NOTE 1: (Continued)

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, as presented in the Core Operating Limits Report,

585.1

T' = $\leq$ 588.2 °F Reference T_{avg} at RATED THERMAL POWER,

K_3 = Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient as presented in the Core Operating Limits Report,

P = Pressurizer pressure, psig,

P' = 2235 psig (Nominal RCS operating pressure),

S = Laplace transform operator, sec^{-1},

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) for $q_t - q_b$ between the "positive" and "negative" $f_1(\Delta I)$ breakpoints as presented in the Core Operating Limits Report; $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent imbalance that the magnitude of $q_t - q_b$ is more negative than the $f_1(\Delta I)$ "negative" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the $f_1(\Delta I)$ "negative" slope presented in the Core Operating Limits Report; and

(iii) for each percent imbalance that the magnitude of $q_t - q_b$ is more positive than the $f_1(\Delta I)$ "positive" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the $f_1(\Delta I)$ "positive" slope presented in the Core Operating Limits Report.

Amendment No. 143 (Unit 1)
Amendment No. 125 (Unit 2)

Attachment 1

T   = As defined in Note 1,

     585.1

T'' = ≤ ~~500.2~~ °F Reference T_{avg} at RATED THERMAL POWER,

S   = As defined in Note 1, and

$f_2(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i)   for $q_t - q_b$ between the "positive" and "negative" $f_2(\Delta I)$ breakpoints as presented in the Core Operating Limits Report; $f_2(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent imbalance that the magnitude of $q_t - q_b$ is more negative than the $f_2(\Delta I)$ "negative" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the $f_2(\Delta I)$ "negative" slope presented in the Core Operating Limits Report; and

(iii) for each percent imbalance that the magnitude of $q_t - q_b$ is more positive than the $f_2(\Delta I)$ "positive" breakpoint presented in the Core Operating Limits Report, the ΔT Trip Setpoint shall be automatically reduced by the $f_2(\Delta I)$ "positive" slope presented in the Core Operating Limits Report.

Note 3:   The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 4.4% of Rated Thermal Power.

Note 4:   The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.0% of Rated Thermal Power.

Proposed revision to Technical Specification Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints

In Note 1

change "T′ = $\leq$ 588.2 ° F Reference T_{AVG} at RATED THERMAL POWER"

to "T′ = $\leq$ 585.1 ° F Reference T_{AVG} at RATED THERMAL POWER"

In Note 2

change "T″ = $\leq$ 588.2 ° F Reference T_{AVG} at RATED THERMAL POWER"

to "T″ = $\leq$ 585.1 ° F Reference T_{AVG} at RATED THERMAL POWER".

Technical Justification

The values for T′ and T″ are defined as Reference T_{AVG} at RATED THERMAL POWER. in Note 1 and 2 of Table 2.2-1. When the steam generators are replaced at McGuire, programmed T_{AVG} at full power will be changed from 588.2 ° F to 585.1 ° F. This temperature was chosen based on returning the secondary side steam pressure to the original value after replacement of the steam generators. 585.1 ° F was the assumed value for nominal full power T_{AVG} in all applicable safety analyses related to replacement of the steam generators.

No Significant Hazards Analysis

The following analysis, required by 50.91, concludes that the proposed amendment will not involve significant hazards considerations as defined by 10 CFR 50.92.

10 CFR 50.92 states that a proposed amendment involves no significant hazards considerations if operation in accordance with the proposed amendment would not:

(1)   Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2)   Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3)   Involve a significant reduction in a margin of safety.

This proposed change to the Technical Specifications does not involve a significant increase in the probability or consequences of an accident previously evaluated. Changing the value for T_{AVG} in Notes 1 and 2 of Table 2.2-1 will update the value to agree with the T_{AVG} assumed in the applicable safety analyses for replacement of the steam generators. Acceptable results were obtained for all required reanalyses. The probability of an accident will not be significantly affected by operation with the new T_{AVG} value, because all equipment will be operated within acceptable design limits. The consequences of previously evaluated accidents which are affected by this change have been evaluated, and have been determined to be within acceptable limits.

This proposed change will not create the possibility of a new or different kind of accident from any previously evaluated. This change does not change the physical configuration of the plant, and all analyses which are affected by replacement of the steam generators have been determined to have acceptable results assuming this value for T_{AVG}.

This proposed change to the Technical Specifications will not involve a significant reduction in the margin of safety. All safety analyses which were affected by replacement of the steam generators assumed this value for T_{AVG} and the results were determined to be within previously acceptable limits.

Attachment 3

DUKE POWER COMPANY
MCGUIRE 1 & 2 AND CATAWBA 1
NUCLEAR GENERATING STATIONS

REPLACEMENT STEAM GENERATOR
TOPICAL REPORT

BABCOCK & WILCOX INTERNATIONAL
BWI-222-7693-LR-01

APRIL 1995
REV. 4

STATION:	MNS/CNS
DPC SPECIFICATION:	DPS-1201.01-00-0002
DPC PURCHASE ORDER:	C23355-68
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LIST OF REVISIONS

REV. NO.	DATE	PAGE	DESCRIPTION
0			Initial Release
1	Sept./94		Table 2.1-1 revised to delete proprietary data.
2	Nov./94	Cover	Revised to Rev. 2.
		All	Editorial change: Added header with page number to each page. Deleted page number at bottom.
		4 through 11	Deleted Section 2.7.4. Revised page numbers.
		15, 16	Revised Title of Figure 2.2.5-5. Added page numbers.
		17	Revised page numbers.
		27	Deleted "typical" in 2 places and changed "Typical" to "The".
		29	Revised Figure 2.2.1-2.
		30	Revised Figure 2.2.1-3.
		31	Revised Figure 2.2.1-4.
		36	Revised text to specific DPC geometry.
		37, 38	Revised medium bar width and revised text in 2.2.4.2.1. Deleted "typical" from 2.2.4.2.2 and 2.2.4.2.3. Made reference to Figure 2.2.4-4.
		40	Text revised in Section 2.2.4.3.3. Added Ref. 3.
		44	Revised Figure 2.2.4-4.
		45	Revised Figure 2.2.4.-5.
		52	Section 2.2.5.7 revised.
		55	Revised text on Figure 2.2.5-2.
		58	Revised text on Figure 2.2.5-5.

REV. NO.	DATE	PAGE	DESCRIPTION
		73	Section 2.2.8 revised. The word "typically" deleted from 2.2.9.
		77, 78	Revised text in Section 2.2.12.
		79	Deleted sentence in Section 2.2.13.1.
		81	Deleted text in 2 places.
		82	Changed "tests" to "experience".
		84	Added last para. to 2.3.1.
		85	Added Table references and 3rd paragraph.
		86	Clarified surface finish in 2.3.2.3 and moved last paragraph of 2.3.3 to Section 2.3.1.
		95	Revised text of 2.4.1.2 to identify Code Edition.
		97	Added text for GDC 4.
		103	Revised text.
		105	Revised text for NUREG 0909.
		108	Added text under "Loose Parts".
		116	Revised solution to problems 32 and 34.
		136	Revised last paragraph of 2.7.2. Revised 2.7.3. Deleted 2.7.4.
		138	Deleted P_m design stress requirement and defined S_u. Added "and NB-3223".
		139	Revised text in 1st paragraph and Level C requirements.
		151	Revised text in 3.2.6.6
3	Dec./94	140	Revised S_u value.
		141	Made typographical corrections and revised S_u value.
4	Apr../95	Cover	Revised to Rev. 4

REV. NO.	DATE	PAGE	DESCRIPTION
		4 through 160	Page numbers revised to reflect additional pages in List of Revisions.
		7 through 14	Revised page numbers in Table of Contents.
		18, 19	Revised page numbers in List of Figures.
		20	Revised page numbers in List of Tables.
		21	Deleted "the same".
		22	Item 4 changed "normally found in" to "found in presently operating". Changed "same" to similar.
		26	Changed "I" to "Alloy".
		29	Revised format.
		39	Added to first sentence of first paragraph. Added description to lattice grid tube supports. Clarified text.
		41	Added "tube lockup" and "via the J-tabs".
		42	Changed "closing" to "operating".
		43	Deleted "Preliminary" and changed "indicates" to "indicate". Changed "41S" to "410S".
		44	Added date to reference 3.
		45	Replaced Figure 2.2.4.1.
		52	Added "as well as... loads" to last sentence of 2nd paragraph.
		53	Revised next to last paragraph for clarity.
		55	Changed "AT CNS" to "For example," and "is" to "can be".

REV. NO.	DATE	PAGE	DESCRIPTION
		56	Added "internal". Revised text in 2nd paragraph of 2.2.5.5 for clarity. Revised text in 2.2.5.6 for clarity. Added last sentence to first paragraph of 2.2.5.7.
		57	Added "the addition of" to the 1st sentence of 2.2.5.8.
		58 and 59	Revised Figures 2.2.5.1 and 2.2.5.2.
		63	Revised text for clarity under item 3 of design features and under item 3 of features that preclude blockage.
		66	Revised text for clarity in 3rd paragraph of 2.2.7.1.
		68	Revised text for clarity in 2.2.7.7 and 2.2.7.8.
		70	Revised Figure 2.2.7.1.
		73	Revised Figure 2.2.7.4.
		75 and 76	Revised Figure 2.2.7.6 and 2.2.7.7.
		77	Changed "addition" to "additional" in first paragraph.
		86	Deleted "typically" from item 3.
		88	Revised last sentence of 1st paragraph of 2.3.1 for clarity.
		89, 90 and 91	Revised Section 2.3.2.2 for clarity. Added "or 690" and changed "0.02" to "0.10" in Section 2.3.2.3. Revised Section 2.3.3 for clarity.
		92 and 93	Changed "316N" to "F316N/316LN".
		96	Revised material specifications.
		97	Revised material specifications and added components.
		98	Added J-tabs.

REV. NO.	DATE	PAGE	DESCRIPTION
		103	Revised last sentence under RG 1.50 for clarification.
		111	Changed "close" to "minimize" and "control boiling... tube bundle" to "low qualities at the tubesheet".
		112	Added "with the alloy 690 interface" and "All threaded... contained.".
		118 thru 121	Added "Alloy 690 material" to solution of problems 8, 9, 12, and 15. Revised "sulphate" to "sulfate" in problem 15. Deleted problem 30 and renumbered following problems accordingly.
		122	Deleted problem 30 from Figure 2.5-1 and renumbered following problems accordingly.
		140	Revised "sulphate" to "sulfate".
		156	Deleted "after installation" from Section 3.2.6.6, item 2.

DUKE POWER COMPANY
REPLACEMENT STEAM GENERATOR
TOPICAL REPORT

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LIST OF ACRONYMS AND GLOSSARY

Terms and acronyms used in this report are defined the first time they are used in the text.
The more significant and widely used acronyms and terms are defined below.

Acron

ALARA	As Low as Reasonably Achievable
ANS	American National Standard
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
B&W	Babcock & Wilcox
BTP	NRC Branch Technical Position (appended to SRPs)
BWI	Babcock & Wilcox Industries
CANDU	Canadian Deterium Uranium heavy water reactor design
CDS	Certified Design Specification
CFR	Code of Federal Regulations
CG	Center of Gravity
CMS	Corrosion Monitoring System
DBE	Design Basis Earthquake
ECT	Eddy Current Test
EP	Electro-chemical Polishing
EPRI	Electric Power Research Institute
F	Degrees Fahrenheit
FEI	Fluid Elastic Instability
FIV	Flow Induced Vibration
FW	Feedwater
FSAR	Final Safety Analysis Report
FUR	Flat bar U-bend restraint
GDC	NRC General Design Criteria
GTAW	Gas Tungsten Arc Welding
hr	Hour
I.D.	Inside Diameter
IGA	Intergrannular Attack
ISI	In-Service Inspection
kips	Thousand pounds (load)
ksi	Thousand pounds per square inch
LBB	Leak Before Break
LBLOCA	Large Break LOCA
LOCA	Loss Of Coolant Accident
MIG	Metal Inert Gas welding process
MFW	Main Feedwater system
MSLB	Main Steam Line Break
MP2	Millstone Plant, unit 2
NRC	United States Nuclear Regulatory Commission
OBE	Operational Basis Earthquake

O.D.	Outside Diameter
OSG	Original Steam Generator
Owner	Utility
psi	Pounds per square inch
psia	Pounds per square inch, absolute
psig	Pounds per square inch, gauge
PWHT	Post Weld Heat Treatment
PWSCC	Primary Water Stress Corrosion Cracking
QA	Quality Assurance
RA	Roughness Average, in micro-inches
RCS	Reactor Coolant System
RG	NRC Regulatory Guide
RSG	Replacement Steam Generator
RUB	Reverse U-bend
SCC	Stress Corrosion Cracking
scfm	Standard cubic feet per minute
SBLOCA	Small Break LOCA
SG	Steam Generator
SMAW	Shielded Metal Arc Welding
SRP	NRC Standard Review Plan (collected in NUREG 0800)
SSE	Safe Shutdown Earthquake
Tech Spec	Technical Specification(s)
TFL	Tube Free Lane
TS	Technical Specifications
UA	Heat transfer capacity (BTU/hr °F)
UT	Ultrasonic test
USNRC	United States Nuclear Regulatory Commission

Glossary

Circulation Ratio - the ratio of steam generator tube bundle (riser) flow to steam flow.

Denting - steam generator tube deformation caused by corrosion product interference at tube support plates.

Downcomer - the annular space between the tube bundle shroud and shell that channels recirculated water to the base of the tube bundle.

Moisture carryover - the percentage of steam mass flow that is entrained as liquid water.

Recirculation Ratio - the ratio of liquid flow separated from the riser flow to steam flow (equal to circulation ratio minus one).

Riser - the flow path through the steam generator tube bundle to the steam separator inlets.

Steam carryunder - the percentage of downcomer mass flow that is entrained steam.

LIST OF FIGURES

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RSG TOPICAL REPORT

EXECUTIVE SUMMARY

The Replacement Steam Generator Topical Report describes the design and manufacture of the BWI Replacement Steam Generators (RSGs) for use in Duke Power Company's McGuire 1 and 2 and Catawba 1 nuclear stations. This report discusses aspects of the RSG design that provide improved reliability from the existing design, addressing failure modes normally found in steam generators and describes features which eliminate or reduce the effects of the failure modes. In addition, it provides general information related to the RSGs describing design characteristics and discusses design criteria including the analysis executed to address the specified requirements.

The RSGs are manufactured by Babcock & Wilcox International (BWI) in Cambridge, Ontario, Canada. The RSGs are designed, manufactured and tested in accordance with the 1986 Edition (no addenda) of Section III of the ASME Code, and will be N-stamped by BWI prior to shipment. The design, procurement, and manufacturing process is performed under a QA Program that complies with the requirements of Appendix B to 10CFR50, and complies with current NRC requirements that relate to steam generator design.

RSGs including the Millstone Unit 2 (MP2) RSGs have been manufactured by BWI, successfully certified by the Authorized Nuclear Agency and are performing satisfactorily. The steam generators described in this report employ similar design features and corrosion resistant materials as the MP2 design. The success of the MP2 steam generator replacement confirms RSG design methods and expected in situ operational performance. △4

The RSGs occupy the same physical envelope as the Original Steam Generator. Differences between the OSG and RSG designs are identified in this report. There are no changes to interfaces with the reactor coolant, main steam systems, or component or piping supports with the exception of the RSG feedwater nozzle which has been relocated to accommodate replacing the integral preheater of the OSG with an internal feedwater header on the RSG and also relocation of the sample taps and the auxiliary feedwater nozzle for CNS. Normal operating conditions and plant transients have been requalified for the RSG design, thus, associated design bases are not affected.

RSG Topical Report

1.   INTRODUCTION AND PURPOSE

This Topical Report describes the McGuire 1 and 2 and the Catawba 1 replacement recirculating nuclear steam generators constructed by Babcock and Wilcox International (BWI) of Ontario, Canada. It describes the superiority of the BWI RSG design and manufacture with respect to generic steam generator failure modes and reliability. Modifications that may be performed during the steam generator replacement outage or the replacement process itself are beyond the scope of this report.

BWI has extensive nuclear steam generator design and fabrication experience, founded on more than a hundred years of heavy-vessel manufacturing capability and experience for the fossil power and petroleum industries. The service record of BWI recirculating steam generators has been excellent. These issues are addressed in Section 2.1.1.

The RSG design is described in Sections 2.2, 2.3, and 2.4 of this report. The RSGs incorporate many improvements. These are discussed in Section 2.5 of this report. Confirmatory analyses and tests, RSG operating restrictions, stress evaluations, and start-up testing are discussed in Sections 2.6, 2.7, 2.8 and 2.9 respectively.

BWI controls RSG design and fabrication to maintain high quality and to maintain the existing plant's design and licensing bases. The BWI RSG design and the quality assurance controls used in RSG construction conform to NRC requirements. The BWI quality plan is described in Section 3 of this report.

The principal objectives of this report are to:

1.   Describe BWI capability to design and build RSGs for pressurized water reactors,

2.   Describe RSG design features, materials, methods of analysis, QA measures, fabrication controls, and demonstrate physical, structural, and thermal-hydraulic compliance with the design requirements,

3.   Identify the RSG design criteria and standards employed, including NRC guidance, and describe conformance to them,

4.   Describe industry steam generator problems and issues considered in RSG design and fabrication and discuss design features that provide improved reliability considering failure modes found in presently operating steam generators.

Section 2.1.2 discusses the Millstone 2 RSG design. These RSGs were designed and fabricated by BWI (not the OSG manufacturer), installed under the provisions of 10CFR50.59 and 10CFR50.90, and approved by the NRC. The RSGs described in this report employ similar corrosion-resistant materials, and similar design features to the Millstone 2 steam generators.

2. REPLACEMENT STEAM GENERATOR DESIGN

2.1 GENERAL DESCRIPTION

The McGuire and Catawba replacement steam generators (RSGs) described in this report are of the recirculating non-preheater U-tube design. They have the following design features:

1. Stainless steel (410S) lattice grid tube supports.
2. Stainless steel (410S) flat bar U-bend supports.
3. High capacity primary and secondary cyclone separators.
4. Circulation ratio of 5.7.
5. Feedwater headers which minimize potential water hammer and thermal stratification effects.
6. Minimum-radius tube U-bends of five times tube diameter or more.
7. Triangular tube pitch.
8. Thermally treated Inconel 690 tubes.

These and other important aspects of the RSGs are described in the following sections of this report. The BWI RSGs accommodate high internal circulation flows with acceptable levels of tube vibration and effective steam separator performance. High internal circulation benefits steam generator performance and longevity by promoting flow penetration across the tubesheet and reducing fluid quality and zones of low velocity thereby reducing sludge accumulations. Through fabrication of steam generators for Canadian heavy water reactor (CANDU) plants and for Millstone 2, and through performance of steam generator repairs and cleaning, BWI has demonstrated its capability to design, manufacture, and maintain steam generators with triangular pitch tube arrangement.

2.1.1 Qualifications of the Steam Generator Supplier

BABCOCK AND WILCOX INTERNATIONAL

Babcock & Wilcox International (BWI), located in Cambridge, Ontario, Canada, has fabricated fossil-fueled boiler components for over 100 years and has fabricated nuclear system components since the late 1950's. Although most of the nuclear system components manufactured have been recirculating steam generators for CANDU nuclear plants, the RSGs are comparable in materials, water chemistry, and fabrication methods. As shown in Figure 2.1-1 the size of these units is also comparable. Therefore, BWI's experience in supplying over 200 CANDU steam generators is directly applicable to the RSGs described in this report.

BWI has strong Project Management, Engineering, Manufacturing, Production Control, Purchasing, and Quality Assurance Departments. These provide close control of the quality of replacement steam generator design, procurement, fabrication, and documentation. Continuous work in the nuclear industry has enabled BWI to maintain a well qualified steam generator design group. Engineers involved with the design and analysis of steam

generators have a thorough knowledge of design by analysis methods and are familiar with the application of the ASME code to nuclear pressure vessel design and analysis. BWI holds ASME certificates of authorization for N, NA and NPT symbol stamps. Subcontractors for material supply and fabrication are all fully qualified under the requirements of the BWI Quality Assurance Program. The BWI quality assurance program is described in Section 3.1.

BWI steam generator manufacturing experience to date includes:

CANDU Steam Generators:

Lattice Grid Type	82
Wolsong 3 and 4 South Korea (under construction)	2
Broached Plate Type	125
Total	209

PWR Steam Generators:

Lattice Grid Type	
Northeast Utilities (Millstone 2)	2
Duke Power Co. (under construction)	12
Florida Power & Light (under construction)	2
Rochester Gas & Electric (under construction)	2
Commonwealth Edison (under contract)	4
Total	22

BWI recirculating steam generators have more than 20 years of operating history. The performance and reliability of BWI steam generators has been excellent. In over 200 steam generators, containing more than 600,000 tubes and having in excess of 6 million tube-years of operation, less than one percent of the tubes had been plugged as of July, 1993.

Additional information on steam generator tube opening experience and BWI measures to preclude primary water stress corrosion cracking, intergranular attack and sludge accumulation are contained in Section 2.5.1.

2.1.2   Millstone 2 Replacement Steam Generator Design and Experience

Two BWI RSGs are in service at Millstone 2. The steam generators described in this report have many features in common with the Millstone 2 design. These include Alloy 690 tubes and other corrosion-resistant materials, weld overlay of all primary side carbon steel surfaces with stainless steel or Inconel, tight packing of tubes, full depth hydraulic expansion of the tubes in the tube sheet, and measures to minimize water hammer and vibration. Unlike the Millstone 2 steam generators, the RSGs are complete replacements, shipped intact to the plant. The Millstone 2 heat exchanger (lower) sections were shipped to the site for use with

the existing steam drums (upper sections).

Successful completion, licensing, installation and startup of the Millstone 2 steam generators demonstrates BWI design and fabrication capability, and the overall acceptability of the RSG design.

2.1.3   Comparison with Existing Design

Parameter changes from the existing (OSG) design are provided in Table 2.1-1 and include differences in steam generator weight, inventory, operating conditions, and major geometrical features. The differences potentially affecting plant safety (water inventories, primary side flow resistance, shell stiffnesses and RSG weight) are beyond the scope of this report.

The RSG is designed, fabricated and analyzed to minimize differences with respect to form, fit, and function as compared to the existing steam generator. Physical comparison of the RSG and OSG are discussed in this report. Compatibility of primary and secondary side materials with the existing design is generally demonstrated in Section 2.3.

TABLE 2.1-1
STEAM GENERATOR COMPARISON

PARAMETER	RSG DATA	OSG DATA
Primary side volume: no tubes plugged (nozzle dams in place) (ft³)	1229.1	955
Secondary side mass: @ 0 % full power (lbm) @ full power (lbm)	144.5 x 10³ 124.6 x 10³	116 x 10³ 104 x 10³
Full power steam flow	3.78 x 10⁶ lbm/hr	3.78 x 10⁶ lbm/hr
Primary Pressure drop across S/G (unplugged) @ 37.0E6 lbm/hr	33.0 psid	33.3 psid
Primary side design pressure psia	2500	2500
Secondary side design pressure psia	1200	1200
Primary side design temperature (°F)	650	650
Secondary side design temperature (°F)	600	600
Primary side operating pressure (psia)	2250	2250

Steam outlet conditions: pressure (psia)
                          maximum carryover
                          (Guarantee)
	1020
	0.25%
		1020
		0.25%

Feedwater temperature @ full power (°F)	440	440
Heat transfer rate @ full power (MW)	857.5	857.5
Steam Outlet Flow restrictor flow area (ft²)	1.374	1.39
Primary side heat transfer surface area: no tubes plugged (based on avg. I.D.) (ft²)	70,480	42,500
Secondary side heat transfer surface area: no tubes plugged (based on avg. O.D.) (ft²)	79,800	48,000
Number of tubes	6633	4674

Tube O.D.:
            upper tolerance
            lower tolerance
	0.6875" nom.
	+0.0"
	-0.005"
		0.750" nom.

Tube wall thickness: nominal tolerance	0.040" ± 0.004"	0.043" -
Tube material: SB-163, Code Case N-20	Alloy 690	Alloy 600

Tube thermal conductivity: @ 400°F
                             @ 500°F
                             @ 600°F
	8.92 Btu/ft-hr-°F
	9.54 Btu/ft-hr-°F
	10.167 Btu/ft-hr-°F
		10.1 Btu/ft-hr-°F
		10.6 Btu/ft-hr-°F
		11.1 Btu/ft-hr-°F

TABLE 2.1-1 (cont'd)
STEAM GENERATOR COMPARISON

PARAMETER	RSG DATA	OSG DATA
Tube pitch	triangular	square
Tube minimum strength (per ASME Code and Code Case N20): yield tensile	40 ksi 80 ksi	35 ksi 80 ksi
Steam Outlet Nozzle Diameter (in)	29.469	29
Shell Side Manways (No. - Dia.(in))	1-21	2-16
Primary Side Manways (No. - Dia.(in))	2-21	2-16
Primary Inlet Nozzles (No. - Dia.(in))	1-31	1-31
Primary Outlet Nozzles (No. - Dia.(in))	1-31	1-31
Feedwater Nozzle Diameter (Nom.) (in)	16	16
Auxiliary Feedwater Nozzle I.D. (in)	5.25	5.3
Bottom Blowdown Nozzles (No.-Dia.(in))	2-3	2-2
Recirculation Nozzle (No.-Dia.(in))	1-3	N/A
Water Level Taps (No. - Dia. (in.))	14-3/4	8-3/4
Handholes (No. - Dia.) (in)	10-6	2-6
Inspection Ports (No. - Dia.) (in)	12-2	4-2
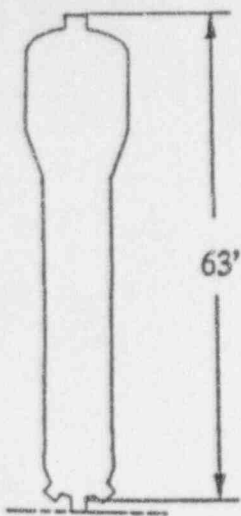
BWI CANDU
Romania
≈ 240 tons ea.
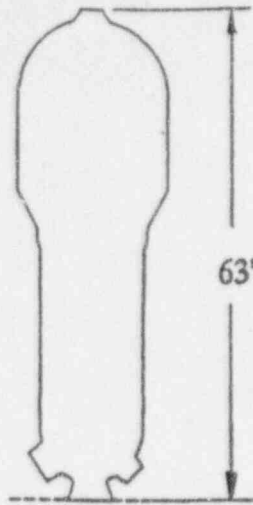
ST. LUCIE 1
BWI Replacement
≈ 527 tons ea.
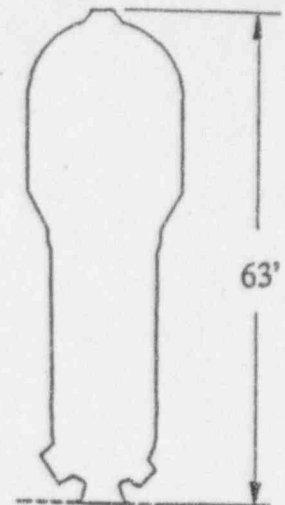
MILLSTONE 2
BWI Replacement
≈ 535 tons ea.
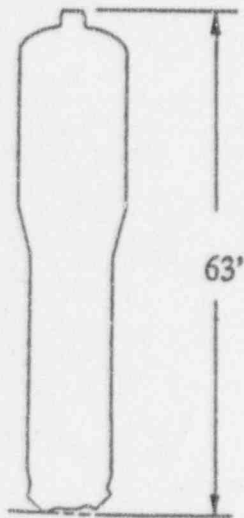
R. E. GINNA
BWI Replacement
≈ 315 tons ea.
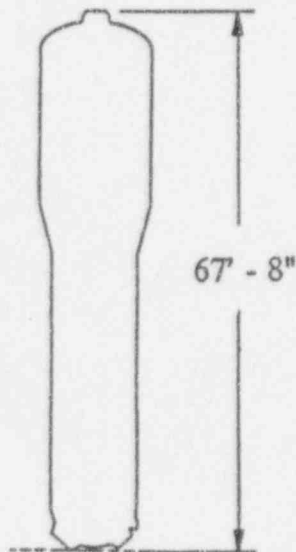
**MCGUIRE 1 & 2 &
CATAWBA 1**
BWI Replacement
≈ 406 tons ea.
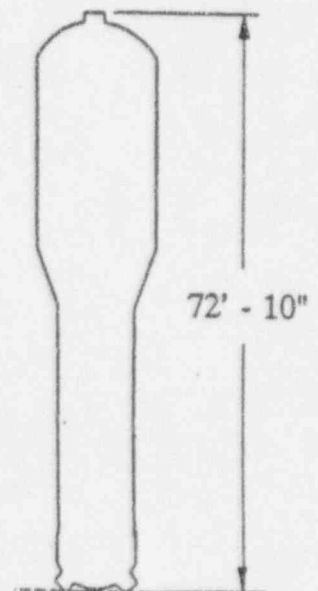
BWI CANDU
Darlington
≈ 385 tons ea.

BWI STEAM GENERATOR PHYSICAL COMPARISON

FIGURE 2.1-1

2.2   STEAM GENERATOR DESIGN HIGHLIGHTS

2.2.1   Pressure Boundary Design

The reactor coolant pressure boundary and secondary side pressure boundary are critical to the safe and reliable operation of the RSG. This section describes the key design features of the RSG portions of these pressure boundaries except for the steam generator tubes which are described in Section 2.2.2. The pressure boundaries withstand internal pressure, seismic, loss of coolant accident (LOCA), main steam line break (MSLB) loads and feedwater break loads. In addition, cyclic loading during normal operation creates the potential for fatigue failures. The pressure boundary components are designed and documented to be in accordance with ASME Code requirements for Nuclear Pressure Vessels, Section III Division 1. Applicable codes and standards are described in Section 2.4.1. Pressure boundary materials are discussed in Section 2.3.1.

Pressure boundary design is analyzed by employing work-station based finite element software. Finite element analysis is used as an analytical tool and a design tool. This permits optimization of important pressure vessel design features while minimizing stresses. Critical design features and dimensions can be reviewed early in the design, accounting for time dependent loads such as operational thermal transients.

Two types of corrosion allowance are considered for design. Corrosion allowances for surfaces that are chemically cleaned include allowances for normal operation and for chemical cleaning. Corrosion allowances for surfaces that are not chemically cleaned include allowances for normal operation only. Allowances vary from zero to 0.0625 inches depending on material and application. Analyses for structural loads, pressure, flow, and flow-induced vibration were performed with corrosion allowances deducted. The corrosion values are verified as part of the BWI Chemical Cleaning Qualification Program. Key elements of this program are presented in Section 2.6.5.

Preparation, revision and issue of design calculations and reports are governed by the BWI Quality Assurance Manual (described in Section 3.1). This ensures that all design and analysis requirements are reviewed for adequacy and approved for release by authorized personnel.

2.2.1.1 Tubesheet Assembly and Primary Divider Plate

The tubesheet/primary head assembly and primary divider plate arrangement is shown in Figure 2.2.1-1. The divider plate is machined from Alloy 690 and welded around its entire periphery to the tubesheet and primary head. At the tubesheet, the plate is welded to a machined Alloy 600 weld build-up along the tube free lane. Along the head, an Alloy 690 weld attaches the divider plate directly to the head base metal rather than to the stainless steel cladding. The stiffening effect of the divider plate is not taken into account when sizing the tubesheet thickness.

2.2.1.2 Closure Design

The RSG is fitted with removable closures at manways, hand holes and inspection ports located to provide access for inspection, repair and maintenance of steam generator internals. Figures 2.2.1-2 and 2.2.1-3 show external and internal manway closure designs. The external cover design provides metal-to-metal contact with the gasket properly seated. This is achieved by controlling the depth of the gasket groove in the inner diaphragm plate. The metal-to-metal contact and use of long flexible bolts reduces the fatigue loading on the bolts during operation. The longer bolts also reduce bolt stress caused by pressure and thermal distortion of the opening. This design can be readily adapted to various stud tensioning systems.

Figures 2.2.1-4 and 2.2.1-5 show locations of primary side manways and secondary side manways, hand holes and inspection ports. These provide access for inspection, maintenance and repair.

2.2.1.3 Shell and Nozzle Design

The RSG shell is fabricated from forgings and plates. Forgings are used for the steam drum head including integrally forged steam outlet nozzle, the primary head including integrally forged primary nozzles and manways, the tubesheet and the conical transition section. Plate is used for the balance of the shell. By maximizing the use of forgings, the RSG design reduces the quantity of weld material requiring in-service inspection and the complexity of in-service inspection. The RSG is supported on support pads which are integrally forged into the primary head (channel head). The lower head and lower tube bundle shell sections are welded to the tubesheet forging.

The RSG primary and secondary side nozzles are the same sizes as those of the OSG. Primary and secondary side manways, however are twenty-one inches in diameter, considerably larger than those on the OSG. This allows easier access to the channel heads and secondary side. The RSG primary nozzles are integrally forged into the primary head. Safe ends are welded to the nozzles to accommodate RSG fit-up to the existing plant piping. The primary manways are integrally forged into the primary head.   Stress concentrations are reduced by contouring all discontinuities and providing large blend radii in these areas. A similar design is used for both the primary and secondary side manways.

The upper head is a single forging which includes the main steam nozzle and an integral flow restrictor that limits internal RSG fluid velocities in the event of a main steam line break. The flow restrictor design and function are discussed further in Section 2.2.9.
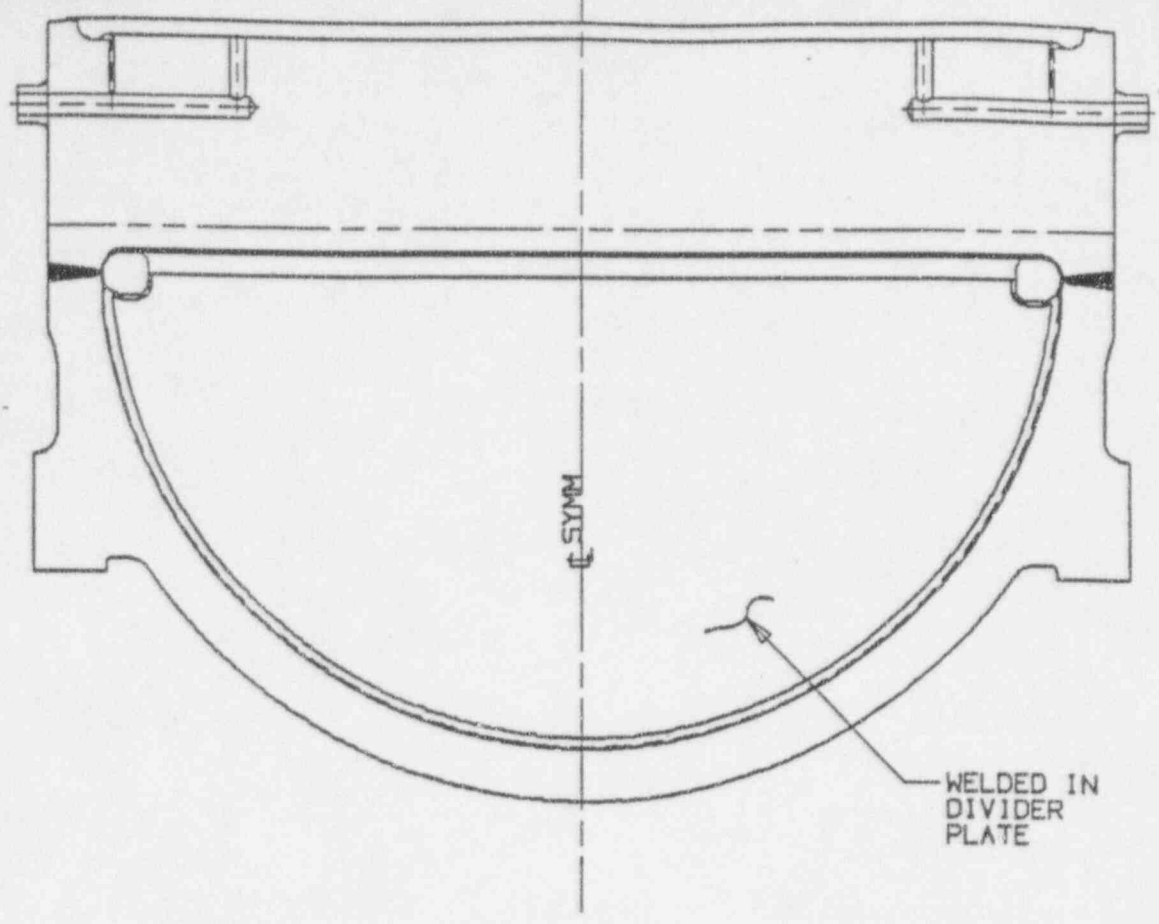
SECTION 'A-A'
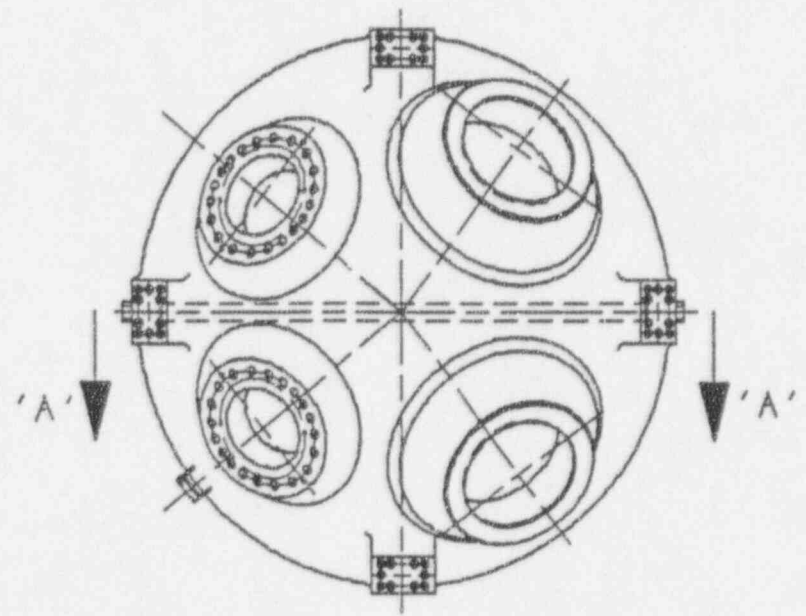
BOTTOM VIEW

WELDED IN DIVIDER PLATE

FIGURE 2.2.1-1
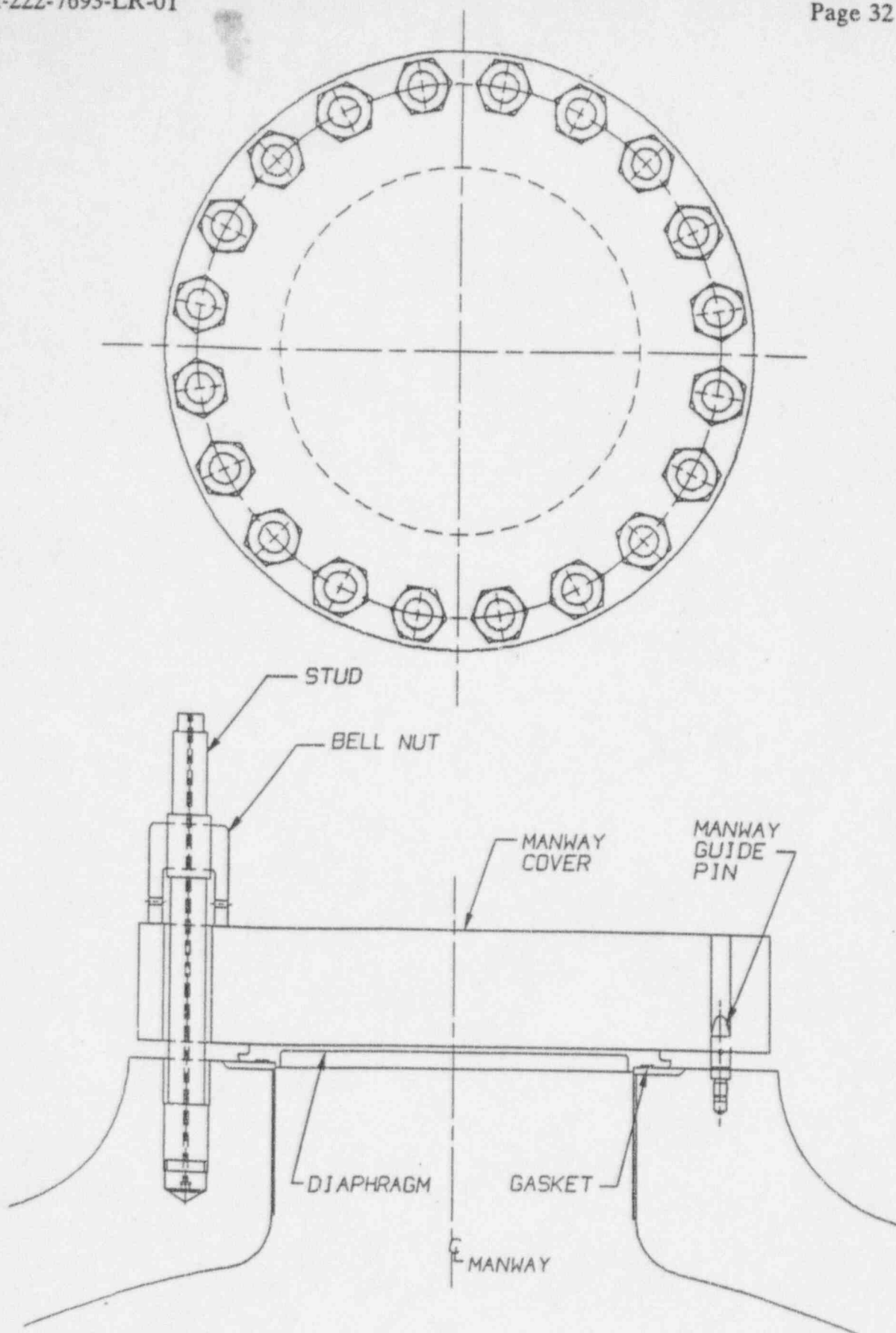
PRIMARY MANWAY CLOSURE DESIGN

FIGURE 2.2.1-2
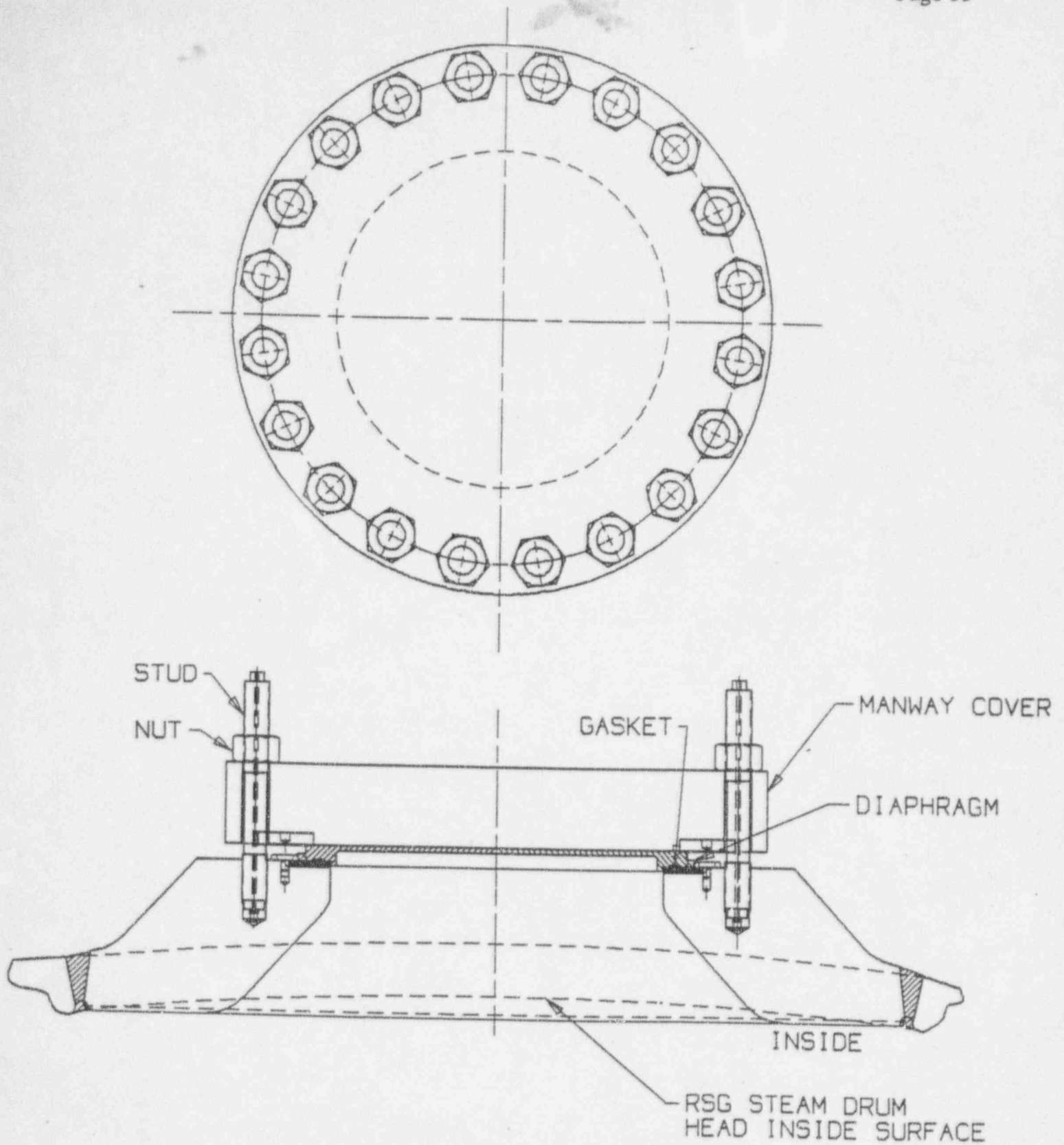
SECONDARY SIDE
MANWAY DESIGN
FIGURE 2.2.1-3
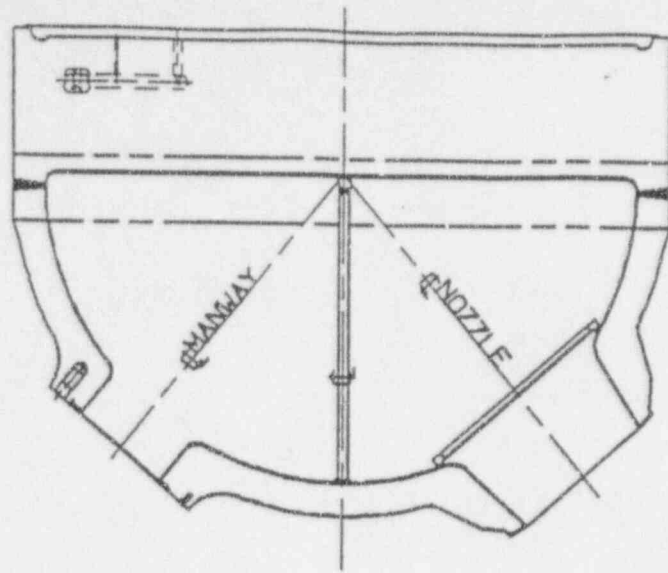
SECTION 'A-A'
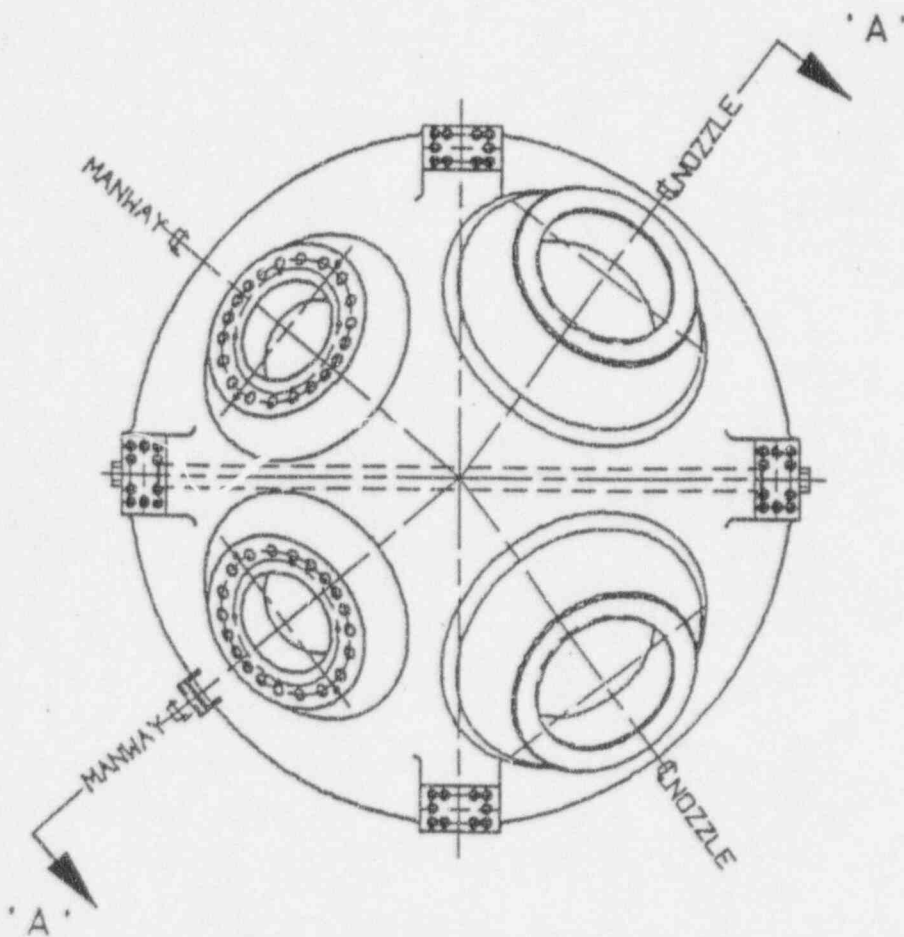
BOTTOM VIEW

PRIMARY HEAD MANWAYS

FIGURE 2.2.1-4
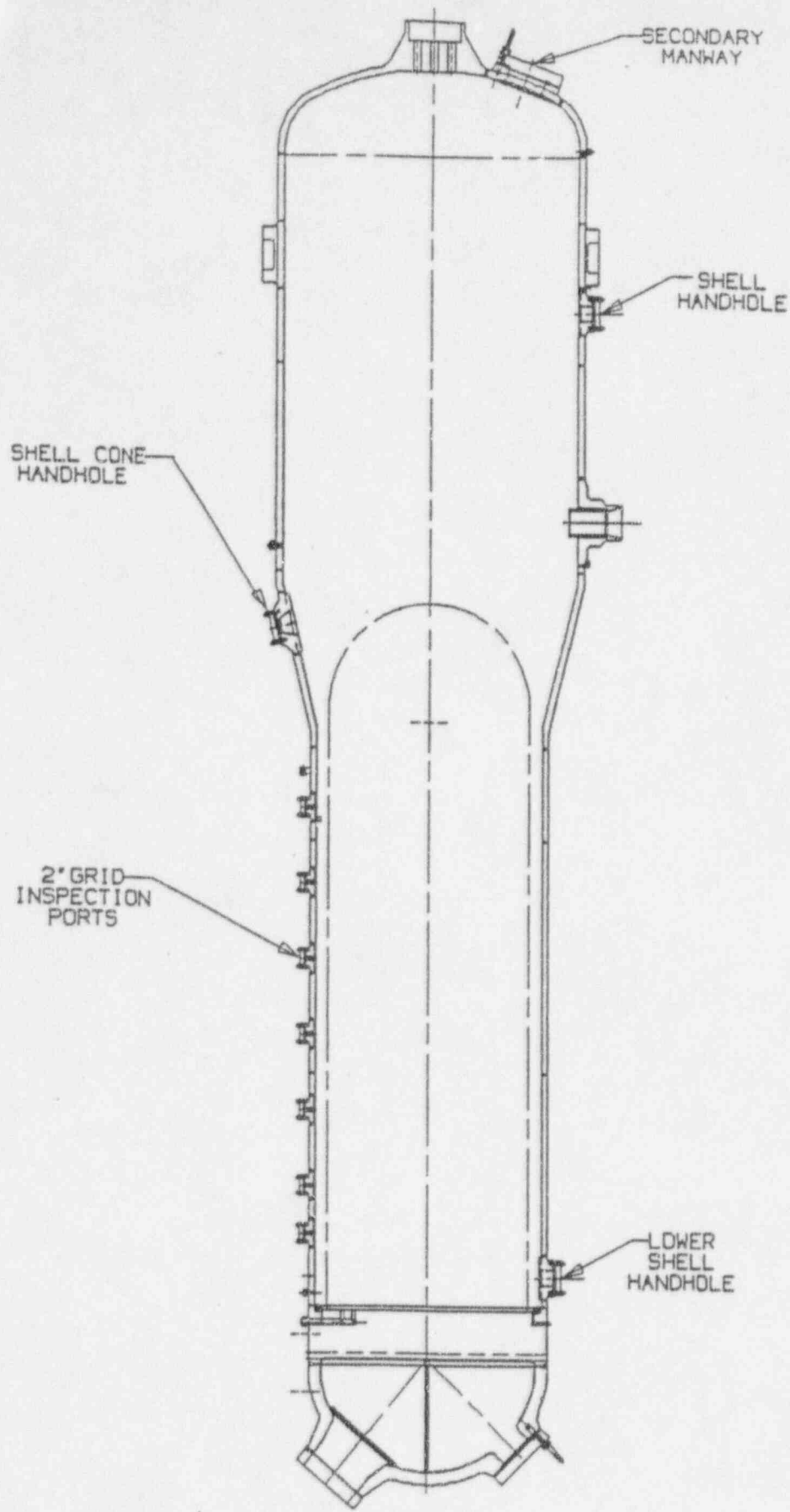
SECONDARY SIDE MANWAYS,
HANDHOLES AND INSPECTION PORTS

FIGURE 2.2.1-5

2.2.2   Steam Generator Tube Design

The RSG tubes are fabricated from thermally treated Alloy 690. This alloy has better overall corrosion resistance than Alloys 600 or 800 in nuclear steam generator environments. Details of the Alloy 690 composition, heat treatment, and mechanical properties are provided in Section 2.3.2. Qualification of the tube-to-tubesheet joining processes are discussed in Section 2.6.3. Pressure stress limits and tube plugging criteria are discussed in Section 2.8.1.

The original and replacement tube bundle designs are geometrically compared. The comparison includes tube outside diameter, tube wall thickness, tube material, average bundle surface area, number of tubes and tube thermal conductivity. Differences exist for the following reasons:

1.   RSG bundle surface area is larger primarily due to the replacement of the integral preheater OSG with a non-preheater RSG.

2.   RSG tubes have lower thermal conductivity due to the change in tubing material from Alloy 600 to Alloy 690.

3.   To facilitate more tubes and corresponding larger surface area the tube O.D. has decreased allowing a thinner wall.

2.2.3   Tube-to-Tubesheet Joint

The RSG tubes are flush welded to the primary face of the tube sheet and hydraulically expanded to maximize mechanical strength and to seal the tube to tubesheet crevice. This precludes crevice or stress corrosion in the tubesheet area. The tubes are installed into the tubesheet after the RSG lower shell and primary head assembly have been welded and received Post Weld Heat Treatment (PWHT). This precludes tube sensitization concerns. The tubes are seal welded to the tubesheet and hydraulically expanded within the full thickness of the tubesheet. Seal welding and expansion of the tubes after PWHT avoids subjecting the tube to tubesheet joint to thermal stresses from these operations and eliminates concern over loosening of tubes or creation of crevices as a result of relaxation of the expanded region.

The tube to tubesheet joint geometry at the secondary face of the tubesheet is shown in Figure 2.2.3-1. The following paragraphs provide further information of the tube to tubesheet joint. Qualifications of the expansion processes are described in Section 2.6.3.

2.2.3.1   Tube-to-Tubesheet Welding

The flush tube-to-tubesheet weld has been applied successfully to twelve steam generators for three 600 MWe power stations (Gentilly and Point Lepreau in Canada and Embalse, Cordoba in Argentina). The generators have been in service since 1983 with no tube joint problems reported. The smooth weld profile has a crown approximately 0.025 in. high and

negligible tube diameter reduction. If necessary, the tube ends are sized by rolling to the minimum expansion diameter to allow subsequent use of tube repair or inspection equipment. The tube-to-tubesheet weld is designed, analyzed, performed and examined in accordance with ASME Section III criteria.

2.2.3.2   Hydraulic Expansion

RSG tubes are hydraulically expanded through essentially the entire thickness of the tubesheet. The length of the expansion mandrel is determined by the thickness of the tubesheet with hydraulic seals positioned on the mandrel to control the length of tube expanded. The hydraulic seals are of elastomeric material and designed so that no metal parts are impressed upon the inside surface of the tube when the hydraulic pressure is applied. The position of the seal at the secondary face of the tubesheet is controlled to ensure that expansion of the tube is as close as possible to the secondary face of the tubesheet without going past the face. This is detailed in Figure 2.2.3-1.

For peripheral tubes, where access is limited by curvature of the primary head, expansion is performed in two overlapping zones, using a shorter expansion mandrel. The shorter mandrel can access the peripheral tubes without interfering with the primary head. The expansion zones overlap near the center of the tubesheet to ensure full depth expansion.

To ensure that all tube-to-tubesheet joining operations can be satisfactorily performed, a ten-tube sample is constructed. It simulates the full tubesheet thickness and uses materials identical to those used in the steam generator. All processes, procedures and inspections approved for use in manufacturing the tube-to-tubesheet joint are performed. Prior to RSG fabrication, the sample is examined by sectioning to verify that manufacturing operations were correctly performed and results are satisfactory.

Tests on hydraulically expanded joints made with Alloy 690 tubes, in closely fitted holes (the BWI practice) have shown that residual stresses exist in the transition region between the expanded and unexpanded tube. The hydraulic expansion process has been designed and qualified to minimize residual stresses while maintaining joint integrity. Qualification of the hydraulic expansion processes is discussed in Section 2.6.3. BWI has successfully hydraulically expanded approximately 334,000 tubes in thirty-eight steam generators. There has been no case where a tube required plugging due to an expansion non-conformance.

After expansion the inside profile of each tube is measured through the entire expanded area of the tubesheet (including the transition) using an eddy current method and recorded. The measurements indicate both the position and condition of the tube expansion, and become a baseline for subsequent inservice inspection. Section 3.2.6.6 describes the baseline inspection. Test results are documented and supplied to the owner.

2.2.3.3   Inservice Inspection

The RSG design provides the capability to perform inservice inspections in accordance with the requirements set forth in ASME Section XI.

$\frac{1}{32}$" MIN. TO $\frac{1}{8}$" MAX.   FINAL HYDRAULIC EXPANSION ZONE TO

OVER-LAP TACK EXPANSION

PRIMARY SIDE

SEAL WELD

SECONDARY SIDE

0.6875" $^{+0.000"}_{-0.005"}$ O.D. × 0.040" $^{±0.004"}$ WALL

0.605" DIA. MIN.

(TUBESHEET HOLE DIA. - 0.6955" $^{+0.0050}_{-0.0040}$)
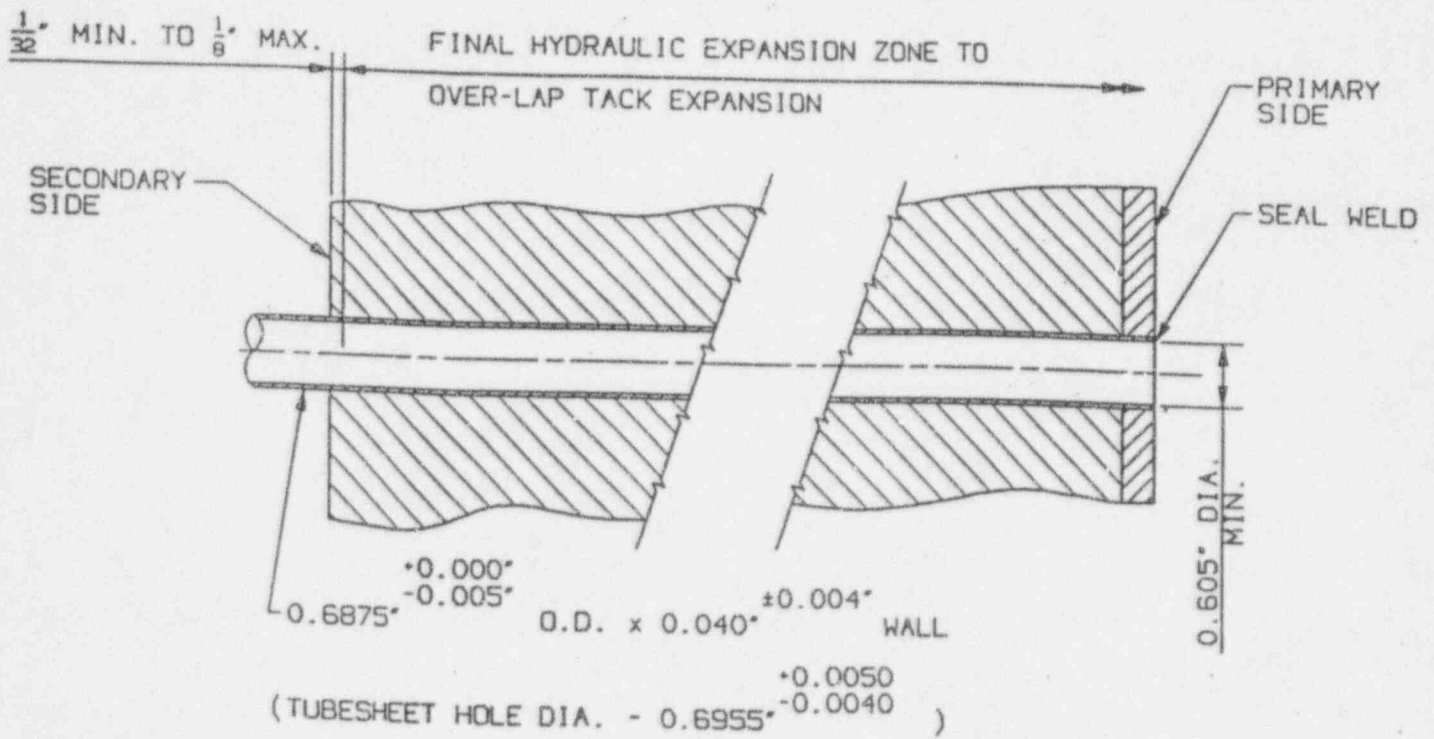
JOINT GEOMETRY AT THE PRIMARY AND
SECONDARY FACE OF THE TUBESHEET

FIGURE 2.2.3-1

2.2.4   Tube Bundle Support System

This section describes the RSG lattice grid and U-bend supports, and measures to minimize flow induced vibration (FIV), dry out potential, flow resistance, wear susceptibility and discusses the structural integrity of the tube support systems.   ⚠4

2.2.4.1   Lattice Grid Tube Supports

The RSG design uses a Type 410S stainless steel lattice grid tube support. BWI has experience with lattice grid and broached plate tube support designs. From this experience BWI concludes that the lattice grid is superior for a recirculating steam generator. The lattice grid provides:

- High circulation rates (through lower flow resistance).

- Superior strength (capable of sustaining very high seismic loads, does not require tie rods)

- Superior vibration restraint and fretting resistance.   ⚠4

- Lower tendency to accumulate deposits than a broached plate (line contact with the tube rather than "area" contact provided by a broached plate).

- Reduced denting potential due to selection of stainless steel.

Examples of the success of this design include Pickering A, with lattice grids, 20 years operation, and tubesheet sludge accumulation (the plant ran 3 years on phosphates) but no lattice grid deposit buildup (determined by visual inspection) and no under-deposit tube failures.

Figures 2.2.4-1 and 2.2.4-2 show the details of a lattice grid. The lattice grid is made up of a series of high bars (approximately 3 inches in width) oriented 30° and 150° to the tube free lane and located every sixth pitch, to accommodate the steam generator loading conditions. Low bars (approximately 1 inch width) are located at every pitch location between the high bars. All low bars flush to the top of the high bars are oriented at 30° to the tube free lane and all low bars flush to the bottom plane of the high bars are oriented 150° to the tube free lane. The bar ends are fitted into precise slots of a specially designed peripheral support ring, which is then sandwiched by two outer retainer rings held together by studs and lock welded acorn nuts. To further enhance stability of the grid, tube ⚠4 free lane support beams and span-breaker bars are secured on the upper and lower surfaces of the grid.

Lattice supports are positioned within the steam generator shroud at elevations selected to prevent flow induced vibration while not creating excessive flow resistance. The tubes are ⚠4 held in position within the diamond-shaped bar opening which provide line support contact. This minimizes the area of "crevices" between the tubes and bars which could trap corrosion

products and eliminates any stagnant spots responsible for "dry-out" caused by local superheat.

All of the lattice supports are identical except that the lowermost lattice incorporates a differential resistance lattice grid (Figure 2.2.4-3) which resembles that of a regular grid. However, the low bars located toward the bundle periphery are replaced by medium bars (approximately 2½" in width). As a result, the flow passages through these regions offers more resistance to flow and the fluid is preferentially directed to penetrate into the central region of the tube bundle. A drilled flow distribution baffle is not used. Since a distribution plate is simply a drilled plate with slightly oversized holes, it may accumulate deposits and possibly become plugged.

Tests conducted by BWI have shown that the in-plane strength of lattice grid supports is higher than that of broached plate supports. This is important to seismic, shipping and handling requirements. Extensive laboratory testing and computer modelling have confirmed that the out-of-plane load handling capability of the BWI lattice grid is superior to the broached plate design.

The tube bundle is analyzed to determine tube vibration characteristics and the effectiveness of lattice grids in suppressing vibration. The results show that lattice grids are the best support system for damping tube vibration and minimizing tube wear due to fretting. Flow-induced vibration modelling is discussed in Section 2.6.1. BWI has refined the grid manufacturing processes to allow very close tube-to-grid clearances so that tube vibration and wear potential are further reduced.

2.2.4.2   U-Bend Supports

Like the lattice grid tube supports, the Type 410S stainless steel Flat bar U-bend Restraint (FUR) system provides effective, close clearance supports of the upper regions of the RSG tubes to prevent flow-induced vibration. The potential for fretting is reduced by compatibility with the tube material and longer contact length than is provided by an AVB support system. All tubes are supported by FURs. The FURs provide open flow paths and line contact support at all locations in the bundle, reducing the potential for sludge build-up.

2.2.4.2.1   Design Configuration

The FURs incorporate a series of flat bar fan assemblies on each side of the bundle, positioned between each layer of tubes as shown in Figure 2.2.4-4. The fans are positioned so that all U-bends are supported at close intervals. Fan assemblies consist of up to four fan finger bars at diagonal positions, connected at their lower ends to a nearly horizontal bar by full-penetration, heat treated welds. The nearly horizontal bar provides support for the smallest radius U-bends. The wide FURs distribute contact force to minimize the possibility of fretting.

All U-bends are supported by the flat bars. The innermost tubes are installed with their U-

bends in a plane that is skewed with respect to the channel head divider and tube-free lane (TFL). This permits larger radius bends than if these tubes were installed perpendicular to the TFL. The FURs do not pass through this part of the tube bundle because of the skewed tube plane. The small-radius tubes are supported below their bend tangent points by the inner ends of the nearly-horizontal bars.

The FURs are made of Type 410S precision cold rolled steel that provides high resistance to fretting wear, excellent strength and high resistance to corrosion-related tube denting. Further information on selection of support system materials appears in Section 2.3.2.

2.2.4.2.2 Flow Characteristics

The FURs are designed with all spaces oriented with an upward slope. This promotes continuous sweeping during operation. FURs do not cross the bundle centerline. This avoids creation of spaces where deposits might collect.

Tubes are supported by line contact and bars are offset within each row to provide more flow area than would exist with in-line bar placement. Bar array position generally follows the pattern of unobstructed U-bend flow. A flow diagram of an unobstructed U-bend and FUR array is shown in Figure 2.2.4-5. Cross flow is low and the FURs do not significantly impede or disrupt flow.

2.2.4.2.3 Flexibility and Thermal Motions

Free expansion of the U-bend during operation is essential in order to avoid tube stress and potential tube damage. The FUR system allows free expansion of the U-bend tubes without sliding between the bars. The FUR assembly is supported by, and moves with the outermost layer of tubes rather than being anchored to the upper lattice support. The FUR and tube bundle move up and down together during heatup and cooldown. During power operation the tube hot- and cold legs have slightly unequal leg temperatures that create a slight angularity, shown (exaggerated) in Figure 2.2.4-6. Analyses show that for a U-bend assembly under the worst case conditions, tube-to-support angularity is easily accommodated by the lattice supports without risk of tube damage, tube lockup, or loss of tube fixity at the top support.

2.2.4.2.4 Support of FUR Assembly

The weight of the fan assemblies is supported by arch bar assemblies which transfer the weight to the outermost layer of tubes via the J-tabs (see Figure 2.2.4-7). The FUR fan finger bars (a) are notched at their upper ends. These bars are collected by a slotted clamping bar (b) which is attached by welded pins to the arch bar (c). The arch bars thereby collect all the weight of the fan assemblies. The weight of the fan assemblies is transferred to the outermost layer of the tubes by "J" tabs (d) installed after completion of the U-bend assembly and positioned to uniformly contact the completed tube assembly. Tube stress resulting from this weight is small. This is confirmed by a tube bundle/FUR interaction analysis.

The arch bar/fan finger assemblies are prevented from splaying apart under dead weight loads during operation by tie tubes that maintain arch bar spacing (Figure 2.2.4-4).

The arrangement described above accommodates all operating loads and motions. Assembly, handling and shipping loads are supported by temporary restraints that are removed at the site by construction personnel after RSG installation, and prior to operating the RSG Seismic tube bundle loads are supported by the FURs and lattice support. As there is no connection between the FUR assembly and the shroud, U-bend deflections during earthquake will not damage the tubing. The flat bars do not absorb the full seismic load, but moderate the deflection of the tubes relative to each other. Main steam line break loads are insufficient to lift the FUR assembly.

2.2.4.3   Design to Minimize Flow Induced Vibration (FIV)

Prevention of excessive FIV and fretting wear is achieved by a combination of design, analysis and testing. The FURs are arranged to meet the design limits established for Fluid Elastic Instability (FEI) and for response to turbulence. These analysis methods and criteria are discussed further in Section 2.6.1.

2.2.4.3.1   Clearances

Small U-bend support clearances are maintained while avoiding tube/bar interference problems (marking of the tubes by the bars, splaying of the bundle due to bar tolerance accumulation, or buildup of bundle thickness) as tubing progresses. The optimum range of flat bar U-bend support clearance was verified by an air flow test (Reference 1). This test compared the effectiveness of flat bar U-bend supports to scalloped bar (360° drilled hole) supports. The test showed that flat bars with small clearances provided more effective support than the scalloped bar design with larger clearances. Tests with larger clearances showed significant response in all directions, including in-plane (the "weak" direction), for either flat bar or scalloped bar supports. The flat bars more effectively suppressed instability and in-plane turbulence response.

2.2.4.3.2   Bar Width

Fan finger bar width is sized to provide line contact that minimizes the potential for fretting. Comparative autoclave fretting tests have shown that the wear rate is substantially reduced as bar width increases (Figure 7 of Reference 2).

2.2.4.3.3   Fretting Assessment

Potential fretting is assessed by performing a FIV sensitivity analysis. FIV methods are discussed in Section 2.6.1. The FUR design is qualitatively compared to other designs by comparing the relevant U-bend support parameters (material selection, bar widths, support clearance and span lengths). Design assurance is achieved by conservatively meeting the FIV analysis parameters for FEI and Random Turbulences Excitation (RTE), and then by assuring that support effectiveness, materials and clearances are optimum.

Fretting is a major design consideration. The relevant parameters are: 1) U-bend flow loading, 2) support positions, 3) support material, 4) support clearance, and 5) support contact length. These parameters are considered in the design of the FUR which is shown schematically in Figure 2.2.4-4 and are addressed below:

Flow loading is determined for a given steam output by the circulation rate and the U-bend tube and support geometry. Having established the desired circulation rate, the velocity and quality distributions are determined by a 3 dimensional thermal hydraulic analysis code (See Section 2.6.2). The optimum geometry is one in which there is least interference with the free release of riser flow. The RSG design achieves this with its open flow configuration and bar orientation which is generally compatible with the flow direction.

Optimization of the position of U-bend supports is based on the FIV analysis for FEI and for turbulence response (Section 2.6.1). The result is a design with short tube spans, high natural frequencies, small response to turbulence and large margin for instability.

Selection of Type 410S as the support material in combination with Alloy 690 tubes provides a high degree of fretting resistance. The Alloy 690/410S combination has the lowest wear rate of any of the available combinations. Fretting wear test results from AECL indicate that the fretting wear rate for the Alloy 690/410S combination is essentially the same as that for Alloy 800/410S and slightly better than Alloy 600/410S (Reference 3).

The mean diametral U-bend support clearance has been set at a very low value. This clearance was selected based on comparative U-bend air flow testing (Reference 1) which indicated that flat bar U-bend supports with small clearances (0.003" to 0.010") provided good "pinned" support conditions and that the effectiveness of such a support was better than that of a scalloped bar support with a 0.020" clearance (even though the scalloped bar provided a 360° drilled support configuration). This mean clearance provides a snug overall design while still permitting thermal motions.

The RSG design provides substantial contact length compared to about 0.40" in other designs. This contact length reduces the contact stresses which result from ongoing turbulent excitation. Comparative autoclave tests have shown that the wear rate is substantially reduced with a greater bar width (Reference 2).

The parameters noted above are the same or better than those used for the Millstone 2 RSG which is operating successfully.

References for Section 2.2.4

1.   "The Effects of Flat Bar Supports on the Crossflow Induced Response of Heat Exchanger U-Tubes", D. S. Weaver, W. Schneider, Journal of Engineering for Power, October, 1983.

2.   Third Keswick International Conference of Vibration in Nuclear Power Plants, England, May 1982, "Heat Exchanger Tube Fretting Wear: Review of Application to Design", P. L. Ko, PhD.

3.   AECL Research, Report RC-1314, "PWR Replacement Steam Generator Fretting-Wear", A. B. Chow and D. A. Grandison, November 1994.   /4\
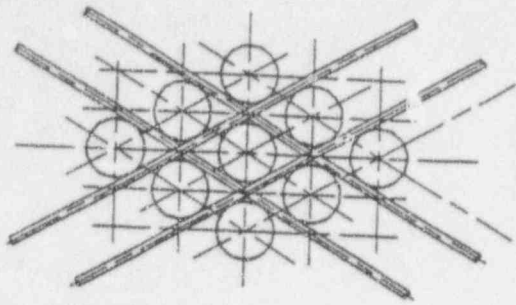
GRID PLAN

ARCH/CLAMPING BAR
FAN BLADE ASSEMBLY
(TYPICAL)

TIE TUBE
ASSEMBLY
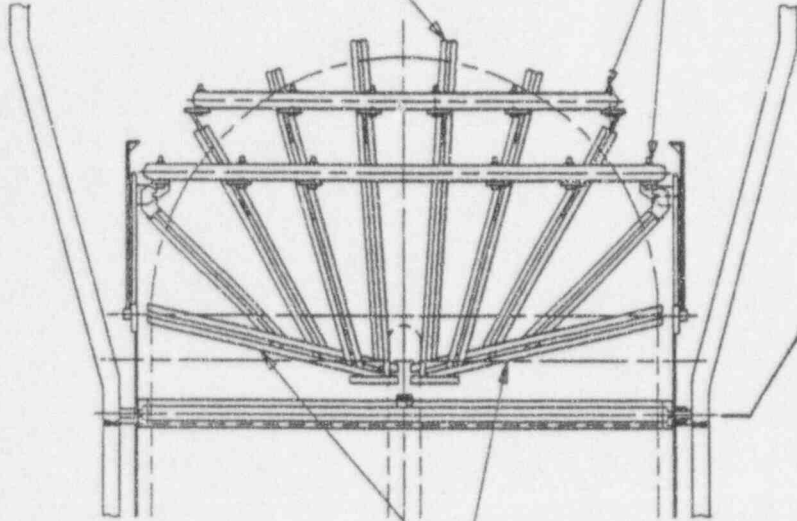
FULL GRID
ISOMETRIC

FAN BLADE ASSEMBLY
CONNECTORS

HIGH BAR

LOW BAR

ISOMETRIC VIEW

DETAILS OF BWI. LATTICE
GRID TUBE SUPPORT

FIGURE 2.2.4-1

"Y-2"

"X-1"

"X-2"

"Y-1"
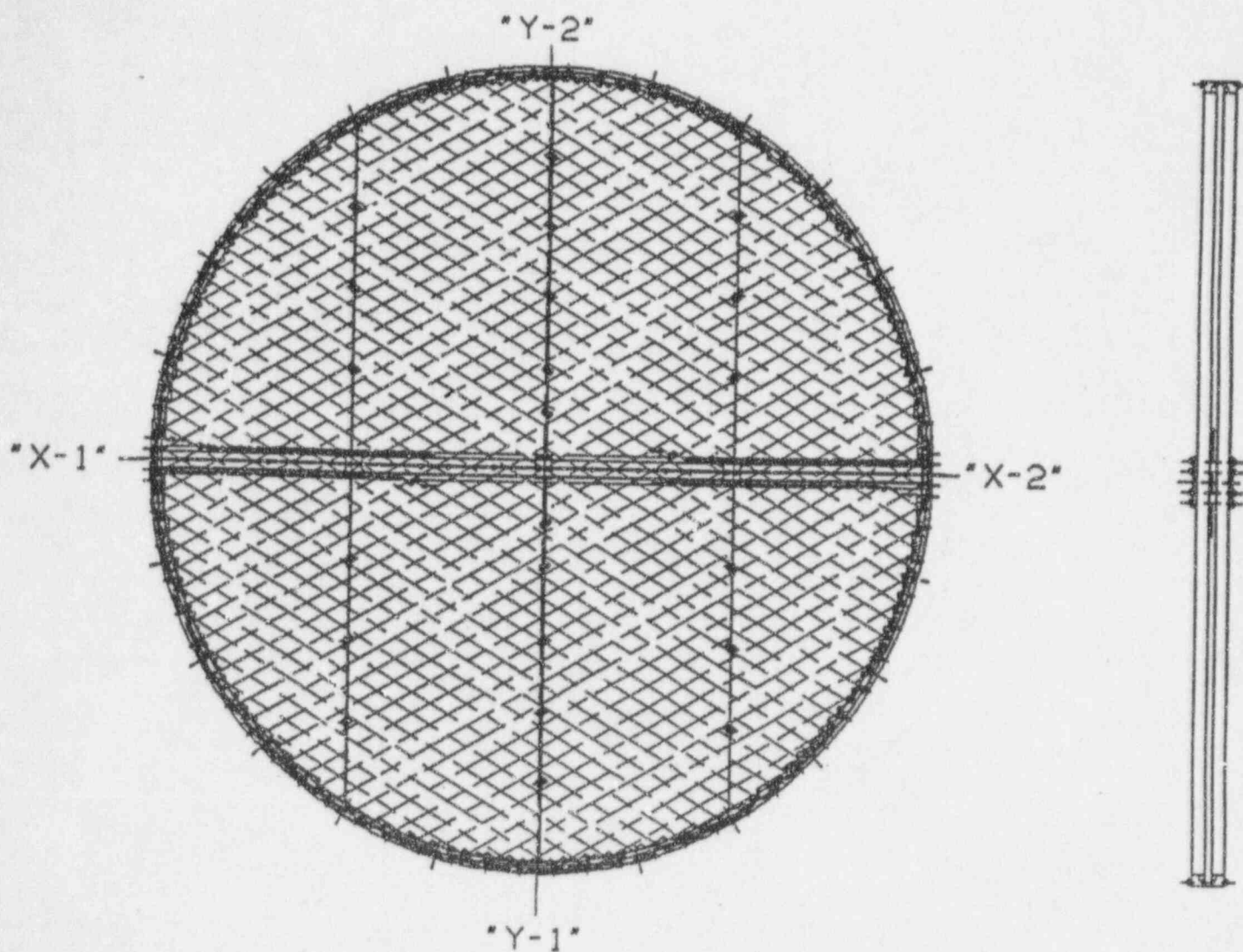
BWI LATTICE GRID ASSEMBLY

FIGURE 2.2.4-2

DIFFERENTIAL RESISTANCE LATTICE GRID

FIGURE 2.2.4-3

ARCH/CLAMPING BAR
FAN BLADE ASSEMBLY
(TYPICAL)

TIE TUBE
ASSEMBLY

FAN BLADE ASSEMBLY
CONNECTORS
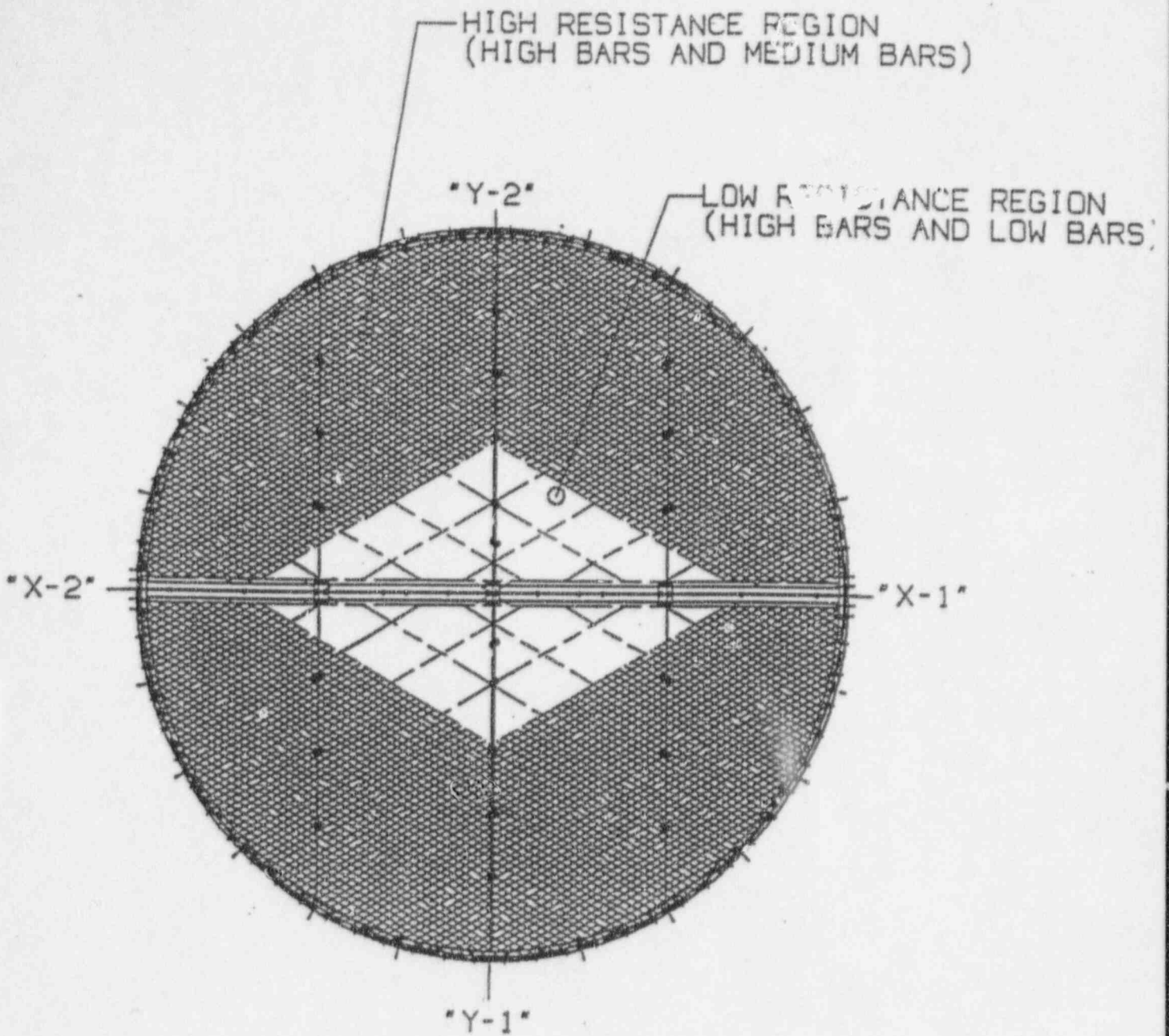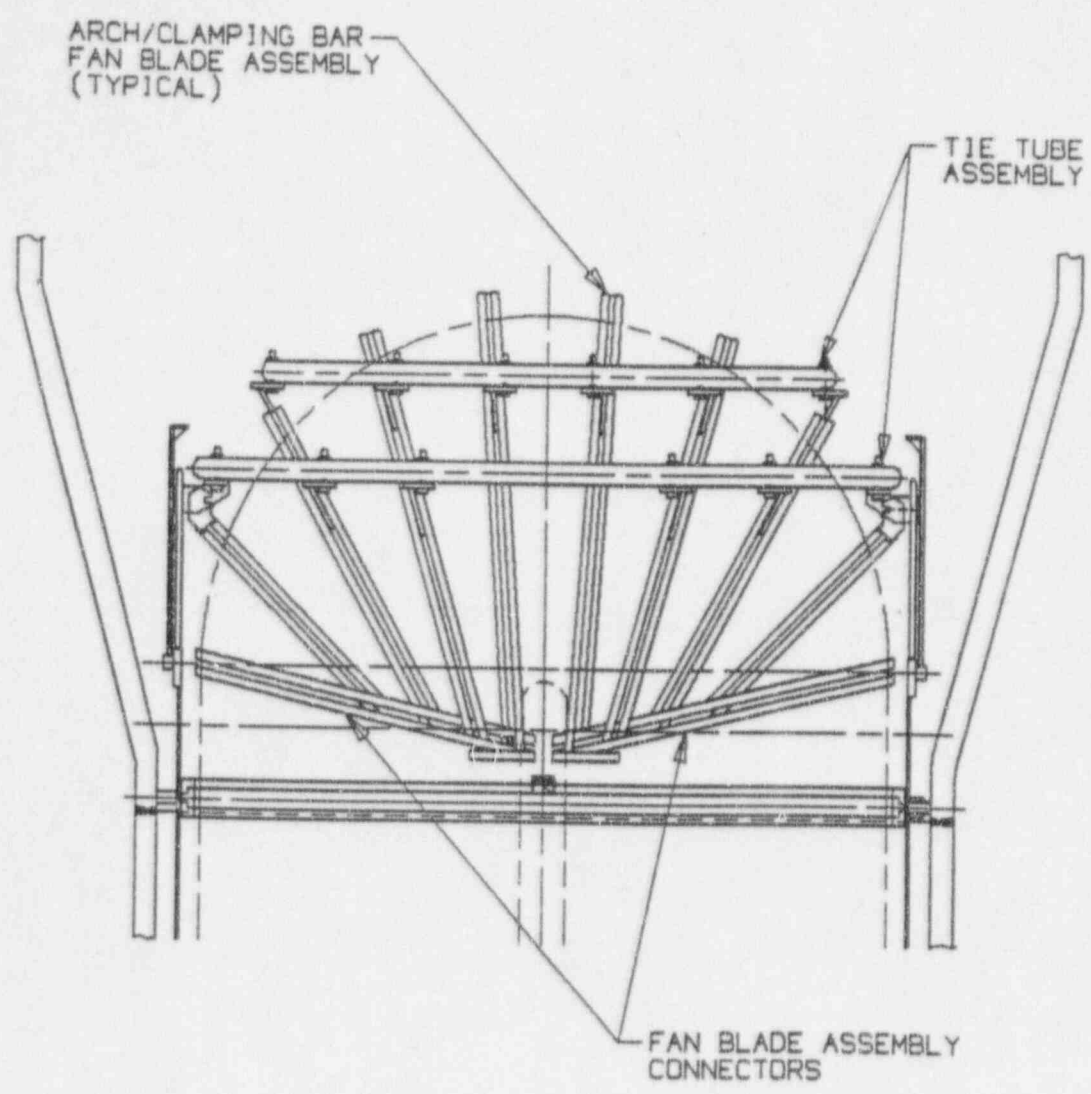
FLAT BAR 'U'-BEND RESTRAINT CONFIGURATION

FIGURE 2.2.4-4

Hot Leg

Cold Leg

4°

16°

29°

45°

Fan Bars

Connector Bar
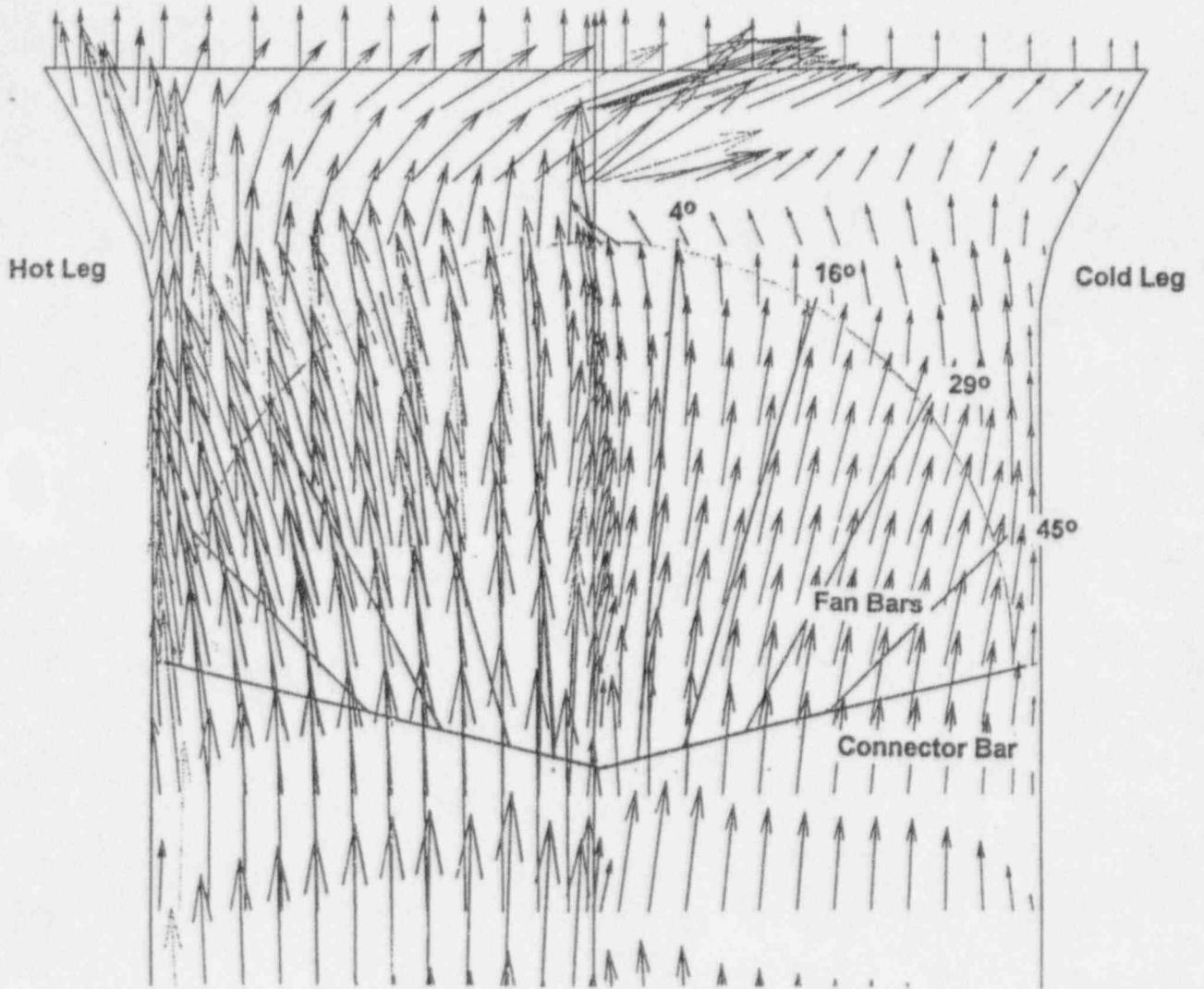
FLOW PATTERN

FIGURE 2.2.4-5
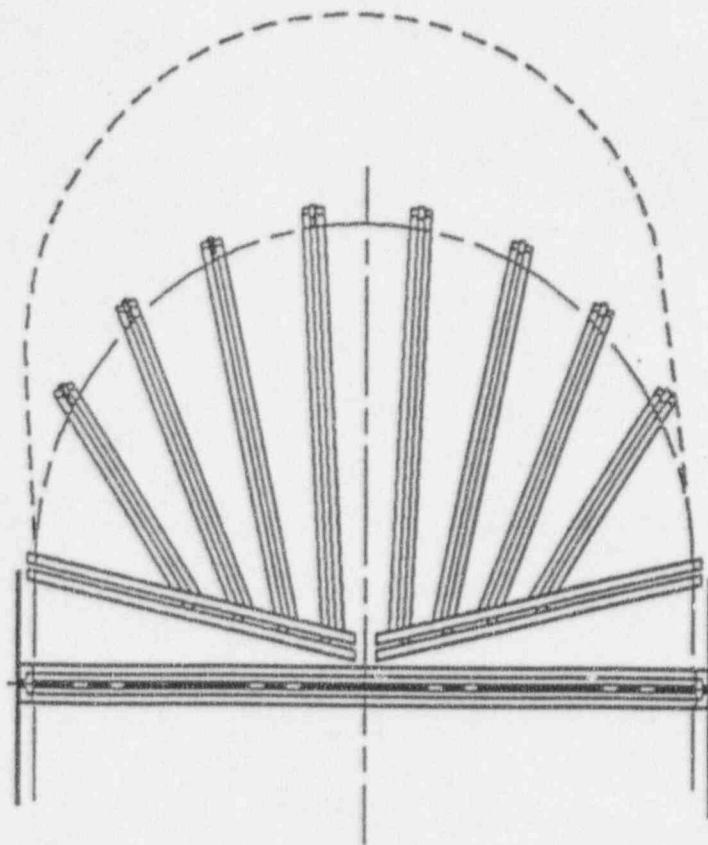
'U'-BEND MOTION (EXAGGERATED)

COLD TO FULL POWER

FIGURE 2.2.4-6
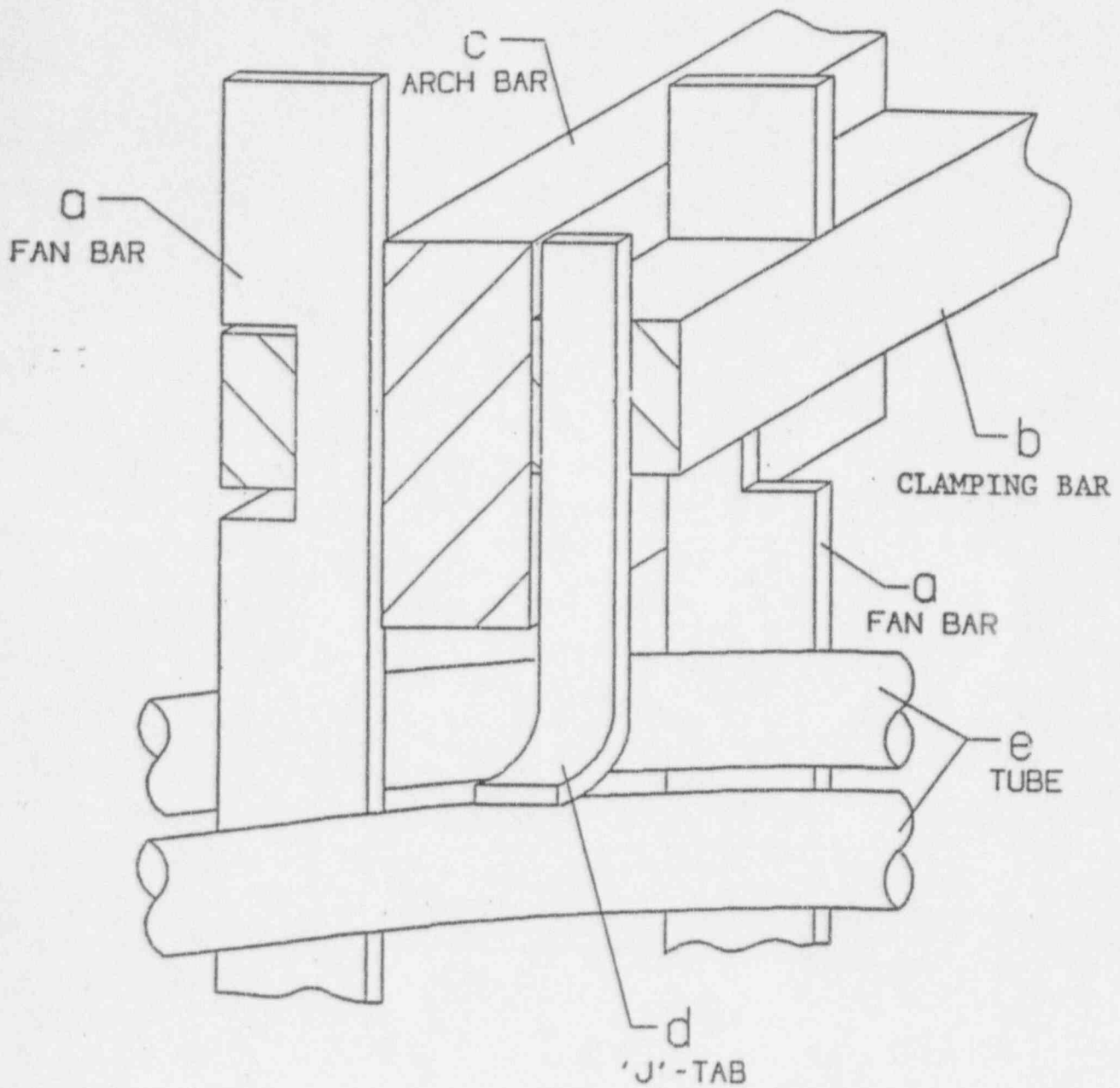
ARCH BAR ASSEMBLY

FIGURE 2.2.4-7

2.2.5   Internal Feedwater System

This section describes the RSG feedwater distribution system and design measures to preclude damage from water hammer, thermal stratification, erosion and internal feedwater header collapse. Water hammer has affected more than half of the operational PWR plants in the U.S. Thermal stratification has caused fatigue cracks on the inner surfaces of thermal sleeves, feedwater nozzles and feedwater piping. High flow velocities and abrupt changes in flow direction have caused erosion where the feedwater flow splits to enter the header rings, and in the feedwater discharge tubes located along the header. Feedwater headers have collapsed due to external pressure. The RSG feedwater distribution system design recognizes these potential problems and includes features to address them.

The RSG feedwater distribution system (shown in Figures 2.2.5-1, 2.2.5-2 and 2.2.5-3) is a split ring design connected via a T-section to a "goose neck" assembly attached to the thermal sleeve in the feedwater piping. The feedwater header is supported by the thermal sleeve/feedwater piping weld interface, and by supporting lugs around the ring circumference attached to the internal shroud. Support lugs are located on the header pipe at approximately 90° to the feedwater nozzle, and at the header ends (near the split in the header ring) opposite the feedwater nozzle. The header lugs are vertical plate structures which have elongated holes through which the header passes. This provides support to restrain motion in the vertical direction, while allowing thermal growth in the horizontal plane. Lateral stability of the feedwater ring is accomplished by restricting motion at the split location in the direction perpendicular to the feedwater nozzle leg. This design provides a support system that accommodates thermal motions and potential loads due to water hammer or system pump pressure pulses as well as all operational, seismic and burst pipe loads. △4

The RSG feedwater distribution system satisfies all current NRC recommendations with respect to water hammer, provides flow stratification mitigation and addresses industry concerns regarding corrosion, corrosion cracking, thermal fatigue and material erosion.

2.2.5.1   The Water Hammer Mechanism

Water hammer in steam generators results from rapid condensation and collapse of steam pockets in the feedwater system. These can cause potentially damaging pressure pulses in feedwater piping. Water hammer can occur under various combinations of operating conditions and piping geometry. Most steam generator water hammer events have involved feedwater headers that discharge downward from a header. During certain plant transients, steam generator water level dropped below the feedwater header, and allowed the header to become partially filled with steam. Increased feedwater flow condensed the steam at the steam-water interface. This caused a counter-flow of steam above the level of the feedwater. Turbulence trapped steam pockets which were condensed. Slugs of water driven by pressure accelerated upstream to fill the void. This sequence is termed "steam-water slugging", and is generally accepted as the initiating mechanism of feedwater header water hammer events.

A total of 27 water hammer events were reported between 1969 and 1982 (Reference 1). In 1977, all recirculating steam generator PWR licensees were requested to submit hardware and procedural proposals to reduce steam generator water hammer susceptibility. The NRC declared water hammer to be an Unresolved Safety Issue in 1978. Design change recommendations were made and implemented, and on-site testing of the new designs was documented for 38 operating plants. In 1982, the NRC considered the top-feeding steam generator water hammer issue to be resolved (Reference 2). Design recommendations addressing steam generator water hammer are described in NRC Branch Technical Position ASB 10-2 (Reference 3). These are addressed in Section 2.2.5.2. The following section discusses these recommendations relative to the RSG design and describes features of the RSG that reduce or eliminate the potential for water hammer.

2.2.5.2   Design to Preclude Water Hammer

NRC Branch Technical Position ASB-10-2, "Design Guidelines for Avoiding Water Hammers in Steam Generators", (Reference 3), for reviews of top-feed steam generator designs identifies four items which serve "to reduce or eliminate the potential for water hammer in the feedwater system:

a)   Prevent or delay water draining from the feed ring following a drop in steam generator water level by means such as top discharge J-Tubes and limiting feed ring seal assembly leakage.

b)   Minimize the volume of feedwater piping external to the steam generator which could pocket steam using the shortest possible (less than seven feet) horizontal run of inlet piping to the steam generator feed ring.

c)   Perform tests acceptable to NRC to verify that unacceptable feedwater hammer will not occur using the plant operating procedures for normal and emergency restoration of steam generator water level following loss of normal feedwater and possible draining of the feed ring. Provide the procedures for these tests for approval before conducting the tests and submit the results from such tests.

d)   Implement pipe refill flow limits where practical."

Items (a) and (b) address steam generator and piping design, while items (c) and (d) address operating and test procedures. Items (c) and (d) have been resolved (see Reference 2). The RSG feedwater header design incorporates J-tubes connected on top of the header to help prevent header draining and formation of steam pockets thus addressing item (a). Item (b) is addressed by minimizing the straight length within the feedwater nozzle prior to the gooseneck.

Figure 2.2.5-1 shows the steam drum internals and the feedwater piping. Operating water levels and the primary separator deck locations are also shown. Design improvements include lowering the primary separator deck (below the normal water level) and the

feedwater sparger ring, a "goose neck" inlet design (detail shown in Figure 2.2.5-2), a schedule 80 header and schedule 160 J-tubes (detail shown in Figure 2.2.5-3).

To show that the RSG design reduces the potential for water hammer damage, both the frequency and consequences of water hammer events must be considered. For water hammer to occur, there must be steam in the feedwater piping. This can occur if the steam drum water level falls below a point of discharge or if a leak exists in the internal feedwater piping system. The potential for these conditions is minimized by:

1. .   Reducing the chance of uncovering the feedwater header by:

    a.   Positioning the header as low in the steam drum as possible.

    b.   Providing a design that maximizes steam drum water inventory above the header.

    c.   Avoidance of the transients that uncover the header.

2.   Minimizing drainage of the header once it is uncovered by:

    a.   Utilizing top discharge header with J-tubes.

    b.   Maintaining feedwater flow to keep the header full.

    c.   Eliminating leakage throughout the header assembly (except at the J-tube discharge).

The BWI feedwater header design incorporates J-tubes, internals with maximum secondary side water inventory between the header and the normal water level, and an all-welded thermal sleeve/header assembly from the thermal sleeve/feedwater pipe interface to the J-tube exit. This eliminates the possibility of steam leakage into the feed ring through header joints.

Because evaporation from the feedwater header during steam generator depressurization can cause steam accumulation, potential header dry-out cannot be totally precluded. If a steam pocket does form, the BWI design is less prone to serious consequences because the feedwater pipe goose neck will retard rapid condensation and water-slug acceleration better than a long, thin steam pocket. Additionally, the feedwater header is designed to prevent collapse if a large steam pocket were to condense and create a near vacuum.

Operating BWI recirculating steam generators have not experienced water hammer problems because the BWI feedwater header design meets NRC guidance and improves upon previous designs with respect to prevention of the occurrence and prevention of damage from water hammer events.

2.2.5.3   Thermal Stratification Mechanism

At low flow rates, thermal stratification of the feedwater may occur in the horizontal section of pipe through the feedwater nozzle. This has caused fatigue cracks on the inside surface of the nozzle and feedwater pipe in some steam generators (Reference 4). Thermal stratification occurs at low power levels, when cold, incoming feedwater flows underneath a warmer, less dense stagnant layer of water. With the low degree of mixing at these low flows, the division between cold and warm feedwater remains well defined. A feedwater flow of approximately 600 gpm of feedwater flow at 70°F is typical for hot standby. This flow is low enough to cause an uneven flow distribution across the horizontal portion of the feedwater pipe (Reference 4).

Changes in local pipe wall temperatures associated with a fluctuating thermal layer cause stress cycles that could lead to fatigue failure. NRC Bulletin 79-13 and Information Notice 91-28 describe thermally induced cracks found in many feedwater nozzle-to-pipe welds. A similar concern exists for any horizontal sections of the external feedwater piping system.

2.2.5.4 Design to Minimize Stratification Susceptibility

The potential for flow stratification exists in any horizontal section of feedwater pipe, including the nozzle. Mixing devices in these sections could reduce stratification, but could cause erosion/corrosion or loose parts at higher flow rates. The potential for flow stratification can be reduced procedurally by preventing the introduction of cold feedwater. For example, reverse purge can be employed to continually draw water out of the generator through the main feedwater nozzle during hot standby and low power operation, at which time flow is routed through the auxiliary nozzle. This procedure eliminates extreme shocks of cold feedwater, since when switching from the auxiliary to the main feedwater nozzle and stagnant slugs of cold water have been eliminated at an intermediate power, the feed flow has already been warmed.

The RSG incorporates a "goose neck" between the feedwater pipe and header (Figures 2.2.5-1 and 2.2.5-2). The goose neck limits the volume of pipe that can be filled with cold water. This design minimizes the time to fill the horizontal runs of external feedwater piping resulting in a rapid rise in the hot/cold dividing layer. This rise occurs quickly enough to prevent establishment of severe temperature distributions in the pipe wall. Figure 2.2.5-4 shows the effect of increasing the fill rate on stress intensity with Braschel, et al.'s graph of normalized stress intensity versus rate of elevation of the thermal dividing layer (Reference 4). The vertical velocity of the thermal interface is an important factor in stress intensification. The faster rate of rise of the thermal interface afforded by the RSG design reduces stress intensity.

The BWI internal feedwater distribution design has considered the potential for thermal stratification and incorporates features which minimize the risk of thermal stratification damage. The main feedwater distribution system goose neck design operates effectively to alleviate thermal stratification.

2.2.5.5   Thermal Sleeve

The BWI feedwater distribution design is an all-welded design. The thermal sleeve (shown on Figure 2.2.5-2) is welded to the internal feedwater piping at the goose neck. The goose neck is welded to a Tee that is welded to the feedwater split ring header components. This provides leak tight joints that protect against header drainage. The attachment point is located away from any pressure boundary thermal or geometric discontinuities to avoid stress concentration. To prevent the attachment point between the thermal sleeve and pressure boundary from thermal shock, an inner thermal sleeve attached to the feedwater header downstream of the nozzle is employed. This double thermal sleeve design further protects the feedwater nozzle, the nozzle to shell juncture and the outer sleeve to nozzle juncture from any detrimental effects due to cold feedwater impingement, or other feedwater thermal variations.

Careful positioning of the attachment of the thermal sleeve to the feedwater piping allows for a design which does not interfere with nozzle or pipe UT examinations. Placement of the feedwater distribution system within the downcomer at the relatively open conical elevation, and the placement of the nozzle within the steam drum access space, provide access for examination and inspection of the feedwater distribution header, goose neck and thermal sleeve components. Secondary and primary deck access doors and access panels in the shroud cone facilitate feedwater system access.

2.2.5.6   Feedwater Distribution

The J-tube discharge is below the gap between the feedwater header and the internal shroud and oriented to avoid impingement of feedwater on internal surfaces. This reduces the possibility of erosion. J-tube arrangement is shown in Figures 2.2.5-2 and 2.2.5-3. Feedwater distribution system materials are selected to optimize resistance to erosion/corrosion, thermal fatigue and corrosion cracking. The Alloy 690 J-tubes provide erosion resistance and withstand the effects of thermal gradients and thermal cycles. Feedwater is distributed axisymmetrically around the downcomer to provide a homogeneous temperature fluid to the bundle riser.

2.2.5.7   Maintenance Features

The BWI steam generator is designed to provide access into the unit for inspection and maintenance. An access tunnel, complete with ladder, is positioned immediately below the secondary manway located in the steam drum head. This provides direct access through the secondary deck and into the steam drum region down to the elevation at the primary deck. From the primary deck, the feedwater nozzle, thermal sleeve, gooseneck, and feedwater ring T-section are readily accessible. In addition a door is provided in the primary deck to permit access to the top of the bundle inside the shroud. From this location access may be made by removal of seal welded windows through the shroud cone.

Access to the feedwater header for remote inspection is provided via a handhole through the pressure boundary. This is located at the elevation of the feedwater header and

oriented approximately 180° to the feedwater nozzle, at the header split location. Access into the header for inspection is made through the J-tube located at each end of the header ring sections (at the header split). This allows for remote fiber optic or camera inspection of the entire header system from inside the header.

2.2.5.8   Auxiliary Feedwater System

An auxiliary feedwater system is used for the addition of cooling water during upset ⚠4 emergency or faulted conditions. In addition, during normal operation feedwater may be introduced through the auxiliary feed system at less than 25% power in order to relieve thermal stresses on the main feedwater nozzle. The auxiliary feedwater system uses many features similar to the main such as a welded thermal sleeve, erosion-corrosion resistant material and an upturn in the header to prevent stratified flow. Unlike the main feedwater header, there are no J-tubes and fluid exits out the end of the header in an elevated section. The auxiliary feedwater arrangement is illustrated in Figure 2.2.5.5.

Reference for Section 2.2.5

1.   Serkiz, A. W., Evaluation of Waterhammer Experience in Nuclear Power Plants - Technical Findings Relevant to Unresolved Safety Issue A-1, NRC Report NUREG-0927, Rev. 1, March 1984.

2.   Anderson, N. and Han, J. T., Prevention and Mitigation of Steam Generator Water Hammer Events in PWR Plants, NRC report NUREG-0918, Nov. 1982.

3.   Branch Technical Position ASB 10-2, "Design Guidelines for Avoiding Water Hammers in Steam Generators", Revision 3, April 1984 (attached to Section 10.4.7 of Standard Review Plan for the review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition, NUREG-0800).

4.   Braschel, R., et al., "Thermal Stratification in Steam Generator Feedwater Lines", Journal of Pressure Vessel Technology, February 1981 pages 78-76.

CENTRIFUGAL
SECONDARY
SEPARATORS

PRIMARY
SEPARATOR

NORMAL
WATER
LEVEL

GOOSENECK

FEEDWATER
NOZZLE
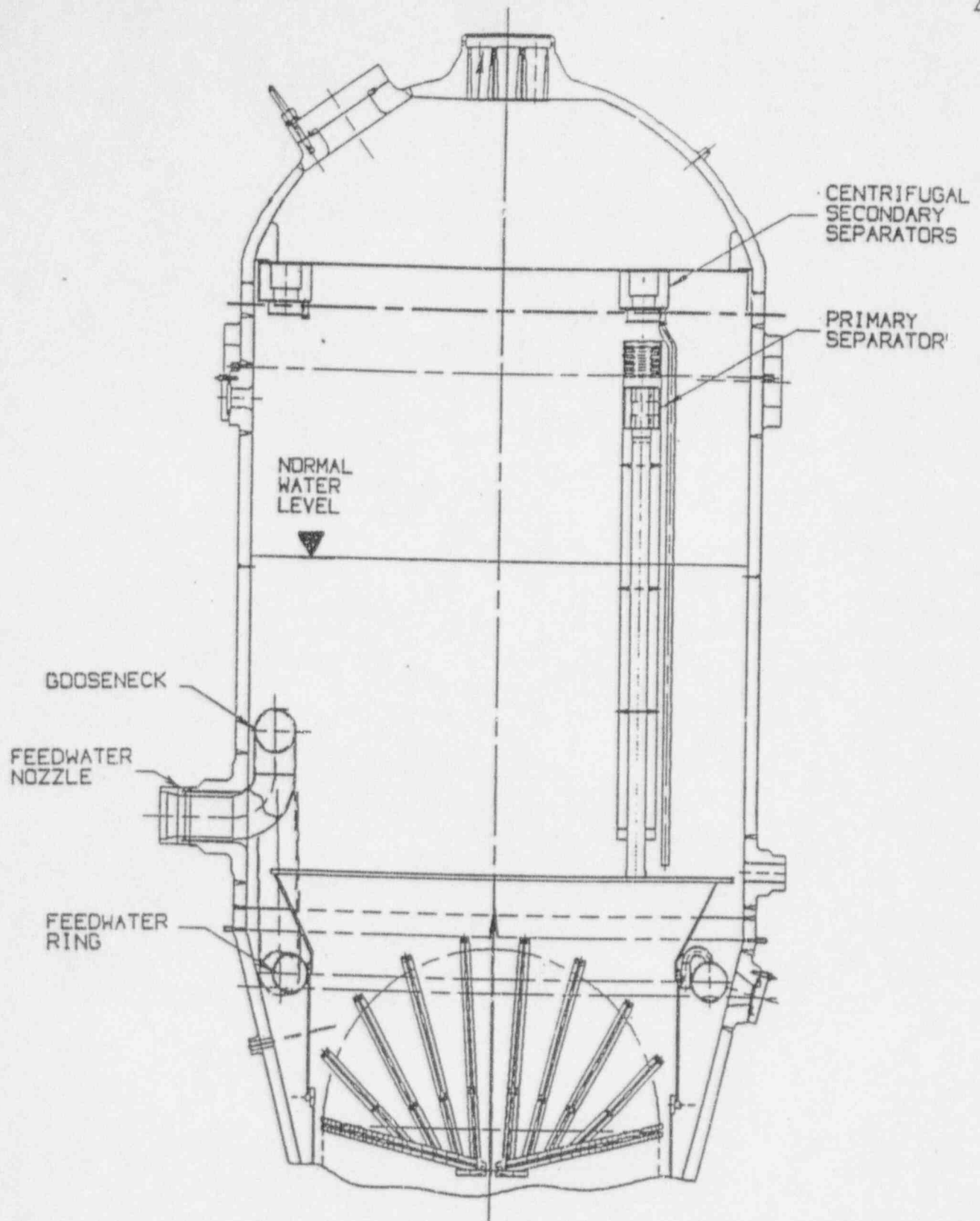
FEEDWATER
RING

STEAM DRUM FEEDWATER
DISTRIBUTION SYSTEM

FIGURE 2.2.5-1
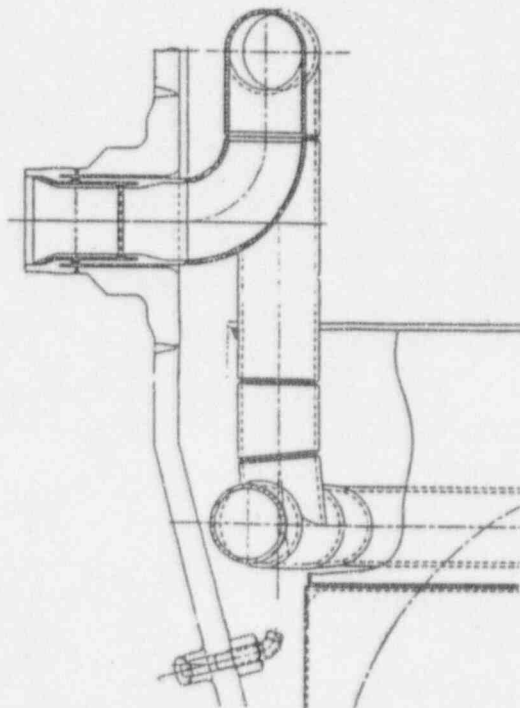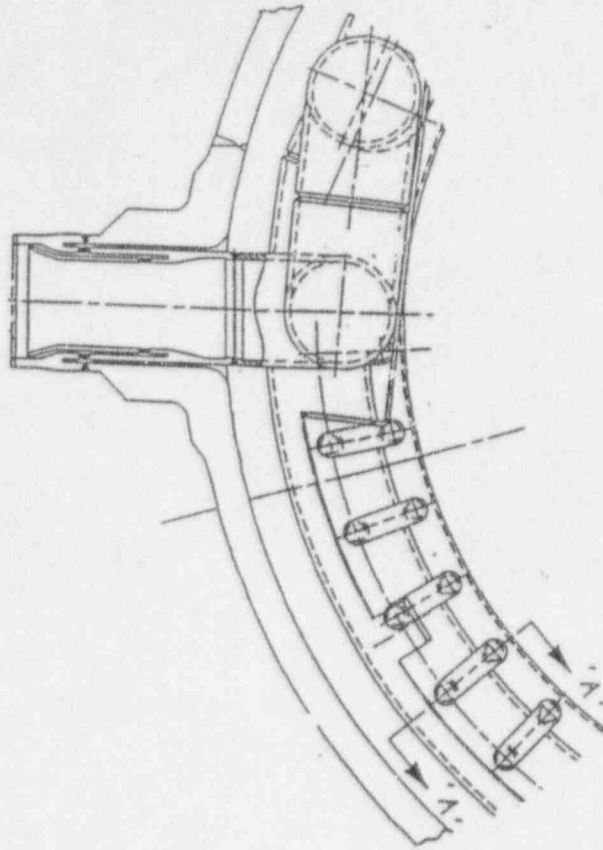
GOOSENECK DESIGN

FIGURE 2.2.5-2

SHROUD

J-TUBE

SHELL

HEADER PLAN VIEW

SECTION "A-A"
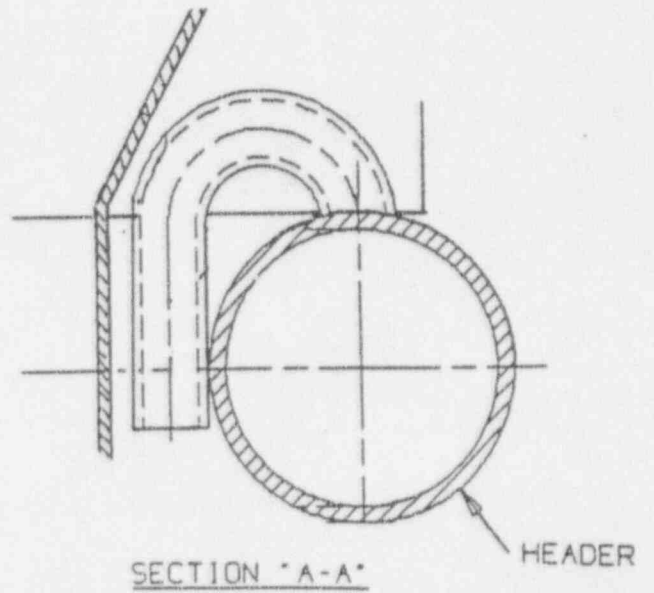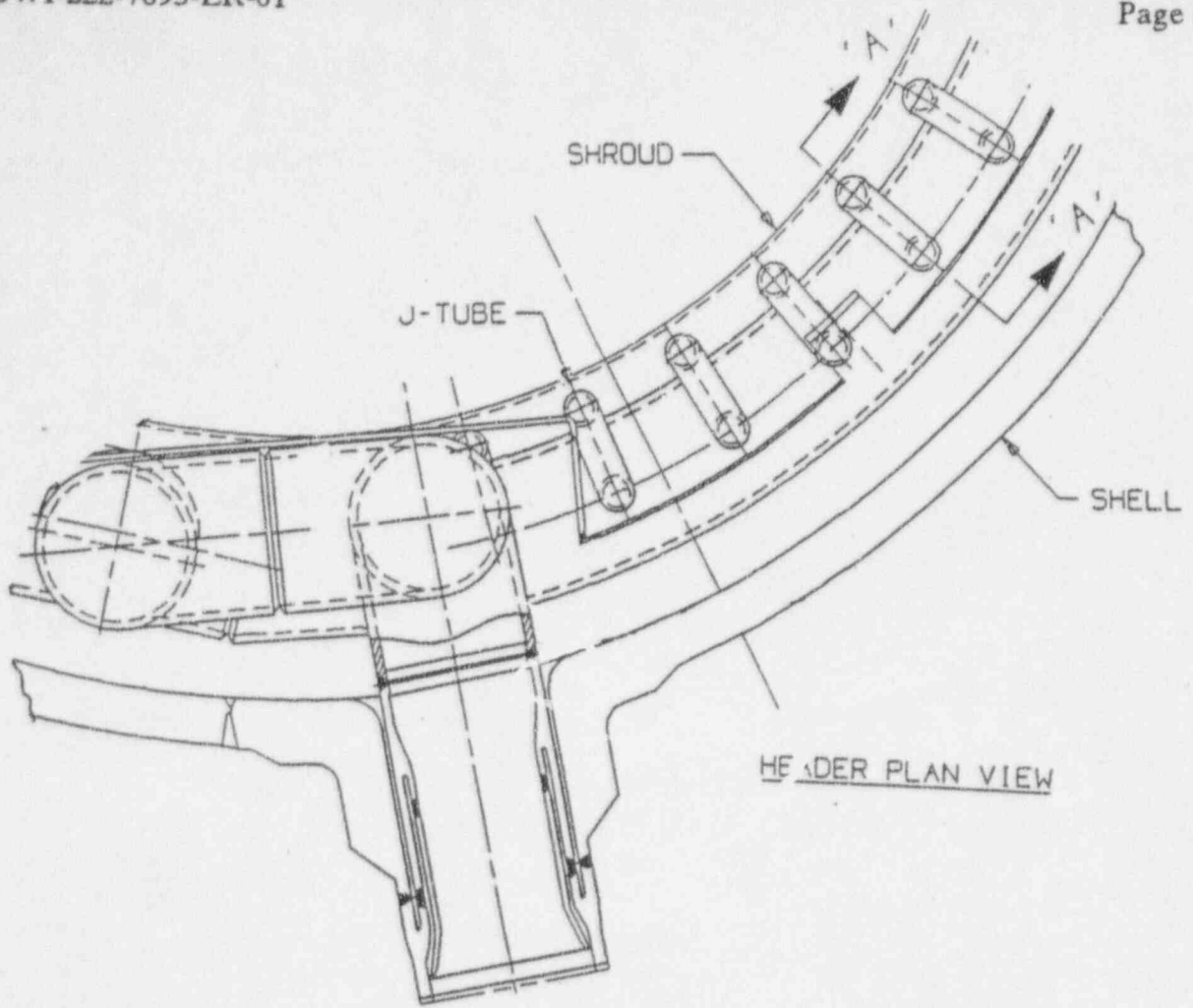
HEADER

'J'-TUBE DESIGN

FIGURE 2.2.5-3
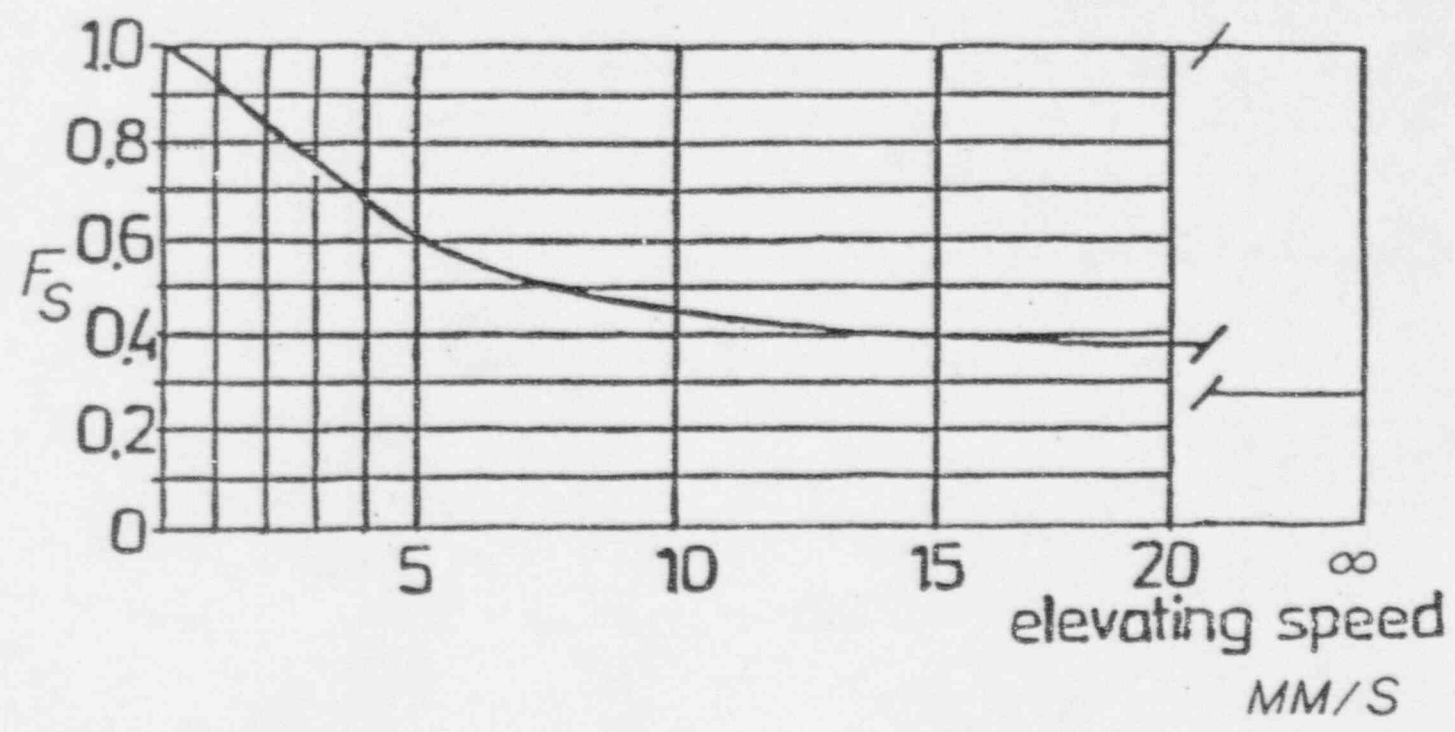
STRESS INTENSITY VERSUS RATE OF LEVEL INCREASE

FIGURE 2.2.5-4

ELEVATION

PLAN
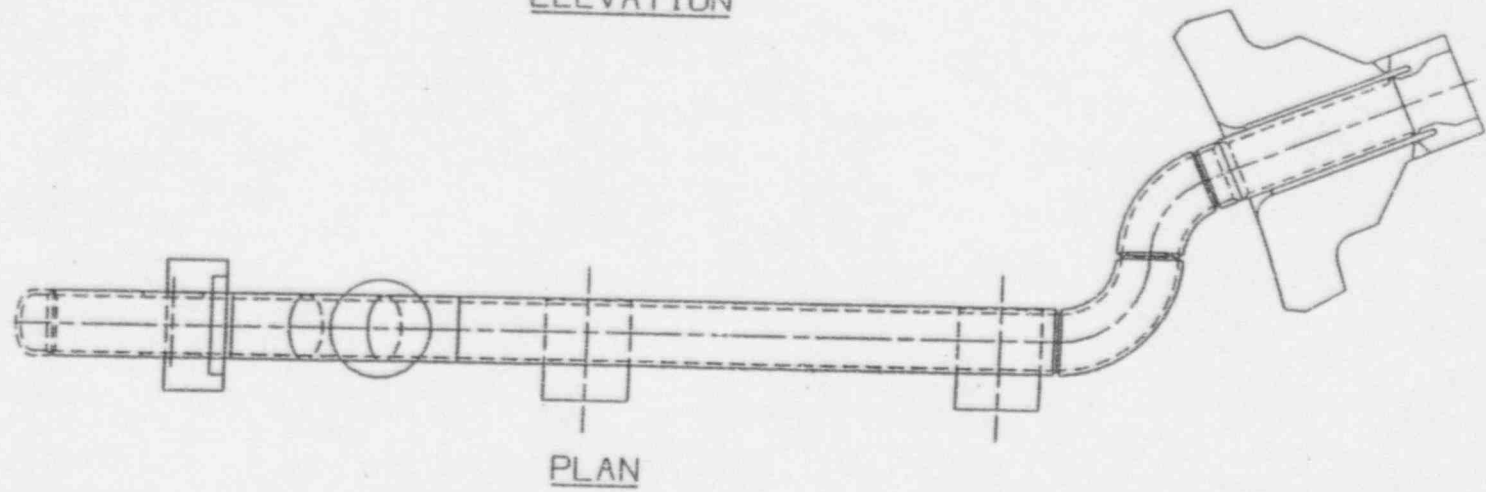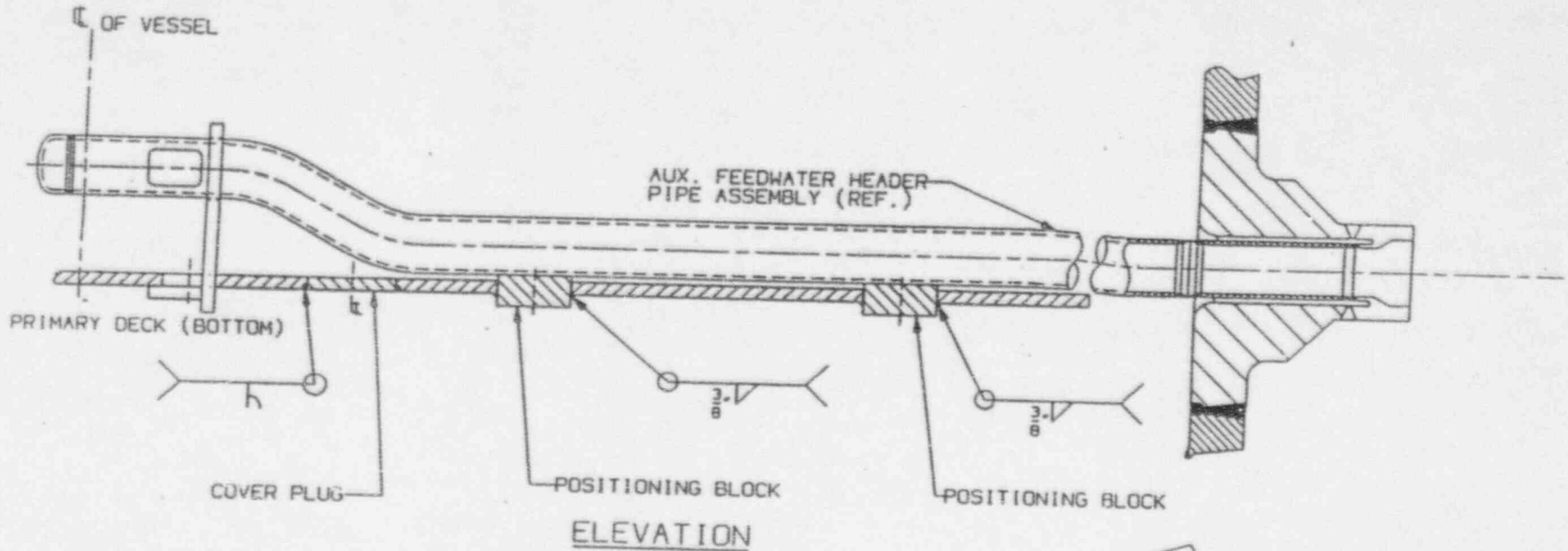
AUXILIARY FEEDWATER NOZZLE & HEADER ASSEMBLY

FIGURE 2.2.5-5

2.2.6   Blowdown System

The RSG's design for blowdown is through tubesheet holes and radial passages drilled in the tube-free lane. These are connected to nozzles at the tubesheet periphery. The RSG blowdown configuration is shown in Figure 2.2.6-1. This arrangement provides the capability for complete RSG drainage and does not obstruct the tubesheet or hinder steam generator recirculation flow. Inspection operations are not hampered by internal blowdown piping or supports. The effectiveness of blowdown is increased by the RSG configuration and by high RSG recirculation and blowdown flow rates.

Corrosion product transport measurements (Reference 1) were made at Point Lepreau Generating Station during steady state operation with blowdown rates of 0.1 to 0.2% (of steam flow by weight). These tests showed that 29% to 45% of incoming iron corrosion products were removed via blowdown. This was at a plant with a high circulation rate and a well designed blowdown system but with a relatively low blowdown rate.

The RSG is designed to have a continuous blowdown rate of at least 1% of full load of steam flow without reducing steaming capacity below the specified value. This rate is nearly an order of magnitude greater than that of the Point Lepreau studies.

Blowdown effectiveness may be enhanced by SG design and by rate of flow. Blowdown enhancement design features are as follows:

1.   Blowdown is at the lowest point in the SG i.e. at the tubesheet level.

2.   Blowdown is via holes drilled down into the tubesheet and connected via radial passages drilled into the tube free lane (TFL) to nozzles at the tubesheet periphery. This provides for the lowest possible takeoff point. It also provides for complete drainage of the SG for maintenance work, etc.

3.   Provision of the above blowdown connections accommodates high rates of blowdown without exceeding erosion limits on the takeoff holes in the tubesheet. Even at a blowdown rate of 3%, the velocity in the two three-inch Schedule 160 blowdown headers will not produce excessive erosion-corrosion that jeopardizes the integrity of the tubesheet.   △4

The design of the blowdown system incorporates features that preclude blockage. These include:

1.   Recommendation that the plant operate using continuous blowdown. This helps prevent sludge build-up on the tubesheet face and over the blowdown holes.

2.   The blowdown holes on the tubesheet face are accessible and can be cleaned.

3.   The system can accommodate nitrogen or water sparging by reverse flow.   △4

Reference for Section 2.2.6

1.   Corrosion Product Transport Studies at Pt. Lepreau, G. Plume, W. Schneider, C. Stauffer, CNS Water Chemistry and Materials Conference, October 1986.
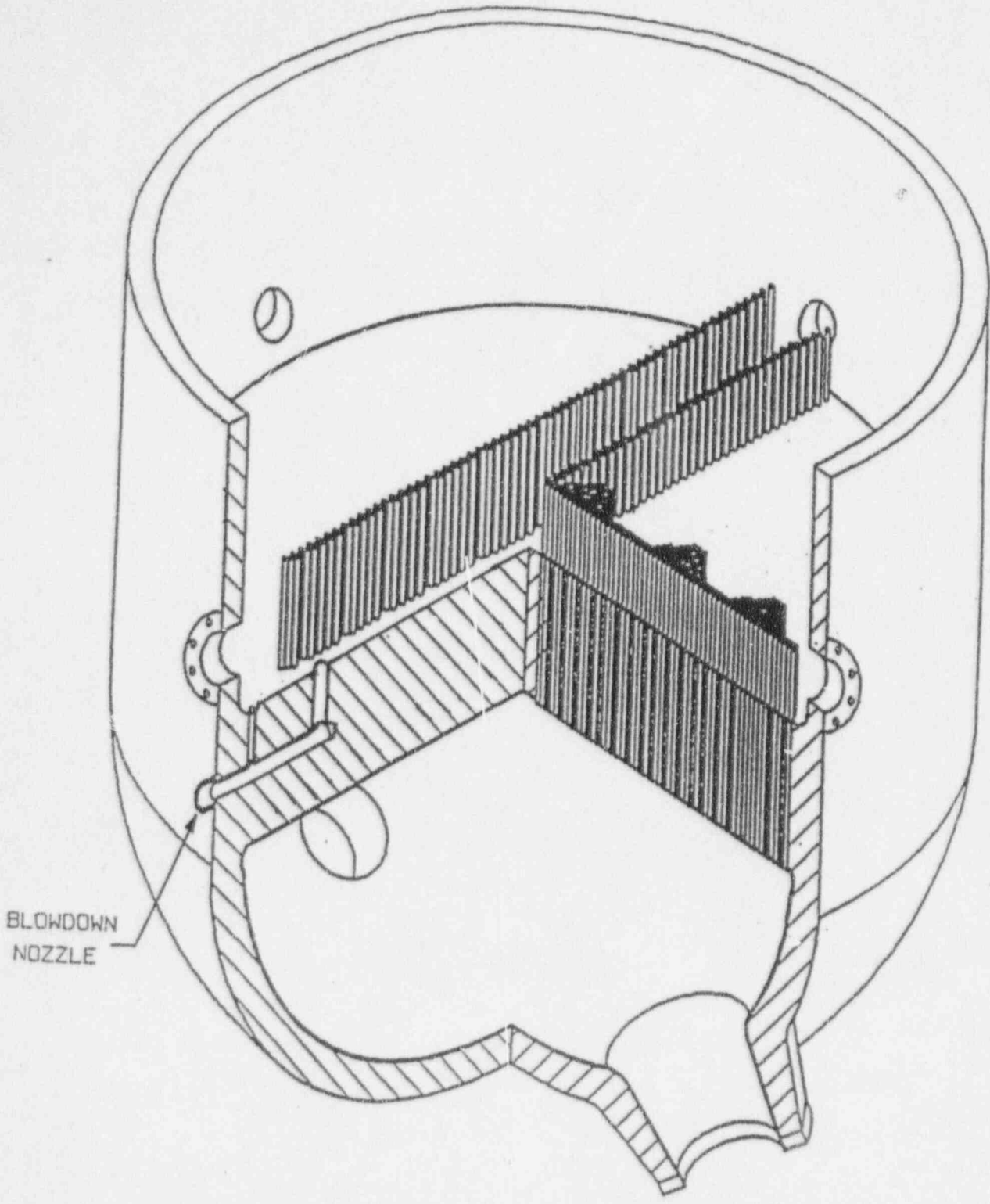
BLOWDOWN
NOZZLE

TUBESHEET BLOWDOWN CONFIGURATION

FIGURE 2.2.6-1

2.2.7   Moisture Separator System

The RSG moisture separators are located in the steam drum. They separate steam from the circulating steam/water mixture from the heat exchanger section. The RSG moisture separator assembly (shown in Figure 2.2.7-1) consists of a "curved-arm" primary stage and a secondary "cyclone" stage. Both are centrifugal type separators and operate with the high steam generator recirculation rates to produce relatively moisture-free steam at the steam generator outlet. Design maximum moisture carryover is 0.25% (by weight), as specified by the contract specifications.

Most of the water is removed from the steam/water mixture in the primary stage, resulting in an exiting quality greater than 90% to the secondary cyclone separators. The remaining water is then removed in the secondary stage. The compact separator design distributes flow more evenly over a larger number of separators, minimizing the potential for overload of any single separator. Their small size allows full-scale testing of a single separator pair, facilitating design optimization and confirmation testing at prototypic conditions.

2.2.7.1   Primary Separators

The primary separator (Figure 2.2.7-2) consists of a riser, four sets of curved-arms, and a return cylinder. The return cylinder extends above the top of the curved-arms where there are several perforations and a retaining lip, which are used to improve the water removal capabilities of the separator at high steam and water flows. The perforations are oriented to accommodate arranging the separators in a square pattern within the steam drum.

The steam/water mixture exiting the tube bundle enters the primary riser at the bottom of the support deck. The mixture enters the curved-arms where a film of water develops on the inner wall of the return cylinder and spirals downward for recirculation. The steam exits the top of the primary separators into the inter-stage region, which distributes the steam prior to the secondary cyclones.

The RSG primary separator provides a steam/water mixture to the secondary separator at greater than 90% quality. Separation of liquid and steam occurs in the region of the curved arms, above the drum water level and minimizes the potential for steam entrainment in the steam drum water inventory. The small inter-stage space and compact separation region allow the primary separators to be positioned relatively high in the steam drum. This allows a higher water level and higher driving head in the downcomer, increases circulation ratio, reduces the chance of feedwater header draining, and reduces the chance of uncovering the tubes. The relatively large flow area through the curved-arms eliminates the need for periodic cleaning. The absence of narrow flow passages which could collect deposits reduces pressure drop and lengthens service life.

2.2.7.2   Secondary Separators

The secondary separator (Figure 2.2.7-2) also operates on the principal of centrifugal separation. The cyclone separator does not have the flow velocity limitations of a scrubber

separator. This allows much higher steam flow per unit area. The steam enters the cyclone through tangential inlet vanes at the bottom of the cyclone which spin the steam. The liquid in the steam is forced to the cyclone wall where it passes through exit vanes and drains back to the main steam drum for recirculation. Flow holes in the top plate of the secondary compartment provide a small steam flow through the secondary skimmer slots, improving separator performance.

The secondary separators are arranged with each separator in its own compartment and with its own drain tube. If unequal separator flow and inlet quality occur, steam exiting each separator is precluded from adversely affecting the performance of the others. This arrangement best matches the BWI separator test configurations. Preliminary moisture carryover information from Millstone 2 RSG has verified the adequacy of this design.

2.2.7.3 Separator Performance

The primary and secondary separator have been extensively evaluated at the B&W Alliance Research Center. The results of these evaluations show the BWI RSG design to produce relatively dry steam over a range of operating conditions.

Steam generator circulation ratio is defined as the mass of mixture entering the steam separators (riser flow) divided by the mass of steam exiting the steam generator (steam flow). High circulation ratio improves water level stability by maintaining a lower void fraction in the steam generator inventory. High circulation ratio also minimizes deposit build-up and tube corrosion. Circulation ratio is increased in the BWI design by raising downcomer head and reducing flow losses. The low primary separator pressure drop increases circulation in the BWI steam generators. Figure 2.2.7-3 shows the relation of circulation flow to steam flow. RSG circulation ratio is highest at low steam flow and decreases as steam flow increases. Riser flow is low at low loads and increases with increasing steam flow, becoming approximately constant between one-third and full load.

Moisture carryover is the amount of liquid exiting the secondary cyclone expressed as a percentage of steam flow by weight. RSG specifications typically require moisture carryover to be less than 0.25%.

Figure 2.2.7-4 illustrates moisture carryover performance versus steam flow for a BWI separator pair at a saturation pressure of 880 psia. The moisture carryover is shown to remain well below 0.25% by weight over the range of tested flows. The following subsections describe the effect of operating pressure, water flow, and water level fluctuations on RSG performance, the effect of steam carryunder, and the design life of the BWI steam separators.

2.2.7.4 Sensitivity to Operating Pressure Fluctuations

Test results show BWI steam separators to be insensitive to operational pressure fluctuations (Reference 1). Figure 2.2.7-5 shows moisture carryover to be insensitive to operating pressure changes for one pair of BWI separators operating at a (high) circulation

ratio of 6.0 for three nominal steam flows.

2.2.7.5 Sensitivity to Water Flow Fluctuations

Test results show BWI steam separators to be insensitive to water flow fluctuations, and therefore insensitive to steam drum flow imbalances (Reference 1). Figure 2.2.7-6 shows that moisture carryover remained insensitive to flow increases up to approximately 160% of full flow.

2.2.7.6 Sensitivity to Water Level Fluctuations

Figure 2.2.7-7 shows moisture carryover versus water level at two different steam flows, and shows the steam separators to be insensitive to changes in water level below the primary separator curved arms (Reference 1). This allows latitude for water level changes.

2.2.7.7 Steam Carryunder

Steam carryunder is steam that becomes entrained in downcomer flow. It is due to ineffective separation of the steam-water mixture from the separators allowing the return of water with entrained steam to drain from the separator back into the downcomer. This reduces the downcomer fluid density by creating steam void which results in a reduced driving head for circulation. It contributes to the swelling potential of the steam drum water inventory and can adversely affect water level control. Carryunder can also increase the downcomer temperature and cause the primary to secondary differential temperature to be reduced. This can impact the steam generator performance. ⚠️4

Under normal operating conditions, the BWI separators have been shown to produce insignificant carryunder. They are highly efficient and configured such that the separation of the steam-water mixture remains above the water level of the steam drum during normal operation. This minimizes the potential for steam entrainment and swelling in the steam drum water inventory. Thus, it improves water level control of the RSG.

2.2.7.8 Separator Design Life

BWI primary and secondary steam separators are designed to last for the life of the steam generator without maintenance or periodic cleaning. The primary separators have large flow passages that preclude plugging even if deposition occurs. The secondary separator inlet body and outlet passages are also large. The relatively small skimmer and vent hole passages of the secondary separators are swept by flow during operation. In the unlikely event of skimmer or vent hole pluggage, they can be cleaned by water lancing. Access is provided from above the secondary separator deck without separator disassembly. Separator design, sizing, material selection and water chemistry control minimize the potential for corrosion to help ensure that the separators will function without maintenance or replacement for the life of the steam generator. ⚠️4

Reference for Section 2.2.7

1.   "High-Efficiency Separator Equipment for Use in Recirculating Steam Generator", M. J. Reed, W. P. Prueter, P. Caple, T. Boyd, ASME Winter Annual Meeting, New Orleans, LA, November 18 - December 3, 1993.

DRY STEAM

CENTRIFUGAL SECONDARY SEPARATORS

PRIMARY SEPARATORS

NORMAL WATER LEVEL

GOOSENECK

FEEDWATER NOZZLE

DOWNCOMER FLOW

FEEDWATER RING

RISER FLOW
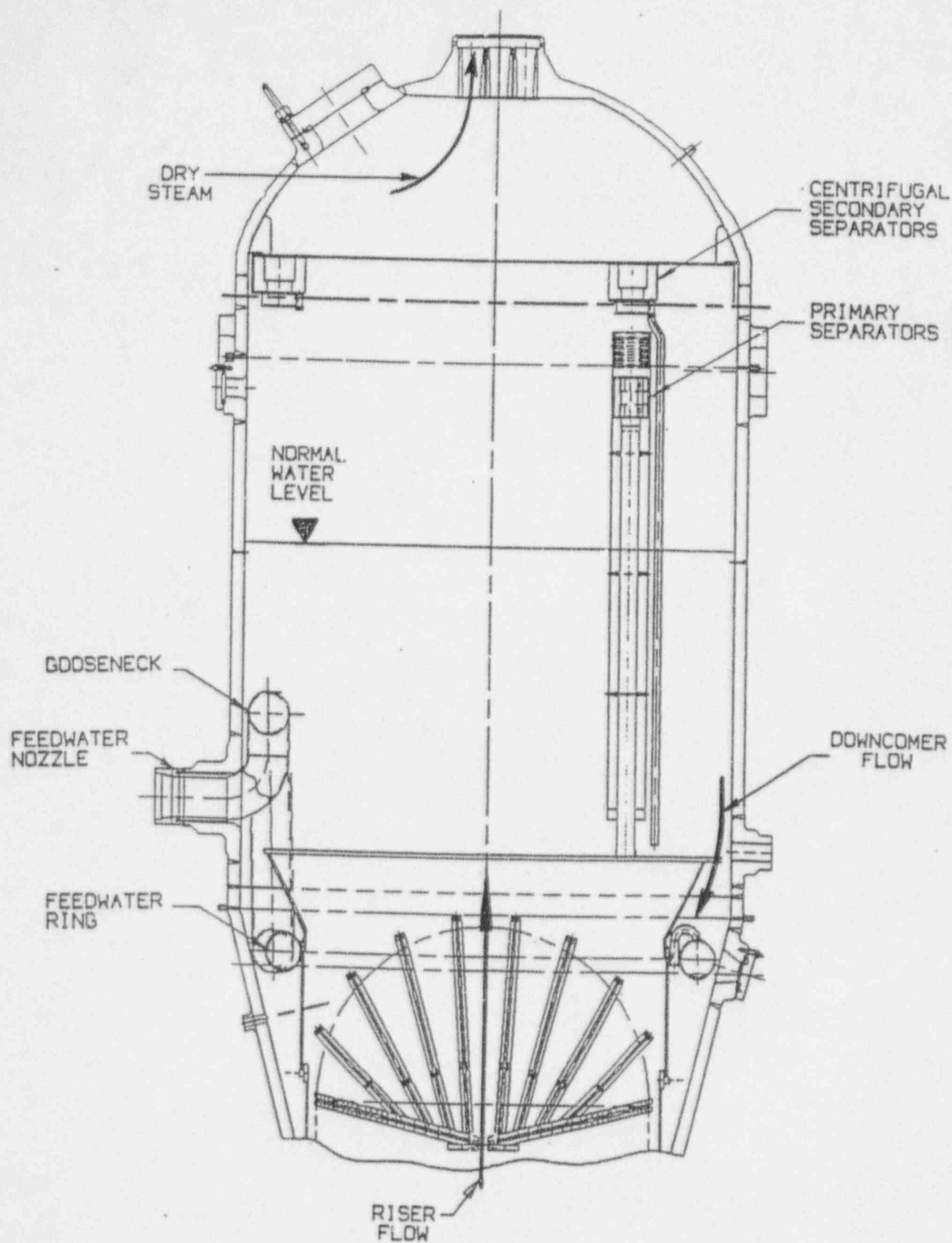
STEAM DRUM ARRANGEMENT

FIGURE 2.2.7-1
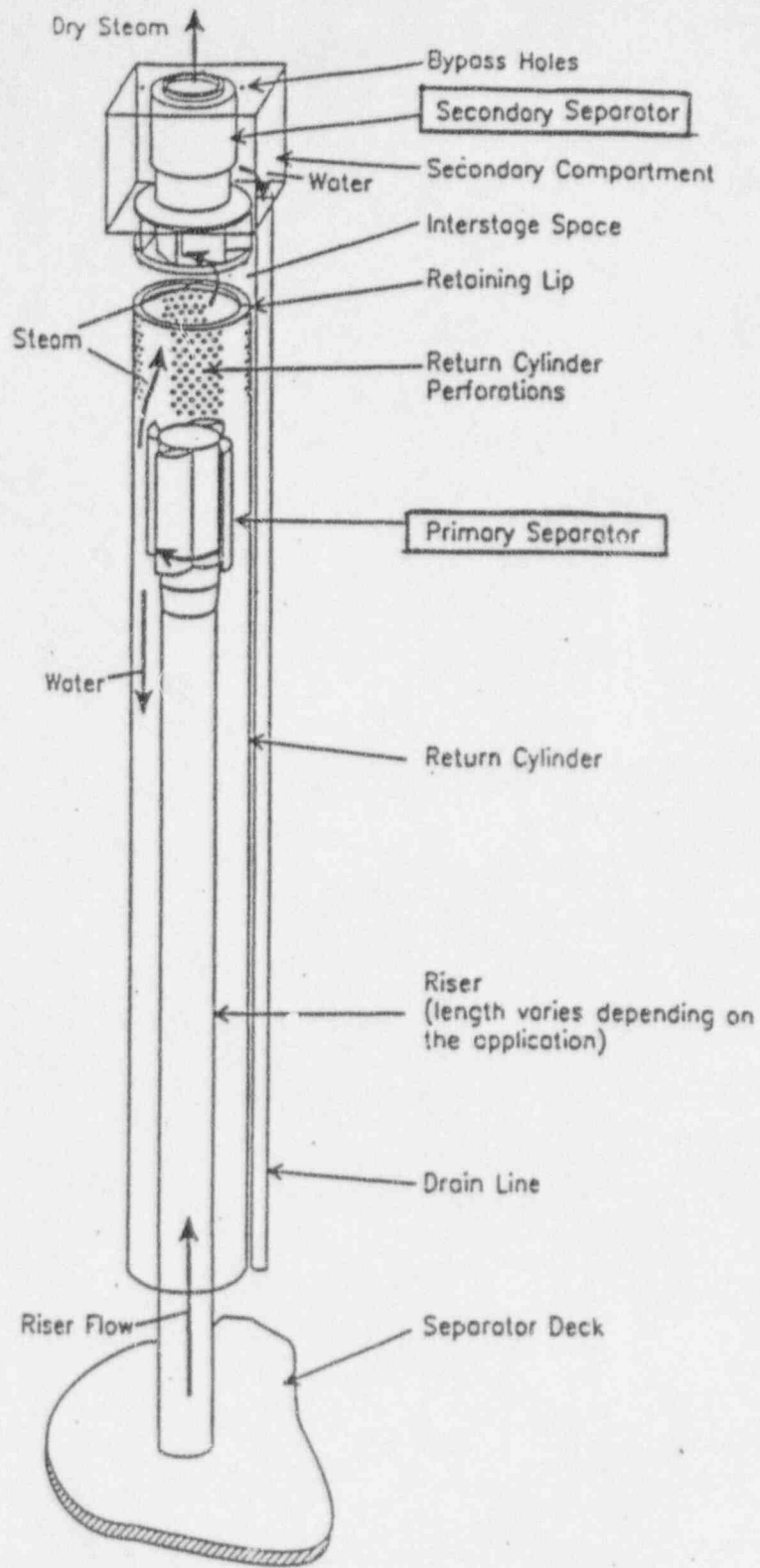
PRIMARY AND SECONDARY SEPARATOR
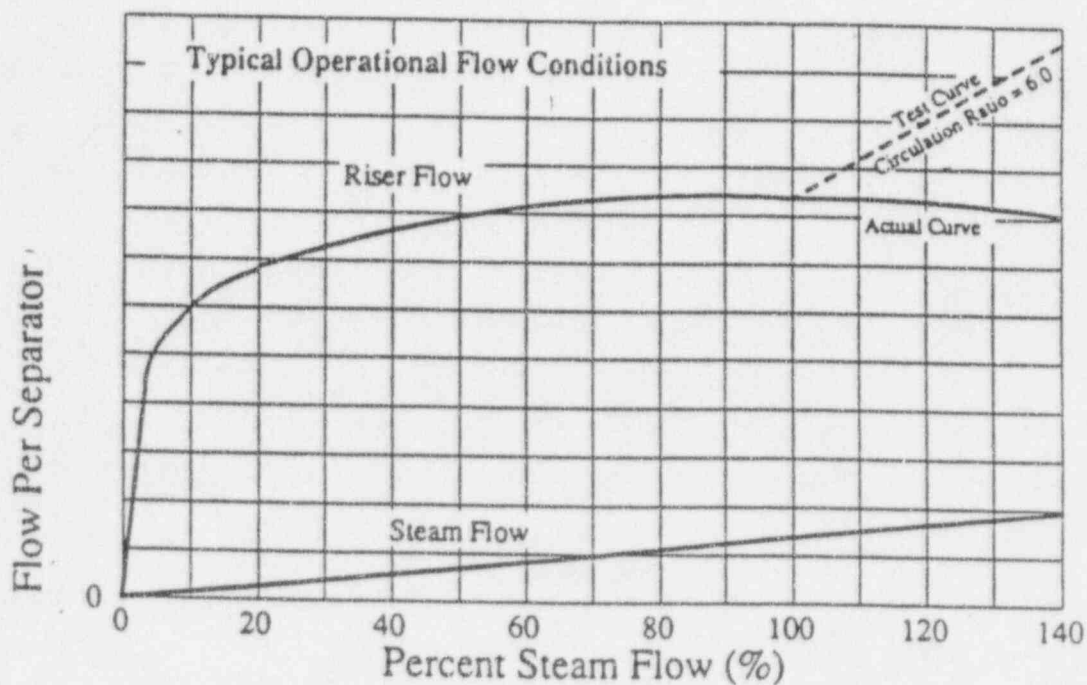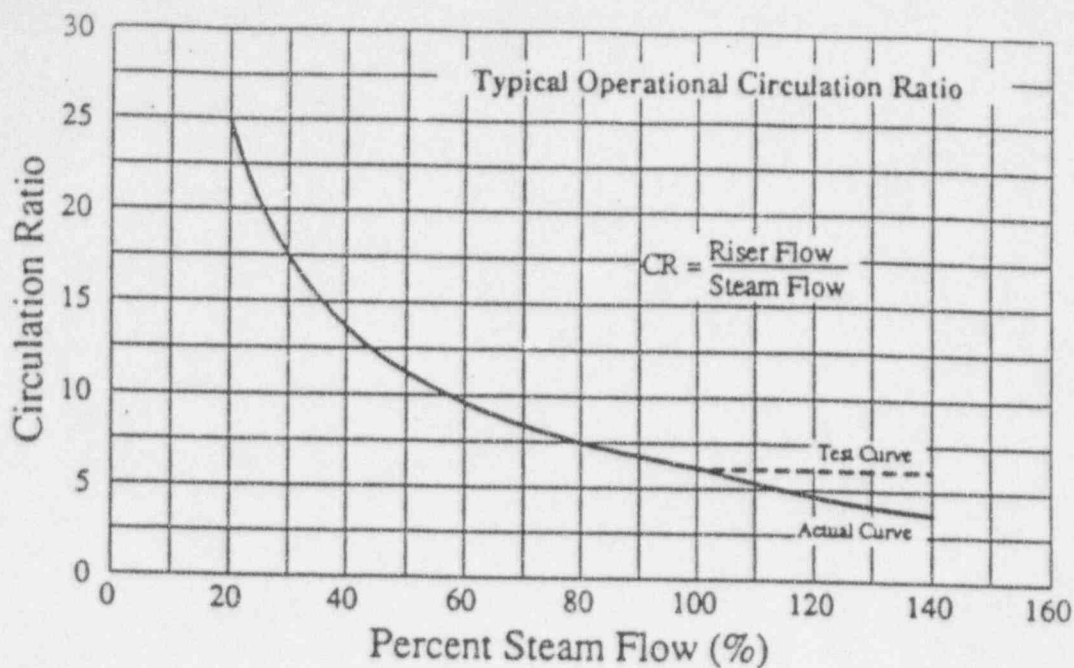
FIGURE 2.2.7-2

TYPICAL OPERATIONAL CIRCULATION RATIO AND FLOW
CONDITIONS

FIGURE 2.2.7-3

BWI SEPARATOR TESTING
CAP-3 & MODULAR SECONDARY SEPARATOR
POWER SERIES TESTING, -20" WL
B&W ARC 1000 PSIA FACILITY, MARCH 1994, 62414
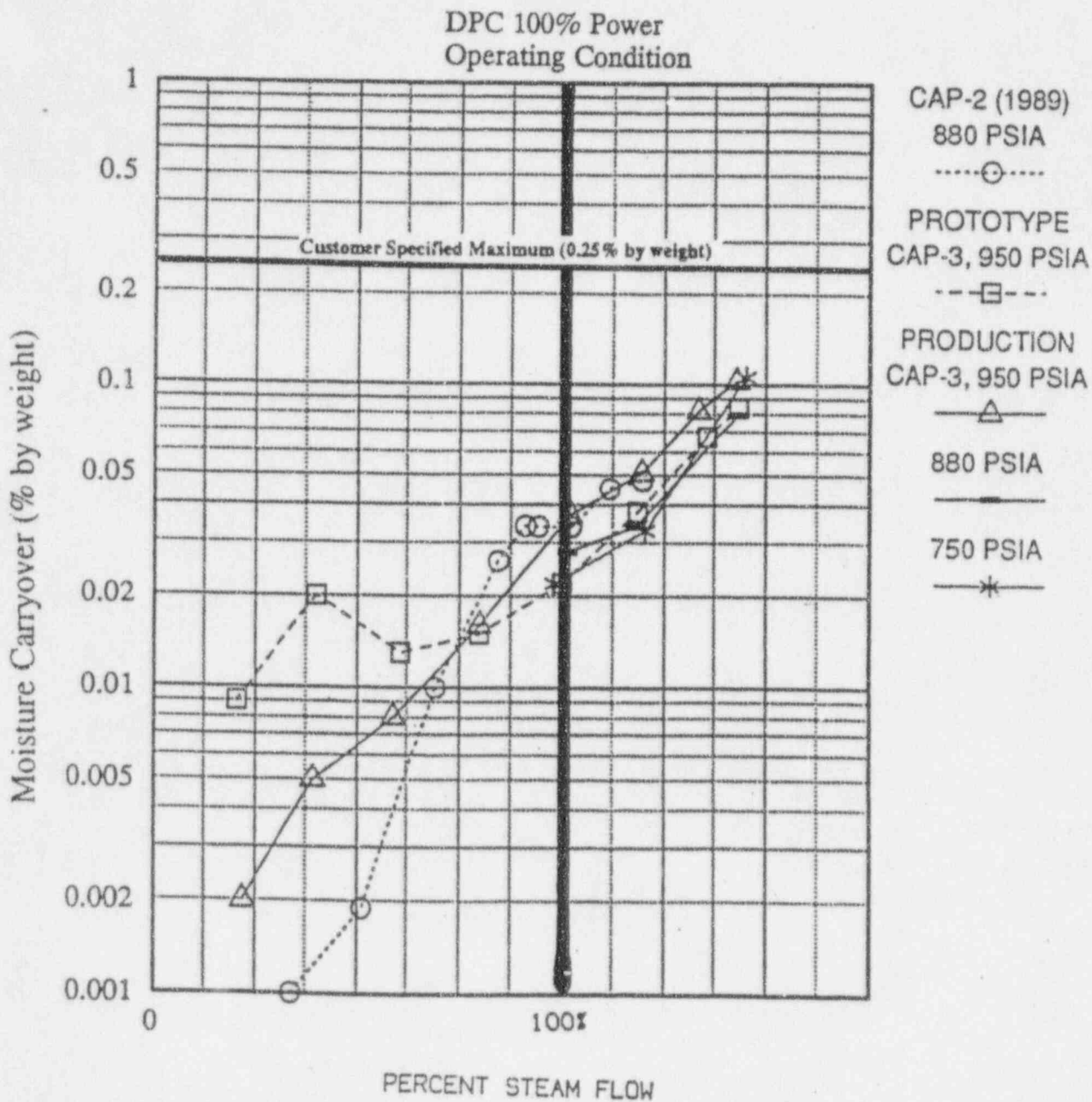
SEPARATOR PERFORMANCE FOR POWER SERIES TESTING

FIGURE 2.2.7-4

B&W Separator Performance
Pressure Sensitivity Test Results
Circulation Ratio = 6.0
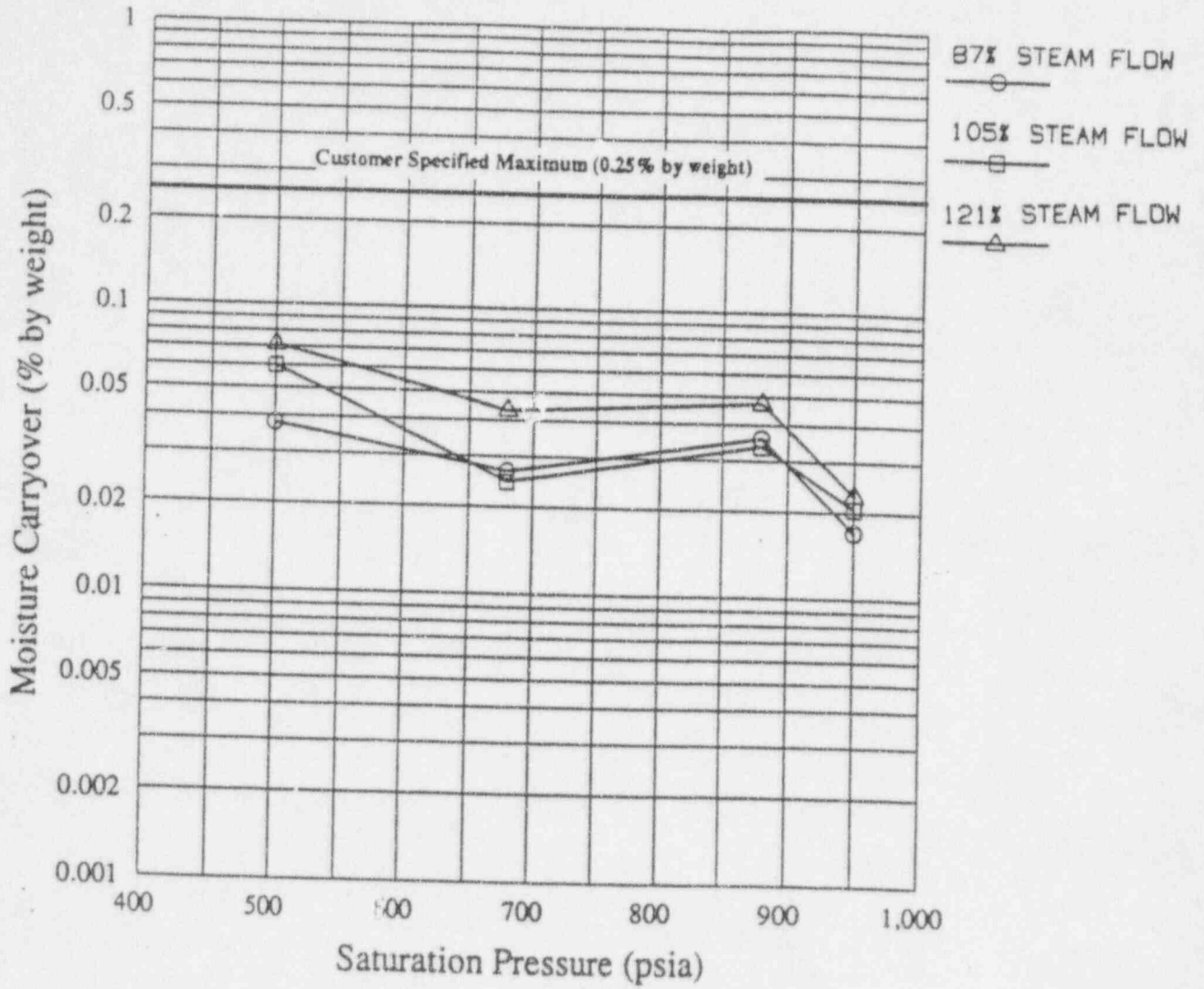
SEPARATOR PERFORMANCE FOR PRESSURE SENSITIVITY TESTING

FIGURE 2.2.7-5

B&W Separator Performance
Water Flow Sensitivity Test Results
Saturation Pressure = 880 psia
Nominal Steam Flow = 41,640 lbm/hr
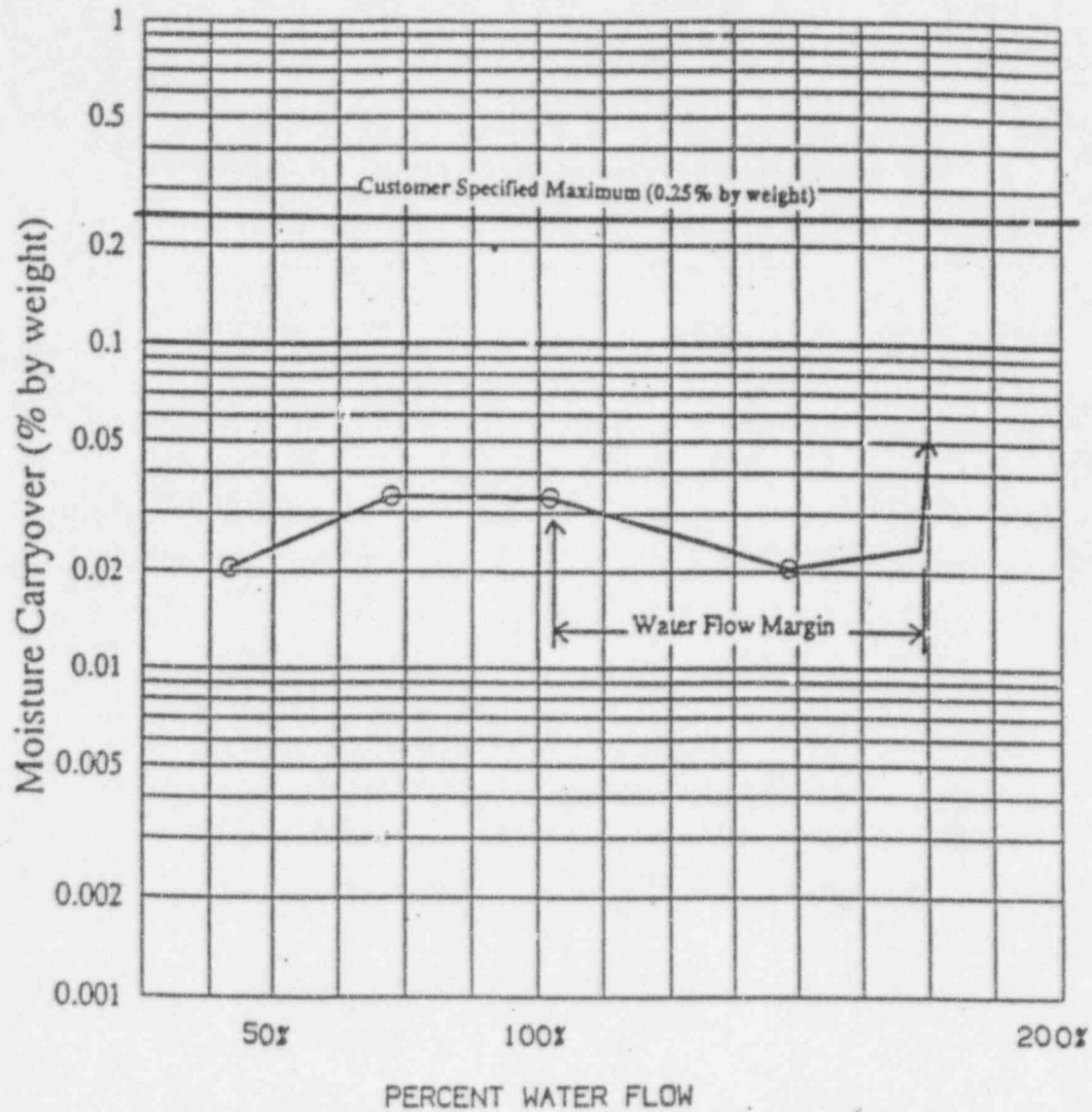
SEPARATOR PERFORMANCE FOR WATER FLOW SENSITIVITY TESTING

FIGURE 2.2.7-6

B&W Separator Performance
Water Level Sensitivity Test Results
Saturation Pressure = 880 psia
Circulation Ratio = 6.0
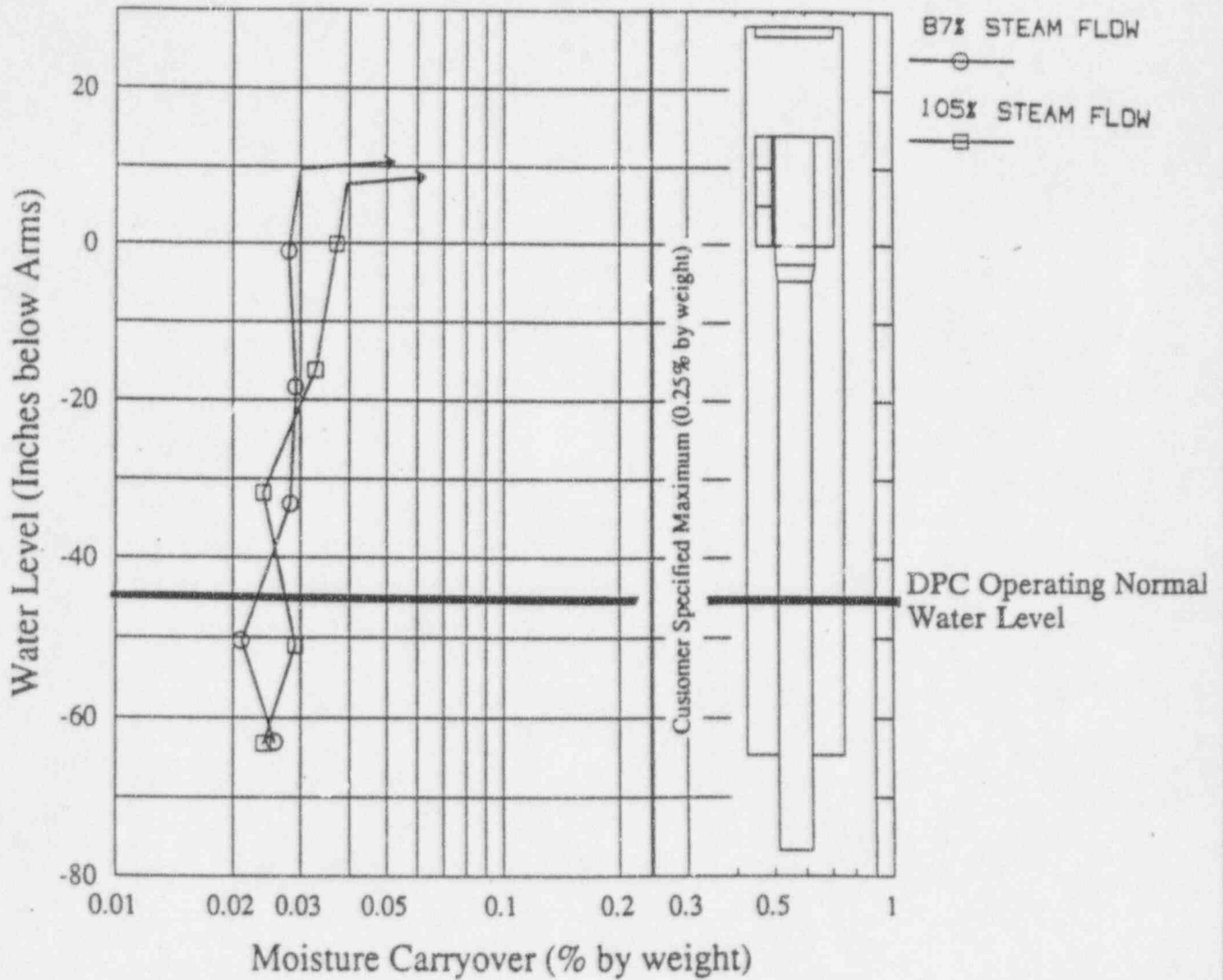
SEPARATOR PERFORMANCE FOR WATER LEVEL SENSITIVITY TESTING

FIGURE 2.2.7-7

2.2.8   Minimized Weld Design

In general, the RSG reduces the amount of ISI required to be performed. One exception is the additional ISI at the nozzle-to-safe end welds. Specifically, the RSG is provided with ⚠4 safe-ends on the feedwater and primary inlet/outlet nozzles to facilitate field fit-up and reduce post weld heat treatment exposure of vessel nozzle welds. The vessel, however, is designed to reduce the number of welds as far as practical and the subsequent ISI as appropriate. Specifically:

1.   The RSG uses a one piece forged steam drum head with an integrally forged nozzle. This eliminates the nozzle-to-head weld used in the OSG and the ISI inspection of this weld.

2.   The RSG conical transition section is a single forging. The OSG conical section is fabricated from three formed plates welded together with longitudinal seams. Thus, three longitudinal welds are eliminated by the RSG. Furthermore, the conical section of the RSG is forged with integral cylindrical sections on each end of the cone. This allows the circumferential cone-to-shell and cone-to-drum welds to be cylinder-to-cylinder configurations rather than cone-to-cylinder configurations as on the OSG. This simpler geometry facilitates the volumetric ISI examinations required by Section XI of the ASME Code.

2.2.9   Integral Flow Restrictor

Certain plant designs include flow restrictors in each main steam line near the steam generator. Their function is to limit steam line break flow for breaks downstream of the flow restrictor and to provide a flow measurement signal. The RSGs include steam flow restrictors as an integral part of the steam generator outlet nozzle to limit steam flow during any steam line break accident, but are not used for flow measurement. The original flow measurement device is retained for that purpose.

If a double ended rupture of the main steam line were to occur, steam flow would become choked at the flow restrictor rather than at the break location. Under this condition, steam flow depends on flow restrictor area rather than break area. Reduction in steam line break flow limits piping loads and energy release rates. BWI designs the flow restrictor to limit internal mass flow rates to four times their normal full load value. The flow restrictor is designed to minimize normal operating pressure drop while fulfilling its flow-limiting function. Steam generator internal pressure differentials during a postulated steam line break are limited to acceptable levels.

2.2.10   Nozzle Dams

The RSG design includes nozzle dams which isolate the RSGs from the hot and cold leg piping to permit maintenance and inspection within the primary head during refuelling operations. The nozzle dams are held in place by locking devices that engage the nozzle dam retention ring. Figure 2.2.10-1 shows RSG nozzle dam details.

The RSG nozzle dams are similar to those in use on the OSG. The nozzle dams are designed so that primary seal failure does not result in catastrophic dam failure. They are designed and fabricated according to the requirements of an owner approved nozzle dam technical specification which provides details of the dams' design, service and test loadings as well as requirements for materials, fabrication, Q.A., cleanliness, and packaging. Although the nozzle dams and rings are not ASME Section III components, their design generally follows Code design philosophy. The nozzle dam vendor provides a design report that details nozzle dam structural design calculations.

The nozzle dam rings are manufactured from Alloy 690 material in accordance with ASME SB-166. They are welded to the head cladding surface concentric with the nozzle by full penetration welds that comply with the requirements of the ASME Code for Section III Class 1 vessel attachment welds. A structural analysis of the weld is performed according to Section III and included in the RSG Code stress report.

DAM SEGMENT (3)

RETENTION RING

LOCK PIN (8)

PASSIVE SEAL

PNEUMATIC DRY SEAL

LEAK CHECK ANNULUS

PNEUMATIC WET SEAL

DIAPHRAGM
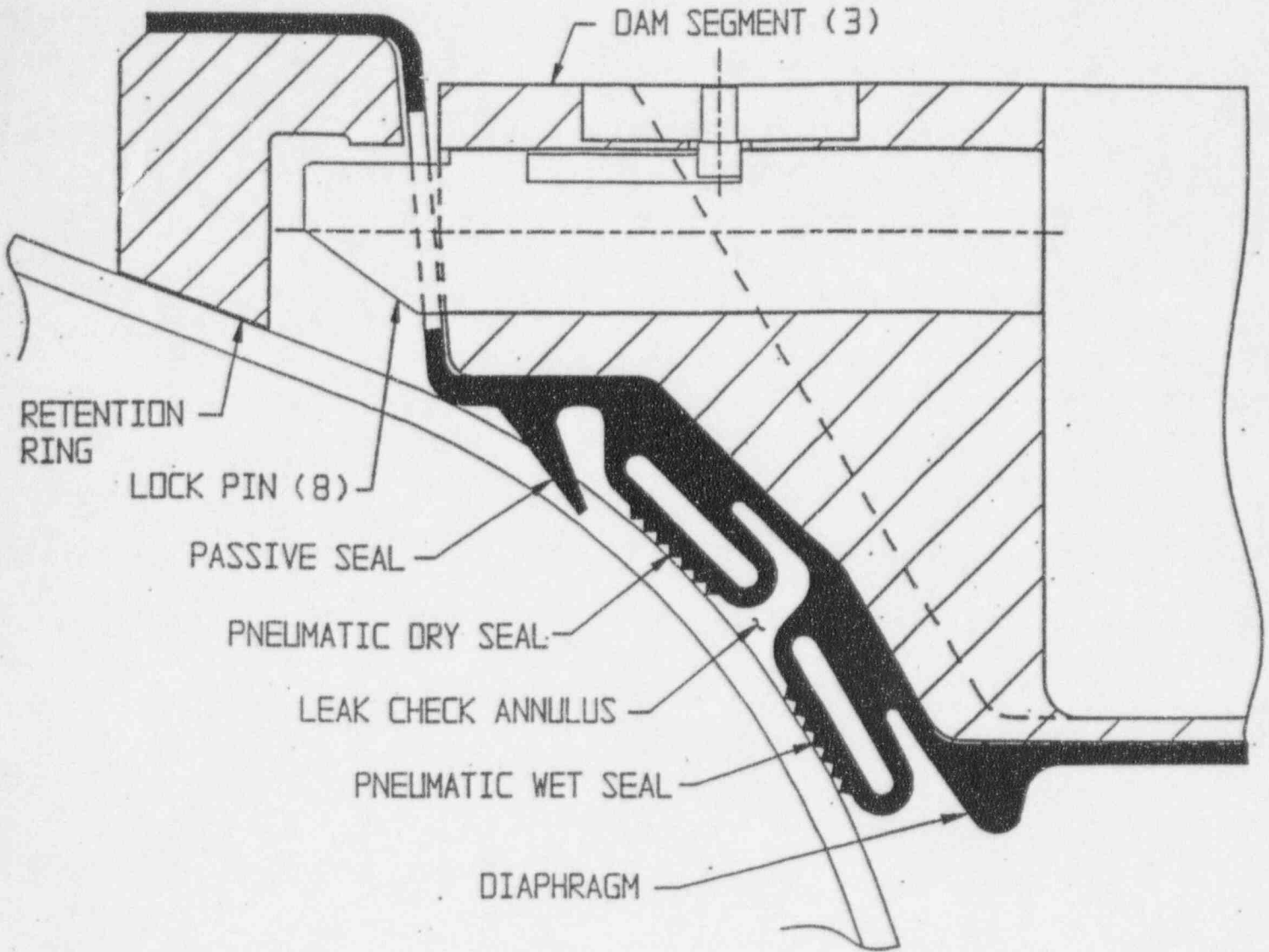
RSG NOZZLE DAM DETAILS

FIGURE 2.2.10-1

2.2.11 Provisions for Inspection

Adequate provisions for RSG inspection for maintenance and repair are provided. These include primary side manways, secondary side manways, hand holes, inspection ports, internal access hatches and ladders. The number and location of primary manways, secondary manways, hand holes and inspection ports are described below and shown on Figures 2.2.1-5 and 6. Internal RSG access is provided by access hatches in the primary and secondary separator decks and access ladders.

Twenty-one inch manways are provided in the hot and cold leg channel heads to provide access to the primary side of the tubesheet. Covers for these manways use a stud design compatible with efficient simultaneous multi-stud hydraulic tensioners, enhancing accessibility to the channel heads. One manway is provided on the secondary side in the steam drum head to provide access to the RSG internals.

The channel head manway covers are provided with a manipulator to ease the handling of covers after stud de-tensioning.

A total f ten(10) six inch handholes provide access to the secondary side of the RSG. Eight c. these are located in the shell between the tubesheet and the lowest lattice grid provid ng access to the ends of the tube free lane, at 90° to the tube free lane and at four other circumferential locations. The internal shroud separating the downcomer and riser sectic is terminates above the secondary face of the tube sheet, and the blowdown system is completely contained within the tube sheet thickness. This provides unobstructed access to the entire tube bundle at the tube sheet elevation and eases inspection and maintenance. One(1) handhole provides access to the steam drum and is located at the approximate elevation of the lifting trunnions. One(1) handhole is located in the conical shell section and provides access to the feedwater header in the vicinity of the header split location.

Access to the tube bundle is provided by a total of twelve(12) two inch inspection ports located at the ends of the tube free lane at the intermediate tube support grids. There are no internal obstructions such as shroud feet, tie rods, lane blockers or blowdown header. This maximizes access for inspection and maintenance.

The two inch inspection ports and hand holes allow inspection of the lattices. These access ports allow fiber-optic lane-by-lane inspection and direct lane-by-lane water lancing of supports. Fiber-optic and water lancing equipment reaches the tubes through the inspection holes at the ends of the tube free lane. Since there are no tie rods or other obstructions within the bundle, one 90° tool can access the intertube lanes on the hot and cold leg sides of the lattice supports.

The internal steam separating subassembly is an all welded design constructed with permanent manway tunnels to the top of the tube bundle and to the steam drum head. Internal hinged doors are secured by a captured, bolted clasp arrangement during generator operation. Steam separation equipment removal is not required for access.

The RSG is designed to accommodate accessibility for ASME Code ISI. Lugs and insulation brackets are arranged to avoid interference with pre-service and in-service inspection and ultrasonic scanning.

2.2.12 Electro-chemical Polishing

A major source of radiation exposure during inspection and service of nuclear steam generators is caused by radioactive nuclides on the internal primary side surfaces. Particles of radioactive material are incorporated into the metal surface oxide films. They are also trapped in the asperities of the metal surfaces. RSG contamination build-up is reduced by polishing the channel head surfaces to a very fine finish. A chemical process, termed electro-chemical polishing, or electropolishing, produces micro-smooth channel head surfaces.

Tests have proven that this is an effective way to reduce occupational exposure (Reference 1). In these tests, manway seal plates showed a contact dose rate reduction by a factor of 7 when the surfaces were electro-polished prior to insertion into the primary system. French steam generator channel heads are routinely electro-polished prior to being put into service as new or replacement units. In France all new or replacement steam generators have electropolished channel head surfaces. In the United States, Palisades, Millstone 2 and other RSGs have electropolished channel head surfaces.

A fine surface is first produced by mechanical polishing. Electro-chemical polishing is then performed to reduce surface "micro-roughness". This process reduces surface asperities by applying an electric current to the metal surface through an electrolyte. Since the surfaces are finished to 63RA (roughness average in micro-inches) by mechanical polishing before electropolishing, the electropolishing does not appreciably alter surface roughness. It does, however, reduce microscopic roughness. The process removes very little material, typically 0.001 to 0.003 inch. The surface layers of metal are removed in a predictable and controllable manner. Electropolished surfaces contain less total surface area, and less tendency to accumulate contamination.

The entire surface of the divider plate and primary head including nozzles and manways are electropolished. Because of the complex geometry, the exposed tubesheet clad face and tube welds are not electropolished. Particular attention is given to manway cladding to minimize personnel exposure. Qualification of the electropolishing process complies with EPRI-NP-6617 and 6618. This is further discussed in Section 3.2.4.3.

Because of the facilities required, electropolishing is applied to the RSGs after final assembly and hydrostatic testing have been completed.

Electropolishing is performed on the steam generator primary side surfaces by a qualified subcontractor to qualified procedures. The electropolishing procedures are qualified to verify and establish the parameters that are used for electropolishing the heads. This includes metallographic evaluations to establish optimum application parameters and to check for any deleterious effects such as intergranular attack, dendrite attack, surface cracks,

and degree of metal loss.

References for Section 2.2.12

1.   EPRI TR-100059 Volume 1 and 2, "Effect of Surface Treatments on Radiation in Steam Generators", Project 2758-02, Final Report, November, 1991.

2.2.13   Provisions for ALARA

The BWI RSGs are designed to reduce radiation exposure to operations and maintenance personnel "As Low As Reasonably Achievable" (ALARA). Measures employed to minimize radiation exposure include selection of materials and design to minimize the number of welds that must be periodically inspected, water chemistry that minimizes sludge accumulation, sufficient hand holes and manways to permit personnel access for inspection, maintenance and repair, and provision for substantial drainage of the channel head. Remote maintenance equipment and procedures can also minimize personnel entry time. Examples of measures employed to minimize personnel exposure and off-site releases are described in the following sections.

2.2.13.1   Material Selection and Design to Minimize Personnel Exposure

The BWI steam generators have been designed to incorporate many features to enhance access and maintainability. Steam generator cladding and tube materials are selected to minimize cobalt content. Detailed information on steam generator materials is presented in Section 2.3. Manway and manway closure design, a steam drum manway tunnel, hand holes, inspection ports, an enhanced blowdown arrangement, and internal shroud support design significantly contribute to ease of access. Efficient design of welded seams, electro-polished channel head, provision for nozzle dam support, flush tube-to-tube sheet welds, in-head tube identification system, welded construction of all internals including locking devices of threaded fasteners, access for sludge lancing and removal, and design compatibility for chemical cleaning contribute to the efficiency of maintenance.

The enhanced blowdown system is a drilled hole configuration at the tube free lane located within the tube sheet thickness - provides the required blowdown, 100% drainage of the secondary side without additional drain connections, and unobstructed access along the tube free lane on the secondary side of the tube sheet.

The removal of sludge from the secondary side of the steam generator is a significant outage activity that can be minimized by careful management of secondary side water chemistry. The RSGs have features to facilitate access and this maintenance activity:

1.   Access to the secondary face of the tube sheet is completely unobstructed from the face to the first grid support by the design decisions to incorporate the blowdown system completely within the tube sheet thickness; to support the shroud with attachments to the lower shell course; and to support the tube grids solely from the shroud. These design features eliminate obstructions such as blowdown headers, shroud support legs, tube support tie rods, and lane blocker devices which would interfere with effective sludge removal.

2.   The BWI enhanced blowdown arrangement permits drainage of all sludge-laden water.

3.   A total of eight hand holes are conveniently located at the secondary side tube sheet face, four of which are on the axes of the RSG.

2.2.13.2   Minimization of Inspected Welds

The RSG design reduces the number of welds that require inservice inspection. This reduces personnel radiation exposure and required inspection time. Examples of design to reduce inspected welds are addressed in Section 2.2.8.

2.2.13.3   Water Chemistry

Primary system water chemistry requirements minimize corrosion products and formation of activated material. Water chemistry is discussed in detail in Section 2.7.

2.2.13.4   Minimization of Personnel Exposure

Reduction of personnel entry time required for steam generator inspection and maintenance activities reduces personnel exposure. Improved access and facilitation of the actions required are incorporated into the RSG design. These include:

1.   Nozzle dam retaining rings have been incorporated into the channel head designs. These contribute to plant outage efficiency by simplifying nozzle dam installation and permit channel head access for steam generator inspection and maintenance activities concurrent with other outage activities. A nozzle dam can be installed manually in 1 to 2 minutes. Nozzle dams are described further in Section 2.2.10.

2.   Large manways are provided with a stud design compatible with simultaneous multi-stud hydraulic tensioning. This enhances accessibility to the channel heads and steam drums.

3.   The design of external manways incorporates weldable seal diaphragms as a back-up to the gasket seal. This permits a readily available alternative sealing method in the unlikely event of damage to gasket sealing surfaces.

4.   The channel head manway covers are provided with a manipulator/hinge which is mounted to the head and capable of being readily detached from the manway. This facilitates handling of manway cover for removal and installation. Internal steam drum manway covers are provided with a hinge assembly.

5.   The tube-to-tube sheet welds are of the "flush" design without reduced tube diameter. This enhances visibility and access to tube ends.

6.   The tube sheet face is permanently stamped with multiple tube identification symbols. This enhances manual location of tubes and calibration and

verification of remotely positioned tooling.

7.   Hand holes and inspection ports are provided to ease inspection and maintenance. Their number and location are described in Section 2.2.11 and shown in Figure 2.2.1-5.

8.   The internal shroud separating the downcomer from the riser boiler sections terminates above the secondary face of the tube sheet. It does not interfere with tube inspection or maintenance.

9.   The internal steam separating subassembly is an all welded design with permanent manway tunnels and ladders to the top of the tube bundle and to the steam drum head. This simplifies access for inspection and maintenance. Removal of steam separating equipment is not required for access to the tube bundle.

10.   Electro-polishing steam generator channel head surfaces reduces occupational radiation exposure. Section 2.2.12 describes the electropolishing process to be employed on the RSGs.

2.2.14   Provisions for Chemical Cleaning

Tube deposits degrade heat transfer and, may promote tube corrosion. Chemical cleaning techniques are effective at removing tube deposits. The RSGs are capable of being chemically cleaned to remove deposits that cannot be removed by sludge lancing.

The RSGs have sufficient corrosion allowance to accommodate six chemical cleanings using the Electric Power Research Institute (EPRI) iron-copper solvent. All RSG materials are qualified for this process (see Section 2.6.5). These materials include the vessel shell, the internals, the tubes and the lattice supports. BWI guidance ensures chemical cleaning in accordance with component requirements and EPRI Steam Generator Owners Group (SGOG) guidelines.

Provisions for chemical cleaning include compatibility of the generator design with the cleaning process and appropriate access. The RSG design allows full solvent access, free drainage and free venting of all surfaces. Access is provided by hand holes and inspection ports for fill and drain connections, monitoring of metal-removal, and inspection. RSG connections available for chemical cleaning are as follows:

1.   Eight six-inch tubesheet hand holes for fill/drain connections.

2.   Tubesheet blowdown holes (or tubesheet level suction line via above hand hole) for final drain-down.

3.   Two six-inch tubesheet hand holes for corrosion monitoring system probe positioning and inspection.

4.   Upper and lower connection points for level measurement.

5.   Upper level connection points for venting of nitrogen blanket.

2.2.15   Water Level Stability and Control

<u>Circulation Stability</u>

Instability (of water level, circulation flows, etc.) occurs for flow systems with a high degree of two-phase losses. Resistances in the circulation loop that have high steam quality associated with them (e.g., upper support plates, U-bend region, steam separators) are the most sensitive to changes in pressure, by virtue of the significant effect of quality on flow resistance. A change in pressure results in some change of flow resistance, and hence circulation ratio, water level, etc. The degree of this effect, and whether oscillations amplify, depend upon many factors. The most notable is the amount of stabilizing single-phase loss in the circuit, as this type of loss does not vary directly with changes in pressure.

Single-phase losses in the circulation loop include the downcomer entrance loss at the primary deck, the loss associated with flow around the feedwater header and cone region, friction loss in the downcomer, and tube bundle entrance loss. For RSG designs, the downcomer friction and tube bundle entrance loss are the most significant.

Based on experience at B&W's Nuclear Equipment Division in Barberton, Ohio, a conservative circulation stability rule was developed, namely

$$\frac{\text{single phase losses}}{\text{two phase losses}} > 0.20$$

This rule was derived in the wake of field observations of instability. Field observations of instability in older equipment were corrected by adding downcomer resistance.

<u>Stability RSG Design</u>

RSG design for circulation stability proceeds as follows:

1.   The downcomer annulus is sized to be as small as possible to maximize bundle size, but not to create unacceptable flow velocities, unacceptably low circulation ratios, or difficulties in shop assembly.

2.   All flow loss coefficients are calculated and put into the B&W design program CIRC.

3.   The single-phase/two-phase loss ratio is computed by CIRC. The stability ratio, based on the design procedure described above, demonstrates that no downcomer orifice is required for this design.

There is no doubt that over decades of operation, fouling and deposits in the SG can change flow patterns and ultimately steam generator circulation stability. A case in point is Bruce Nuclear Station A, Unit 2, Boiler 3.

Significant oscillations in water level were first observed at Bruce A for power levels above 95% in October, 1986. By November, 1988, the threshold power for observations of the oscillation was 78%. Various analyses, including simulation and site measurement confirmed that the problem was caused by excessively high blockage of the uppermost broached plates (75% for the top broached plate). The problem was eventually solved by water lancing the uppermost three broached plates.

There are two points to this example. First, the kind of blockage experienced at Bruce A is one of the main reasons that BWI has chosen the lattice grid support structure for the RSGs. BWI experience with lattice grid supports (at Pickering), and the experience of other manufacturers (C-E and KWU) have proven the superiority of the lattice grid design due to its open flow area. Second, BWI's model of the Bruce design simulating the pluggage prior to water lancing yields a ratio of single-phase to two-phase flow of less than 0.05 at the threshold of instability. After waterlancing, which reduced the blockage to 55%, the unit operated stably with a calculated stability ratio of 0.1. This confirms that the design rule of single-phase losses to two-phase losses ≥ 0.2 is conservative.

The factors that contribute to potential water level instability in the steam generator are identified and understood. Provisions are made in the RSG design to maximize water level stability. Operation and consequent deposit build-up can alter flow patterns and decrease water level stability. Operating experience has shown the superiority of the lattice grid design in resisting problems that lead to water level instability. Therefore, RSG water level will be stable, and will remain stable during plant operation.

2.3   STEAM GENERATOR MATERIALS

This section discusses RSG pressure boundary, critical-to-function, and steam generator internals materials. These categories are defined below, and the materials in each category are described. RSG materials are listed in Table 2.3-5.

2.3.1   Pressure Boundary Materials

RSG pressure boundary materials consist of ferritic steels, either carbon steel or low alloy steel, and weld material to join them. The chemical analysis and mechanical properties of these materials are contained in Tables 2.3-1 and 2.3-2. Low alloy steels such as SA-508 Class 3 and SA-533 Type B Class 1 are supplied in the quenched and tempered condition, and are qualified on the basis of their mechanical properties after a simulated post-weld heat treatment (PWHT). This simulated PWHT conservatively bounds all anticipated PWHTs and complies with the requirements of the Certified Design Specification. ⚠️

Welding materials are selected on the basis of their mechanical properties after a simulated PWHT, and exhibit equivalent or higher strength levels than the base metals they join.

Strength and toughness are the critical selection criteria for all pressure boundary materials. Materials comply with the ASME Boiler & Pressure Vessel Code Section II and Section III. Toughness prevents brittle fracture during high load conditions, such as hydrostatic testing and upset conditions. RSG material toughness is specified by a reference temperature RT NDT (reference temperature for nil-ductility transition temperature) defined by ASME III, Subsection NB-2300. RT NDTs depend on impurity levels, such as sulphur, material fabrication, such as rolling or forging, and heat treatment, including the simulated PWHT. The RT NDT for each RSG pressure boundary plate, forging or weld is equal to or less than 0°F. Typically, these temperatures range from -70°F to -20°F.

Materials for bolting applications are ASME Section II SA-193 Gr. B7 for bolts and studs, and SA-194 Gr. 7 for nuts. These materials are supplied in the quenched and tempered condition and have adequate strength and toughness for their application.

2.3.2   Critical-to-Function Materials

The term "Critical-to-Function material" is applied to materials used in components essential to preserve RSG internal structural integrity or emergency heat removal capability. These components include steam generator tubes, tube supports, cladding and feedwater headers. Chemical analysis requirements and mechanical properties of materials used in these components are provided in Tables 2.3-3 and 2.3-4 respectively.

2.3.2.1   RSG Tube Material

RSG tube material is a nickel-chromium-iron alloy, ASME Section II SB-163, Code Case N-20-3, Alloy 690, that exhibits high resistance to corrosion and stress corrosion cracking in primary and secondary side environments. Code Case N-20-3 permits the use of Alloy

690 in the construction of Class I components in accordance with Section III, Division I of the ASME Code and gives requirements for strength and design stress intensities to meet Code requirements.

The chemical composition and mechanical properties of Alloy 690 are shown in Tables 2.3-3 and 2.3-4. Corrosion resistance is derived primarily from the higher chromium content and heat treatment that produce a corrosion-resistant microstructure.

The cobalt content of the RSG tubing and cladding material is a major contributor to the radioactivity level in the primary head. In accordance with ALARA guidelines, cobalt content is limited in the tubing material specification to an average of 0.014% with a maximum of 0.016%.

The tubes are procured from a BWI approved supplier who must demonstrate by manufacture and examination of pre-production tube samples the ability to manufacture tubing to BWI's requirements. These requirements are delineated in detail in a tubing specification to which the tube manufacturer must comply. This specification describes all technical requirements which must be met for the tubing, including the bent regions. In addition to the material requirements, criteria are specified for dimensions, properties, cleanliness, heat treatment, defects and surface finish as well as inspection, testing, Q.A. non-destructive examination and packing/shipping.

RSG tubes receive a 100% volumetric ultrasonic inspection and a 100% eddy current surface inspection designed to detect indications of 0.002 inches deep. Tubes are rejected if they have one or more flaws in excess of 0.002 inches deep. This is more stringent than the Code requirement of 0.004 inches for indication depth and contributes to long tube life. RSG production tubes also receive an eddy current examination (ECT) for signal-to-noise ratio. Strip chart recordings of the test form part of the documentation. The criteria for acceptance is a minimum signal-to-noise ratio of 15:1 in the straight lengths. This allows detection of small flaws during inservice inspections and monitoring of flaw growth over time. The ability to detect small flaws and monitor their growth aids in planning maintenance activities. Tube bend geometry restricts use of the signal-to-noise criteria to straight runs of tube. The EPRI "Guidelines for Procurement of Alloy 690 Steam Generator Tubing", Report NP-6743-L, Vol. 2 are used as a basis to develop procurement requirements for RSG tubes.

2.3.2.2   Tube Support Materials

The tube support material for the RSGs is SA-240 Type 410S, a 12% chromium martensitic stainless steel (see Tables 2.3-3 and 2.3-4). It is supplied in the quenched and tempered, cold rolled, stress relieved condition. The tube support material resists corrosion, has adequate strength to support design loads and effectively resists wear when coupled with Alloy 690 tubing. Type 410S is compatible with manufacturing operations such as welding, machining, and assembly operations of the U-bend tube support and the lattice grid tube support structures as applicable. This material forms a tight, adherent oxide in secondary side water which is not greater in volume than the original metal. This greatly reduces the

potential for tube denting. After any applied welding processes, Type 410S stainless steel is stress relieved to reduce the hardness of the weld joint and to maintain adequate stress corrosion resistance. Yield strengths above 50,000 psi are easily achieved with 410S. ◢4◣

2.3.2.3 Corrosion Resistant Cladding

All primary side ferritic steel surfaces (primary side of the tubesheet and inside surfaces of the primary head) are clad with austenitic stainless steel (Type 308L Type 309L) or Alloy 600 or 690 weld metal to prevent corrosion. Tables 2.3-3 and 2.3-4 show the chemical ◢4◣ composition and mechanical properties of these materials. Critical aspects of this cladding are the thickness, cobalt content and surface finish. A maximum cobalt content of 0.10% ◢4◣ is specified to reduce residual radioactivity in areas where maintenance personnel will be working. A smooth surface finish of 63 microinches roughness average (RA) is specified by the customer to reduce accumulation of radioactive material on primary side surfaces.

The tubesheet is clad with Alloy 600 and has a minimum thickness of 0.312 inches in accordance with customer specifications. The primary head is clad with Type 308L and 309L stainless steel with a minimum thickness of 0.20 inches in accordance with customer specifications.

2.3.2.4 Feedwater Headers

RSG feedwater headers and associated components (gooseneck, thermal sleeve, header and J-tubes) carry feedwater at high velocities and accommodate temperature gradients that occur between the feedwater, the RSG water, and the RSG shell. The material chosen must resist erosion and corrosion, and be thermally compatible with the nozzle and feedwater piping materials. The use of low alloy (2¼ Cr 1 Mo) steel for the gooseneck, thermal sleeve, and header provides erosion and corrosion resistance and thermal compatibility with the nozzle and attached piping. Alloy 690 is used for the J-tubes to resist erosion because fluid velocities are high.

2.3.3 SG Internals Materials

Materials for various internal components such as shrouds, decks, lugs and steam separators, are carbon steels in a variety of product forms. These materials are readily weldable and have the required strength for their specific applications. Internal components in close proximity with tubing such as tube support components are fabricated from stainless materials. High chrome materials are also employed for components potentially susceptible to flow assisted corrosion such as some downcomer and feedwater components. When the material is being selected, sufficient corrosion allowances are applied to ensure that the ◢4◣ components are compatible with all operating/shutdown conditions including six chemical cleanings during the 60 year design life. The selected materials have performed well in other steam generators.

Materials for bolting applications are ASME Section II SA-193 Gr. B7 for bolts and studs, and SA-194 Gr. 1 or SA-194 Gr. 8 M.A. for nuts. These materials are supplied in the

quenched and tempered condition and have adequate strength and toughness for their applications. To minimize the potential for loose parts bolting applications have been limited. ⟨4⟩

2.3.4 Archive Samples

BWI provides archive samples of selected materials for later reference. Archive materials include pressure boundary base metals, pressure boundary weldments, selected internals, and tube samples. Pressure boundary base metal archives are provided from the steam drum head forging, secondary shell plate, the conical transition forging, tubesheet, forged primary head, and various forged inserts such as the handholes, manways, and feedwater nozzles. Weldment archives are provided for representative pressure boundary long seam welds. Single archive samples of the U-bend anti-vibration bars and lattice tube support bars are provided.

The following archive samples of tubing are provided:

1.   U-bends with 3 ft straight legs, two per row (ie. radius) from each of the 119 rows, for a total of 238 pieces, per station.

2.   One foot sections of continuous straight tubing from different tubes from each of the approximately 44 heat/lot combinations just prior to the thermal treatment.

3.   One foot sections of continuous straight tubing from different tubes of each of the approximately 44 heat/lot combinations after the thermal treatment.

4.   Two simulated tubesheet pieces each including 77 tubes fully expanded and seal welded into the tubesheet block (Mockup Requirement).

The tubing archive samples have undergone all the manufacturing process steps (in-process cleaning, grit blasting, acid cleaning, stress relief, etc.) applied to the production tubing installed in the generators except as noted in Item 2 above.

Table 2.3-1
CHEMICAL ANALYSIS REQUIREMENTS FOR PRESSURE BOUNDARY MATERIALS[1]

SPECIFICATION	CHEMICAL ANALYSIS, %										
	C	Mn	P	S	Si	Ni	Cr	Mo	V	Cu	Others
Base Material											
SA-508 Cl. 3	0.24	1.20-1.50	0.010	0.005	0.15-0.40	0.40-1.00	0.25	0.45-0.60	0.01	0.1	Al 0.04
SA-533 Tp B Cl.1	0.24	1.15-1.50	0.035	0.005	0.15-0.40	0.40-0.70	-	0.45-0.60	-		-
SA-106 Gr C	0.35	0.29-1.06	0.048	0.058	0.10	-	-	-	-	-	-
SA-336-F316N/316LN[2]	0.030	2.00	0.040	0.030	1.00	10.0-14.0	16.0-18.0	2.0-3.0	-		N 0.10-0.16
Small Nozzles											
SB-166 N06690	0.05	0.50	-	0.015	0.050	58.0 (min.)	27.0-31.0	-		0.50	Fe 7.0-11.0
Weld Consumable											
SFA 5.23 (EF2)	0.12-0.18	1.70-2.40	0.025	0.010	0.20	0.40-0.80	-	0.40-0.65	-	0.35	-
SFA 5.5 (E8018-C3)	0.12	0.40-1.25	0.03	0.010	0.80	0.80-1.10	0.15	0.15-0.35	0.05	-	-
UNS W86152	0.05	5.0	0.03	0.015	0.75	Bal.	28.0-31.5	0.50	-	.50	Cu 0.5 Fe 7.0-12.0 Ti 0.5 Col. 1.0-2.5
SFA 5.28 (ER80S-D2)	0.07-0.12	1.60-2.10	0.025	0.01	0.50-0.80	0.15	-	0.40-0.60	-	0.50	

[1] Maximum values or range unless stated otherwise.
[2] Procured to 316LN chemistry requirements.

Reference: ASME Code Section II, 1986.

Table 2.3-2
MECHANICAL PROPERTIES OF PRESSURE BOUNDARY MATERIALS[1]

Specification	Ultimate Tensile Strength (psi)	Yield Strength (psi)	Elong. (%)	Reduction in Area (%)	Max.[2]RT NDT (°F)
Base Material					
SA-508 Cl. 3	80,000-105,000	50,000	18.0	38.0	0
SA-533 Tp B Cl. 1	80,00-100,000	50,000	18.0	-	0
SA-106 Gr. C	70,000	40,000	20	-	0
SA-336-F316N/316LN	80,000	35,000	25.0	45	0
Small Nozzles					
SB-166 N06690	85,000	35,000	30.0	-	-
Weld Consumable					
SFA 5.23 (EF3)	80,000-100,000	68,000	20.0	-	0
SFA 5.5 (E8018-C3)	80,000	68,000-80,000	24.0	-	0
UNS W86152	80,000	-	30.0	-	0
SFA 5.28 (ER 80S-D2)	80,000	68,000	17.0	-	0

[1]   Minimum properties unless a range is shown.

[2]   Typical customer requirement.

Table 2.3-3
CHEMICAL ANALYSIS REQUIREMENTS OF CRITICAL-TO-FUNCTION MATERIALS[1]

SPECIFICATION	CHEMICAL ANALYSIS, %															
	Ni	Cr	Fe	Cu	Mn	C	Si	S	P	Co	N	Al	B	Ti	Mo	Nb+Ta
Tubes																
SB-163 Alloy 690 Code Case N-20-3	58.0 min.	28.5-31.0	8.0-11.0	0.5	0.50	0.015-0.025	0.50	0.002	0.015	0.016	0.03	0.50	0.004	0.40	0.2	0.1
Tube Support Materials																
SA-24C Tp 410S	0.60	11.5-13.0			1.00	0.05-0.08	1.00	0.030	0.040							
Cladding Materials[2]																
SFA 5.9 ER 308L	9.0-11.0	18.0-21.0		0.75	0.05-2.5	0.03	1.0	0.010	0.03	0.02					0.75	
SFA 5.9 ER309L	12.0-14.0	22.0-25.0		0.75	1.0-2.5	0.02	1.0	0.010	0.03	0.02					0.75	
N06082	67.0 min	18.0-22.0	3.0	0.5	2.5-3.5	0.16	0.50	0.015	0.03					0.75		
N06052	Rem.	28.0-31.5	7.0-11.0	0.3	1.0	0.04	0.50	0.015	0.020			1.1		1.0	0.5	
W86152	Rem.	28.0-31.5	7.0-12.0	0.5	5.0	0.05	0.75	0.015	0.030			0.5		0.5	0.5	
Feedwater Header Material																
SA-335 P22		1.9-2.6			0.30-0.60	0.15	0.50	0.030	0.030						0.87-1.13	

[1]   Maximum values unless a range is shown.
[2]   As-deposited weld metal.

Table 2.3-4

TABLE 2.3-4
MECHANICAL PROPERTIES OF CRITICAL-TO-FUNCTION MATERIALS[1]

	Ultimate Tensile Strength (ksi)	Yield Strength (ksi)	Elong. (%)	Reduction of Area (%)	Hardness Rockwell "B"
Tubes					
SB-163 Alloy 690	89.0	40-55	30	-	95 Max.
Tube Support Materials					
SA-240 Tp 410S	80.0	50.0	18	-	95 Max.
Feedwater Headers					
SA-335 P22	60.0	30.0	30	-	-

[1]   Minimum properties unless a range is shown.

TABLE 2.3-5
TYPICAL REPLACEMENT STEAM GENERATOR MATERIALS

Component	Material
Pressure Boundary	
Primary Head	SA-508 Cl 3
Tubesheet	SA-508 Cl 3
Boiler Shells	SA-533 Tp B Cl 1
Conical Shells	SA-508 Cl 3
Steam Drum	SA-533 Tp B Cl 1
Steam Dome	SA-508 Cl 3
Tubes	SB-163 Code Case N-20-3 Alloy 690
Primary Head Components	
Primary Nozzles	SA-508 Cl 3
Primary Nozzle Safe Ends	SA-336-F316N/316LN
Primary Manway Cover Plate	SA-533 Tp B Cl 1
Primary Manway Studs	SA-193 Gr B7
Primary Manway Diaphragm	SB-168 UNS N06690
Primary Manway Gasket Seating Surface	SFA 5.4E309L/308L SS Weld Build Up (Equivalent to 304L)
Primary Head Cladding	SFA 5.9 ER 309L/ER 308L (equivalent to 304L)
Tubesheet Primary Side Cladding	SFA 5.14 ER Ni-Cr3 (equivalent to Alloy 600, Inco 82)
Primary Divider Plate	SB-168 N06690
Nozzle Dam Rings	SB-168 N06690

TABLE 2.3-5 (cont'd)
TYPICAL REPLACEMENT STEAM GENERATOR MATERIALS

Secondary Shell Components

Handhole Forging Inserts	SA-508 Cl 3
Handhole Studs	SA-193 Gr B7
Handhole Diaphragm	SB-168 N06690
Handhole Gasket Seating Surface	SFA 5.4 E309L/308L SS Weld Buildup (Equivalent to 304L)
Handhole Cover Plate	SA-533 Tp B Cl 1
Inspection Ports	SFA 5.5 (E8018-C3) or SFA 5.23 (EF2)
Inspection Port Studs	SA-193 Gr B7
Inspection Port Diaphragms	SB-168 N06690
Inspection Port Cover Plate	SA-533 Tp B Cl 1
Inspection Port Gasket Seating Surface	E309L/308L SS Weld Buildup (Equivalent to 304L)
Secondary Manway Forging Insert	SA-508 Cl 3
Secondary Manway Studs	SA-193 Gr B7
Secondary Manway Gasket Seating Surface	SFA 5.4 E309L/308L SS Weld Buildup (Equivalent to 304L)
Secondary Manway Cover Plates	SA-533 Tp B Cl 1
Small Nozzles (3/4 in)	SB-166 N06690 and SFA 5.5 (E8018-C3) Weld Build-Up
3" Recirculation Nozzle	SFA 5.5 (E8018-C3) and UNS W86152
Blowdown Nozzles	SFA 5.5 (E8018-C3) and UNS W86152
Steam Outlet Nozzle/Flow Restrictor	SA-508 Cl 3 (integral with Dome)
Steam Flow Restrictor Venturies	SA-312 304L
Steam Outlet Nozzle Safe End	SA-106 Gr. C
Auxiliary Feedwater Nozzle	SA-508 Cl 3
Auxiliary Feedwater Nozzle Safe End	SB-166 UNS N06690
Main Feedwater Nozzle	SA-508 Cl 3
Main Feedwater Nozzle Transition Piece	SA-508 Cl. 3
Secondary Shell Lugs	SA-106 Gr. B
Vessel Lifting Trunnions	SA-508 Cl 3
Lateral Support Trunnion	SA-106 Gr. B

TABLE 2.3-5 (cont'd)
TYPICAL REPLACEMENT STEAM GENERATOR MATERIALS

Secondary Internals

Wrapper (Shroud)	SA-516 Gr 70
Shroud Pins	SA-193 Gr B7
Lattice Ring	SA-516 Gr 70
Lattice Bars	SA-240 Tp 410 S
Lattice Ring Studs	SA-193 Gr B7
U-Bend Flatbars	SA-240 Tp 410 S
U-Bend Archbars	SA-516 Gr 70
J-Tabs	SA-240 TP316L
Feedwater Nozzle Thermal Sleeve	SA-335 P22
Feedwater Distribution Ring	SA-335 P22
Auxiliary Nozzle Thermal Sleeve	SA-234 P22 Class 1
Auxiliary Feedwater Distribution System	SA-335 P22
Steam Drum Internals Structural Components	SA-516 Gr 70
Primary Cyclone Moisture Separators	A569/A36/A500 Gr B
Secondary Cyclone Moisture Separators	A569/A36/SA 106 Gr B

(next to J-Tabs row, in margin): △4

2.4   RSG DESIGN BASES

The following sections describe development of the RSG design bases, compliance with design codes and standards, and conformance with regulatory guidance.

2.4.1   Codes and Standards

This section describes the analysis of key design features of the RSG pressure boundary and outlines the experience and capability of BWI in design and analysis of pressure boundaries capable of withstanding internal pressure, seismic loads and loads from postulated accident events and cyclic loading that occur during operation. The RSGs are designed in accordance with the ASME Code requirements for Nuclear Pressure Vessels, Section III Division 1 (Subsection NB) and supported by comprehensive documentation.

BWI has considerable experience in designing and analyzing pressure vessels to meet Subsection NB, having performed the design and analysis on over 200 nuclear steam generators, in addition to other nuclear pressure vessels for domestic and foreign markets. In 1991, BWI completed the ASME Design Report for the Millstone 2 Replacement Steam Generator project, meeting the ASME Code, regulatory, and customer specification design requirements.

This section describes reconciliation of the code versions applied to the replacement steam generators with the existing plant code requirements. It also describes the certified design specification process, tests and inspections, and application of the ASME N-Stamp.

2.4.1.1   Code Reconciliation

The RSG specification requires that the steam generators be purchased to the latest edition of the ASME code and applicable addenda approved by 10CFR50.55a at the time of the purchase order issue date. ASME Section XI requires reconciliation of the design, fabrication, and examination for use of later codes on replacement components.

Relevant ASME III code technical changes from the edition to the OSG through the edition applied to the RSG are evaluated for significance to RSG fabrication. The results of these evaluations will be documented, and typically conclude that all ASME Section III changes that pertain to steam generators from the OSG edition up to and including the RSG edition are reconciled. The detailed reconciliations for technical changes that are different or less restrictive include technical resolutions of each difference.

2.4.1.2   ASME Certified Design Specification

The RSGs are designed, fabricated, inspected and tested according to the requirements of the 1986 ASME Boiler and Pressure Vessel Code, Section III (with no addenda) as defined in the Certified Design Specification provided by the owner in accordance with paragraph NCA-3250 of the Code.

2.4.1.3   Tests and Inspections

The RSGs are tested and inspected according to the requirements of the applicable ASME Boiler and Pressure Vessel Code, Sections III and V and applicable addenda.

Code-required tests include materials, fabrication and leak tests. Materials testing includes chemistry and mechanical property testing, and may include other tests such as ultrasonic, hydrostatic and surface finish tests depending on the material form and intended application. These are performed by BWI or material suppliers.

Fabrication testing includes magnetic particle, dye penetrant, radiographic, eddy current, and ultrasonic testing depending on the materials used, their form, application, and the specified manufacturing processes and sequence.

Leak testing includes hydrostatic testing of tube material and of the completed RSG. Additionally, BWI performs a helium leak test to confirm tube seal weld leak tightness.

2.4.1.4   N-Stamp

The RSGs are designed and fabricated at BWI's facilities located in Cambridge, Ontario, Canada. This facility holds the ASME N-Stamp and complies with the requirements and rules governing its use. The RSGs are inspected by the Authorized Nuclear Inspector according to the requirements of NCA-5000 of the ASME Code. The RSGs will be given an ASME Code N-Stamp prior to their release to ship from the Cambridge facility. The Certified Date Report is submitted to the owner.

2.4.2   Comparison to NRC Guidance

This section compares the RSG design to NRC General Design Criteria, NRC Regulatory Guides, applicable portions of the NRC Standard Review Plan, and other NRC information.

2.4.2.1   NRC General Design Criteria

The RSG design complies with the NRC General Design Criteria as follows:

GDC 1 - Quality Standards and Records -

> The RSG design meets the requirements of GDC 1. The RSGs are designed, fabricated, and tested to quality standards commensurate with their safety functions. The codes and standards used are defined elsewhere in this section, and have been selected on the basis of their applicability and adequacy. The BWI quality assurance program provides adequate assurance that the RSGs will satisfactorily perform their safety functions by meeting ASME NQA-1, and the requirements of 10CFR50, Appendix B. Appropriate records of RSG design, fabrication, and testing will be maintained, as described in Section 3.2 of this report.

GDC 2 - Design Bases for Protection Against Natural Phenomena -

> The RSG design meets the requirements of GDC 2. The RSGs are designed to withstand the effects of earthquakes and are protected from the effects of other natural phenomena by their location in the reactor building. RSG seismic design is described in Section 2.4.3 of this report. The RSG seismic design envelopes the existing plant seismic design basis.

GDC 4 - Environmental and Dynamic Effects Design Basis

> The RSGs are designed to accommodate the effects of and be compatible with the normal operation, maintenance, testing and postulated accidents including loss-of-coolant accidents. Protection against dynamic effects, including the effects of missiles, pipe whipping and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit are provided by the existing plant structures and systems. The RSG meets the requirements of GDC 4.

GDC 14 - Reactor Coolant Pressure Boundary -

> The RSG design meets the requirements of GDC 14. The RSG portions of the Reactor Coolant Pressure Boundary are designed, fabricated, and tested to have an extremely low probability of abnormal leakage, rapidly propagating failure or gross rupture. The RSG design meets ASME Section III requirements, and complies with 10CFR50.55a.

GDC 15 - Reactor Coolant System Design -

> The RSG design meets the requirements of GDC 15. The RSG portions of the reactor coolant system are designed with sufficient margin to assure that RSG limits are not exceeded during any condition of normal operation or anticipated operational occurrences.

GDC 30 - Quality of Reactor Coolant Pressure Boundary -

> The RSG design meets the requirements of GDC 30. RSG portions of the reactor coolant pressure boundary are designed, erected, and tested to the highest practical quality standards by meeting the ASME Code and 10CFR50, Appendix B. Detection and identification of the location of RSG leakage is through existing plant instrumentation and procedures.

GDC 31 - Fracture Protection of Reactor Coolant Pressure Boundary -

> The RSG design meets the requirements of GDC 31. The RSG portions of the reactor coolant pressure boundary are designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and

postulated accident conditions (1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design considers service temperatures and other operating, maintenance, testing and postulated accident conditions; uncertainties in determining material properties; affects of radiation on material properties; residual, steady state and transient stresses; and size of flaws.

GDC 32 - Inspection of Reactor Coolant Pressure Boundary -

The RSG design meets the requirements of GDC 32. RSG portions of the reactor coolant pressure boundary are designed to permit periodic inspection and testing of important areas and features to assess structural and leak-tight integrity. RSG provisions for inspection are provided in Section 2.2.11.

2.4.2.2   NRC Regulatory Guides

NRC Regulatory Guides were reviewed to identify those potentially applicable to the RSG replacement. Regulatory Guides identified as applicable to the licensing basis are met by the RSG design except as specified below:

RG 1.29, Rev. 3, 9/78, "Seismic Design Classification"

The RSG design complies with the NRC regulatory position. The replacement steam generator is a seismic Category I component. It is designed to withstand the effects of the SSE event and still perform its safety functions.

RG 1.31, Rev. 3, 4/78, "Control of Ferrite Content in Stainless Steel Weld Metal"

The RSG design complies with the regulatory position. All austenitic stainless steel welding consumables are procured and qualified in conformance to the ASME Code Section III, as well as this Regulatory Guide. In addition, stainless cladding welding procedure qualifications are subject to the minimum and maximum Ferrite Numbers of this Regulatory Guide and Section III.

RG 1.37, 3/73, "Quality Assurance Requirement for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants"

The RSG design complies with the NRC regulatory position. This NRC Regulatory Guide applies to the tubing material used in the RSG. The Regulatory Guide's positions are imposed on the tubing supplier in the RSG tube specification. Note that BWI uses ANSI 45.2.1 - 1980 rather than ANSI 45.2.1 - 1973 as referenced in RG 1.37.

RG 1.38, Rev. 2, 5/77, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants"

> The RSG design complies with the NRC regulatory position. This Regulatory Guide applies to the tubing from the tube mill to BWI as well as to the completed RSG itself.

RG 1.43, 5/73, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components"

> The RSG design complies with the NRC regulatory guidelines. The ferritic base metals which are clad (SA508, Cl.3, equivalent to SA533 Grade B, Cl.1, and SA508 Cl.1), are procured to fine grain practice and are not considered susceptible to underclad cracking. Weld procedure qualification is performed on material of the same specification (or equivalent) as used in production. BWI performs a 70 degree longitudinal UT examination for underclad cracking on all primary side clad inside RSG surfaces.

RG 1.44, 5/73, "Control of the Use of Sensitized Stainless Steel"

> The RSG design complies with the NRC regulatory position. Sensitized stainless steels are used for cladding in the primary head assembly, for cladding on gasket surfaces or for diaphragms, for the feedwater ring assembly and for the channel divider plate. In each instance the cladding is not a pressure retaining component and is L grade material on all primary surfaces; the feedwater assembly on the secondary side is not subject to post-weld heat treatment temperature and the divider plate is L grade material and is not subject to post-weld heat treatment temperatures.

RG 1.50, 5/73, "Control of Preheat Temperature for Welding of Low-Alloy Steel"

> The RSG design complies with the NRC regulatory position. This NRC Regulatory Guide applies to weld fabrication for low alloy components specified in Sections III and IX of the ASME Code. For production welds, BWI does not maintain preheat temperature until post-weld heat treatment as required by regulatory position C.2. Instead either the maximum interpass temperature is maintained four hours or the minimum preheat temperature is maintained eight hours after welding. However, as permitted in C.4, this deviation from requirement C.2 is permitted since the soundness of the welds are verified by an acceptable examination procedure appropriate to the weld under consideration. | △4

RG 1.60, Rev. 1, 12/93, "Design Response Spectra for Seismic Design of Nuclear Power Plants"

> BWI complies with the NRC regulatory position.

RG 1.61, 10/93, "Damping Values for Seismic Design of Nuclear Power Plants"

BWI complies with the NRC regulatory position.

RG 1.71, 12/73, "Welder Qualification for Areas of Limited Accessibility"

BWI complies with the regulatory position.

RG 1.83, Rev. 1, 7/75, "In-Service Inspection of Pressurized Water Reactor Steam Generator Tubes"

The RSG design complies with the regulatory position with the following clarifications:

The Regulatory Guide addresses both new and in-service components. The RSGs are new components and as such comply with the appropriate sections of this regulatory guide. Specifically C.1.a, C.1.b, C.2, C.3.a, and C.4.a. A 100 percent baseline inspection of the RSG is performed prior to the unit being put into service. BWI acceptance criteria exceeds the NRC guidelines for wall thickness reductions in that BWI limits wall thickness reductions to no more than 15% versus 20% allowed in the NRC guidelines.

RG 1.84, 4/92, "Design and Fabrication Code Case Acceptability ASME Section III Division 1"

This regulatory guide lists those Section III ASME Code Cases oriented to design and fabrication that are generally acceptable to the NRC staff for implementation in the licensing of light-water-cooled nuclear power plants. The following Code Case is used on the RSG:

N-411-1   Alternative Damping Values for Response Spectra Analysis of Class 1, 2 and 3 Piping Section III, Division 1.

RG 1.85, Revision 28, 1992, "Material Code Case Acceptability ASME Section III, Division 1"

The RSG design complies with the regulatory position. The following code cases are used on the RSG's:

N-10   This Code Case addresses the use of ultrasonic examination following RSG PWHT in lieu of radiographic examination. This Code Case was incorporated into ASME Section III by the 1986 addenda and endorsed by the NRC in RG 1.85, Revision 26, dated 1986. Through inclusion into the ASME Code, and annulment as a Code Case. N-10 is now listed in paragraph C.2 of RG 1.85, and deleted from the list of acceptable Code

Cases. However, paragraph D.3 of RG 1.85 permits components ordered to a previously approved Code Case to remain unchanged.

N-20-3   This Code Case addresses the use of Nickel-Chromium-Iron Alloy 690 tubing in the construction Class 1 components in accordance with ASME Section III.

N-71-15   This Code Case addresses additional materials for subsection NF Classes 1, 2, 3 and MC Component Supports fabricated by Welding Section III Division 1.

N-474-1   This Code Case addresses the use of Nickel-Chromium-Iron UNS N06690 material (Inconel 690) with a minimum yield strength of 35 KSI in the construction of Class 1 components in accordance with ASME Section III.

2142   This Code Case was approved by ASME on November 25, 1992 and addresses the use of Nickel-Chromium-Iron UNS W86052 welding filler metal (this material is similar to Inconel 690 material) in the construction of Class 1 components in accordance with ASME Section III.

2143   This Code Case was approved by ASME on November 25, 1992 and addresses the use of Nickel-Chromium-Iron UNS W86152 weld rod (this material is similar to Inconel 690 material) in the construction of Class 1 components in accordance with ASME Section III.

RG 1.92, Rev. 1, 1976, "Combining Modal Responses and Spatial Components in Seismic Response Analysis"

The RSG design complies with the regulatory position. Modal responses which are not closely spaced (greater than 10 percent) are combined by using the square root of the sum of the squares (SRSS) method to determine the maximum response for the seismic direction being considered. Dynamic systems that exhibit closely spaced modes (if any) are analyzed in accordance with the Regulatory Guide position.

RG 1.121, 8/76, "Bases for Plugging Degraded PWR Steam Generator Tubes"

The RSG design complies with the regulatory position.

RG 1.147, Rev. 5, 8/86, "In-service Inspection Code Case Acceptability ASME Section XI, Division 1"

> The RSG design complies with the regulatory position. Code Case N401 is used on the RSG to permit digitized collection and storage of data for permanently recording eddy current examination of pre-service exam.

RG 8.8, 1982, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonably Possible"

> The RSG design complies with the regulatory position as it relates to the design and supply of the RSG component. Design features are incorporated to minimize access time, to allow for rapid removal of equipment, to permit use of robotics, to provide surfaces which minimize crud traps, and to minimize welds requiring in-service inspection.

2.4.2.3   Comparison to NRC Standard Review Plans

NUREG 0800 was reviewed to identify Standard Review Plan (SRP) acceptance criteria that potentially affect RSG design or fabrication. The relation of each potentially applicable SRP acceptance criteria to RSG design or fabrication is described below. This section is intended to acknowledge existence of SRP guidance, and to document review and evaluation of potentially relevant items. Other than the statements below, blanket commitment to SRP acceptance criteria is not intended.

Reactor Coolant Pressure Boundary Materials (SRP 5.2.3)

The RSGs meet the requirements of GDC 1 and 30 as described in Section 2.4.2.1 and RGs 1.31, 1.43, 1.44, 1.71 and 1.85 as described in Section 2.4.2.2. Low alloy and carbon steels used as pressure retaining components are clad with austenitic stainless steel or Alloy 600. The fracture toughness requirements of 10CFR50, Appendix G are met.

Steam Generator Materials (SRP 5.4.2.1)

RSG materials are those permitted by Appendix I of Section III of the ASME Code and specified in detail in Parts A, B and C of Section II. RSG materials meet the requirements of RG 1.85 as described in Section 2.4.2.2, and meet the requirements of Appendix G to 10CFR50 as augmented by sub article NB-2300, Section II of the Code, and Appendix G, Article 2000 of the Code. The RSGs are designed to minimize crevice areas where the tubes pass through the tubesheets as described in Section 2.2.3, and to minimize crevice areas where the tubes pass through the lattice bar supports as described in Section 2.2.4.1. Compatibility of the RSG tube material with primary and secondary side fluids is discussed in Section 2.3.2.

Steam Generator Tube Inservice Inspection (SRP 5.4.2.2)

The design of the RSGs provides access for an inservice inspection program of all tubes. This is to permit detection of imperfections in the tube wall. A baseline tube inspection is scheduled prior to startup as described in Section 3.2.6.6.

2.4.2.4 Generic Letters, Bulletins, Notices and NUREGs

Indexes of NRC Generic Letters, Bulletins, Information Notices and NUREGs were reviewed to identify those potentially applicable to SG replacement. Documents potentially applicable to RSG replacement were evaluated. The relation of each potentially applicable document to RSG design or fabrication is described below. This section is intended to acknowledge existence of these NRC documents, and to document review and evaluation of potentially relevant information. Other than specific statements below, blanket commitment to recommendations or other provisions contained in the listed documents is not intended.

NRC Generic Letters:

81-16   Steam Generator Overfill

This generic Letter discusses the potential for steam line and safety valve damage from events that fill the main steam lines with water. It describes a potential concern that B&W (once through) steam generators may be subject to failure of weakened tubes by thermal transients resulting from overfill events. This generic letter does not affect RSG design, construction or installation.

81-28   Steam Generator Overfill

This generic Letter is a re-issue of GL 81-16 and attaches a report from the Office of Analysis and Evaluation of Operating Data (AEOD). The report identifies steam generator overfill events as an Unresolved Safety Issue. The AEOD report focused on operating procedures and analyses rather than design changes. This generic letter does not affect RSG design, construction or installation.

NRC Bulletins:

79-13   Cracking in Feedwater System Piping

NRC Bulletin 79-13 required thorough testing of feedwater-to-steam generator nozzle welds as a result of cracks detected in this area on numerous CE and Westinghouse plants. These inspections were conducted and results reported to the NRC. NRC Information Notice 91-28 (see below) concluded that because the prescribed inspections appeared to reliably detect cracking

at the feedwater piping-to-nozzle weld no expanded inspections or other actions were necessary. The required inspections can be performed on the RSGs. No further RSG design or fabrication actions are required to respond to the concerns identified in Bulletin 79-13.

88-02   Rapidly Propagating Cracks in Steam Generator Tubes

Bulletin 88-02 described a tube rupture attributed to high cycle fatigue cracking at the top tube support plate in a Westinghouse steam generator. The cracking was caused by flow induced vibration (FIV). Owners of Westinghouse plants with carbon steel tube support plates were requested to review inspection data for evidence of tube denting, and implement a program to minimize the probability of rapidly propagating fatigue failure. This Bulletin is not directly applicable to the BWI RSGs because they have stainless steel support bars rather than tube support plates. However, tube denting is addressed in Section 2.5 of this report, and FIV of SG tubes is addressed in Section 2.6.1.

NRC Information Notices:

91-28   Cracking in Feedwater System Piping

Information Notice 91-28 describes NUREG/CR-5285, a document that closed out Bulletin 79-13. NUREG/CR-5285 discusses feedwater piping-to-steam generator nozzle weld cracking, inspections confirming that cracks existed at many plants, repairs, and augmented inspections in the future. The NRC concluded that because the inspections appear to reliably detect degradation in feedwater piping that no additional actions are needed. The RSG design includes a thermal sleeve that is welded to the feedwater piping and prevents header drainage. The weld is located to avoid geometric and thermal transitions in the pressure boundary that could concentrate stress. Sections 2.2.5.3 and 2.2.5.4 discuss causes of cracks and the design features incorporated into the RSG in consideration of minimizing these causes of cracking.

93-20   Thermal Fatigue Cracking of Feedwater Piping to Steam Generators

Information Notice 93-20 states that additional feedwater line cracks have escaped detection although augmented inspections are being performed. It states that ASME Section XI does not appear adequate to detect these thermal fatigue feedwater piping cracks. It also states that replacement of piping with the same material does not solve the cracking problem. To facilitate periodic inspection, the RSG thermal sleeve is positioned to avoid interference with feedwater pipe and nozzle weld inspection. Also a detailed fatigue analysis in accordance with ASME Section III is performed to analyze the effects of both mechanical and thermal induced stresses such as

stratification.

NRC NUREGs:

0909   Ginna Event

Review of this report revealed no specific RSG design impacts. The report includes a discussion of steam generator tube operating history, with past tube plugging information, historical inspection results, and descriptions of foreign particles found during inspections of both steam generators after the tube rupture event. The cause of the Ginna event appears to have been tube fretting caused by foreign material that had accumulated in the lower tubesheet area. The thickness of the ruptured tube at the break was only five percent of its original nominal value. Steam generator design was not an issue in the Ginna event. During fabrication, cleanliness procedures are employed to ensure that debris and foreign materials are excluded from the RSG. Full accountability is maintained of all tools and loose parts used during assembly. An inspection for foreign objects is performed just prior to final closure before shipment, to further minimize the potential for loose secondary side parts (see Sections 3.2.7 and 3.2.8).

2.4.3   Seismic Requirements

The design basis for seismic requirements is detailed in the Certified Design Specification (CDS) according to the requirements of the ASME Code Section III. The CDS provides the seismic, performance and transient loadings which are used to design the RSG. Included are the mechanical design data, thermal-hydraulic design data, service loads and test loads. The loadings are identified by service level (levels A, B, C, D or test) and service limits are applied according to the requirements of Section III. These service levels include requirements for OBE and SSE conditions and are further described in Section 2.8. In order to Code stamp the vessel, the requirements of the ASME Code must be met and verified by the Authorized Inspector.

2.4.4   Performance Requirements

The design basis for performance requirements is detailed in the Certified Design Specification (CDS) according to the requirements of the ASME Code Section III. The CDS requires that the RSGs generate steam that meets or exceeds its pressure, temperature, and flow rate requirements when supplied with reactor coolant and feedwater at the full load conditions. It requires that the RSGs not exceed the design basis limit of 0.25% moisture content when operating at full load conditions.

The RSG performance requirements duplicate or exceed those of the original steam generator design. Additional RSG performance considerations include minimization of flow induced vibration and avoidance of loose parts.

2.4.5   Accidents and Transients

The design basis for accidents and transients is detailed in the Certified Design Specification (CDS) according to the requirements of the ASME Code Section III. The CDS provides the seismic, performance and transient loadings which are used to design the RSG. Included are the mechanical design data, thermal-hydraulic design data, service loads and test loads. The RSG design transients duplicate or exceed those used for the OSG. The loadings are identified by service level (levels A, B, C, D or test) and service limits are applied according to the requirements of Section III. In order to Code stamp the vessel, the requirements of the ASME Code must be met and verified by the Authorized Inspector.

2.5   DESIGN IMPROVEMENTS

BWI studies world-wide steam generator performance to identify industry problems and potential steam generator design improvements. BWI has identified 45 potential steam generator problem areas. Most are PWR issues, fewer are heavy water reactor problems. Each potential problem area was considered in designing the RSG. Table 2.5-1 summarizes these problem areas and identifies the RSG features that preclude or minimize each. Figure 2.5-1 lists the problem areas and identifies the potentially affected part of the steam generator.

The following paragraphs highlight the more important problem areas and identify RSG design measures to preclude or minimize their occurrence.

- **Tubesheet Cladding Separation** is avoided through the use of MIG cladding processes, ultrasonic testing of applied cladding, and a channel head divider plate that does not impose high stresses on the cladding at the tubesheet or primary head connections.

- **Tube-to-Tubesheet Crevice IGA** is avoided by selection and control of the tube alloy and tube expansion to minimize the crevice at the tubesheet secondary face.

- **Tube-to-Tubesheet Crevice and Primary Side SCC** is avoided by using low residual stress expansion techniques. BWI employs full depth hydraulic expansion and monitors tube and tubesheet hole tolerances and tubesheet thickness variations. Additionally, once expanded the tube is not subjected to thermal stress by post weld heat treatment processes that could relax the tube recreating the crevice.

- **Tube Sensitization** at the tube-to-tubesheet weld is avoided by stress relieving the pressure boundary of the steam generator, including the primary head-to-tubesheet weld (except for the closing seam with the steam dome portion) prior to tubing the generator. The closing seam, located in the conical transition section connecting the steam drum and tube bundle shell section is stress relieved after tubing the RSG. Stress relief of this weld is performed locally, using a controlled process and does not affect the tubesheet area.

- The **Tubesheet Sludge Pile** is minimized through the use of a high circulation ratio in the steam generator, high cross flow at the tubesheet secondary face, low qualities at the tubesheet, high capacity blowdown capability, water chemistry limits, and provision of access ports for sludge lancing.

- **Tube Support Crud Accumulation** and consequent undesirable increases in tube support pressure drop over the 60-year design life of the steam generator is avoided through the use of "open-flow" lattice grids.

- **Denting by Tube Supports** is precluded by open-flow lattice grid supports, line contact with tubes, high circulation flows to keep the tube-to-support areas clean, and selection of a tube support material (410S) that resists corrosion.

- **Tube Vibration Fretting Wear** at lattice grid and U-bend supports is avoided by maintaining small clearances, installing U-bend restraints as the tubing process proceeds, applying conservative analytical predictive techniques, and selecting tube support material that resists wear with the Alloy 690 interface. /4\

- **U-Bend Cracking of Inner Row Tubes** is avoided by use of large minimum radius U-bends and stress relieving the small, tight-radius inner row bends.

- **Water Hammer** is minimized by avoiding potential sites for the steam pockets that cause water hammer. Feedwater inlet piping volume is limited with discharge through inverted "J-tubes".

- **Loose Parts** are avoided by minimizing the number of RSG parts and by securely capturing the applied parts that are used. All threaded connections are encapsulated such that hypothetically severed studs remain contained. /4\ Loose parts and control of parts during fabrication are discussed further in Sections 2.5.2., 3.2.7 and 3.2.8.

- **Steam Separator Moisture Carryover** is reduced by testing prototype steam separators to 130% of required capacity prior to release for manufacture. These tests are described in Section 2.6.4.

- **Pressure Boundary Weld Cracking** is prevented by selection and testing of shell material and weld consumables for resistance to cracking, by control of welding processes, and by control of preheat and post-weld heat treatment.

The RSG is designed to resist the steam generator failure mechanisms that have been identified through industry operating experience. By following industry problems, the failure mechanisms listed above were identified and are considered in the RSG design. The following subsections describe design measures that further contribute to minimization of these problems, maximization of performance and reliability, and facilitation of maintenance.

2.5.1   Minimization of Corrosion

2.5.1.1   Materials Employed to Minimize Corrosion

PWR steam generators have experienced primary side stress corrosion cracking (SCC) in the small-radius U-bends and in the expanded zones of tubes. These problems have been less severe in heavy water reactors, perhaps partly due to the lower operating temperatures in these plants. But temperature differences do not fully explain the SCC differences. Typical PWR steam generator tubing heat treatment practices of the late 1960's and early

1970's used lower annealing temperatures than those used for heavy water reactor steam generators, and did not use subsequent heat treatment. Typical PWR values are compared below with values for a typical heavy water reactor of the same vintage (Bruce A):

	Typical PWR Steam Generator	Bruce A
Tubing Alloy	600	600
Anneal Temperature	1760°F	1850°F (min)
Subsequent Heat Treatment	None	Stress relief in the vessel at 1125°F

The higher annealing temperature (improves grain structure) and stress relief (improves grain boundary carbide precipitation) improved the primary SCC resistance at Bruce A. Laboratory results show that heat treatment of the Alloy 600 material significantly increases its resistance to pure water SCC. Higher annealing temperature and stress relief, assisted by the lower heavy water reactor temperatures have yielded excellent tube performance.

Comparison of the Bruce A (Alloy 600) steam generator tubes to those of a typical PWR steam generator, shows the following temperature differences:

Plant	T-hot	T-Cold	T-U-bend	Maximum heat flux
Bruce A	580	507	535	107,000
Typical PWR	610	550	570	105,000

Heat treatment of Alloys 690 and 600 for optimum SCC resistance involve a mill annealing at temperatures sufficient to put all of the carbon into solid solution, followed by a thermal treatment to precipitate carbides on the grain boundaries in the tube metal microstructure. Resistance to SCC is greatest when the grain boundaries are well decorated with carbides.

The development of Alloy 690 steam generator tubing was driven by stress corrosion cracking (SCC) of Alloy 600 in both primary side and secondary side water environments. These nickel-chromium-iron alloys differ in that the Alloy 690 has 27-31% chromium and Alloy 600 has 14-17% chromium. A considerable amount of work has gone into evaluating the effects of thermo-mechanical processing and environment on the SCC resistance of Alloy 690 and two other popular steam generator materials - Alloy 600 and Alloy 800.

The SCC testing has demonstrated that Alloys 690 and 800 are highly resistant to cracking in primary side water environments, while Alloy 600 is susceptible to primary water stress corrosion cracking (PWSCC). Alloy 690 resists SCC as well or better than Alloy 600 or Alloy 800 in most secondary side water environments (Reference 3). Alloy 690 has somewhat greater SCC resistance than Alloy 600 in concentrated caustic environments. Alloy 690 resists pitting and general corrosion as well or better than Alloy 600 or Alloy 800.

Many tests have been performed which compare the PWSCC behaviour of candidate steam generator tubing. Figure 2.5-2 displays test results on statically loaded Reverse U-Bend (RUB) specimens which were presented at the 1985 EPRI Workshop on Alloy 690. These results indicate that both the mill annealed and thermally treated conditions of Alloy 600 are susceptible to PWSCC. Cracking was observed in the mill annealed specimens in approximately 300 hours while 800 hours were required to crack the thermally treated material. Alloy 690 and Alloy 800 RUBs did not exhibit PWSCC in this 12,000 hour test (Reference 1).

Figure 2.5-3 presents data from statically loaded tube tensile specimens tested in 680°F primary water. In this test, both Alloys 600 and 800 exhibited PWSCC within 2,900 hours. Alloy 690 did not exhibit PWSCC in this test after 7,000 hours (Reference 1).

Additional results collected on highly stressed Alloy 690 specimens tested in a variety of pure and primary water environments for times up to 31,000 hours indicate that Alloy 690 is highly resistant to PWSCC (References 2 through 6).

Alloy 690 specimens were also compared to Alloy 600 specimens in "steam tests" which produce accelerated PWSCC. Steam tests are performed in 760°F steam produced from hydrogenated pure water. As in the water tests, Alloy 600 cracked within 1,000 hours while Alloy 690 displayed no PWSCC after exposure times up to 6,000 hours (References 4 and 5).

The above results indicate that Alloy 600 is susceptible to PWSCC while Alloy 800 has generally displayed resistance to PWSCC. PWSCC of Alloy 690 has not been reported in the open literature.

2.5.1.2   Design Features Employed to Minimize Corrosion

Several design features that have been developed and included in the BWI steam generator design are directed toward avoiding corrosion problems which have been observed in recirculating steam generators. These features include the selection of corrosion-resistant materials of construction, an improved tube support design, improvement of fluid dynamics for introduction of feedwater, high circulation ratio, blowdown header design and high blowdown rate capability. All wetted surfaces of the steam generator primary side are constructed of stainless steel or Alloy 690, or are clad with weld-deposited stainless steel or Alloy 600 (tubesheet cladding). RSG materials are described in Section 2.3. Material compatibility is evaluated as part of the BWI Chemical Cleaning Qualification Program (Section 2.6.5).

Tube-to-tubesheet crevice IGA is avoided by material selection and expansion of the tube to close the crevice at the tubesheet secondary face. Primary Side SCC is avoided by using low residual stress expanding techniques detailed in Section 2.5.1.1. Tube sensitization due to welding stress relief has been eliminated by stress relief prior to commencement of the tubing operation.

The Tubesheet Sludge Pile is minimized through high circulation, use of a special two-zone high/low resistance lower lattice grid, high capacity blowdown and water chemistry recommendations. The dual resistance lower lattice grid has an outer region of higher resistance to vertical cross flow through it which enables the downcomer flow to penetrate deep into the tube bundle along the secondary face of the tubesheet. The benefit of deeper penetration is to minimize the zone of net boiling and low velocities, that contribute to sludge pile formation. Three-dimensional flow simulation is used to quantify the boiling and low velocity zones. This is accomplished by the ATHOS Computer Code discussed in Section 2.6.2.2. Such zones can be graphically illustrated in figures similar to those shown in Figures 2.5-4 through 2.5-7. These design measures are complemented by periodic cleaning of the tubesheet. Tube support crud accumulation and consequent increases in tube support pressure drop is avoided through the use of "open flow" lattice grids.

References for Section 2.5.1

1.   T. Yonezawa, et al., "Evaluation of the Corrosion Resistance of Alloy 690", EPRI NP-4665S-SR Proceedings: Workshop on Thermally Treated Alloy 690 Tubes for Nuclear Steam Generators, Pittsburgh, Pennsylvania, June 26-28, 1986, paper no. 12.

2.   R.M. Rentler, "Laboratory Corrosion Test Results of Alloy 600 and 690 Steam Generator Tubing Exposed to Faulted Secondary Chemistry Environments" EPRI NP-4665S-SR Proceedings: Workshop on Thermally Treated Alloy 690 Tubes for Nuclear Steam Generators, Pittsburgh, Pennsylvania, June 26-28, 1986, paper no. 11.

3.   R.J. Jacko, A.W. Klein and C.E. Sessions, "An Overview of Laboratory Tests Conducted by Westinghouse to Qualify Alloy 690 Steam Generator Tubing" EPRI NP-4665S-SR Proceedings: Workshop on Thermally Treated Alloy 690 Tubes for Nuclear Steam Generators, Pittsburgh, Pennsylvania, June 26-28, 1986, paper no. 9a.

4.   R.G. Aspden, T.F. Grand and D.L. Harrod, "Corrosion Performance of Alloy 690" EPRI NP-6750-M Proceedings: 1989 EPRI Alloy 690 Workshop, New Orleans, Louisiana, April 12-14, 1989.

5.   G. Santarini, et. al., "Alloy 690: Recent Corrosion Results" EPRI NP-6750-M Proceedings: 1989 EPRI Alloy 690 Workshop, New Orleans, Louisiana, April 12-14, 1989.

6.   K. Norring, J. Engstorm, and H. Tornblom, "Intergranular Stress Corrosion Cracking of Steam Generator Tubing: 25,000 Hours Testing of Alloy 600 and Alloy 690" Proceedings of the Fourth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors. Jekyll Island, Georgia, August 6-10, 1989, pg. 12-1.

2.5.2   Minimization of Loose Parts

The existence of a loose part within a steam generator, whether originating from outside or within the unit, can cause significant damage to the steam generator particularly the tube bundle components. The RSG design employs specific criteria to minimize the potential for loose parts, namely,

1.   The total number of parts making up the steam generator assembly is minimized within the overall design constraints. The only installed parts on the primary side are the divider plate, and nozzle dam retaining rings. The retaining rings are attached by full penetration welds to the head. The divider plate is attached to the tubesheet and primary head by full penetration weld around its entire periphery. On the secondary side, most installed parts are weldments, others are securely captured using the measures described below.

2.   Where possible welded joints are used in lieu of bolted joints.

3.   Where fasteners must be used, the design fastener material is specified with a chemistry that permits lock-welding of the component.

4.   Where internal fasteners that are not lock welded are used, the bolt or nut is locked in place with a corrosion resistant locking tab. To avoid cracking of high strength bolting materials, fasteners with ultimate tensile strengths over 150 ksi are not used.

5.   Manufacturing and quality control procedures (described in Section 3.2.7.1) ensure that loose parts are not left in the steam generator before shipment.

The above criteria ensures that the completed steam generator has minimal loose parts potential. Typically, each steam generator is equipped with provisions for mounting acoustic sensors for detection of loose parts during operation. There are pads on the inlet side of the channel head and on the shell side of the RSG for mounting the acoustic sensors in the same locations as on the OSG.

2.5.3   RSG Performance Improvements

Several sections of this report discuss improvements in the RSG compared to the OSG. RSG performance is typically improved by increased heat transfer surface, higher circulation rate, lower moisture carryover and better water level stability during transient conditions. The specification and design changes that produce these improvements are discussed in Section 2.2.2 (higher heat transfer surface), Section 2.2.7.3 (high circulation rate), Section 2.2.7 (lower moisture carryover) and Section 2.2.15 (water level stability).

2.5.4   Maintenance and Reliability Improvements

RSG design aspects that are expected to reduce maintenance or improve reliability include

reduced corrosion (leading to tube wall thinning or cracking), less release of material to the reactor coolant that could become activated, and secondary side design and materials selection that reduce sludge accumulation and promote sludge removal. The paragraphs below describe the more significant maintenance and reliability design improvements.

Steam generator reliability can be reduced by tube thinning and cracking. The Alloy 690 tubing used in the RSGs is less prone to stress corrosion cracking than the Alloy 600 tubing used in most current PWR steam generators. Additional information on Alloy 690's resistance to stress corrosion cracking is presented in Section 2.3.2.

Alloy 690 forms tightly adherent oxide films. General corrosion does not pose a large problem for Alloys 690, 600 and 800, however, in high temperature flowing ammoniated water that simulated secondary side water chemistry, the metal release is related to the alloy's chromium content. Therefore the metal release rate of Alloy 690 (30% Cr) is less than Alloy 800 (21% Cr) and less than Alloy 600 (15% Cr).

Because of its high chromium content, Alloy 690 is expected to exhibit less general corrosion than Alloys 600 or 800 in most secondary side environments. Testing performed in an environment that simulated a condenser leak indicated that Alloy 690 had the greatest resistance to wastage, followed by Alloy 800 and Alloy 600.

The lower metal release rate of Alloy 690 benefits both the primary and secondary sides of the steam generator. On the primary side, the low metal release rate of Alloy 690 leads to less cobalt transfer and lower radiation levels in steam generators tubed with Alloy 690 than those tubed with Alloy 600.

2.5.5   Inservice Inspection Improvements

The RSG design accommodates inservice inspection (ISI) with uncluttered design and ample access. Section 2.2.11 describes access provisions for the RSG primary and secondary side inspection, maintenance and repair. Section 2.2.8 describes the reduced number of pressure boundary welds that require ISI. Reduced inspection requirements and improved accessibility of areas requiring inspection reduce personnel time, expected dose and calendar time for ISI.

Table 2.5-1

Industry Steam Generator Problem Areas and RSG Measures to Address Them

Problem		BWI Heavy Water S.G. Experience	BWI Solution to Generic Industry Problem
1.	Secondary SCC/IGA in crevice	None. All crevices closed.	Hydraulic expansion to close crevice. Corrosion resistant 690 tubing.
2.	Denting at tube sheet face	None.	Closed crevices. High circulation ratio to limit sludge buildup.
3.	Tube SCC at ID of expanded region	None. Stress-relieved Alloy 600 or 800 material.	Hydraulic expansion (low residual stress). Alloy 690 material resists SCC.
4.	Caustic IGA under sludge	None.	High circulation ratio to limit sludge buildup. Water chemistry recommendations. Alloy 690 material resists IGA.
5.	Sludge accumulation	Present in older steam generators. Phosphate and copper based.	High circulation ratio. Enhanced bundle flow penetration. Accessibility for sludge lancing.
6.	Phosphate wastage-sludge	None despite phosphates in older units.	Water chemistry recommendations. Limit sludge accumulation.
7.	Secondary loose parts damage	3 minor incidents. Approximately 10 tubes (total) plugged.	Limit part count. All-welded structure. Rigorous tool control. Cleanliness check during assembly.
8.	Open span pitting	None.	Water chemistry recommendations. Alloy 690 material.
9.	Pitting under support plate deposits	Experience at Pickering Reactor 5. Approximately 2000 tubes plugged.	Use lattice grids. Water chemistry recommendations. Alloy 690 material.
10.	Tube damage at supports, improper heat treatment	None. No local stress reliefs done. BWI manufacturing sequence	

11.	Support Plate clogging	Extensive at Bruce A and Pickering B.	Use lattice grids Water chemistry recommendations
12.	Acid wastage	None.	Water chemistry recommendations. High circulation ratio. Alloy 690 material.
13.	Phosphate wastage	None.	Water Chemistry recommendations.
14.	Denting at tube support plate	Very minor, if any.	Stainless lattice grids.
15.	Sulphur/sulfate attack	May have happened at Bruce A.	Water chemistry recommendations. Alloy 690 material
16.	Secondary IGA/SCC, at tube support	None.	Stainless lattice grids. Alloy 690 tubing.
17.	High cycle fatigue	Very small occurrence in early operation, Bruce A.	Ample U-bend support. Alloy 690 tubing.
18.	Wastage under anti-vibration bar deposits	None.	Non-accumulating flat-bar design. Water chemistry recommendations. Alloy 690 tube material.
19.	U-bend fretting	Bruce B scallop bar designs.	Flat-bars lattice. Ample support. Small clearances.
20.	Failures due to improperly installed anti-vibration bars	None.	Install FURs as bundle is tubed. Small clearances.
21.	Waterhammer	None.	Gooseneck feedwater ring inlet. J-tube header discharge. Low header elevation.
22.	J-tube weld failures	None.	Large size J-tubes. Substantial, heat treated welds.
23.	Stratification and cracking of feed nozzle	None.	Gooseneck inlet. Extended thermal sleeve.
24.	Moisture carryover	Some in oldest steam generators due to clogging of scrubber type driers.	Cyclone separators.

25.	Secondary drier clogging	Some in oldest steam generators due to clogging of scrubber type driers.	Centrifugal drier steam velocity precludes deposits.
26.	Water level trips	Some in older units. Operating procedures improved.	High circulation ratio reduces level variation. Separators have wide tolerance range.
27.	Water level stability	Experienced at Bruce A, after tube supports severely clogged.	Balance single and two phase losses. Non-clogging lattices maintain balance.
28.	Steam carryunder	None.	Cyclone separators eliminate. Feed ring design assists.
29.	Pressure boundary weld failures	None.	Forged shells. No corner welds. Material controls. Preheat and post-heat controls. Multiple inspections.
30.	Secondary SCC/IGA at anti-vibration bars	Bruce A, traced to acid excursions.	Flat-bar lattice design has no deposition sites. Flat bars cannot dent or restrict tubes. Alloy 690 material.
31.	Lead induced IGA/TGC	Bruce A, units 1 and 2.	Water chemistry recommendations. Foreign material exclusion controls.
32.	Impingement Erosion	None.	Considered N/A to RSG. Lattice grid supports.
33.	Primary SCC, tight bends	None.	Large minimum radius bends in innermost tubes. Stress relieve inner row tubes. Tube material selection (Alloy 690).
34.	Tube support plate deterioration	None.	Stainless steel lattices.
35.	Surface fouling	Bruce A. Water chemistry related.	High circulation ratio. Water chemistry recommendations.
36.	Fretting at top of preheater	None.	Flow mixing region design.

37.	Preheater baffle sludge accumulation	BWI not aware of any.	Axial flow preheaters using lattice grids as tube supports.
38.	Combined vibration/ corrosion	None.	Design to eliminate vibration. Water chemistry recommendations.
39.	Tube fretting in preheater	None in BWI units. Two loose part fretting failures in Argentina.	Inlet flow distribution belt. Analysis of entrance region. Conservative fretting criteria.
40.	Fretting against lane blocker	None.	RSG has no lane blockers.
41.	Primary side loose parts damage	One incident at Gentilly 2.	Flush welds (less susceptible to damage). Primary side designed to facilitate robotic repair.
42.	Plug failures	None.	Recommend state-of-the art plugs. RSG should need fewer plugs.
43.	Cladding Separation	None.	Clad application control. Stringent inspection. Low divider plate stress on cladding.
44.	Manway/handhole gasket failures	One handhole gasket failure at Bruce A.	Advanced gasket designs. Covers and tooling designed for effective installation.

24. MOISTURE CARRYOVER

25. SECONDARY DRIER CLOGGING

26. WATER LEVEL TRIPS

27. WATER LEVEL STABILITY

28. STEAM CARRYUNDER

29. PRESSURE BOUNDARY
WELD FAILURES

30. SECONDARY IGA/SCC AT
TUBE SUPPORTS

31. LEAD INDUCED IGA

32. IMPINGEMENT EROSION

33. PRIMARY SCC-TIGHT BENDS

34. TSP (CARBON STEEL)
DETERIORATION

35. SURFACE FOULING

36. FRETTING TOP OF PREHEATER

37. PREHEATER BAFFLE SLUDGE
ACCUMULATION

38. COMBINED VIBRATION/CORROSION

39. TUBE FRETTING, PREHEATER

40. FRETTING AGAINST LANE
BLOCKER

41. PRIMARY SIDE LOOSE
PARTS DAMAGE

42. PLUG FAILURES

43. CLADDING SEPARATION

44. MANWAY/HANDHOLE GASKET FAILURES

23. STRATIFICATION AND CRACKING

22. J-TUBE WELD FAILURES

21. WATER HAMMER

20. FAILURES DUE TO IMPROPERLY
INSTALLED AVB'S

19. U-BEND FRETTING

18. WASTAGE, DEPOSITS AVB'S

17. HIGH CYCLE FATIGUE

16. SECONDARY IGA/SCC AT TSP

15. SULFUR/SULPHATE ATTACK

14. DENTING AT TSP

13. PHOSPHATE WASTAGE

12. ACID WASTAGE

11. SUPPORT PLATE CLOGGING

10. TUBE DAMAGE AT SUPPORTS,
IMPROPER HEAT TREATMENT

9. PITTING-UNDER SUPPORT
PLATE DEPOSITS

8. PITTING-OPEN SPAN

7. SECONDARY LOOSE PARTS DAMAGE

6. PHOSPHATE WASTAGE UNDER SLUDGE

5. SLUDGE ACCUMULATION

4. CAUSTIC IGA UNDER SLUDGE

3. I.D. SCC-EXPANDED REGION

2. DENTING AT TUBESHEET FACE

1. SECONDARY SIDE IGA/SCC
IN CREVICE

INDUSTRY WIDE STEAM GENERATOR PROBLEMS

FIGURE 2.5-1
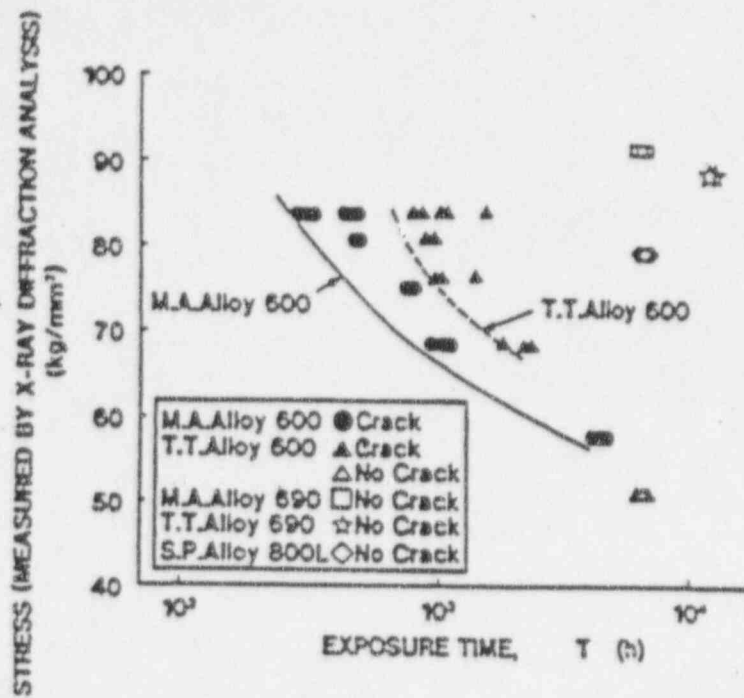
STRESS VS. DETECTION TIME FOR SCC
IN TEMPERATURE ACCELERATED
(360°C) PRIMARY WATER

FIGURE 2.5-2
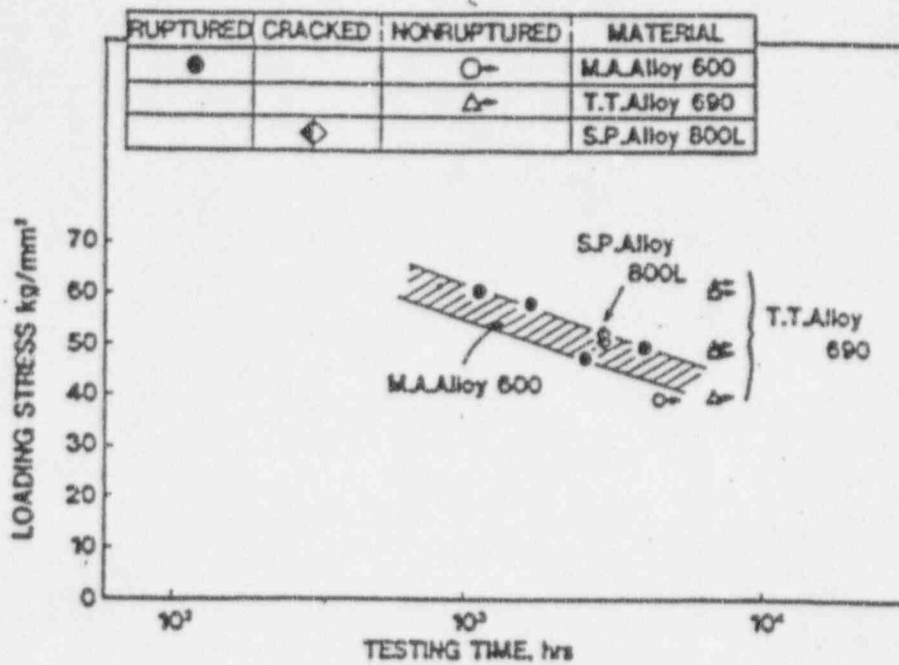
CONSTANT LOADING STRESS CORROSION CRACKING TEST
RESULTS IN TEMPERATURE ACCELERATED
(360°C) PRIMARY WATER

FIGURE 2.5-3

STEAM QUALITY CONTOURS

FIGURE 2.5-4

GAP VELOCITY CONTOURS

FIGURE 2.5-5

TUBESHEET HOT LEG FLUID QUALITY PROFILE
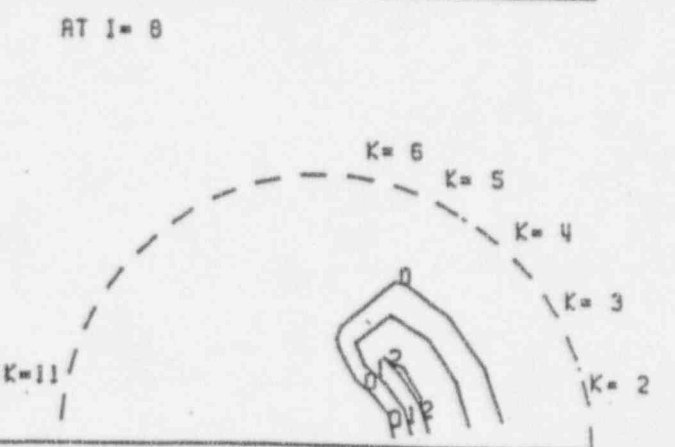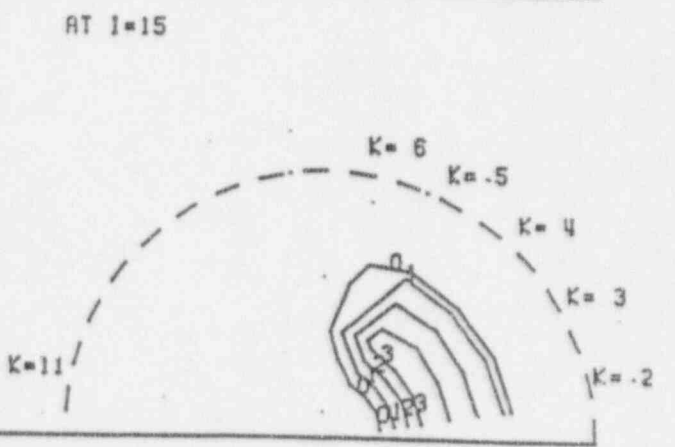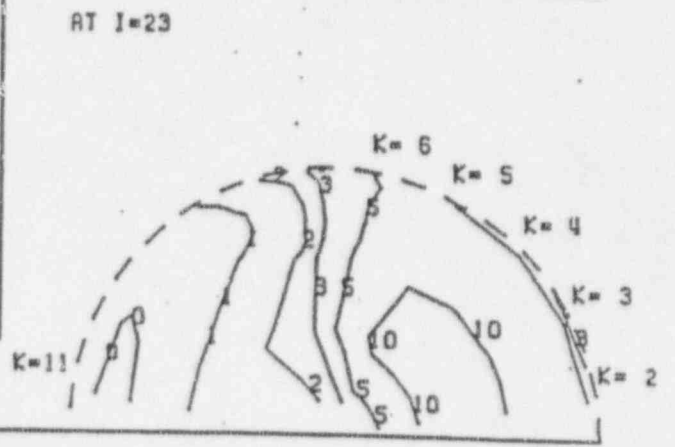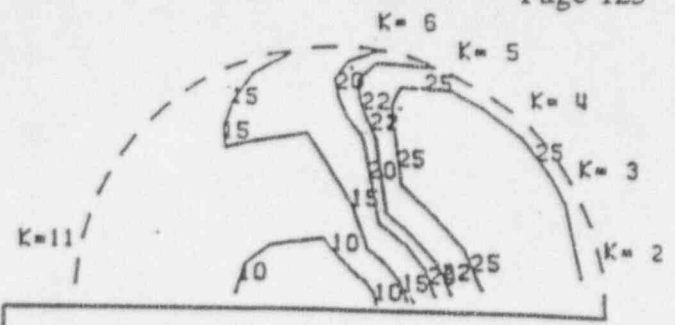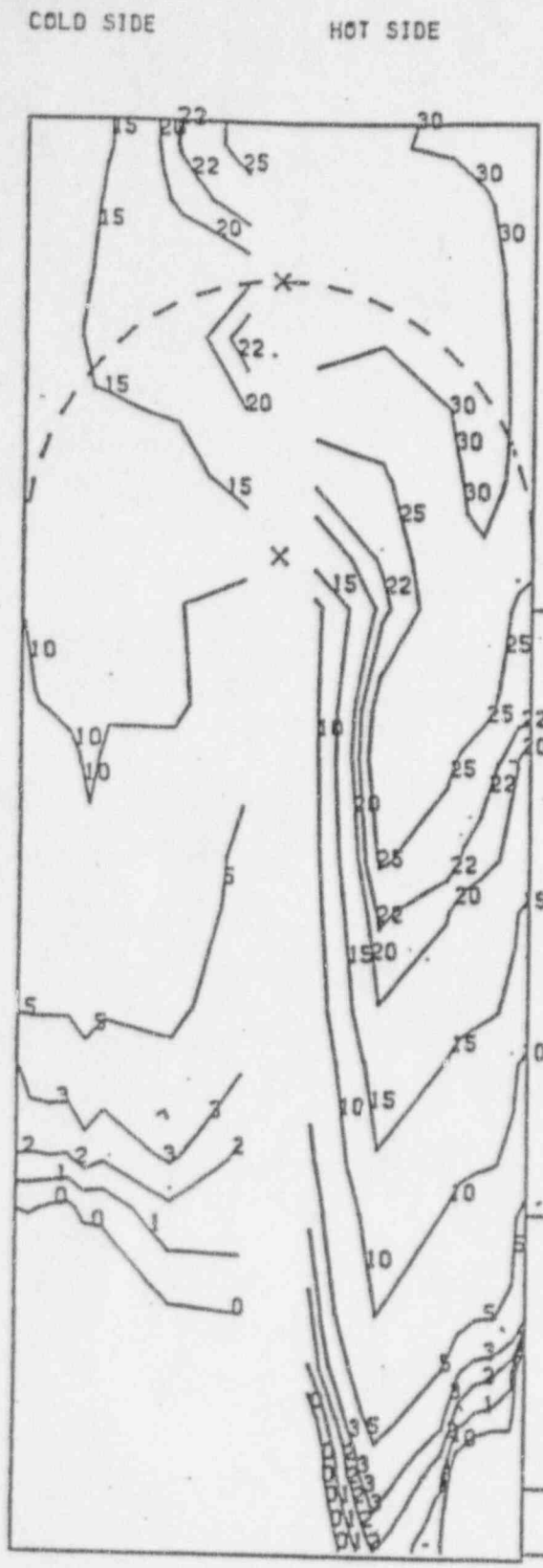
FIGURE 2.5-6
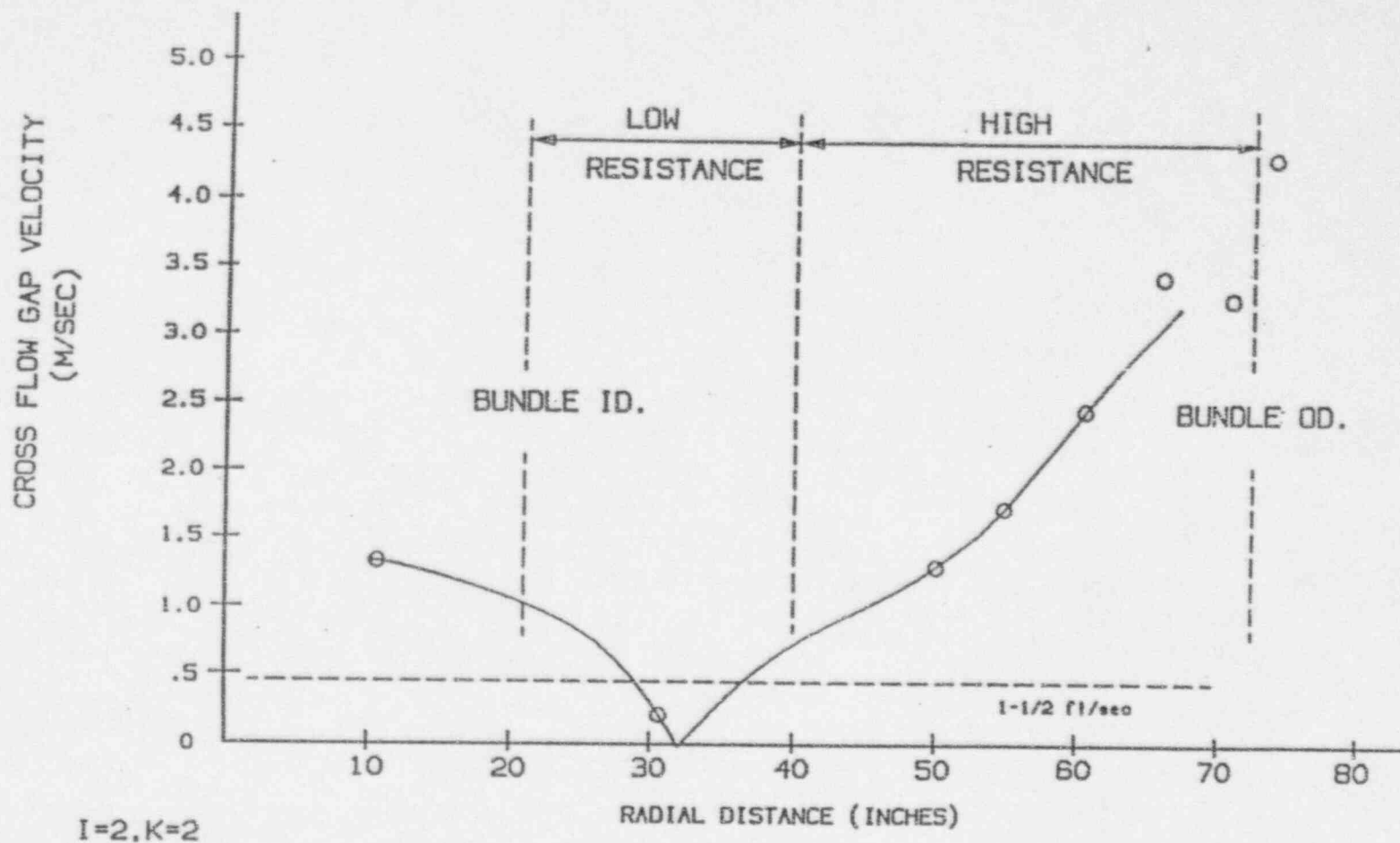
I=2,K=2

I=ANGULAR PLANE
K=CROSS SECTIONAL (ELEVATION) PLANE

TUBESHEET HOT LEG GAP VELOCITY PROFILE

FIGURE 2.5-7

2.6   CONFIRMATORY ANALYSIS AND TESTING

Extensive analysis and testing support the BWI RSG design. This section discusses analysis and test programs that support design, manufacturing and operational aspects of the RSGs.

2.6.1   Flow-Induced Vibration

BWI performs a flow-induced vibration (FIV) analysis in order to confirm that the tube bundle is adequately supported to avoid significant levels of tube vibration. FIV reports will be provided to the owner to verify that the vibration of the RSG internals does not result in excessive wear or fatigue throughout the tube bundle and U-bend regions. This section addresses the FIV methods and evaluations supporting the RSG design. RSG design measures to minimize FIV are discussed in Section 2.2.4.3.

The three pertinent cross-flow FIV mechanisms in the RSG are vortex shedding resonance, random turbulence excitation and fluid elastic instability. The FIV analysis verifies that excessive tube vibration from these sources is avoided. Particular areas of emphasis are the tube bundle entrance and the U-bend region.

Fluid elastic instability is a mechanism which causes the vibration amplitudes to increase sharply when a certain critical flow velocity is exceeded. The acceptance criteria for fluid elastic instability is that the maximum cross-flow velocity at any point in the bundle is less than the critical velocity.

Another FIV excitation mechanism is vortex shedding resonance. When fluid flows across a circular cylinder, the wake behind the cylinder contains vortices. The vortices detach from the cylinder in a regular manner, i.e. at a certain frequency, and cause the tubes to vibrate at the same frequency in a direction perpendicular to the flow direction. When, at a critical cross-flow velocity, the vortex shedding frequency happens to be close to a tube natural frequency, the vibration of the tube can organize the wake, causing it to synchronize (lock in) with the tube motion at the tube natural frequency. This phenomenon is called vortex shedding resonance, and on a plot of amplitude (y) vs. flow velocity (x) it would show up as a hump. For flow in tube bundles, vortex shedding resonance has only been observed to occur at the first few (three or four) outermost tube rows, and is limited to single-phase cross-flows. This means that for a feed ring design RSG, vortex shedding resonance can only occur at the bundle entrance region, which is the region where the downcomer flow enters the base of the tube bundle. A vortex shedding analysis in this region is carried out by first calculating the vortex shedding frequency from the equation

$$f_s = \frac{S*V}{D}$$

where f_s = vortex shedding frequency (Hz)
S = Strouhal number (proportionality constant, determined experimentally)
V = maximum cross-flow velocity
D = tube outside diameter (O.D.)

Subsequently, this frequency is compared to the tube natural frequencies. Vortex shedding resonance is assumed to occur if for any mode of vibration the vortex shedding frequency is within 30% of the tube natural frequency.

For the modes where resonance is predicted, resonant amplitudes of vibration are calculated. The maximum allowable vortex shedding amplitude is 2% of the tube outer diameter.

The third mechanism, random turbulence excitation, is the buffeting of the tubes primarily from the turbulence in the flow, and is the "background" mechanism which accounts for tube vibration below fluid elastic instability (FEI) and outside regions of vortex shedding resonance. It results in relatively low levels of vibration which increase with increasing flow velocity, with amplitudes and mode shapes varying randomly in time and in direction. The maximum allowable amplitude is 10 RMS mils.

A three-dimensional analysis is performed to derive a detailed flow distribution in the U-bend area. From this analysis, velocity and density profiles are determined for each of the five longest tube spans (i.e. longest span with five supports, longest with four supports, etc.). A finite element analysis is used to predict mode shapes for each case for the various mode types and frequencies.

The B&W FIV computer code EasyFIV (Reference 2), developed by B&W Alliance Research Center, and the finite element macro "MSC/pal 2" are used to determine if the FEI threshold velocity is avoided and to analyze response to random turbulent excitation. These analyses are repeated until the optimum number and position of support locations is achieved that conservatively meet the design criteria.

Conservatism of the B&W FIV analysis was demonstrated in two projects with McMaster University. Hot leg bundle entrance region FIV was measured on a full-scale model of a Darlington steam generator tube bundle (with lattice grids). The FIV response of a full-scale model U-bend section, simulating Darlington conditions, was measured. In both cases, the measured FIV responses were below those predicted by B&W.

The potential for fretting is assessed by FIV sensitivity analysis. The FIV analysis is used to confirm that the tube bundle is adequately supported to prevent excessive tube motion due to FIV excitation mechanisms.

The RSG bundle design parameters that are the most important for controlling FIV are:

1.   Tube and support materials.

2.   Tube outside diameter, thickness and pitch/diameter ratios, and diametrical clearance at the lattice bars.

3.   Bundle height.

4.   Radius of outermost tube.

5.   Number of lattice grids.

6.   Number of U-bend supports.

7.   Width of fan bar and high bars.

8.   Steam flow at full power.

9.   Circulation ratio.

These parameters are compared for the RSG and previous BWI steam generators. Typically this comparison shows similarity with existing units and indicates that all regions of the tube bundle are adequately supported to prevent excessive tube motion due to flow induced vibration. This comparison provides a basis on which to conclude that the RSGs will be adequately resistant to FIV.

2.6.2   Thermal-Hydraulic Performance

Verification of the CIRC Computer Code

The computer code CIRC has been the principal design tool for nuclear steam generators at the Babcock & Wilcox International Division in Canada for nearly twenty years. The code is one-dimensional and was written primarily to allow rapid data manipulation and to allow the operator to run a multiplicity of cases of design alternatives to arrive at an optimum design conclusion quickly. The code is capable of analyzing the following alternatives.

1.   Steady-state heat transfer, any power level.

2.   Steady-state circulation, any power level.

3.   Feed ring or integral preheater type steam generators.

4.   Heavy water or light water primary fluid.

The only difference to the CIRC code between heavy water and light water is that the code is directed to use one or the other of two tabulations of property values such as enthalpy, viscosity, etc. as a function of temperature. The secondary side characteristics are identical and vary only in accordance with actual RSG operating pressure.

The CIRC code has been used in performing the thermal hydraulic design of the majority of the BWI steam generators operating in the field today. These include the Pickering B steam generators, the Bruce B steam generators, the 600 Mw units (Point Lepreau 1, Gentilly 2, and Embalse, Cordoba), the Darlington and Cernavoda steam generators, and

Millstone Unit 2. All of the operating steam generators have performed satisfactorily and have seen no adverse performance due to design aspects verified by the CIRC program. Further, operational characteristics determined by the CIRC program and used to calculate velocities for the flow-induced vibration analysis of the tube bundle have been more than adequate as evidenced by no tube failure due to vibration in the operating steam generators to date.

The CIRC code has been used to simulate the operational parameters of many of the existing plants in operation today. In this capacity, the CIRC code uses measured field data and as-built steam generator geometric characteristics to simulate conditions measured in the operating unit. In general, the CIRC code accurately simulated the observed site performance thus confirming its applicability and accuracy as a design tool for new steam generators. The above listed analysis covers the full range of Steam Generator designs with a wide range of geometric and performance parameters expected to bound all current and future applications of the CIRC code.

2.6.2.1 Three-dimensional Thermal Hydraulic Analysis

A 3-D Thermal Hydraulic Analysis is performed to assess the following design features:

1. Flow penetration at the tubesheet face.

2. Steam quality at the tubesheet face.

3. Maximum bundle cross flow gap velocities.

4. Flow expansion at the shroud window exit.

5. Flow distribution in the U-bend region.

6. Primary cyclone entrance quality distribution.

7. Overall heat transfer rate and circulation ratio.

Design features 1 and 2 above are critical for estimating sludge pile size and rate of growth. As a general rule, the sludge pile is expected to form in regions of net steam quality and low cross flow velocities at the tubesheet face. As expected, previous analyses indicate that these two regions are located at approximately the same location.

Design features 3 to 5 are important for assessing the ability of the design to preclude excessive tube motion due to flow-induced vibration. Feature 5 is also used to reduce flow resistance in the U-bend region by aligning the U-bend supports with the U-bend flow streams.

Design feature 6 is used to ensure adequate loading of the primary steam separation equipment.

Design feature 7 is used as additional confirmation of the heat transfer analysis results obtained from the single-dimension thermal hydraulic (CIRC code) analysis.

2.6.2.2 The ATHOS Computer Code

The ATHOS (Analysis of Thermal-Hydraulics Of Steam Generators) computer program is used to analyze the thermal-hydraulics and the three dimensional flow distribution in steam generators at steady state conditions. The RSG riser is modeled in "volume elements" and the "finite differences" method is used to solve the equations of mass, momentum and energy conservation. Extensive documentation is available for ATHOS and its applications.

The flow field in the tube bundle is represented as a three dimensional model. To deal with two phase flow it is possible to employ either the homogeneous flow model or a model with slip in the steam generator axial direction according to a drift flux model. The homogenous flow model is used. ATHOS uses "distributed resistances" to simulate pressure losses in the steam generator, particularly the tube bend region. The free flow area in the tube bundle is calculated with the aid of "porosity". The pressure losses across the differential resistance lattice grid, the typical grids, the steam generator downcomer and the moisture separators are treated as locally concentrated resistances and assigned a pressure loss coefficient. Refer to Section 2.2.4.1 of this report for a description of both typical and differential lattice grid tube supports.

The velocity field calculated with ATHOS shows the asymmetry between the hot and cold legs of the riser due to dissimilar steam quality. Asymmetric profiles are illustrated in Figure 2.5-5. In keeping with the homogenous two-phase model, the velocities presented are mixture velocities (equal water and steam velocities). The reference area for velocity is based on the volume porosities of the respective cells. The radial flow displacement in the tube bend region is correctly simulated by the program insofar as the pressure loss in each cell is considered as appropriate to the direction of incidence on the tube (axial flow, cross flow or a combination of both). The velocities corresponding to the volume porosity provided by ATHOS give no information on the velocities in the narrowest gaps between adjacent tubes. However, these gap velocities are crucial to vibration excitation. For this reason, the gap velocities parallel and normal to the tube axis (axial flow gap velocity and cross flow gap velocity, respectively) are determined in a post-processing conversion routine.

In addition to the geometric input data required by the ATHOS pre-processor programs, ATHOS input includes process data for various load step conditions, pressure loss coefficients, information on the type of flow model and on the sections for which flow parameters are required, initial values and convergence criteria. ATHOS output includes a summary of the input data as well as a summary of the calculated thermal-hydraulic operating data.

Much of the thermal-hydraulic sizing of the steam generator is performed using classical analysis techniques and one-dimensional thermal-hydraulic code analysis. Three-dimensional analysis results are utilized in very specific areas that require detailed knowledge of the various flow parameters. In addition the heat transfer results from the

three-dimensional thermal-hydraulic analysis are used for additional confidence in the results obtained from the one-dimensional CIRC code results.

An RSG area requiring details for the three-dimensional analysis is the tubesheet secondary face region. This is the most common location for sludge deposition. The three-dimensional analysis evaluates the effects of various bundle entrance geometries and differential resistance lower tube supports. It determines the flow distribution and can determine areas of low cross-flow and potential stagnation. This allows minimization of low cross-flow and stagnant areas. The deposition of sludge is affected by the amount of boiling at or near the tubesheet secondary face. Sludge deposition can occur at these locations due to the inability of the steam phase to transport corrosion products that drop out of solution and form sludge piles. By minimizing the areas of high net quality and low cross-flow at the tubesheet secondary face, the size and rate of growth of the sludge pile can be minimized.

From the three-dimensional analysis, flow velocities on the outermost tubes at the bundle entrance region, flow imbalances at the periphery of the bundle, and variations in vertical velocity at the bundle entrance can be determined. These are important to the ability of the design to preclude FIV.

As the flow rises through the tube bundle and steam quality increases, interaction of the hot leg and cold leg produce some flow imbalance. The U-bend tube support system, and flow resistance of the tube bundle in cross-flow tend to create flow imbalances as well. The three-dimensional thermal-hydraulic analysis aids the design process in two ways. First, the position of the U-bend supports can be optimized so there is minimal effect on flow distribution. This reduces the pressure drop in the U-bend region and, by aligning U-bend supports with U-bend flow streams, additional flow imbalance is avoided. Secondly the three-dimensional analysis identifies flow imbalance in the U-bend region of the final design. The flow and local quality distribution (and therefore density), determine the U-bend gap velocities. These are used in FIV analysis of the U-bend tubes. The flow forcing function distribution is used to quantitatively confirm that the design has no tendency for excessive FIV excitation.

Directly above the U-bend region is the primary separator deck. The flow leaving the U-bend region has a steam quality distribution and a mass flow distribution. This produces variation in steam separator component loading. The three-dimensional thermal-hydraulic analysis of quality- and mass distribution determines the range of expected water- and steam loadings, and circulation ratios. BWI uses this information to determine the appropriate ranges for confirmatory testing of steam separation equipment.

2.6.3 Tube-to-Tubesheet Joint Qualification Program

The function of the tube-to-tubesheet joint is to fasten the tube to the tubesheet such that it sustains the forces associated with pressure and thermal transients while not creating conditions which could cause the tube to degrade in service through crevice corrosion or stress corrosion mechanisms. Industry wide problems associated with this region of the steam generator include secondary stress corrosion cracking (OD SCC), intergranular attack

(IGA) in the crevice and ID SCC in the expanded region. BWI steam generators have to date suffered no tube failure due to tube corrosion at this location. Several key factors have contributed to this success:

1. Water Chemistry Control.

2. Materials Selection.

3. Design Optimization.

Primary and secondary water chemistry are addressed in Sections 2.7.2 and 2.7.3. Materials selection for corrosion resistance is addressed in Section 2.5.1. The design integrity is evaluated by BWI's tube-to-tubesheet joint qualification program.

Based on growing knowledge and understanding of the failure mechanisms associated with industry problems with the joint, BWI has developed design parameters and fabrication techniques that are directed at minimizing potential for joint problems.

Expanded Joint Integrity Requirements: An assessment of the integrity of expanded tube-to-tubesheet joints considers the following:

1. Pullout Strength

    Pullout strength is the force required to pull the expanded tube through the full thickness of a tubesheet. Pullout may consider the unflexed tubesheet and/or the integrated effect of pullout resistance over the full thickness fully flexed tubesheet. This relates to the ability of the tube to structurally resist pullout without benefit of the seal weld.

2. Leak Tightness

    Leak tightness is the ability of the expanded joint to resist net leakage of fluid through the full thickness of tubesheet. Leakage may consider the unflexed tubesheet and/or the integrated effect of the full thickness/fully flexed tubesheet.

    This relates to the ability of the tube to resist primary to secondary side leakage without benefit of the seal weld.

3. Secondary Fluid Ingress Resistance

    Secondary fluid ingress resistance is the ability of the tube to resist ingress/weeping of secondary fluid into the crevice. Ingress resistance and the depth of ingress will be affected by dilation of the tube holes due to tubesheet flexing as well as by the basic parameters of the expanded joint.

This relates to the minimization of corrosion within the crevice area due to contaminants in the secondary side fluids.

4.   Control of Residual Stresses

Surface residual tensile stresses are a by-product of the hydraulic expansion process that creates the joint by yielding the tube plastically into the elastically yielded tubesheet hole. High tensile stresses on the outside diameter leave the tube vulnerable to failure through in-service stress corrosion cracking.

This relates to the ability of the tube to obtain design longevity.

BWI fabrication processes and technologies applied to the joints address these design objectives and result in:

1.   Full-depth expansion through tubesheet to eliminate crevices.

2.   High SCC resistance to both primary and secondary water through selection of I-690 material.

3.   Minimization of microscopic crevice.

4.   Hydraulic expansion to minimize and accurately control residual stresses.

5.   Full-depth expansion to minimize strains, maximize pullout strength and leak tightness.

6.   Stringent QA standards and fabrication controls assure that design requirements are met.

The tube-to-tubesheet joint qualification program consists of studies, evaluations, and testing that consider the following:

1.   Measurement of residual tensile surface stresses on the outside diameter of the transition area (and resulting SCC susceptibility). Because of the uncertainties inherent in this type of measurement, independent techniques are applied:

    a.   An X-ray diffraction residual stress survey of the transition region.

    b.   Finite element predictions of the stress state in the transition region.

    c.   A study of SCC susceptibility of the joint transition, seal welds, and scratched tubes.

The residual stress experiments are complemented by existing analytical data on the BWI tube-to-tubesheet joint.

2.   An evaluation of the degradation of the structural (pullout) load capacity and leak tightness and secondary fluid ingress as a result of experimentally simulated plant heat-up and cool-down transients.

2.6.4 Separator Testing Experience

The performance of the primary and secondary steam separators has been extensively evaluated at the B&W Alliance Research Center. The following paragraphs discuss performance results for the BWI steam separation equipment, including the test facility used, instrumentation, and results. Testing experience shows the RSG steam separation equipment to be effective and relatively insensitive to variations in operating pressure, water flow, water level, and steam carryunder. Additional information on steam separator design and performance is found in Section 2.2.7.

At typical operating conditions, the moisture carryover was shown to remain below the specified design value of 0.25 percent by weight. Testing at flows ranging from 15 to 30% above the design steam flow showed that the design value moisture content was achieved.

The Darlington steam generators incorporate the same replacement steam generators except that the secondary separators are not enclosed in separate isolated compartments. The enclosures of the secondary separators ensures a greater compatibility between the full scale prototype test configuration where a single pair of primary and secondary separators are tested and the actual in-service conditions. Also by providing isolated secondary separators potential performance deficiencies due to flow maldistribution between the separators is avoided. The Millstone 2 RSGs have a isolated secondary separators and have demonstrated moisture carryover well below 0.1 percent by weight.

Steam carryunder describes steam which is carried downward with the downcomer flow. This reduces downcomer flow density and the available driving head for steam generator circulation. Carryunder can also raise downcomer temperature, reducing steam generator performance. Tests have shown the BWI steam separators to produce no appreciable carryunder at normal operating conditions.

BWI performs steam separator testing at its facilities located in Alliance, Ohio. BWI separator development capabilities rely heavily on experimental research utilizing state-of-the-art measurement, diagnostic, and analysis equipment supplemented by advanced analytical modelling techniques using mainframe and microcomputer hardware. Experimental research includes both quantitative and qualitative assessments of separators using both an Air/Water Test Facility and a Steam Test Facility.

The steam test facility is designed for a pressure of 1000 psia and a temperature of 544°F at steam flows up to 58,000 lbm/hr. Performance parameters such as pressure drop, moisture carryover, water level, liquid film height, steam carryunder, temperatures, and secondary cyclone drain flow are measured. Technology available for testing includes a gamma densitometer for determining local density and optical techniques for obtaining high-speed videos in the steam/water environment.

Figure 2.6-1 illustrates the steam flow and water flow capacity curve for a BWI separator pair relative to a typical operating curve for a BWI steam generator. There are substantial steam and water flow margins beyond normal operating conditions before the moisture carryover limit of 0.25% by weight is reached.

2.6.5   Chemical Cleaning Qualification of Materials

BWI qualifies RSG materials for corrosion resistance considering the effects of normal operation and multiple chemical cleanings. This is done through a program of testing and assessments of the various RSG materials. This program is documented in the Chemical Cleaning Qualification Report. Key elements of the program include:

1.   Laboratory screening tests of the tube (Alloy 690) and tube support material (SA240TY 410S) against materials already qualified in industry chemical cleaning database (i.e. Alloy 690, carbon steel, and a SMAW electrode).

2.   Assessment of the steam generator to identify materials, welds and/or joints.

3.   Laboratory testing of materials, welds and/or joints not qualified within the BWI Chemical Cleaning Database, but identified in the RSG material survey.

4.   An estimate of expected corrosion losses for the RSG materials during multiple chemical cleanings, as well as during normal operation.

5.   Specification of corrosion allowance design values based on the estimates of expected corrosion.

Reference for Section 2.6

1.   "Verification of the ATHOS3 Code Against Feedring and Preheater Steam Generator Test Data", EPRI report NP-5728, Project 1066-10, May 1988.

2.   Flow Induced Vibration Analysis Program EasyFIV Rev. 0 Verification Package.

B&W Separator Performance
at 880 psia
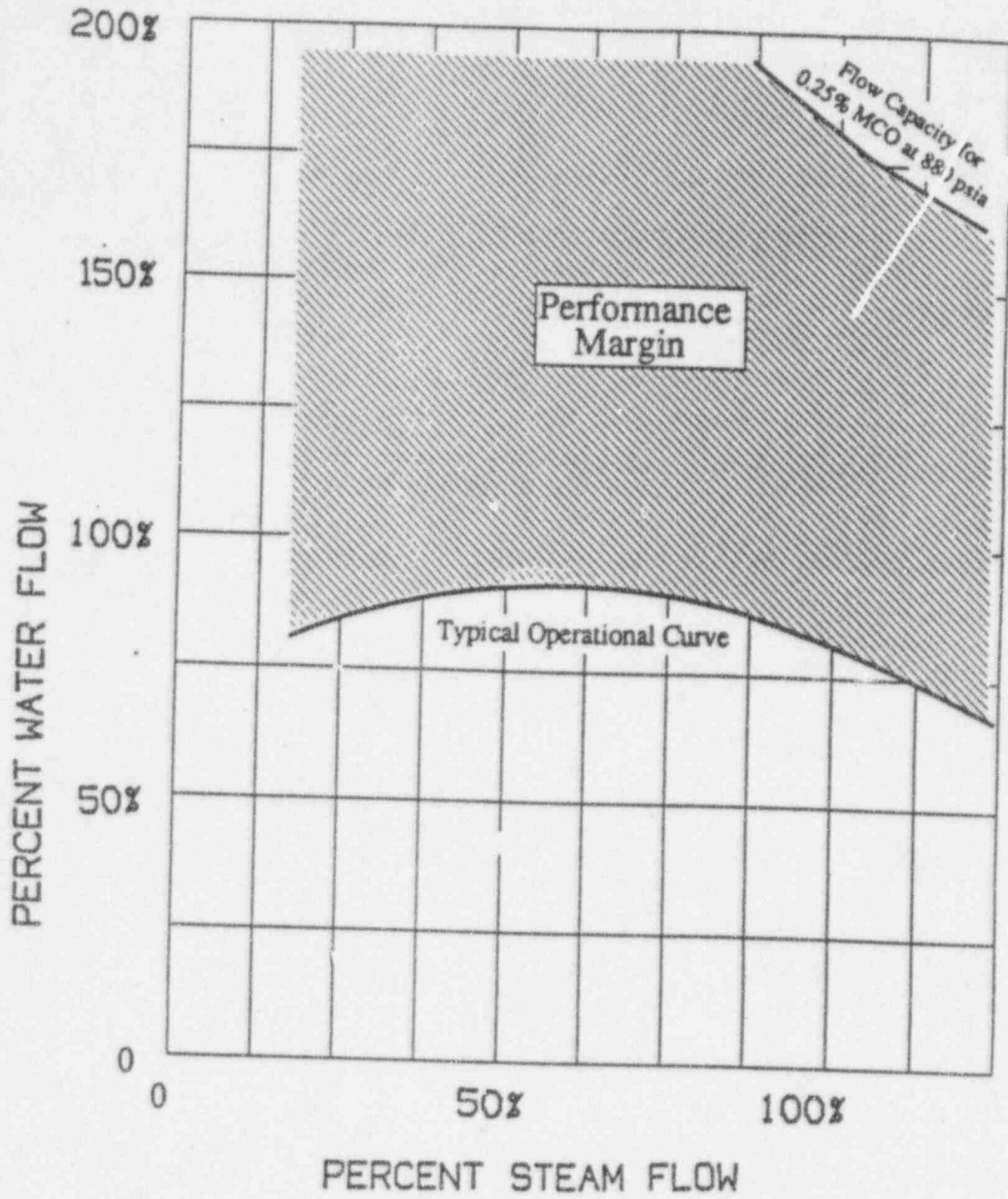
PERFORMANCE MARGIN FOR B&W SEPARATOR EQUIPMENT
AT SATURATION PRESSURE OF 880 PSIA

FIGURE 2.6-1

2.7   OPERATING RESTRICTIONS WITH RSG DESIGN

2.7.1   Removal of Temporary U-Bend Shipping Restraints

Temporary restraints are applied to the RSG tube U-bends to prevent shipping damage. These must be removed prior to RSG operation. Removal requires RSG secondary side entry. The temporary supports are accessed through the steam separator deck.

2.7.2   Primary Water Chemistry

There have been relatively few corrosion problems associated with the primary system chemistry environment. In recent years, however, a phenomenon called primary water stress corrosion cracking (PWSCC) has been observed on the primary side of certain RSG Alloy 600 tubing. However, this corrosion phenomenon has less to do with the primary water chemistry environment than the metallurgical condition and stress levels of the tubing. This phenomenon is most prevalent in Alloy 600 tubing having high residual tensile stresses, e.g. U-bends and expansion and bending transitions. In addition, this phenomenon occurs most frequently in tubing that has been mill-annealed at relatively low temperatures (1750F). Thermally treated or stress-relieved Alloy 600 tubing has been proven to be far less susceptible to PWSCC.

All U.S. PWR plants use boric acid for chemical shim reactivity control and lithium hydroxide to raise pH. Each nuclear fuel vendor normally provides the owner with guidelines relative to the proper concentrations of these chemicals, since the boric acid concentration must be reduced as burnup of the fuel progresses. EPRI has developed and updated guidelines for PWR Primary Water chemistry (Reference 1). These guidelines minimize the impact of the boric acid and lithium hydroxide on primary system materials and fuel cladding. Reference 1 presents principles for each plant to use in developing its own boron/lithium hydroxide control scheme. From this information, the owner can develop the optimum primary system chemistry control scheme in consultation with their fuel vendor.

Other parameters that are controlled by the EPRI guidelines include chloride, fluoride, dissolved hydrogen, and dissolved oxygen. Diagnostic parameters include sulfate and ⚠4 suspended solids. A water chemistry control program which covers all modes of operation is documented in the station chemistry manuals.

2.7.3   Secondary Water Chemistry

The water chemistry requirements for the secondary system are dependent upon the operational mode of the plant, as well as system materials. The operational modes that require environmental control are cold shutdown, heatup/startup/hot standby, and normal power operation. A water chemistry control program which covers all modes of operation is documented in the station chemistry manuals.

References for Section 2.7

1.   NP-7077, "PWR Primary Water Chemistry Guidelines: Rev. 2", Electric Power Research Institute, November, 1990.

2.   NP-6239, "PWR Secondary Water Chemistry Guidelines: Rev. 2", Electric Power Research Institute, December, 1988.

2.8 RSG STRUCTURAL EVALUATION

Structural and seismic evaluation of RSG primary and secondary side pressure boundaries demonstrate that these components satisfy ASME III, Division 1, Class 1 design requirements for service levels A, B, C, and D (normal, upset, emergency and faulted conditions, respectively). Steam generator internal components are not governed by the ASME Boiler & Vessel Code . However, ASME III Subsection NB for Class 1 components is used as a guide for structural analysis of RSG internal components. RSG internal components are required to withstand all specified loadings to maintain heat transfer capability during and following a design basis earthquake. This helps to ensure that safe shutdown capability is maintained. The RSG structural evaluation will be documented in a Code Stress Report.

Conservative hand calculations and finite element modelling (where required for pressure and thermal transients) are employed to prove that the components examined meet the ASME Code allowable stresses. For seismic loading, an equivalent static load analysis is performed to determine seismic loads on components for subsequent stress analysis. The design and hydrotest primary stresses in the RSGs meet the design and hydrotest allowables of ASME III as shown in the following sections.

The requirements of Subsection NB-3221 for design stresses are met as follows:

$$P_m \quad < S_m \text{ at design temperature}$$

$$P_l \quad < 1.5 S_m \text{ at design temperature}$$

$$P_l + P_b \quad < 1.5 S_m \text{ at design temperature}$$

where:

P_m = General primary membrane stress

P_l = Local primary membrane stress

P_b = Primary bending stress

S_m = Design stress intensity value

S_y = Yield strength

S_u = Ultimate strength

The criteria for normal and upset loads are the ASME level A & B allowables for the range of primary plus secondary stress. The requirements of Subsection NB-3222 and NB-3223 are met as follows:

$$\text{Range of } (P_m + P_b + Q) < 3 S_m \text{ @ operating temperature}$$

where: Q = Secondary Stress

For pressure boundary components and tubing, it is also shown that the cumulative fatigue usage factor remains below 1.0 for all Level A, B, and test condition operating cycles.

The criterion for level C loading conditions is to maintain integrity of tube, tube supports (lattice grid) and steam drum internals for emergency conditions of level C. The requirements of Subsection NB-3224 are met as follows:

$$P_m \qquad\qquad < \text{ greater of } (1.2 \, S_u \text{ or } S_y) \text{ at operating temperature}$$

$$P_m + P_b \qquad < \text{ greater of } (1.8 \, S_u \text{ or } 1.5 \, S_y) \text{ at operating temperature}$$

The criterion for level D loading condition (combined main steam line break and design basis earthquake) is to ensure tube integrity by proving that tube rupture and leakage cannot occur. The requirements of Subsection NB-3225 are met as follows:

$$P_m \qquad\qquad < \text{ lesser of } (2.4 S_m \text{ or } 0.7 S_u) \text{ at operating temperature}$$

$$P_m + P_b \qquad < \text{ lesser of } (3.6 S_m \text{ or } 1.05 S_u) \text{ at operating temperature}$$

The requirements of Subsection NB-3226 for hydrotest are met as follows:

$$
\begin{aligned}
\text{for} \quad & P_m && < 0.67 S_y \text{ at test temperature} \\
& P_m + P_b && < 1.35 S_y \text{ at test temperature}
\end{aligned}
$$

$$
\begin{aligned}
\text{for} \quad & P_m && < 0.9 S_y \text{ at test temp.} \\
& P_m + P_b && < (2.15 S_y - 1.2 P_m) \text{ at test temperature}
\end{aligned}
$$

2.8.1 Tubing

The structural analysis demonstrates that for an instantaneous full rupture of the steam line downstream of the steam outlet nozzle occurring during normal full power operation, the tube integrity is maintained. The structural evaluation of the tubing for level D is in accordance with the ASME Boiler and Pressure Vessel Code Section III requirements as explained in this section. Furthermore, the tube material selection and size exceed the strength requirements of the existing steam generators.

Comparison of the RSG Alloy 690 used in the RSGs with the typical Alloy 600 tube material shows that the RSG material strength characteristics are as good or better than those of the existing design.

The existing steam generator has a nominal tube wall thickness of 0.043 inches, compared with the RSG nominal of 0.040 inches. The ASME III allowable stresses (in ksi) based on S_m, S_y, and S_u for Alloy 600 and 690 are:

Temperature	Alloy 600			Alloy 690		
	S_m	S_y	S_u	S_m	S_y	S_u
70°F	23.3	35.0	80.0	26.6	40.0	85.0 (as ordered)
650°F	23.3	27.4	80.0	26.6	35.2	85.0 (as ordered)

where:

$$S_u \quad = \quad \text{Ultimate Strength}$$

Pressure-induced stress in the tubing is calculated as $\sigma_h = PR_i/t$.

where:

$$\sigma_h \quad = \quad \text{Hoop stress}$$
$$t \quad = \quad \text{Tube wall thickness}$$
$$P \quad = \quad \text{Pressure}$$
$$R_i \quad = \quad \text{Tube inside radius}$$

A stress margin, defined as the ratio of the ASME III allowable stress to the actual pressure-induced stress is expressed as S/σ_h.
where:

$$S \quad = \quad \text{Allowable stress}$$

A comparison of the stress margins of the RSG and OSG is expressed as a stress margin ratio defined as:

$$\text{Stress Margin Ratio} \quad = \quad \frac{(S/\sigma_h)_{(RSG)}}{(S/\sigma_h)_{(RSG)}}$$

$$= \quad \frac{t_{(RSG)} \times S_{(RSG)} \times R_{i\,(OSG)}}{t_{(OSG)} \times S_{(OSG)} \times R_{i\,(RSG)}}$$

A stress margin value of 1.0 or greater indicates that the RSG tubing has strength

characteristics with respect to pressure-induced stresses that are equal to or better than those of the OSG.

For S_m @ 70°F or 650°F:

$$\text{Stress Margin Ratio} = \frac{0.040}{0.043} \times \frac{26.6}{23.3} \times \frac{0.332}{0.304} = 1.16$$

For S_y @ 70°F:

$$\text{Stress Margin Ratio} = \frac{0.040}{0.043} \times \frac{40.0}{35.0} \times \frac{0.332}{0.304} = 1.16$$

For S_y @ 650°F:

$$\text{Stress Margin Ratio} = \frac{0.040}{0.043} \times \frac{35.2}{27.4} \times \frac{0.332}{0.304} = 1.31$$

For S_u @ 70°F or 650°F:

$$\text{Stress Margin Ratio} = \frac{0.040}{0.043} \times \frac{85}{80} \times \frac{0.332}{0.304} = 1.08$$

2.9   STARTUP TESTING REQUIREMENTS

After installation, tests will be performed to verify that the replacement steam generators comply with the requirements of the equipment specification and are capable of satisfactorily performing their intended function. These tests form the basis for determining compliance with the terms of the replacement steam generator contract performance requirements.

The following parameters are to be verified by start-up testing:

1.   Thermal and Hydraulic Performance.

2.   Moisture Carryover Testing.

3.   Reactor Coolant System Flow Rate and Pressure Drop Measurements.

4.   Primary to Secondary Leakage.

Only minor differences are expected in shrink and swell of the RSG as compared to the OSG. Therefore, the effect on the feedwater control system will be minor and no specific tests, beyond those required to verify warranted performance values, are required. DPC will also perform additional tests as required by the ASME Code Section XI and to monitor startup performance.

3.   REPLACEMENT STEAM GENERATOR FABRICATION

3.1   QUALITY ASSURANCE PROGRAM

This section describes the BWI Quality Assurance (QA) program and controls applied during RSG design and construction. The BWI QA program is implemented by the "Quality Assurance Manual for Nuclear Products" and by supporting procedures and instructions that govern the design and construction of nuclear steam generators and other components. The program conforms to the requirements of ASME Section III; the applicable sections of ASME NQA-1 invoked by Section III; 10CFR50, Appendix B; 10CFR21 and other international codes and standards. BWI holds ASME certificates of authorization for N, NA and NPT symbol stamps. BWI obtained ASME certification initially in 1986 and has successfully passed ASME surveys in 1988 and 1991.

3.1.1   Design Control

BWI has established measures to ensure that applicable code and regulatory requirements and the owners' design specifications are correctly translated into BWI design documents (design analysis, design reports, drawings, etc.) and that the design documents are verified against the design specifications.

Design control measures include:

1.   Assignment of a cognizant project engineer responsible for coordinating design document preparation.

2.   Review for suitability and application of design methods, materials, parts, equipment and processes essential to RSG safety and performance functions.

3.   Planned and controlled design analysis, using appropriate analysis plans, outlines or drawings; material specifications and review schedules, based on the complexity of the analysis and previous experience.

4.   Legible documentation suitable for reproduction, filing, and retrieval, covering design analysis and required ASME Code design reports. Typical design report contents include:

        a.   definition of objective, design inputs and their sources

        b.   results of literature searches and other background data

        c.   identification of assumptions and indication of those that must be verified as the design proceeds

        d.   the main body section of calculations including Acceptance Criteria, Loads, Material Properties, Boundary Conditions, Model Description, Computer

input, Analysis and Results as appropriate for the calculation.

e. identification of any computer type, computer program name, evidence of/or reference to computer program verification and the basis supporting the application of the computer program to the specific physical problem.

f. review and approval by personnel other than the preparer.

g. Design verification by a qualified engineer not responsible for the design. Verification may be by means of design review, alternate calculations or qualification tests.

h. Formal documented design reviews of first-of-a-kind features or features that are major extrapolations of BWI designs. These are conducted by experienced BWI or outside engineers not involved in the design process.

i. Control of changes to design documents by the same controls used on the original documents, including necessary reviews and approvals. Design changes are controlled by means of revisions to the original documents.

The as-built conditions are formally reconciled with the ASME Code Design Report by the project engineer.

j. Identification and control of design interfaces in accordance with documented procedures. Information transmitted across an interface is controlled and documented with regard to the information transmitted and its status.

3.1.2 Document Control

Documents that specify quality requirements or describe activities affecting quality (such as QA program procedures, inspection and non-destructive examination procedures, inspection & test plans, manufacturing procedures, welding & heat treatment and other special process procedures, material ordering standards, drawings) are issued under a formal Document Control system which ensures that all documents and revisions are reviewed for adequacy and released by authorized personnel prior to use.

3.1.3 Corrective Action

Conditions adverse to quality detected during audits, inspections or other activities are addressed under a formal corrective action program. The program requires for significant conditions that the cause of the condition be determined and corrective action be taken to preclude recurrence.

The program documents the responsibilities for initiation, evaluation and acceptance of corrective actions. It establishes time limits for these activities to ensure timely corrective action.

Periodic reports are provided to BWI senior management documenting each condition adverse to quality, its cause and corrective action(s) taken.

3.1.4 Non-conforming Items

A program has been established to detect non-conformances to drawings and specifications to prevent unauthorized use or shipment. Non-conforming items are segregated from the normal production flow. Further processing of non-conforming items is controlled, pending evaluation and disposition by authorized personnel.

The program defines the responsibility and authority of personnel responsible for the disposition of non-conformances. Design Engineering, Manufacturing Engineering and Quality Assurance are involved in the disposition of all non-conformances. Supplier non-conformances are handled under the same program.

3.1.5 QA Records

QA Records required by the applicable codes and standards, owner's specification or for other reasons are generated, supplied and maintained under a formal records program. The program ensures that records are legible, accurate, accepted by authorized personnel, identifiable to the item or activity to which they apply and retrievable.

For each contract a Documentation Checklist identifies the records required, their classification, the personnel responsible for obtaining and storing them, and record retention and distribution requirements.

3.1.6 Audits

Planned and periodic audits are conducted to verify compliance to and the effectiveness of the QA program. Audits are conducted by QA personnel qualified to the requirements of ASME NQA-1. Audits are performed in accordance with written procedures and checklists by personnel not having direct responsibilities for the areas being audited. Results are documented in audit reports that define audit scope, identify auditors, identify persons contacted, summarize audit results, assess program element effectiveness, and describe findings. Audit reports are forwarded to management personnel including the Division General Manager. The QA Manager is responsible for the initiation of corrective action requests and other actions based on the audit findings.

Follow-up action including re-audit of deficient areas is performed as appropriate. Resolution of audit report findings are documented.

3.2   FABRICATION CONTROL

This section describes the measures used to control fabrication of the RSGs from purchase of materials through shipping to the owner.

3.2.1   Control of Purchased Items and Services

Procurement documents (material ordering standards, purchase orders, etc.) are prepared under a controlled program which ensures that the requirements of the design basis documents, specification requirements and applicable codes and standards are included and that procurement documents and revisions are reviewed and accepted by cognizant personnel before use.

Purchased items and services are procured from vendors who meet applicable quality program standards. Vendor evaluation is performed by the QA Vendor Control Group based on reviews of vendor quality program manuals and audits of program implementation. An "Approved Supplier List" is published and issued as a controlled document. It lists acceptable vendors and their approval status.

Vendor performance is verified by a combination of surveillance, source inspection and incoming inspection. The methods used to verify performance are selected and documented on BWI and vendor inspection and test plans. The vendor is required to submit inspection and test plans, and manufacturing, inspection and test procedures for review and approval by cognizant BWI personnel.

A historical file is maintained for each vendor. It contains survey and audit reports, source inspection reports, non-conformance reports and other documentation relative to the vendor's items and services. Vendor performance is assessed at least annually and approved supplier status revoked if quality is not acceptable.

BWI has implemented a Commercial Grade Dedication procedure to allow materials for safety related components to be procured without imposing 10CFR21 and 10CFR50, Appendix B. This procedure is modelled after EPRI guideline NP-5652. In addition, procedures indicate that for safety related items, either 10CFR21 be imposed or commercial grade dedication be performed.

All items and services for steam generator construction are subject to receipt inspection for conformance to procurement documents. Acceptance is documented to ensure that only conforming items are used for construction.

3.2.2   Control of Manufacturing Processes

Control of quality-related manufacturing processes to ensure performance in accordance with documented procedures, instructions and drawings is achieved through a shop traveller (Route Sheet) system. The Route Sheet controls and documents the status of shop operations and performs the following functions:

1.   Lists the sequence of operations.

2.   Describes each operation.

3.   Identifies drawings and procedures & instructions to be followed with revision levels.

4.   Provides space for indicating inspection, witness, documentation and hold points.

5.   Provides space for sign-off of completion of fabrication operations and inspection points.

6.   Documents the fabrication history of the product.

Input for the preparation of Route Sheets is obtained from Fabrication Outlines, Inspection & Test Plans, drawings and lists of weld procedures prepared by cognizant engineers. Route Sheets and revisions are reviewed by cognizant engineers before issue.

Special processes such as welding, non-destructive examination and heat treatment are performed in accordance with documented procedures developed by technical specialists. Procedures and personnel are qualified as required by applicable codes and standards.

3.2.3 Control of Consumables

Consumable products are nonmetallic, non-permanent products which come into contact with the RSGs during manufacture, inspection or testing. Because these products may contain materials that could be detrimental to the RSGs, the use of consumable products is controlled.

Limits are placed on the amounts of certain materials that may be present in consumable materials. Consumable materials include:

1.   Cleaning solvents and agents.

2.   Non-destructive testing compounds and agents.

3.   Adhesives and adhesive tapes.

4.   Insulation and refractory materials.

5.   Cutting, drilling and tapping compounds.

6.   Other consumables that are capable of transferring detrimental materials to nuclear hardware.

All consumable products are legibly labelled with the product and manufacturer. Three levels of control are established for consumable materials:

1.   Prohibited Products - These are materials that are not allowed to contact nuclear hardware, because they contain elements and/or compounds known to be detrimental. Typical examples of prohibited materials are:

   a.   Lead and lead-based alloys.

   b.   Copper and alloys containing more than 50% copper.

   c.   High sulphur compounds, especially molybdenum disulphide.

   d.   Alloys based on, or containing significant amounts of, cadmium, mercury, arsenic, zinc, antimony, bismuth and tin.

   e.   Halogenated solvent, aerosol propellants or similar highly halogenated compounds.

2.   Acceptable Products - shown to contain low levels of elements known to have deleterious effects on nuclear materials, especially nickel-based alloys and stainless steels. These elements include chlorides, fluorides, sulphates, mercury, lead, antimony, bismuth, copper, zinc, tin, arsenic and cadmium.

   Records are kept on all acceptable products. These include the manufacturer's specifications, certificates of analysis, identification of low melting point constituents (where applicable) and special use restrictions (e.g. "must be removed if temperature exceeds 200F"). These records also identify any special cleaning procedures that may be required to remove the material.

   Only items on the "Acceptable Products List" may be used in contact with corrosion resistant materials (nickel-based alloys and stainless steels) during RSG assembly. Only items on this list are allowed contact with final cleaned surfaces, or during processes involving elevated temperatures (welding or heat treatment).

3.   Controlled Products - products that contain (or may contain) potentially detrimental materials in excess of the amounts allowed in the Acceptable Products List. Use of controlled products is restricted to applications where there is either no transfer of the potentially detrimental material to nuclear materials, or where the potentially detrimental material can be removed and the surface condition can be verified. An example of this latter condition is use of high sulphur cutting fluid. After machining, the fluid is removed and verified to have been removed. Controlled products are discussed in Section 3.2.5.

3.2.4   Control of Specialized Processes

The following paragraphs describe control of specialized RSG manufacturing processes.

3.2.4.1 Tube-to-Tubesheet Welding

Each heat of welding wire is tested for weldability before use. Each operator makes a test tube-to-tubesheet weld each shift. These are sectioned and examined to ensure a satisfactory weld has been made. Should a test weld prove unsatisfactory, welding is halted and all welds made by the operator prior to the stop are subject to a non-conformance report for evaluation.

The origin of the problem is determined and corrected, and another test weld is made by the operator before he resumes welding. The second test weld is examined in the same manner as the first. Sectioning and polishing equipment, and a metallurgical microscope are dedicated to this examination in the clean room.

Each completed tube-to-tubesheet weld is visually and dye penetrant examined.

3.2.4.2 Hydraulic Tube Expansion

Close control of the hydraulic expansion process is maintained throughout the operation. Detailed instructions are prepared and operators are trained to use the process before working on the steam generator. Quality control checks are made on all critical parameters, including:

1. Pressure measured at the expansion mandrel.

2. Time of applied pressure.

3. Position of the hydraulic seal at the secondary face of the tubesheet.

4. Verification that all tubes have been expanded.

3.2.4.3 Electro-polishing

The electro-polishing (EP) process, used to improve channel head surfaces, is qualified by performing the process steps on sample specimens and microscopically evaluating the resulting surface finish. Qualification ensures that the entire EP process does not degrade the RSG primary side surfaces. The surfaces polished include primary head, stay cylinder and nozzle stainless steel weld overlay, and stainless steel divider plate material. Because of the complex geometry, the tubesheet cladding and tube-to-tubesheet welds are not electropolished.

Qualification involves electro-chemical polishing of samples representing the primary side surface materials, using the electrolyte, polishing equipment and electrical parameters proposed for the RSG. Procedures define prerequisite operations, precautions to be taken, and the mechanical polishing parameters to prepare the surface. They list materials and chemicals that can be used, specify cladding thickness requirements, and specify cleanliness requirements. During qualification, measurements of the treated surface are made. These include scanning electron microscopy to characterize surface profilometry, amount of

cladding removal, and dye penetrant testing to detect excessive metal removal or surface finish problems prior to production EP. These examinations ensure that the EP process will not result in any degradation of the primary side surfaces.

3.2.5 Material Control

Measures are established to identify and control materials, parts, and components to ensure that only the correct materials are used and that proper records are maintained from initial receipt of the material through shipment of the finished component. Identification is maintained either on the component or on documentation traceable to the component. BWI verifies material identification prior to shipment under the material control system.

As required by the ASME Code or owner's Design Specification, material control measures ensure that materials are traceable by heat and lot number, or by other appropriate means, to the Material Test Reports.

3.2.6 SHOP TESTS AND INSPECTIONS

The following paragraphs summarize shop tests and inspection requirements applied to key RSG components and tooling.

3.2.6.1 Test and Inspection Equipment

Tools, gauges and other measuring and test equipment used for activities affecting quality are controlled to assure their calibration and adjustment to maintain accuracy within acceptable limits. Measuring and test equipment is calibrated by comparison to certified standards which are traceable to National Standards. Equipment found to be out of calibration tolerances are physically segregated until repairs are made. Equipment beyond repair is replaced.

Documented procedures establish the responsibility, calibration methods, frequency, and notification requirements for calibration and the requirements for handling discrepant equipment including validation of items checked with equipment.

Subcontracted calibration is performed by approved suppliers. Supplier approval is described in Section 3.2.1.

3.2.6.2 Tests and Inspections of Forgings

Forgings used for steam generator pressure boundary components are examined and tested in accordance with ASME Section III requirements. Additional requirements are imposed by BWI on critical forgings such as tubesheets. These include restrictive chemistry requirements (Sulphur, Phosphorous, etc.) and additional ultrasonic and magnetic particle examination requirements. Their purpose is to ensure that critical forgings (primarily tubesheets) are free from inclusions or defects which could affect the structure, cladding, tube-to-tubesheet welds, or welds in highly stressed areas of the tubesheet and lead to in

service problems.

3.2.6.3 Tests and Inspection of Tubing

Tubing quality is critical to long-term steam generator performance and integrity. For this reason, BWI tubing requirements significantly exceed ASME Code and industry standards. Tubing is procured to the requirements of ASME Section III and EPRI NP 6743-L, Volume 2 guidelines. BWI chemistry requirements are more restrictive than those required by either ASME or EPRI. Special requirements are placed on content of iron, carbon, sulfur, chromium and cobalt. In addition, BWI imposes requirements for chemistry, nondestructive examination (multi-directional UT with both outside- and inside diameter calibration notches), and rejectable defect size (max. 0.002 inches).

Prior to manufacturing tubes for an order, the tubing vendor must qualify the manufacturing process and inspection techniques on a pre-production tubing lot. These tubes are examined by BWI using enhanced NDE techniques and destructive examination to assess whether they meet BWI standards.

At all times during tube manufacture, the vendor's processes are monitored by resident BWI inspectors. In addition a statistical sample of tubes from each lot are subject to enhanced NDE and destructive examination by BWI personnel to ensure that tubing quality is maintained.

3.2.6.4 Welds

All pressure boundary welds are examined to ASME Code requirements using trained and qualified personnel. In addition significantly more stringent requirements are imposed on welds critical to long-term integrity and performance. Tubesheet overlay cladding is ultrasonically inspected to an acceptance standard that is more stringent than that required by Section V of the ASME Code. Tube-to-tubesheet welds are required to pass a no-indication acceptance standard for liquid penetrant examination.

3.2.6.5 Steam Generators

Nuclear steam generators are tested and examined in accordance with ASME Section III requirements with additional requirements based on the experience of BWI and associated B&W divisions.

3.2.6.6 Baseline Eddy Current Inspection

Eddy current inspection is performed on the RSG tubing prior to operation to document tube condition and to form a baseline for comparison with future (inservice) tube inspections. Important features of the RSG eddy current inspection are:

1.   Each tube is inspected end to end with an internal bobbin probe prior to installation in the steam generator and after fabrication is complete.

2.   The inspection provides a complete cross section capable of showing flaw indications $\triangle_4$
and wall thinning. All indications are reported and dispositioned as either being
acceptable or requiring removal, replacement or plugging.

3.   A profilometry inspection of each tube is made through the length of the tubesheet
to assure proper and complete expansion.

4.   Data is collected and stored on optical disk for future reference.

The eddy current inspection equipment used for the baseline inspection is the MIZ 18/30
Eddy Net Acquisition and Analysis Systems. This equipment provides examination
reliability in the presence of extraneous test variables and greater flexibility in data
manipulation to provide thorough signal detection and analysis capability. Additional
specialized equipment (such as Motorized Rotating Pancake Coil (MRPC)) is available to
inspect areas of special interest.

Inspection techniques, personnel qualification and procedures are prepared using the
guidelines identified in the EPRI Report Summary NP-6201, the ASME Sections V and XI,
and NRC Regulatory Guide 1.83.

3.2.7   Handling, Storage and Shipping

Detailed BWI procedures ensure that the handling, storage, cleaning, packaging, shipping
and preservation of items are controlled to prevent damage or loss and to minimize
deterioration. Cognizant engineers provide drawings and instructions for critical operations.
Important steps for handling, storage and shipping are indicated in the inspection and test
plan. These activities are inspected and documented.

3.2.7.1   Cleanliness

BWI combines a cleanliness policy, cleanliness procedures, assembly in a nuclear clean
room, and a consumables control policy to ensure that RSGs are clean and free from
contamination when shipped. The BWI cleanliness policy guides overall conduct of
fabrication and material handling activities and maintains awareness of the importance of
cleanliness. Procedures ensure all equipment remains free of debris and potentially
deleterious materials. The clean room is used exclusively for assembly of nuclear steam
generators and similar equipment. Initial heavy fabrication operations, such as the welding
of shells, tubesheets and heads is carried out in other areas of the BWI plant. BWI also
implements a consumables control policy to ensure that expendable materials utilized during
the manufacturing and assembly processes do not contaminate primary or secondary wetted
surfaces. These measures ensure that the RSG meets the NRC and customer standards.
The following paragraphs detail BWI cleanliness procedures, clean room and consumable
material control.

3.2.7.1.1   Procedures

Cleanliness procedures ensure that debris and foreign material are excluded from the RSGs during assembly. These procedures were developed in conjunction with the manufacturing procedures and maintain and verify cleanliness during manufacturing, assembly and testing operations. Cleaning and cleanliness inspection points are incorporated into the shop routing instructions and are used, with the appropriate shop instruction sheets, to ensure that the components are clean prior to and during assembly and ensure that the final RSG meets cleanliness criteria.

The requirements of ANSI N45.2-1 (1980 edition) and NRC Regulatory Guide 1.37 are included in the cleanliness procedures. These procedures are designed to obtain N45.2-1 cleanliness level B for the primary (or tube) side and level C for the steam (or shell) side of the RSGs. Cleanliness inspections are conducted at critical stages of fabrication and assembly including:

1.   After cleaning operations.

2.   Prior to operations involving elevated temperatures (pre-heating before welding, and post-weld heat treatment).

3.   Prior to assembly operations, especially any operation which will result in a loss of access (fitting the lattice support grids into the shell assembly).

4.   After completion of final assembly and prior to the sealing of openings in preparation for shipment.

Implementation of cleanliness procedures ensures that loss of cleanliness is rare, but would be detected. The Quality Assurance Manual provides procedures to detect and rectify such a situation, and to reverify that the required degree of cleanliness is reestablished.

Full accountability is maintained of all tools and loose parts used during assembly. This includes personal effects such as eyeglasses. Additionally, all hand tools, including electrically or air powered tools, are maintained in a clean condition. This ensures that dirt, debris, oil, etc. are not transferred from the tools to the RSG.

3.2.7.1.2   Clean Room

Before the RSG or sub-assembly is moved to the clean room, it is cleaned and inspected to ensure that debris and foreign materials are not transferred into the clean room. Assembly operations, such as installation of the shroud and tube support grids, installation of tubes, and the tube-to-tubesheet welding are performed in the clean room. Filtered and heated air is provided to the clean room to maintain a positive pressure relative to outside ambient conditions. This prevents ingress of contaminants.

Internal combustion engines are not permitted inside the clean room, eliminating the

potential for oil fumes, lead and other materials. Most welding operations are performed before components are moved into the clean room. Tube-to-tubesheet welding is a Gas Tungsten Arc Welding (GTAW) process which generates a minimum amount of fumes. It is carried out within sealed and air-conditioned enclosures. If it is necessary to use another welding process, such as Shielded Metal Arc Welding (SMAW), adequate precautions are taken, using temporary enclosures, fume hoods and extractors to prevent significant release of weld fume into the atmosphere. Clean-up procedures ensure that slag and debris are contained and removed after completion of the welding operation.

The clean room is equipped with its own laboratory facility to monitor activities such as tube-to-tubesheet welding. It also has a dedicated document control center and tool crib for storage of hand tools and consumables. The operation of these facilities is governed by procedures which are compatible with the clean room cleanliness requirements.

3.2.8   Receipt Inspection Requirements

Packaging, shipping, receiving, storage and handling are in accordance with standardized procedures which meet the requirements of ANSI N45.2.2-1972 as supplemented by Regulatory Guide 1.38 and customer specifications. The packaging procedure considers the method of transportation and handling as well as possible storage environment.

3.2.8.1   Preparation for Shipment

Prior to shipment, the RSG tube and shell sides are cleaned. A foreign object inspection is performed just prior to final closure of all openings. Equipment is stored, inspected, handled, installed, and cleaned by methods which ensure that harmful contaminants do not remain on any component surface in contact with process fluids. Protection of internal cleanliness is achieved by sealing all openings with plugs, caps, or covers. All threaded plugs used to seal auxiliary nozzles are removed after site installation. These items are also protected to preclude damage that could result in loss of the nitrogen blanket or contamination of RSG internal surfaces. Covers are designed and installed for removal without damaging the vessel or pipe nozzle weld preparation. The primary nozzle covers installed for the shop hydrostatic test are left in place for shipping.

RSG internal surfaces are required to meet the following criteria prior to shipment:

1.   Surfaces having free access must pass visual, wipe test, leach sample, and rust examinations. Visual techniques include boroscopes, mirrors, supplementary lighting, or other aids when needed to properly examine hard-to-see surfaces:

       a.   The surface must appear "metal clean" when examined without magnification under a lighting level of at least 100 foot-candles.

       b.   The surface must be free of particulate contaminants such as sand, packing materials, sawdust, metal chips, wire, weld spatter, tape, and tape residue.

c. The surface must have no evidence of organic material or films such as oil, grease, paint, crayon, moisture, chemical residue, or preservatives. In addition to visual examination, the surface is wiped with a solvent-dampened, white, lint-free cloth, using a clean portion for each wipe. A visible discoloration of the cloth is unacceptable unless it is established that the deposit is not detrimental.

2. If visual examination is not possible, but the surface is accessible, inspection consists of wiping the surface with a dry, white, lint-free cloth. Visual discoloration on the cloth is unacceptable unless it is established that the deposit is not detrimental.

3. The cause of rust shall be determined to prevent recurrence.

The RSG primary and secondary sides are drained and dried immediately after hydrotesting and cleanliness inspection, and are evacuated to eliminate residual moisture (dew point $\leq$ -20°F). Each RSG is sealed and pressurized on both sides with dry nitrogen to a pressure between 5 and 10 psig. The nitrogen is maintained at this pressure for shipping.

Each RSG is shipped with a connected nitrogen supply. Redundant pressure gauges are used to indicate the nitrogen pressure in each circuit. Valved connections allow adding nitrogen as necessary. Calibration requirements and gauge ranges to monitor nitrogen pressure, the nitrogen addition procedure for supplied valving, and cleaning controls for caps are also provided to help assure that nitrogen blanket is properly maintained.

3.2.8.2 Handling and Shipping

The RSG is designed to withstand the associated loads, including lifting and upending, and environments without damage during shipping, installation, and handling. Appropriate instruments on the carrier are provided to monitor and record vibration and shock to which the RSG is subjected during shipment. Continuously recording accelerometers are installed to measure accelerations in all three directions during transit. A report characterizing the loads and effects on the shipment is prepared and submitted to the owner.

The weight, center of gravity, and lifting points for all handling procedures are provided. Limitations imposed when the RSG is lifted or moved, including maximum allowable three-dimensional accelerations, including maximum internal ambient temperatures and pressures, during shipping are also provided. These are included in the equipment Operation and Maintenance Manual (O&M Manual) which is provided to the customer.

3.2.8.3 Inspection at Jobsite

Upon receipt of a steam generator at the jobsite, the surface shall be inspected by the owner to ensure that no damage has occurred. Nozzle caps are inspected for shipping damage and to ensure that a positive nitrogen pressure has been maintained on both the primary and secondary sides.

3.2.8.4   Storage

Each RSG is prepared for long-term storage prior to shipment. Exterior surfaces are protected against rust. Interior surfaces are protected against oxidation or corrosive attack by an inert dry nitrogen gas blanket. These storage provisions should be maintained at the jobsite until the RSG is installed.

3.2.8.4   Storage