50-498

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 19, 1997

LICENSEE: STP Nuclear Operating Company

FACILITY: SOUTH TEXAS PROJECT, UNIT 1

SUBJECT: SUMMARY OF DECEMBER 10, 1997, MEETING ON STEAM GENERATOR (SG) REPLACEMENT FOR UNIT 1

On December 10, 1997, the licensee met with the NRC to discuss their progress towards SG replacement at Unit 1. Meeting attendees are listed in Attachment 1. The handout provided by the licensee is in Attachment 2.

The fabrication of the replacement SGs is about 45% complete (delivery scheduled for mid-1999), the licensing engineering is about 78% complete (with proposed technical specification changes to be submitted in May 1998), SG replacement engineering is about 70% complete (all design change packages to be issued by June 1998), and plant preparations are underway. SG replacement is planned for refueling outage 9 in early 2000.

The design characteristics of the delta 94 replacement SGs, compared to the current SGs, include increased reactor coolant system (RCS) flow, increased RCS volume, increased heat transfer area, increased mass/weight, increased secondary inventory and an elevated feedring instead of a preheater. The licensee presented the status of the various evaluations and analyses to support the delta 94 SG design. The licensee is performing evaluations when existing analyses remain bounding with the delta 94 design. New analyses are performed when the existing analyses are no longer bounding.

The licensee plans to submit technical specification changes in May 1998 for RCS flow, RCS volume, SG water level, auxiliary feedwater storage tank volume, and SG surveillances. The licensee also plans to submit an unreviewed safety question on dose consequences (they indicated there is a small increase in doses). In addition, the licensee noted that the implementation methodologies in WCAP-14882 (for the Retran Code) and WCAP-14565 (for the Vipre Code), which are being used for various reanalyses, are currently under NRC review.

During the course of the meeting the staff asked questions and requested clarifications on the material presented. In addition, the staff indicated a desire for further discussion on the circumferential temperature distribution and resulting stress distribution on the delta 94 SG tubes due to the broached hole configuration. The staff also indicated that the dose consequence results will need to be adequately supported with the analytical assumptions and other data (i.e., describe methodology, show calculations) sufficient for the staff to perform confirmatory calculations.

DF01 1/1

The licensee indicated a desire for a follow-up meeting to discuss progress on the supporting small-break and large-break loss-of-coolant accident (LOCA) analyses, and a desire to meet again with the staff to present the remaining analytical work and installation details in March 1998.

Thomas W. Alexion

Thomas W. Alexion, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-498

Attachments: 1. List of Meeting Attendees
2. Meeting Handout

cc w/atts: See next page

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ORIGINAL SIGNED BY:
Thomas W. Alexion, Project Manager
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Office of Nuclear Reactor Regulation

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MEETING BETWEEN STP NUCLEAR OPERATING COMPANY AND NRC

REPLACEMENT OF UNIT 1 STEAM GENERATORS

December 10, 1997

Name	Organization
T. Alexion	NRC
A. Keim	NRC
M. Hartzman	NRC
C. Hinson	NRC
I. Barnes	NRC
M. Shuaibi	NRC
J. Tsao	NRC
J. Conly	STP
C. Albury	STP
C. Pham	STP
S. Patel	STP
M. McBurnett	STP
A. McIntyre	STP
D. Dominicis	Westinghouse
B. McFetridge	Westinghouse
D. Bhowmick	Westinghouse
C. Reid	Bechtel
F. Farzam	Bechtel
R. Pernisi	Bechtel
A. Papadopoulos	Bechtel
J. Weil	McGraw-Hill

ATTACHMENT 1

SOUTH TEXAS PROJECT
ELECTRIC GENERATING STATION

UNIT 1 STEAM GENERATOR
REPLACEMENT PROJECT

NRC
BRIEFING
DECEMBER 10, 1997

ATTACHMENT 2

SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION
UNIT 1 STEAM GENERATOR REPLACEMENT PROJECT

AGENDA

INTRODUCTION	Chet McIntyre
LICENSING ENGINEERING APPROACH	Chet McIntyre
FOLLOW-UP TO APRIL 1997 BRIEFING	Chet McIntyre
NON-LOCA ANALYSIS	Charlie Albury
PRESSURE/TEMPERATURE ANALYSIS	Charlie Albury
INTRODUCTION TO PIPING AND SUPPORTS ANALYSIS	Sam Patel
REACTOR COOLANT SYSTEM PIPING ANALYSIS	Sam Patel
MAIN STEAM PIPING STRESS EVALUATION	Cong Pham
FW/AF PIPING REROUTING	Cong Pham
NSSS/BOP INTERFACE SYSTEMS EVALUATION	Bob McFetridge
RCP AND CRDM EVALUATIONS	Bob McFetridge
STEAM GENERATOR REPLACEMENT SEQUENCE	Sam Patel
LICENSING APPROACH	Mark McBurnett

INTRODUCTION

INTRODUCTION

DESIRED BRIEFING OUTCOME

- COMMUNICATE PROJECT STATUS AND TECHNICAL INFORMATION
- DETERMINE AREAS OF SPECIAL INTEREST TO THE NRC

PROJECT STATUS

- RSG FABRICATION ~ 45% COMPLETE; DELIVERY MID-1999
- LICENSING ENGINEERING ~ 78% COMPLETE; TECH SPEC CHANGE SUBMITTAL MAY 1998
- SGR ENGINEERING ~ 70% COMPLETE; ALL DCPs ISSUED BY JUNE 1998
- PLANT PREPARATIONS UNDERWAY

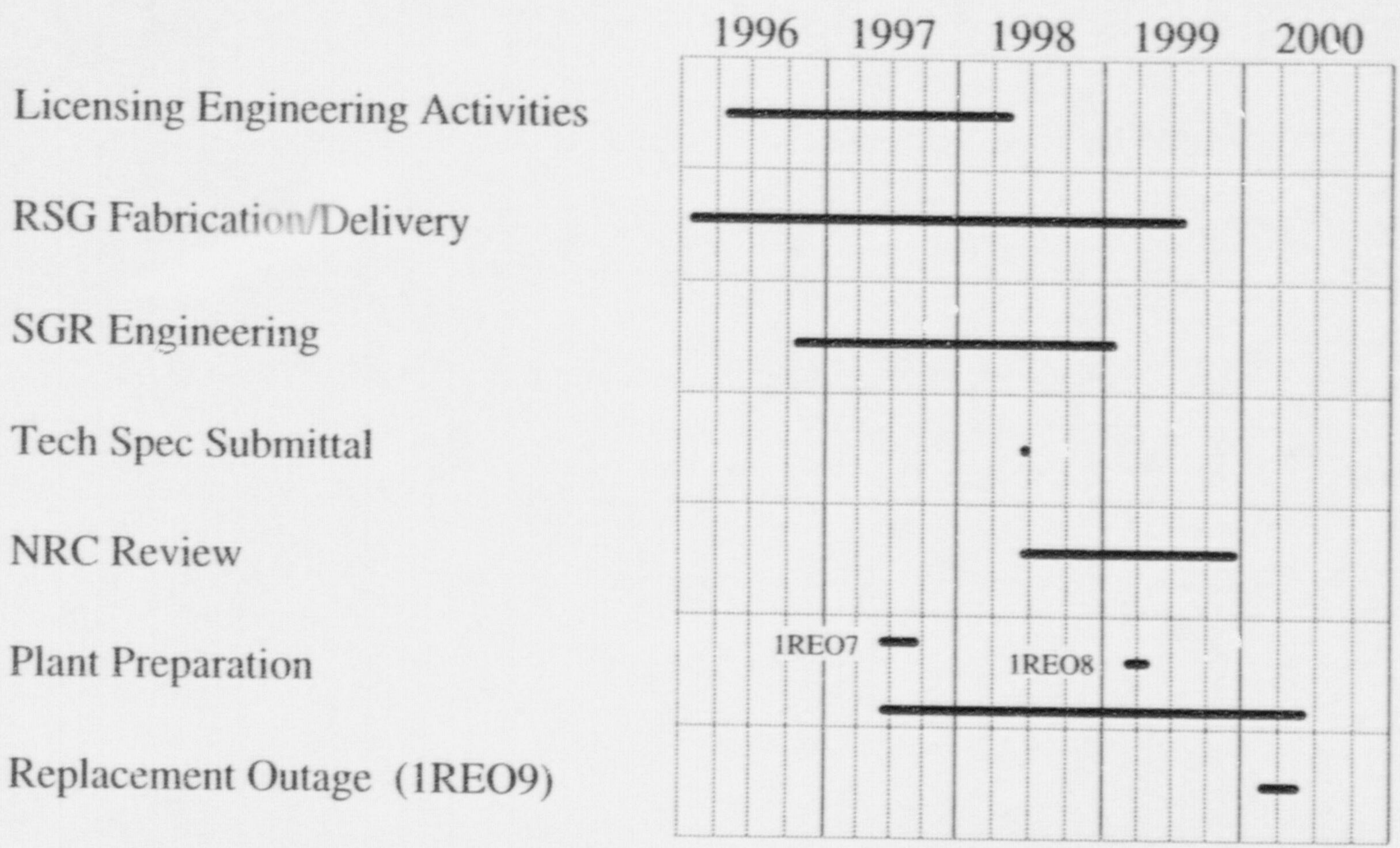
  -- TEMPORARY FACILITIES

  -- BARGE SLIP

  -- CONTRACTOR MOBILIZATION JUNE 1998

SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION
UNIT 1 STEAM GENERATOR REPLACEMENT PROJECT

PROJECT SCHEDULE OVERVIEW



LICENSING ENGINEERING APPROACH

LICENSING ENGINEERING APPROACH

EVALUATIONS PERFORMED TO DEMONSTRATE EXISTING ANALYSES REMAIN BOUNDING WHEN Δ94 DESIGN DOES NOT ADVERSELY AFFECT RESULTS

ANALYSES PERFORMED WHEN Δ94 DESIGN ADVERSELY AFFECTS RESULTS

<u>Δ94 RSG DESIGN CHARACTERISTICS</u>

- INCREASED REACTOR COOLANT FLOW
- INCREASED REACTOR COOLANT VOLUME
- INCREASED HEAT TRANSFER AREA
- INCREASED MASS/WEIGHT
- INCREASED SECONDARY INVENTORY
- ELEVATED FEEDRING INSTEAD OF PREHEATER

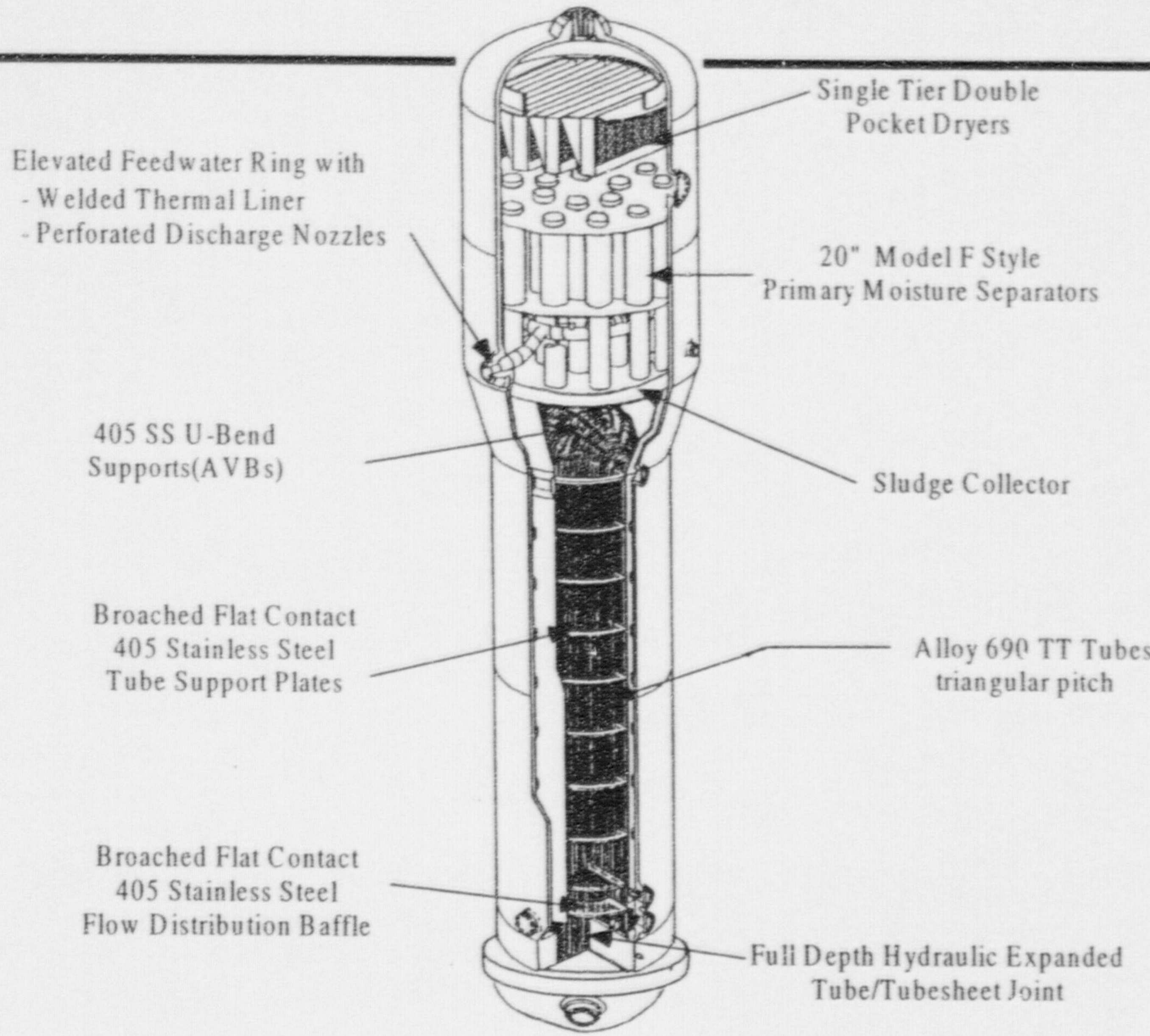
FOLLOW-UP TO APRIL 1997 BRIEFING

1REO7 STEAM GENERATOR TUBE INDICATION SUMMARY

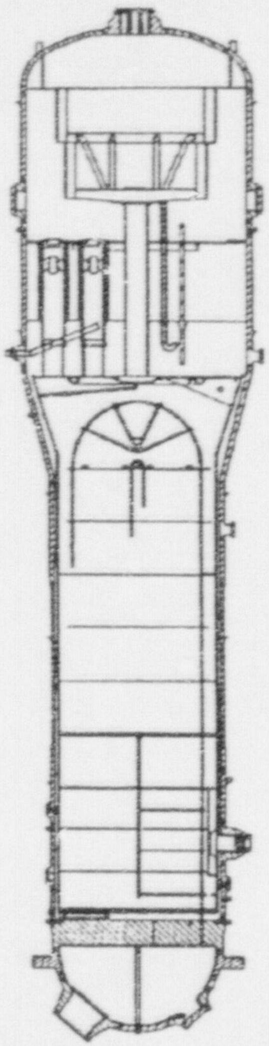
Steam Generator	Circumferential Indications	Axial Indications	Volumetric Indications	Preventive Plugging	Tubes Plugged
A	19[1]	3	8[2]	5	36
B	14	11	3	0	28
C	22	9	14	3	47
D	20	8[3]	15	0	43
Total	75	32	39	8	154

(1) One tube had a circumferential indication and a volumetric indication at the hot leg tubesheet
(2) One tube had a volumetric indication at both the hot and cold leg tubesheet
(3) One tube had an axial indication at the hot leg tubesheet and in the cold leg freespan

Enhanced Design Features

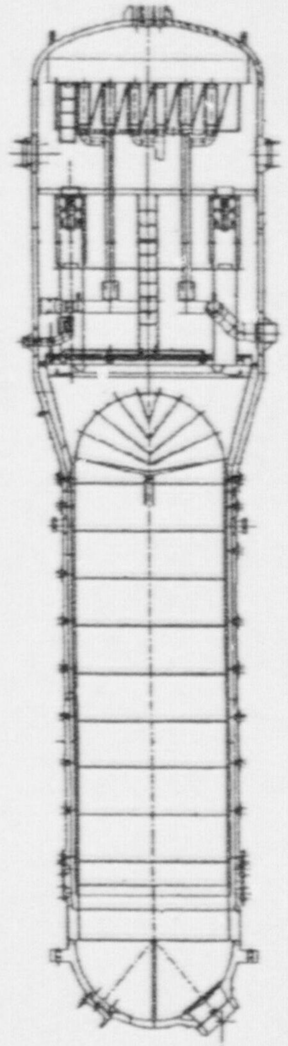


Delta 94/Model E Comparison



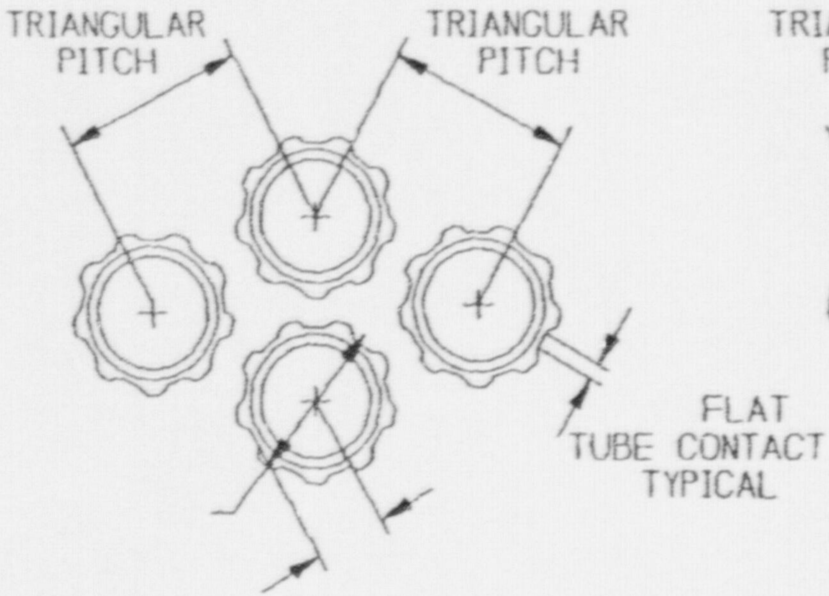
Original Unit

Model E		Delta 94
Preheater	**SG Design**	Feedring
4,864	**No. of Tubes**	
0.750"	**Tube O.D.**	
.043"	**Tube Wall**	.040"
Alloy 600 MA	**Tube Material**	Alloy 690 TT
Square	**Tube Pitch**	Triangular
68,000 ft^2	**H. T. Area**	94,500 ft^2
440"	**Bundle Height**	
3.25"	**Min U-Bend Radius**	3.25"

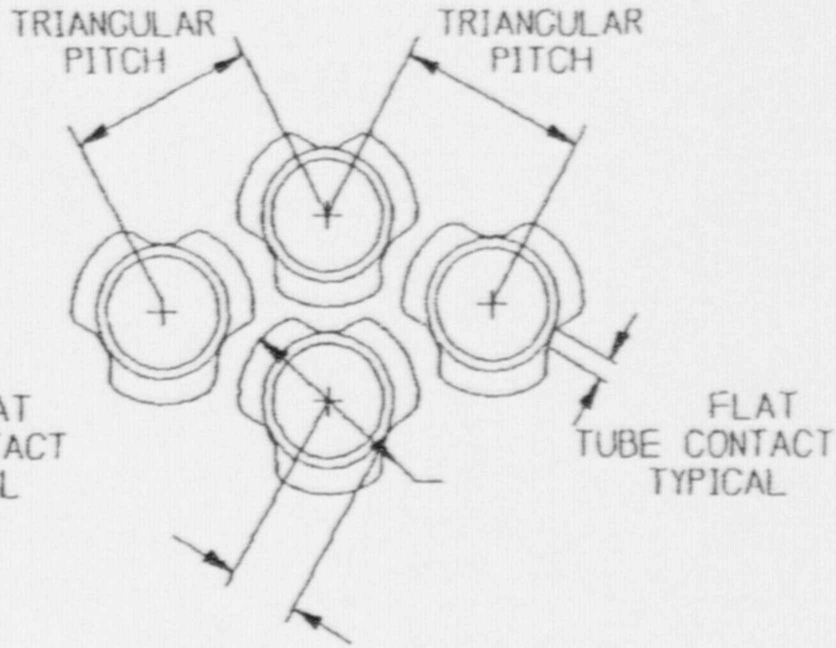


Replacement Unit

Broached Hole Configuration



Flow Distribution Baffle



Tube Support Plate

NON-LOCA ANALYSIS

SCOPE

Analyses

- Feedwater Malfunction
- Steam Line Break
- Loss of Load/Turbine Trip
- Loss of Normal FW/LOOP
- Feedwater Pipe Break
- Fire Hazards/LOOP
- CVCS Malfunction/Boron Dilution

Evaluations

- Loss of RCS Flow
- Locked RCP Rotor
- RCS Depressurization
- Excessive Load Increase
- Rod Withdrawal from Power
- Rod Ejection
- Dropped Rod
- Startup of Inactive Loop
- OT/ΔP, OP/ΔT Trip Setpoints
- Feedwater Temperature Reduction
- Increased Inventory from CVCS
- Inadvertent ECCS Operation

CONCLUSION

RESULTS CONTINUE TO COMPLY WITH EXISTING DESIGN BASES

COMPARISON OF OPERATING PARAMETERS

	Model E	Δ94
Core Power (MWth)	3800	3800
NSSS Nominal Power (MWth)	3817	3821
RCS Pressure (psia)	2250	2250
Thermal Design Flow (gpm)	381,600	392,000
Core Bypass Flow (%)	4.5	8.5
Cold Leg Temperature (°F)	549.0 - 560.4	549.8 - 561.2
Average Temperature (°F)	582.3 - 593.0	582.3 - 593.0
Hot Leg Temperature (°F)	615.6 - 625.6	614.8 - 624.8
No-Load Temperature (°F)	567	567
Pressurizer Level, Hi T_{av} (% span)	25 - 60	25 - 57
Pressurizer Level, Lo T_{av} (% span)	32.9 - 47.0	25 - 40
S/G Temperature (°F)	542.9 - 558.0	539.3 - 552.4
Steam Pressure (psia)	986 - 1115	957 - 1066
S/G Nominal HFP Fluid Mass (lbm)	~137,000	~162,000
Steam Flow (10^6 lbm/hr)	16.86 - 16.96	15.74 - 16.92
Feedwater Temperature (°F)	440	390 - 440
S/G Tube Plugging (%)	0 - 10	0 - 10

NON-LOCA ANALYSIS

<u>Δ94 DESIGN DIFFERENCES AFFECTING NON-LOCA ANALYSIS</u>

- FEEDRING vs. PREHEATER
- INCREASED RCS VOLUME
- INCREASED HEAT TRANSFER SURFACE AREA

<u>INCREASED RCS FLOW</u>

<u>INCREASED CORE BYPASS FLOW</u>

- DELETED THIMBLE PLUGS
- UPPER HEAD T-COLD CONVERSION

<u>REDUCED MINIMUM FEEDWATER TEMPERATURE</u>

NON-LOCA ACCEPTANCE CRITERIA

- MINIMUM DNBR LIMIT
- REACTOR COOLANT SYSTEM OVERPRESSURIZATION
- MAIN STEAM OVERPRESSURIZATION
- PRESSURIZER OVERFILL

SELECTION CRITERIA FOR EVALUATION OR ANALYSIS

- SECONDARY-SIDE INITIATED EVENTS
- APPARENT MARGIN WITH RESPECT TO ACCEPTANCE CRITERIA

RETRAN CODE

- **RETRAN-02/MOD 5.2** CORRECTS ERRORS IN MOD 5.1 (NRC APPROVED)
- IMPLEMENTATION METHODOLOGY IN WCAP-14882

VIPRE CODE

- **VIPRE-01** APPROVED BY NRC
- IMPLEMENTATION METHODOLOGY IN WCAP-14565

RETRAN/VIPRE USED TO REANALYZE

- FEEDWATER MALFUNCTION (INCREASED FEEDWATER FLOW)
- STEAM LINE BREAK (CORE RESPONSE)
- LOSS OF EXTERNAL ELECTRICAL LOAD/TURBINE TRIP
- LOSS OF NORMAL FEEDWATER/LOSS OF OFFSITE POWER
- FEEDWATER PIPE BREAK
- FIRE HAZARDS/LOSS OF OFFSITE POWER

REANALYZED CVCS MALFUNCTION WITH BORON DILUTION EVENT

NON-LOCA EVENT EVALUATIONS

- LOSS OF REACTOR COOLANT SYSTEM FLOW
- LOCKED REACTOR COOLANT PUMP ROTOR
- REACTOR COOLANT SYSTEM DEPRESSURIZATION
- EXCESSIVE LOAD INCREASE
- ROD WITHDRAWAL FROM POWER
- ROD EJECTION
- DROPPED ROD
- STARTUP OF INACTIVE REACTOR COOLANT LOOP
- OVERTEMPERATURE/OVERPOWER ΔT REACTOR TRIP SETPOINTS
- FEEDWATER TEMPERATURE REDUCTION
- INADVERTENT ECCS OPERATION DURING POWER OPERATION
- INCREASED INVENTORY FROM CVCS MALFUNCTION

SUMMARY OF NON-LOCA WORK

FOR WORK COMPLETED TO DATE, RESULTS CONTINUE TO COMPLY WITH EXISTING DESIGN BASES

LOCA

- LARGE BREAK EVALUATED DUE TO PEAK CLADDING TEMPERATURE BENEFIT FROM RSG INCREASED NUMBER OF TUBES
- SMALL BREAK TO BE DISCUSSED IN FUTURE MEETING

PRESSURE/TEMPERATURE ANALYSIS

PRESSURE/TEMPERATURE ANALYSIS

SCOPE

- INCREASED RCS VOLUME MORE LIMITING FOR LOCA M/E RELEASES, POTENTIALLY LEADING TO INCREASED PEAK CONTAINMENT PRESSURE
- INCREASED SECONDARY INVENTORY MORE LIMITING FOR MSLB M/E RELEASES, POTENTIALLY LEADING TO INCREASED PEAK CONTAINMENT TEMPERATURE
- OTHER ISSUES ADDRESSED FOR COMPLETENESS

CONCLUSIONS

- CONTAINMENT P/T RESPONSE CONTINUES TO COMPLY WITH EXISTING EQ LIMITS
- CONTAINMENT PEAK PRESSURE LOWER THAN CURRENT DESIGN BASIS

LOCA LONG-TERM CONTAINMENT P/T ANALYSIS

- M/E RELEASE METHODOLOGY

  SATAN, **REFLOOD**, **FROTH**, AND **EPITOME** COMPUTER CODES

  NRC-APPROVED METHODOLOGY (WCAP-10325-P-A)

    IMPROVED INITIAL CORE STORED ENERGY ESTIMATE

    DECAY HEAT MODEL CHANGED FROM ANS 1971 + 20% TO ANS 1979 + 2σ

    NO SUPERHEAT OCCURS BECAUSE OF CONDENSATION IN S/G TUBES

    IMPROVED ESTIMATE OF SUMP WATER TEMPERATURE IN M/E RELEASES

- P/T METHODOLOGY

  CONTEMPT4/MOD5 COMPUTER CODE

  SATISFIES ANSI/ANS 56.4-1983, NUREG-0800, AND NUREG-0588

  BENCHMARKED AGAINST EXISTING ANALYSIS OF RECORD (**COPATTA** CODE)

- RESULTS

  LOWER PEAK PRESSURE

  LONG-TERM P/T RESPONSE CONFIRMED WITHIN EQ LIMITS

MSLB LONG-TERM CONTAINMENT P/T ANALYSIS

- M/E RELEASE METHODOLOGY

  DOUBLE-ENDED RUPTURE AND SPLIT BREAK

  RETRAN-02/MOD 5.2 USED

- P/T METHODOLOGY

  CONTEMPT4 METHODOLOGY UNCHANGED

- RESULTS

  -- SIMILAR M/E RELEASE RATES

  -- PEAK CONTAINMENT TEMPERATURE WITHIN 1°F

  -- CONTAINMENT TEMPERATURE PROFILE REMAINS BOUNDED BY EXISTING EQ LIMIT

OTHER P/T ISSUES

- SHORT-TERM (PEAK) PRESSURE ANALYSIS
- OUTSIDE CONTAINMENT P/T ANALYSIS
- GENERIC LETTER 96-06 EVALUATION

PIPING AND SUPPORTS ANALYSES

INTRODUCTION TO PIPING AND SUPPORTS ANALYSES

SCOPE

- RCS PIPING
- LARGE BORE SECONDARY PIPING
- AUXILIARY PIPING ATTACHED TO RCS
- PIPING SUPPORTS
- EQUIPMENT SUPPORTS
- BUILDING STRUCTURES

CONCLUSIONS

- CONFIRMED THAT RCS PIPING AND SUPPORTS, EQUIPMENT SUPPORTS, AND SECONDARY PIPING REMAIN IN COMPLIANCE WITH EXISTING DESIGN BASES

REACTOR COOLANT SYSTEM ANALYSES

- RCB SEISMIC ANALYSIS WITH WESTINGHOUSE NSSS MODEL

  -- LESS THAN 1% CHANGE IN TOTAL MASS

  -- CONCLUSIONS

    DYNAMIC CHARACTERISTICS OF RCB AND INTERNAL STRUCTURE UNCHANGED

    DESIGN BASIS RESPONSE SPECTRA FOR RCB AND INTERNAL STRUCTURE UNCHANGED

- RSG FINITE ELEMENT MODEL

  -- RSG MASS ~11% GREATER

  -- MINOR CHANGE IN RSG CENTER OF GRAVITY LOCATION

  -- FREQUENCIES OF OSG AND RSG IN FUNDAMENTAL MODES ARE ANALYTICALLY EQUIVALENT

  -- PARTICIPATION FACTORS FOR OSG AND RSG IN FUNDAMENTAL MODES ARE SIMILAR

  -- CONCLUSIONS

    EXISTING DESIGN BASIS RESPONSE SPECTRA CURVES UNCHANGED

    DESIGN BASIS RESPONSE SPECTRA AT RSG NOZZLES UNCHANGED

LARGE BORE SECONDARY PIPING ANALYSIS

- NOZZLE SPECTRA
- NOZZLE DISPLACEMENT
  - SEISMIC ANCHOR MOVEMENTS
  - THERMAL DISPLACEMENTS
  - LOCA DISPLACEMENTS
- DYNAMIC ANALYSIS REQUIRED WHEN MASS REDISTRIBUTED
- DISPLACEMENTS REVISED

REACTOR COOLANT SYSTEM ANALYSIS

EXISTING PIPING ANALYSIS REVIEWED FOR CHANGES

- REANALYZED FOR RSG INCREASED MASS
- REANALYZED FOR HOT AND COLD LEG TEMPERATURES AND DESIGN BASIS TRANSIENTS
- REANALYZED SEISMIC INERTIA FOR RSG INCREASED MASS
- DESIGN RESPONSE SPECTRA FOR ALL RCB ELEVATIONS UNCHANGED
  - -- USED ENVELOPED RESPONSE SPECTRA
  - -- CRITICAL DAMPING UNCHANGED

LOCA

- SURGE AND ACCUMULATOR NOZZLE BREAK ANALYSES NOT REQUIRED DUE TO LBB IN EXISTING LICENSING BASIS
- REANALYZED LOCA BREAKS TO INCORPORATE RSG HYDRAULIC FORCES
  - -- RESIDUAL HEAT REMOVAL
  - -- SAFETY INJECTION
  - -- PRESSURIZER SPRAY
  - -- NORMAL LETDOWN
  - -- NORMAL CHARGING

PRELIMINARY RESULTS

- RCS PIPING STRESSES REMAIN IN COMPLIANCE WITH EXISTING DESIGN BASES
- EQUIPMENT NOZZLE LOADS REMAIN IN COMPLIANCE WITH EXISTING DESIGN BASES

WORK IN PROGRESS

- ADDITIONAL EVALUATION REQUIRED AFTER COMPARTMENT PRESSURIZATION FORCES ARE FINALIZED
- EVALUATING AUXILIARY LINES FOR INCREASED SEISMIC ANCHOR MOVEMENTS

RCS EQUIPMENT SUPPORTS ANALYSIS

- INCLUDED RSG, RCP, AND REACTOR VESSEL SUPPORTS
- SUPPORTS REANALYZED FOR RSG DEADWEIGHT, THERMAL, SEISMIC, AND PIPE RUPTURE LOADS
- RESULTS

  SUPPORT LOADS INCREASE

  SUPPORTS REMAIN IN COMPLIANCE WITH EXISTING DESIGN BASES

  NO SUPPORT MODIFICATIONS ANTICIPATED

RCB INTERNAL STRUCTURES BEING EVALUATED FOR INCREASED LOADS

LEAK-BEFORE-BREAK EVALUATION

- SRP 3.6.3 METHODOLOGY WITH RCS ANALYSIS LOADS AND PLANT SPECIFIC CMTRs
- PIPE GEOMETRY, LOADING, AND FRACTURE TOUGHNESS DETERMINED CRITICAL LOCATION
- CRITICAL LOCATION REMAINS IN COMPLIANCE WITH SRP 3.6.3 MARGINS
- EXISTING LBB ANALYSIS CONCLUSIONS REMAIN VALID

MAIN STEAM PIPING STRESS EVALUATION

MAIN STEAM PIPING STRESS EVALUATION

EXISTING PIPING ANALYSIS REVIEWED FOR CHANGES

- LOAD COMPONENTS EVALUATED
  - -- WEIGHT
  - -- THERMAL
  - -- SEISMIC INERTIA
  - -- DESIGN BASIS ACCIDENT
  - -- STEAM HAMMER
  - -- LOCA EFFECTS
- SEISMIC ANCHOR MOVEMENTS ANALYZED

PRELIMINARY RESULTS

- SUPPORT LOADS AND PIPING STRESSES REMAIN IN COMPLIANCE WITH EXISTING DESIGN BASES

WORK IN PROGRESS

- SEISMIC ANCHOR MOVEMENTS TO BE REVIEWED AFTER COMPARTMENT PRESSURIZATION FORCES ARE FINALIZED
- IMPACT OF FW/AFW BREAK JET IMPINGEMENT

FEEDWATER/AUXILIARY FEEDWATER PIPING REROUTING

FEEDWATER/AUXILIARY FEEDWATER PIPING REROUTING

INTRODUCTION OF VIDEOTAPE

- REROUTING DUE TO ELEVATED FEEDRING DESIGN AND AUXILIARY FEEDWATER NOZZLE AZIMUTH CHANGE

FW/AF PIPING, SUPPORT, AND MATERIAL MODIFICATIONS

- ADDED NEW FW AND AF SUPPORTS
- MODIFIED EXISTING FW AND AF SUPPORTS
- PIPING MATERIAL CHANGED AT NOZZLES TO PROVIDE ADDITIONAL STRENGTH AND REDUCE FLOW-ACCELERATED CORROSION EFFECTS

SUMMARY OF FW/AF STRESS ANALYSIS

- ANALYSIS PERFORMED TO EXISTING DESIGN BASES (DESIGN CODES, ASSOCIATED CODE CASES, AND COMMITMENTS)
- PIPING AND SUPPORT ANALYSIS RESULTS REMAIN IN COMPLIANCE WITH EXISTING DESIGN BASES
- HELB EFFECTS AT RSG NOZZLE LOCATIONS BEING EVALUATED

NSSS/BOP INTERFACE SYSTEMS EVALUATION

NSSS/BOP INTERFACE SYSTEMS EVALUATION

EVALUATED DUE TO Δ94 OPERATING PARAMETERS

	Model E	Δ94
S/G Temperature (°F)	542.9 - 558.0	539.3 - 552.4
Steam Pressure (psia)	986 - 1115	957 - 1066
Steam Flow (10^6 lbm/hr)	16.40 - 16.96	15.74 - 16.92
Feedwater Temperature (°F)	440	390 - 440
S/G Tube Plugging (%)	0-10	0 - 10

NSSS/BOP INTERFACE COMPONENTS EVALUATED

- MAIN STEAM SAFETY VALVES
- STEAM GENERATOR PORVs
- MAIN STEAM ISOLATION VALVES
- STEAM DUMP VALVES
- FEEDWATER ISOLATION VALVES
- FEEDWATER CONTROL VALVES
- AUXILIARY FEEDWATER STORAGE TANK
- S/G BLOWDOWN FLOW CONTROL VALVES

RESULTS

- EXISTING S/G SAFETY VALVE AND PORV SETPOINTS AND CAPACITIES REMAIN IN COMPLIANCE WITH EXISTING DESIGN BASES
- EXISTING MSIV CLOSURE TIME AND MAXIMUM PIPING LOADS REMAIN BOUNDING
- STEAM DUMP VALVE CAPACITY SUPPORTS 50% DESIGN BASIS LOAD REJECTION WITHOUT REACTOR TRIP
- EXISTING FWIV CLOSURE TIME AND MAXIMUM PIPING LOADS REMAIN BOUNDING
- REQUIRED TECH SPEC MINIMUM AUXILIARY FEEDWATER STORAGE TANK INVENTORY INCREASES SLIGHTLY
- EXISTING S/G BLOWDOWN CONTROL VALVE CAPACITY REMAINS BOUNDING

RCP AND CRDM EVALUATIONS

RCP AND CRDM EVALUATIONS

PRESSURE BOUNDARY CONSIDERATIONS

- MINOR NSSS DESIGN OPERATING PARAMETER RANGE REVISIONS
- REVISIONS TO RCP AND CRDM DESIGN TRANSIENTS DUE TO RSG

DESIGN OPERATING PARAMETER AND DESIGN TRANSIENT REVISIONS

- MINOR CHANGES IN HOT AND COLD LEG OPERATING TEMPERATURES BOUNDED BY EXISTING ANALYSES
- FOUR UPSET AND THREE FAULTED DESIGN TRANSIENTS POTENTIALLY IMPACTED RCP AND CRDM
- TRANSIENT REVISIONS ENVELOPED BY CONSERVATIVE RANGE OF TRANSIENT CONDITIONS IN EXISTING DOCUMENTATION

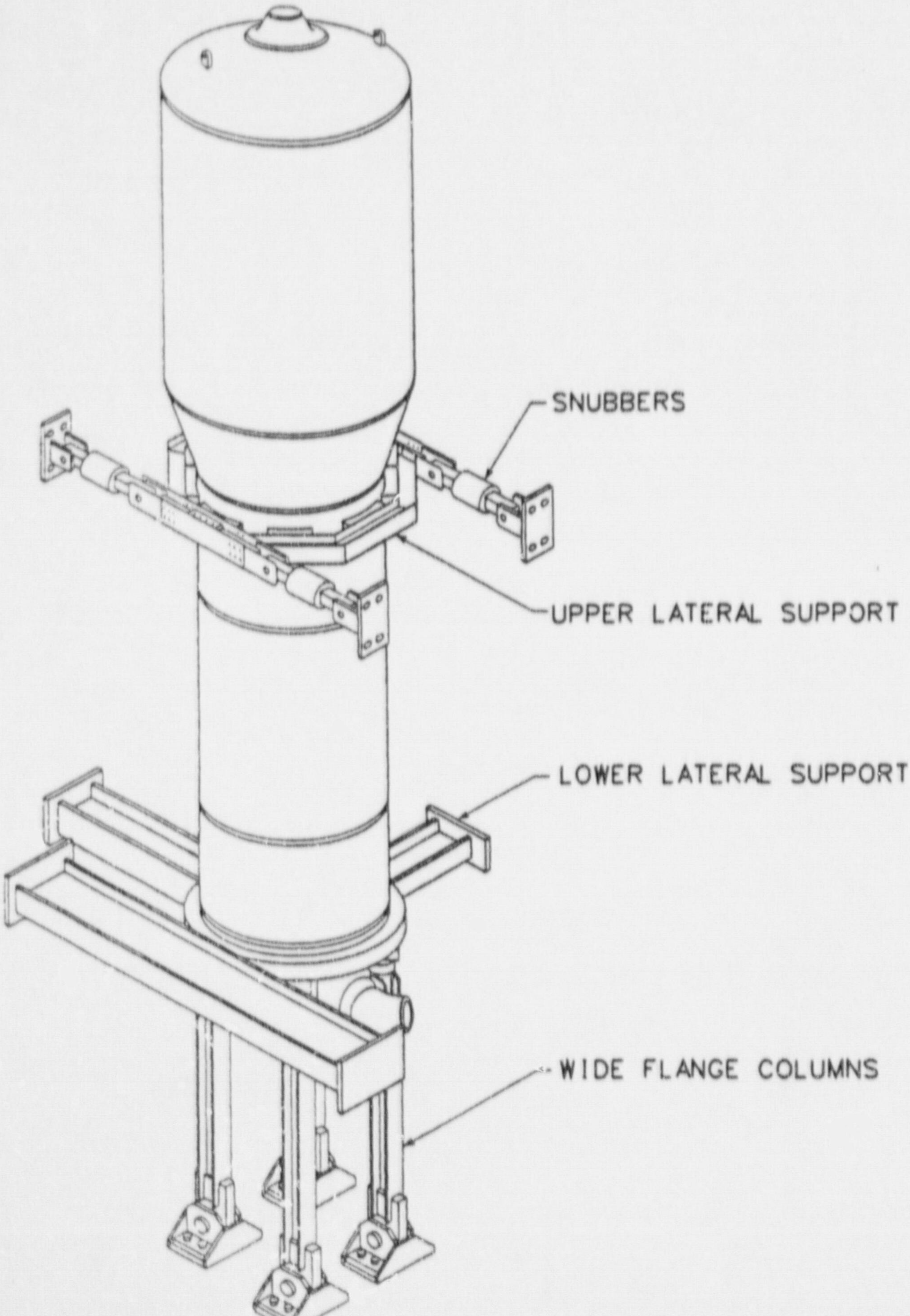
RCP BRAKE HORSEPOWER AND HYDRAULIC THRUST

- BRAKE HORSEPOWER REDUCED SLIGHTLY
- HYDRAULIC THRUST LOAD INCREASE INCLUDED IN ANALYSIS

CONCLUSION

RCPs AND CRDMs REMAIN IN COMPLIANCE WITH DESIGN BASES

STEAM GENERATOR REPLACEMENT SEQUENCE


SNUBBERS
UPPER LATERAL SUPPORT
LOWER LATERAL SUPPORT
WIDE FLANGE COLUMNS

M0502.001

STEAM GENERATOR REPLACEMENT SEQUENCE

OSG REMOVAL – MODE 6 AND DEFUELED MODE

- REMOVE S/G AND RC PIPING INSULATION
- CUT SECONDARY PIPING AT S/G NOZZLES
- REMOVE S/G UPPER SUPPORT SNUBBERS
- UNBOLT S/G UPPER SUPPORT RING GIRDERS AT SPLICE PLATE
- INSTALL TEMPORARY S/G SUPPORTS (WEDGES, SCREWJACKS, AND CABLES) AT LOWER AND UPPER LATERAL SUPPORTS
- INSTALL TEMPORARY S/G COLUMN SUPPORTS
- INSTALL TEMPORARY HOT LEG DEADWEIGHT SUPPORTS
- INSTALL TEMPORARY CROSSOVER LEG DEADWEIGHT SUPPORTS AND TIE-DOWNS
- INSTALL TEMPORARY FW/AF PIPING SUPPORTS

OSG REMOVAL

- CUT PRIMARY PIPING AT S/G INLET AND OUTLET NOZZLES
- ATTACH TEMPORARY LIFTING DEVICE
- REMOVE COLUMN-TO-LOWER SUPPORT FLANGE BOLTS
- REMOVE TEMPORARY S/G SUPPORTS
- RIG S/G OUT OF CUBICLE

REACTOR COOLANT SYSTEM PIPING PREPARATION

- DECONTAMINATE PIPING
- TEMPLATE PIPING FOR NOZZLE WELD
- TEMPLATE RSG NOZZLES
- DETERMINE WELD PREP LOCATIONS BASED ON BEST FIT TEMPLATING
- MACHINE PIPING FOR S/G NOZZLE WELDS

RSG INSTALLATION SEQUENCE

- RIG S/G INTO CUBICLE
- SECURE BEST FIT CONDITIONS BETWEEN PIPING AND NOZZLES
- SHIM COLUMN SUPPORTS; BOLT LOWER SUPPORT FLANGE
- INSTALL TEMPORARY RESTRAINTS
- REMOVE TEMPORARY LIFTING DEVICE
- MAKE INITIAL WELD PASS AT NOZZLES
- COMPLETE FIRST THIRD OF REQUIRED WELDING
- REMOVE TEMPORARY SUPPORTS AS WELDING IS COMPLETED
- CONNECT MS, FW, SGBD PIPING AND INSTRUMENTATION
- INSTALL UPPER RING GIRDER SUPPORTS, SPLICE CONNECTIONS, SHIMMING, HYDRAULIC SNUBBERS

THERMAL EXPANSION TESTING

MONITOR/ADJUST THERMAL GAPS DURING PLANT HEATUP

- REACTOR COOLANT PUMP SUPPORTS
- S/G UPPER AND LOWER LATERALS
- REACTOR COOLANT PUMP TIE RODS

LICENSING APPROACH

LICENSING APPROACH

TECH SPEC CHANGE TO BE SUBMITTED MAY 1998

- REACTOR COOLANT SYSTEM FLOW
- REACTOR COOLANT SYSTEM VOLUME
- STEAM GENERATOR WATER LEVEL
- AUXILIARY FEEDWATER STORAGE TANK VOLUME
- STEAM GENERATOR SURVEILLANCES

TECH SPEC CHANGE PACKAGE APPLIES TO BOTH UNITS

MODIFICATIONS TO BE EVALUATED UNDER 10CFR50.59

UNREVIEWED SAFETY QUESTION ON DOSE CONSEQUENCES WILL BE SUBMITTED

RETRAN WCAP-14882 AND **VIPRE** WCAP-14565 UNDER NRC REVIEW