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Application of the EPRI Standard Radiation Monitoring Program for PWR Radiation Field Reduction

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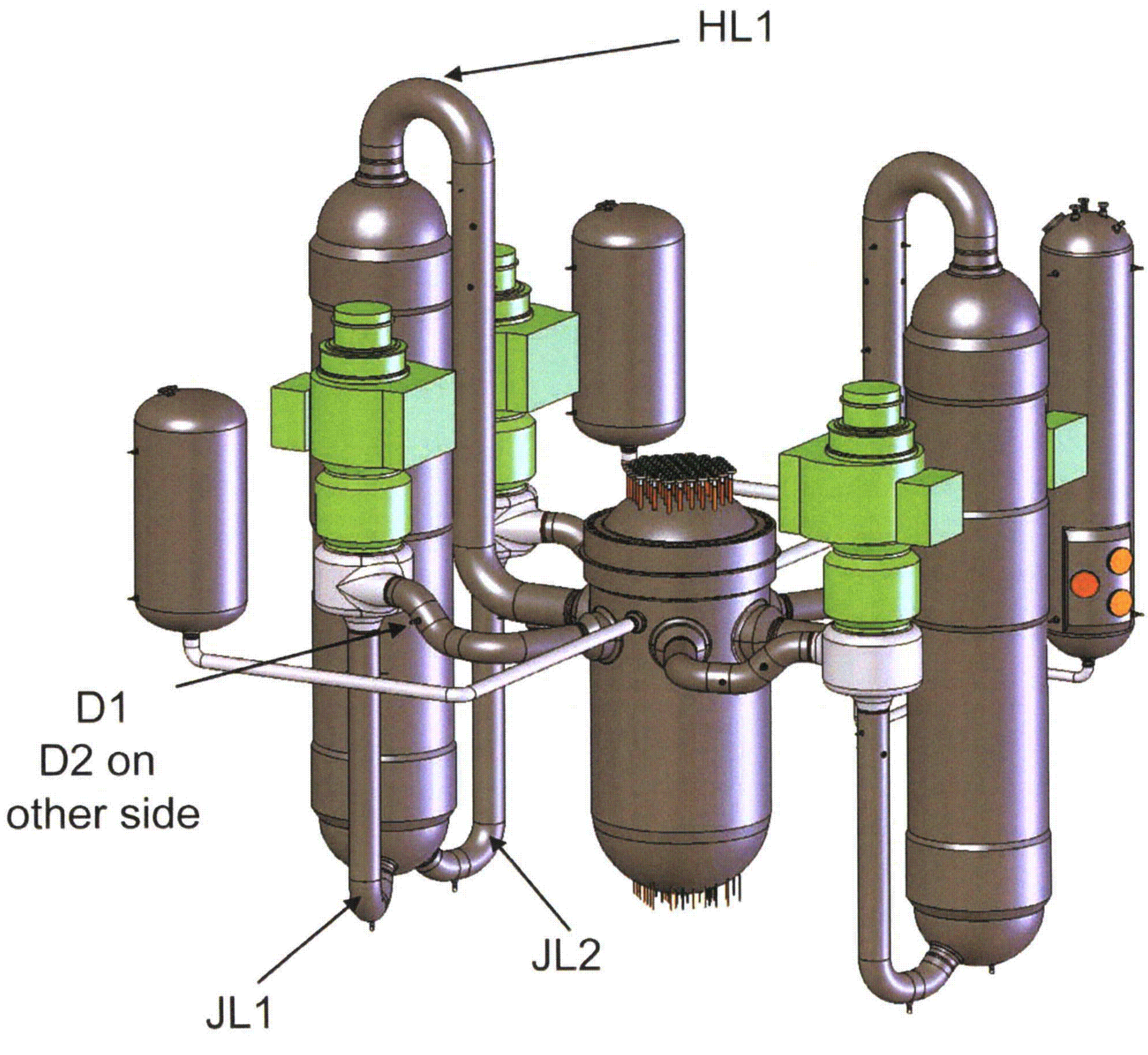

Application of the EPRI Standard Radiation Monitoring Program for PWR Radiation Field Reduction

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REPORT SUMMARY

The NEI/INPO/EPRI RP 2020 Initiative was developed to promote radiation dose reduction by emphasizing radiological protection fundamentals and reducing radioactive source term. EPRI was charged as the technical lead in the area of source term reduction. EPRI's Radiation Management program initiated a multi-year program to develop an understanding of source term generation and transport with the eventual goal of providing plant specific recommendations for source term reduction. Reinstatement of the Standard Radiation Monitoring Program (SRMP) is one of the major program components of the EPRI Source Term Reduction Program.

Background
Originally started in 1978, the EPRI Standard Radiation Monitoring Program was discontinued in 1996 due to lack of funding. Recent interest in reducing radiation fields has prompted the EPRI Technical Advisory Committee to recommend the reinstatement of the program. Since 2005, EPRI has worked with the industry to revise the data procedures, develop the data collection mechanism, and integrate the data with the PWR Chemistry Monitoring and Assessment Database. This report presents the results of this three-year effort.

Objectives
- To inform radiation protection and chemistry staff, engineers and managers about the status of the radiation fields in PWRs
- To describe the revisions and updates to the Standard Radiation Monitoring Program
- To provide a theoretical basis for activity generation and incorporation into the ex-core surfaces
- To perform a plant benchmarking comparison of most-recent available cycle radiation fields
- To perform an analysis of selected plant data that highlight effects of plant changes on radiation field behavior
- To summarize current work and offer recommendations for program improvement
- To provide graphical data for reporting plants

Approach
The project team narrated the history of the Standard Radiation Monitoring Program and discussed changes in the industry that impact the radiological behavior of the plants. The team described the theoretical causes for radiation fields and offered evidence to support that description in the results sections.

Results

The EPRI Standard Radiation Monitoring Program has been successfully re-instated with active participation from the utilities. Participation is not 100%, but contacts have been developed with all plants to begin data collection.

In conjunction with the PWR Monitoring and Assessment database, the project developed a relational database linking steam generator design, chemistry, and the SRMP radiation data. The result is a powerful tool to understand causal links between plant changes and radiation fields. Reviewing the radiological history of a single unit provides significant insight into cause-and-effect relationships of chemistry, operations, and plant design to radiation fields. Plant-to-plant benchmarking comparisons are useful to identify strong performers in radiation field reduction; but they may offer misleading indications about the state of a plants source term reduction program because of interference sources, measurement technique differences, and operator training.

The data clearly show zinc injection and steam generator replacement with Alloy 690 tubing have significant impacts on ex-core radiation fields. Electropolishing steam generator channel heads is an effective method to reduce channel head radiation fields after replacement. Interestingly, there is little to no correlation of radiation fields with PWR coolant radiocobalt concentrations or forced oxidation peaks.

Diablo Canyon gamma spectroscopy data indicate a strong need for inclusion of gamma spectroscopy for radiation benchmarking; the current total gamma fields offer valuable insights but do not provide information about nuclide distribution.

EPRI Perspective

The EPRI PWR Source Term Reduction Program is a multi-year program that seeks to first understand the causes of radioactivity generation and transport and then, through data analysis, operations review, and technology evaluations, provide utilities with plant specific recommendations for source term reduction. The program began with the publication Dose Rate Impacts of Activity Transport in Primary Coolant Systems in 2005 (EPRI report 1011736) and continued with Source Term Reduction: Impact of Plant Design and Chemistry on PWR Shutdown Releases and Dose Rates (EPRI report 1013507). It will continue through 2008 with the further data collection in the Standard Radiation Monitoring Program and PWR Chemistry Monitoring and Assessment database for the development of plant-specific source term reduction recommendations.

Keywords

Shutdown dose rates
Standard Radiation Monitoring Program
Activity release
Source term reduction
Gamma spectroscopy

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1 INTRODUCTION

The buildup of radiation fields on plant primary and secondary components has a direct impact on maintenance, repair, and in-service inspection operations at nuclear power plants. The reduction of these fields can be a significant step toward lower operational and maintenance costs at such plants. To support the radiation control program, a broad-based set of radiation field data is required.

The Standard Radiation Monitoring Program (SRMP), sponsored by EPRI, was first instituted in 1978, as part of a more general program with the major emphasis on improving plant reliability and availability. The objectives of this program in 1978 were as follows:

- To provide a meaningful, consistent, and systematic approach to monitoring the rate of PWR radiation field buildup and to provide the basis for projecting the trend of those fields.
- To provide a reliable set of radiation field data for each participating plant, from which comparisons can be made.
- To monitor certain plant parameters that affect or may affect observed radiation fields.
- To use the information from this program to identify plant design features, material selection, and operational techniques that present opportunities for radiation control.

The objectives of the reinstatement of the SRMP have not changed. Previous EPRI reports published as a result of the SRMP program list the factors that affect plant dose rates and quantitatively evaluate the effect of these factors [1, 2, 5]. The most important factors were found at that time to be operational coolant chemistry and variations in cobalt input based on Inconel fuel grids.

The SRMP program had consistent data collection efforts for Westinghouse and Combustion Engineering plants through 1985 and 1996, respectively. Afterwards, SRMP data collection had been limited primarily to plants that had implemented elevated primary coolant pH, zinc injection or replaced steam generators with Alloy 690 tubing.

In 2005, adverse industry trends in Radiation Protection were a key factor in the development of the NEI/EPRI/INPO RP 2020 Initiative, which had the stated goal of 'Taking Radiation Off The Table.' EPRI was charged with taking the technical lead for Radiation Source Term Reduction. In response to this initiative, the EPRI Chemistry and LLW Technical Advisory Committee strongly recommended that EPRI restart PWR radiation field data collection efforts, known as the Standard Radiation Monitoring Program, SRMP to help quantify the effects of plant changes such as replacement steam generators, core uprating, adverse radiological incidents, and various changes in shutdown and normal chemistry procedures. These changes have caused unpredictable fluctuations in dose rates throughout the ex-core surfaces, and a more fundamental understanding is required.

The reinstatement of the SRMP includes:

- A revision of the SRMP General Survey Procedure, which includes the development of a procedure for Babcock and Wilcox plants.
- Collection of the available industry data and re-establishing the data collection mechanism from each operating PWR.
- An update of the radiation field analysis to include recent changes in plant design operation and operation.

In addition, the interactions of the procedure has been streamlined by defining survey locations as 'Required Points,' 'Recommended Points,' and 'Optional Information.' The definitions of these terms are below:

- Required points are those that must be taken.
- Recommended points are those that are requested, but may be skipped in cases of personnel safety, poor accessibility, or significant ALARA impact.
- Optional information is information that is requested if available.

To avoid confusion because of the differences between plants, this General Procedure has been revised to have separate procedures for Westinghouse, Combustion Engineering, and Babcock and Wilcox plant designs. Copies of the updated procedures are in Appendices C, D, and E.

Previously, the SRMP collected shutdown radiation field surveys at locations in contact with the reactor coolant loop components as well as primary chemistry data. To relieve the burden on health physics staff, the chemistry data are no longer requested. Instead, these data are collected by the PWR Chemistry Monitoring and Assessment Program.

To date, 58 plants have contributed data to the SRMP. These plants and their data submission summaries are listed in Table 1-1. The table lists the cycles for which there are any data entries, it does not imply that the complete survey was submitted.

There are some instances where the plant had submitted data after the report queries were developed; these will be added to additional future SRMP reports. Utilities are requested to review the data in this section and the appendices to assess the accuracy the accuracy of the data. The detailed database is not given in this report; each plant will receive their data in separate communications.

Table 1-1
Summary Of Data Submitted To EPRI For This Report

Plant Name	RCS Loop Piping Data	SG Channel Head Data
Arkansas Nuclear One 1	19,20	18,20
Arkansas Nuclear One 2	None	None
Beaver Valley 1	2, 15-17	None
Beaver Valley 2	10-12	3-12
Braidwood 1	None	1-6, 8, 10-12
Braidwood 2	None	1-12
Byron 1	None	3-7, 9-11, 13
Byron 2	None	2-13
Callaway	1-5, 7-15	1-13, 15
Calvert Cliffs 1	None	13,14
Calvert Cliffs 2	None	10,12,13
Catawba 1	15,16	14,15
Catawba 2	14	11-14
Comanche Peak 1	3-12	3-11
Comanche Peak 2	1-9	1-8
DC Cook 1	15, 18-20	15, 18, 19
DC Cook 2	11,13-15	11,12, 14,15
Crystal River 3	None	8-14
Davis Besse	6-14	6-14
Diablo Canyon 1	8-14	8-14
Diablo Canyon 2	7-13	7-13
Farley 1	17-20	17,20
Farley 2	13-18	15,18
Fort Calhoun	23	None
Ginna	None	None
Harris	11,12	1-9, 11,13
Indian Point 2	None	None
Indian Point 3	14	None
Kewaunee	None	None
McGuire 1	16-18	12-14, 16
McGuire 2	15-17	12;13
Millstone 2	13-17	13-16
Millstone 3	6-8, 10, 11	6-11
North Anna 1	7-18	10-15, 17, 18

Plant Name	RCS Loop Piping Data	SG Channel Head Data
North Anna 2	6-9, 11-18	11-15, 18
Oconee 1	17-23	17-20, 22, 23
Oconee 2	15-22	15-19, 21, 22
Oconee 3	16-22	16-20,22
Palisades	14-18	None
Palo Verde 1	None	None
Palo Verde 2	None	None
Palo Verde 3	None	None
Point Beach 1	None	None
Point Beach 2	None	None
Prairie Island 1	16-23	16-21,23
Prairie Island 2	16-20,22-23	17,18, 20-23
Robinson	14,20-24	11,14,17,21,22,24
Salem 1	None	16
Salem 2	None	13,14
Seabrook	4,5	4,5
Sequoyah 1	None	None
Sequoyah 2	None	None
San Onofre 2	10-13	2-14
San Onofre 3	10-13	2-14
St. Lucie 1	None	None
St. Lucie 2	None	None
South Texas Project 1	2-9, 11-13	1,3-8,12,13
South Texas Project 2	1-12	1-8,10
Summer	None	9-12, 15
Surry 1	13-20	13-20
Surry 2	13, 15-21	13-21
Three Mile Island-1	11, 13-15	11-15
Turkey Point 3	8-13	12-20
Turkey Point 4	7,9-13	13-16,18,20
Vogtle 1	None	None
Vogtle 2	None	None
Waterford 3	None	8-13
Watts Bar	1,2	1,2
Wolf Creek	13-15	13-15

Graphical compilations of the data can be found in Appendix A and Appendix B.

International Radiation Units

The following conversions will be useful for international members to convert radiation fields and activity measurements to SI units.

Table 1-2
Radiation Conversion Factors

To convert from	to	Multiply by
Ci (Curies)	Becquerel (Bq)	3.7E10
Bq (Becquerel)	Disintegration/second	1
μCi/ml	Bq/ml	3.7E4
μCi/ml	Bq/liter	3.7E7
μCi/ml	MBq/m^3	3.7E4
MBq/m^3	μCi/ml	2.7E-5
Bq/m^2	$\mu Ci/cm^2$	2.70E-9
$\mu Ci/cm^2$	Bq/m^2	3.70E8
Sievert	Rem	1E2
Rem	Sievert	1E-2
R/h	Sievert/h	1E-2
R/h	milliSievert/h	10
mR/h	milliSievert/h	1E-2
Rad	Gray(Gy)	1E-2

2
THEORETICAL BASIS FOR RADIOACTIVITY GENERATION AND DEPOSITION IN EX-CORE SURFACES

Background

This section provides a theoretical basis for the generation and incorporation of activated corrosion products into the ex-core surfaces.

Mechanism for Activity Incorporation into Ex-Core Surfaces

Radiation fields in ex-core surfaces induced by activated corrosion products develop, as a minimum, through a five step process derived from mass transport theory.

1. Parent nuclide (nickel and elemental cobalt) corrodes on the exposed surfaces.
2. The parent nuclide releases from the surface into the coolant. The expected mechanism is soluble (ionic) transport.
3. The coolant transports the parent to the core where it deposits on the fuel. The nuclide will activate on the core. It may also activate while being transported in the coolant, but the time in core will be much less than if deposited.
4. The activated corrosion product is released from the core, again with the expected mechanism of soluble transport.
5. The soluble activated corrosion product is incorporated into the growing oxide film.

The following sub-sections discuss several of the relevant issues related to activity generation and incorporation into the ex-core surfaces.

Corrosion Rates and Film Growth

The first step of activity generation and the last step of activity incorporation relate to the corrosion rate and release rate of the ex-core surfaces. Corrosion rates of PWR materials have been measured in several experiments. Corrosion rates of unfilmed stainless steels and high nickel alloys rapidly decrease with time reaching levels in the range of 1 to 10 mg/dm^2-mo after several months of exposure to PWR primary coolant. They then gradually decrease to 1 to 2 mg/dm^2-mo after about a year [1]. Further decreases are expected as exposure continues and film thicknesses increase, but the decreases are expected to be small, and it is difficult to quantify their extent. Correspondingly, the film thickness on corroding surfaces increases rapidly as the protective film develops. The film thickness continues to increase with time as the corrosion process continues. However, a fraction of the film is released to the coolant thus reducing the residual film thickness. If only minor changes in coolant chemistry occur, the corrosion and corrosion product release rates are expected to remain nearly constant leading to a linear increase of film thickness with time. However, long term data from operating plants are not available to validate this premise.

Corrosion product films on Alloy 600 and Alloy 690 remain relatively thin even after years of operation. For example, the film thickness on Alloy 600 after 5 years of operation at relatively low pH has been reported to be in the range of 3 microns [4]. Based on laboratory corrosion rate comparisons, films on Alloy 690 after 5 years will be much lower (•1 micron) because of its much lower corrosion rate (Table 2-1) [5]. Zinc injection will further reduce corrosion rates of both stainless steels and high nickel alloys (Table 2-2 and Table 2-3) [6].

Table 2-1
Corrosion and Corrosion Release Rates for Inconel 600 and Inconel 690 (5)

Reference	**Material/ Type**	**Corrosion Rate (mg/dm^2-mo)**	**Ratio 600/690**	**Corrosion Release Rate (mg/dm^2-mo)**	**Ratio 600/690**
Westinghouse	600 MA/Tubing	4.5	5.0	1.4	4.1
STR Test(9)	690 TT/Tubing	0.9		0.34	
Westinghouse	600 MA/Tubing	2.6	2.0	0.8	1.3
Zinc Test(10)	690 TT/Tubing	1.3		0.6	
Yamada(11)	600 TT/Tubing	5.1	3.4	1.4	2.8
	690 TT/Tubing	1.5		0.5	
Average			3.5		2.7

a) Releases measured at 2-3 months

Table 2-2 Effects of Zinc Addition on Corrosion and Release Rates From PWR System Materials

Period	Zinc	Corrosion Rate, mg/dm²-mo	Release Rate, mg/dm²-mo	Ratio Zn/No Zn Corrosion Rate	Ratio Zn/No Zn Release Rate
0-500 h 304SS	N	5.94	2.47	0.42	0.26
	Y	2.5	0.63		
500-2500 h 304	N	3.50	1.43	0.24	--
	N	2.56	0.56		
	Y	0.75	0		
0-2500 h 304 SS	N	3.99	1.64	0.30	0.09
	N	3.23	0.94		
	Y	1.10	0.12		
0-2500 h 316 SS	N	3.62	1.44	0.35	0.10
	Y	1.27	0.14		
0-500 h 600 MA	N	6.26	0.69		
0-500 h 600 TT	N	1.47	0	0.32	--
0-500 h 600 TT	Y	0.48	0.15		
0-500 h 690 TT	N	1.19	--	0.62	--
	Y	0.74	--		
0-2500 h 600 MA	N	2.22	0.7	0.58	0.35
	N	3.07	0.96		
	N	3.17	1.05		
	N	2.0l	0.70		
	Y	1.95	0.42		
	Y	1.10	0.19		
0-2500 h 600 TT	N	2.06	0.91	0.26	0.22
	Y	0.53	0.20		
0-2500 h 690 TT	N	1.32	0.63	0.18	--
	Y	0.24	--		
500-1300 h 600	N	5.16	1.82	0.18	0.03
	Y	0.91	0.06		
1300-2500 h 600	N	0.50	0.67	0.60	0.45
	Y	0.30	0.30		
1300-2500 h 690	N	0.74	0.79	0.26	--
	Y	0.19	--		

Some insights into the corrosion and corrosion product release processes can be obtained from the limited available data from operating plants. For example, Farley 2 Alloy 600 steam generator tubing corrosion films were characterized by Auger electron spectroscopy at EOC9 and EOC10 to assess the impact of zinc injection which began in Cycle 10 [5, 7]. Results are shown in Figure 2-1 and Figure 2-2. The EOC 9 sample exhibited a minimal corrosion product film thickness (< 1 micron) after 10.1 EFPY of operation. The film was depleted in nickel and enriched in chrome and iron compared to the base alloy composition. The EOC 10 specimen (11.3 EFPY) showed clear evidence of zinc incorporation into the oxide films and a minimal change in film thickness compared to the pre-zinc injection film. The zinc concentration at the surface at EOC10 was greater than 20% and decreased rapidly with depth into the oxide film. This result was similar to that found for Alloy 600 specimens during laboratory testing [9]. However, the depth of penetration was less for the Farley 2 EOC10 tube. It was hypothesized that this difference reflected the higher laboratory exposure temperature and the thickness of the initial oxide (greater for the Farley 2 tubes) [8].

Note that the film thickness (based on a nickel concentration of 70%) is approximately 500 microns in both cases. This is significantly less than the film thicknesses predicted after 10 years of operation based on extrapolation of laboratory data. This infers a very high corrosion product release rate (on the order of 90% of the corrosion rate) or a much lower corrosion rate than predicted based on the laboratory testing.

Film compositions also may vary and affect releases [10] as alloying elements such as carbon vary in percentage. Each of these effects will require more study to assess their impact in operating plants.

Quantification of the corrosion and corrosion product release processes in the PWR primary system has not been possible to date because of the complexity of the processes and the minimal data available on primary system materials film thickness and composition. However, some general insights can be developed relative to film growth on PWR primary system piping surfaces and activity incorporation rates from available data. For example, assume that the corrosion rate of the primary system materials at the average coolant temperature can be expressed as follows for a pH_T in the range of 6.9 to 7.4:

$$CR = A_1 \exp(-A_2 t) + 1\exp(-B_2 t)$$

Where CR = Corrosion Rate, mg/dm^2-mo (mdm) [1]

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Figure 2-1
Normalized Atomic Percent Of Steam Generator ID Surfaces At Farley 2 Without Zinc (Top) And After Zinc (Bottom) [7]

Content deleted - EPRI Proprietary Information.

Figure 2-2
A (top) Results of Auger Analysis for Farley 2 SG-C Tube R27C54 Following Exposure to Zinc in Cycle 10
B (bottom) Results of Auger Analysis for Farley 2 SG-B Tube R26C46 Removed After Cycle 9 – Not Exposed to Zinc [7]

A_1(mg/dm^2-mo), A_2 (yr^{-1}) and B_2(yr^{-1}) are constants that are adjusted to provide reasonable fits to short and long term corrosion behavior of stainless steels, Alloy 600 and Alloy 690 on exposure to the primary coolant; "t" is time expressed as EFPY. The first term of Equation 1 is used to approximate the corrosion rate variation following initial exposure to the coolant when the rate rapidly decreases with time over approximately the first year of exposure. The second term is used to approximate the corrosion rate variation over extended times when a gradual decrease in rate is expected as the film thickness slowly increases. A reasonable fit to laboratory corrosion data for Alloy 600 and stainless steel without zinc addition is given by the following relations:

Alloy 600 Corrosion Rate, mg/dm^2-mo (mdm)

$$CR_{600} = 5\exp(-6t) + 1\exp(-0.02t)$$

Stainless Steel Corrosion Rate, mg/dm^2-mo (mdm)

$$CR_{SS} = 10\exp(-6t) + 2\exp(-0.02t)$$

Corrosion rates based on these expressions are summarized in Figure 2-7. Constants in the above equation change with zinc addition, but the general trend with time is expected to be reasonably similar.

For Alloy 600, the above relation yields an initial corrosion rate of 6 mdm, and rates of 4, 1.3, 1.0, and 0.9 mdm after 0.1, 0.5, 1 and 5 EFPY, respectively (Figure 2-3). Assuming negligible release from the corroding surface, Alloy 600 surface film weights of approximately 22 and 67 mg/dm^2 are predicted after operating times of 1 and 5 EFPY (Figure 2-4). These values correspond to film thicknesses of approximately 0.8 and 2.4 microns at 50% porosity and a density of 2.8 g/cm^3 (Figure 2-5), and total film weights of 62 and 188 Kg (as metals) for a total primary system surface area of 300,000 ft^2 assuming negligible releases (Figure 2-6).

Predictions for Alloy 600 and stainless steel are summarized in Table 2-4. Film thickness and weight per unit area decrease as the corrosion product release rate increases.

Table 2-3
Effect Of Zinc On Approximate Corrosion And Corrosion Release Rates At 3.5 Months (mg/dm^2-mo [mdm])

Material	Corrosion		Corrosion Release	
	with Zn	w/o Zn	with Zn	w/o Zn
304 SS	1.1	3.5	0.1	1.3
316 SS	1.3	3.5	0.1	1.4
600 MA	1.5	2.6	0.3	0.8
600 TT	0.5	2.1	0.2	0.9
690 TT	0.2	1.3	0.1	0.6
X-750	0.6	2.6	0.2	1.2
STELLITE	0.4	14.7	0.1	12.0

Table 2-4
Summary of Expected Corrosion Behavior of Alloy 600 and Stainless Steel 304 in PWR Primary Coolant at pH_T ~ 7.0

Exposure Time, Years	Total Corrosion mg/dm^2		Corrosion Rate, mg/dm^2-mo		Film Thickness, microns [a]		Total Film Weight, kg as metals[a, b]	
	Alloy 600	SS 304	Alloy 600	SS 304	Alloy 600	SS 304	Alloy 600	SS 304
0.1	-	-	6	12	-	-	-	-
0.5	-	-	4	7.5	-	-	-	-
1	22	45	1.2	2.5	0.8	1.6	62	8
5	67	130	0.9	1.8	2.4	4.8	188	25
20	-	-	0.7	1.3	-	-	-	-

a. No Release, = 2.8 gms/cm^3
b. 300,000 ft^2 Alloy 600; 20,000 ft^2 SS304

Table 2-5
Examples of Steam Generator Replacement Areas

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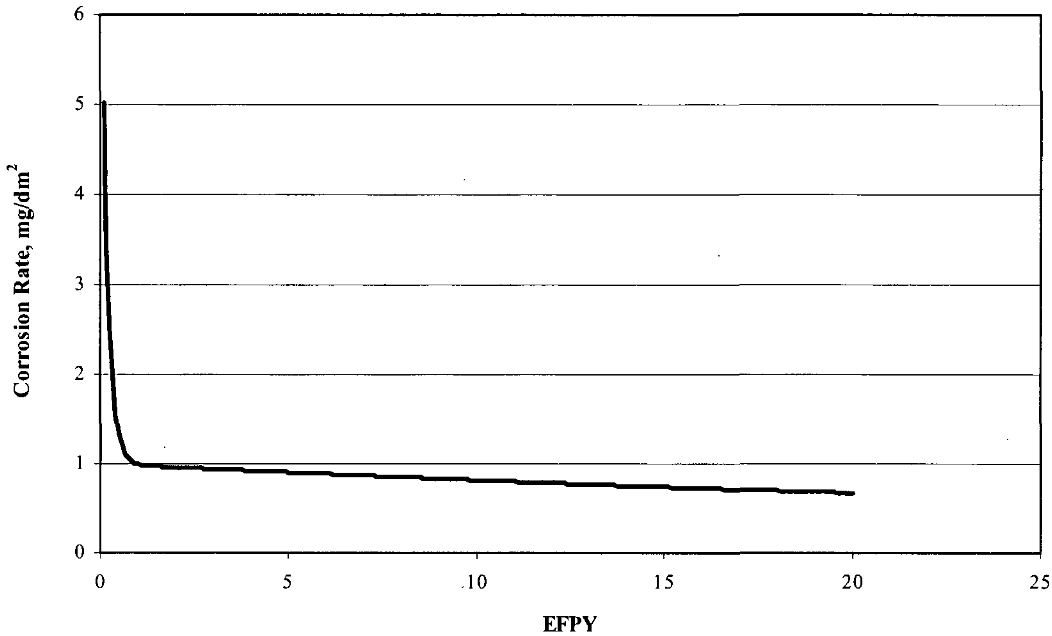

Figure 2-3
Approximate Variation in Alloy 600 Corrosion Rate on Exposure to PWR Primary Coolant (pH = 6.9-7.4; no zinc)

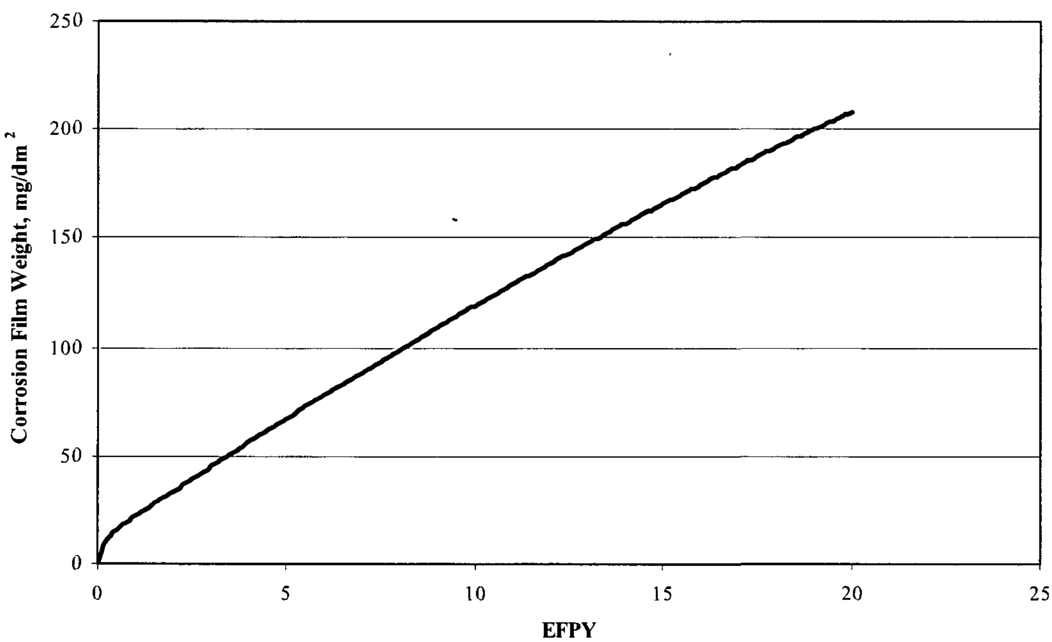

Figure 2-4
Approximate Variation in Alloy 600 Corrosion Film Weight (as metals, no release) on Exposure to PWR Primary Coolant (pH = 6.9-7.4; no zinc)

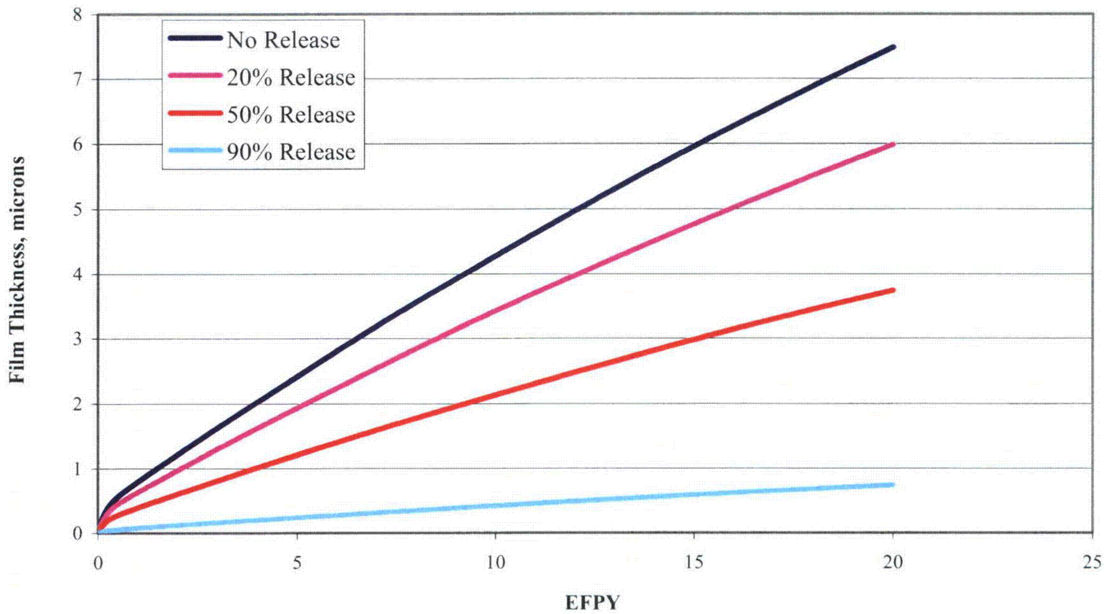

Figure 2-5 Approximate Variation in Alloy 600 Film Thickness on Exposure to PWR Primary Coolant (pH = 6.9-7.4; no zinc)

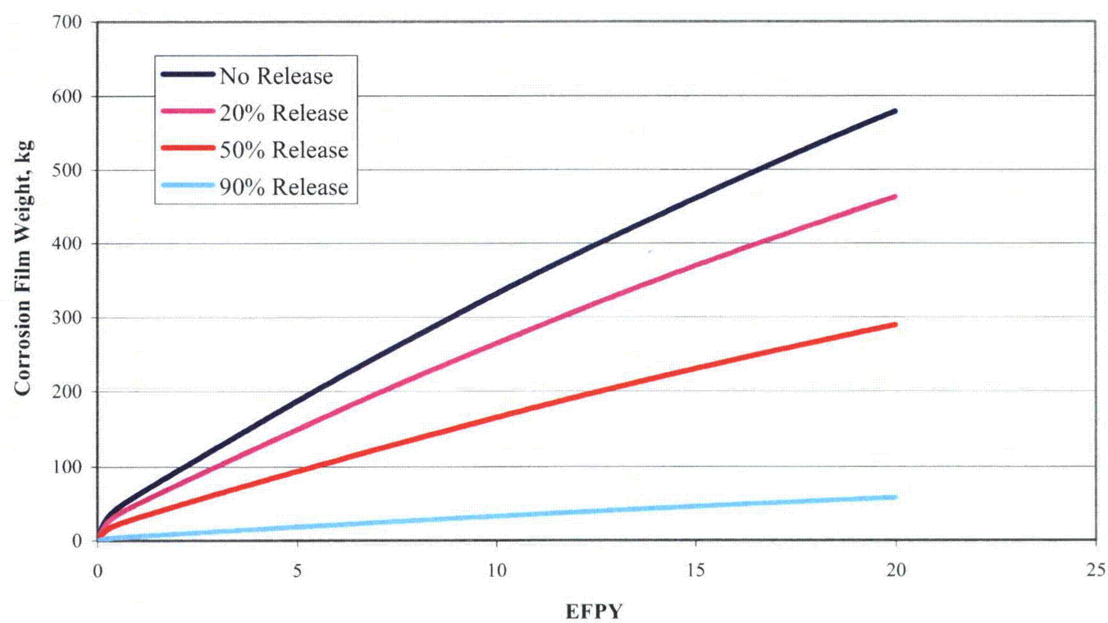

Figure 2-6
Approximate Variation in Primary System Alloy 600 Corrosion Film Weight (as metals) on Exposure to PWR Primary Coolant (pH = 6.9-7.4; no zinc)

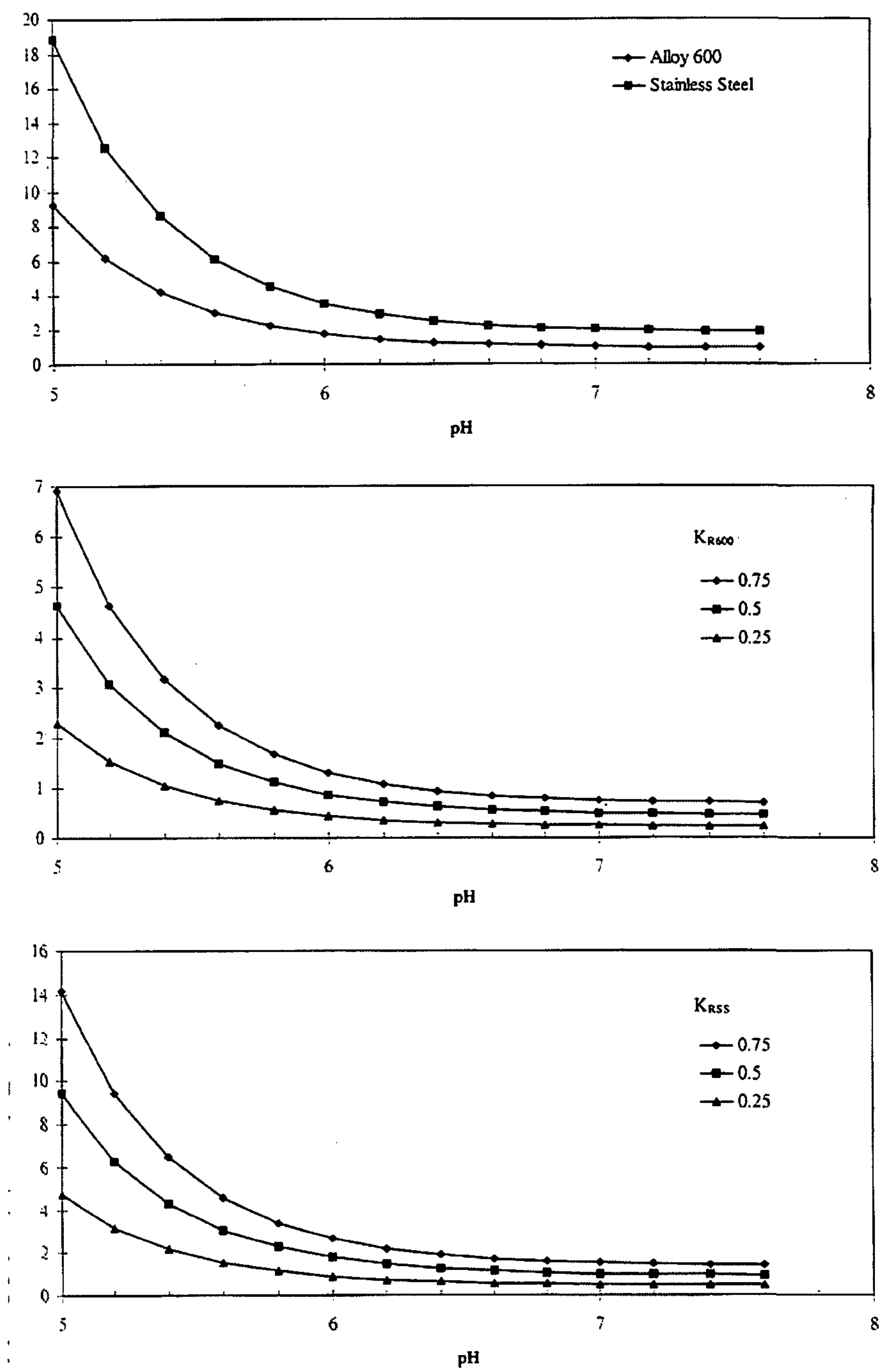

Figure 2-7
Estimated Corrosion Rates and Metal Release Rates for Alloy 600 and 300 Series Stainless Steel at 288º to 316ºC (1) (KR = Release Rate Fraction; no zinc)

Surface Activity Incorporation

Limited insights into the buildup of activity on primary system surfaces can be obtained from the corrosion rate and film growth estimates if a simple activity incorporation model is assumed. For example, if it is assumed that the incorporation of ionic activity into the surface films only occurs at corrosion sites during the corrosion process, as a result of soluble transport, surface activity estimates can be developed as a function of the percentage of the available oxide lattice locations into which an isotope such as Co-58 is incorporated. Note that a correlation of soluble Co-60 coolant concentration and activity incorporation rate has been demonstrated at BWRs [12]. However, a correlation of surface activity buildup rates with soluble Co-58 and Co-60 concentrations has not been developed for PWRs because of the difficulty of obtaining representative coolant samples and the limited database on piping and steam generator surface activities.

In Figure 2-8, the predicted variation in Co-58 surface concentration as a function of time after initial exposure of a corroding surface to PWR primary coolant is shown for a Co-58 incorporation rate of 1E-5% of the oxide matrix sites as they are formed. Following incorporation, the Co-58 is allowed to decay, and additional incorporation occurs only into new oxide matrix locations. No additional incorporation occurs into sites initially occupied by Co-58 because these sites become occupied by the stable daughter. In Figure 2-9, similar results considering simultaneous incorporation of 50% Co-58 and 50% Co-60 into the growing oxide matrix are shown. This ratio is based on a coolant Co-58 to Co-60 activity ratio of 30 and an atomic ratio of 1:1. The soluble activity ratio of 30 is based on the data from Sizewell B as reported by Garbett [11]. Normalized dose rate trends for a hard gamma ratio of 2.5 for Co-60 to Co-58 are shown in Figure 2-10.

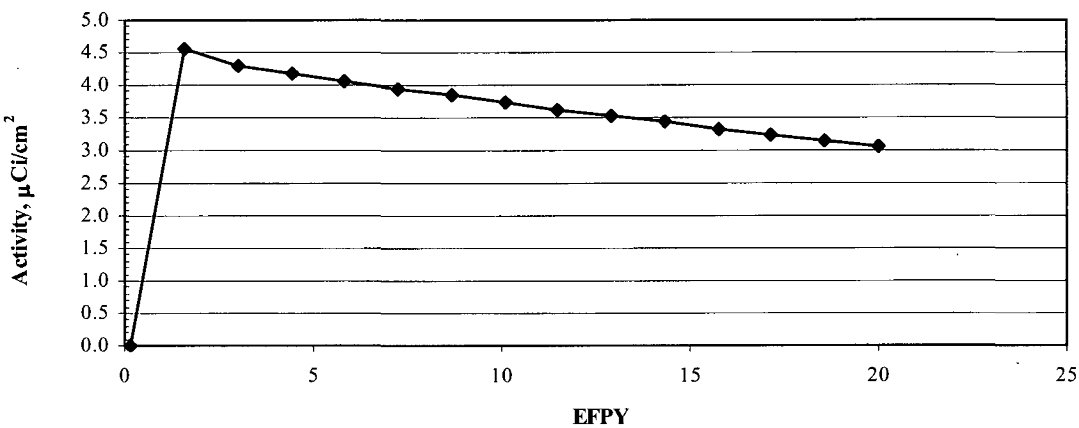

Figure 2-8
Estimation of Co-58 Surface Activity Buildup for 0.00001% Site Incorporation

The results of the above simplistic modeling approach to the activity incorporation process are reasonably consistent with observations of shutdown dose rate **trends** at most plants following initial startup as well as at plants following steam generator replacement (e.g., Figure 2-11). Since the selected incorporation fraction value was completely arbitrary and the predicted trend is based on a very simple model, validity of the basic hypothesis cannot be assessed based on these results.

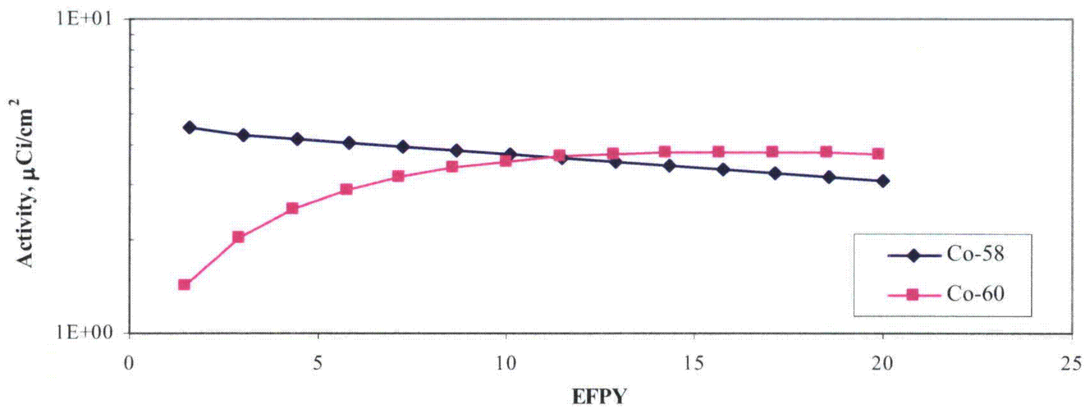

Figure 2-9
Estimation of Co-58 and Co-60 Surface Activity Buildup for 0.00001% Site Incorporation and Equal Coolant Co-58 and Co-60 Atom Concentrations

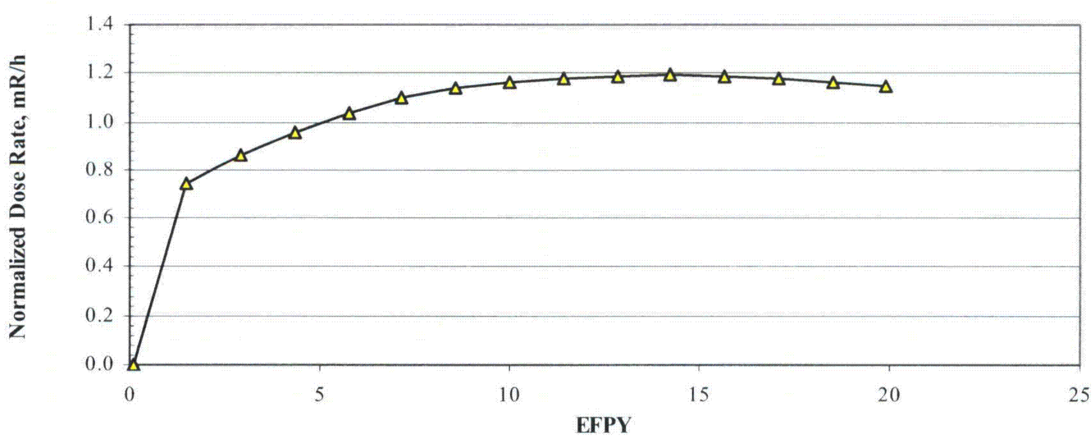

Figure 2-10
Dose Rate Trend Approximation (Normalized to 1.0 at 5 EFPY)

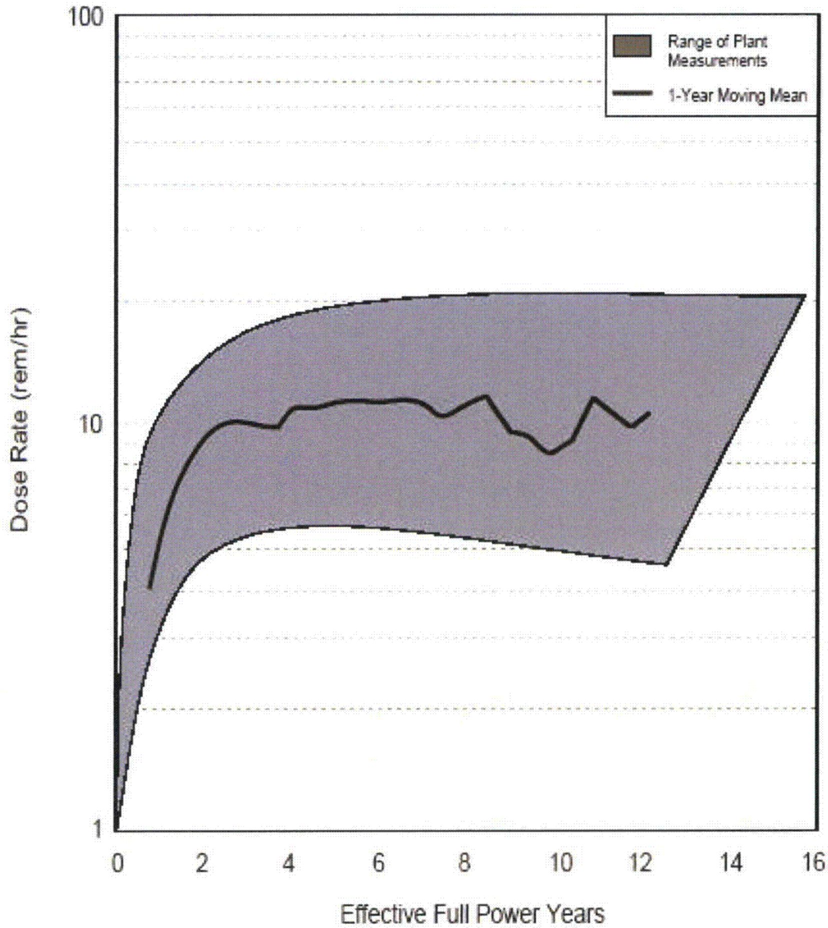

Figure 2-11
Radiation Field Trends in PWRs [2]

Effect of Material Composition on Activity Incorporation

As shown in Table 2-1 to Table 2-3, average corrosion and release rates measured in the laboratory for Alloy 690 are approximately 4 times lower than those for Alloy 600. Thus, film thicknesses on Alloy 690 tubing with initial surface characteristics similar to those of the laboratory coupons used to develop the corrosion and release rate values would be expected to be significantly less than for Alloy 600 tubing. If the basic hypothesis is correct, i.e., activity incorporation is dependent on the soluble Co-58 and Co-60 concentrations in the primary coolant and available lattice sites, surface activity levels and dose rates for 690 replacement steam generators should be significantly less than those experienced with 600 steam generators if exposed to the same chemistry and operating conditions. However, note that the area of the replacement steam generator tubing can be as much as 70% greater than the original steam generators (Table 2-5). The simple hypothesis also would lead to a prediction of much higher surface activity levels on stainless steel piping than on Alloy 600 or 690 surfaces since the corrosion rate of the piping is significantly greater than that of the high nickel alloys. However, the film composition will no doubt vary with alloy composition, and the fraction incorporation could vary correspondingly.

Effect of Zinc Addition

Zinc has a divalent spinel matrix preference similar to cobalt and at a 5 ppb concentration is present at an atom concentration approximately 1,000,000 times the soluble cobalt concentration. Thus, activity incorporation will be greatly reduced in the presence of zinc. In addition, zinc reduces the corrosion rates of stainless steels and Alloys 600 and 690 thus decreasing the number of available lattice positions amenable to activity incorporation.

Coolant Chemistry

Although primary coolant pH_T affects corrosion and release rates of out-of-core surfaces, deposition on fuel surfaces and soluble iron, nickel, cobalt, Co-58 and Co-60 concentrations, assessments of available laboratory data for Alloy 600 and stainless steels indicate that the effect on corrosion and release rates is not significant [6] in the range of current interest for PWR systems (pH_T of approximately 7.0 to 7.4). However, pH_T has a significant impact on the tendency for precipitation of corrosion products from the coolant onto the fuel surfaces as a result of its effect on the solubility of iron and nickel. The maximum solubility of iron is limited to that in equilibrium with the major stable iron bearing solid phase in the primary system, i.e., nickel iron spinel. The solubility cannot exceed this value at equilibrium. However, it can be less than this value if there is a removal process from the primary coolant, e.g., precipitation on the fuel surfaces. The solubility of iron gradually increases across the core as temperature is increased at pH_{300} >6.9 [11]. If precipitation occurs in the core due to subcooled nucleate boiling, the solubility at the core outlet will decrease resulting in release from the steam generator tubing and piping surfaces. Note that the predicted solubility change across the core is relatively small at equilibrium, i.e., <40%.

At normal hydrogen concentrations, the stable nickel phase is nickel metal. In the range of interest, pH_T has a minimal effect on nickel solubility, i.e., there is a predicted increase across the core of ~30% [11].

The above expectations are based on MULTEQ Database Version 5.0 calculations. This database is currently under review for possible revision since an alternate solubility model has been developed by the EPRI Fuel Reliability Program [11].

At the present time, cobalt compounds are not considered in MULTEQ. Although it would normally be expected that the behavior of nickel should reasonably simulate that of cobalt, this may not be the case in the PWR since it is believed that cobalt is primarily incorporated into spinel oxide phases while nickel is believed to be present primarily as nickel metal in PWR fuel deposits.

Shutdown Evolution Practices

Release of activity from primary system piping surfaces and steam generator tubing is not expected to be significant during the shutdown evolution, and surface activity concentrations measured by gamma spectroscopy following shutdown are believed to provide reasonable estimates of the values present at shutdown [11]. Refer to Section 6 for a comparison of forced oxidation peaks to shutdown radiation fields.

3 RADIATION FIELD MEASUREMENT CONSIDERATIONS

Radiation Field Sources and Interferences

As discussed in Section 2, the primary contributors to out-of-core radiation fields have been identified as Co-58 and Co-60. The large majority of this activity is incorporated into the oxide film of the ex-core surfaces through soluble oxide growth. Gamma ray spectroscopy measurements of the reactor coolant loop piping have shown that these two radionuclides generally control shutdown dose rates as discussed in Section 6.

Because the half-life of cobalt-58 (70.8 days) is relatively short in comparison to that of cobalt-60 (5.3 years), the time elapsed between the plant shutdown and survey dates can have an impact on the measured radiation fields. This is why the procedures request the data should be taken within 24 hours after completion of the forced oxidation cleanup.

Localized high radiation areas or hot spots located near a survey location can also influence the radiation field measurements. Examples of such hot spots are drain lines, the regenerative heat exchanger, and resistance temperature detector (RTD) manifolds, valves, and associated piping. Unfortunately, much of the historical data did not have labels indicating hot spots or other interferences.

Attenuation Effects

A major factor which can influence the radiation field measurements at the reactor coolant loop survey locations is the wall thickness of the reactor coolant loop piping. For Westinghouse-designed plants, the nominal sizes (inside diameter) of reactor coolant loop piping are listed below:

- Cold leg - 0.698 m (27.5 inches.)
- Hot leg - 0.74 m (29 meters)
- Crossover leg - 0.79 m (31 inches)

The wall thickness of CE and B&W plants were not known at publication time; however, they are expected to be of similar magnitude.

Table 3-1
Reactor Coolant Loop Piping Nominal Dimension Summary for Westinghouse Plants

SRMP Survey Location	Nominal Pipe ID		Nominal Pipe Wall Thickness	
	Meters	Inches	Meters	Inches
C1	0.79	31	0.0654	2.575
C2	0.79	31	0.0654	2.575
C3	0.79	31	0.0654	2.575
C4	0.79	31	0.0654	2.575
C5	0.79	31	0.0654	2.575
HL	0.74	29	0.0615	2.42
CL	0.698	27.5	0.0584	2.3

Instrumentation

Examination of the radiation field data compiled in Appendices A and B of this report shows that significant variations can exist in the data measured at the same location but on different occasions.

To effectively present the plant radiation fields trends, the factors that influence the data must be identified, quantified, and considered. These factors can include the source strength, attenuation effects, and instrumentation utilized.

Unfortunately, many of the historical data that were used in this report do not have adequate descriptions of the attenuation and instrumentation effects; the data were often presented as tables of values without qualification information.

There are a variety of radiation measurement devices available to the industry, and this report could not identify all of them by make and model. However, most are either Geiger-Muller tubes or Ion Chambers that are calibrated to known sources.

Many utilities have chosen to use Electronic Dosimeters (EDs) because they conveniently interface with plant data collection systems. They are ion chamber detectors, except they often have an electronic bias programmed for conservatism. These data are corrected if the bias is known.

Because the instrument type is unknown in much of the data, a detailed comparison between instrument types could not be performed. The limited data suggest that no one type of instrument reads consistently higher or lower than any other type. The only exception is those measurements performed with a shielded versus a conventional (unshielded) probe. For this report, only measurements performed with conventional (unshielded) probes are considered in the presentation of the data.

Most of the data compiled in this report were taken with an unshielded probe survey instrument. A partial list of available instrumentation is given below in Table 3-2.

Table 3-2
Instrumentation Summary

Name/Model Number	Vendor	Detector Type	Company Website	Currently available?
Teletector-6112	Eberline(Thermo)	Geiger-Mueller	www.thermo.com	No
RO-2A	Eberline(Thermo)	Ionization chamber	www.thermo.com	No
RO-20A	Eberline(Thermo)	Ionization chamber	www.thermo.com	Yes
E-530	Eberline (Thermo)	Geiger-Mueller	www.thermo.com	Uncertain
E-530N with shielding probe (HP-220A)	Eberline(Thermo)	Geiger-Mueller	www.thermo.com	Uncertain
Xetec-330	Xetec, Inc.	Geiger-Mueller	N/A	Uncertain
PIC-6	Eberline(Thermo)	Ionization chamber	www.thermo.com	No
MG-90 (Electronic Dosimeters)	MGP Instruments (SynOdsys)	Geiger-Mueller	http://synodys.com/ portal/	Yes
RSO-50	Bicron (Thermo)	Ionization chamber	www.thermo.com	No
RSO-500	Bicron (Thermo)	Ionization chamber	www.thermo.com	No

Note: EPRI does not endorse any vendor or product. Instrument type is collected for statistical purposes only, and the list above was collected from industry surveys and web research.

4

UPDATE OF THE STANDARD RADIATION MONITORING PROGRAM PROCEDURES

Background

The SRMP survey procedures define the methodology needed to collect radiation surveys at well-defined locations and to record pertinent plant conditions. The data gathered in the surveys give a better understanding of the parameters that influence RCS radiation fields. This information will, in turn, provide the potential for reducing plant radiation fields.

During the planning of the reinstatement of the SRMP, it was noted that there is a significant amount of historical data already available, and changes in the procedures should be limited to ensure that these data are still relevant for comparison purposes. Therefore, most of the technical requirements of the program is relatively unchanged.

The updated procedures have several programmatic changes in order to reduce the burden on the Radiation Protection staff. These changes include

- The elimination of chemistry information from the data request— Coolant chemistry data, which are significant in relation to the deposition, release, and transport of activated corrosion products, are collected under the PWR Chemistry Monitoring and Assessment (PWR CMA) program.

- The SRMP General Procedure has been revised to have separate procedures for Westinghouse, Combustion Engineering, and Babcock and Wilcox plants in order to reduce confusion in the procedure. This is different from the earlier version where the Westinghouse and Combustion Engineering plant designs were included in the same procedure.

- The development of the Babcock and Wilcox monitoring points—As noted in Section 1, Babcock and Wilcox designed plants did not participate in the earlier program. Consultation with B&W plant Health Physics staff, and application of the similar methodology led to the selection of loop piping and channel head points that are reasonably comparable to those at Westinghouse and CE plants.

- Developing a data collection priority system—Several comments from utilities indicated that many of the survey points were difficult to access and posed both safety and ALARA concerns.

- Data from the previous SRMP reports [1, 2, 4] have often not been included for this report because they were listed as averages of the loop values. These data may be included in future versions of this report, but the current database is designed to have individual measurements.

Copies of the procedures are given in Appendices C, D, and E.

Survey Point Priority

As noted above, several concerns about worker safety and ALARA led to the prioritization of the survey points. The survey locations were re-defined as 'Required Points,' 'Highly Recommended Points,' 'Recommended Points,' and 'Optional Information.' The definitions of these terms are below:

- Required points are those that must be taken.
- Highly Recommended are those that are strongly requested, but may be skipped in only cases of personnel safety, poor accessibility, or significant ALARA impact. The points have significant research value and the plants are asked to make the best possible effort to obtain them.
- Recommended points are those that are requested, but may be skipped in cases of personnel safety, poor accessibility, or significant ALARA impact.
- Optional information is information that is requested if available.

The procedures provide a controlled measurement program for assessing radiation field trends of RCS components.

The radiation surveys are conducted during plant shutdowns and collect dose rate readings at permanent markers located on the outside surfaces of RCS components. Surveys are also specified for the internal surfaces of the steam generator channel heads when maintenance or inspection activities are performed.

Westinghouse Survey Points

The following section discusses the survey points and requirements of the radiation survey procedures for Westinghouse designed plants.

Reactor Coolant Loop Piping Survey Procedure

The reactor coolant loop piping survey locations are given in Figure 4-1 and Figure 4-2 and are summarized below.

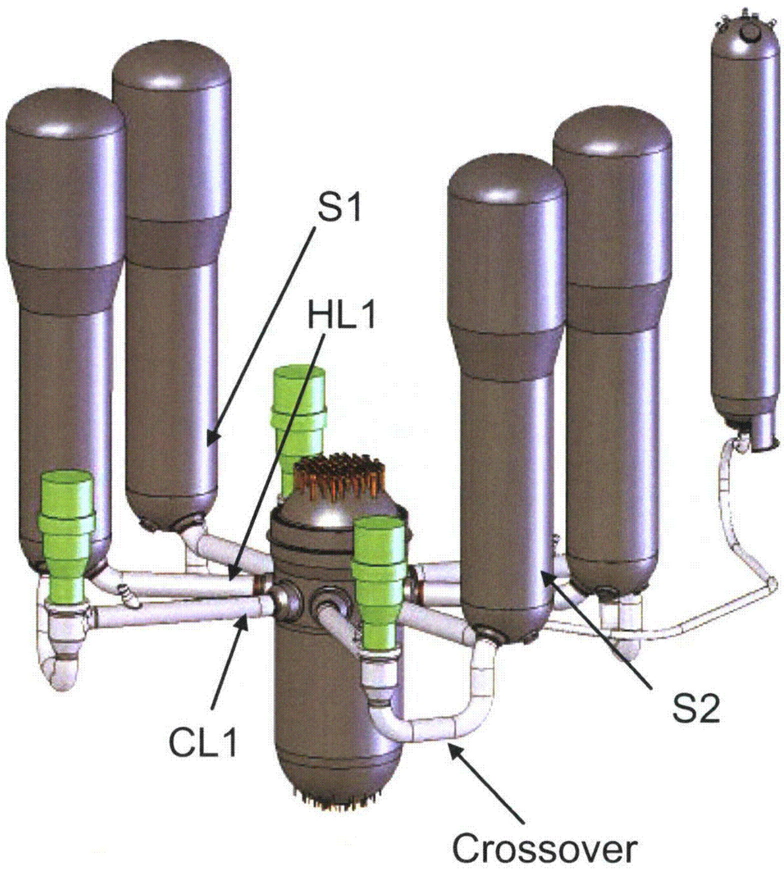

Figure 4-1
Typical Westinghouse 4-Loop Plant With Piping and Steam Generator Survey Points Marked.

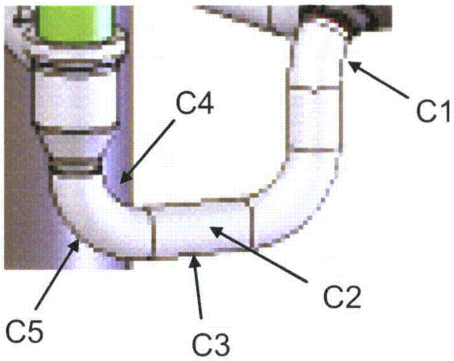

Figure 4-2
Expanded View Of Crossover Survey Points For Westinghouse Plants

Required Points

- C2 - Straight section of crossover piping, side of pipe (generally away from primary concrete shield)
- HL1 - Bottom of hot leg piping between steam generator inlet and reactor vessel shield
- CL1 - Bottom of cold leg piping between reactor coolant pump and reactor vessel shield
- S1 & S2 if taken previously (See below)

Recommended Points

- C1 - Above crossover piping elbow, midway along vertical section of piping from the steam generator
- C3 - Straight section of crossover piping, bottom
- C4 - Crossover piping elbow to RCP, midway along inside radius
- C5 - Crossover piping elbow to RCP, midway along outside radius

- S1 - Outside of steam generator hot leg side, approximately 1 meter above top of channel head tube sheet and approximately midway between secondary side hand-hole cover and hot leg piping (90 degrees radially from the tube lane)
- S2 - Same as S1 but approximately midway between secondary side hand-hole cover and cold leg piping (90 degrees radially from the tube lane)

Optional Information Points

Note: Specify location of measurements, e. g., on letdown piping, one foot downstream of regenerative heat exchanger

- Letdown piping
- CVCS heat exchanger (on the shell)
- RHR piping
- RHR heat exchangers (on the shell)
- Refueling water surface

Steam Generator Channel Head Survey Procedure

If access to the steam generator channel head(s) occurs during the shutdown period, the results of the channel head survey are to be recorded on an appropriate survey form included in procedure. The steam generator channel head survey locations are given Figure 4-3 and are summarized below.

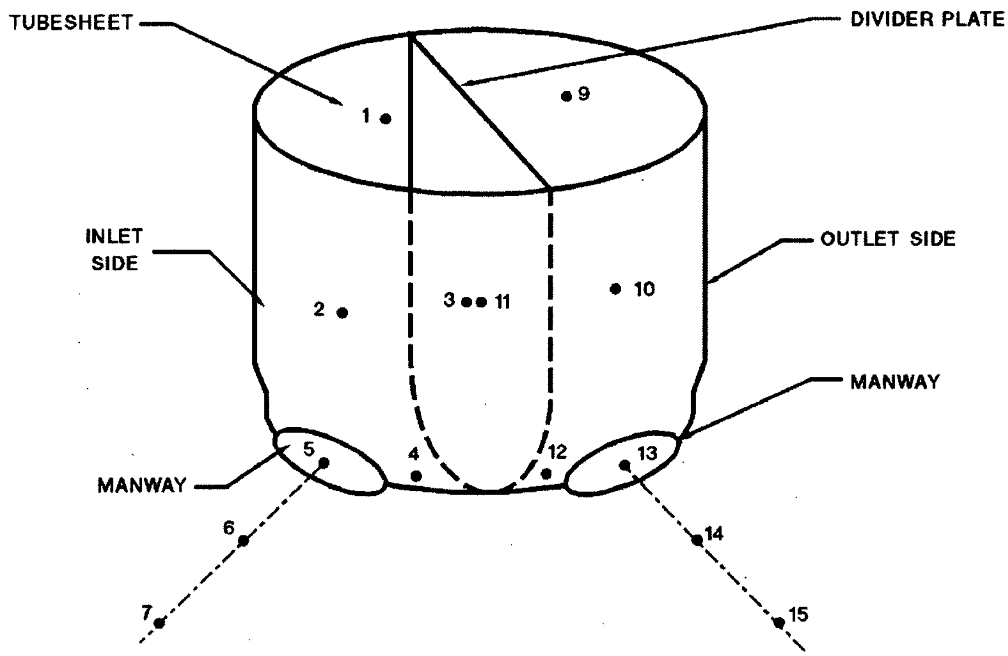

Figure 4-3
Westinghouse Plant Channel Head Survey Points

Required Points

- Midpoint of Tubesheet (Hot Leg & Cold Leg ,points 1 and 9)
- Channel Head Center (Hot Leg & Cold Leg, points 2 and 10)
- Center Divider Plate (Hot Leg & Cold Leg, points 3 and 11)
- Bottom of Channel Head (Hot Leg & Cold Leg, points 4 and 12)

Recommended Points

- Manway Entrance (Hot Leg & Cold Leg, points 5 and 13)
- 30 centimeter from Manway (Hot Leg & Cold Leg, points 6 and 14)
- One meter from Manway (Hot Leg & Cold Leg, points 7 and 15)

Combustion Engineering Plants

The following section contains information on the radiation survey procedures for Combustion Engineering designed plants.

Reactor Coolant Loop Piping Survey Procedures

The reactor coolant loop piping survey locations are given in Figure 4-4 and Figure 4-5, and are summarized below.

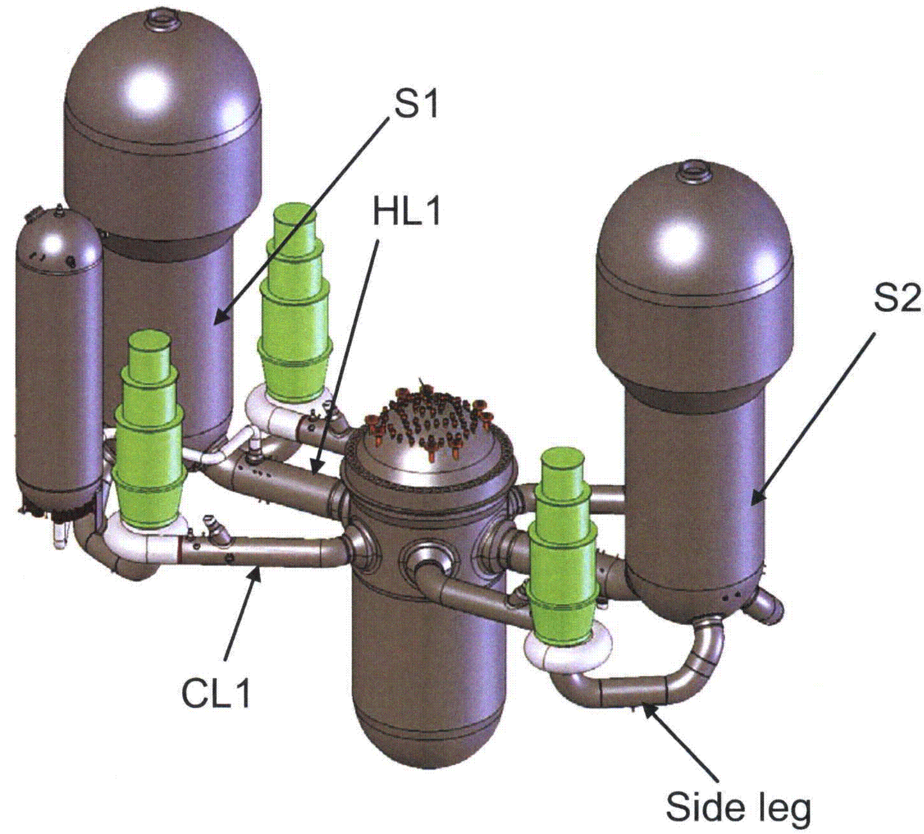

Figure 4-4
Combustion Engineering Loop Piping and Steam Generator Survey Points

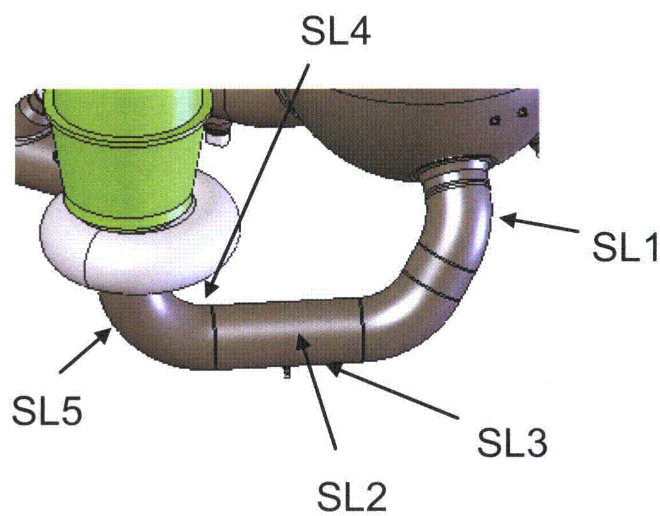

Figure 4-5
Expanded View Of Side-Leg Survey Points For CE Plants

Required Points

- SL2 - Straight section of the side-leg (crossover) piping, side of pipe (generally away from primary concrete shield)
- HL1 - Bottom of hot leg piping between steam generator inlet and reactor vessel shield
- CL1 - Bottom of cold leg piping between reactor coolant pump and reactor vessel shield

Recommended Points

- SL1 - Above side leg piping elbow, midway along vertical section of piping from the steam generator
- SL3 - Straight section of side leg piping, bottom
- SL4 – Side leg piping elbow to RCP, midway along inside radius
- SL5 – Side leg piping elbow to RCP, midway along outside radius
- S1 - Outside of steam generator hot leg side, approximately 1 meter above top of channel head tube sheet and approximately midway between secondary side hand-hole cover and hot leg piping (90 degrees radially from the tube lane)
- S2 - Same as S1 but approximately midway between secondary side hand-hole cover and cold leg piping (90 degrees radially from the tube lane)

Optional Information Points

Note: Specify location of measurements, e. g., on letdown piping, one foot downstream of regenerative heat exchanger

- Letdown piping
- CVCS heat exchanger (on the shell)
- RHR piping
- RHR heat exchangers (on the shell)
- Refueling water surface

Steam Generator Channel Head Survey Points

The steam generator channel head survey locations are given Figure 4-6 and summarized below.

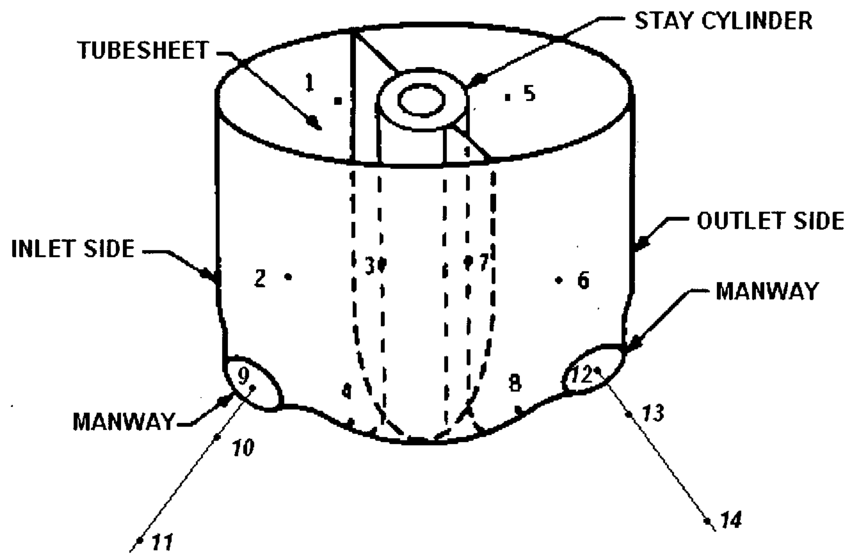

Figure 4-6
Steam Generator Channel Head Survey Points For CE Plants

Required Points

- Midpoint of Tubesheet (Hot Leg & Cold Leg, points 1 and 5)
- Channel Head Center (Hot Leg & Cold Leg, points 2 and 6)
- Center of Stay Cylinder (Hot Leg & Cold Leg, points 3 and 7)
- Bottom of Channel Head (Hot Leg & Cold Leg, points 4 and 8)

Recommended Points

- Manway Entrance (Hot Leg & Cold Leg, points 9 and 12)
- 30 centimeter from Manway (Hot Leg & Cold Leg, points 10 and 13)
- One meter from Manway (Hot Leg & Cold Leg, points 11 and 14)

Babcock and Wilcox Plants

The following section contains information on the radiation survey procedures for Babcock and Wilcox designed plants.

Reactor Coolant Loop Piping Survey Procedures

The reactor coolant loop piping survey locations are given Figure 4-7 and are summarized below. Note the D2 point is not shown, but is located on the other discharge line of the same generator.

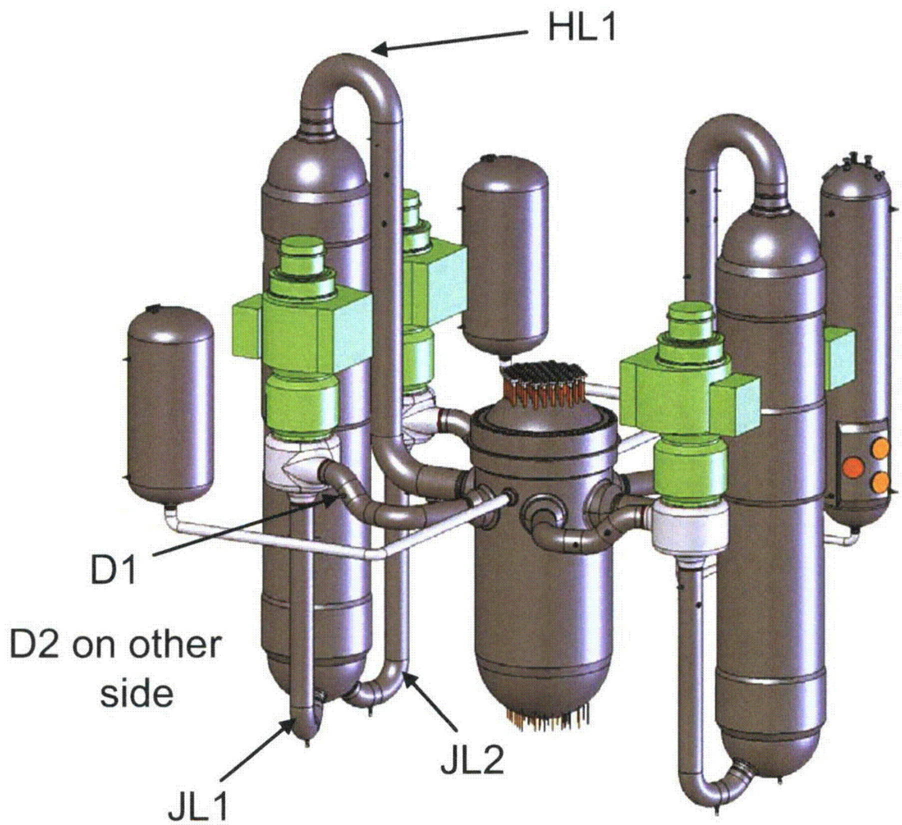

Figure 4-7
Reactor Coolant Loop Piping Survey Points For B&W Plants

Required Points

- JL1/JL2 - 30 centimeters above the elbow of the cold leg (J-leg) opposite the steam generator (Two points required for each steam generator)

Recommended Points

- D1/D2 - Opposite of the high pressure injection (HPI) nozzle (Two points for each steam generator)
- HL1 - Top of the hot leg elbow (One point for each steam generator)

Optional Information Points

Note: Specify location of measurements, e. g., on letdown piping, one foot downstream of regenerative heat exchanger

- Decay heat piping
- Cooler inlet piping
- Decay heat pump suction

- Seismic plate (reactor vessel head service structure)

Steam Generator Channel Head Survey Procedures

If access to the steam generator channel head(s) occurs during the shutdown period, the results of the channel head survey are to be recorded on an appropriate survey form included in procedure. The steam generator channel head survey locations are given Figure 2-6 and summarized below.

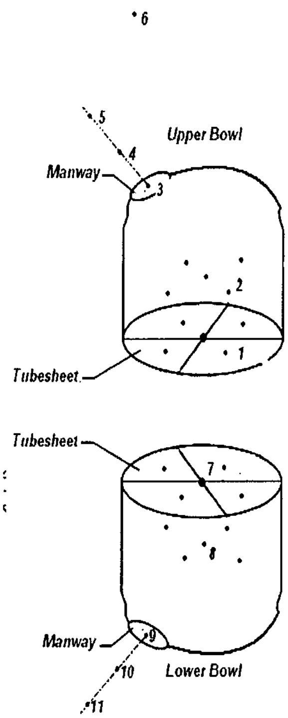

Figure 4-8
B&W Channel Head Survey Locations

Required Points

- Highest General Area Dose Rate of Four Quadrants and Contact with the Center of the Tubesheet (Upper Bowl & Lower Bowl)
- 30 centimeters Above Highest Dose Rate Tubesheet Point of Upper Bowl
- 30 centimeters Below Highest Dose Rate Tubesheet Point of Lower Bowl

Recommended Points

- Plane of Manway (Upper Bowl & Lower Bowl)
- 30 centimeters outside of Plane of Manway (Upper Bowl & Lower Bowl)
- One meter outside Plane of Manway (Upper Bowl & Lower Bowl)
- Three meters above top of the Upper Manway

5
CHANNEL HEAD AND LOOP PIPING DOSE RATE BENCHMARKING

Introduction

This section performs a benchmarking analysis of the available loop piping and channel head dose rate data. The data presentation has several caveats, and while it may be useful to highlight plants with low dose rates, plants with higher dose rates may actually have shown strong improvements over several cycles through aggressive source term reduction efforts. As discussed in Section 2, plants that have had historically high dose rates may require several Co-60 half-lives (5.3 years) in order for the incorporated activity to decay.

In addition, plant-to-plant comparisons are hindered by differences in measurement instruments, measurement techniques, and plant specific factors such as hot spots and monitoring point accessibility.

It should be noted that not all plants have responded with data. This is especially true with loop piping dose rates. Most plants recorded channel head data because of the need for nozzle dam installation and eddy current testing; however, since the SRMP had been discontinued since 1997, many plants had no longer recorded loop piping measurements.

Finally, the reinstatement of the program involved collecting thousands of records in electronic and paper form and transcribing them to a complex relational database. While every effort has been made to accurately report the data, errors may exist within this report. *Plants are encouraged to review their data and send comments to EPRI.*

Plant Benchmarking

The data shown for benchmarking should be considered for illustration only; the data set is incomplete. Table 5-1 summarizes the data that were used for the benchmarking comparisons. It lists the cycle number, date of the cycle outage, tubing material and age of the tubing if less than 5 years, whether the plant injected zinc that cycle, and if the steam generator channel heads are electropolished. The tubing age, zinc status, and electropolishing columns reflect the plant status at the time of the outage.

Table 5-1
Plants Cycle Information Used for Benchmarking Comparison

. **Content deleted - EPRI Proprietary Information.**

Content deleted - EPRI Proprietary Information.

The hot leg and cold leg loop piping and steam generator channel head radiation fields of the three plant designs are compared in Figure 5-1 through Figure 5-4. No consideration was given to other major factors that affect dose rate such as plant age or chemistry contol methodology. The following observations are made about the loop piping:

- The cold leg loop piping dose rates are generally higher than the hot leg dose rates. The median value of the hot leg is 41 mR/hr and the cold leg is 73 mR/hr.
- The variation in the hot leg loop piping radiation fields are relatively narrow, most points fall within 25 mR/hr of the median.
- The B&W loop piping dose rates are all below the median, but it is uncertain if there are attenuation effects because of thicker pipe walls. Data from Crystal River 3 or Davis-Besse are not available.

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Figure 5-1
Average Hot Leg Loop Piping Dose Rates for Most Recent Available Cycle

Content deleted - EPRI Proprietary Information.

Figure 5-2
Average Cold Leg Loop Piping Dose Rates for Most Recent Available Cycle[1]

[1] Millstone 2 and Beaver Valley 1 cold leg loop piping data were not available for latest cycle.

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Figure 5-3
Average Hot Leg Center Channel Head Dose Rates for Most Recent Available Cycle

Content deleted - EPRI Proprietary Information.

Figure 5-4
Average Cold Leg Center Channel Head Dose Rates for Most Recent Available Cycle

A similar review of the channel head data was performed. Note the data from Calvert Cliffs are before steam generator replacement.

The lack of correlation between plant types is expected because the plants have similar materials of construction and operate under similar chemistries and operating temperatures. Differences in coolant pump design and steam generator configuration should have little impact on the incorporation of activity in the corrosion films.

Several plant variables were compared including steam generator tubing alloy, zinc injection, and electropolishing. Plants that have electropolished steam generators heads generally exhibit lower dose rates as show below in Figure 5-5.

Content deleted - EPRI Proprietary Information.

Figure 5-5
Comparison Of Steam Generator Center Channel Head Hot Leg Dose Rates For Plants That Have And Have Not Electropolished Channel Head Bowls

There also are differences between plants that have steam generators with Alloy 690 tubing and those that have Alloy 600 (Figure 5-6). As expected, because of the lower cobalt content and the lower corrosion and corrosion product release rates, plants with Alloy 690 tubing tend to have lower dose rates. However, it is possible for existing Co-60 and Co-58 on the fuel after replacement to contaminate the channel heads and cause high radiation fields. It should also be noted that the plants with Alloy 600 tubing have been operational for more years.

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Figure 5-6 Comparison Of Channel Head Center Hot Leg Dose Rates For Plants That Have Alloy 600 Tubing, Recently Installed Alloy 690 Tubing, And Alloy 690 Tubing For Greater Than 5 Years.

The previous figures indicate that while it is possible to observe general trends, a simplistic Pareto analysis is not an ideal method for assessing differences in plant to plant radiation fields. For example, major factors that impact on shutdown dose rates such as operating time, chemistry, shutdowns and startups, and the cobalt source term are not considered. Another reason for the difficulties is the lack of consistency between measurements at similar type locations in the same plant, e.g., hot and cold leg piping. See Figure 5-7 and Figure 5-8. Notice that the difference between the cold leg and hot leg measurements for both channel heads and loop piping is inconsistent both in magnitude and direction. However, cold leg channel head dose rates were greater than the hot leg values at >80% of the plants.

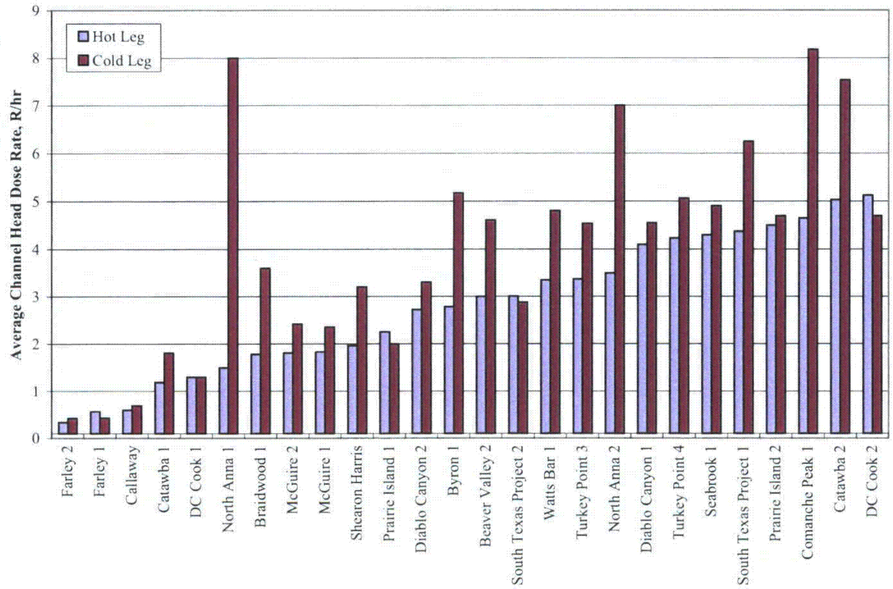

Figure 5-7
Comparison Of Paired Cold Leg/Hot Leg Radiation Fields From The Most Recent Available Cycle For Westinghouse Center Channel Heads

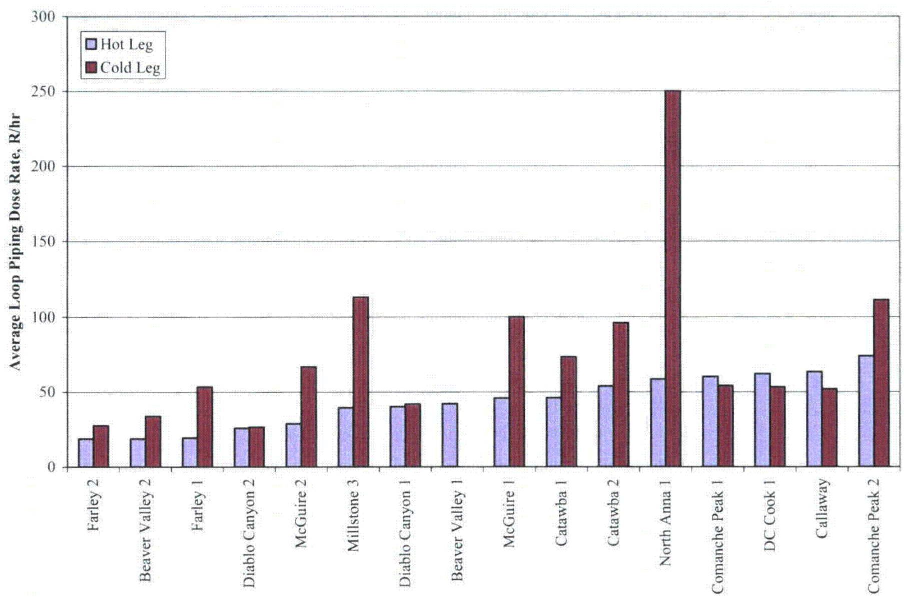

Figure 5-8
Comparison Of Paired Cold Leg/Hot Leg Radiation Fields From The Most Recent Available Cycle For Westinghouse Loop Piping

Effects of Radiocobalt Concentrations on Radiation Fields

The BWR BRAC program has shown a correlation between the reactor water soluble Co-60 to soluble zinc ratio for plants that have implemented NMCA or have sufficient hydrogen [12]. This trend prompted a review of the effects of normal operations reactor coolant cobalt-58 and cobalt-60 concentrations on shutdown dose rates.

Data from the PWR Chemistry Monitoring and Assessment database were compared to the last two available cycles of data for several plants. The results are shown in Figure 5-9 through Figure 5-11, which show the radiation fields of the hot leg loop piping and channel head as a function of cycle average total cobalt-58 and cobalt-60 concentration. There is no apparent correlation of the shutdown radiation fields with respect to total radiocobalt concentrations. This appears counter-intuitive to BWR experience, but there are several complicating factors. For example, obtaining a representative sample of soluble and particulate radiocobalt species at PWRs is more difficult than at BWRs due to the chemistry changes that occur on sample cooldown and depressurization at a PWR. These changes can significantly affect the observed soluble to particulate ratio. Further research in this area is needed.

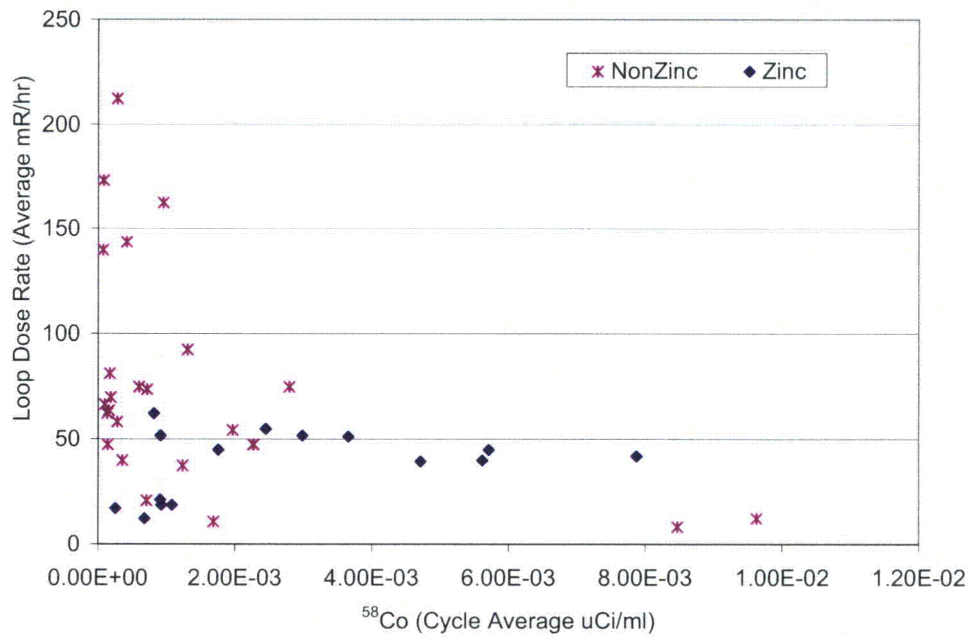

Figure 5-9
Hot Leg Loop Piping Shutdown Dose Rates As A Function Of Cycle Average Cobalt-58 Coolant Concentrations For Zinc And Non-Zinc Plants

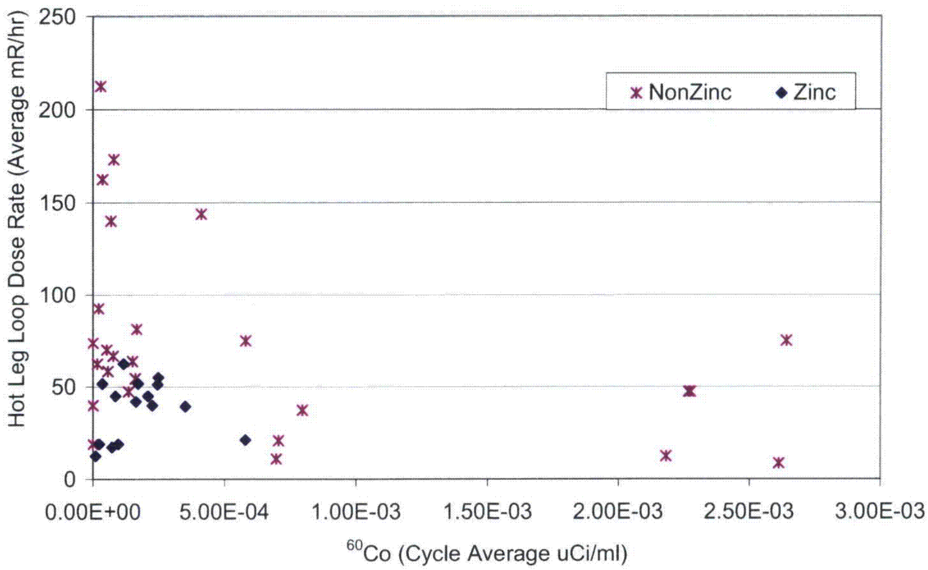

Figure 5-10
Hot Leg Loop Piping Dose Rates As A Function Of Cycle Average Cobalt-60 Coolant Concentrations For Zinc And Non-Zinc Plants

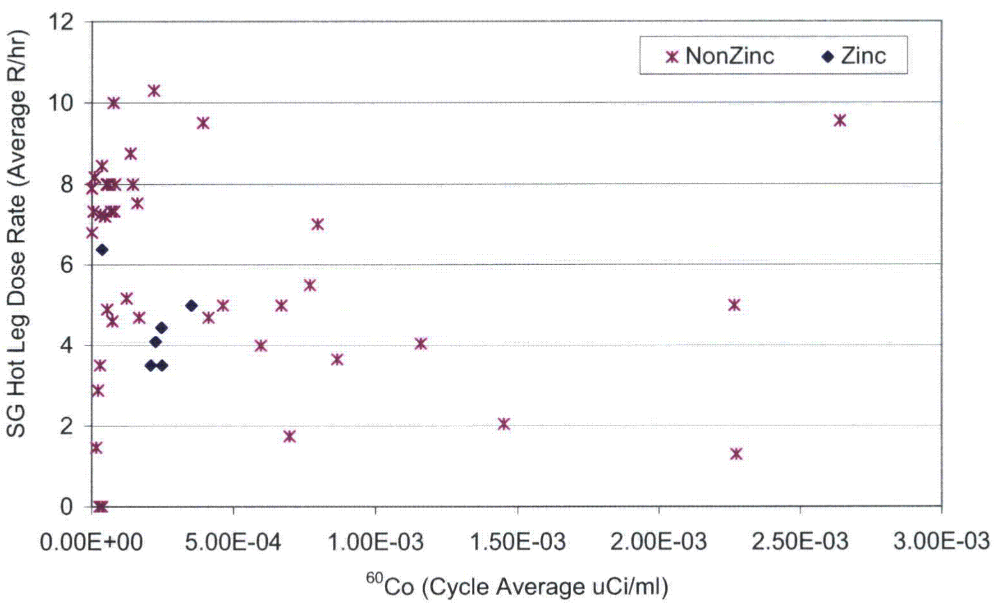

Figure 5-11
Hot Leg Center Channel Head Dose Rates As A Function Of Cycle Average Cobalt-60 Coolant Concentration For Zinc And Non-Zinc Plants

Summary of Fleet Benchmarking Results

As noted in the introduction, the limited benchmarking examples presented in this section have considerable uncertainty because of the lack of consistency between measurement methods and instruments, the limited available database, and the lack of consideration of major variables that can significantly affect shutdown dose rates. For example, consideration has not been given to critical variables such as EFPY, coolant chemistry, cobalt source terms, startups and shutdowns, etc. As the database is updated and expanded, consideration of these and other factors is expected to allow firm relations of shutdown dose rates to specific plant design and operating variables to be developed.

The following preliminary conclusions were drawn from the limited benchmarking effort performed to date:

- In general, the cold leg locations have higher radiation fields than the hot leg locations. The cause for this effect is uncertain, and further work in this area is needed.
- There is no apparent correlation of shutdown radiation fields to cycle average total radiocobalt concentrations.
- Plants that have electropolished steam generator channel heads and alloy 690 tubing generally have lower dose rates than those that do not. Of these plants, those that implemented zinc injection early have considerably lower dose rates.

6
CHANNEL HEAD AND LOOP PIPING DOSE RATE VARIATION WITH EFFECTIVE FULL POWER YEARS

Section 5 showed that while plant benchmarking can be used on a limited basis, the technique is not recommended for investigating cause and effect relationships between plant operations and configuration and radiation fields.

A more effective approach is to monitor the radiation fields of a single plant over several outages and see how the fields change over time after plant changes. This style of analysis has been the basis of the previous SRMP reports, and the following sections apply this technique to the relevant PWR plant and chemistry changes.

Alloy 600 Plants Without Steam Generator Replacement Or Zinc Injection

Two of the major impacts on a plants radiological performance are steam generator replacement and implementation of zinc injection [13], hence, plants that have not implemented either may be chosen as a reasonable base case for single plant comparisons. Figure 6-1 shows the cold leg center channel head radiation fields of plants that have not replaced generators or implemented zinc injection. The radiation fields follow the pattern that was established in Figure 2-10 and Figure 2-11.

The effects of major variables such as operating temperature, fuel cleaning, core duty, etc., have not been quantified. For example, Callaway radiation fields increased as the unit began the core uprates in Cycles 6-10, and then decreased after they reduced the operating temperature, implemented ultrasonic fuel cleaning, and eventually zinc injection in later cycles.

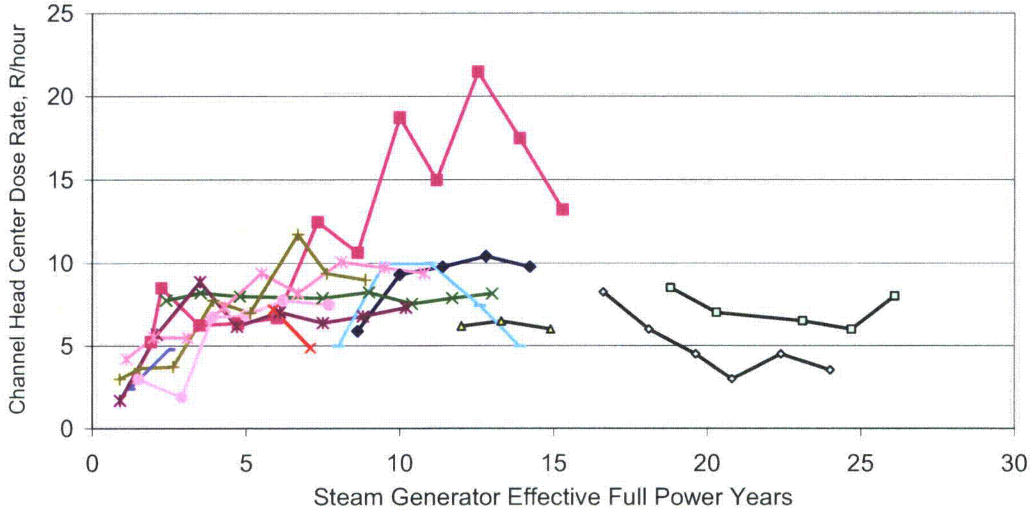

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Figure 6-1
Cold Leg Channel Head Center Dose Rates as a Function of Time for Plants with Alloy 600 Tubing without Zinc Injection

Effects of Steam Generator Replacement

Steam generator replacement with Alloy 690 reduces the elemental cobalt source term in the RCS system, and correspondingly the amount of Co-59 that is deposited on the fuel. The changes in loop piping dose rates vary by plant, but in general, for plants that have not implemented zinc injection, the dose rates remain constant for three cycles after replacement and then decrease. Refer to North Anna 1 and 2 (Appendix A, Figure A-25 and Figure A-26).

Figure 6-2 shows the hot leg center channel head dose rates for three types of plants: 1) replaced steam generators with Alloy 600 tubing and no electropolishing, 2) replaced with Alloy 690 TT tubing and no electropolishing, and 3) replaced with Alloy 690 TT with electropolished channel head bowls.

The non-electropolished 600 plants, in general, had higher dose rates than those with Alloy 690. One interesting trend that is common to all of the data is that the radiation fields were constant (or slowly increasing in the case of the electropolished bowls) for the first three cycles after replacement. A plausible explanation for this trend is the reduction in the steam generator tubing corrosion rate and corrosion product release rate that occurs over time following exposure to the reactor coolant. The beneficial effect of electropolishing is also apparent.

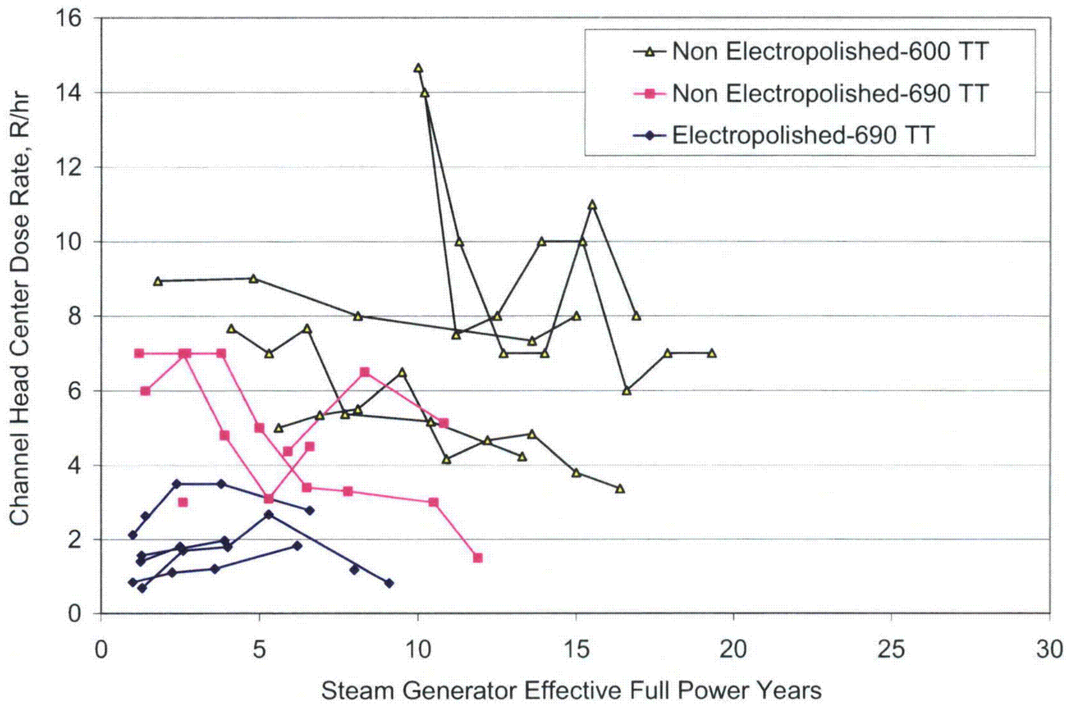

Figure 6-2 Hot Leg Center Channel Head Dose Rates for Plants That Have Replaced With and Without Alloy 690 TT and Electropolished Channel Heads

Effects of Zinc Injection on Shutdown Dose Rates

The overall effects of zinc injection on radiation fields throughout the industry have been well-documented in the PWR Zinc Application Guidelines [14]. The previous works have shown the average radiation field reduction as a function of cumulative exposure. Review of individual cases is also illustrative of the benefits of zinc injection. Two specific examples of interest include Farley 1 as compared to North Anna 2, and Diablo Canyon.

Farley 1 and North Anna 2 Comparison

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Table 6-1
Comparison of Farley 1 and North Anna 2 Design and Chemistry Parameters

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Figure 6-3
Loop Piping Dose Rates As A Function Of Time For Farley 1 And North Anna 2 After Steam Generator Replacement

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Zinc Injection and Diablo Canyon Gamma Spectroscopy Measurements

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Table 6-2
Diablo Canyon Zinc Injection History

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Table 6-3
Gamma Scan Results for Diablo Canyon Unit 1 Crossover Piping

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Table 6-4
Gamma Scan Results for Diablo Canyon Unit 2 Crossover Piping

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Effects of Forced Oxidation Peaks on Shutdown Dose Rates

Development of the PWR Chemistry Monitoring and Assessment and SRMP databases allows comparison of the Co-58 forced oxidation peak and radiation fields as a function of effective full power years. This analysis was performed for Callaway station before steam generator replacement, and North Anna 2 after steam generator replacement with Alloy 690 tubing. In the Callaway comparison, there is no correlation between the crud burst peak and the shutdown dose rates. For example at 6.01 EFPY, the dose rates increase with a decreased crud burst, while they reverse the trend at 10 EFPY. For North Anna 2, there appears to be a slight correlation for the first three cycles after steam generator replacement; however, from the fourth cycle onward the crud burst peaks drop dramatically while the dose rates fall along the Co-60 decay curve.

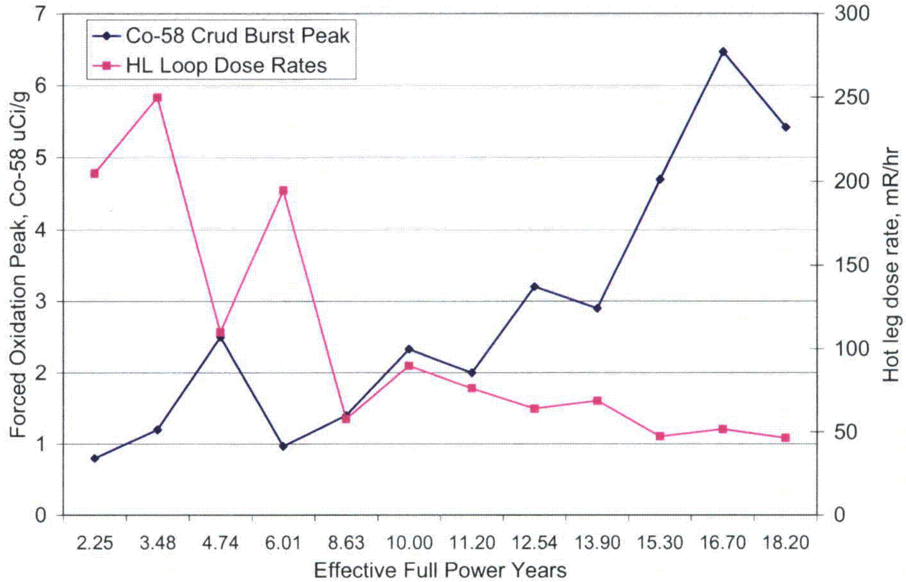

Figure 6-4
Forced Oxidation Co-58 Peaks And Hot Leg Loop Piping Dose Rates As A Function Of Effective Full Power Years For Station

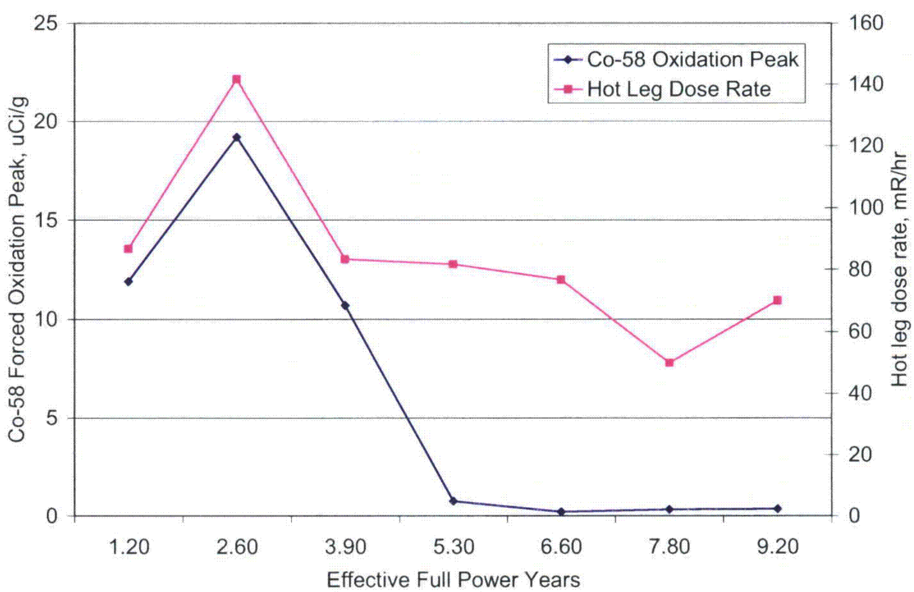

Figure 6-5
Forced Oxidation Co-58 Peaks And Hot Leg Loop Piping Dose Rates As A Function Of Effective Full Power Years For

7 CONCLUSIONS AND RECOMMENDATIONS

The following conclusions/observations are made from this work:

- The EPRI Standard Radiation Monitoring Program has been successfully reinstated with active participation of most United States utilities.
- A relational database linking steam generator design, chemistry, and radiation data provides a powerful tool to understand causal links of plant changes to radiation fields.
- Reviewing the radiological history of a single unit provides significant insight into cause-and-effect relationships of chemistry, operations, and plant design to radiation fields.
- Plant-to-plant benchmarking comparisons are useful to identify strong performers in radiation field reduction, but they may offer misleading indications about the state of a plants source term reduction program. *Ranking plant performance based on benchmarking data is not recommended.*
- Gamma spectroscopy data from Diablo Canyon 1 and 2 and many other PWRs indicate Co-60 and Co-58 are the largest contributors to ex-core radiation fields.
- Zinc injection and steam generator replacement with Alloy 690 tubing have significant impacts on ex-core radiation fields
- Electropolishing steam generator channel heads is an effective method to reduce channel head radiation fields after replacement.
- There is no observable correlation of RCS coolant total Co-58 and Co-60 concentrations and shutdown radiation fields.
- There is no observable correlation between forced oxidation peaks and shutdown radiation fields.

Based on these conclusions and commentary from the utility advisors, the following recommendations are made.

- Strongly encourage complete program participation by the utilities that have not yet submitted data.
- Review and improve the selected points for auxiliary systems (e.g. letdown heat exchanger outlet; the current 'Optional' points in the procedures should be streamlined and upgraded to 'Recommended' points.
- Develop a method to include gamma spectroscopy in the Standard Radiation Monitoring Program. This can include 'smears', gamma scan campaigns, or coupon analysis.

Knowledge of nuclide distributions in surface films is expected to provide valuable insights into radiation field behavior over time.

- Strengthen the data relationships among chemistry, fuel reliability, steam generator design, and operating conditions. Current areas that require investigation include the effects of operating temperature, hydrogen concentration, core design, and steam generator tubing manufacturing process. At this time, there is not an effective means to correlate these results.

To support these conclusions, in the next two to three years EPRI is sponsoring a communication effort to increase participation, including with EDF, developing a gamma spectroscopy program with chemistry, communicating with the Fuel Reliability Program and Steam Generator Management Program to consolidate data, and will hold Standard Radiation Monitoring Program workshops to help define auxiliary system radiation points more consistently.

8
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A
REACTOR COOLANT LOOP DOSE RATE DATA

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Figure A-1 Average Loop Piping Radiation Fields As A Function Of Effective Full Power Years For All Westinghouse Plants

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Figure A-2 Average Loop Piping Radiation Fields As A Function Of Effective Full Power Years For Combustion Engineering Plants

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Figure A-3 Average Loop Piping Radiation Fields As A Function Of Effective Full Power Years For Babcock And Wilcox Plants

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Figure A-4 Reactor Coolant Loop Dose Rate Trends for ANO 1

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Figure A-6 Reactor Coolant Loop Dose Rate Trends for Beaver Valley 2

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Figure A-8 Reactor Coolant Loop Dose Rate Trends for Catawba 1

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Figure A-9 Reactor Coolant Loop Dose Rate Trends for Catawba 2

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Figure A-10 Reactor Coolant Loop Dose Rate Trends for Comanche Peak 1

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Figure A-11 Reactor Coolant Loop Dose Rate Trends for Comanche Peak 2

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Figure A-12 Reactor Coolant Loop Dose Rate Trends for Davis Besse

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Figure A-13 Reactor Coolant Loop Dose Rate Trends for DC Cook 1

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Figure A-14 Reactor Coolant Loop Dose Rate Trends for DC Cook 2

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Figure A-15 Reactor Coolant Loop Dose Rate Trends for Diablo Canyon 1

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Figure A-16 Reactor Coolant Loop Dose Rate Trends for Diablo Canyon 2

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Figure A-17 Reactor Coolant Loop Dose Rate Trends for Farley 1

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Figure A-18 Reactor Coolant Loop Dose Rate Trends for Farley 2

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Figure A-19 Reactor Coolant Loop Dose Rate Trends for Fort Calhoun

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Figure A-20 Reactor Coolant Loop Dose Rate Trends for Indian Point 3

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Figure A-21 Reactor Coolant Loop Dose Rate Trends for McGuire 1

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Figure A-22 Reactor Coolant Loop Dose Rate Trends for McGuire 2

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Figure A-23 Reactor Coolant Loop Dose Rate Trends for Millstone 2

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Figure A-25 Reactor Coolant Loop Dose Rate Trends for North Anna 1

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Figure A-26 Reactor Coolant Loop Dose Rate Trends for North Anna 2

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Figure A-27 Reactor Coolant Loop Dose Rate Trends for Oconee 1

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Figure A-28 Reactor Coolant Loop Dose Rate Trends for Oconee 2

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Figure A-29 Reactor Coolant Loop Dose Rate Trends for Oconee 3

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Figure A-30 Reactor Coolant Loop Dose Rate Trends for Palisades

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Figure A-32 Reactor Coolant Loop Dose Rate Trends for Prairie Island 2

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Figure A-33 Reactor Coolant Loop Dose Rate Trends for Robinson

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Figure A-34 Reactor Coolant Loop Dose Rate Trends for San Onofre 2

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Figure A-35 Reactor Coolant Loop Dose Rate Trends for San Onofre 3

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Figure A-36 Reactor Coolant Loop Dose Rate Trends for Shearon Harris

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Figure A-37 Reactor Coolant Loop Dose Rate Trends for South Texas Project 1

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Figure A-38 Reactor Coolant Loop Dose Rate Trends for South Texas Project 2

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Figure A-39 Reactor Coolant Loop Dose Rate Trends for Surry 1

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Figure A-40 Reactor Coolant Loop Dose Rate Trends for Surry 2

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Figure A-41 Reactor Coolant Loop Dose Rate Trends for Three Mile Island 1

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Figure A-42 Reactor Coolant Loop Dose Rate Trends for Turkey Point 3

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Figure A-43 Reactor Coolant Loop Dose Rate Trends for Turkey Point 4

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Figure A-44 Reactor Coolant Loop Dose Rate Trends for Wolf Creek

B
STEAM GENERATOR CHANNEL HEAD DOSE RATE DATA

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Figure B-1 Average Channel Head Radiation Fields As A Function Of Effective Full Power Years For Westinghouse Plants

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Figure B-2 Average Center Channel Head Radiation Fields As A Function Of Effective Full Power Years For Combustion Engineering Plants

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Figure B-3 Average Center Channel Head Radiation Fields As A Function Of Effective Full Power Years For Babcock And Wilcox Plants

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Figure B-4 Steam Generator Channel Head Dose Rate Trends for ANO 1

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Figure B-5 Steam Generator Channel Head Dose Rate Trends for Beaver Valley 2

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Figure B-6 Steam Generator Channel Head Dose Rate Trends for Braidwood 1

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Figure B-9 Steam Generator Channel Head Dose Rate Trends for Byron 2

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Figure B-10 Steam Generator Channel Head Dose Rate Trends for Callaway

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Figure B-13 Steam Generator Channel Head Dose Rate Trends for Catawba 1

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Figure B-18 Steam Generator Channel Head Dose Rate Trends for Davis Besse

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Figure B-21 Steam Generator Channel Head Dose Rate Trends for Diablo Canyon 1

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Figure B-29 Steam Generator Channel Head Dose Rate Trends for North Anna 1

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Figure B-30 Steam Generator Channel Head Dose Rate Trends for North Anna 2

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Figure B-31 Steam Generator Channel Head Dose Rate Trends for Oconee 1

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Figure B-32 Steam Generator Channel Head Dose Rate Trends for Oconee 2

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Figure B-33 Steam Generator Channel Head Dose Rate Trends for Oconee 3

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Figure B-34 Steam Generator Channel Head Dose Rate Trends for Prairie Island 1

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Figure B-36 Steam Generator Channel Head Dose Rate Trends for Robinson

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Figure B-37 Steam Generator Channel Head Dose Rate Trends for Salem 2

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Figure B-38 Steam Generator Channel Head Dose Rate Trends for San Onofre 2

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Figure B-39 Steam Generator Channel Head Dose Rate Trends for San Onofre 3

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Figure B-40 Steam Generator Channel Head Dose Rate Trend at Shearon Harris

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Figure B-49 Steam Generator Channel Head Dose Rate Trends for Waterford

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Figure B-50 Steam Generator Channel Head Dose Rate Trends for Watts Bar 1

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Figure B-51 Steam Generator Channel Head Dose Rate Trends for Wolf Creek

C
REVISED SRMP PROCEDURE FOR WESTINGHOUSE PLANTS

RP20060015
March 2006

EPRI SRMP GENERAL PROCEDURE FOR WESTINGHOUSE DESIGNED PWR PLANTS

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RP20060015
March 2006

EPRI SRMP GENERAL PROCEDURE FOR WESTINGHOUSE DESIGNED PWR PLANTS

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SECTION 1 INTRODUCTION

1.1 BACKGROUND

The Standard Radiation Monitoring Program (SRMP), first instituted in 1978, is part of a more general program, sponsored by the Electric Power Research Institute (EPRI), with the major emphasis on improving plant reliability and availability. The purpose of this program is as follows:

- To provide a meaningful, consistent, and systematic approach to monitoring the rate of PWR radiation field buildup and to provide the basis for projecting the trend of those fields.
- To provide a reliable set of radiation field data for each participating plant, from which comparisons can be made.
- To monitor certain plant parameters that affect or may affect observed radiation fields.
- To use the information from this program to identify plant design features, material selection, and operational techniques that present opportunities for radiation control.

Previous EPRI reports have been published as a result of the SRMP program. The reports list the factors that affect plant dose rates and quantitatively evaluate the effect of these factors. The most important factors were found to be operational coolant chemistry and variations in cobalt input.

The SRMP program had consistent data collection efforts for Westinghouse and Combustion Engineering plants through 1996 and 1985 respectively; afterwards, SRMP data collection had been limited primarily to plants that had implemented zinc injection or replaced steam generators with low-cobalt tubing. The EPRI Chemistry and LLW TAC asked to start these data collection efforts because recent trends in replacement steam generators, core uprating, adverse radiological incidents, and various changes in shutdown and normal chemistry procedures have caused unpredictable fluctuations in dose rates throughout the ex-core surfaces.

This revision to the SRMP General Procedure updates the survey procedures and forms. The procedure has been streamlined by defining survey locations as 'Required Points,' 'Recommended Points,' and 'Optional Information.' The definitions of these terms are below:

- Required points are those that must be taken.
- Recommended points are those that are requested, but may be skipped in cases of personnel safety, poor accessibility, or significant ALARA impact.
- Optional information is information that is requested if available.

Reactor coolant chemistry data are collected under the PWR Chemistry Performance Monitoring and Assessment (PWR CPMA) program. To avoid confusion because of the

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1.2 SUMMARY

The procedures provide a controlled measurement program for assessing radiation field trends of Reactor Coolant System (RCS) components. The Standard Radiation Monitoring Program (SRMP) involves the performance of radiation surveys at well-defined locations and recording certain plant conditions. The data gathered in the surveys will lead to a better understanding of the parameters that influence RCS radiation fields. This information will, in turn, provide the potential for reducing plant radiation fields.

The radiation surveys are conducted during plant shutdowns and collect dose rate readings at permanent markers located on the outside surfaces of RCS components. Surveys are also specified for the internal surfaces of the steam generator channel heads when maintenance or inspection activities are planned. In addition to the radiation survey information, certain coolant chemistry data that may be significant in relation to the deposition, release, and transport of activated corrosion products are to be collected under the PWR CPMA program.

In the specification of data requirements and survey frequencies, an attempt has been made to minimize the additional workload on operating personnel. To reach this goal, this General Procedure has been revised to more clearly define the data requirements (i.e., 'Required Points', 'Recommended Points', and 'Optional Information'), to eliminate the collection of redundant data, and to streamline the data collection forms.

EPRI Solutions will compile and reduce the survey data for the duration of the program. As the processes and parameters that have a major impact on radiation levels are more clearly defined and understood, the data specification may be modified as indicated by the program results.

SECTION 2 DATA REQUIREMENTS

2.1 GENERAL

The data requirements of the Standard Radiation Monitoring Program fall into two basic categories, i.e., 1) radiation surveys and 2) primary coolant chemistry regimes. The primary coolant chemistry information will be collected under the PWR CPMA program. The radiation survey parameters to be monitored and schedules for each of the measurements are discussed below. However, it should be emphasized that any additional observations by plant personnel of conditions, occurrences, or other items which might affect radiation levels or which would provide further insight or understanding of those levels should be recorded. Accordingly, the data input sheets allow for entry of more of this type of information.

2.2 RADIATION SURVEYS

Radiation surveys (at permanent markers located on the outer surfaces of RCS loop components) are to be performed during plant shutdown, except when the duration of the outage is too short to make the survey practical. Radiation surveys of the steam generator channel heads should also be recorded if primary side inspection or maintenance of steam generators is scheduled during a plant shutdown. The recommended frequencies for performing the radiation surveys are outlined in Table 2-1. In the interest of getting useful and consistent data for this program, a list of the more relevant data requirements/screening criteria is included in the table.

Guidelines and information relative to the installation and location of the survey markers are included in Section 5. Forms for recording the reactor coolant loop and steam generator survey results are reproduced in Appendix A. As noted on the Appendix A data sheets, the total number of survey locations is small (less than 10 locations per reactor coolant loop). Because in-containment radiation surveys are generally performed during shutdown conditions, the additional burden on plant personnel and associated radiation exposure is expected to be minimal. Further, some operating plants have implemented electronic dosimetry at selected locations. If electronic dosimetry instruments are used for contact dose rate monitoring, instrument bias information should be noted in the survey forms. Such equipment can provide dose rate data on a continuous basis that can be recorded remotely with a minimum of radiation exposure to plant personnel.

TABLE 2-1. DATA REQUIREMENTS FOR WESTINGHOUSE PLANTS

Content deleted - EPRI Proprietary Information.

SECTION 3 RADIATION SURVEY INSTRUCTIONS

3.1 REACTOR COOLANT LOOP PIPING SURVEY PROCEDURES

Plant personnel should perform the shutdown radiation surveys of the reactor coolant loops in accordance with the schedule and requirements presented in Table 2-1. The results should be recorded on an appropriate survey form included in Appendix A. Note that the reactor coolant loop shutdown survey should preferably be performed with the steam generator manway covers in place.

In order to make the surveys be more consistent and effective throughout the industry, the SRMP working committee defined survey locations as 'Required Points', 'Recommended Points', and 'Optional Information'. Required points are those that must be taken. Recommended points are those that are requested, but may be skipped in cases of personnel safety, poor accessibility, or significant ALARA impact. Optional information is information that is requested if available. The reactor coolant loop piping survey locations are summarized below.

3.1.1. Required Points

- C2 - Straight section of crossover piping, side of pipe (generally away from primary concrete shield)
- HL1 - Bottom of hot leg piping between steam generator inlet and reactor vessel shield
- CL1 - Bottom of cold leg piping between reactor coolant pump and reactor vessel shield

3.1.2. Recommended Points

- C1 - Above crossover piping elbow, midway along vertical section of piping from the steam generator
- C3 - Straight section of crossover piping, bottom
- C4 - Crossover piping elbow to RCP, midway along inside radius
- C5 - Crossover piping elbow to RCP, midway along outside radius

- S1 - Outside of steam generator hot leg side, approximately 1 meter above top of channel head tube sheet and approximately midway between secondary side hand-hole cover and hot leg piping (90 degrees radially from the tube lane)
- S2 - Same as S1 but approximately midway between secondary side hand-hole cover and cold leg piping (90 degrees radially from the tube lane)

Note: S1 and S2 are strongly recommended if taken previously

3.1.3. Optional Information Points

Note: Specify location of measurements, e.. g., on letdown piping, one foot downstream of regenerative heat exchanger

- Letdown piping
- CVCS heat exchanger (on the shell)
- RHR piping
- RHR heat exchangers (on the shell)
- Refueling water surface

The survey instrument to be employed in the surveys should generally be equipped with an extending or telescoping probe that encompasses the range from 2 milliroentgens per hour to 20 roentgens per hour. Battery and response checks of the portable monitoring equipment employed should be performed prior to the survey. It must be ascertained that the instrument has been calibrated according to the calibration procedures and manufacturer's recommendations for that particular instrument. It is desirable to utilize portable monitoring equipment that has been calibrated by exposure to known gamma radiation fields (e.g. a commercially built calibrator equipped with radioactive sources traceable to NIST standards). The instrumentation used for this procedure should be properly calibrated according to plant specific or utility specific procedures. Any special features or conditions associated with the equipment should be noted in the survey form.

Also, as noted in Section 2-2, electronic dosimetry at survey locations can be used in lieu of manual surveys using portable survey equipment. In this case, instrument bias information should be noted in the survey form.

3.2 STEAM GENERATOR CHANNEL HEAD SURVEY PROCEDURES

If access to the steam generator channel head(s) occurs during the shutdown period, the results of the channel head survey are to be recorded on an appropriate survey form included in Appendix A. The steam generator channel head survey locations are summarized below.

3.2.1. Required Points

- Midpoint of Tubesheet (Hot Leg & Cold Leg)
- Channel Head Center (Hot Leg & Cold Leg)
- Center Divider Plate (Hot Leg & Cold Leg)
- Bottom of Channel Head (Hot Leg & Cold Leg)

3.2.2. Recommended Points

- Manway Entrance (Hot Leg & Cold Leg)
- 30 centimeter from Manway (Hot Leg & Cold Leg)
- One meter from Manway (Hot Leg & Cold Leg)

SECTION 4 PRECAUTIONS

The radiation surveys should be conducted in accordance with the following precautions.

- The marker locations relative to a fixed reference point should be recorded when the markers are first installed. It may be necessary to reinstall markers on insulation removed for in-service inspection or maintenance of the component; the fixed reference point notation will allow positioning at the initially established location.

- Radiation surveys of the reactor coolant loop should not be performed with steam generator manway covers removed because the background dose rate from the steam generator channel heads has been found to influence the loop piping measurements. If not possible, a notation should be made on the form that the manway covers were removed.

- If the results of the radiation survey indicate marked inconsistencies with previous survey results and/or if damage to the survey instrument is suspected, the instrument calibration should be rechecked after completion of the survey.

- Prior to entering the survey locations, worker should note hotspots near by. Potential crud traps are summarized below.

    - Drain lines
    - Pressurizer surge line
    - Pressurizer spray line
    - Sample/transmitter lines
    - Loops stop valves
    - Check valves
    - RTD wells
    - Loop drain valves
    - Letdown lines
    - RHR Lines
    - Any low point in a pipe section

- Also note the Data Requirements described in Table 2-1.

SECTION 5 INSTALLATION OF SURVEY MARKERS

Figures 5-1 is a sketch of the markers that should have been installed at all of the PWR plants participating in the SRMP. These markers, which consist of a thin (20 gage) stainless steel plate backing, are attached to the outer insulation cover or stainless steel lagging with standard 1/8-inch diameter steel pop rivets.

The locations at which radiation levels are to be monitored are based on the following selection criteria:

- The location should be representative of the crud layer source on out-of-core surfaces and should reflect the behavior of these surface sources.
- The locations should be common from loop to loop and plant to plant to facilitate comparison of the various plants' radiation levels and buildup.
- The survey points should be easily accessible to plant personnel.
- The total number of survey points to be monitored should be limited so that the personnel exposures received during the survey are low.

Figure 5-2 shows recommended survey marker locations on the external surfaces of the reactor coolant loops. Although these locations are highly desirable, experience has shown that they are not always practical for installation of markers. For example, the lack of access to a certain point may require some modification of the survey point. In such cases, be sure to note any modifications in the location in the comments section of the survey form.

It is important that the locations be chosen in such a manner that the radiation level at those points is due to the RCS piping or the steam generator tubes, and not to adjacent pipe runs and crud traps (i.e., valves) which are located in the same general area. The influence of such outside sources can be minimized by performing a careful survey of the area to determine the source and magnitude of the background radiation levels.

The recommended procedure is as follows:

5.1.1. Choose a location in the approximate vicinity of each of the points illustrated in Figure 5-2.

Content deleted - EPRI Proprietary Information.

- HL1 - Bottom of hot leg piping between steam generator inlet and reactor vessel shield
- CL1 - Bottom of cold leg piping between reactor coolant pump and reactor vessel shield
- S1 - Outside of steam generator hot leg side, approximately 1 meter above top of channel head tube sheet and approximately midway between secondary side hand-hole cover and hot leg piping (90 degrees radially from the tube lane)
- S2 - Same as S1 but approximately midway between secondary side hand-hole cover and cold leg piping (90 degrees radially from the tube lane)

5.1.2. Monitor piping and valves near the prospective marker location to identify those which carry radioactive fluid and/or those which are potential crud traps. Potential crud traps are summarized below.

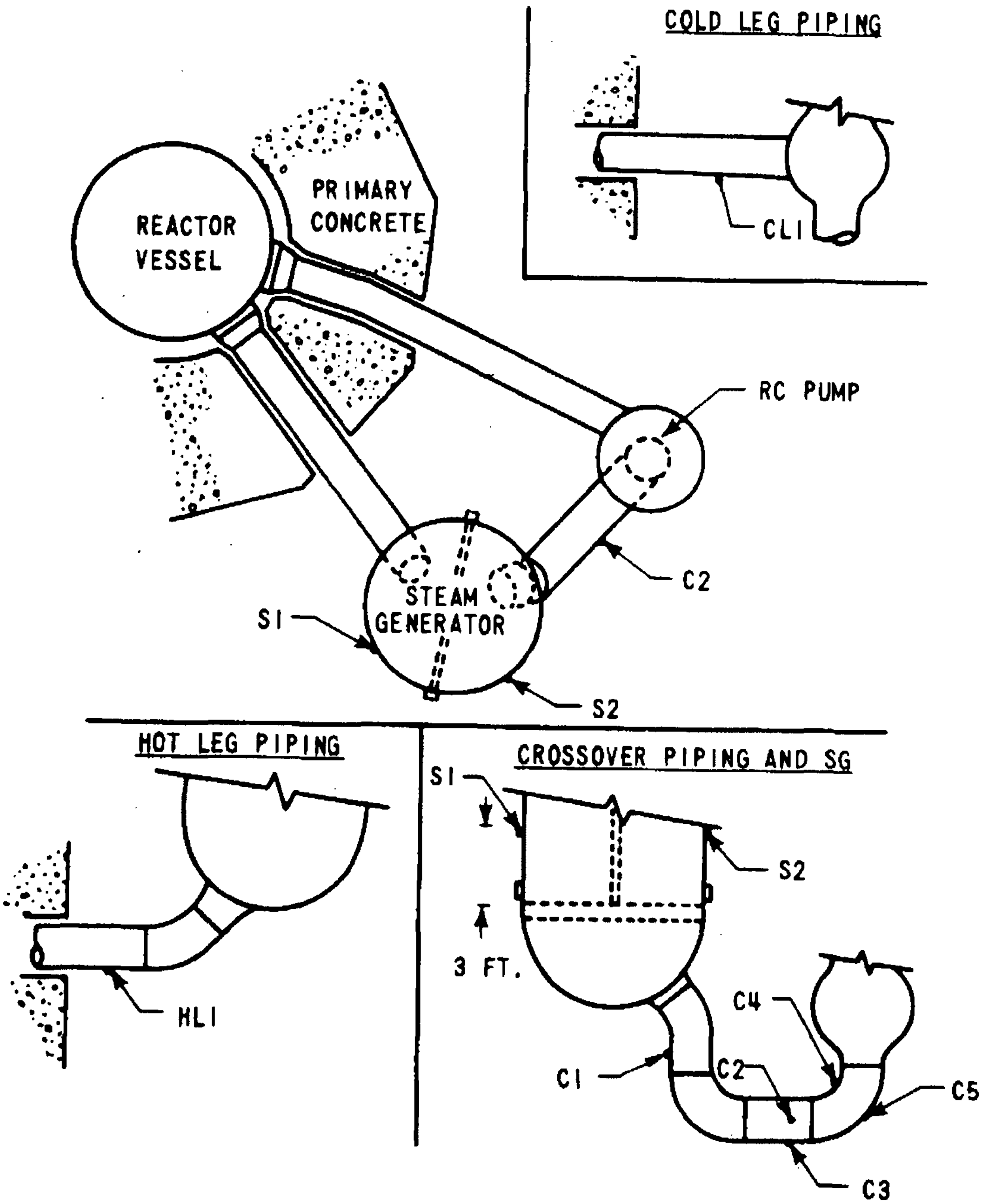
- Drain lines
- Pressurizer surge line
- Pressurizer spray line
- Sample/transmitter lines
- Loops stop valves
- Check valves
- RTD wells
- Loop drain valves
- Letdown lines
- RHR Lines
- Any low point in a pipe section

5.1.3. Estimate the contribution of such piping/valves to the total dose rate at the prospective marker location by surveys at various distances away from and around the background source.

5.1.4. Locate the marker at a position such that the dose contributions from background sources are minimized.

FIGURE 5-1. RADIATION SURVEY MARKER

FIGURE 5-2. REACTOR COOLANT LOOP SURVEY MARKER LOCATIONS WESTINGHOUSE DESIGNED PWR PLANTS



APPENDIX A - STANDARD DATA SHEETS

This Appendix contains copies of the EPRI PWR radiation survey data sheets. Please carefully and completely gather your data according to this General Procedure and then complete these data sheets.

Provide your input to EPRI at the address listed below.

Samuel Choi

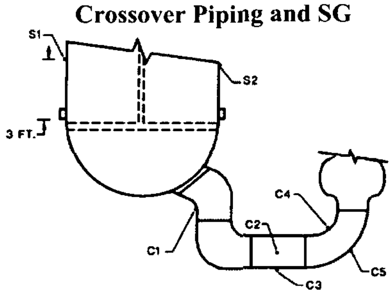
EPRI
3420 Hillview Ave
Palo Alto, CA 94304
Tel: (650) 855-2940
schoi@epri.com

EPRI-PWR STANDARD RADIATION MONITORING PROCEDURE
DATA SHEET 1
REACTOR COOLANT LOOP PIPING SURVEY – WESTINGHOUSE DESIGNED PLANT

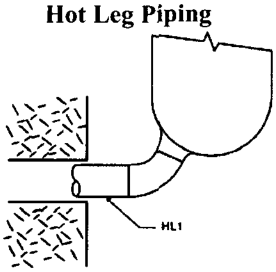
PLANT / UNIT			
SHUTDOWN (Breaker Open)	DATE	**SURVEY**	DATE
	TIME		TIME

Plant Condition			
Reactor Coolant Piping	Full	Empty	% Drained =________ (if applicable)
Steam Generator Primary Side	Full	Empty	% Drained = ________(if applicable)
Steam Generator Secondary Side	Full	Empty	% Drained = ________(if applicable)
Survey Performed By			

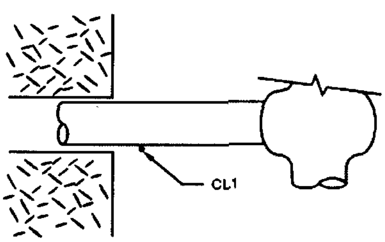
Crossover Piping and SG



Hot Leg Piping



Cold Leg Piping



RADIATION SURVEY DATA

			CONTACT DOSE RATE AT MARKER (mR/hr)			
Survey Point	**Survey Instrument Type**	**ED Bias**	**Loop #**	**Loop #**	**Loop #**	**Loop #**
S1 (c)						
S2 (c)						
C1 (b)						
C2 (a)						
C3 (b)						
C4 (b)						
C5 (b)						
HL1 (a)						
CL1 (a)						

Comments:

a. Required Points

b. Recommended Points

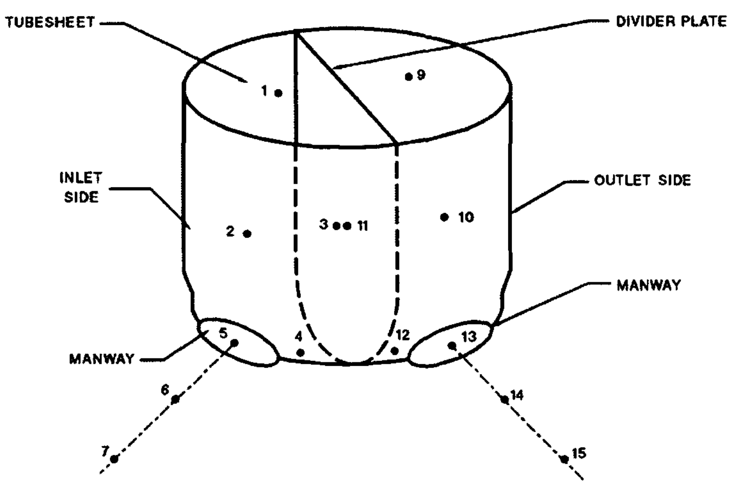
c. Strongly Recommended if taken previously

EPRI-PWR STANDARD RADIATION MONITORING PROCEDURE
DATA SHEET 2
STEAM GENERATOR CHANNEL HEAD SURVEY – WESTINGHOUSE DESIGNED PLANT

PLANT / UNIT				
SHUTDOWN (Breaker Open)	DATE	**SURVEY**	DATE	
	TIME		TIME	

Survey Instrument Type	
ED Bias (if applicable)	
Survey Performed By	

Steam Generator Channel Head (Manway Cover Removed)



RADIATION SURVEY DATA

DOSE RATE AT SURVEY POINT (R/hr)					
Survey *Point*	**Location**	**Loop #**	**Loop #**	**Loop #**	**Loop #**
	Survey Date for Each Loop				
Inlet Channel (Hot Leg)					
1	Midpoint of Tubesheet (a)				
2	Channel Head Center (a)				
3	Center Divider Plate (a)				
4	Bottom of Channel Head (a)				
5	Manway Entrance (b)				
6	30 cm from Manway (b)				
7	One meter from Manway (b)				
Outlet Channel (Cold Leg)					
9	Midpoint of Tubesheet (a)				
10	Channel Head Center (a)				
11	Center Divider Plate (a)				
12	Bottom of Channel Head (a)				
13	Manway Entrance (b)				
14	*30 cm from Manway (b)*				
15	One meter from Manway (b)				

Comments: a. Required Points b. Recommended Points
*** Note if TLDs were used to obtain data**

EPRI-PWR STANDARD RADIATION MONITORING PROCEDURE
DATA SHEET 3 (OPTIONAL)
OPTIONAL INFORMATION POINTS SURVEY – WESTINGHOUSE DESIGNED PLANT

PLANT / UNIT			
SHUTDOWN (Breaker Open)	DATE	**SURVEY**	DATE
	TIME		TIME

Optional Information Points Survey				
Survey Point	**Location where data is taken**	**Survey Instrument Type**	**ED Bias**	**DOSE RATE (mR/hr)**
Letdown Piping				
CVCS Heat Exchanger (On the Shell)				
RHR Piping				
RHR Heat Exchangers (on the shell)				
Refueling Water Surface				
Comments:				

Dose Rate at Survey Point Before Forced Oxidation (mR/hr)						
Survey Point	**Survey Instrument Type**	**ED Bias**	**Loop #**	**Loop #**	**Loop #**	**Loop #**
S1						
S2						
C1						
C2						
C3						
C4						
C5						
HL1						
CL1						

D
REVISED SRMP PROCEDURE FOR COMBUSTION ENGINEERING PLANTS

RP20060018
March 2006

EPRI SRMP GENERAL PROCEDURE FOR COMBUSTION ENGINEERING DESIGNED PWR PLANTS

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RP20060018
March 2006

EPRI SRMP GENERAL PROCEDURE FOR COMBUSTION ENGINEERING DESIGNED PWR PLANTS

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SECTION 1 INTRODUCTION

1.1 BACKGROUND

The Standard Radiation Monitoring Program (SRMP), first instituted in 1978, is part of a more general program, sponsored by the Electric Power Research Institute (EPRI), with the major emphasis on improving plant reliability and availability. The purpose of this program is as follows:

- To provide a meaningful, consistent, and systematic approach to monitoring the rate of PWR radiation field buildup and to provide the basis for projecting the trend of those fields.
- To provide a reliable set of radiation field data for each participating plant, from which comparisons can be made.
- To monitor certain plant parameters that affect or may affect observed radiation fields.
- To use the information from this program to identify plant design features, material selection, and operational techniques that present opportunities for radiation control.

Previous EPRI reports have been published as a result of the SRMP program. The reports list the factors that affect plant dose rates and quantitatively evaluate the effect of these factors. The most important factors were found to be operational coolant chemistry and variations in cobalt input.

The SRMP program had consistent data collection efforts for Westinghouse and Combustion Engineering plants through 1996 and 1985 respectively; afterwards, SRMP data collection had been limited primarily to plants that had implemented zinc injection or replaced steam generators with low-cobalt tubing. The EPRI Chemistry and LLW TAC asked to start these data collection efforts because recent trends in replacement steam generators, core uprating, adverse radiological incidents, and various changes in shutdown and normal chemistry procedures have caused unpredictable fluctuations in dose rates throughout the ex-core surfaces.

This revision to the SRMP General Procedure updates the survey procedures and forms. The procedure has been streamlined by defining survey locations as 'Required Points,' 'Recommended Points,' and 'Optional Information.' The definitions of these terms are below:

- Required points are those that must be taken.
- Recommended points are those that are requested, but may be skipped in cases of personnel safety, poor accessibility, or significant ALARA impact.
- Optional information is information that is requested if available.

Reactor coolant chemistry data are collected under the PWR Chemistry Performance Monitoring and Assessment (PWR CPMA) program. To avoid confusion because of the

differences between plants, this General Procedure has been revised to have separate procedures for each plant designed by Westinghouse, Combustion Engineering, and Babcock and Wilcox. This document contains information on the radiation survey procedures for Combustion Engineering designed plants.

1.2 SUMMARY

The procedures provide a controlled measurement program for assessing radiation field trends of Reactor Coolant System (RCS) components. The Standard Radiation Monitoring Program (SRMP) involves the performance of radiation surveys at well-defined locations and recording certain plant conditions. The data gathered in the surveys will lead to a better understanding of the parameters that influence RCS radiation fields. This information will, in turn, provide the potential for reducing plant radiation fields.

The radiation surveys are conducted during plant shutdowns and collect dose rate readings at permanent markers located on the outside surfaces of RCS components. Surveys are also specified for the internal surfaces of the steam generator channel heads when maintenance or inspection activities are planned. In addition to the radiation survey information, certain coolant chemistry data that may be significant in relation to the deposition, release, and transport of activated corrosion products are to be collected under the PWR CPMA program.

In the specification of data requirements and survey frequencies, an attempt has been made to minimize the additional workload on operating personnel. To reach this goal, this General Procedure has been revised in an effort to more clearly define the data requirements (i.e., 'Required Points', 'Recommended Points', and 'Optional Information'), to eliminate the collection of redundant data, and to streamline the data collection forms.

EPRI Solutions will compile and reduce the survey data for the duration of the program. As the processes and parameters that have a major impact on radiation levels are more clearly defined and understood, the data specification may be modified as indicated by the program results.

SECTION 2 DATA REQUIREMENTS

2.1 GENERAL

The data requirements of the Standard Radiation Monitoring Program fall into two basic categories, i.e., 1) radiation surveys and 2) primary coolant chemistry regimes. The primary coolant chemistry information will be collected under the PWR CPMA program. The radiation survey parameters to be monitored and schedules for each of the measurements are discussed below. However, it should be emphasized that any additional observations by plant personnel of conditions, occurrences, or other items which might affect radiation levels or which would provide further insight or understanding of those levels should be recorded. Accordingly, the data input sheets allow for entry of more of this type of information.

2.2 RADIATION SURVEYS

Radiation surveys (at permanent markers located on the outer surfaces of RCS loop components) are to be performed during plant shutdown, except when the duration of the outage is too short to make the survey practical. Radiation surveys of the steam generator channel heads should also be recorded if primary side inspection or maintenance of steam generators is scheduled during a plant shutdown. The recommended frequencies for performing the radiation surveys are outlined in Table 2-1. In the interest of getting useful and consistent data for this program, a list of the more relevant data requirements/screening criteria is included in the table.

Guidelines and information relative to the installation and location of the survey markers are included in Section 5. Forms for recording the reactor coolant loop and steam generator survey results are reproduced in Appendix A. As noted on the Appendix A data sheets, the total number of survey locations is small (less than 10 locations per reactor coolant loop). Since in-containment radiation surveys are generally performed during shutdown conditions, the additional burden on plant personnel and associated radiation exposure is expected to be minimal. Further, some operating plants have implemented electronic dosimetry at selected locations. If electronic dosimetry instruments are used for contact dose rate monitoring, instrument bias information should be noted in the survey forms. Such equipment can provide dose rate data on a continuous basis that can be recorded remotely with a minimum of radiation exposure to plant personnel.

TABLE 2-1. DATA REQUIREMENTS FOR COMBUSTION ENGINEERING PLANTS

SHUTDOWN RADIATION SURVEYS		
LOCATION	FREQUENCY	ASSUMPTIONS
Reactor Coolant Loop Piping	Once after shutdown plant evolutions associated with crud solubilization (i.e., hydrogen peroxide addition or other coolant oxygenation measures) have been completed. Surveys should be taken as soon as possible after the forced oxidation cleanup end point concentration limit in the primary coolant is reached (e.g. 0.05 μCi/g) **Recommended Survey:** If possible, once after the plant is at zero power for at least 12 hours and prior to primary coolant oxygenation evolutions.	1. Surveys performed with a directional shielded probe such as the Eberline HP-220A should be noted clearly. 2. For surveys in contact with the steam generator shell insulation, the RCS should preferably be 100% full and the secondary side should be ≥ 20% full. 3. For surveys in contact with primary piping, the RCS loop should preferably be 100% full. 4. Survey data affected by local hot spots should be noted clearly. 5. Survey data taken while the steam generator manway covers are not in place should be noted clearly.
Steam Generator Channel Head	Prior to primary side inspection or maintenance of the steam generator. Additional surveys as required by plant staff or plant procedures.	1. Surveys performed with a directional shielded probe such as the Eberline HP-220A should be noted clearly. 2. Surveys performed prior to and during the first refueling outage will only be considered if they occur within the first three (3) weeks following shutdown.

SECTION 3 RADIATION SURVEY INSTRUCTIONS

3.1 REACTOR COOLANT LOOP PIPING SURVEY PROCEDURES

Plant personnel should perform the shutdown radiation surveys of the reactor coolant loops in accordance with the schedule and requirements presented in Table 2-1. The results should be recorded on an appropriate survey form included in Appendix A. Note that the reactor coolant loop shutdown survey should preferably be performed with the steam generator manway covers in place.

In order to make the surveys be more consistent and effective throughout the industry, the SRMP working committee defined survey locations as 'Required Points', 'Recommended Points', and 'Optional Information'. Required points are those that must be taken. Recommended points are those that are requested, but may be skipped in cases of personnel safety, unavailable resources, poor accessibility, or significant ALARA impact. Optional information is information that is requested if available. The reactor coolant loop piping survey locations are summarized below.

3.1.1. Required Points

- SL2 - Straight section of crossover piping, side of pipe (generally away from primary concrete shield)
- HL1 - Bottom of hot leg piping between steam generator inlet and reactor vessel shield
- CL1 - Bottom of cold leg piping between reactor coolant pump and reactor vessel shield

3.1.2. Recommended Points

- SL1 - Above crossover piping elbow, midway along vertical section of piping from the steam generator
- SL3 - Straight section of crossover piping, bottom
- SL4 - Crossover piping elbow to RCP, midway along inside radius
- SL5 - Crossover piping elbow to RCP, midway along outside radius

- S1 - Outside of steam generator hot leg side, approximately 1 meter above top of channel head tube sheet and approximately midway between secondary side hand-hole cover and hot leg piping (90 degrees radially from the tube lane)
- S2 - Same as S1 but approximately midway between secondary side hand-hole cover and cold leg piping (90 degrees radially from the tube lane)

Note: S1 and S2 are strongly recommended if taken previously

3.1.3. Optional Information Points

Note: Specify location of measurements, e. g., on letdown piping, one foot downstream of regenerative heat exchanger

- Letdown piping
- CVCS heat exchanger (on the shell)
- RHR piping
- RHR heat exchangers (on the shell)
- Refueling water surface

The survey instrument to be employed in the surveys should generally be equipped with an extending or telescoping probe that encompasses the range from 2 milliroentgens per hour to 20 roentgens per hour. Battery and response checks of the portable monitoring equipment employed should be performed prior to the survey. It must be ascertained that the instrument has been calibrated according to the calibration procedures and manufacturer's recommendations for that particular instrument. It is desirable to utilize portable monitoring equipment that has been calibrated by exposure to known gamma radiation fields (e.g. a commercially built calibrator equipped with radioactive sources traceable to NIST standards). The instrumentation used for this procedure should be properly calibrated according to plant specific or utility specific procedures. Any special features or conditions associated with the equipment should be noted in the survey form.

Also, as noted in Section 2-2, electronic dosimetry at survey locations can be used in lieu of manual surveys using portable survey equipment. In this case, instrument bias information should be noted in the survey form.

3.2 STEAM GENERATOR CHANNEL HEAD SURVEY PROCEDURES

If access to the steam generator channel head(s) occurs during the shutdown period, the results of the channel head survey are to be recorded on an appropriate survey form included in Appendix A. The steam generator channel head survey locations are summarized below.

3.2.1. Required Points

- Midpoint of Tubesheet (Hot Leg & Cold Leg)
- Channel Head Center (Hot Leg & Cold Leg)
- Center of Stay Cylinder (Hot Leg & Cold Leg)
- Bottom of Channel Head (Hot Leg & Cold Leg)
- Manway Entrance (Hot Leg & Cold Leg)
- 30 centimeter from Manway (Hot Leg & Cold Leg)

3.2.2. Recommended Points

- One meter from Manway (Hot Leg & Cold Leg)

SECTION 4 PRECAUTIONS

The radiation surveys should be conducted in accordance with the following precautions.

- The marker locations relative to a fixed reference point should be recorded when the markers are first installed. It may be necessary to reinstall markers on insulation removed for in-service inspection or maintenance of the component; the fixed reference point notation will allow positioning at the initially established location.
- Radiation surveys of the reactor coolant loop should not be performed with steam generator manway covers removed because the background dose rate from the steam generator channel heads has been found to influence the loop piping measurements. If not possible, a notation should be made on the form that the manway covers were removed.
- If the results of the radiation survey indicate marked inconsistencies with previous survey results and/or if damage to the survey instrument is suspected, the instrument calibration should be rechecked after completion of the survey.
- Prior to entering the survey locations, worker should note hotspots near by. Potential crud traps are summarized below.
    - Drain lines
    - Pressurizer surge line
    - Pressurizer spray line
    - Sample/transmitter lines
    - Loops stop valves
    - Check valves
    - RTD wells
    - Loop drain valves
    - Letdown lines
    - RHR Lines
    - Any low point in a pipe section
- Also note the Data Requirements described in Table 2-1.

SECTION 5 INSTALLATION OF SURVEY MARKERS

Figures 5-1 is a sketch of the markers that should have been installed at all of the PWR plants participating in the SRMP. These markers, which consist of a thin (20 gage) stainless steel plate backing, are attached to the outer insulation cover or stainless steel lagging with standard 1/8-inch diameter steel pop rivets.

The locations at which radiation levels are to be monitored are based on the following selection criteria:

- The location should be representative of the crud layer source on out-of-core surfaces and should reflect the behavior of these surface sources.
- The locations should be common from loop to loop and plant to plant to facilitate comparison of the various plants' radiation levels and buildup.
- The survey points should be easily accessible to plant personnel.
- The total number of survey points to be monitored should be limited so that the personnel exposures received during the survey are low.

Figure 5-2 shows recommended survey marker locations on the external surfaces of the reactor coolant loops. Although these locations are highly desirable, experience has shown that they are not always practical for installation of markers. For example, the lack of access to a certain point may require some modification of the survey point. In such cases, be sure to note any modifications in the location in the comments section of the survey form.

It is important that the locations be chosen in such a manner that the radiation level at those points is due to the RCS piping or the steam generator tubes, and not to adjacent pipe runs and crud traps (i.e., valves) which are located in the same general area. The influence of such outside sources can be minimized by performing a careful survey of the area to determine the source and magnitude of the background radiation levels.

The recommended procedure is as follows:

5.1.1. Choose a location in the approximate vicinity of each of the points illustrated in Figure 5-2.

- SL1 - Above crossover piping elbow, midway along vertical section of piping from the steam generator
- SL2 - Straight section of crossover piping, side of pipe (generally away from primary concrete shield)
- SL3 - Straight section of crossover piping, bottom
- SL4 - Crossover piping elbow to RCP, midway along inside radius
- SL5 - Crossover piping elbow to RCP, midway along outside radius

- HL1 - Bottom of hot leg piping between steam generator inlet and reactor vessel shield
- CL1 - Bottom of cold leg piping between reactor coolant pump and reactor vessel shield
- S1 - Outside of steam generator hot leg side, approximately 1 meter above top of channel head tube sheet and approximately midway between secondary side hand-hole cover and hot leg piping (90 degrees radially from the tube lane)
- S2 - Same as S1 but approximately midway between secondary side hand-hole cover and cold leg piping (90 degrees radially from the tube lane)

5.1.2. Monitor piping and valves near the prospective marker location to identify those which carry radioactive fluid and/or those which are potential crud traps. Potential crud traps are summarized below.

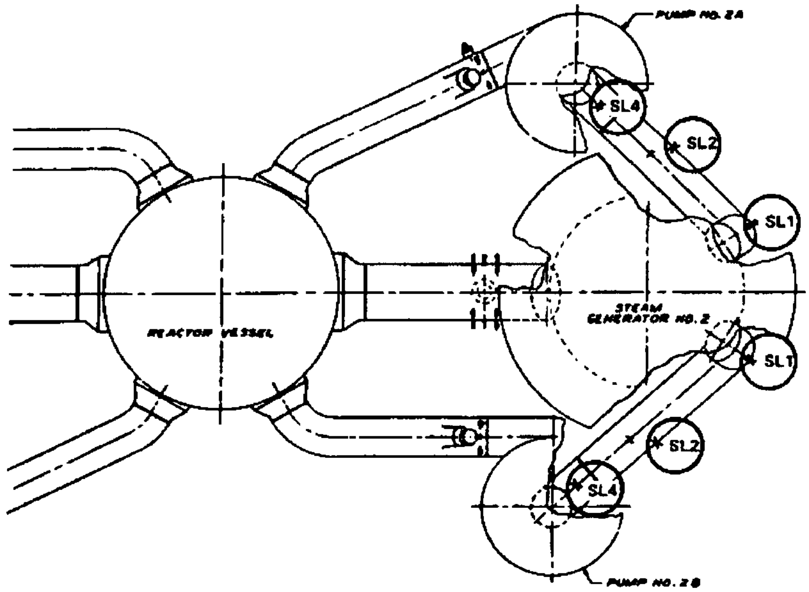
- Drain lines
- Pressurizer surge line
- Pressurizer spray line
- Sample/transmitter lines
- Loops stop valves
- Check valves
- RTD wells
- Loop drain valves
- Letdown lines
- RHR Lines
- Any low point in a pipe section

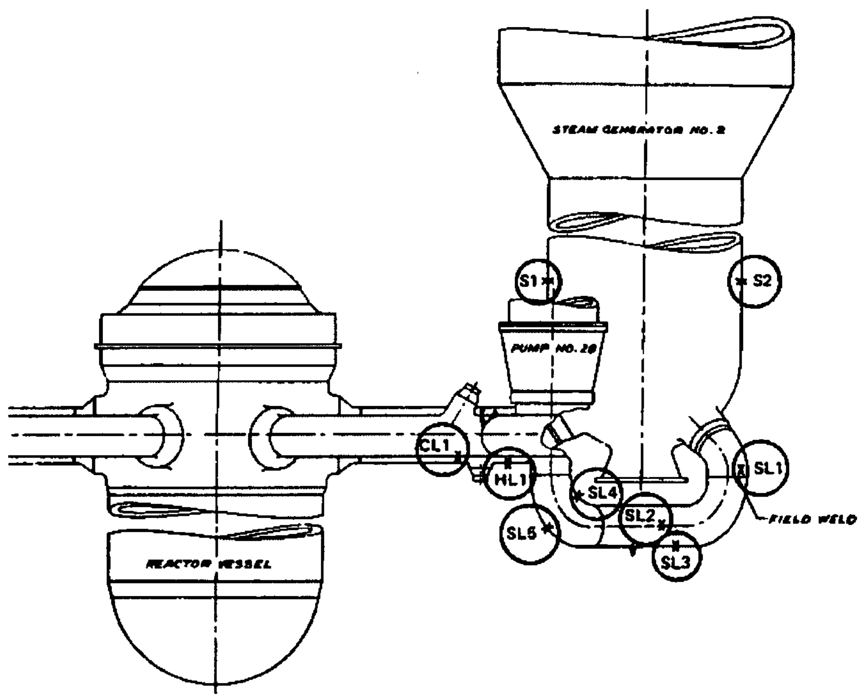
5.1.3. Estimate the contribution of such piping/valves to the total dose rate at the prospective marker location by surveys at various distances away from and around the background source.

5.1.4. Locate the marker at a position such that the dose contributions from background sources are minimized.

FIGURE 5-1. RADIATION SURVEY MARKER

FIGURE 5-2. REACTOR COOLANT LOOP SURVEY MARKER LOCATIONS COMBUSTION ENGINEERING DESIGNED PWR PLANTS





APPENDIX A - STANDARD DATA SHEETS

This Appendix contains copies of the EPRI PWR radiation survey data sheets. Please carefully and completely gather your data according to this General Procedure and then complete these data sheets.

Provide your input to EPRI at the address listed below.

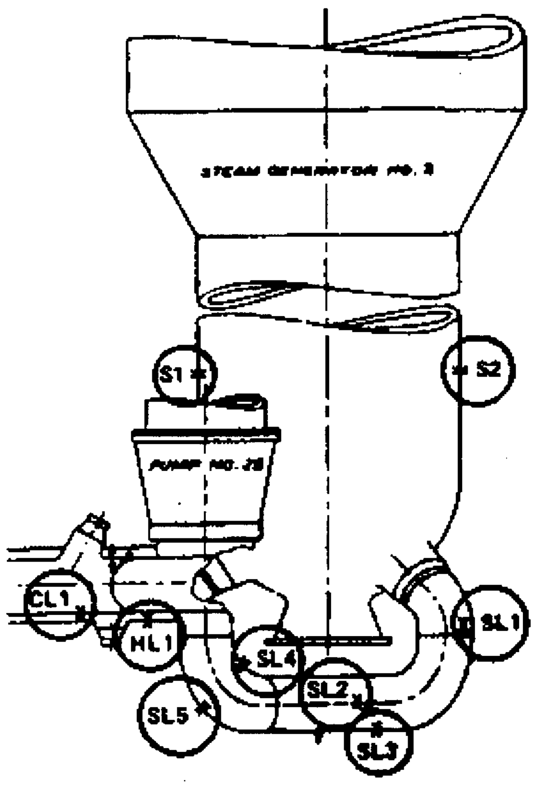
Samuel Choi

EPRI
3420 Hillview Ave
Palo Alto, CA 94304
Tel: (650) 855-2940
schoi@epri.com

EPRI-PWR STANDARD RADIATION MONITORING PROCEDURE
DATA SHEET 1
REACTOR COOLANT LOOP PIPING SURVEY – CE DESIGNED PLANT

PLANT / UNIT			
SHUTDOWN (Breaker Open)	DATE	**SURVEY**	DATE
	TIME		TIME

Plant Condition			
Reactor Coolant Piping	Full	Empty	% Drained =_________ (if applicable)
Steam Generator Primary Side	Full	Empty	% Drained = _________(if applicable)
Steam Generator Secondary Side	Full	Empty	% Drained = _________(if applicable)
Survey Performed By			



RADIATION SURVEY DATA

			CONTACT DOSE RATE AT MARKER (mR/hr)			
			Steam Generator #		**Steam Generator #**	
Survey Point	**Survey Instrument Type**	**ED Bias**	**Loop #**	**Loop #**	**Loop #**	**Loop #**
S1 (c)						
S2 (c)						
SL1 (b)						
SL2 (a)						
SL3 (b)						
SL4 (b)						
SL5 (b)						
HL1 (a)						
CL1 (a)						

Comments:

a. Required Points

b. Recommended Points

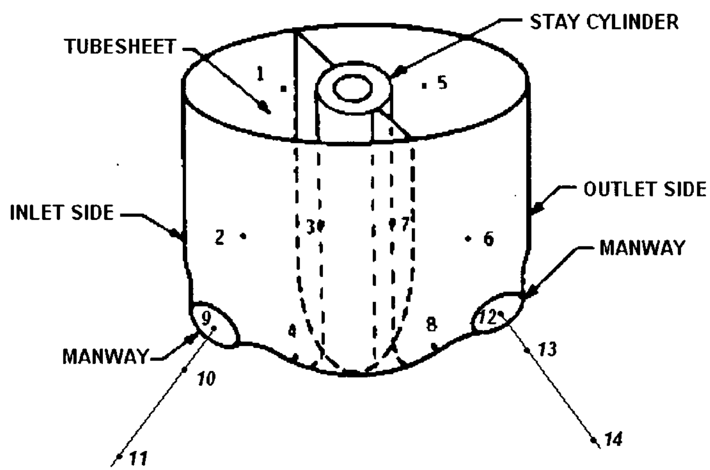
c. Strongly recommended if taken previously

EPRI-PWR STANDARD RADIATION MONITORING PROCEDURE
DATA SHEET 2
STEAM GENERATOR CHANNEL HEAD SURVEY – CE DESIGNED PLANT

PLANT / UNIT			
SHUTDOWN (Breaker Open)	DATE	**SURVEY**	DATE
	TIME		TIME

Survey Instrument Type	
ED Bias (if applicable)	
Survey Performed By	

Steam Generator Channel Head (Manway Cover Removed)



RADIATION SURVEY DATA

DOSE RATE AT SURVEY POINT (R/hr)			
Survey Point	**Location**	**Steam Generator #**	**Steam Generator #**
	Survey Date for Each Loop		
Inlet Channel (Hot Leg)			
1	Midpoint of Tubesheet (a)		
2	Channel Head Center (a)		
3	Center of Stay Cylinder (a)		
4	Bottom of Channel Head (a)		
9	Manway Entrance (a)		
10	30 cm from Manway (a)		
11	One meter from Manway (b)		
Outlet Channel (Cold Leg)			
5	Midpoint of Tubesheet (a)		
6	Channel Head Center (a)		
7	Center of Stay Cylinder (a)		
8	Bottom of Channel Head (a)		
12	Manway Entrance (a)		
13	30 cm from Manway (a)		
14	One meter from Manway (b)		

Comments: a. Required Points b. Recommended Points
* Note if TLDs were used to obtain data

EPRI-PWR STANDARD RADIATION MONITORING PROCEDURE
DATA SHEET 3 (OPTIONAL)
OPTIONAL INFORMATION POINTS SURVEY – CE DESIGNED PLANT

PLANT / UNIT			
SHUTDOWN (Breaker Open)	DATE	**SURVEY**	DATE
	TIME		TIME

Optional Information Points Survey

DOSE RATE AT SURVEY POINT (mR/hr)				
Survey Point	**Location where data is taken**	**Survey Instrument Type**	**ED Bias**	**DOSE RATE (mR/hr)**
Letdown Piping				
CVCS Heat Exchanger (On the Shell)				
RHR Piping				
RHR Heat Exchangers (On the shell)				
Refueling Water Surface				
Comments:				

Dose Rate at Survey Point Before Forced Oxidation (mR/hr)						
			Steam Generator #		**Steam Generator #**	
Survey Point	**Survey Instrument Type**	**ED Bias**	**Loop #**	**Loop #**	**Loop #**	**Loop #**
S1						
S2						
SL1						
SL2						
SL3						
SL4						
SL5						
HL1						
CL1						

E

REVISED SRMP PROCEDURE FOR BABCOCK AND WILCOX PLANTS

RP20060019
March 2006

EPRI SRMP GENERAL PROCEDURE FOR BABCOCK AND WILCOX DESIGNED PWR PLANTS

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RP20060019
March 2006

EPRI SRMP GENERAL PROCEDURE FOR BABCOCK AND WILCOX DESIGNED PWR PLANTS

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SECTION 1 INTRODUCTION

1.1 BACKGROUND

The Standard Radiation Monitoring Program (SRMP), first instituted in 1978, is part of a more general program, sponsored by the Electric Power Research Institute (EPRI), with the major emphasis on improving plant reliability and availability. The purpose of this program is as follows:

- To provide a meaningful, consistent, and systematic approach to monitoring the rate of PWR radiation field buildup and to provide the basis for projecting the trend of those fields.
- To provide a reliable set of radiation field data for each participating plant, from which comparisons can be made.
- To monitor certain plant parameters that affect or may affect observed radiation fields.
- To use the information from this program to identify plant design features, material selection, and operational techniques that present opportunities for radiation control.

Previous EPRI reports have been published as a result of the SRMP program. The reports list the factors that affect plant dose rates and quantitatively evaluate the effect of these factors. The most important factors were found to be operational coolant chemistry and variations in cobalt input.

The SRMP program had consistent data collection efforts for Westinghouse and Combustion Engineering plants through 1996 and 1985 respectively; afterwards, SRMP data collection had been limited primarily to plants that had implemented zinc injection or replaced steam generators with low-cobalt tubing. The EPRI Chemistry and LLW TAC asked to start these data collection efforts because recent trends in replacement steam generators, core uprating, adverse radiological incidents, and various changes in shutdown and normal chemistry procedures have caused unpredictable fluctuations in dose rates throughout the ex-core surfaces.

This revision to the SRMP General Procedure updates the survey procedures and forms. The procedure has been streamlined by defining survey locations as 'Required Points,' 'Recommended Points,' and 'Optional Information.' The definitions of these terms are below:

- Required points are those that must be taken.
- Recommended points are those that are requested, but may be skipped in cases of personnel safety, poor accessibility, or significant ALARA impact.
- Optional information is information that is requested if available.

Reactor coolant chemistry data are collected under the PWR Chemistry Performance Monitoring and Assessment (PWR CPMA) program. To avoid confusion because of the

differences between plants, this General Procedure has been revised to have separate procedures for each plant designed by Westinghouse, Combustion Engineering, and Babcock and Wilcox. This document contains information on the radiation survey procedures for Babcock and Wilcox designed plants.

1.2 SUMMARY

The procedures provide a controlled measurement program for assessing radiation field trends of Reactor Coolant System (RCS) components. The Standard Radiation Monitoring Program (SRMP) involves the performance of radiation surveys at well-defined locations and recording certain plant conditions. The data gathered in the surveys will lead to a better understanding of the parameters that influence RCS radiation fields. This information will, in turn, provide the potential for reducing plant radiation fields.

The radiation surveys are conducted during plant shutdowns and collect dose rate readings at permanent markers located on the outside surfaces of RCS components. Surveys are also specified for the internal surfaces of the steam generator channel heads when maintenance or inspection activities are planned. In addition to the radiation survey information, certain coolant chemistry data that may be significant in relation to the deposition, release, and transport of activated corrosion products are to be collected under the PWR CPMA program.

In the specification of data requirements and survey frequencies, an attempt has been made to minimize the additional workload on operating personnel. To reach this goal, this General Procedure has been revised in an effort to more clearly define the data requirements (i.e., 'Required Points', 'Recommended Points', and 'Optional Information'), to eliminate the collection of redundant data, and to streamline the data collection forms.

EPRI Solutions will compile and reduce the survey data for the duration of the program. As the processes and parameters that have a major impact on radiation levels are more clearly defined and understood, the data specification may be modified as indicated by the program results.

SECTION 2 DATA REQUIREMENTS

2.1 GENERAL

The data requirements of the Standard Radiation Monitoring Program fall into two basic categories, i.e., 1) radiation surveys and 2) primary coolant chemistry regimes. The primary coolant chemistry information will be collected under the PWR CPMA program. The radiation survey parameters to be monitored and schedules for each of the measurements are discussed below. However, it should be emphasized that any additional observations by plant personnel of conditions, occurrences, or other items which might affect radiation levels or which would provide further insight or understanding of those levels should be recorded. Accordingly, the data input sheets allow for entry of more of this type of information.

2.2 RADIATION SURVEYS

Radiation surveys (at permanent markers located on the outer surfaces of RCS loop components) are to be performed during plant shutdown, except when the duration of the outage is too short to make the survey practical. Radiation surveys of the steam generator channel heads should also be recorded if primary side inspection or maintenance of steam generators is scheduled during a plant shutdown. The recommended frequencies for performing the radiation surveys are outlined in Table 2-1. In the interest of getting useful and consistent data for this program, a list of the more relevant data requirements/screening criteria is included in the table.

Guidelines and information relative to the installation and location of the survey markers are included in Section 5. Forms for recording the reactor coolant loop and steam generator survey results are reproduced in Appendix A. As noted on the Appendix A data sheets, the total number of survey locations is small (less than 10 locations per reactor coolant loop). Because in-containment radiation surveys are generally performed during shutdown conditions, the additional burden on plant personnel and associated radiation exposure is expected to be minimal. Further, some operating plants have implemented electronic dosimetry at selected locations. If electronic dosimetry instruments are used for contact dose rate monitoring, instrument bias information should be noted in the survey forms. Such equipment can provide dose rate data on a continuous basis that can be recorded remotely with a minimum of radiation exposure to plant personnel.

TABLE 2-1. DATA REQUIREMENTS FOR BABCOCK AND WILCOX PLANTS

SHUTDOWN RADIATION SURVEYS		
LOCATION	FREQUENCY	ASSUMPTIONS
Reactor Coolant Loop Piping	Once after shutdown plant evolutions associated with crud solubilization (i.e., hydrogen peroxide addition or other coolant oxygenation measures) have been completed. Surveys should be taken as soon as possible after the forced oxidation cleanup end point concentration limit in the primary coolant is reached (e.g. 0.05 µCi/g). If surveys are taken prior to forced oxidation or while running RCPs, it should be noted as such. **Recommended Survey:** If possible, once after the plant is at zero power for at least 12 hours and prior to primary coolant oxygenation evolutions.	1. Surveys performed with a directional shielded probe such as the Eberline HP-220A should be noted clearly. 2. For surveys in contact with the steam generator shell insulation, the RCS level should be above the point of the survey, preferably 100% full and the secondary side should be ≥ 20% full. 3. For surveys in contact with primary piping, the RCS loop should preferably be above the point of the survey. 4. Survey data affected by local hot spots should be noted clearly. 5. Survey data taken while the steam generator manway covers are not in place should be noted clearly.
Steam Generator Channel Head	Prior to primary side inspection or maintenance of the steam generator. Additional surveys as required by plant staff or plant procedures.	1. Surveys performed with a directional shielded probe such as the Eberline HP-220A should be noted clearly. 2. Surveys performed prior to and during the first refueling outage will only be considered if they occur within the first three (3) weeks following shutdown.

SECTION 3 RADIATION SURVEY INSTRUCTIONS

3.1 REACTOR COOLANT LOOP PIPING SURVEY PROCEDURES

Plant personnel should perform the shutdown radiation surveys of the reactor coolant loops in accordance with the schedule and requirements presented in Table 2-1. The results should be recorded on an appropriate survey form included in Appendix A. Note that the reactor coolant loop shutdown survey should preferably be performed with the steam generator manway covers in place.

In order to make the surveys be more consistent and effective throughout the industry, the SRMP working committee defined survey locations as 'Required Points', 'Recommended Points', and 'Optional Information'. Required points are those that must be taken. Recommended points are those that are requested, but may be skipped in cases of personnel safety, unavailable resources, poor accessibility, or significant ALARA impact. Optional information is information that is requested if available. The reactor coolant loop piping survey locations are summarized below.

3.1.1. Required Points

- JL1/JL2 - 30 centimeters above the elbow of the cold leg (J-leg) opposite the steam generator (Two points required for each steam generator)

3.1.2. Recommended Points

- D1/D2 - Opposite of the high pressure injection (HPI) nozzle (Two points for each steam generator)
- HL1 - Top of the hot leg elbow (One point for each steam generator)

3.1.3. Optional Information Points

Note: Specify location of measurements, e. g., on decay heat piping, one foot downstream of regenerative heat exchanger

- Decay heat piping
- Cooler inlet piping
- Decay heat pump suction
- Seismic plate (reactor vessel head service structure)

The survey instrument to be employed in the surveys should generally be equipped with an extending or telescoping probe that encompasses the range from 2 milliroentgens per hour to 20 roentgens per hour. Battery and response checks of the portable monitoring equipment employed should be performed prior to the survey. It must be ascertained that the instrument has been calibrated according to the calibration procedures and manufacturer's recommendations for that particular instrument. It is desirable to utilize portable monitoring equipment that has been calibrated by exposure to known gamma radiation fields (e.g. a commercially built calibrator equipped with radioactive sources traceable to NIST standards). The instrumentation used for this procedure should be properly calibrated according to plant specific or utility specific procedures. Any special features or conditions associated with the equipment should be noted in the survey form.

Also, as noted in Section 2-2, electronic dosimetry at survey locations can be used in lieu of manual surveys using portable survey equipment. In this case, instrument bias information should be noted in the survey form.

3.2 STEAM GENERATOR CHANNEL HEAD SURVEY PROCEDURES

If access to the steam generator channel head(s) occurs during the shutdown period, the results of the channel head survey are to be recorded on an appropriate survey form included in Appendix A. The steam generator channel head survey locations are summarized below.

3.2.1. Required Points

- Highest General Area Dose Rate of Four Quadrants and Contact with the Center of the Tubesheet (Upper Bowl & Lower Bowl)
- 30 centimeters Above Highest Dose Rate Tubesheet Point of Upper Bowl
- 30 centimeters Below Highest Dose Rate Tubesheet Point of Lower Bowl
- Plane of Manway (Upper Bowl & Lower Bowl)
- 30 centimeters outside of Plane of Manway (Upper Bowl & Lower Bowl)

3.2.2. Recommended Points

- One meter outside Plane of Manway (Upper Bowl & Lower Bowl)
- Three meters above top of the Upper Manway

SECTION 4 PRECAUTIONS

The radiation surveys should be conducted in accordance with the following precautions.

- The marker locations relative to a fixed reference point should be recorded when the markers are first installed. It may be necessary to reinstall markers on insulation removed for in-service inspection or maintenance of the component; the fixed reference point notation will allow positioning at the initially established location.

- Radiation surveys of the reactor coolant loop should not be performed with steam generator manway covers removed because the background dose rate from the steam generator channel heads has been found to influence the loop piping measurements. If not possible, a notation should be made on the form that the manway covers were removed.

- If the results of the radiation survey indicate marked inconsistencies with previous survey results and/or if damage to the survey instrument is suspected, the instrument calibration should be rechecked after completion of the survey.

- Prior to entering the survey locations, worker should note hotspots near by. Potential crud traps are summarized below.

    - Drain lines
    - Pressurizer surge line
    - Pressurizer spray line
    - Sample/transmitter lines
    - Loops stop valves
    - Check valves
    - RTD wells
    - Loop drain valves
    - Letdown lines
    - RHR Lines
    - High Pressure Injection Nozzles
    - Any low point in a pipe section

- Also note the Data Requirements described in Table 2-1.

SECTION 5 INSTALLATION OF SURVEY MARKERS

Figures 5-1 is a sketch of the markers that should have been installed at all of the PWR plants participating in the SRMP. These markers, which consist of a thin (20 gage) stainless steel plate backing, are attached to the outer insulation cover or stainless steel lagging with standard 1/8-inch diameter steel pop rivets.

The locations at which radiation levels are to be monitored are based on the following selection criteria:

- The location should be representative of the crud layer source on out-of-core surfaces and should reflect the behavior of these surface sources.
- The locations should be common from loop to loop and plant to plant to facilitate comparison of the various plants' radiation levels and buildup.
- The survey points should be easily accessible to plant personnel.
- The total number of survey points to be monitored should be limited so that the personnel exposures received during the survey are low.

Figure 5-2 shows recommended survey marker locations on the external surfaces of the reactor coolant loops. Although these locations are highly desirable, experience has shown that they are not always practical for installation of markers. For example, the lack of access to a certain point may require some modification of the survey point. In such cases, be sure to note any modifications in the location in the comments section of the survey form.

It is important that the locations be chosen in such a manner that the radiation level at those points is due to the RCS piping or the steam generator tubes, and not to adjacent pipe runs and crud traps (i.e., valves) which are located in the same general area. The influence of such outside sources can be minimized by performing a careful survey of the area to determine the source and magnitude of the background radiation levels.

The recommended procedure is as follows:

5.1.1. Choose a location in the approximate vicinity of each of the points illustrated in Figure 5-2.

- JL1/JL2 - 30 centimeters above the elbow of the cold leg (J-leg) opposite the steam generator (Two points required for each steam generator)
- D1/D2 - Opposite of the high pressure injection (HPI) nozzle (Two points for each steam generator)
- HL1 - Top of the hot leg elbow (One point for each steam generator)

5.1.2. Monitor piping and valves near the prospective marker location to identify those which carry radioactive fluid and/or those which are potential crud traps. Potential crud traps are summarized below.

- Drain lines
- Pressurizer surge line
- Pressurizer spray line
- Sample/transmitter lines
- Loops stop valves
- Check valves
- RTD wells
- Loop drain valves
- Letdown lines
- RHR Lines
- High Pressure Injection Nozzles
- Any low point in a pipe section

5.1.3. Estimate the contribution of such piping/valves to the total dose rate at the prospective marker location by surveys at various distances away from and around the background source.

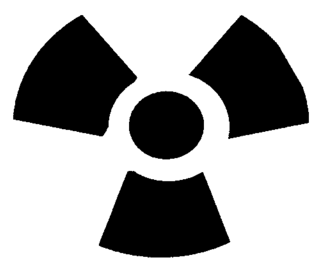
5.1.4. Locate the marker at a position such that the dose contributions from background sources are minimized.

FIGURE 5-1. RADIATION SURVEY MARKER

EPRI - PWR

STANDARD RADIATION MONITORING PROGRAM

SURVEY LOCATOR

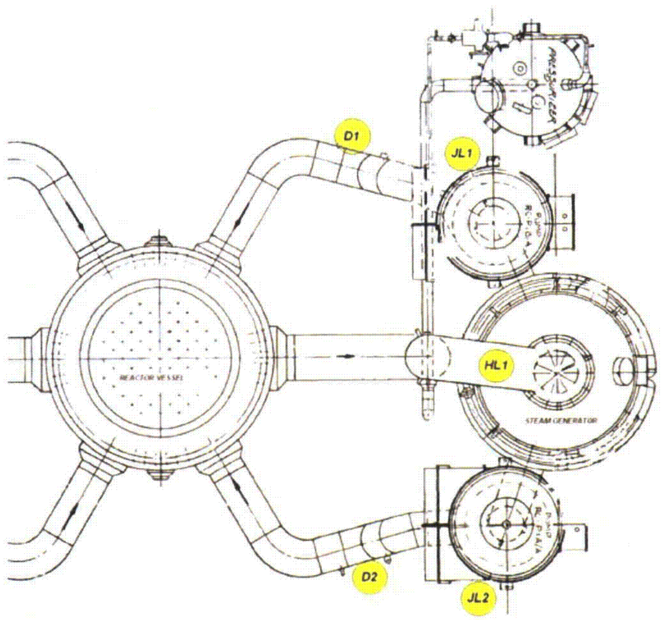


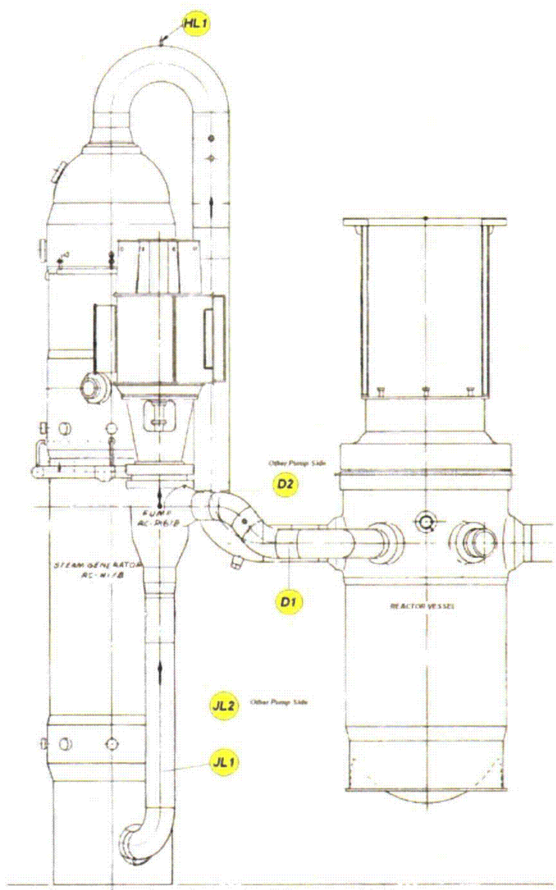
PLACE INSTRUMENT ON ABOVE SYMBOL DURING MEASUREMENT

LOCATION #

LOOP #

FIGURE 5-2. REACTOR COOLANT LOOP SURVEY MARKER LOCATIONS BABCOCK AND WILCOX DESIGNED PWR PLANTS





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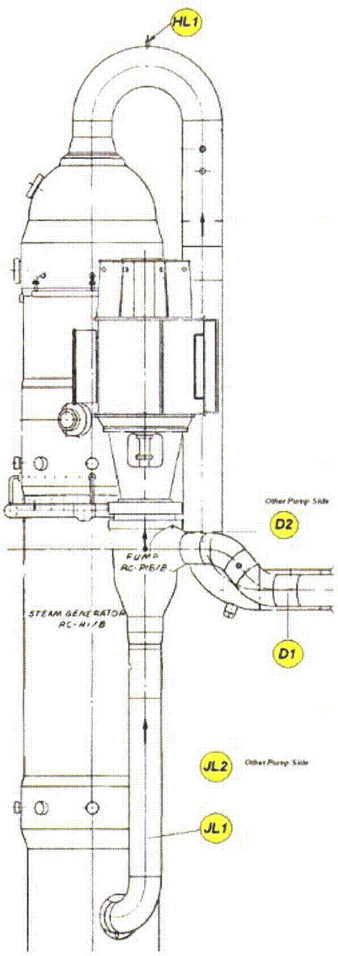
schoi@epri.com

EPRI-PWR STANDARD RADIATION MONITORING PROCEDURE
DATA SHEET 1
REACTOR COOLANT LOOP PIPING SURVEY – B&W DESIGNED PLANT

PLANT / UNIT			
SHUTDOWN (Breaker Open)	DATE	**SURVEY**	DATE
	TIME		TIME

Plant Condition			
Reactor Coolant Piping	Full	Empty	% Drained =________ (if applicable)
Steam Generator Primary Side	Full	Empty	% Drained = ________(if applicable)
Steam Generator Secondary Side	Full	Empty	% Drained = ________(if applicable)
Survey Performed By			



RADIATION SURVEY DATA

			CONTACT DOSE RATE AT MARKER (mR/hr)	
Survey Point	**Survey Instrument Type**	**ED Bias**	**Steam Generator #**	**Steam Generator #**
JL1 (a)				
JL2 (a)				
D1 (b)				
D2 (b)				
HL1 (b)				

Comments:

a. Required Points

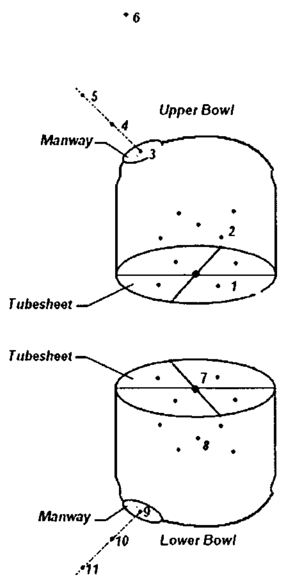
b. Recommended Points

EPRI-PWR STANDARD RADIATION MONITORING PROCEDURE
DATA SHEET 2
STEAM GENERATOR CHANNEL HEAD SURVEY – B&W DESIGNED PLANT

PLANT / UNIT			
SHUTDOWN (Breaker Open)	DATE	**SURVEY**	DATE
	TIME		TIME

Survey Instrument Type	
ED Bias (if applicable)	
Survey Performed By	

Steam Generator Channel Head (Manway Cover Removed)



RADIATION SURVEY DATA

DOSE RATE AT SURVEY POINT (R/hr)			
Survey Point	**Location**	**Steam Generator #**	**Steam Generator #**
	Survey Date for Each Loop		
Inlet Channel (Upper Bowl)			
1	Highest Contact on Tubesheet (a)		
2	30 cm above highest contact dose rate (a)		
3	Manway Entrance (a)		
4	30 cm from Manway (a)		
5	One meter from Manway (b)		
6	3 meter from Manway (b)		
Outlet Channel (Lower Bowl)			
7	Highest Contact on Tubesheet (a)		
8	30 cm below highest contact dose rate (a)		
9	Manway Entrance (a)		
10	30 cm from Manway (a)		
11	One meter from Manway (b)		

Comments: a. Required Points b. Recommended Points
*** Note if TLDs were used to obtain data**

EPRI-PWR STANDARD RADIATION MONITORING PROCEDURE
DATA SHEET 3 (OPTIONAL)
OPTIONAL INFORMATION POINTS SURVEY – B&W DESIGNED PLANT

PLANT / UNIT			
SHUTDOWN (Breaker Open)	DATE	**SURVEY**	DATE
	TIME		TIME

Optional Information Points Survey

SURVEY POINT	LOCATION WHERE DATA IS TAKEN	SURVEY INSTRUMENT TYPE	ED BIAS	CONTACT DOSE RATE (mR/hr)
Decay Heat Piping				
Cooler Inlet Piping				
Decay Heat Pump Suction				
Seismic Plate (Reactor Vessel Head Service Structure)				
Comments:				

Dose Rate at Survey Point Before Forced Oxidation (mR/hr)				
Survey Point	Survey Instrument Type	ED Bias	Steam Generator #	Steam Generator #
JL1				
JL2				
D1				
D2				
HL1				

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