximum – Average respectively. Additional graphs were plotted and are contained in the attached Excel file. This includes a BWR Combined Average. For tasks with data for only one or two plants the calculation used those one or two values. Therefore if this task was performed, this would be the average expected exposure. Single plant graphs were generated for several stations for BWR Task Dose Data. Those graphs appeared to be of limited value so this was not performed for all stations.

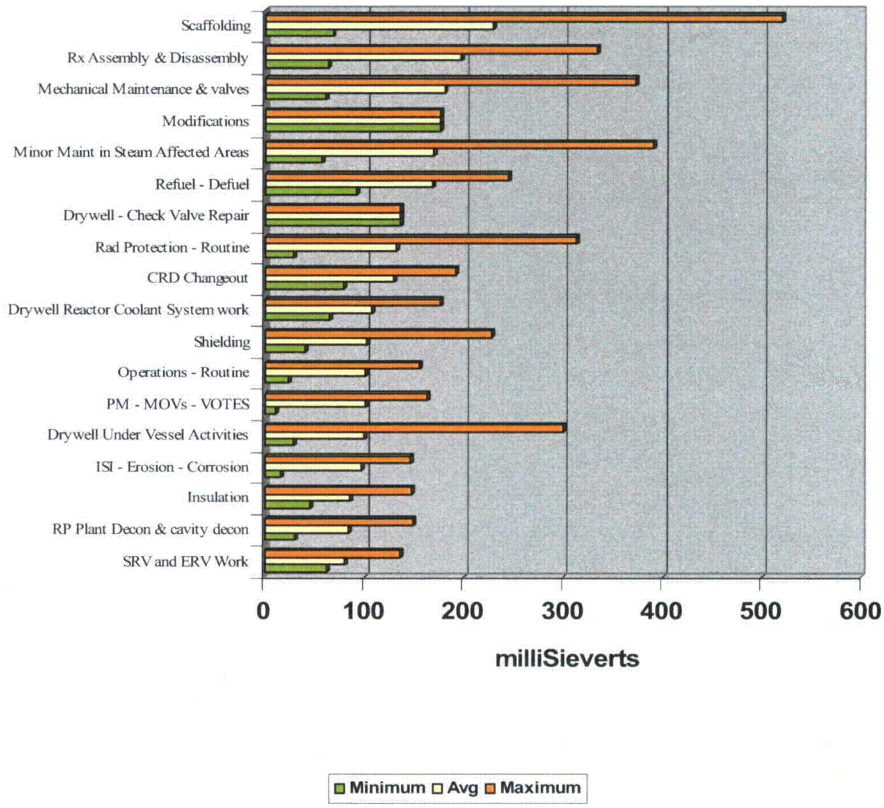

Figure 1-1
BWR Combined Minimum – Maximum - Average Graph

For BWRs the three most significant tasks are:

1. Scaffolding

   This includes labor-intensive staging, scaffold installation and inspection, removal, decontamination and storage. This is of particular concern for work in the drywell area where the physical equipment configuration requires work in close proximity to radioactive components and piping.

   The results are consistent with historical industry data. Currently, EPRI conducts an annual industry Vertical Access/Scaffolding workshop to share industry lessons learned, review alternate technologies, and to identify areas for improvement.

2. Minor maintenance in steam affected spaces

   This typically involves leak repair at power in locations with high general area dose rates that are dependent on reactor power level. Frequently reactor power reduction cost-benefit scenarios are evaluated to reduce the work area dose rates. For plants employing hydrogen water chemistry regimes, this may also include an evaluation of a hydrogen injection "dial down" to reduce their significant gamma contributions to those areas.

3. Valve and equipment maintenance

BWRs have historically been challenged by equipment reliability issues. Technology and maintenance practices improvements have resulted in reduced maintenance requirements. However, because of the sheer volume of BWR radiologically controlled areas, much of the work performed results in exposure accumulation.

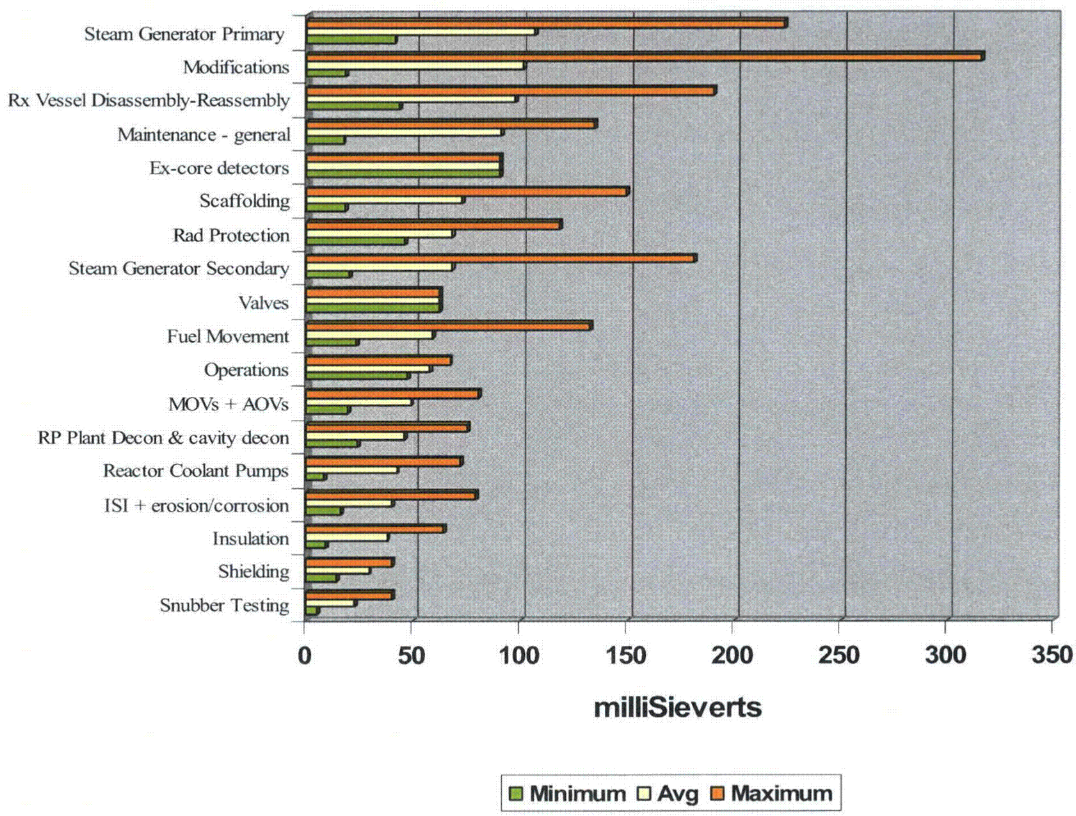

Figure 1-2
PWR Combined Minimum – Maximum - Average Graph

For PWRs the three most significant tasks are:

1. Scaffolding

   Similar to BWRs, this includes labor intensive staging, scaffold installation and inspection, removal, decontamination and storage. Many of the tasks inside the containment biological shield walls require work in the overhead. This involves construction of numerous large towers that may be 15 – 18 meters in height. Much of that work is adjacent to reactor coolant piping, steam generators, reactor coolant pumps and associated valving.

2. Modifications

   Historically modifications were related to reactor safety initiatives driven by the Three Mile Island incident. Currently they are typically required as a result of industry events (material or equipment failure) or to satisfy license extension commitments. Many modifications requires work in close proximity to radioactive components and piping for cutting, welding or otherwise opening and closing systems in addition to pre and post modification inspections, engineering and testing evolutions.

3. Refueling

   This is a large task and includes reactor disassembly, fuel movements, testing, new fuel loading, reactor reassembly and in some instance post refueling cavity decontamination. Most of the exposure is due to work is in close proximity to the reactor head and partially exposed highly radioactive components (e.g., reactor head on the head stand, setting the reactor head, etc.). The second most significant dose is attributable to fuel movement. Work over the reactor cavity and spent fuel pool is completed in relatively low dose rate areas, but its execution requires multiple persons for a significant period of time.

Area Dose Rates

The industry data were evaluated for plotting to assist with identification of trends or anomalies. However, the industry uses a very broad spectrum of area identification labels that are reactor vendor, site, and/or reactor specific. Therefore it was not feasible to accurately align or compare individual site data for the majority of areas. Background and summary PWR data captured via EPRI's recently re-instated Standard Radiation Monitoring Program (SRMP) was evaluated. These data clearly illustrates the source term (hence, the dose rate) variability between reactors.

The BWR Radiation Assessment and Control (BRAC) program yields similar results with dramatic variations in source term impacted by factors including chemistry, purification system efficiency and materials of construction. The BWR data, captured in the accompanying data file, show that areas containing or adjacent to reactor water cleanup systems (piping, tanks, phase separators, etc.) and the reactor drywell and vessel (bottom drains, nozzles, etc.) have the highest general area dose rates. The drywell is heavily accessed during outages. The phase separator rooms are not routinely accessed.

For PWRs, heat exchangers (regenerative and decay heat removal), the containment piping penetration area and loop piping and bypass valve areas have the highest dose rates. The pipe penetration and loop piping areas are frequently accessed for operation, maintenance and testing activities. Heat exchanger enclosures do not typically require routine personnel entry or maintenance evolutions.

Summary

In summary, it appears that the task dose related to scaffold and "routine" equipment maintenance are the most frequent high dose tasks. These challenges are somewhat related in that reductions to maintenance will result in reductions in scaffolding requirements. Area dose rates vary dramatically by reactor type and location with very few consistent trends. Physical plant system configuration, materials of construction, and source term controls all play a significant role in defining these data.

2
IMPACT OF REGULATION ON DOSE REDUCTION

There are many differences between the United States and Japanese regulations, and several of these have impacts on the cumulative exposure. This section describes several of the United States regulations that have impacts on the cumulative radiation exposure.

Background

The following aspects of regulation of U.S. utilities have or will have an effect on the radiation dose performance of the nuclear plants. However, this review of the impact of regulation will concentrate on items 4 and 5, Risk-Informed ISI and the Materials Reliability Program.

1. Deregulation of the electric utility industry
2. Significance determination process
3. Potential for reduction in dose limits
4. Risk-Based and Risk-Informed ISI
5. Alloy 600 Materials Reliability Program

Federal, state and local regulations impact nuclear power plant programs and processes in a variety of ways including resource allocation, operating boundaries, cost-efficiency, and exposure management. For example, integrity validation and control for Class 1, 2 and 3 systems and components is managed using American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI guidance. Historically, plants relied on very conservative reported design stress limitations that did not adequately address actual failure potential. As a result, the industry has adopted a new strategy that is believed to more accurately represent material condition and its related risk of failure.

In 1995, the Nuclear Regulatory Commission (NRC) published a policy statement on the use of probabilistic risk assessment (PRA) methods in nuclear regulatory activities. In the statement, the Commission stated its belief that the use of PRA technology in NRC regulatory activities should be increased to the extent supported by the state of the art in PRA methods and data and in a manner that complements the NRC's deterministic approach. To implement this policy, the staff developed a PRA implementation plan, together with a timetable for developing regulatory guides (RGs) and standard review plans (SRPs), which describes an acceptable approach for assessing the nature and impact of changes by considering engineering issues and applying risk insights. A part of this effort involves developing an RG and the corresponding SRP to provide guidance to power reactor licensees and the NRC staff on an acceptable approach for utilizing risk information to support requests for changes in the plant's ISI program for piping. The RG and the SRP were issued for trial use in October of 1998.

The nuclear industry has developed two methodologies for implementing RI-ISI. One methodology has been jointly developed by the American Society of Mechanical Engineers (ASME) and the Westinghouse Owners Group (WOG). The other methodology is being sponsored by the EPRI. In addition, ASME has developed three ASME Code cases for alternate examination requirements to ASME Section XI, Division 1, for piping welds. ASME Code Case N-577 is based on the WOG methodology and ASME Code Case N-578 is based on the EPRI methodology. ASME Code Case N-560 is based on the EPRI methodology for Class 1 B-J welds and is being revised to encompass both methodologies.

NRC PRA policy statement encourages great use of PRA to improve safety decision making and improve regulatory efficiency. Nuclear Regulatory Commission has developed and published several Regulatory Guides and Standard Review Plan sections which provide guidance on use of PRA findings and risk insights in support of licensee requirements for changing plant licensing basis.

Regulatory Guides:

- 1.174 An Approach for using Probabilistic Risk Assessment in Risk- Informed Decisions On Plant-Specific Changes to the Licensing Basis
- 1.175 An Approach for Plant-Specific, Risk informed Decision-making: Inservice Testing
- 1.176 An Approach for the Plant Specific, Risk-Informed Decision-making: Graded Q
- 1.177 An Approach for the Plant-Specific, RI-Decision Making: Technical Specifications
- 1.178 An Approach for the Plant-Specific, Risk-Informed Decision-making: ISI of Piping
- 1.182 Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants

Standard Review Plans

- Chapter 19 Use of PRA in Plant-Specific, Risk-Informed Decision-making: General Guidance
- Chapter 3.9.7. Risk Informed In-service Testing
- Chapter 16.1. Risk-Informed Decision-making: Technical Specifications
- Chapter 3.9.8. Trial Use for the Review of Risk-Informed In-service Inspection Piping

Recent activities on Risk-Informed In-service Inspection include the extension of RI-ISI methodology to Break Exclusion Zone Region Piping (BER) through the 10 CFR 50.59.

Since its USNRC acceptance in 1999, the EPRI Risk-Informed Inservice Inspection (RI-ISI or RISI) methodology has been used by many nuclear plants to categorize and select piping components for inspection based on risk significance. The EPRI RI-ISI methodology serves as an alternative to the in-service inspection requirements for piping cited in the ASME Code.

As illustrated in Table 2-1 below, the EPRI technical approach involves assigning segments of piping into risk categories based on the probability of a large break occurring in the piping and the consequences of a break in that segment. Probability of a break is based on an assessment of the susceptibility of the piping to the degradation mechanisms known to affect such piping. The EPRI methodology provides quantitative screening criteria for determining the susceptibility of the piping to each mechanism based on the specific materials of construction, operating conditions and piping geometry. The consequence assessment utilizes conventional failure modes and effects criticality analysis (FMECA) techniques used in plant Probabilistic Risk Assessments, which are simplified for this application in the EPRI methodology. Once classified in this manner, piping elements (typically welds) are selected for inspection from the highest risk segments.

Table 2-1
EPRI Piping ISI Risk Categories and Consequence Table

<table>
<tr><th colspan="2">Risk Categories</th><th colspan="4">Consequence Category</th></tr>
<tr><td colspan="2">High - Cat 1, 2, & 3</td><td rowspan="3">None</td><td rowspan="3">Low</td><td rowspan="3">Medium</td><td rowspan="3">High</td></tr>
<tr><td colspan="2">Medium - Cat 4 & 5</td></tr>
<tr><td colspan="2">Low - Cat 6 & 7</td></tr>
<tr><td rowspan="3">Degradation Mechanism Category</td><td>Large Break</td><td>Category 7</td><td>Category 5</td><td>Category 3</td><td>Category 1</td></tr>
<tr><td>Medium Break</td><td>Category 7</td><td>Category 6</td><td>Category 5</td><td>Category 2</td></tr>
<tr><td>Small Break</td><td>Category 7</td><td>Category 7</td><td>Category 6</td><td>Category 4</td></tr>
</table>

The overriding goal of this concept is to reduce the plant inspection scope while maintaining an effective pipe/component integrity management program. Related benefits include reduced regulatory burden, improved plant safety and reduced personnel exposure.

In direct contrast to that opportunity for exposure reduction, other regulatory required inspections, regardless of their basis, can have a dramatic impact on personnel exposure. One of the most recent examples is related to Alloy 600. Several weld and piping failures have been identified and the resultant inspection requirements challenge even the most seasoned ALARA professionals. A significant portion of the inspection and/or mitigation activities require extended periods of work in relatively high radiation fields. These areas include, but are not limited, to the pressurizer spray and surge lines, pressurizer heater penetrations, reactor piping safe ends, and under and on top of the reactor head.

Approach

The EPRI team requested information from over 20 reactor sites. The intent was to obtain data that qualified and quantified to the extent possible, the impact of regulation on exposure reduction programs. The information gathering process included discussion of applied risk informed regulation such as RI-ISI. A list of mandatory and non-mandatory inspections by components and systems was requested as was data related to reductions to radiation area work, and the associated labor and exposure reductions.

ISI Programs and Procedures

Two documents were reviewed that were representative of the types of programmatic documents used to implement the in-service inspection program, including risk-informed ISI. The first document was a "Nuclear Plant Risk Informed ISI Program", and it specifically addresses methods to utilize the risk-informed methodology. The second document was a corporate level program document that sets general standards for performing ISI, including the requirement to perform testing with low radiation exposure.

Results

ISI on reactor vessel welds, nozzle welds, and reactor coolant system piping inspections are tasks that result in dose to non-destructive examination (NDE) inspectors, as well as scaffold builders (often carpenters), insulators, health physics technicians, and decontamination workers. The majority of these are not automatic inspection devices. However, several advances in technology have resulted in development of semi-remote devices; they require a reduced amount of hands-on time in the relatively high radiation field.

In all plant correspondence it was clear that the station staff knew that RI-ISI resulted in tangible exposure reduction. Additionally, preliminary EPRI studies indicate that plants can reduce their ISI requirements by 60% to 80%. Those results are in line with related NRC and Westinghouse Owners Group studies.

Site specific ISI data were very difficult to obtain. The majority of stations polled do not track exposure savings relative to regulatory requirements. For example, ISI exposure is tracked by some stations as a segregated task, but the number and type of sub-tasks contributing to that "segregated" exposure reduce the accuracy of benchmarking activities to unacceptable levels. One plant may include all aspects of a regulatory required inspection on a single RWP including scaffolding management, radiation protection coverage, support services, and insulation removal and replacement; another station may only include the exposure for those persons performing the ISI activity. Similarly, the software, databases, and radiation work permit (RWP) strategies employed at plants change frequently; this further complicates the ability to compare pre-RISI to post RISI performance. Two stations provided data for comparison, but early in the review process it was determined that the data was not sufficient or accurate enough for inclusion in this analysis.

Data related to other regulatory relief actions and materials reliability programs (MRP) were readily available.

The data in this section were derived from station interviews, data request responses and published reports and presentations.

Alternate Test Methods

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ISI

Content deleted - EPRI Proprietary Information.

Table 2-4
ISI Exposure and Resource Loading Estimate versus Actual

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Figure 2-1
ISI Exposure Trend

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Table 2-5
Inspection Exposure Detail in person-Rem

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Alloy 600 Materials Reliability Program

Alloy 600 degradation is a one of the most significant regulatory compliance and exposure management issues currently challenging U.S. PWRs (and to a much lesser extent, and more historically, BWRs). This nickel base alloy containing 14-17% Cr, 6-10% Fe and various minor elements has recently been found to be degrading at a rate that is unacceptable to the industry and regulators. Alloy 600 was originally selected for SG tubes for its resistance to transgranular Stress Cracking Corrosion induced by chlorides on the secondary side. It has a similar coefficient of thermal expansion with low alloy steel used in construction of reactor pressure vessels, pressurizers and steam generator shells

Causal Factor(s)

Primary/pure water stress corrosion cracking (PWSCC) is the main failure mechanism; it is highly dependent upon:

- Operating temperature
- Material heat treatment (i.e. microstructure)
- Cold working
- Aggressive environment

It is caused by critical combination of material and/or environment and/or stress; the stress is often residual from fabrication and/or installation. The incubation times for crack initiation can be very long, but are reduced by cold work. The industry had many reactor years of operation before the first sign of PWSCC. The exact causal mechanism(s) is still unclear after over 30 years of research. Theories include anodic dissolution, hydrogen embrittlement, environmental modification of mechanical properties at grain boundaries, and internal oxidation, etc. This lack of fundamental knowledge hampers the success of the industry mitigation effort.

Locations

The issue applies to all Alloy 600 base material (with the exception of SG tubing and SG divider plates) and Alloy 82/182 weld metal locations in PWR primary systems. There are several common locations where this alloy was used in Westinghouse, Combustion Engineering (CE) and Babcock and Wilcox (B&W) reactors.

Common

- Large Diameter (≥4") Reactor Vessel Head Nozzles
  - Top Head CRDM /ICI
- Small Diameter Nozzles (<4")
  - Reactor Vessel Head Leak Monitor Tubes
  - Reactor Vessel Bottom Head Instrument
  - Steam Generator Bowl Drain
  - Pressurizer Steam Space Instrument
  - Pressurizer Liquid Space Instrument
  - Hot Leg Instrument
  - Cold Leg Instrument
- Pressurizers
  - Surge Line Nozzles
  - Spray Nozzles
  - Safety & Relief Valve Nozzles
- Non Pressure Boundary Locations
  - Steam Generator Tube Sheet Cladding
  - Core Support Blocks/Alignment Lugs
- Pressurizer Heater Sleeves/Bundles
- Reactor Vessels
  - CRDM Motor Housing

- Main Coolant Piping Loop
  - RCP Suction & Discharge Nozzles
- Branch Line Connections
  - Pipe-to-Surge Nozzle Connection
  - Charging Inlet Nozzles
  - Safety Injection and SDC Inlet
  - Shutdown Cooling Outlet Nozzle
  - Spray Nozzles
  - Let-Down and Drain Nozzles
- Non Pressure Boundary
  - Steam Generator Nozzle Dam Ring

There are also locations specific to reactor supplier design.

Westinghouse and CE Designs

- Small Diameter Nozzles (<4")
  - Reactor Vessel Top Head Vent
  - SG Divider Plate Welds

Westinghouse Design

- Reactor Vessel
  - Inlet and Outlet Nozzles
- Main Coolant Piping Loop
  - SG Inlet and Outlet Nozzles

Babcock & Wilcox Design

- Reactor Vessel
  - Core Flood Tank Nozzles
- Small Diameter Nozzles (<4")
- Reactor Vessel Top Head Thermocouple
  - Flow Element
  - Flow Meter

History

Figure 2-2 plots the history of Alloy 600 in a time versus cumulative degradation starting with BWR failures early in the 1970s followed closely by PWR steam generators. This data is significant from both a mitigation and exposure control standpoint. Most of the inspection and mitigation tasks involve work in radiologically controlled areas. This work is typically in some of the highest dose rate accessible areas of the plant including BWR risers, steam generator primary channel heads, under the reactor pressure vessel head and large bore pressurizer surge and spray line piping.

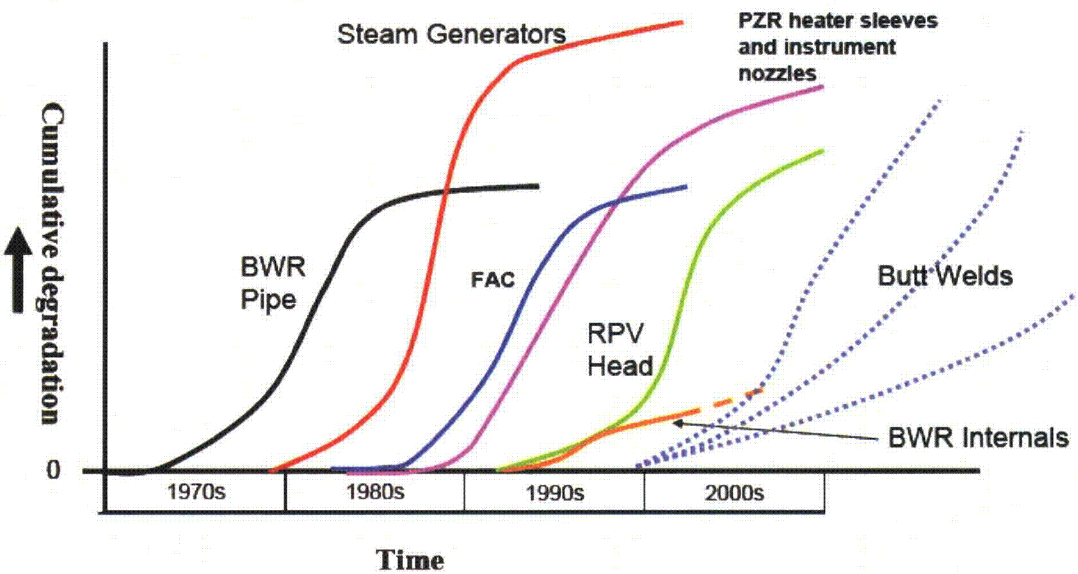

Figure 2-2
Alloy 600 Impact: Degradation vs. Time

The BWR piping history is directly applicable to PWR 82/182 Welds. The failures were preceded by other components cracking through same mechanism. Specific cracks of concern were found at two BWRs and at the VC Summer and Tsuruga PWRs.

The historically acceptable inspection technology, procedures, and personnel training were inadequate to find SCC. The processes were also similar in that the repairs were costly, especially when required as emergent work. As a result of the significance of this issue, the NRC has issued related inspection requirements in the form of:

- GL 88-01 and a Technical Specification change committing to it
- NRC head inspection order –revised individual plant licenses

General industry guidance has been developed for all locations. The guidance includes inspection/mitigation of:

- Reactor Pressure Vessel Top Head Nozzles

- Pressurizer Heater Sleeves
- Alloy 82/182 Butt Welds
- Reactor Pressure Vessel Bottom Mounted Nozzle
- Steam Generator Bowl Drains
- Other Location

MRP-126

MRP-126 is a mandatory Alloy 600 program requiring each plant to develop and document an Alloy 600 management plan. The plan needs to define the processes it intends to use to maintain the integrity and operability of each Alloy 600/82/182 component for the remaining life of the plant. Implementation of the plan was required for all U.S. PWRs by June 2006. The document "MRP Letter 2003-039, January 20, 2004" outlines the general requirements:

- Insulation removal and inspection of all Alloy 600/82/182 pressure boundary components >350°F
- Once within next 2 RFOs
- Priority to highest temperature (PZR and Hot Leg)
- ASME CC N-722 (Pending approval)
- Insulation removal and BMV inspection of all Alloy 600/82/182 pressure boundary components
- Prioritize on temperature and operating experience
- Inspections are split over interval (some inspections every period)

Reactor Pressure Vessel Top Head Nozzles: Top Head CRDM / ICI

The inspections related to this issue are currently being conducted to meet the NRC First Revision Order EA 03-009. This work involves RPV head penetration safety assessment using "Reactor Vessel Closure Head Penetration Safety Assessment for U.S. PWR Plants (MRP-110), 1009807".

The current guidance requires an NDE re-inspection frequency of every 10 calendar years for "resistant" materials beyond the first 10 years and specifies the NDE technique and personnel qualifications.

Reactor Pressure Vessel Bottom Mounted Nozzles

The B&W Owners Group, MRP and Westinghouse Owners Group are completing Safety Assessment calculations using a similar approach to that used for RPV Head Nozzles.

Reactor Pressure Vessel Bottom Mounted Nozzles MRP 2003-17 Letter

This guidance requires that during the current or next refueling outage, a bare metal visual examination of any Alloy 600 nozzles penetrating the bottom head of the reactor vessel be performed. This was issued in 2003 and the requirement was that this be completed within two RFOs. For plants operating on an 18 Month Cycle this requires completion by the end of Spring 2006. For those stations on a 24 Month Cycle the completion date is the end of Spring 2007. All plants that have bottom mounted nozzles have completed this inspection and there were no indications of leakage.

Butt Weld Safety Assessment Conclusions

The industry concluded that butt welds posed no immediate safety concern. This was based on the very small number of leaks/cracks given large numbers of locations worldwide. It was concluded that axial cracking is much more likely than circumferential cracking and probabilistic analyses showed that the impact of butt weld PWSCC is insignificant.

Current Required Inspections

The MRP related welds are within the ISI population and are inspected per ASME Section XI. This involves visual inspections for leakage and boric acid, NDE inspections for >1"NPS, and volumetric NDE for sizes >4"NPS. Ultrasonic testing (UT) must meet ASME Section XI Appendix VIII requirements.

MRP Exposure Impact

The following data represent the impact that Alloy 600 inspection and mitigation efforts have on station exposure.

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Table 2-6
MRP Inspection Exposures

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Table 2-7
MRP Inspection Exposures

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Table 2-8
MRP Inspection Exposures

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Table 2-9
MRP Inspection Exposures

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The exposure associated with this effort is summarized in Table 2-10.

Table 2-10
MRP Inspection Exposures

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Table 2-11
MRP Inspection Exposures

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Table 2-12
MRP Inspection Exposures

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Table 2-13
MRP Inspection Exposures

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Table 2-14
Industry MRP Exposure Survey June 2006 – 1 of 2

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Table 2-15
Industry MRP Exposure Survey June 2006 – 2 of 2

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The data in Table 2-16 were derived using the industry experience through July 2006.

Table 2-16
Industry Exposure Impact Estimates

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Conclusions

US PWRs cumulative exposure has continued to trend downward with few exceptions. The industry has put in place a per-reactor annual exposure goal of 65 person-Rem (60 person-Rem beginning in 2010). The limited data that was available for publication clearly show that regulatory mandated inspections and testing can have a significant impact on exposure management programs. Expending 15 – 40 person-Rem for Alloy 600 weld inspection and overlay processes each cycle will have a significant negative impact on both the trend and industry goal. However, industry experience demonstrates that using qualified processes and risk based analyses, the ability to defer or reduce the scope or frequency of those efforts will result in direct reductions to personnel exposure.

References

1. *Risk-Informed Inservice Inspection Evaluation Procedure*. EPRI, Palo Alto, CA: 1996. TR-106706.

2. *Applications of Risk and Performance Technology, Volume 1, Application of the EPRI Risk-Informed Inservice Inspection (RI-ISI) Methodology to Break Exclusion Region (BER) Programs*. EPRI, Palo Alto, CA: 2002. 1006837.
3. ASME. Boiler and Pressure Vessel Code, Section XI.
4. Plant data.
5. ERPI/INPO 2006 Alloy 600 Workshop meeting minutes.

3
REMOTE MONITORING TECHNOLOGY

Background

Today's Nuclear Plant Radiation Protection Programs face numerous major challenges including:

- Shortened outage schedules with increasing outage work
- Reduction in contract personnel
- Increase in first time radiation workers
- Challenging oversight goals
- Aging workforce

As a result, radiation protection (RP) programs across the industry are undergoing a major redefinition. The old RP paradigm is being replaced with more streamlined organizational structures and work practices. The industry is incorporating Remote Monitoring Technology as a means of extending its resources and increasing its worker productivity. These major changes are being performed within RP organizations while maintaining worker protection and radioactive material control.

EPRI sees remote monitoring as one of the most significant RP development within the past decade. This technology has the potential to bring significant change to the present nuclear plant organizational structure and work procedures and practices, which translates into improved worker efficiency and lower personnel exposure. The integration of video, audio communication, equipment performance and teledosimetry into remote monitoring (RMT) packages continues to result in a major transformation in many nuclear plant programs, e.g., radiation protection, operations, maintenance, planning and scheduling.

Applications

The following is a partial list of RMT and supporting technologies and several of the specific applications. The technologies and applications represent those that currently exist and/or are being evaluated for use in nuclear plants:

- Infrastructure and software
- Central monitoring stations
- Personnel exposure and area dose rate monitoring

- Audio/visual communication
- RFID (radio frequency identification)
- Electronic surveys, Tablet personal computer (PC), PDA (personal digital assistants)
- Dimensional measuring
- GPS positioning
- Personnel vital statistics
- Equipment performance
  - Vibration
  - Temperature
  - Pressure/differential pressure

Expected Benefits

The following is a brief summary of documented and expected benefits associated with remote monitoring technology. They are derived directly from reactor experience and are grouped into six categories:

1. Exposure and radiological control performance
2. Worker efficiency
3. Work data and process quality
4. Resource (staffing) optimization
5. Equipment reliability
6. Cost reductionHighlights of each of these are discussed in more detail in the following subsections.

Plant Program Status and Deployment

This section provides a partial industry program status update; the industry's deployment of remote technologies is very dynamic, summarizing 100% of applications would be impractical. This section also addresses several generic considerations related to long and short term planning. As stated previously, the majority (most likely all) US reactors are currently employing RMT. The applications, coverage, and future plans vary significantly by site. The following summarizes information from several sources.

- EPRI has recognized the value of RMT for radiation protection purposes, and has simultaneously realized that the technology has direct cross-ties and can add value to other plant programs and processes. EPRI has begun an integration of their industry team's radiation protection efforts with those programs through industry meetings such as the EPRI Wireless Conference; this conference targets all plant wireless applications.

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- Many plants would like to use Electronic Dosimeters (ED) or RP controlled radiation monitors - not plant installed equipment, to decrease radiation survey frequencies. Several plants have incorporated this measure and have increased the time period between surveys if certain conditions are met, such as, no noted changes and no entries into these specific areas.

- All stations agreed that when developing a plan, it is critical to account for all potential end users and to evaluate, and accept or reject, a wide variety of remote technology options.

The balance of this section summarizes additional plant experience.

Infrastructure and Software

Technology Overview

RMT infrastructures vary dramatically by site. Ideally this would entail a combination of fiber optics, hard wired, and wireless technologies routed throughout the plant.

Fiber optic cabling provides a reliable, high-quality infrastructure on which to base the RMT system. Fiber optics has several advantages to conventional copper wire including:

- **Thinner** - Optical fibers can be drawn to smaller diameters than copper wire.
- **Less signal degradation** - The loss of signal in optical fiber is less than in copper wire.
- **Light signals** - Unlike electrical signals in copper wires, light signals from one fiber do not interfere with those of other fibers in the same cable. This means clearer audio conversations or video reception.
- **Digital signals** - Optical fibers are ideally suited for carrying digital information, which is especially useful in computer networks.
- **Non-flammable** - Because no electricity is passed through optical fibers, there is no fire hazard.
- **Lightweight** - An optical cable weighs less than a comparable copper wire cable. Fiber-optic cables take up less space in the ground.

Permanently installed fiber optic cabling results in no additional set up setup time of an RMT system during an outage and frequently allows for use of the system during operations as well as outage. Using permanent fiber optic penetrations for areas where the cable must be removed during plant operations results in reduced setup time of an RMT system for use during an outage.

The infrastructure should also include at least one permanently connected containment/drywell penetration, and dedicated radio frequencies.

Finally, the infrastructure also includes both permanently mounted and junction boxes for removable hardware devices that are specific to their intended use (e.g., cameras, radiation monitors, temperature detectors, repeater stations), and data capture and retrieval hardware and software (e.g., PCs, database(s), LAN connections, archives, etc.).

The RMT software package should provide a powerful interface that supports viewing, evaluation and capture of incoming information. The software should be written to support integration with standard network configurations and protocol, taking into account projected changes to programming and internet platform options. The RMT software should be linked to the plant's telemetry system to provide automated worker login and logout via EDs. The software should also facilitate viewing live data on the LAN. Alarms should be incorporated into the system to alert the operator/monitor to abnormal, undesirable conditions.

Finally, in some applications, a digital image of plant areas serves as the background over which real time data (e.g., temperature, radiation field) can be transposed. Data capture locations incorporated on the drawings allows evaluation using real-time data and generation of hardcopies following data validation.

Application

Historically in the US, infrastructures to support RMT have been very limited requiring significant upgrades to monitoring stations, fiber optic and other networking options, monitors (all types) and data capture. The following figure represents a very basic arrangement to illustrate the interaction complexities associated with RMT.

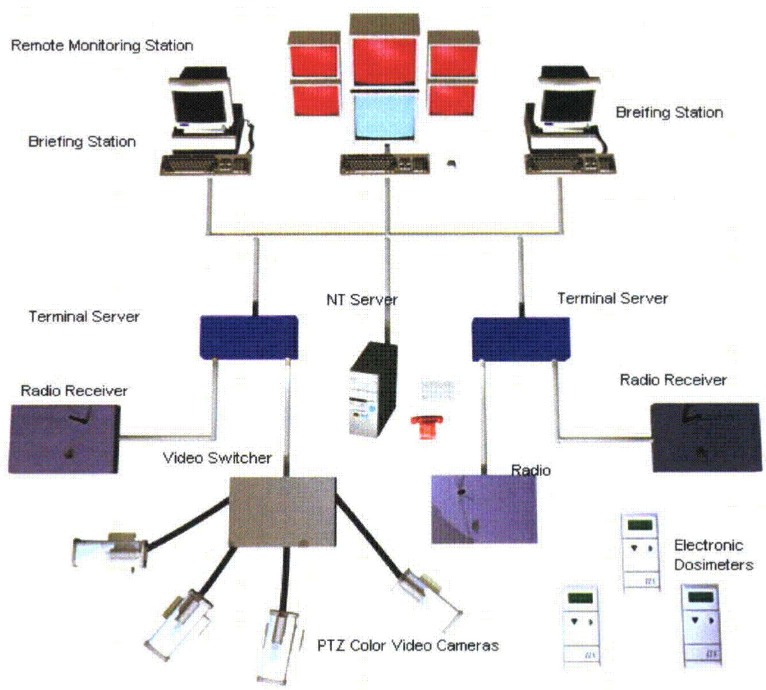

Figure 3-1
Simple Application Schematic

Industry Experience

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- During the initial installation of telemetry equipment or even during the process making the system more efficient, it is imperative to map the plant work areas to ensure proper reception from wireless transmission equipment.

Central Monitoring Stations

In the mid 1990's, remote monitoring moved from stand-alone projects to a centralized function with the creation of Central Remote Monitoring Stations.

Technology Overview

The design of central monitoring stations varies by plant, the location inside the plant, and the intended use. A typical master station will include 6 – 10 monitors arranged to support 3 – 4 persons monitoring those stations, camera/monitor switching stations, camera (pan and tilt, zoom, resolution) and audio controls, speakers, microphones, wireless headset transmitters, wireless headsets, data capture/ recording devices, alarm/annunciator panels, and a current time display. A typical outage installation will support 10 – 20 cameras, up to several hundred remote telemetry devices for area/component dose rate monitoring, and up to several thousand personal dosimetry devices. The following figure illustrates North Anna's central monitoring station.

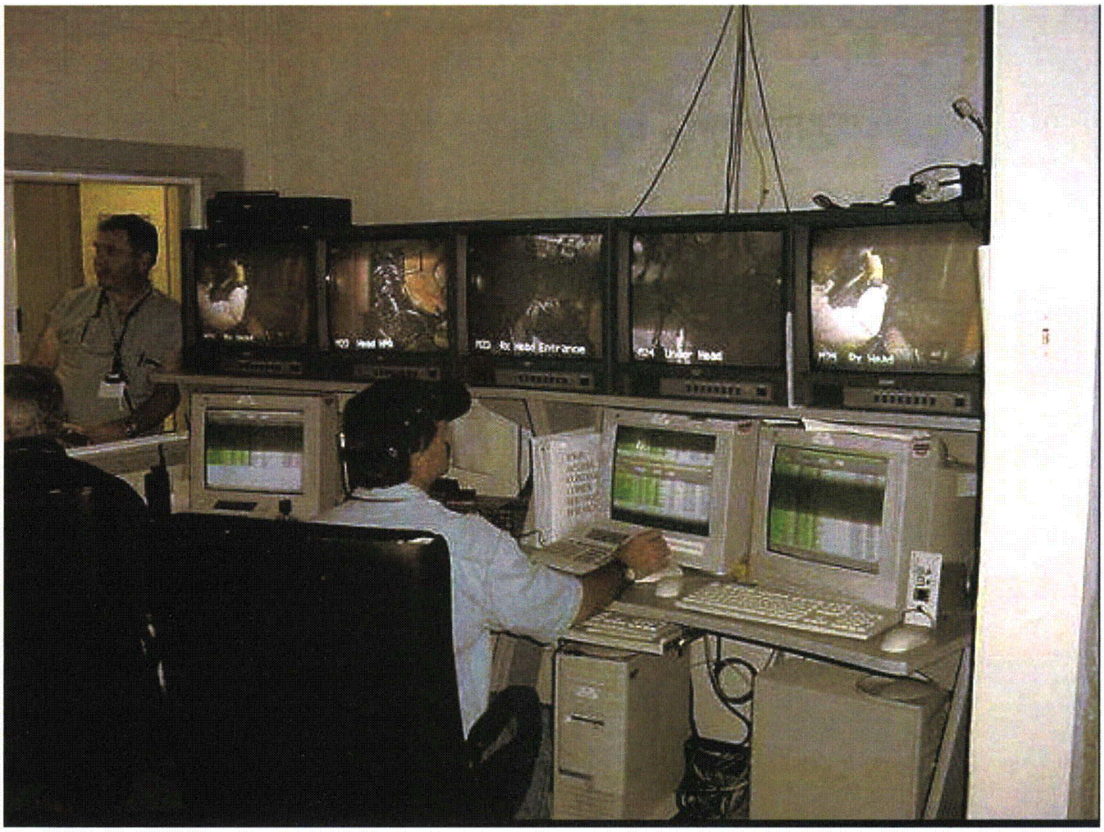

Figure 3-2
North Anna's Central Monitoring Station (Work Control Center)

Application

These facilities provide a "Control Room" for the explicit purpose of monitoring work performed by personnel at all key locations during normal operations and plant outages. Real time information transmitted back to the Control Room normally includes; video of the work location, communication with workers at the location, general field dose rates, air radioactivity concentration, worker exposure dose rate and accumulated dose.

Industry Experience

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Personnel Exposure and Area Dose Rate Monitoring

Technology Overview

Remote personnel exposure monitoring has been used worldwide for at least the past decade. The technology continues to advance with enhancements to audible alarms, control features, battery life and integrated options such as proximity sensors and log-in/log-out dialogue. Currently, the majority of systems employ a personal unit that contains exposure monitoring and radio frequency data transmitters. The cumulative dose and dose rate can be transmitted to remote displays and databases via a radio receiver. Typical limiting factors include radio frequency interference, the distance between monitoring device and receivers, and the number of units that can be used in parallel.

Application

Personnel RMT is typically used for high activity or radiologically significant work, multiple badging for complex source term geometries, and most recently as temporary area radiation monitors for monitoring exposure trends during shutdown crudburst cleanup or other transient source term, or significant radiation control evolutions.

Many plants have installed area monitors and continuous airborne radioactivity monitors and have incorporated this data into area radiological condition trending programs. Additionally, some stations are using tele-dosimeters at specific locations for similar purposes. The following figure illustrates one application of this technology to monitor primary system purification filter activity buildup.

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Figure 3-3
Primary Filter Dose Rate Monitoring Using Telemetry

The inclusion of dose rates in plant monitoring programs that provide indications such as valve position, pump status, temperature, flow readings, etc. is also being implemented.

An advantage of using telemetric dose rate instruments is that continuous, real-time dose rates are being provided rather than a single data point taken on a specified frequency (weekly, monthly, quarterly).

Industry Experience

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- RMT is used in many stations in conjunction with dose rate-to-activity analyses to monitor activity buildup in process components to ensure the resultant waste activity does not exceed a desired waste classification (US regulations: Class A, B, C or >C).
- Area dose rate data are routed to emergency response command and control centers, eliminating the need for deploying teams into potentially radiologically significant areas.
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Audio and Visual Communication

Technology Overview

Audio and visual communications represent the most commonly applied technologies for radiation protection programs. The majority of installations include multiple cameras located strategically throughout the plant. Most cameras have pan and tilt and zoom/focus capabilities; some require only a fixed mount. Monitors are typically high definition and several stations employ large screen plasma monitors for viewing critical or high activity areas (refuel floor, steam generator primary platform).

Audio systems typically incorporate the use of wired or wireless (preferred) headsets and for some applications, loud speakers at the job site. The following figure illustrates a basic audio and visual communication system.

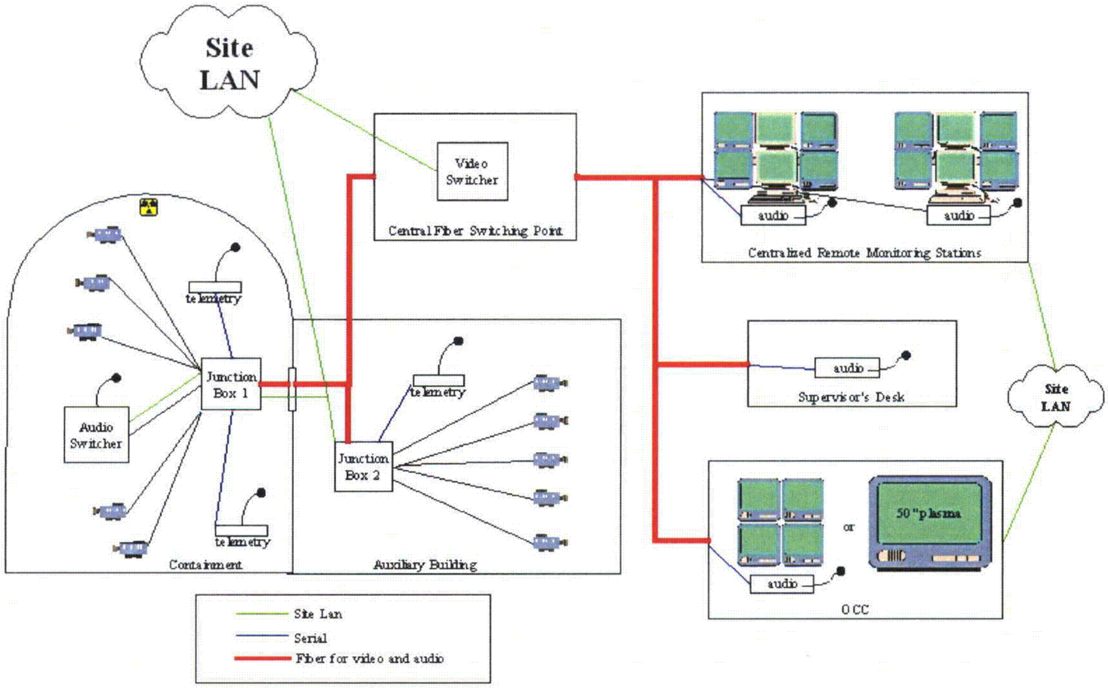

Figure 3-4
Audio/Visual Schematic

Application

Video cameras are an integral part of any remote monitoring system. They provide the necessary visual link between a worker and personnel monitoring an area, component or person in the remote location.

Many plants supplement their RMT system with a mobile cart equipped with full remote monitoring capability. These carts are available for unusual specific work locations to provide localized monitoring. The mobile cart supports area, component, or personnel monitoring from a remote location. The following figures represent a mobile application used by the Diablo Canyon Power Plant and a typical permanent camera application.

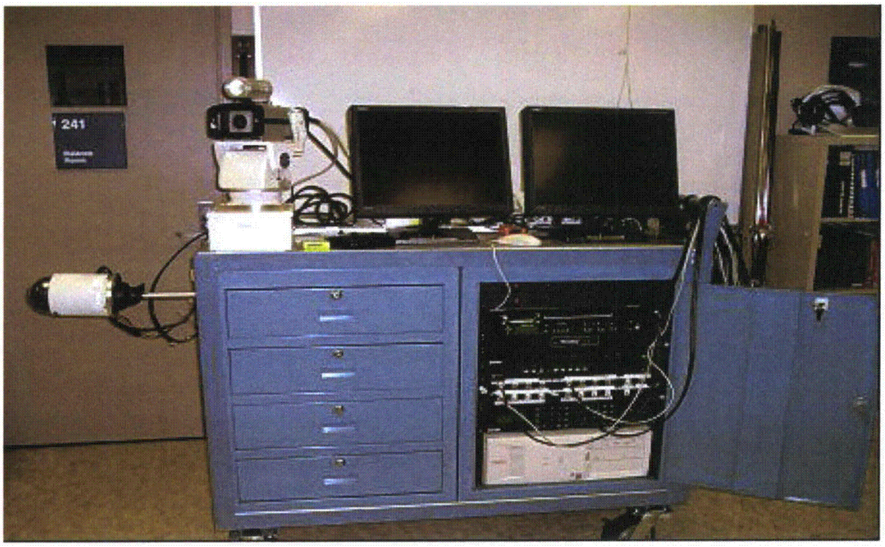

Figure 3-5
Mobile Cart Unit used for Localized Monitoring of Work

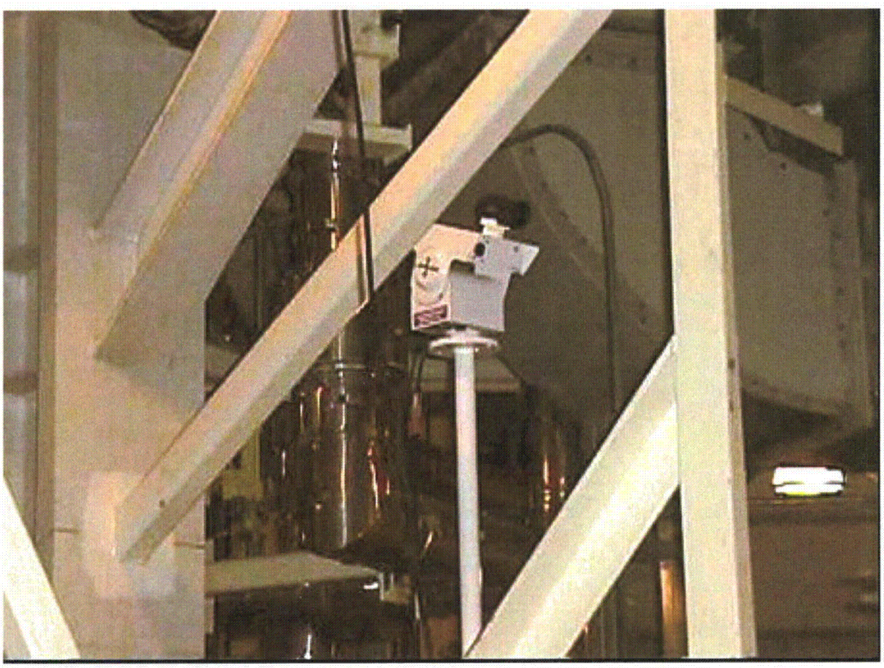

Figure 3-6
Typical Camera Mount

Industry Experience

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Other plants have done the following

- Fiber optic based visual systems on the refuel floor at Exelon were used to eliminate nine contractors from that area. This reduces the time on site, cost, work area congestion, and exposure. Activities are viewed from remote, off-site offices. This can also reduce the volume of contractor 'in-processing' during peak outage periods.
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- Most stations report that cameras and other remote monitoring equipment in containment and drywell areas are installed and removed each outage.
- RMT cameras are used at many stations to observe the media level in waste containers during fill operations--For example, filling spent resin high integrity containers with spent resin.
- Audio and visual information can be routed to emergency response command and control centers, eliminating the need for deploying teams into potentially radiologically significant areas.

Electronic Surveys via Tablet PCs and PDAs, and Status Boards

Technology Overview

Electronic survey technology is simply the use of tablet PCs or PDAs with either docking stations or internet or RF data transmission capabilities. There is nothing unique to a nuclear application with the exception of the software and database templates used for data input and retrieval. Status boards are electronic monitors (plasma, LED, LCD) that display information from a source PC. The following is one example of an electronic survey screen. Note the tabs at the bottom of the screen for dose rate, smear (contamination), and instrumentation.

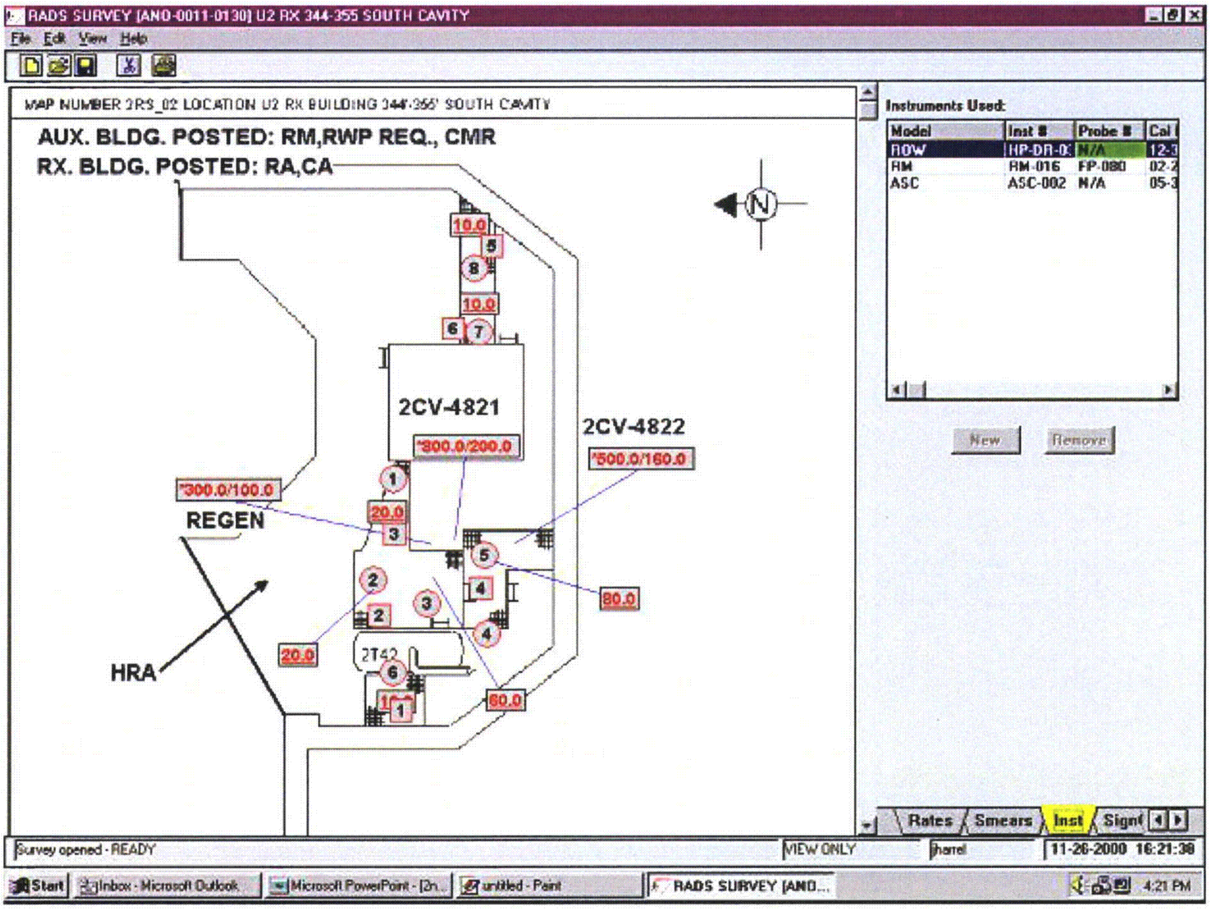

Figure 3-7
Sample Electronic Survey

Often it is important to visualize the radiation conditions over larger plant areas and components. The following image illustrates real time radiation field data superimposed over a significant portion of a containment building. This format can be extremely useful during periods of transient radiological conditions such as during forced oxygenation, crudburst cleanup, or post accident.

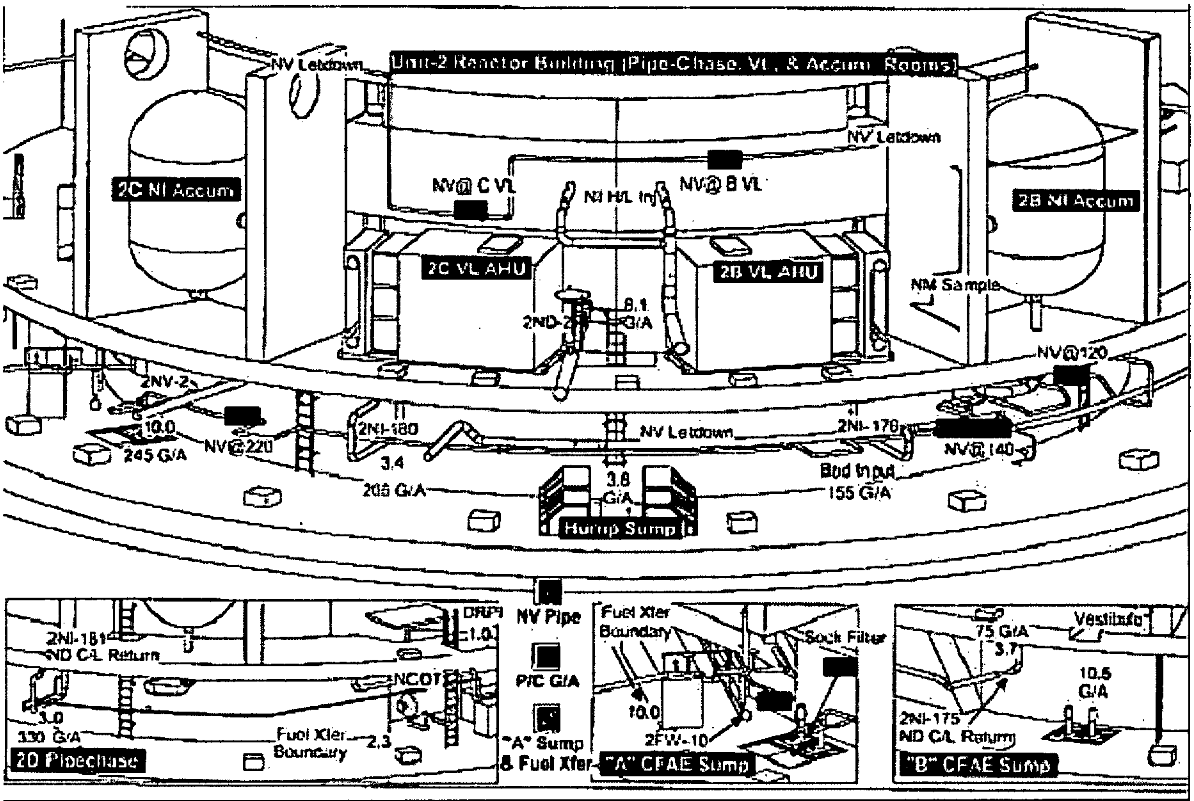

Figure 3-8
Containment Building Dose Rate Summary

Application

Several plants have initiated RMT programs to optimize survey frequencies to reflect area conditions and the need to update the survey status. Additionally, some plants have integrated their survey program with real time updating of surveys by a technician in the plant.

Status boards are typically located in locations adjacent to the work site, work site access, controlled area access, control stations, RP offices, and other key plant locations. They are also used to convey warnings associated with abnormal or otherwise significant information.

Industry Experience

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- Several stations report that using electronic surveys has improved the survey program's quality and accuracy; previous digital surveys are readily available to compare with new survey data.
- Several suppliers have incorporated user defined thresholds and alerts that help minimize the potential for missing or incorrect survey entries.
- Multi-site use of a standard electronic survey and notification system improves worker efficiency and reduces the potential for human performance errors.
- Survey data is readily available for prejob briefs and general worker updates minimizing the potential for regulatory non-compliance (NRC Occupational Radiation Exposure Cornerstone).
- Electronic survey data is readily retrievable for life of station and for potential radiation litigation cases.
- Several plants use PDAs for Chemistry data capture.
- The majority of US reactors employ or are implementing dose rate status boards.
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Radio Frequency Identification (RFID)

Technology Overview

RFID is an automatic identification method, relying on storing and remotely retrieving data using devices called RFID tags or transponders. An RFID tag is an object that can be applied to or incorporated into a product or component for the purpose of identification using radiowaves. Some tags can be read from several meters away and beyond the line of sight of the reader.

RFID systems use a unique identifier for each data point. The data tag can be linked to a specific parameter, component, equipment, or tool. A technology called chipless RFID allows for discrete identification of tags without an integrated circuit, thereby allowing tags to be printed directly onto assets at lower cost than traditional tags.

Application

This proven technology is widely used in numerous applications and is gaining acceptance in plant applications. It is used for equipment identification for operator rounds, logging maintenance, test, and radiation protection equipment and instruments in and out of inventory/storage/issue areas, and for waste containers. The output is used for tracking inventory, calibration dates, volumes, etc.

Industry Experience

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Personnel Vital Statistics

Technology Overview

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Application

None documented for power reactor application-to-date.

Industry Experience

None documented for power reactors-to-date.

Equipment Performance

Technology Overview

The technology available for equipment performance monitoring varies dramatically in capabilities, cost, technology, complexity and application. The most common performance parameters of interest include vibration, temperature, and pressure/differential pressure. Data are retrieved on a user-defined basis and transmitted and stored on plant computers.

Application

Many performance parameters are monitored by original, as-built detection and transmitting devices. The availability, capabilities and cost efficiency of new monitoring technologies continues to improve; as a result, plants are more aggressively pursuing this technology. Plant staff reductions further increase the potential benefit associated with this technology. Finally, as plants migrate towards predictive, rather than preventative, maintenance for key components, the availability of data that is live, more voluminous, and more accurate supports this change in strategy. However, the actual system upgrade to this technology at US reactors is evolving slowly because of the significant cost and resources required for implementation.

Industry Experience

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Advanced Imaging and Dimensional Measurement Technology

This section addresses two primary technologies that support area mapping, engineering, work control, emergency response, troubleshooting, and radiological control programs.

Surrogate Tour Systems

Technology Overview

This system uses digital photos to construct a simulated tour of plant areas. A specially designed camera shoots thousands of images in a 360 degree format. The photos are then seamlessly stitched and catalogued for retrieval. The retrieval process uses a joystick for manipulating the photos to provide a simulated tour from and to any direction. Additional data including dose rate and component identifiers and hyperlinks can be added to enhance the product's usefulness. Access across the Web allows multiple users to view site information simultaneously.

Application

This technology was the original remote tour technology of choice for many years. Recent programming and photography enhancements have increased the value of this product. However, unlike laser scanning, this technology does not currently offer the added benefit of accurate dimensioning or overlay (superimposing) of CAD drawings.

Industry Experience

- This technology is used at several US reactors.

Laser Scanning and 3D Models

Technology Overview

Laser scanning technology is used to provide a high-resolution (360° horizontal and over 300° vertical) and realistic representation of the scanned space. The technology provides an accurate depiction of piping, components, and structures. This technology can be used with as-built drawings and CAD systems to generate accurate 3D models.

Application

The resultant scan and/or 3D model can be used to identify critical equipment clearances during all phases of inspection and mitigation tasks to within ¼ to ½ inch (0.64 to 1.27 cm) tolerance. The laser map provides an effective tool for visualization and training for task specific crews. It can also be used to develop shielding strategies and to provide a map for post work interference and area recovery. Typically, this technology is used in conjunction with other software programs to integrate the laser scanning database with data from existing plant drawings and CAD models. The scanned database can also be used for other future applications such as radiological control applications, integration with plant documents and databases, outage support planning, equipment removal, etc.

At least one vendor offers the option to integrate a plant 3D model with a Primavera Project Schedule. Orthographic drawings in AutoCAD format, in a selected scale, can be automatically generated from the scan database. The fundamental tool is shown in Figure 3-9 with an actual resultant scan illustrated in Figure 3-10 that includes dimensions calculated by the scanning software.

Figure 3-9
Laser Scanner (typical)

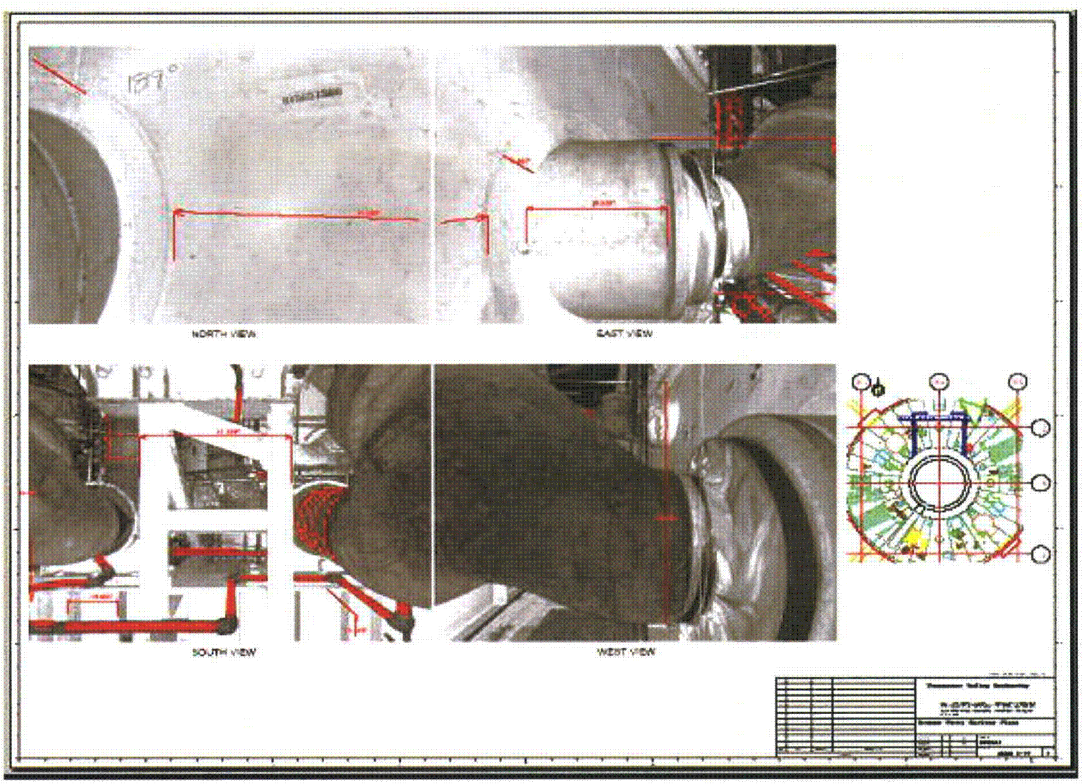

Figure 3-10
Laser Scan Image with Dimensions

Industry Experience

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Robotics Applications

Technology Overview

Typical robotic applications include wheeled, tracked, fixed and submersible technologies. The robotic base unit can include end-effectors such as:

brushes	orbital welding devices
cutting devices	ribbon lifts
grinders	telescopic booms
inspection devices	ultrasound probes
manipulators	vacuum heads

The tools can be driven by geared, belt, track or hydraulic/pneumatic motive forces. Typical specifications for a wheeled robotic base unit from one of the primary US reactor suppliers are captured in the following table.

Table 3-1
ROV Scarab I and Scarab II Specifications

Construction	Solid cast aluminum-black anodized
Motors	(3) 115V DC w/heavy duty transmission reduction
Drive Shafts	Hardened stainless steel
Body Seals	Buna-N
Shaft Seals	Custom polymer lip seals
Weight	75lbs out of water
Dimensions	Approx. 18"L, 15"W, 18"H (w/turret device horizontal)
Speed	0 to 30 feet per minute
Turning Radius	Pivots on center axis
Final Drive	4 solid rubber tires – 65 durometer, (tracks optional)
Body Lights	(2) 75W, 110V, wet or dry
Turret Lights	(1) 75W or 250W, 110V wet/dry (app. specific)
Accessory Connectors	Heavy duty rubber underwater pluggable
Umbilical Connector	37 pin underwater
Umbilical	37 conductor, urethane jacket, 125 feet long
Internal Cameras (Driving Cameras)	>700 horizontal TV lines, rad tolerant, monochrome
External Cameras (Turret Cameras)	>700 horizontal TV lines, rad tolerant, monochrome or color, remote operated zoom-focus w/ optional 6X zoom and macro-zoom capability (mono) - or - 25x optical (color)
Turret Movement	270 degrees pan, 90 degrees of tilt
Audio	Electret microphone mounted in external camera tube
Vehicle Operators Control Console	R.O.V. Technologies, Inc. standard configuration
Main Control Console	R.O.V. Technologies, Inc. standard Tri-Con System

The following figure illustrates one typical wheeled robot designed for general, multiple use applications. Tooling can be connected to the robot base unit.

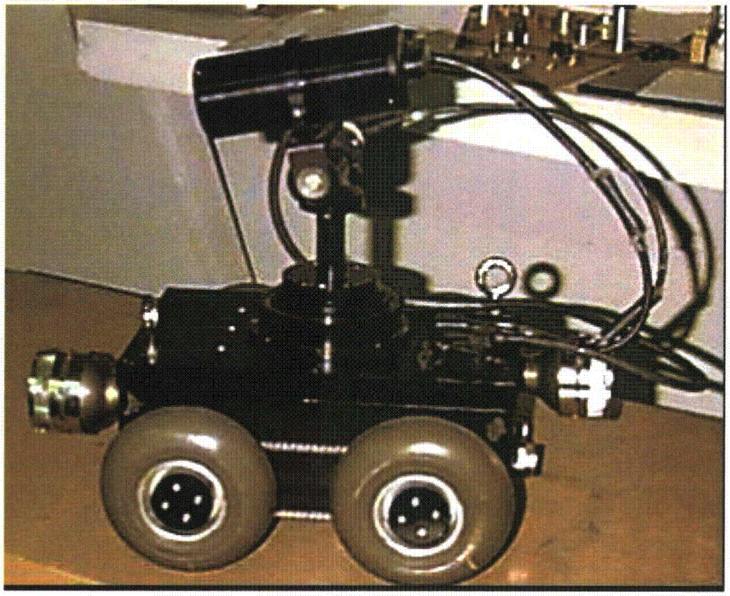

Figure 3-11
Wheeled Robot with Observation Camera (R.O.V. Technologies)

Fixed system technology is typically designed for a specific application such as reactor head inspections, steam generator tube inspections, etc. They are most often manually installed, then calibrated to the location prior to use. The following figures illustrate two different approaches to fixed inspection of control rod drive nozzles under a PWR head.

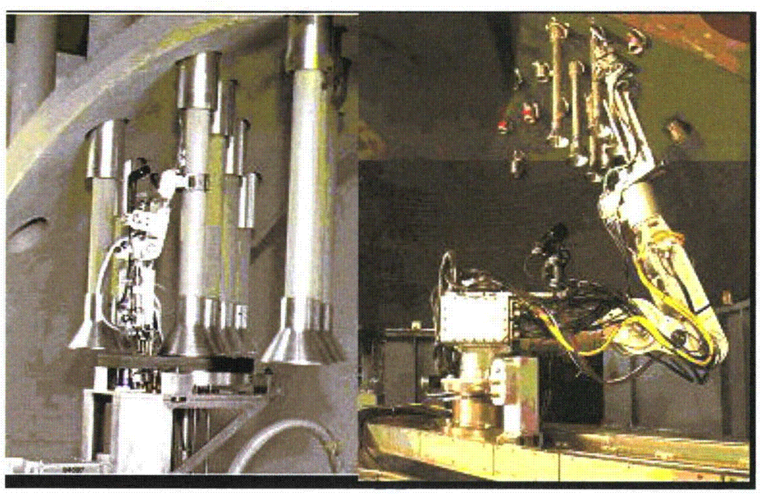

Figure 3-12
Under Reactor Head - Robotic Inspection Tooling

Submersible technology involves integration of buoyancy and thruster and directional controls into a single unit. The units are typically tethered to supply the required services (e.g., electrical, air, water) and to allow emergency retrieval in the event of an equipment failure. The following figures illustrate submersible used for inspections and observation.

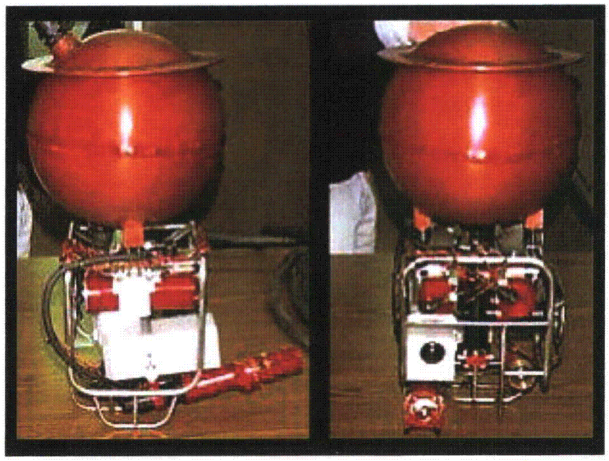

Figure 3-13
Mini Submersible (ROV Technologies)

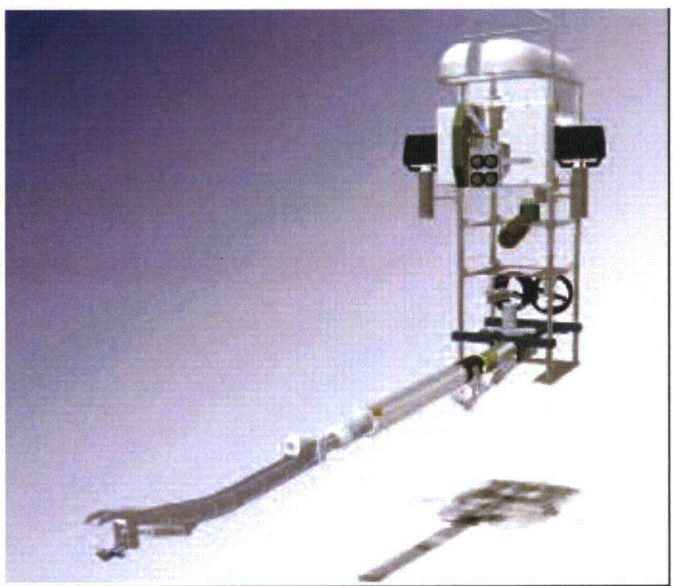

Figure 3-14
Submersible for BWR In-Vessel Inspection (GE Nuclear)

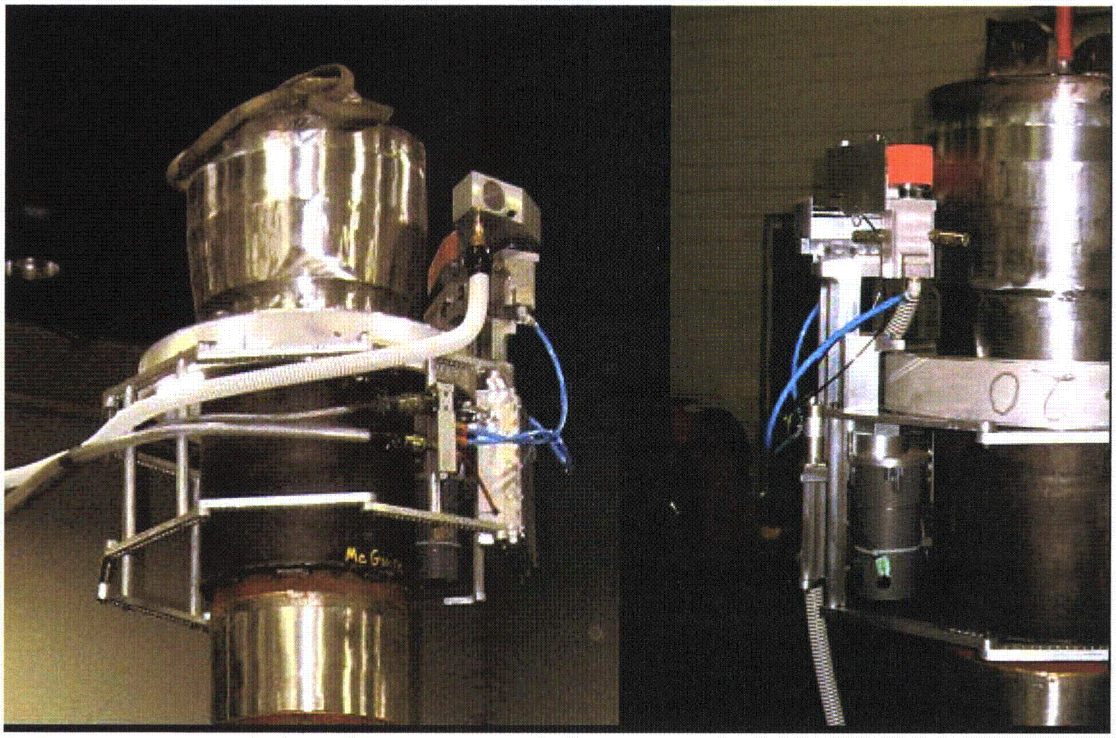

Figure 3-15
Semi-Automated Belt Sander Application (WSI[2])

Application

Robotics are used in many applications in a variety of plant components and areas.

- Visual Inspections
- Video Measurement
- Retrievals
- Vacuuming
- High Pressure Jet-Spraying
- Remote Manipulation
- Debris Collection And Retrieval
- Brushing
- Radiation Surveying (Both Dose Rate And Smears

Submersible technology is typically used for underwater inspection, tank cleaning and desludging, welding, cutting and other task-specific functions.

Industry Experience

- Wheeled, tracked, submersible, and fixed mount robots are used by the majority of US reactors in some capacity at some point in plant life.
- Several stations have historically implemented aggressive robotic equipment strategies. The majority of those no longer use them to the extent they had in the past. This is primarily attributable to a lack of clear ownership that impacts their use, material condition and overall effectiveness.
- The majority of station currently contract robotic development and/or deployment for special projects such as tank cleanings, weld inspections, etc.
- The "simpler is better" concept applies to this technology. More complex designs have the potential for more frequent operability issues, potentially reducing the overall exposure and cost benefit.
- Robotic Head Inspection
  - The robot requires personnel access at the worksite for mounting, adjustment and removal. The duration varies dramatically and is impacted by pipe diameter, obstructions, width, indications and equipment operability. This work is usually performed by a technician with a lower qualification rating. Automated UT offers the following advantages and disadvantages:
  - Advantages
    - Reduced total time at worksite for qualified personnel, resulting in reductions in personnel exposure. The exposure savings will vary dramatically and are dependent on the specific application, robotic technology, and work site source term.
    - Better quality inspection records.
    - High quality data that can be used more effectively to evaluate an indication in its context.
    - Depth sizing is qualified for this technology.
  - Disadvantages
    - The required scanner inspection configuration cannot be achieved for all applications.
    - Higher cost than manual options.

GPS Positioning

Technology Overview

Thermo-Fisher and other companies are developing systems for the location and tracking of personnel and equipment within buildings. Several companies currently have an outdoor system that is GPS based to track and locate personnel such as when surveys are being performed in large areas. The knowledge gained from the outdoor system is being used to enhance its indoor counterpart

Application

No indoor applications documented for power reactors-to-date.

Industry Experience

No indoor applications documented for power reactors-to-date.

Cost and Cost Justification

Cost

Cost data was solicited from several industry suppliers and utilities. The consistent response was that the cost will vary significantly based on equipment options and installation requirements.

A standard system with video, audio, telemetry software and hardware was estimated by one supplier to cost approximately $ 750,000 - $1,000,000 (US). An electronic dosimeter system with telemetry capability that supports 1000 devices with readers and software was estimated at $500,000. These costs are based on a single reactor installation and assume that a fiber optic or wireless infrastructure is installed.

To provide a realistic estimate for fiber installation would require a site assessment that involves both IT and tele-communication engineers. In some applications, the conduit used for phone services can be used for fiber. A very rough estimate for a fiber or category 6 twisted pair hardwire installation at an existing nuclear facility could range from 1 to 3 million dollars.

Cost Justification (Savings)

Building a business case for implementing remote technologies involves a host of factors. It is impossible and impractical to define a universal cost savings metric with quantifiable results. However, the factors that should be considered in the analysis can be defined. Several of those are listed below:

Incurred Cost

- Design
- Engineering
- Installation
- IT resources to design and implement
- Project management
- Direct cost- Software
- Direct cost – Hardware – servers, etc.

- Future O&M
- Other labor resources (e.g., RP, Training development and implementation)

Incurred Exposure

- Design and Engineering
- Installation and testing
- Outage specific installations/setup (recurring)

Avoided Cost

- Recurring setup of entire system
- Personnel resource reduction
- Error/rework reduction
- Live equipment performance/predictive maintenance
- Avoided liability associated with regulatory non-compliance
- Maintenance associated non-standard system (e.g., multiple components from multiple suppliers)

Avoided Exposure

- Reduced staff time in RCA
- Improved planning and estimating

The following table summarizes one utility's annual savings analysis for three technologies. The table includes both a high and low estimate for each factor.

Table 3-2
Examples Annual Savings per Reactor

Technology		Low ($ US)	High ($ US)
Electronic Dosimetry			
Actual Cost Reduction:		57,000	114,000
Efficiency Cost Savings:		66,500	76,000
ALARA Cost Savings:		87,500	150,000
Electronic Dosimetry Total		211,000	340,000
Video Tour System		Low	High
Efficiency Cost Savings:		68,640	68,640
Electronic Survey		Low	High
Efficiency Cost Savings		113,818	113,818
TOTAL		393,458	522,458

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Table 3-3
Electronic Survey Cost Analysis

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Table 3-3 (continued)
Electronic Survey Cost Analysis

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This demonstrates a very important aspect of successful RMT programs. That is the sponsorship of senior management and the 'buy-in' of all Stakeholders.

Cautions and General Considerations

The following list was developed from industry experience.

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- General data management including storage, end user access and data format, and data security should be considered in RMT application planning.
- Wireless frequencies should be carefully evaluated for use and interference. Interference from devices on the same frequency led to failure or shutdown of teledosimetry in at least one instance. One example involved the use of a wireless underwater camera for fuel inspection. In this case the EDs were powered off to eliminate the 2.4 MHz interference.
- Many RMT systems have the ability to record images on videotape or computer, and some plants are doing this now. A standard should be established for recordings based on task type and end use of material. This standard should recommend recordings that are required for training, documentation of a task as in the case of insurance purposes, and as a reference for repeating or similar future tasks. This standard should also address the recording format and how long the recording should be saved.
- Many plants continue to struggle with the oversight and management of their RMT program. Ownership of the equipment is a major issue and governs the purchase, installation, operation, repair, and replacement of the system and its components. This issue is particularly noteworthy for non-RP applications. In some plants, the responsibilities for the remote monitoring system are shared among the I&C, RP and IT departments.
- Many stations continue to develop protocols and standards related to the interface of the RMT system with the plant network, frequencies used, and network security.

Plants Applying RMT

The majority of US utilities use at least some minimal remote monitoring technology. Several plants are considered the leaders in applying this technology. The following list summarizes several stations that are involved in RMT application. Known contact information has been provided for several individuals.

Table 3-4
Plants Using Remote Monitoring Technologies

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U.S. Reactor Remote Monitoring Benchmarking Trip

A team representing JNES completed a remote monitoring and ALARA Program benchmarking trip during the week of February 10, 2008. Southern Nuclear's Plant Vogtle and Entergy's ANO stations were visited by five individuals. The stations' remote monitoring technology applications were reviewed in detail including hardware, training, and field application programs. Additional information was provided related to ALARA program management. The following are summary notes from the benchmark trip.

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Remote Monitoring Technology History/General Performance

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RMT Procedures

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Use of Remote Monitoring

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RadIS discussion

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CMS

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Camera/System Description

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ALARA Committee

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ANO Site Notes

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Remote Monitoring Comparison Summary

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Table 3-5
Remote Monitoring System Comparison

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4
PRIMARY SYSTEM CLEANUP OPTIONS

This section address two primary cleanup system topics; 1] primary system cleanup with filtration resin and 2] primary system cleanup filtration using cartridge filters.

Primary System Cleanup with Filtration Resin

PWR Chemical Volume and Control Systems (CVCS) have effectively removed both soluble and particulate contaminants and radioactivity. Recent industry experiences have claimed significant improvements in overall radiological performance such as reduced forced oxidation peaks and reduced radiation fields because of increased removal efficiency of the CVCS system. The following information summarizes an analysis from a 'black-box' approach, and the results show that cleanup systems have no effect on forced oxidation peaks, soluble activity can be removed to nearly 100% efficiency, and while removal of as much insoluble activity is desirable, the radiation impacts of insoluble activity are generally limited to local deposition and airborne contamination issues.

There has been considerable discussion in the industry about using 'specialty media' as an overlay in the CVCS demineralizers during forced oxidation cleanup. Users of specialty media claim that the media may enhance cleanup of particulate corrosion products and activity, which then reduces shutdown dose rates for the current outage and reduces the forced oxidation peak for the subsequent outage.

Review of Cleanup System Fundamentals

The following figure illustrates a demineralizer with a single influent and single effluent stream. Other unit operations such as filters and parallel or series demineralizers are omitted for simplicity.

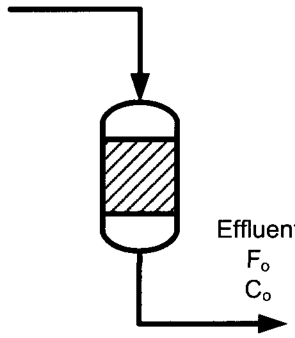

Figure 4-1
Demineralizer with Ion Exchange Media

Viewing the demineralizer (or entire cleanup system) as a black-box, the contaminant mass or activity of a single specie removed from the influent is defined by the following equation:

$$m_r = \int_{t_o}^{t_f} F(t)\left(C_i(t) - C_e(t)\right)dt \qquad \textbf{Equation 4-1}$$

where

m_r = contaminant mass (or activity) removed (g, Ci)
$F(t)$ = flow rate as a function of time (liters/second)
$C_i(t)$ = influent concentration of a contaminant as a function of time (g or Ci/liter)
$C_e(t)$ = effluent concentration of a contaminant as a function of time (g or Ci/liter)
t_o = initial time (seconds)
t_f = final time (seconds)

When viewing the removal in small increments of time, the flow rate and concentrations can be considered constant and the equation can be simplified to

$$m_r = F(C_i - C_e)\Delta t \qquad \textbf{Equation 4-2}$$

A parametric analysis shows that maximizing the removed mass, m_r, requires increased flow rate, increased time, increased influent concentration, or decreased effluent concentration. Applying parameters shows that the sensitivity of the removal term to effluent concentration is quite small. Figure 4-2 shows a typical RCS cleanup curve for a plant with the effluent concentration plotted (note the log scale). The area between the influent and effluent concentration curves multiplied by the flow rate yields the total curies.

Table 4-1 shows the results of a numerical integration that uses the influent curve in Figure 4-2 and varies the effluent concentration and flow rate. It is clear from the table that the effluent concentration has little effect on the total removal (a difference of 10 or 14 curies), while if the flow rate were increased by 33%, the removal increases proportionately. This indicates that although it is desirable to have near-zero effluent concentrations, maximizing the total removal is more dependent on maximizing the influent concentrations, flow rate, and clean-up time.

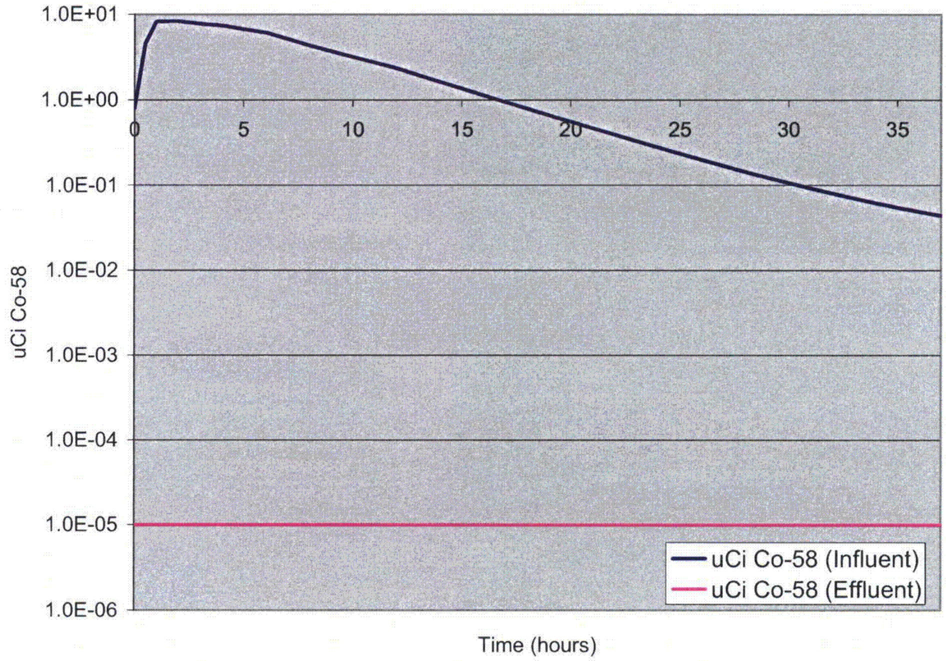

Figure 4-2
Typical RCS Cleanup Curve

Table 4-1
Sensitivity Analysis of Demineralizer Removal to Flow Rate and Effluent Concentrations

DF (based on peak concentration	Effluent Concentration (µCi/g)	Co-58 Removed (Curies)
10	0.8	1350
100	0.08	2097
1000	0.008	2171
10000	0.0008	2179

Effects of Cleanup Systems on Forced Oxidation Peaks

A review of Table 4-1 shows that the effect of the cleanup system on the magnitude of a forced oxidation peak (crud burst) is minimal. The potential extra removal is on the order of 10 to 20 curies (or in the case of metallic nickel removal, 10 to 20 grams), and a typical cleanup is 2000 curies, which is less than 1% of the total source. Therefore, claims that enhanced cleanup can remove parent nuclides such as nickel-58 from the system and reduce subsequent outage crud bursts are false. A recent EPRI report has indicated PWR crud bursts are a result of the combination of steam generator tubing material, surface area, manufacturing process, as well as the core duty and shutdown operations that can affect the release of crud, such as multiple reactor coolant pump (RCP) operation after forced oxidation.

Effects of Cleanup Systems on Activity Removal

Although definitions vary, radioactive contaminants are loosely grouped into soluble (ionic) and particulate (filterable) contaminants. Soluble contaminants may be removed with ion exchange resins, while particulate contaminants may or may not be removed by filtration systems, depending on the particle size and filtration threshold.

A given atom can be both ionic and particulate throughout the lifecycle in the plant. For example, nickel can be in particulate form until it reacts with oxygen to form ionic nickel. The dissolution of nickel may release ionic cobalt-58, which has a slightly lower solubility than nickel, and the cobalt-58 may precipitate back into the solid form. The solubilities of nickel, iron, and cobalt vary with temperature, hydrogen concentration, and solution pH; hence the interactions with the cleanup system are complex. While there is considerable interaction between ionic and particulate activity, the kinetics of the reactions are often slow, and the removal mechanisms for ions and particulates can be discussed separately.

Soluble Contaminant Removal

Soluble contaminants and activity are effectively part of the coolant, and given sufficient time and/or turbulent flow, the contaminants and activity will mix perfectly. Because of this mixing, if there is not precipitation or rapid exchange mechanism into the oxide layers of the ex-core surfaces, it is clear from equation [2] that the lower limit of a soluble RCS contaminant concentration is the effluent concentration of the demineralizer.

For a mixed bed ion exchange column that is not kinetically limited, the ionic specie effluent concentration is determined by the ionic loading at the bottom of the bed. Therefore, when practical, it is a good operational practice to replace mixed bed resin before shutdown operations, and also not disturb ion exchange vessels once they have been placed in service so as much of the removed activity as possible remains in the top of the bed.

Particulate Activity

Particulate activity is unlike soluble activity in several ways:

- Particulates are of varying size and they do not uniformly distribute in the solution according to diffusion laws,
- They have inertia and may not necessarily follow the direction of flow.
- They have a charge and may be attracted to or repelled from surfaces.

In order for the CVCS system to remove enough particulate activity with filters or specialty media to reduce dose rates, the following criteria must be satisfied:

- There must be sufficient particulate activity
- The activity must reach the filtration system, i.e. it cannot deposit on the piping surfaces or heat exchangers before the filter/media system
- There must be a removal mechanism, i.e. either mechanical filtration or a charge based driving force.

The following sub-sections review the expected particle size characteristics, and consider the factors that affect the removal of particulates from the coolant.

Particle Size Distribution

The particle size distribution for nuclear power plant systems is difficult to quantify accurately. Measurements from grab samples exposed to air will be skewed by iron precipitation and nickel solubilization. Particle size distribution research has been performed and parallel research indicates that for liquid system that did not have disturbances, the particle count is proportional to the diameter of the particle:

$$N = c(d_p)^p$$ **Equation 4-3**

Where,

N = number of particles in a measured sample
d_p = particle diameter
c, p = empirically derived constants.

The research noted that for liquid systems, $p = -3.01$, or effectively, the particle count is proportional to the inverse of the volume of the particle.

Table 4-2 shows the calculations used to estimate the particle mass distribution. The power law expression indicates a nearly uniform mass distribution across the particle diameter range. The largest particle size was assumed to be 5.0 microns (based on the most common nominal filters used in US CVCS systems). A Curie estimate of particulate activity was developed assuming 0.5 ppm of particulate nickel (which is the approximate average of insoluble nickel taken from the EPRI Shutdown Template), system volume of 100,000 gallons (379,000 liters), spherical particles, a nickel density of 8.0 g/cm^3, a saturated activation of 1.2 Ci Co-58 per gram Ni, and a particle size distribution ranging from 0.01 to 5 microns (centered around 0.6 microns). The harmonic mean of the particle size ranges is used to determine the average diameter of the particle ranges.

Table 4-2
Parameters Used to Estimate the Log-Normal Distribution of Particulate Sizes

Particle size distribution (lower-upper in microns)	Harmonic Mean Diameter (cm)	Particle Count %	Mass Fraction	Nickel Mass (g)	Co-58 Curies	Cum mass percent	Cum Co-58 Curies
2.5	5	3.33E-04	0.0000%	0.163	30.9	38.61	100.0%
1	2.5	1.43E-04	0.0002%	0.164	31.2	38.94	83.7%
0.45	1	6.21E-05	0.0024%	0.166	31.4	39.27	67.3%
0.25	0.45	3.21E-05	0.0175%	0.167	31.6	39.53	50.7%
0.1	0.25	1.43E-05	0.2015%	0.168	31.9	39.85	34.0%
0.01	0.1	1.82E-06	99.7783%	0.172	32.5	40.68	17.2%
Total				**1**	**189.5**	**236.88**	

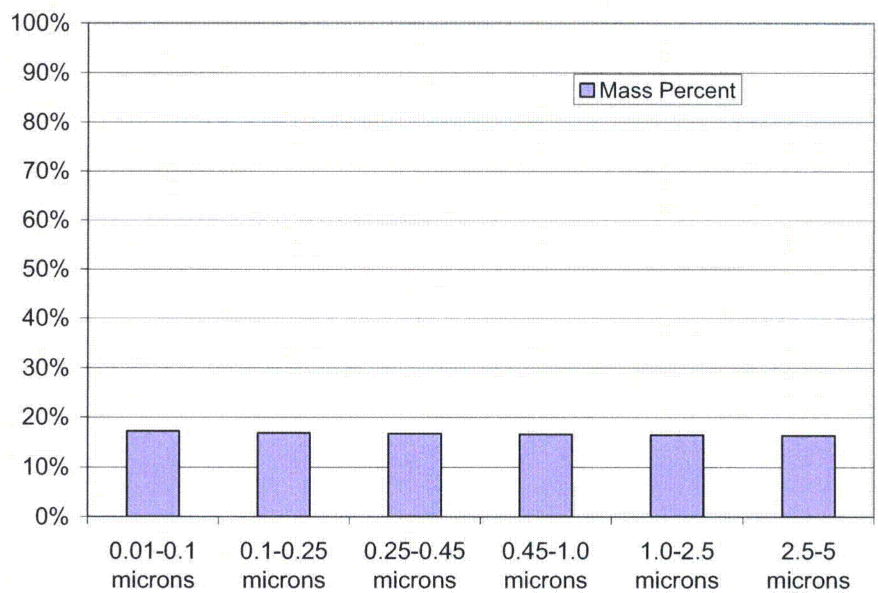

Figure 4-3
Expected Mass Distribution of Particles Based on Diameter Range

From the stated assumptions and the results of

Table 4-2 and Figure 4-3, the activity in particulate form is roughly 10% of the total activity. While the number of particles smaller than 0.1 microns will dominate the total particulate count, the mass and activity of particles less than 0.1 microns will be less than 20% of the total activity in the particulate phase, and less than 2% of the activity in the coolant during shutdown. Therefore, in CVCS systems with enhanced filtration (defined as a filtration threshold less than 0.1 microns) and complete transport of the particles to the filtration system, the expected increased removal is expected to be approximately 2%, and for the sample case, about 40 Curies of particulate activity can be removed.

Particle Removal with CVCS Systems

For the sample case, the previous section showed that up to 40 Curies are available for supplemental removal if all of the particles smaller than 0.1 microns are suspended in the coolant and reach the filtration system. However, as noted earlier, particles have charge that is defined by the zeta potential. Zeta potential is a measure of the effective charge of a particle in the solution. Piping surfaces, in general, have a positive charge; however, negative ions and particles collect on the surface and there is an effective surface charge at a finite distance from the surface. The charge of the particle dictates whether the particle will be repelled from the surfaces or stick to them.

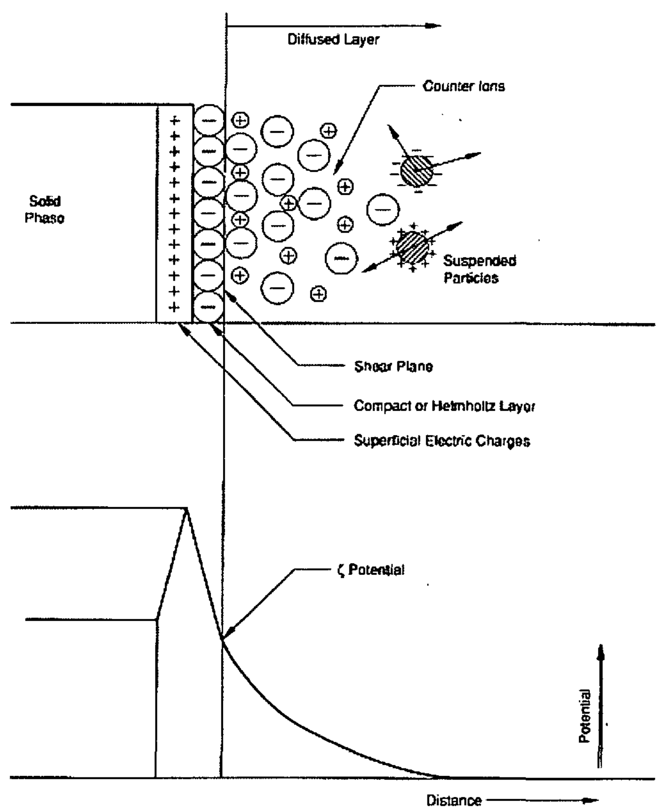

Figure 4-4
Illustration of Zeta Potential

The charge of the particle is known to change with temperature, pH, and chemical composition. A parameter of interest is the point of zero charge (PZC), which states when the particle changes charge sign and has effectively zero charge. This is of concern because at that point the particle may be attracted to, or repelled from, surfaces as the conditions change.

The end result of this analysis is that the overall particulate removed from the coolant by the CVCS system should be less then the maximum predicted particulate removal because of deposition of particulates on the surfaces before reaching the filters and demineralizers. Therefore, for the sample case considered, the total curies that can be removed from the system by enhanced filtration will be less than 40 Curies.

The distribution of the remaining particulate activity throughout the system raises other questions. A rough estimate of the total surface area of piping surfaces is 1e9 cm^2. Assuming an even distribution of 40 Curies of cobalt-58, this would lead to an activity coverage of 40 nCi/cm^2, which would have minimal impact on the overall radiation fields. If the particulates were preferentially deposited in a smaller area (such as auxiliary systems), then it is possible that the local radiation fields could be impacted. However, this mechanism is still under investigation.

Conclusions

The following conclusions may be drawn from this information:

- The CVCS system, while effective for cleaning up existing activity, has no effect on PWR shutdown forced oxidation peak magnitudes.
- Soluble (ionic) activity comprises of approximately 90% of the total activity in the coolant, and the CVCS demineralizer is capable of removing the soluble activity to nearly 100% efficiency.
- Based on particle size estimates, insoluble activity with diameters less than 0.1 microns comprises about 2-5% of the total activity in the reactor coolant.
- Insoluble (particulate) activity may not be removed to 100% efficiency. Particulates should have a relatively small contribution to the ex-core radiation fields; however, their removal is desirable in situations where the particulate may become airborne or cause personnel contamination events.

Primary System Cleanup Filtration using Cartridge Filters

RCS Purification

PWR RCS purification system designs vary for each reactor; however the systems generally apply filtration in one of three configurations:

1. Upstream of the demineralizer to protect the media from insoluble particulate

2. Downstream of the demineralizer to capture resin fines and/or resin in the event of a retention element failure

3. Filters in both upstream and downstream locations

Cooled RCS letdown is routed through one or more letdown demineralizers. Figure 4-5 is a detailed schematic of a system showing the ion exchangers, filter and associated support equipment for one plant application.

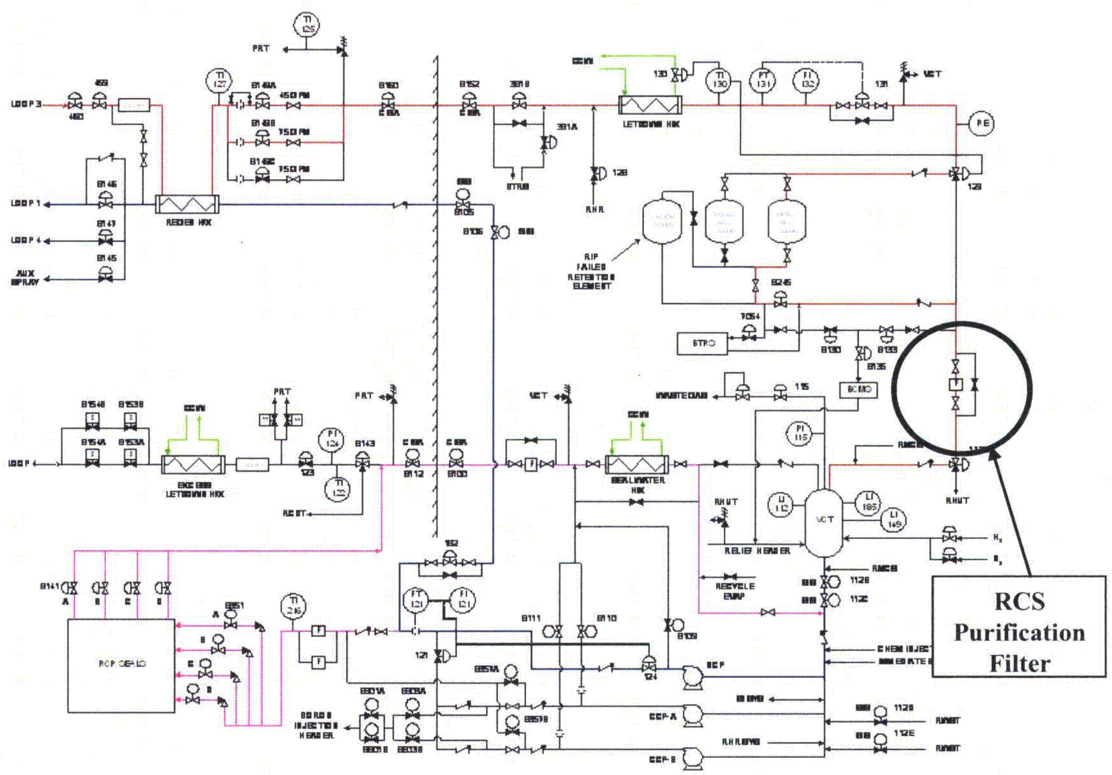

Figure 4-5
RCS Purification System Diagram

Historically, filtration and ion exchange have proven to be marginally effective for reducing the concentration of activity available for deposition in primary systems. The low flow rate and media materials and design for the vast majority of installed cleanup system applications have proven to be less than adequate for the volume of liquid requiring treatment. This resulted in poor turnover ratios (time required to completely process one full system volume) and less than desirable particulate removal. Recent advancements in both materials and element design have improved the overall effectiveness of primary filtration systems.

A typical implementation program includes a particle size analysis, an evaluation of the filter housing(s), flow rate and differential pressure considerations, and spent filter disposition. The majority of existing filter housings require a one time modification; this is typically achieved without modifying or impacting the pressure boundary. The initial filter rating should be based on the results of the implementation analysis. A micron size reduction plan is then developed. The majority of plants using this technology have progressed to the use of submicron ratings for power operation, typically using 0.1 to 0.45 micron elements. Some plants increase the micron rating just prior to outages and/or startup to prevent premature fouling of the elements. In many configurations, the filters cannot be isolated during shutdown crudburst cleanup evolutions without isolating the ion exchanger; this would negatively impact the cleanup efficiency and outage dose rates.

Plant RCS and SFP Filter Generation

Indian Point Unit 3 and Callaway plant were the industry leaders in application of this technology. With a few notable exceptions, the Callaway data in Figure 4-6 clearly show a downward trend in all primary filter generation in spite of filter micron rating reductions. The trend is more pronounced for the SFP and RCS filters since the end of cycle 10. While the data clearly indicate the benefits associated with this technology, it is known that other factors impact filtration efficiency and loading including operational primary chemistry controls and shutdown and startup chemistry, purification and operating strategies.

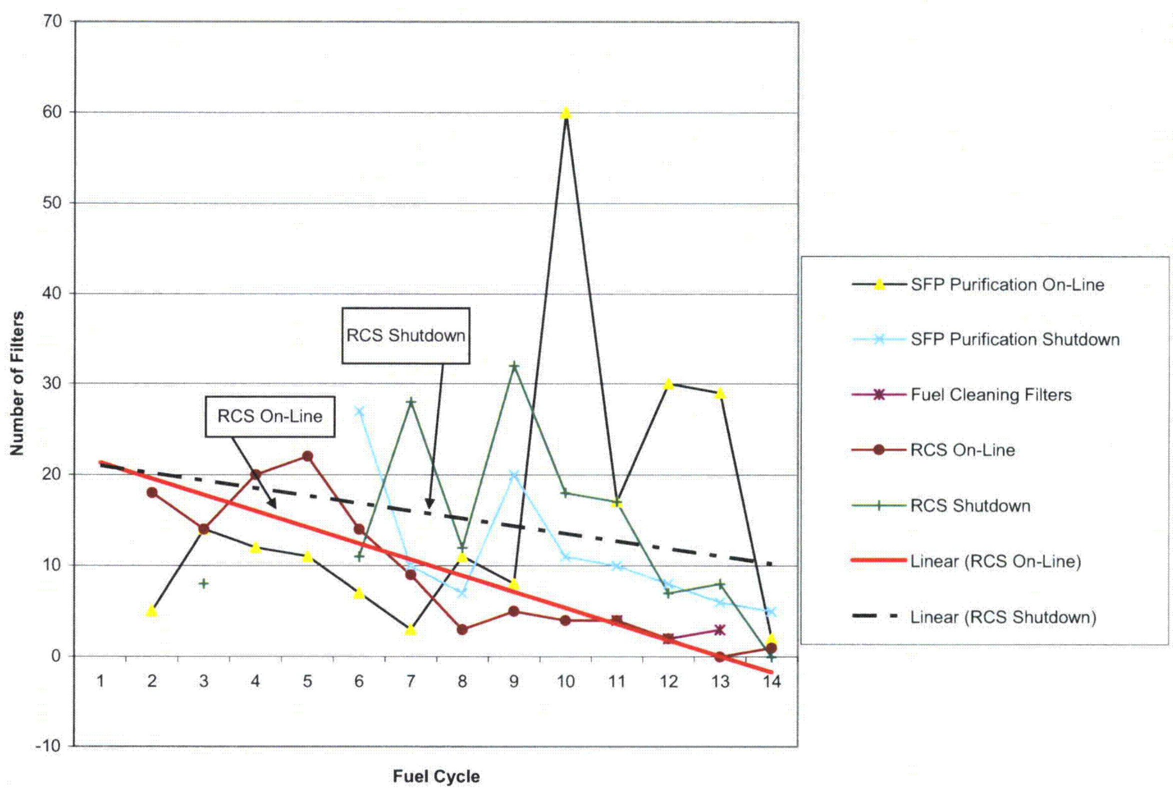

Figure 4-6
Callaway Filter Performance

Note 1: Initial RCS filters (FBG06) were 30 micron nominal. When filter downsizing began, absolute filters were employed as replacement elements.

Note 2: Five RCS filters were changed during Cycle 13 due to the Summer Reliability Outage during March 2003. One filter has been changed since that time to upsize for Refuel 13 shutdown. During Refuel 13 only one RCS filter was replaced during shutdown cleanup. One filter was replaced following head lift and swapping RHR trains. The remaining filters were replaced to support hydrazine additions in preparation for startup.

Note 3: Fourteen 1 micron absolute rated RCS filters were replaced during the plant heatup following SGR during Refuel 14.

Filter micron reduction is intended to reduce insoluble particulate in the RCS systems. This should result in a reduction in filter generation during steady-state operation and a parallel downward trend in program costs. The periods of increased RCS filter generation in cycles 13 (operation) & 14 (refuel) were a result of system perturbations and major maintenance activities that generated additional solids in addition to baseline conditions. These upsets were temporary and the RCS returned to the desired baseline quality afterward. The high SFP filter generation during cycles11, 12 & 13 reflect the generation of solids from SFP activities associated with refueling and fuel cleaning.

Plant Filter Program Cost

Table 4-3 summarizes a typical filter generation and cost analysis.

Table 4-3
Filter Generation and Cost – <u>Including</u> Upsets

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Table 4-4
Filter Generation and Cost – Excluding Upsets

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- Chemistry regime changes (e.g., zinc addition) will impact crud concentrations and characteristics and will therefore impact filter generation rates.
- On-line purification system's cleanup flow rate.
- Shutdown cleanup efficiency (duration, flow rate, etc.)

Shutdown cleanup efficiency and start-up chemistry should be evaluated for impact to particulate content of the primary system and filter loading.

Benefits and Conclusions

The use of a submicron, absolute rated filtration strategy can reduce the insoluble particulate in RCS system assuming the particulates can reach the filters without sticking to the piping walls. This provides benefits that include reduced particle impingement on reactor coolant pump seals and reductions in the potential for personnel contamination during system opening or cavity decontamination evolutions.

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5 GROUNDWATER CONTAMINATION

Background

Experience at numerous operating and decommissioning commercial nuclear power stations has positively identified unintentional releases of small quantities of radionuclides from plant structures. This is a high visibility regulatory issue in the U.S. To date, this has not represented a scientific risk, however it is having a very significant societal impact and an equally significant impact on individual sites' costs and resource allocation related to research, mitigation and long-term regulatory compliance efforts. It is not clear if this is an issue in Japan. However, Japan is constructing many new reactors and those designs should build on recent U.S. experience by incorporating those lessons learned that minimize the potential for groundwater contamination.

Any one or a combination of component, system, and structural integrity failures has ultimately led to unmonitored activity being released to both the site and local environment. This in turn results in a major challenge to utilities to identify and control the source, and remediate and/or monitor the contamination over the life of the plant. In several instances, a significant effort was required to mitigate the public trepidation and negative environment resulting from leakage of radioactivity into public drinking water supplies or bodies of water used for recreational purposes. Additionally, all operating nuclear power stations will eventually become uneconomic for continued operation, will reach the end of their useful life cycle, and will be decommissioned. The level of effort required to monitor and/or recover structures, soil, and hydrogeological formations due to tritium and other radioactive contamination is very significant and costly.

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Design Principles

There are several important design principles related to ground water contamination control.

Containment

Containing liquids at their point of generation is the most critical design element. Design or unintentional leakage that is contained and dispositioned in a controlled manner eliminates the potential for unmonitored, unplanned releases. This is accomplished using appropriately designed barriers (structural, system, component and environmental), berms, moats, and sealing systems.

Structure, System and Component Integrity

These design considerations include specific features such as material selection that is compatible with the contained liquid, 1) system, component and structure fabrication, 2) system, component, and structural joint technology, and 3) management of overpressure or overflow conditions. They apply to both interior and exterior applications.

Exterior Systems and Components

Minimizing the volume of radioactive liquids routed in exterior areas either in an above or below grade configuration, reduces the potential for leakage of any magnitude. Where required, the design, placement, and interconnections to structures and systems requires careful consideration to minimize the potential for barrier failure and to contain design and incidental leakage.

Leak Detection and Mitigation

Operating under the assumption that some leakage is inevitable at some point in the plant life, the ability to monitor, inspect, test, and repair barriers is a crucial design element. This is particularly true for those applications that involve highly active or large volume liquids such as spent fuel pools, refueling water storage tanks and below grade piping.

Terminology

Recommendations for controlling liquids may relate to any combination of plant design, operation or decommissioning considerations. In most instances the term "exterior" means it is located outside and/or exposed to the environment.

General definitions for the terms below grade, buried, and embedded follow; they are not absolute and are intended to define boundaries and position relative to an elevation, surface or structure.

Below Grade: Systems, structures or components (SSCs) whose elevation is below a reference ground level at that point on the site whether the location is inside or outside of a building (e.g., a building whose lowest elevation is lower than personnel or equipment access roads)

Underground or Buried: SSCs that are outside of any building or other accessible structure and are surrounded by soil, fill, pavement, etc. on all sides and are generally inaccessible without excavation (e.g., a tank with or without a concrete containment structure that is completely covered by soil).

Embedded: A pipe or component that is partially or fully surrounded by concrete (e.g., floor drain piping and traps).

Access for Inspection: This varies by application. Basic requirement is provision of access for the appropriate, proven inspection technology with detection capabilities to achieve the desired result (e.g., boroscope or camera for spent fuel pool void space inspection between double barrier systems, personnel access for inspecting structural joints)

When evaluating design requirements, the it is imperative to assess planned and unplanned liquid considerations related to the following categories:

- Structures That Contain Radioactive, or Potentially Radioactive Systems or Components
- Spent Fuel Building, Pool, and Transfer Canal
- Sumps and Associated Piping/Components that Contain Radioactive or Potentially Radioactive Liquids
- Outside Areas (Exposed to the Environment)
- Storm and Curtain Drains
- Ponds and Basins
- Piping Containing Radioactive or Potentially Radioactive Liquids
- Tanks Containing Radioactive or Potentially Radioactive Liquids
- Valves in Systems that Contain Radioactive or Potentially Radioactive Liquids
- HVAC
- Live Steam Systems
- Monitoring Wells and Site Water Sources
- Barrier Inspection and Integrity Testing
- Site Counting Facility Location
- Specific AE/Construction Considerations

6
MAIN CONTROL ROOM HABITABILITY

Background

The control room HVAC system serves the control room habitability zone, that is, those spaces that must be maintained habitable following a postulated accident to allow the orderly shutdown of the reactor. The control room HVAC system performs the following functions:

- Controls the environmental conditions in the main control room
- Pressurizes or isolates the control room to prevent infiltration
- Reduces the radioactivity level in the room
- Protects the area from hazardous chemical fume intrusion
- Protects the area from noxious fumes, such as smoke from surrounding or outside areas

The required aspects of a control room for nuclear power reactors are established by General Design Criterion (GDC-19), "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50. This addresses ventilation/pressurization, inleakage from smoke or chemicals and radioactivity control to limit exposure to operating personnel required to maintain the plant.

GDC-19 was the original criteria used when plants were designed and used inleakage numbers as low as 10 CFM for ingress and egress by personnel thru doors when entering and exiting the control room. It was generally assumed that a positive pressure control room only leaked out and never in. The general assumption of no in-leakage was disproved with tracer gas tests, proving that it is highly likely that even with positive pressure in the control room you will have in-leakage from adjacent areas of lower pressure.

Since around 1980, the NRC staff and industry have been working on concerns with control room habitability. Control room habitability is the required capability of the control room to be safe for humans to remain there and work during both normal operations and accident conditions. This capability includes protection from the effects of radiation, hazardous chemicals, fire and smoke, as well as having breathable air, heating and cooling, and other facilities for human comfort. In its review of license amendment submittals over the past several years, the staff has identified numerous problems associated with the assessment of control room habitability. These problems have included the overall soundness of the control room envelope (the plant area that includes the main control room and other rooms and areas to which the operators must go to control the plant during an accident) and the manner in which licensees have demonstrated the ability of their control room designs to meet GDC-19.

This section of the report evaluates the impact of recent regulatory requirements related to pressurization of the control room to prevent CO2 in-leakage from the fire suppression system (e.g., in the cable spreading room.) Industry experience documents, the regulatory requirement, industry implementation plans, and known cautions and challenges were used during the evaluation.

Regulatory Requirement

Since 1998, the staff has been working with industry through the Nuclear Energy Institute (NEI) and the Nuclear Heating,.Ventilation, and Air Conditioning Users Group to develop guidance to address control room habitability. Also, during this time, industry and NEI have developed an industry guidance document NEI 99-03, "Control Room Habitability Assessment Guidance," which was issued in June 2001. Although the staff agreed in concept with NEI 99-03, because of several policy and technical issues, such as frequency and method of testing, it could not endorse it fully. In May and June 2003, final regulatory guides were issued to provide guidance on:

- Control room envelope habitability (RG 1.196)
- Integrity testing (RG 1.197)
- Atmospheric dispersion (RG 1.194)
- Radiological dose assessment with the traditional source term (RG 1.195).

Also, two other regulatory guides offer guidance on control room habitability assessment:

- RG 1.183 on radiological dose assessment with an alternative source term
- RG 1.78 on control room habitability during a hazardous chemical release.

The staff also developed generic letter, GL 2003-01, "Control Room Habitability," that was issued on June 12, 2003. This letter requests that licensees provide confirmation that their facility's control room meets the applicable regulatory requirements and that control room habitability systems are designed, constructed, configured, operated, and maintained according to the facility's design and licensing bases.

This combined affect of these regulatory changes has resulted in utilities performing bases documentation reviews and development, walk-downs to establish actual configuration and material condition for the affected systems, review of radiological dose assessment and accident analyses, development of qualitative smoke assessments, development of a current toxic gas analysis, integration of tracer gas test vendor's procedures to meet specific plant procedural requirements for test execution, and development of Control Room Envelope Integrity Program (CRIP) to satisfy the regulatory requirements.

The following list summarizes most of the applicable regulations:

GL 2003-01, Control Room Habitability

RG 1.194, Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants

RG 1.195, Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors

RG 1.196, Control Room Habitability at Light-Water Nuclear Power Reactors

RG 1.197, Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors

RG 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors

RG 1.78, Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release

Testing Requirements

The testing is treated as a self assessment and must encompass the control room envelope (total volume). Each licensee must demonstrate that they have performed a comprehensive test and have identified and repaired any source of unfiltered in-leakage.

There are several test methods; the ASTM 741 is the preferred and most prevalent. An alternate method is developing a site specific correlation to the preferred (741) method. The regulatory requirements are to perform a baseline inspection, then retest within 6 years of that successful test. Following the first successful baseline test, periodic pressure validation tests are used to verify the control room envelope boundaries have not degraded. This is a performance based testing program. If a test fails, the subsequent test must be a full system test. If that test is successful, the plant returns to a 6 year full test frequency.

Plant Experience

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A
ABBREVIATIONS AND ACRONYMS

ALARA	As low as is reasonably achievable
ANO	Arkansas Nuclear One
BWR	Boiling water reactor
CVCS	Chemical and volume control system
DECON	Decontamination
DR	Dose rate
ED	Electronic dosimetry
EPRI	Electric power research institute
GTCC	Greater than class c (waste)
I&C	Instrumentation and controls
INPO	Institute of nuclear power operations
IT	Information technology
LAN	Local area network
NPV	Net present value
NRC	Nuclear regulatory commission
NRR	Nuclear reactor regulation
PC	Personal computer
PDA	Personal digital assistant
PWR	Pressurized water reactor
RAM	Radioactive material
RCA	Radiologically controlled area
RF	Radio frequency
RFID	Radio frequency identification
RFO	Refueling outage
RG	Regulatory guide
RHR	Residual heat removal
RMT	Remote monitoring technology
RP	Radiation protection
RPV	Reactor pressure vessel

RWP	Radiation work permit
RX	Reactor
SG	Steam generator
VOIP	Voice over internet protocol
VR	Volume reduction

B
REFERENCES

RMT

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Cobalt Source Term Reduction

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PWR Primary Coolant Chemistry

Fuel Integrity Monitoring and Failure Evaluation Handbook. EPRI, Palo Alto, CA: 1998. TR-108779.

PWR Shutdown Chemistry Practices, EPRI, Palo Alto, CA: 1998. TR-109569.

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C
GENERIC LETTER 2003-01, CONTROL ROOM HABITABILITY

OMB Control No.: 3150-0011

UNITED STATES NUCLEAR REGULATORY COMMISSION OFFICE OF NUCLEAR REACTOR REGULATION WASHINGTON, DC 20555-0001

June 12, 2003

NRC GENERIC LETTER 2003-01: CONTROL ROOM HABITABILITY

Addressees

All holders of operating licenses for pressurized-water reactors (PWRs) and boiling-water reactors (BWRs), except those who have permanently ceased operations and have certified that fuel has been permanently removed from the reactor vessel and more than 1 year has elapsed since fuel was irradiated in the reactor vessel.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this generic letter to:

(1) alert addressees to findings at U.S. power reactor facilities suggesting that the control room licensing and design bases, and applicable regulatory requirements (see section below) may not be met, and that existing technical specification surveillance requirements (SRs) may not be adequate,
(2) emphasize the importance of reliable, comprehensive surveillance testing to verify control room habitability,
(3) request addressees to submit information that demonstrates that the control room at each of their respective facilities complies with the current licensing and design bases, and applicable regulatory requirements, and that suitable design, maintenance and testing control measures are in place for maintaining this compliance, and
(4) collect the requested information to determine if additional regulatory action is required.

Background

The control room is the plant area, defined in the facility licensing basis, from which actions are taken to operate the plant safely under normal conditions and to maintain the reactor in a safe condition during accident situations. For most facilities, the habitability criteria of General Design Criterion 19 (GDC 19) in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," apply to this area. The control room envelope (CRE) is the plant area, defined in the facility licensing basis, that encompasses the control room and may encompass other plant areas. The structures that make up the CRE are designed to limit the inleakage of radioactive and hazardous materials from areas external to the CRE. Control room habitability

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systems (CRHSs) typically provide the functions of shielding, isolation, pressurization, heating, ventilation, air conditioning and filtration, monitoring, and the sustenance and sanitation necessary to ensure that the control room operators can remain in the control room and take actions to operate the plant under normal and accident conditions. The personnel protection features incorporated into the design of a particular plant's CRHSs depend on the nature and scope of the plant-specific challenges to maintaining control room habitability. In the majority of the CRHS designs, isolation of the normal supply and exhaust flow paths and pressurization of the CRE relative to adjacent areas are fundamental to ensuring a habitable control room.

During the design of a nuclear power plant, licensees perform analyses to demonstrate that the CRHSs, as designed, provide a habitable environment during postulated design basis events. These design analyses model the transport of potential contaminants into the CRE and their removal. The amount of inleakage of assumed contaminants is important to these analyses. Unaccounted-for contaminants entering the CRE may impact the ability of the operators to perform plant control functions. If contaminants impair the response of the operators to an accident, there could be increased consequences to the public health and safety.

There are two typical CRE designs. These designs are referred to as positive-pressure and neutral-pressure CREs. Both designs focus on limiting the amount of contaminants entering the CRE. For radiological challenges, the positive-pressure CRE intentionally pressurizes the CRE with air from outside the CRE. The pressurization air is treated by a high-efficiency particulate air filter and iodine adsorption media to remove contaminants. The neutral-pressure CRE does not intentionally pressurize the CRE, but limits inleakage of contaminants by isolating controlled flow paths into the CRE. Most plants with a positive-pressure CRE have a technical specification SR to verify that those ventilation systems serving the CRE can maintain the CRE at a positive differential pressure relative to adjacent areas. These surveillance tests (typically referred to as a ΔP surveillance) are generally implemented through a technical specification SR for the CRHSs. Plants with a neutral-pressure CRE design typically do not have a CRE integrity testing program. (The term "neutral-pressure" means only that the CRE is not intentionally pressured. The actual pressure of the CRE may be positive, neutral, or negative relative to adjacent areas.)

In addition to the ΔP surveillance described above, licensees have performed CRE integrity testing at approximately 30 percent of the power reactor facilities using the standard test method described in American Society for Testing and Materials (ASTM) consensus standard E741, "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution." Unlike the ΔP surveillance, the ASTM E741 test determines the total CRE inleakage from all sources. It is well suited for assessing the integrity of positive-pressure or neutral-pressure CREs. The test basically involves homogeneously dispersing a nontoxic tracer gas throughout the CRE and measuring the dilution of the tracer gas caused by inleakage.

The results of the ASTM E741 tests indicate that the ΔP surveillance is not a reliable method for demonstrating CRE integrity. For all but one facility tested using the ASTM E741 standard, the measured inleakage was greater than the inleakage assumed in the design basis analyses. In some cases, even though the licensees had routinely demonstrated a positive ΔP relative to adjacent areas at their facilities, the measured inleakage was several orders of magnitude greater than the value previously assumed. Affected facilities were subsequently able to achieve compliance with the control room radiation protection regulatory requirements by sealing, adding new ductwork, changing their CRE, or reanalyzing their control room habitability.

Use of the ΔP surveillance as an indicator of CRE integrity has two inherent deficiencies. First, it does not measure CRE inleakage. The ΔP surveillance infers that no contamination can enter the CRE if the CRE is at a higher pressure than adjacent areas. Second, the ΔP surveillance cannot determine whether there may be unrecognized sources of pressurization of the CRE that could introduce contaminants into the CRE under accident conditions. Two possible unrecognized contamination pathways are the CRHS fan suction ductwork that is located outside the CRE, and the pressurized ducts that traverse the lower pressure CRE en route to another plant area.

The ASTM E741 testing has helped to identify a spectrum of CRHS deficiencies that affect (1) system design, construction, and quality, (2) system boundary construction and integrity, and (3) technical specification SRs. Licensees have determined that the performance of the CRHSs can be affected by (1) the gradual degradation in associated equipment such as seals, floor drain traps, fans, ductwork, and other components, (2) the drift of throttled dampers, (3) maintenance on the CRHSs, and (4) inadvertent misalignments of the CRHSs. Since inleakage is influenced by pressure differentials between the CRE and adjacent areas, changes in ambient pressure in these adjacent areas can affect the CRE inleakage. These changes can be the result of a modification, the degradation of the ventilation systems serving these areas, or inadequate preventive and corrective maintenance programs.

Licensees and NRC staff have identified other deficiencies in CRHS design, operation, and performance from the review of license amendments, licensee event reports, and records and reports prepared pursuant to 10 CFR 50.59. These deficiencies showed that the licensees' CRHSs did not meet their design bases. Some of these deficiencies are discussed in Regulatory Issue Summary 2001-19, "Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests." For example, some licensees credited the operation of CRHSs based upon actuation of high-radiation signals from instrumentation. Further investigation revealed that for some licensees the system would not be actuated due to incorrect setpoints or placement of the instrumentation. Other CRHS designs appear not to have considered unfiltered or once-filtered inleakage through idle CRHS ventilation trains. Without adequate consideration of such design issues, design basis radiation exposure limits may be exceeded.

Previous to the ASTM E741 testing, a group of licensees had trouble meeting the control room criteria in Three Mile Island (TMI) Action Item III.D.3.4, "Control Room Habitability Requirements," that the NRC ordered most licensees to implement after the accident at TMI. At that time, radiological source term research suggested that the distribution of the chemical forms of iodine released during an accident could be different from the distribution in the traditional source term defined in U.S. Atomic Energy Commission Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites." Because of the possible differences, the staff allowed licensees to postpone changing their control rooms until the ongoing source term research was completed or until a generic letter on control room habitability was issued. The staff believed that postponing changes was reasonable since the source term research or improved methods of analyses might prove that the changes were unnecessary. Many of these licensees that postponed changes incorporated compensatory actions into their operating procedures to assure that the control room operators would be protected in case of an accident. Since then, some licensees have found that they could not meet the thyroid dose limits for habitability without using compensatory actions. The NRC also allowed these facilities to use compensatory actions until completion of the source term research. In August 2000, the NRC staff incorporated the results of the source term research into Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," which is now available for use by licensees.

Although many CRE integrity testing programs focus on radiological concerns, radiation is only one potential design basis challenge to the protection of the operators. The inleakage of other contaminants may have a greater impact on control room habitability. An inleakage rate that is tolerable for one contaminant may not be tolerable for another. The control room licensing basis describes the hazardous chemical releases considered in the CRE design, the design features, and the administrative controls implemented to mitigate the consequences of these releases to the control room operators. Smoke and other byproducts of fire within the CRE or in adjacent areas are among the contaminants that can have an adverse impact on control room habitability.

Discussion

Information obtained by the NRC indicates that some licensees have not maintained adequate configuration control over their CREs and have not corrected identified design and performance deficiencies. The primary design function of CRHSs is to provide a safe environment in which the operator can control the nuclear reactor and auxiliary systems during normal operations and can safely shut down these systems during abnormal situations to protect the health and safety of the public. It is important for the operators to be confident of their safety in the control room to minimize errors of omission and commission. Errors of omission and commission are more likely if CRHSs do not properly perform as intended in response to challenges from off-normal or accident situations. The control room must be safe so that operators can remain in the control room to monitor plant performance and take appropriate mitigative actions. This is an underlying assumption in both the design basis and severe accident risk analyses. It is, therefore, imperative to the health and safety of the public that operators are safe in the control room at all times.

The scope and magnitude of the problems that NRC staff and certain licensees have identified raise concerns about whether similar design, configuration, and operability problems exist at other reactor facilities. The NRC staff is particularly concerned about whether licensees' programs to maintain configuration control of CRHSs are sufficient to demonstrate that the physical and functional characteristics of CRHSs are consistent with and are being maintained according to their design bases. It is emphasized that the NRC's position has been, and continues to be, that it is the responsibility of individual licensees to know the licensing basis for the CRHSs. Licensees should also have appropriate documentation of the design basis and procedures in place, in accordance with NRC regulations, for performing necessary assessments of plant or procedure changes that may affect the performance of the CRHSs. The technical specifications for about 75 percent of the control rooms (mostly positive-pressure CREs) have an SR to measure the ΔP from the CRE to adjacent areas. The bases of the Improved Standard Technical Specifications state that this SR demonstrates control room integrity with respect to unfiltered inleakage. The ASTM E741 integrated testing proves that it does not. Because 10 CFR 50.36 requires technical specifications to be derived from the safety analyses, the staff believes that the existing deficiency should be corrected. This correction is consistent with NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications That Are Insufficient To Assure Plant Safety," which describes the staff's expectation that licensees correct technical specifications that are found to "contain non-conservative values or specify incorrect actions."

Because of the importance of ensuring habitable control rooms under all normal and off-normal plant conditions, the addressees are requested to provide certain information that will enable the NRC staff to verify whether addressees can demonstrate and maintain the current design bases for the CRHSs at their facilities. Addressees are encouraged, but not required, to work closely with industry groups on the coordination of their responses. Coordinating the responses promotes efficiency since it leads to a uniform approach to demonstrating compliance with the design bases of their CREs.

NEI 99-03, "Control Room Habitability Assessment Guidance," provides industry generic guidance on control room habitability. The NRC staff reviewed NEI 99-03, but rather than fully endorse NEI 99-03, the NRC staff developed its own guidance. Regulatory Guide 1.196 (formerly DG-1114), "Control Room Habitability at Light-Water Nuclear Power Reactors," endorses NEI 99-03 to the extent possible and provides additional guidance. Licensees are not required to comply with Regulatory Guide 1.196, but may find it useful in responding to this generic letter. Licensees that are unable to confirm item 1 under the Requested Information section may use Regulatory Guide 1.196 to develop and implement corrective actions.

<u>Requested Information</u>

Addressees are requested to provide the following information within 180 days of the date of this generic letter.

1. Provide confirmation that your facility's control room meets the applicable habitability regulatory requirements (e.g., GDC 1, 3, 4, 5, and 19) and that the CRHSs are designed, constructed, configured, operated, and maintained in accordance with the facility's design and licensing bases. Emphasis should be placed on confirming:

(a) That the most limiting unfiltered inleakage into your CRE (and the filtered inleakage if applicable) is no more than the value assumed in your design basis radiological analyses for control room habitability. Describe how and when you performed the analyses, tests, and measurements for this confirmation.
(b) That the most limiting unfiltered inleakage into your CRE is incorporated into your hazardous chemical assessments. This inleakage may differ from the value assumed in your design basis radiological analyses. Also, confirm that the reactor control capability is maintained from either the control room or the alternate shutdown panel in the event of smoke.
(c) That your technical specifications verify the integrity of the CRE, and the assumed inleakage rates of potentially contaminated air. If you currently have a ΔP surveillance requirement to demonstrate CRE integrity, provide the basis for your conclusion that it remains adequate to demonstrate CRE integrity in light of the ASTM E741 testing results. If you conclude that your ΔP surveillance requirement is no longer adequate, provide a schedule for: 1) revising the surveillance requirement in your technical specification to reference an acceptable surveillance methodology (e.g., ASTM E741), and 2) making any necessary modifications to your CRE so that compliance with your new surveillance requirement can be demonstrated.

If your facility does not currently have a technical specification surveillance requirement for your CRE integrity, explain how and at what frequency you confirm your CRE integrity and why this is adequate to demonstrate CRE integrity.

2. If you currently use compensatory measures to demonstrate control room habitability, describe the compensatory measures at your facility and the corrective actions needed to retire these compensatory measures.
3. If you believe that your facility is not required to meet either the GDC, the draft GDC, or the "Principal Design Criteria" regarding control room habitability, in addition to responding to 1 and 2 above, provide documentation (e.g., Preliminary Safety Analysis Report, Final Safety Analysis Report sections, or correspondence) of the basis for this conclusion and identify your actual requirements.

Requested Response

If an addressee cannot provide the information or cannot meet the requested completion date, the addressee should submit a written response indicating this within 60 days of the date of this generic letter. The response should address any alternative course of action the addressee proposes to take, including the basis for the acceptability of the proposed alternative course of action and the schedule for completing the alternative course of action.

The written response should be addressed to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001. A copy of the response should be sent to the appropriate regional administrator.

NRC staff will review the responses to this generic letter and, if concerns are identified, will notify affected addressees. The staff may conduct inspections to determine licensees' effectiveness in addressing this generic letter.

Applicable Regulatory Requirements

Several provisions of the NRC regulations and plant operating licenses (technical specifications) pertain to the issue of control room habitability. The general design criteria for nuclear power plants (10 CFR Part 50, Appendix A), or, as appropriate, the quality assurance requirements in the licensing basis for a reactor facility (stated in 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"), and the technical specifications, are the bases for the NRC staff's assessment of control room habitability.

Appendix A to 10 CFR Part 50 and the plant safety analyses require or commit licensees to design and test safety-related structures, systems, and components (SSCs) to provide adequate assurance that they can perform their safety functions. The NRC staff applies these criteria to plants with construction permits issued on or after May 21, 1971, and to those plants whose licensees have committed to them. The applicable GDC are GDC 1, 3, 4, 5, and 19. GDC 1 requires quality standards commensurate with the importance of the safety functions performed. GDC 3 requires SSCs to be designed and located to minimize the effects of fires. GDC 4 requires SSCs to be designed to accommodate the effects of accidents. GDC 5 requires that an accident in one unit will not significantly impair orderly shutdown and cooldown of the remaining unit.

GDC 19 specifies that a control room be provided from which actions can be taken to operate the nuclear reactor safely under normal conditions and maintain the reactor in a safe condition under accident conditions, including a loss-of-coolant accident. There must be adequate radiation protection to permit personnel to access and occupy the control room under accident conditions without receiving radiation exposures in excess of specified values.

Before the issuance of the GDC, proposed GDC (sometimes called "principal design criteria") were published in the *Federal Register* for comment. As they evolved, several of the proposed GDC addressed control room habitability. A facility may have been licensed before the issuance of the GDC, but the licensee may have committed to the proposed GDC as they existed at the time of licensing.

Following the accident at TMI, TMI Action Plan Item III.D.3.4, "Control Room Habitability Requirements," as clarified in NUREG-0737, "Clarification of TMI Action Plan Requirements," required all licensees to assure that control room operators would be adequately protected against the effects of accidental releases of toxic and radioactive gases and that the nuclear power plant could be safely operated or shut down under design basis accident conditions. When licensees proposed modifications, the NRC issued orders confirming the licensees' commitments. As a result, most plants licensed before the GDC were formally adopted were then subsequently required to meet the TMI Action Plan III.D.3.4 requirements.

Appendix B to 10 CFR Part 50 establishes quality assurance requirements for the design, construction, and operation of those SSCs that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. Appendix B, Criterion III, "Design Control," requires that design control measures be provided for verifying or checking the adequacy of design. A suitable testing program is identified as one method of accomplishing this verification. Appendix B, Criterion XVI, "Corrective Action," requires measures to be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, defective material and equipment, and nonconformances are promptly identified and corrected.

The regulations in 10 CFR 50.36, "Technical Specifications," require plant technical specifications to be derived from the safety analyses.
If, in the course of preparing a response to the requested information, an addressee determines that its facility is not in compliance with the Commission's requirements, the addressee is expected to take appropriate action in accordance with requirements of Appendix B to 10 CFR Part 50 and the plant technical specifications to restore the facility to compliance.

<u>Reasons for Information Request</u>

This generic letter transmits an information request that is necessary to permit the assessment of plant-specific compliance with applicable regulatory requirements. Specifically, this information will enable the NRC staff to determine whether the control rooms at power reactor facilities comply with the current licensing bases and whether additional regulatory actions are required.

The habitability of the control room and the operability of the CRHSs in the event of adverse environmental conditions external to the CRE have a direct link to maintaining public health and safety. Plant design bases and severe accident risk analyses both assume that the control room operators can remain safely within the control room to monitor plant performance and take appropriate mitigative actions. It is essential that operators be confident of their safety within the control room at all times.

Backfit Discussion

This generic letter transmits an information request for the purpose of verifying compliance with existing applicable regulatory requirements (see the Applicable Regulatory Requirements section of this generic letter). This generic letter does not constitute a backfit as defined in 10 CFR 50.109(a)(1) since it does not impose modifications or additions to structures, systems, and components or to the design or operation of an addressee's facility. Nor does it impose an interpretation of the Commission's rules that is either new or different from a previous staff position. Therefore, no backfit is either intended or approved by this generic letter, and the staff has not performed a backfit analysis.

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