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MECREDY,R.C. Rochester Gas & Electric Corp.
RECIP.NAME RECIPIENT AFFILIATION
JOHNSON,A.R. Project Directorate I-3

See Rpts

SUBJECT: Summary of 940324 public meeting w/util to discuss Steam generator replacement & fuel reload changes for 1996.Agenda encl.

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RG AND E

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April 13, 1994

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Allen R. Johnson
Project Directorate I-3
Washington, D.C. 20555

Subject: Public Meeting on March 24, 1994 to discuss Steam Generator Replacement and Fuel Reload Changes for 1996
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Johnson:

On March 24, 1994 a public meeting was held with Rochester Gas and Electric and members of the NRC Staff to update the staff on the evaluation of the steam generator replacement and fuel reload changes at Ginna Station, currently scheduled for the spring of 1996. This letter provides a summary of the meeting.

J. Cutchin (NRC/OGC) began the meeting with a discussion of the use of 10CFR50.59 for steam generator replacement. It was noted that the NRC Staff determined that 10CFR50.59 could not be used for replacement at V.C. Summer because several changes to the Technical Specifications were intimately connected with the installation of the new larger steam generators. Mr. Cutchin stated that this did not signal a change in staff philosophy. Under the terms of 10CFR50.59, even if a replacement does not involve an unreviewed safety question, prior NRC approval is required if it involves a change in the Technical Specifications. Mr. Cutchin indicated that if the plant cannot be operated following a change without certain Technical Specification amendments, then prior approval would be required. However, a replacement project could be broken down into several discrete plant changes (as has been done at other plants), with the result that prior approval via the Technical Specification amendment process would only be required for certain specific changes, not the entire project. The staff observed that Ginna appears to be different from the V.C. Summer case in that the new steam generators will be physically similar to the existing ones, and that no necessary Technical Specification amendments have been identified.

RG&E presented the current status of the Steam Generator Replacement Project. RG&E noted that the project would be broken down into two distinct parts: the component design; and the

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installation activities. There are currently no intentions to further break down the component design aspect of the project. The installation activity aspect, however, would be further divided into discrete work activities, notably the containment openings, rigging and handling of heavy loads, pipe cutting and restoration, insulation of the generator, and storage of the old steam generators. Current evaluations indicate that all changes associated with the steam generator replacement can be accomplished without prior staff approval. A schedule was presented indicating when final safety evaluations for these items would be available. These will be provided to the staff for information only.

RG&E presented a comparison of the existing and replacement steam generators. The staff noted that the generators appear very similar. RG&E discussed the preliminary safety evaluation of the replacement steam generator design, and the detailed thermal hydraulic models of the primary system and containment being developed to reanalyze some of the UFSAR transients. The staff questioned the need for detailed models, given the similarities of the generators.

RG&E presented an overview of changes associated with the 1996 Fuel Reload. Ginna will transition from annual fuel cycles to eighteen month fuel cycles with the 1996 reload, and therefore there would be some Technical Specification amendments associated with the reload. A schedule for these submittals was presented.

RG&E presented an overview of some of the design features incorporated in the replacement steam generator to correct deficiencies identified in the current generators. The staff observed that these improvements seemed prudent, however they cautioned that close attention should be paid to the materials used for weld transitions.

RG&E outlined the intended methods to be used for the structural evaluation of the restored reactor coolant system. Specifically, RG&E intends to include explicit modeling of walls and supports when analyzing the restored system. The staff cautioned that while this appears acceptable, close attention should be given to this to ensure that consistent and conservative modeling of the wall-RCS interfaces are made, particularly with respect to damping factors. RG&E also presented information on a displacement model of the reactor coolant system developed to help predict pipe movement during cutting activities. The staff was questioned as to the appropriateness of using temporary supports or restraints, but it was agreed that further detail would be needed before a response could be provided.

RG&E presented the efforts to date to model the containment to analyze the effects of the temporary construction openings on the structure both during construction and after restoration of the containment. The staff noted that the model appeared to be of

sufficient detail to analyze all effects of the openings.

Copies of the overheads presented by RG&E at this meeting are attached. RG&E appreciates the valuable input and cooperation that was provided by the staff at this meeting.

Very truly yours,

Robert C. Mecredy

Robert C. Mecredy

Attachment
BJF/327

xc: Mr. Allen R. Johnson (Mail Stop 14D1)
Project Directorate I-3
Washington, D.C. 20555

U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

USNRC Ginna Senior Resident Inspector

[illegible]

RG&E

Ginna Station — BWI — Bechtel — Lampson — Rochester Gas and Electric

GINNA STATION

STEAM GENERATOR REPLACEMENT

NRC STATUS UPDATE

MARCH 24, 1994

AGENDA

1.0	INTRODUCTION	GEORGE WROBEL
2.0	SAFETY AND LICENSING	BRIAN FLYNN BOB ELIASZ
	2.1 PRELIMINARY SAFETY EVALUATION 2.2 RELAP MODEL 2.3 CONTEMPT MODEL 2.4 FUEL CONTRACT	
3.0	STATUS / STRESS ANALYSIS UPDATE	JOHN SMITH BERNIE CARRICK
	3.1 EQUIPMENT / INSTALLATION STATUS 3.2 METHODOLOGY AND APPROACH 3.3 ACCEPTANCE CRITERIA 3.4 STATUS AND SCHEDULE 3.5 DISPLACEMENT MODEL	
4.0	S/G DESIGN FEATURES	JOHN SMITH
5.0	CONTAINMENT OPENING MODEL	JOHN SMITH
6.0	SUBMITTAL SCHEDULE	GEORGE WROBEL
7.0	REPLACEMENT VIDEO	JOHN SMITH

9404190372

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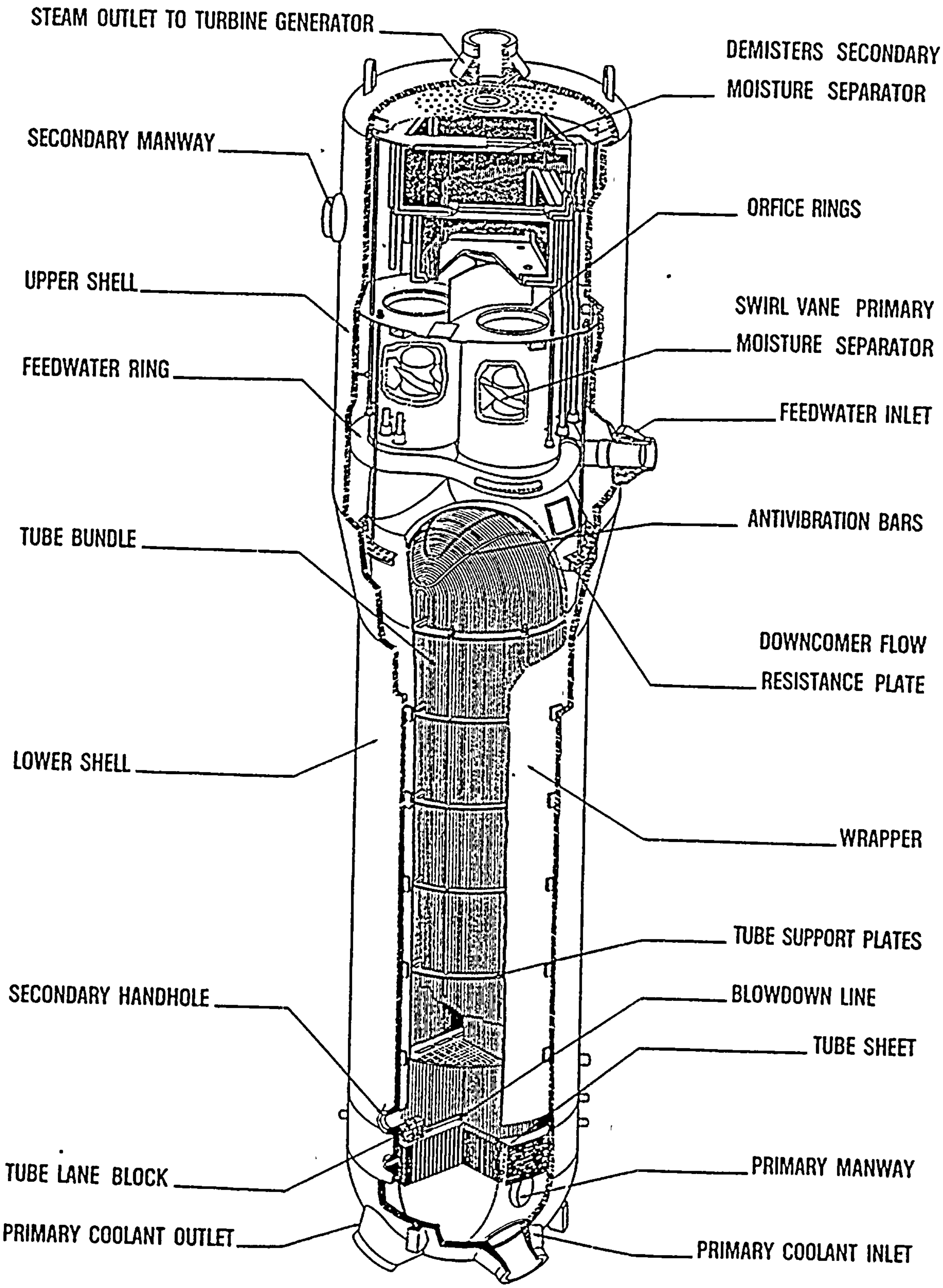
R.E. Ginna Steam Generator Replacement
Comparison of Existing vs. Proposed

	Existing	Replacement
Manuf./Model	W/44 (feedring)	BWI (feedring)
Primary Side Pressure Drops (0% plug)		
Nozzle inlet to Nozzle outlet	33.5 psi	31.1 psi
Primary Side Flow for above dp's	34.6 E06 lbm/hr	34.9 E06 lbm/h
Heat Transfer Areas		
0% Plugging	44430 sq. ft.	54000 sq. ft.
15% Plugging	37765 sq. ft.	-
20% Plugging	-	43200 sq. ft.
Tubing		
OutsideDiameter	0.875 in	0.750 in
Avg. Wall Thickness	0.050 in	0.0431 in
Number of Tubes	3260	4765
Material	Inconel 600, MA	Alloy 690, TT
Volumes, primary side		
Inlet Plenum	133 cu. ft.	132.5 cu.ft.
Tubes	654.5 cu. ft.	710 cu.ft.
Outlet Plenum	133 cu. ft.	132.5 cu. ft.
Secondary Volume, Total	4580 cu. ft.	4513 cu. ft.
Secondary Water Mass, nominal		
100% (1520 MWt)	84,500 lbm	86,200 lbm
0% (HZP)	118300 lbm	115,100 lbm
Secondary Mass Flow, 100%	3.3 E06 lbm/hr	3.3 E06 lbm/hr
Steam Line Orifice Size	4.37 sq. ft.	1.4 sq. ft.
Initial Steam Pressure, 100%	800 psia	875 psia

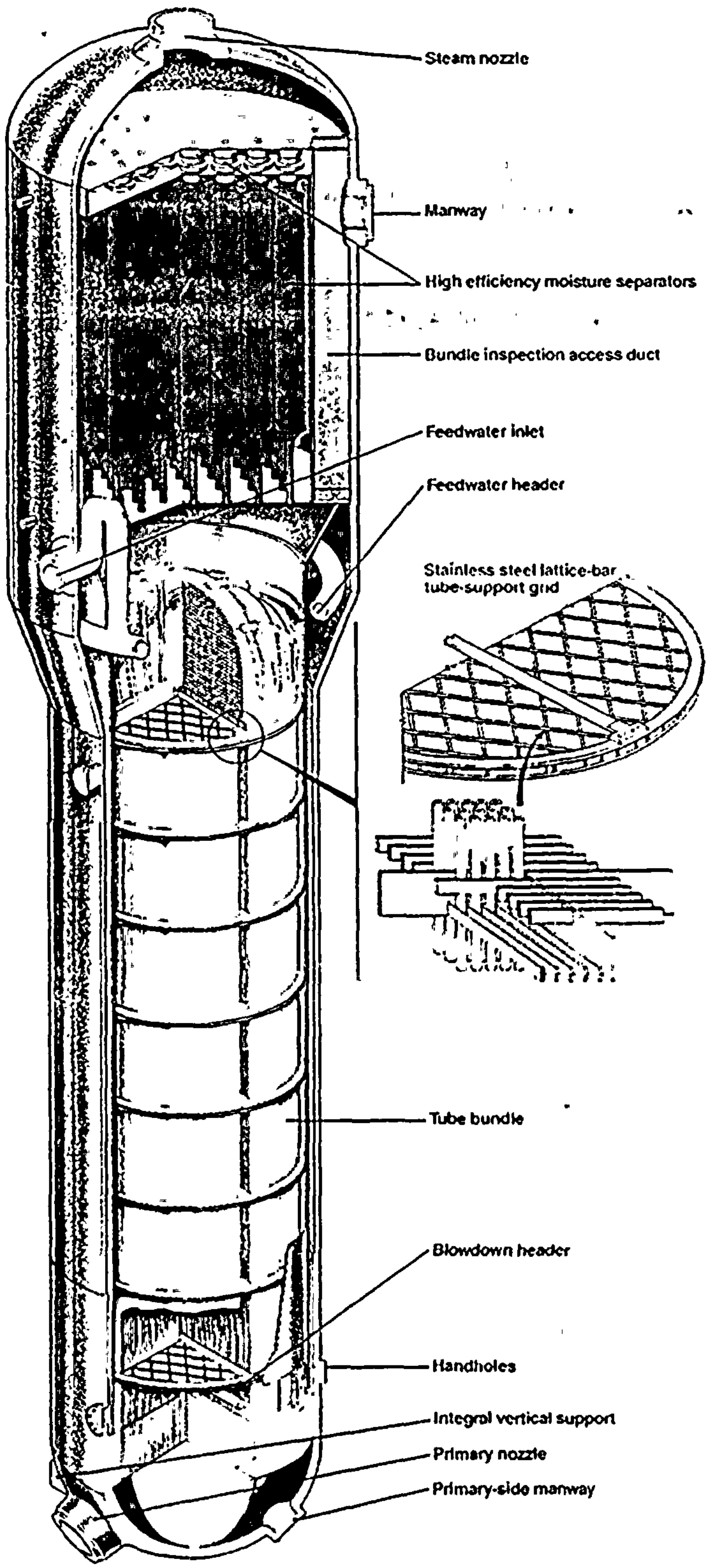
24 March 94



SERIES 44 STEAM GENERATOR

NOT TO SCALE
9221

Babcock & Wilcox Canada Advanced Series PWR Replacement Steam Generator



Design Objectives

- Retain all terminal points, outside dimensions and overall performance consistent with existing steam generators and reactor requirements
- Design for maximum reliability in operation
- Achieve high circulation ratios
- Eliminate crevices and potential flow stagnation areas
- Use non-rigid tube supports
- Maintain pressure boundary integrity during seismic and burst pipe events
- Minimize tube residual stress
- Avoid flow-induced vibration
- Assure high steam quality (above 99.75%) under all operating conditions
- Prevent loose parts
- Facilitate inspection and maintenance

Ginna Accident Analysis

EVALUATE	ANALYZE		
		15.1	Increase in Heat Removal by the Secondary System
X		15.1.1	Decrease in Feedwater Temperature
X		15.1.2	Increase in Feedwater Flow
X		15.1.3	Excessive Load Increase Incident
X		15.1.4	Inadvertent Opening of a SG Relief/Safety Valve
	X	15.1.5	Steam Line Breaks Inside and Outside Containment
	X	15.1.6	SG Relief Valve and Feedwater Control Valve Failure
		15.2	Decrease in Heat Removal by the Secondary System
X		15.2.1	Steam Pressure Regulator Malfunction
	X	15.2.2	Loss of External Electrical Load
X		15.2.3	Turbine Trip
X		15.2.4	Loss of Condenser Vacuum
X		15.2.5	Loss of Offsite Power to the Station Auxiliaries
X		15.2.6	Loss of Normal Feedwater Flow
X		15.2.7	Feedwater System Pipe Breaks
		15.3	Decrease in RCS Flowrate
X		15.3.1	Flow Coastdown Accidents
X		15.3.2	Locked Rotor Accident
		15.4	Reactivity and Power Distribution Anomaities
X		15.4.1	Uncontrolled RCCA Withdrawal from Subcritical
X		15.4.2	Uncontrolled RCCA Withdrawal at Power
X		15.4.3	Startup of an Inactive Reactor Coolant Loop
X		15.4.4	CVCS Malfunction
X		15.4.5	RCCA Ejection
X		15.4.6	RCCA Drop
X		15.5	Increase in RCS Inventory
X		15.6	Decrease in RCS Inventory
X		15.6.1	Inadvertent Opening of a Pressurizer Safety or Relief Valve
		15.6.2	Radiological Consequences of Small Lines Carrying RC
X			Outside Containment
	X	15.6.3	Steam Generator Tube Rupture
		15.6.4	Primary System Pipe Ruptures
		15.6.4.1	SBLOCA
X		15.6.4.2	LBLOCA
X		15.7	Radiological Release From a Subsystem or Component
		15.7.1	Radiological Gas Waste System Failure
X		15.7.2	Radiological Liquid Waste System Failure
X		15.7.3	Fuel Handling Accidents
X		15.8	Anticipated Transients Without Scram
X			
			Chapter 6, Chapter 5
	X	6.2.1.2	Containment Integrity
	X	5.2.2	Low Temperature Overpressurization

RG&E

Ginna Station
BWI
Bechtel
Lampson
Rochester Gas and Electric

STEAM GENERATOR REPLACEMENT TRANSIENT ANALYSIS STATUS

ACTIVITY	COMPLETION
MODELS	
RELAP	COMPLETE
CONTEMPT	COMPLETE
ANALYZE CONTAINMENT	5/94
ANALYZE ACCIDENTS	9/94
10CFR50.59 EVALUATION	12/94

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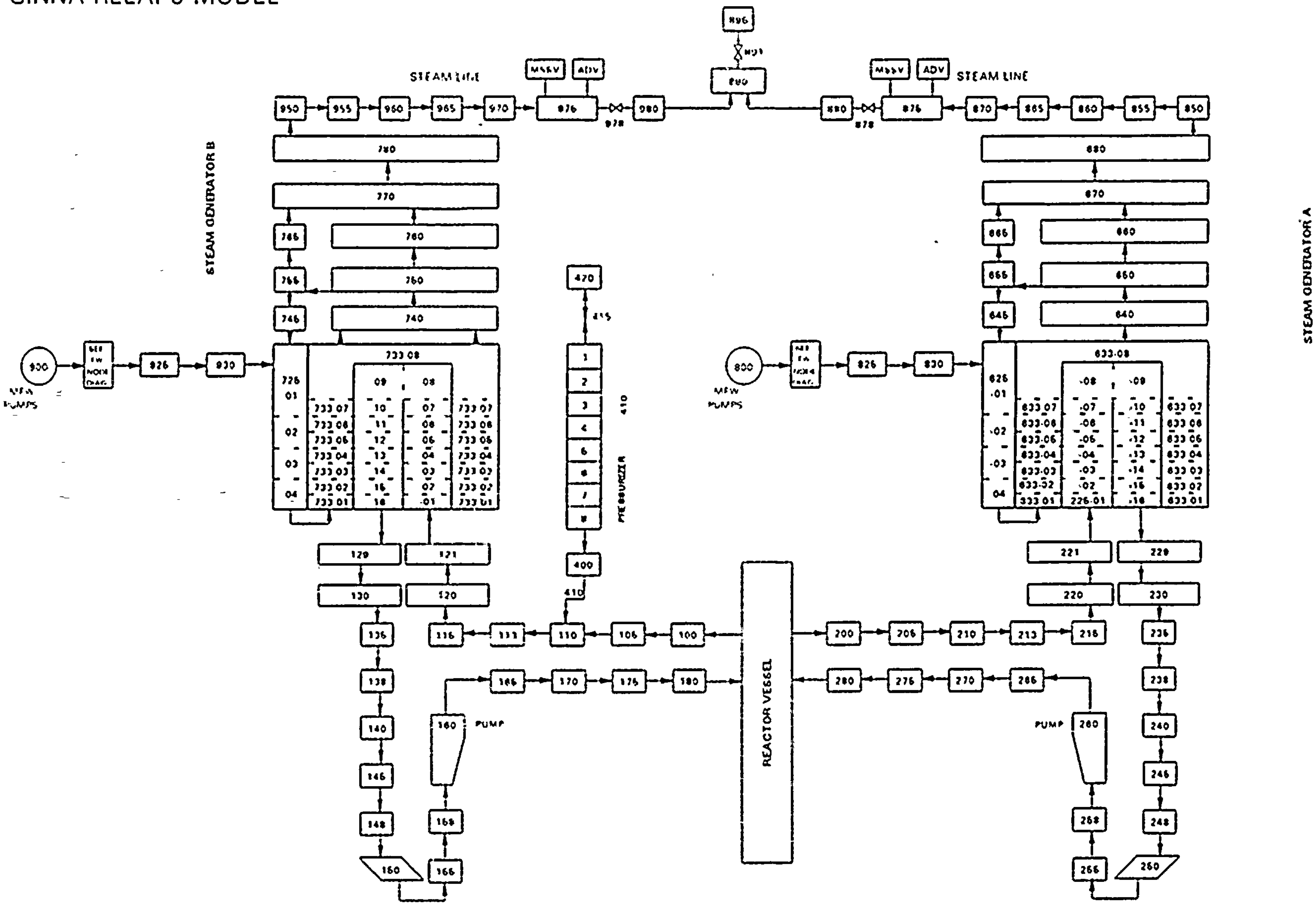
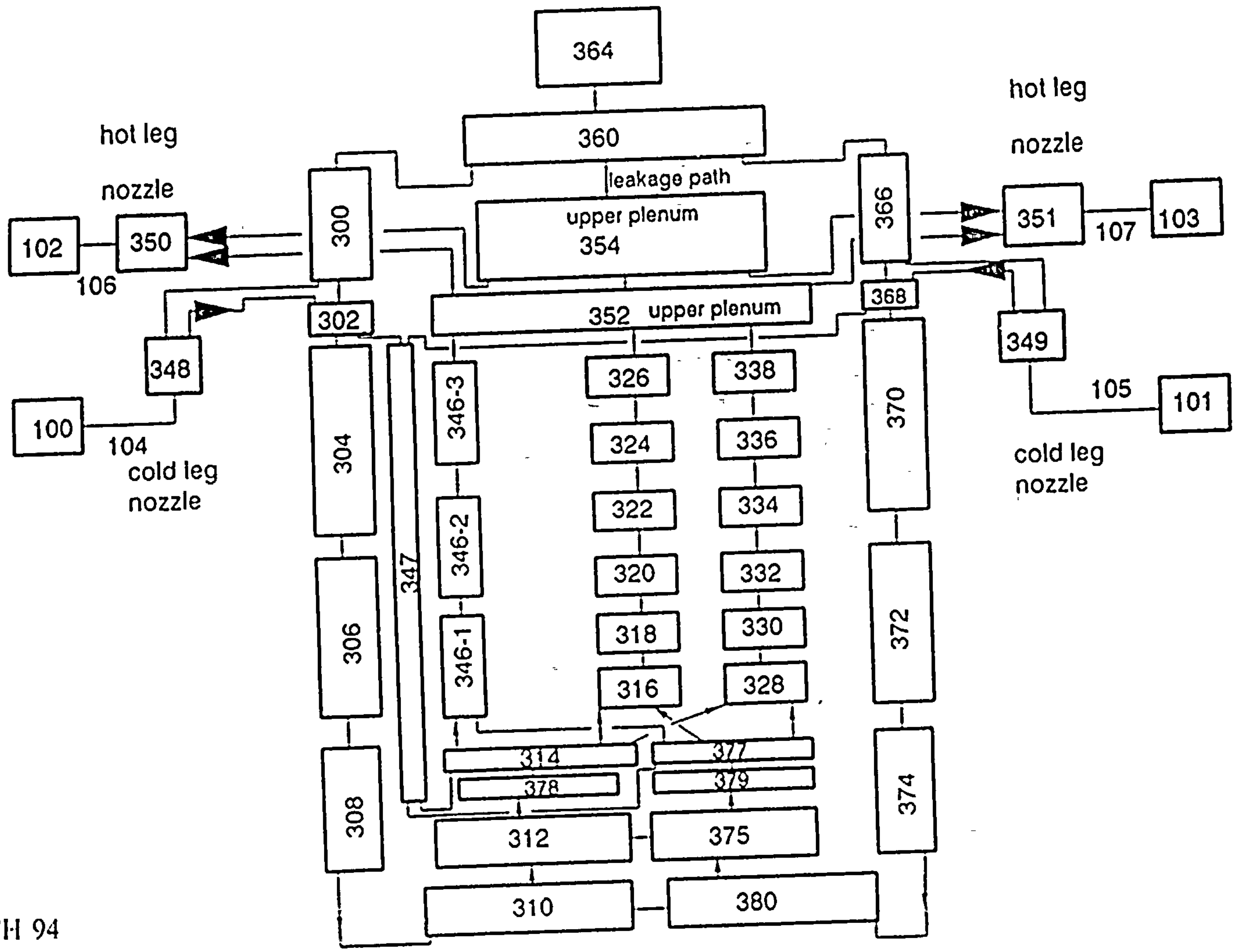
GINNA RELAP5 MODEL



24 MARCH 94

GINNA REACTOR VESSEL AND CORE MODEL



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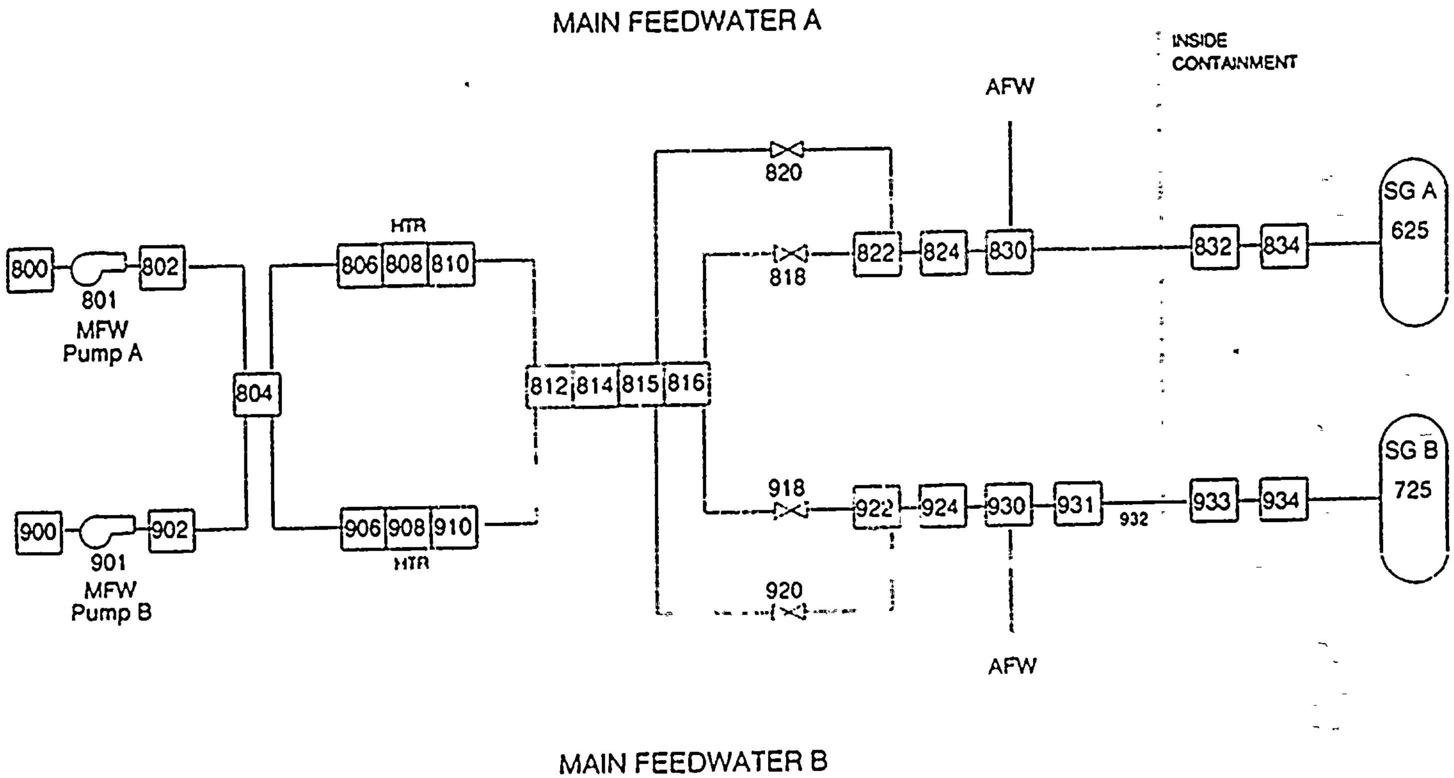

Ginna MFW RELAP5 Node Diagram

24 MARCH 94

1996 FUEL RELOAD

- NEW WESTINGHOUSE FUEL FABRICATION CONTRACT
- INSERT VANTAGE -5 FUEL VS. OFA FUEL
- $F_{\Delta H}$ INCREASE FROM 1.66 TO 1.75
- F_Q INCREASE FROM 2.32 TO 2.50
- CYCLE LENGTH INCREASE FROM ANNUAL TO 18 MONTHS
- INCREASE SFP ENRICHMENT FROM 4.25 W/O TO 5.0 W/O
- POSSIBLE T_{ave} DECREASE UP TO 15°F

24 MARCH 94

FUEL ASSEMBLY CHARACTERISTICS
COMPARISON OF CURRENT VS. 1996

CHARACTERISTIC	CURRENT	1996
TYPE	14 X 14 OFA	VANTAGE 5
FUEL ROD O.D.	0.40 IN	0.40 IN
FUEL CLADDING MATERIAL	Z_r-4	Z_r-4
ACTIVE FUEL LENGTH	141.4 IN	141.4 IN
BLANKET REGION/ ENRICHMENT	6 IN/NAT.	6 IN/2.6 W/O
CENTER REGION ENRICHMENT	UP TO 4.25 W/O	UP TO 5.0 W/O
BOTTOM NOZZLES	DFBN	DFBN
GRIDS		
2 TOP AND BOTTOM	INCONEL-718	INCONEL-718
7 MID	Z_r-4	Z_r-4
INTERMEDIATE FLOW MIXING	NONE	NONE
DISCHARGE BURNUPS	LOW 40s GWD/MTU	MID 50s GWD/MTU

24 MARCH 94

Ginna Accident Analysis

15.1 Increase in Heat Removal by the Secondary System
- 15.1.1 Decrease in Feedwater Temperature
- 15.1.2 Increase in Feedwater Flow
- 15.1.3 Excessive Load Increase Incident
- 15.1.4 Inadvertent Opening of a SG Relief/Safety Valve
- 15.1.5 Steam Line Breaks Inside and Outside Containment
- 15.1.6 SG Relief Valve and Feedwater Control Valve Failure

15.2 Decrease in Heat Removal by the Secondary System
- 15.2.1 Steam Pressure Regulator Malfunction
- 15.2.2 Loss of External Electrical Load
- 15.2.3 Turbine Trip
- 15.2.4 Loss of Condenser Vacuum
- 15.2.5 Loss of Offsite Power to the Station Auxiliaries
- 15.2.6 Loss of Normal Feedwater Flow
- 15.2.7 Feedwater System Pipe Breaks

15.3 Decrease in RCS Flowrate
- 15.3.1 Flow Coastdown Accidents
- 15.3.2 Locked Rotor Accident

15.4 Reactivity and Power Distribution Anomaities
- 15.4.1 Uncontrolled RCCA Withdrawal from Subcritical
- 15.4.2 Uncontrolled RCCA Withdrawal at Power
- 15.4.3 Startup of an Inactive Reactor Coolant Loop
- 15.4.4 CVCS Malfunction
- 15.4.5 RCCA Ejection
- 15.4.6 RCCA Drop

15.5 Increase in RCS Inventory

15.6 Decrease in RCS Inventory
- 15.6.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve
- 15.6.2 Radiological Consequences of Small Lines Carrying RC Outside Containment
- 15.6.3 Steam Generator Tube Rupture
- 15.6.4 Primary System Pipe Ruptures
  - 15.6.4.1 SBLOCA
  - 15.6.4.2 LBLOCA

15.7 Radiological Release From a Subsystem or Component
- 15.7.1 Radiological Gas Waste System Failure
- 15.7.2 Radiological Liquid Waste System Failure
- 15.7.3 Fuel Handling Accidents

15.8 Anticipated Transients Without Scram

6.2.1.2 Containment Integrity

5.2.2 Low Temperature Overpressurization

ANALYSES THAT WILL BE UPDATED WITH RELOAD

15.1.1	DECREASE IN FEEDWATER TEMPERATURE
15.1.2	INCREASE IN FEEDWATER FLOW
15.1.3	EXCESSIVE LOAD INCREASE INCIDENT
15.1.4	INADVERTENT OPENING OF A SG RV
15.1.5	SLB (BOTH CORE AND M&E)
15.1.6	SG RV & FW CONTROL VALVE FAILURE
15.2.7	LOSS OF EXTERNAL LOAD/TURBINE TRIP
15.3.1	FLOW COASTDOWN ACCIDENTS
15.6.3	SG TUBE RUPTURE
15.6.4.1	SBLOCA
15.6.4.2	LBLOCA
15.7.3	FUEL HANDLING ACCIDENTS
5.2.2	LOW TEMP. OVERPRESSURIZATION (BWNT)

SCHEDULE

-	DATA PREPARATION	4/1/94
-	FINALIZE INPUT DATA	6/1/94
-	START ANALYSIS	6/1/94
-	DRAFT REPORT	6/1/95
-	FINAL REPORT	7/1/95
-	SUBMIT REPORT TO NRC	8/1/95
-	CYCLE 26 STARTUP	5/1/96

EQUIPMENT STATUS

- **FABRICATION BY B&W INTERNATIONAL**

- **ALL MAJOR COMPONENTS ORDERED**

  - MAJOR FORGINGS, JAPAN STEEL WORKS
  - TUBING, VALINOX
  - SHELL PLATE, CREUSOT-LOIRE

- **PRIMARY HEADS**

  - CLADDING COMPLETE
  - NOZZLE DAM RINGS BEING INSTALLED
  - PRIMARY NOZZLE BUTTERING UNDERWAY

- **TUBESHEETS**

  - CLADDING COMPLETE
  - READY FOR GUNDRILLING OF TUBEHOLES

- **SECONDARY SHELLS**

  - LOWER SHELL CONES WELDED
  - HANDHOLES AND INSPECTION PORTS BEING INSTALLED

- **TRANSITION CONE FORGINGS**

  - HANDHOLE OPENINGS CUT

EQUIPMENT STATUS CONT'D

- **TUBING**

- PRE PRODUCTION UNDERWAY AT VALINOX
- PREPRODUCTION COMPLETE JUNE 1994
- PRODUCTION MATERIAL BEING MELTED AT INCO
- PRODUCTION COMPLETE DECEMBER 1994

INSTALLATION STATUS

- **INSTALLATION CONTRACTOR BECHTEL POWER**

  - DETAILED DESIGN
  - INSTALLATION

- **DETAILED ENGINEERING 1994**

- **PROCEDURE PREPARATION 1995**

- **ACTIVITIES TO DATE**

  - PROJECT INTERFACE PROCEDURES
  - VIDEO PREPARATION
  - CONTAINMENT OPENING STUDY
  - QA PROCEDURE MANUAL
  - PROJECT ENGINEERING PROCEDURES MANUAL
  - INSULATION STUDY
  - DRAFT DESIGN CRITERIA FOR CONTAINMENT STRUCTURAL WORK

- **MAJOR SUBCONTRACTORS**

  - POWER CUTTING
  - LAMPSON
  - PSI

RGE

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Rochester Gas and Electric

R. E. GINNA

STEAM GENERATOR REPLACEMENT

STRUCTURAL EVALUATION

OF

EFFECTED COMPONENTS & SYSTEMS

24 March 94

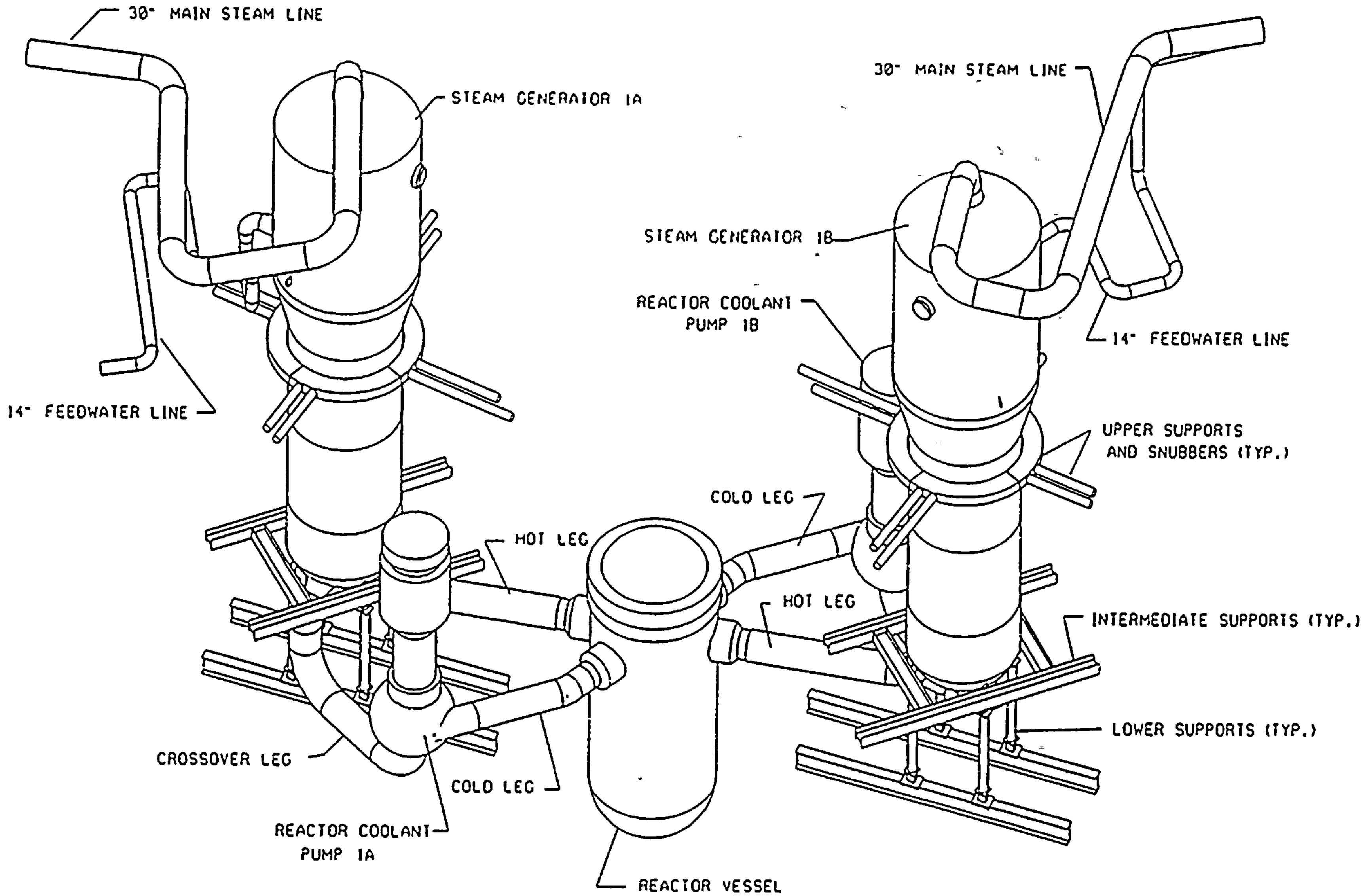
REACTOR COOLANT SYSTEM

LATEST ANALYSIS

- 1988 SNUBBER REPLACEMENT
- IMPLEMENT LBB/HELB CRITERIA
- RIGID STRUTS
- S/G COLD SPRING

ACCEPTANCE CRITERIA

● PIPING	USAS\ANSI B31.1
● EQUIPMENT	ASME SECTION III
● SUPPORTS	ASME SECTION III



VIEW OF NSSS [illegible]TEM FOR GINNA NUCLEAR STATION

STRUCTURAL MODELS

BWSPAN - STRUCTURAL CODE

3 MODELS
- BENCHMARK
- ENHANCED W/OLD S/G
- ENHANCED W/NEW S/G

DEMONSTRATE UNDERSTANDING OF CURRENT BASIS AND LOOP BEHAVIOR

GENERATE DETAILED LOADING/STRESS INFO FOR CURRENT S/G

DISTINGUISH EFFECTS OF MODEL ENHANCEMENTS AND S/G DIFFERENCES

CONFIRM/CALCULATE LOADS, STRESSES, AND THERMAL MOTIONS FOR NEW S/G

LOADING CONDITIONS

BENCHMARK
- DEADWEIGHT
- OBE

ENHANCED MODEL W/OLD S/G
- DEADWEIGHT
- OBE

ENHANCED MODEL W/NEW S/G
- DEADWEIGHT
- THERMAL
- OBE/SSE
- LOCA/HELB

- COLD SHUTDOWN EARTHQUAKE

ACCEPTANCE CRITERIA

1. COMPARE TO CURRENT ANALYSIS

   NEW LOADS < OLD LOADS = OK

2. COMPARE TO ALLOWABLES

PIPING	- B31.1 ALLOWABLES - CURRENT LBB CRITERIA
EQUIPMENT	- NOZZLES LOADS - SUPPORT LOADS
AUX LINES	- < $^1/_{16}$ " ADDED DEFLECTION

24 March 94

MODEL ENHANCEMENTS

NEW	OLD
CONSISTENT MASS MODELING	LUMPED MASS
EXPLICIT MODELING OF SUPPORTS	STIFFNESS MATRIX
FREQUENCY CUTOFF @ 30Hz	100 Hz
N-411 DAMPING	2%/4% DAMPING
SINGLE ANCHOR Pt./SINGLE SPECTRA	MULTIPLE ANCHOR Pts./ENVELOPE SPECTRA
CLOSELY SPACED MODES VIA 10% RULE	EPSILON RULE
EXPLICIT ACP ANALYSIS FOR HELB	FACTOR ON DISCHARGE COEFFICIENT

LOADING METHODS
SEISMIC

DW\THERMAL
- STATIC

OBE\SSE
- RESPONSE SPECTRA
- 3 AXIS EXCITATION
- CURRENT DESIGN SEISMIC SPECTRA
- MULTIPLE CASES FOR SUPPORTS
- MODES COMBINED SRSS
- CLOSELY SPACED MODES VIA 10% RULE

- TIME HISTORIES FOR ARSs ON S/G SHELL

LOADING METHODS
LOCA\HELB

CRAFT & COMPAR2 CODES

LINEAR TIME HISTORY ANALYSIS
- PIPING\INTERNALS TRANSIENT LOADS
- M/E RELEASE FOR ACP ANALYSIS
- ACP ON COMPONENTS
- CONFIRM SUPPORTS ACTIVE

RCS : LEAK-BEFORE-BREAK
- RHR LINE
- SURGE LINE
- SI LINE

HELB : TERMINAL LOCATIONS ONLY
- MAIN STEAM
- FEEDWATER
- BLOWDOWN
- RECIRC NOZZLE

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CURRENT EXPECTATION

DEADWEIGHT	- INCREASE (<5%)
THERMAL	- SAME
SEISMIC	- DECREASE [1]
LOCA	- INCREASE (< 15%) [2]
HELB	- DECREASE

1. MODEL ENHANCEMENTS = MORE MARGIN
INCREASED WEIGHT = LESS MARGIN

   OVERALL EFFECT - EXPECT MORE MARGIN

2. BLOWDOWN INITIAL CONDITIONS WILL USE 15°F REDUCED T_{AVE}

24 March 94

STATUS & SCHEDULE

MODELS NEARING COMPLETION

STRUCTURAL BENCHMARKING STARTED

SEISMIC ANALYSIS:	COMPLETE 7/94
BLOWDOWN ANALYSES:	COMPLETE 10/94
LOAD COMBINATION & COMPARISONS:	COMPLETE 1/95

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DISPLACEMENT MODEL

PURPOSE

RG&E TECHNICAL OVERSIGHT OF RCS PIPING CUT & WELD

CONCERNS

COLD SPRING (BEFORE \ AFTER)
WELD FIT-UP
RCS TEMPORARY SUPPORT DESIGN

MODEL

ANSYS FINITE ELEMENT MODEL OF RCS
PLATE & BEAM ELEMENTS
MODEL RCS PIPE AS SHELLS
3400 ELEMENTS / 3400 NODES
ONE LOOP MODEL

STATUS

MODEL NEAR COMPLETION

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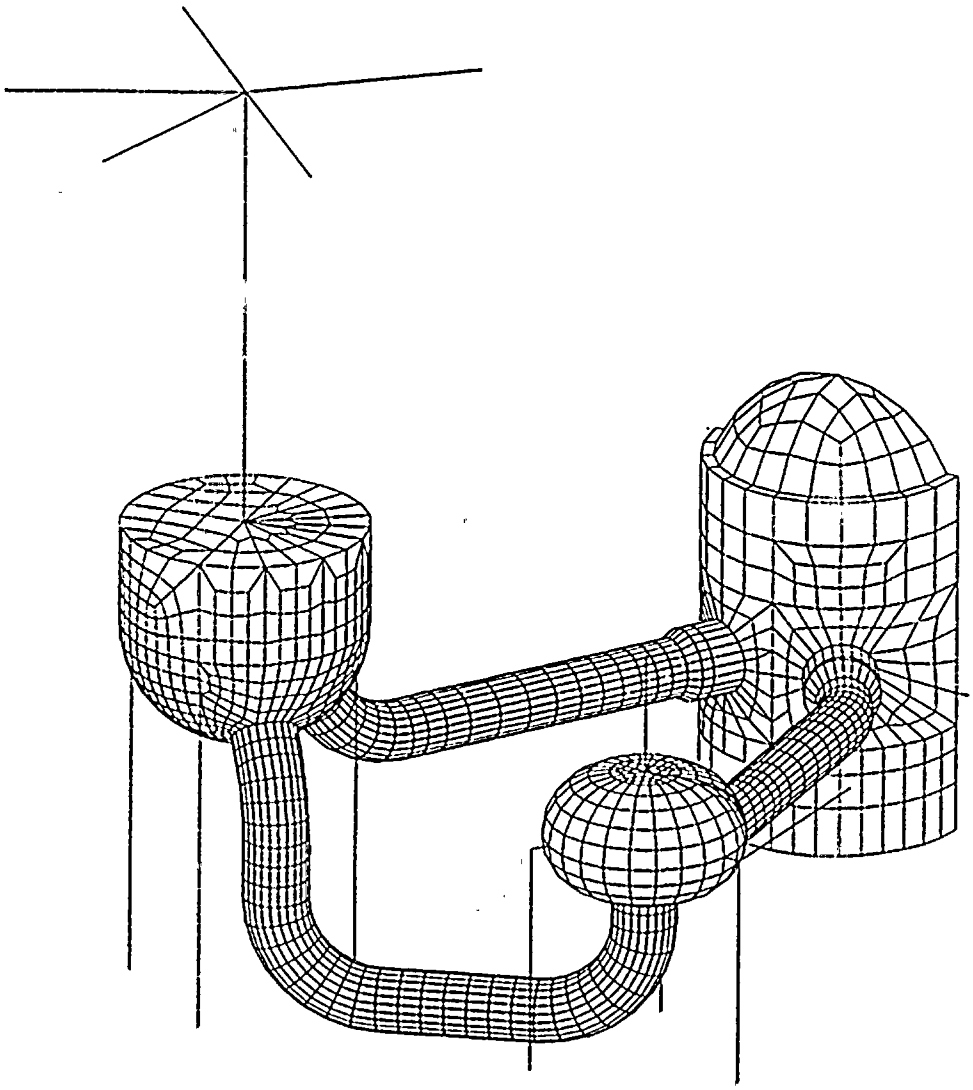

S/G DESIGN FEATURES

PROBLEM	DESIGN FEATURES
TUBESHEET DEFECTS	CLOSED CREVICE HYDRAULIC EXPANSION INCONEL 690 TUBING
SLUDGE ACCUMULATION ON TUBESHEET	HIGH CIRCULATION RATIO INSPECTION/MAINTENANCE PORTS ACCESSIBLE FOR SLUDGE LANCING
DEFECTS AT TUBE SUPPORT PLATES	LATTICE GRIDS STAINLESS STEEL CONSTRUCTION INCONEL 690 MATERIAL
HIGH CYCLE FATIGUE	FAN BAR SUPPORT SYSTEM
WATER HAMMER	GOOSE NECK AT FEEDWATER RING INLET J-TUBES
J-TUBE FAILURES	INCONEL 690 FOR EROSION RESISTANCE

S/G DESIGN FEATURES

PROBLEM	DESIGN FEATURES
MOISTURE CARRYOVER	HIGH EFFICIENCY SEPARATORS 0.10% GUARANTEE
PRESSURE BOUNDARY WELD FAILURES	FORGED AND PLATE COMPONENTS NO CORNER WELDS STRICT PRE AND POST HEAT REQUIREMENTS
PWSCC OF U-BENDS	LARGE MINIMUM RADIUS BENDS STRESS RELIEF OF FIRST 8 ROWS
SECONDARY LOOSE PARTS	NO FASTENERS, 100% WELDED STRUCTURE
PRIMARY SIDE ACCESS	18" DIAMETER MANWAYS
SECONDARY SIDE ACCESS	6-8" HANDHOLES 14-2" INSPECTION PORTS 1-18" MANWAY
PRIMARY NOZZLE WELDING	316 LN SAFE ENDS NARROW GAP WELDING SPARE ELBOWS

CONTAINMENT OPENING MODEL

- **CONTAINMENT OPENING DESIGN INCLUDED IN BECHTEL WORKSCOPE**

- **RGE MODEL FOR OVERCHECK OF BECHTEL DESIGN**

- **ANSYS FINITE ELEMENT MODEL**

- **INCLUDES ENTIRE CONTAINMENT STRUCTURE**

  - ROCK ANCHORS
  - BASE MATS
  - WALLS AND TENDONS
  - SPRING LINE AND TRANSITION
  - DOME STEEL, CONCRETE AND LINER PLATE

- **MODEL DEVELOPMENT COMPLETE**

- **VERIFIED AGAINST CLASSICAL SOLUTIONS**

- **WILL BE AVAILABLE TO VERIFY BECHTEL DESIGN AND FOR CONSTRUCTION.**

<u>SCHEDULE FOR INFORMATIONAL SUBMITTALS</u>

COMPONENT ACTIVITIES

PRELIMINARY SAFETY EVALUATION	MAY 1994
FINAL REPORT/50.59 EVALUATION	MAY 1995

INSTALLATION ACTIVITIES

SAFETY EVALUATION OF CONTAINMENT OPENING	AUGUST 1994
SAFETY EVALUATION OF RIGGING AND HANDLING	OCTOBER 1994
SAFETY EVALUATION OF STEAM GENERATOR PIPING	DECEMBER 1994
SAFETY EVALUATION OF STEAM GENERATOR INSULATION	DECEMBER 1994
TESTING AND INSPECTION PLAN	MARCH 1995

RG&E

Ginna Station
BWI Bechtel Lampson
Rochester Gas and Electric

SCHEDULE FOR SUBMITTALS FOR REVIEW

I-690 RELIEF REQUEST	MAY 1994
CURRENT STEAM GENERATOR TUBE RUPTURE ANALYSIS	JULY 1994
FUEL RELOAD REPORT	AUGUST 1995

24 March 94