UPRATING OF CANDU REACTORS - A FEASIBILITY STUDY

S.S. Dua, Manager
Quality Department
AECL CANDU
Sheridan Park Research Community
Mississauga, Ontario, L5K 1B2

T. Lee
Stress Analyst
Fuel Channels Department
Technical Division
AECL CANDU

J. Mistry
Design Engineer Specialist
Nuclear Engineering Department
Ontario Hydro

C. Lee
Design Engineer Specialist
Nuclear Safety Department
Ontario Hydro

1. ABSTRACT

A feasibility study was undertaken to determine if Pickering B reactors could be uprated to provide extra power without any major equipment modifications or replacements. The power uprating for the study phase has been defined as 103% of the normal full power output achieved by changing the operating flux in the reactor. The power distribution towards the core periphery is both flattened and increased which mainly affects the thermal conditions in the reactor.

The paper provides an overview of the key systems and components reviewed for the uprating parameters. Although the main focus of this study was to qualify the existing design and equipment for the uprated power conditions (103% of full power), the program did make in-depth assessment of the important safety considerations for jurisdictional approval.

The results of the study phase indicate that the power uprating of the Pickering B reactors to 103% power is feasible without any major design changes to the key systems and components. It is also shown that there are no safety related impediments to uprating, and that this uprating can be accomplished with the present safety system equipment. The study does indicate the need for some additional assessment for a few highly stressed components before this program is implemented.

2. INTRODUCTION

This feasibility study for the uprating of Pickering B reactors was started in mid 1987. The objective was to determine if Pickering B reactors could be uprated to provide extra power without any major equipment modifications or replacements. A brief economic assessment concluded that uprating is cost effective if it can be achieved without any major design changes to the equipment and systems.

To minimize the need for changes to the current systems, several constraints to the uprated operating parameters were adopted at the start of this study such as:

- no increase in the nominal reactor outlet header pressure;
- no changes to the channel and bundle power license limits;
- no bulk boiling in any of the channels, outlet feeders and outlet headers.

After an initial assessment the limit for power uprating was defined as 103% of the normal full power (FP) output. The main factors limiting uprating to 103% FP are generator temperature limits, turbine blade stress levels and reactor channel outlet temperature limits. This 3% power increase can be achieved by changing the operating flux in the reactor such that the power distribution towards the reactor core periphery can be flattened and increased. For the revised flux, the reactor physics and thermohydraulic calculations were performed to determine the heat disposition rates and total heating in the reflector, the calandria, the vault water, the vault liner and concrete, end shields and calandria tubes.

Based on the reactor physics and thermohydraulic calculations, various key systems and components were assessed to determine their ability to handle the additional heat loads due to 103% reactor power. The major systems considered in this study include various process systems such as the heat transport system, pressure and inventory control system, moderator and shield cooling system, etc., and the special safety systems. These systems were analyzed for the steady state and enveloping transient conditions. The results of these analyses in terms of bulk and peak water temperatures were used to check the structural integrity of the calandria and its components.

Although the slightly higher flux associated with 103% uprated conditions does not have any significant effect on the fuel channel creep, a reassessment of the channel creep was done based on the latest creep predictions. Since the channel creep predictions based on latest equations were about 30-50% higher for some of the critical channel locations, this required an assessment of the feeders and fuel channels.

This paper provides an overview of the feasibility study related to the nuclear stream supply system (NSSS) of the Pickering 'B' reactor.

3. OVERVIEW OF MAJOR SCOPE ITEMS

The following provides a brief assessment of the major systems and equipment (Figure 1) covered by the feasibility study.

3.1 Reactor Physics

Reactor physics calculations were carried out to determine the heat disposition rate distributions and the total heating in the calandria assembly components, D_2O moderator reflector and vault water of the shield coolant system at 103% of full power operation.

3.2 Process Systems

3.2.1 Heat Transport System (HTS). Heat transport operating conditions at 103% power were simulated using SOPHT computer program. The results of simulations show very minor changes in the HTS operating conditions for a 3% increase in reactor power. The reactor outlet header temperature increases by 1°C; the reactor inlet header temperature decreases by 0.5°C and the HTS pressure remains unchanged.

An assessment of the HTS service limits concluded that there is sufficient conservatism in the existing service limits to envelope the 103% uprated conditions.

3.2.2 Pressure and Inventory Control System. Although the Heat Transport System (HTS) swell depends on HTS temperature, the average change in temperature due to uprating is negligible and hence has no effect on the swell rate and total swell of the HTS.

Uprating therefore has no appreciable effect on the pressure and inventory control system.

3.2.3 Moderator and Shield Cooling Systems. Based on the heat disposition rates obtained from reactor physics calculations, the moderator and shield cooling systems were analyzed for the following conditions:

- Steady state full power operation at 103%.
- Complete loss of shield cooling pumps.
- Loss of Class IV power.
- Partial loss of service water flow to the moderator heat exchangers.
- Partial loss of moderator pumping.

Based on the understanding of the original (100% FP) design, and an engineering judgement on the expected structural behaviour under the uprated power conditions, it was decided to limit the analyses for these systems to the following two enveloping conditions:

a) 103% Full Power Steady State (FPSS)
b) Complete Loss of Shield Cooling Pumps (LSCP)

The analysis results in terms of the average and transient bulk water temperatures were used for the calandria thermal and structural analysis.

3.3 Safety Analysis

Instead of analyzing all the design basis events for the uprated conditions, a review of all accident scenarios considered in the Safety Report was first carried out to identify accidents whose trip coverage and consequences may be affected by changes in the initial operating conditions.

A preliminary safety analysis was then performed for these accidents to determine whether the existing safety analysis is capable of mitigating the consequences of postulated accident scenarios during operation under the uprated conditions. The following cases were identified for further analyses:

a) loss of flow

Only the total loss of Class IV power case was analyzed as it represents the limiting case for heat transport system overpressure.

b) loss of regulation

These accidents were assessed as the flux shapes and reactor header conditions are somewhat different at the uprated power level.

c) small break, loss of coolant accident (LOCA)

Even though the accident scenarios considered in the safety report are not expected to be affected by the slight increase in the reactor power, these accidents were analyzed to confirm the effectiveness of the special safety systems for the entire range of small break LOCAs at the uprated conditions.

d) large break, loss of coolant

The large break LOCA was analyzed because operation at an increased power level tends to:

(i) increase the core voiding rate;
(ii) increase the fission product source term as a result of a larger number of channels in the high channel power groups;
(iii) increase the steady state and transient heat loads to the moderator;
(iv) increase the peak overpressure and the duration of containment overpressure due to the higher stored energy in the heat transport system.

As a result, a detailed assessment was performed to evaluate the required moderator outlet temperature in order to ensure channel integrity, and to demonstrate that the radiological consequences meet the appropriate Siting Guide dose limits.

e) steam and feedwater supply system failure.

Steam and feedwater supply system failures were analyzed to demonstrate that trip effectiveness is maintained and that containment peak overpressure following a large steam main break is within acceptable limits.

Except for the large break LOCA, the analysis was performed at 106.2 percent full power (1764 MW th) which is about 3% above the desired uprated power of 103% FP. This 3% additional power is included, as in all recent safety analysis, to allow for control variations and heat balance uncertainties. For the large break LOCA, the initial reactor power was assumed to be at 110% FP (including control and heat balance uncertainties). This limiting accident analysis was performed at 110% FP to show that, even at a power level above the targeted uprating power level, large safety margins are available in terms of meeting Siting Guide dose limits and maintaining fuel channel integrity.

The preliminary safety analysis demonstrates that there are no safety related impediments to uprating Pickering NGS B, and that this uprating can be accomplished with the present safety system equipment. Existing safety system operating setpoints can remain unchanged, except for the neutron overpower trip setpoint, which is decreased by an acceptable 2% FP from the setpoint for 100% FP. Trip coverage at the uprated power level (103% FP) is similar to that at 100% FP, and has been shown to be acceptable at the uprated power level for all classes of accident considered in the Safety Report.

Accidents expected to be strongly influenced by the increase in power level have been considered in detail and shown to have consequences within allowable limits. Overpressures following loss of flow and turbine trip events are within the limits defined by the ASME code, and sufficient heat sink exists following secondary side accidents at the uprated full power to allow ample time for provision of an alternative heat sink.

The maximum dose values that may result from the limiting accidents of a large break LOCA are also demonstrated to be well below the Siting Guide Limits.

3.4 Structural Analysis

3.4.1 Calandria Assembly Analysis. The loads for the uprated conditions in terms of moderator/shield coolant and metal temperatures, piping and nozzle loads, accident analysis loads etc., were compared with the original loads and it was concluded that some of the uprated loads for the calandria assembly (Figure 2) are more severe than that at 100% full power (FP).

The analysis for the calandria assembly was therefore performed for the following conditions (Figure 3).

a) Full Power Steady State at 103% power (FPSS)
b) Complete Loss of Shield Cooling Pumps (LSCP)
c) Pressure Tube Burst (in-core LOCA)
d) Pressure Tube Burst with a SDE at 24 hours
e) Header Failure (large LOCA)

Finite element analysis technique (Figure 4) were employed using STARDYNE (Ref. 1) and MARC (Ref. 2) computer codes and stresses at critical locations were calculated. Based on the provisions of the ASME code, the stresses for the uprated conditions were categorized and compared against the service limits for Normal, Upset and Emergency Conditions (levels A, B and C as per the current ASME code). The results of the uprated analysis are summarized in the Calandria Vessel and End Shields Design Reports and it is shown that stresses are acceptable.

3.4.2 Fuel Channels Assessment. As the HTS service limits are not affected by the uprated conditions, there is no direct effect on the fuel channel design. However, for the uprated conditions, it was recommended to reassess the predicted creep for the fuel channels based on the latest creep equations. Fuel channel creep analysis was performed and channel loads affected by uprating such as garter spring loads, feeder loads, fuelling machine loads, end fitting bearings and positioning assembly loads were recalculated (Figure 5).

The stresses at the critical locations of the fuel channel assembly were calculated and compared against the ASME code allowables. The Design Report, as per the requirements of the ASME codes and CSA standards, was prepared and it is concluded that the fuel channel pressure boundary, except for the feeder coupling, is not affected by these loads (Ref. 6).

For the feeder coupling, a leak test is recommended to obtain a high level of confidence that the Pickering B joints meet the design criteria as per the original design specifications.

3.4.3 Feeders/Headers Assessment. An assessment of the feeder pipes for the uprated loads including the higher creep predictions for the fuel channels was made (Figure 6). Although there are negligible changes in the pressure and temperature of the HTS, the predicted movements of the feeders at the feeder couplings increase significantly due to higher creep associated with the latest creep equations.

Thirty (30) representative feeders on the west face of the reactor were analyzed for the 103% uprated conditions to cover all significant design parameters and variations. The assessment indicates that nine (9) representative feeders exceed the ASME code allowable limit of 3 Sm. This assessment is conservative and it is felt that a detailed finite element analysis can provide a more accurate stress distribution in the critical areas. Based on the work done on other projects, it seems that a more realistic finite element analysis will be able to qualify these feeders without any design changes.

The feeders assessment also examined the feeder clearances and concluded that the changes due to uprating will require only minor adjustments to ensure that feeder-to-feeder fretting is avoided toward the end of reactor life.

The stresses in the headers are within the allowable limits.

3.4.4 Boiler Safety and Atmospheric Steam Reject Valves. For the uprated conditions, the total steam flow rate from the boilers will increase. This assessment was done to check if the boiler safety and atmospheric steam reject valves have sufficient capacity to handle the increased steam flow rate. It is shown that both boiler safety valves (16) and atmospheric steam reject valves (10) have adequate capacity for the 103% uprated conditions.

4. CONCLUSIONS

The feasibility study reported in this paper is limited to the nuclear steam supply system (NSSS) of the Pickering B reactors. The results show that power uprating of the Pickering B reactors to 103% power is feasible without any major design changes to the key systems and components. The preliminary safety assessment covered by this study indicates that uprating can be accomplished with the present safety system equipment. It is expected that uprating will reduce the present neutron overpower (NOP) trip margins. To operate the reactor at the present 100% FP reliability level, it will be necessary to provide the operators with an effective means of monitoring the status of SDS#1 and SDS#2 NOP trip channels.

The study also indicates the need for some additional assessment of a few components prior to the start of an implementation program for the uprating of Pickering B reactors.

The current station activities are concentrated on improvements in reactor safety and reliability at the 100% power level rather than uprating. Nevertheless, some important prerequisite work is proceeding and the actual timing for uprating is dependent on the availability of station resources and the major work priorities.

ACKNOWLEDGEMENTS

The authors gratefully acknowledge the efforts of R.J. Dee, R. Lior, J. Sherin, J. Wills, A. Usmani, A. Merritt and several other Ontario Hydro and AECL CANDU staff who contributed towards the success of this study.

References

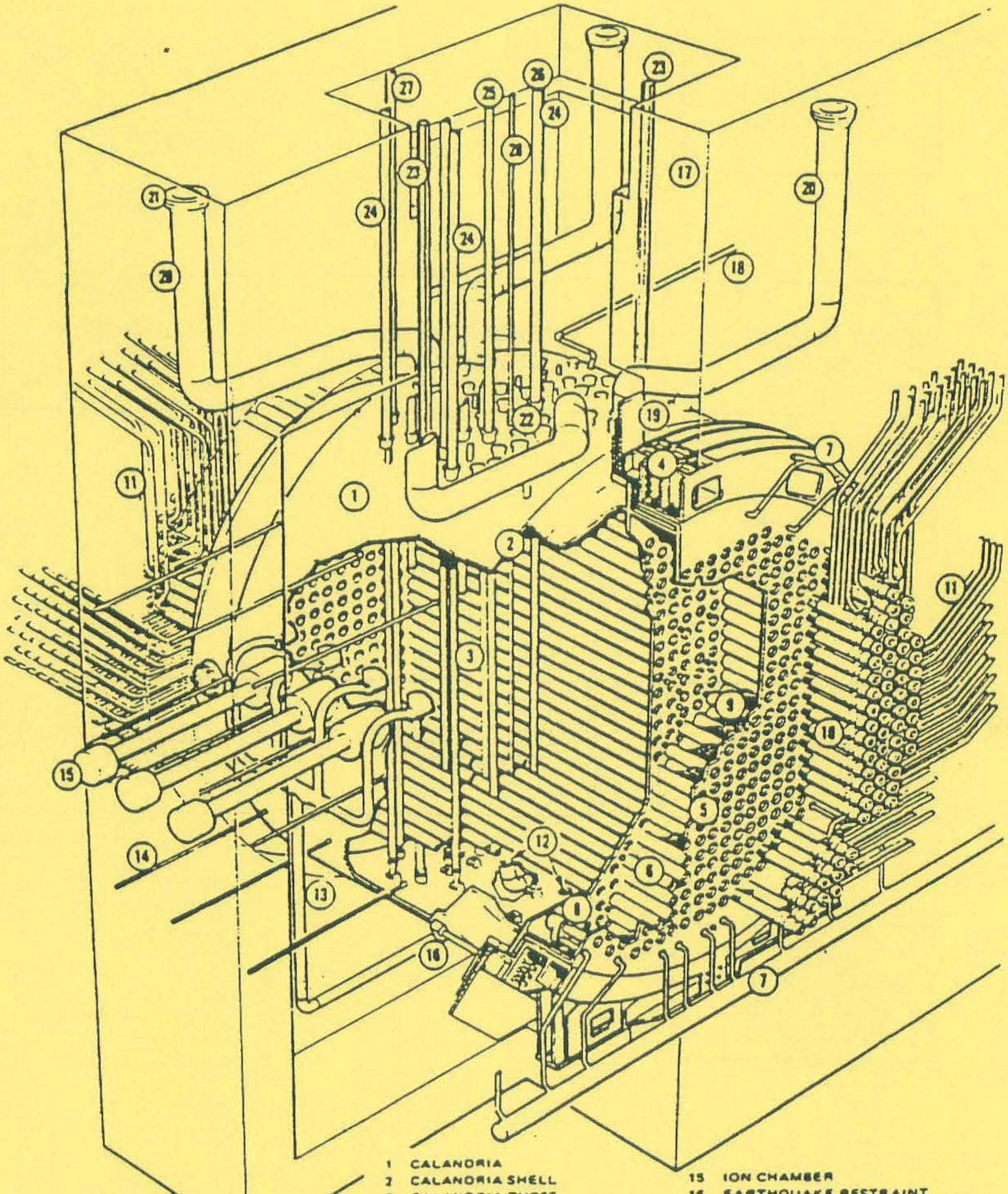
1. STARDYNE User Information Manual, Revision E, 1984, Control Data Corporation

2. MARC User Information Manuals, Revision K3, July 1988, Marc Analysis Research Corporation

REACTOR PHYSICS	REACTOR PHYSICS		
SYSTEMS	HEAT TRANSPORT SYSTEM	PRESSURE AND INVENTORY CONTROL	MODERATOR SYSTEM
	SHIELD COOLING SYSTEM		
EQUIPMENT	CALANDRIA ASSEMBLY	FUEL CHANNEL ASSEMBLY	FEEDERS/ HEADERS
SAFETY	SAFETY ASSESSMENT OF POSTULATED ACCIDENTS		

FIGURE 1

UPRATING STUDY PHASE - MAJOR SCOPE ITEMS



1 CALANDRIA
2 CALANDRIA SHELL
3 CALANDRIA TUBES
4 EMBEDMENT RING
5 FUELLING TUBESHEET
6 END SHIELD LATTICE TUBES
7 END SHIELD COOLING PIPES
8 INLET-OUTLET STRAINER
9 STEEL BALL SHIELDING
10 END FITTINGS
11 FEEDER PIPES
12 MODERATOR OUTLET
13 MODERATOR INLET
14 FLUX DETECTOR AND LIQUID INJECTION NOZZLE
15 ION CHAMBER
16 EARTHQUAKE RESTRAINT
17 VAULT WALL
18 MODERATOR EXPANSION TO HEAD TANK
19 CURTAIN SHIELDING SLABS
20 PRESSURE RELIEF PIPES
21 PRESSURE RELIEF DISC
22 REACTIVITY CONTROL ROD NOZZLES
23 VIEWING PORT
24 SHUTOFF ROD
25 ADJUSTER ROD
26 CONTROL ABSORBER ROD
27 ZONE CONTROL ROD
28 VERTICAL FLUX DETECTOR

FIGURE 2 REACTOR ASSEMBLY

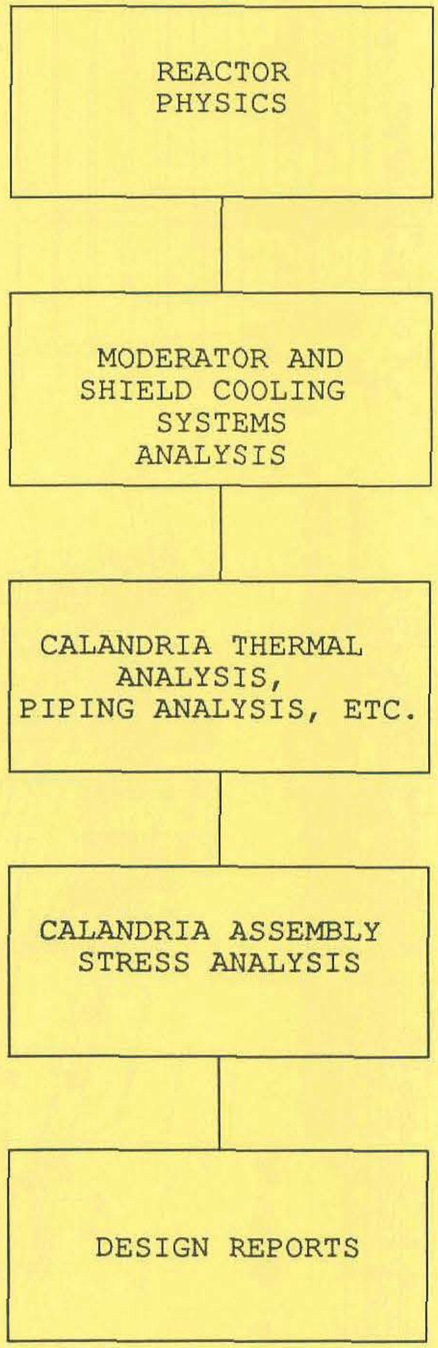
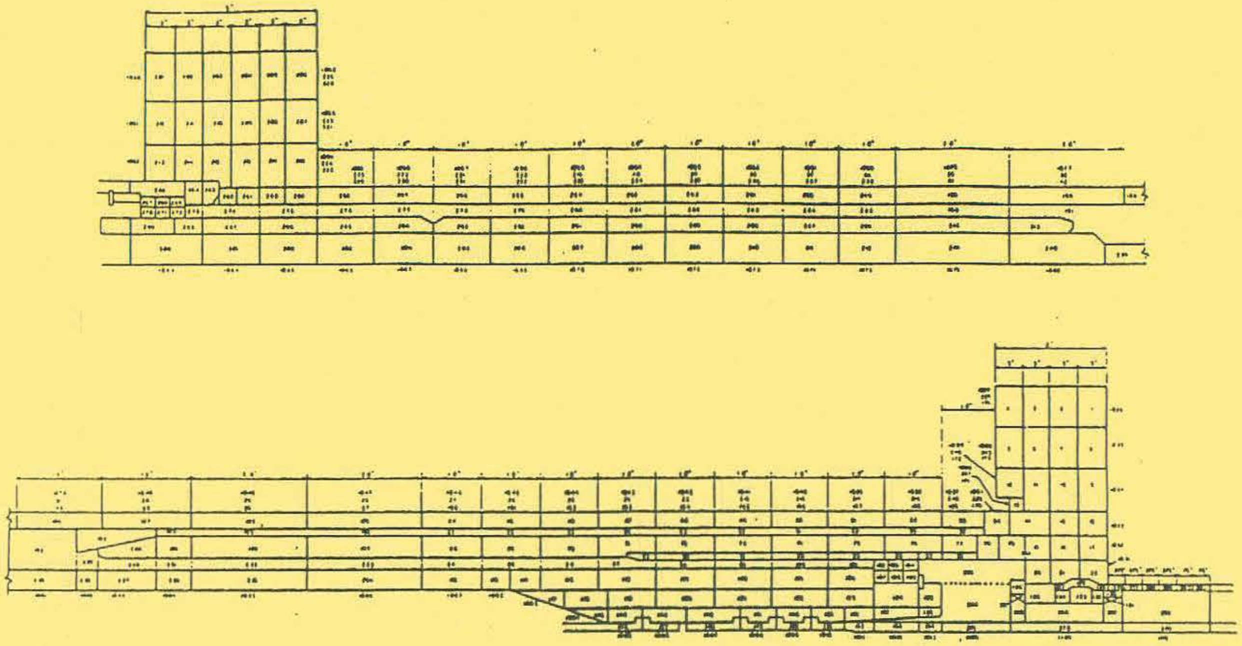
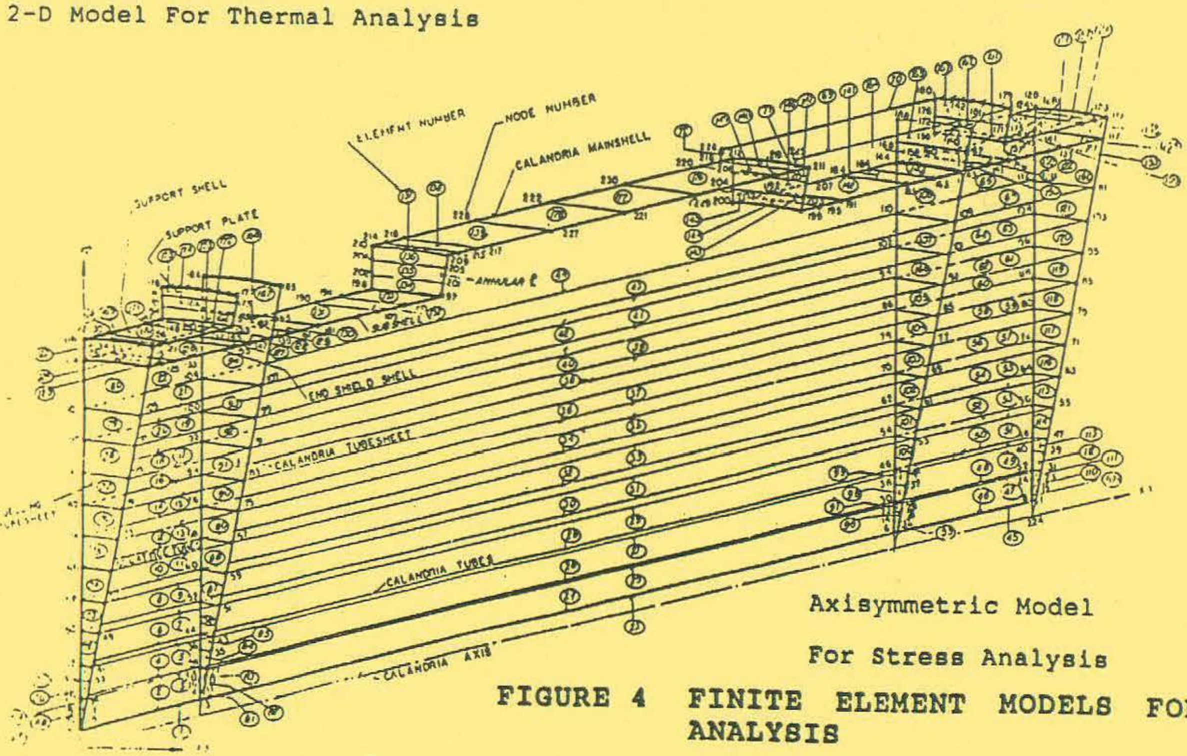


FIGURE 3

CALANDRIA ASSEMBLY STRESS ANALYSIS FOR UPRATING STUDY PHASE

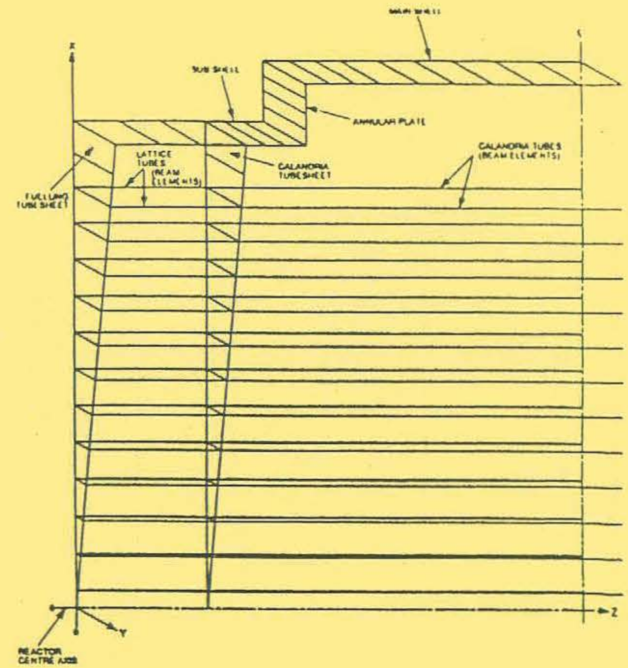


2-D Model For Thermal Analysis



Axisymmetric Model

For Stress Analysis



Axisymmetric Sector Model

For Elastic-Plastic Analysis

FIGURE 4 FINITE ELEMENT MODELS FOR CALANDRIA ASSEMBLY STRESS ANALYSIS

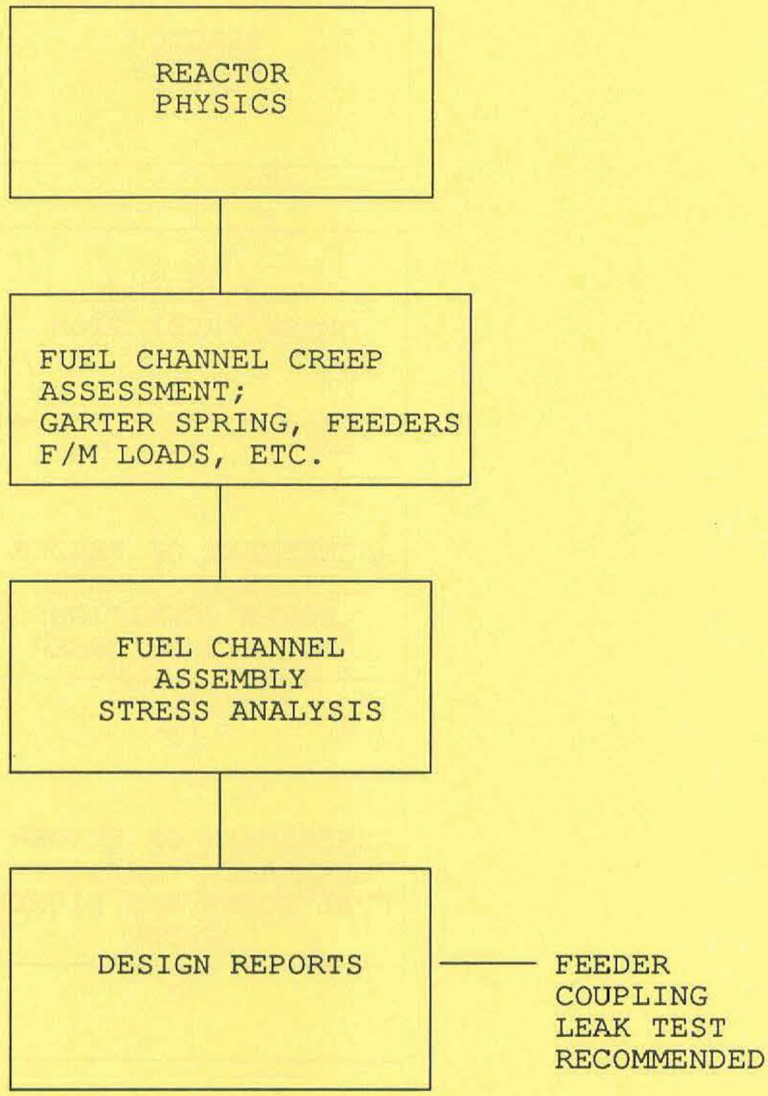


FIGURE 5

FUEL CHANNEL ASSEMBLY STRESS ANALYSIS
FOR UPRATING STUDY PHASE

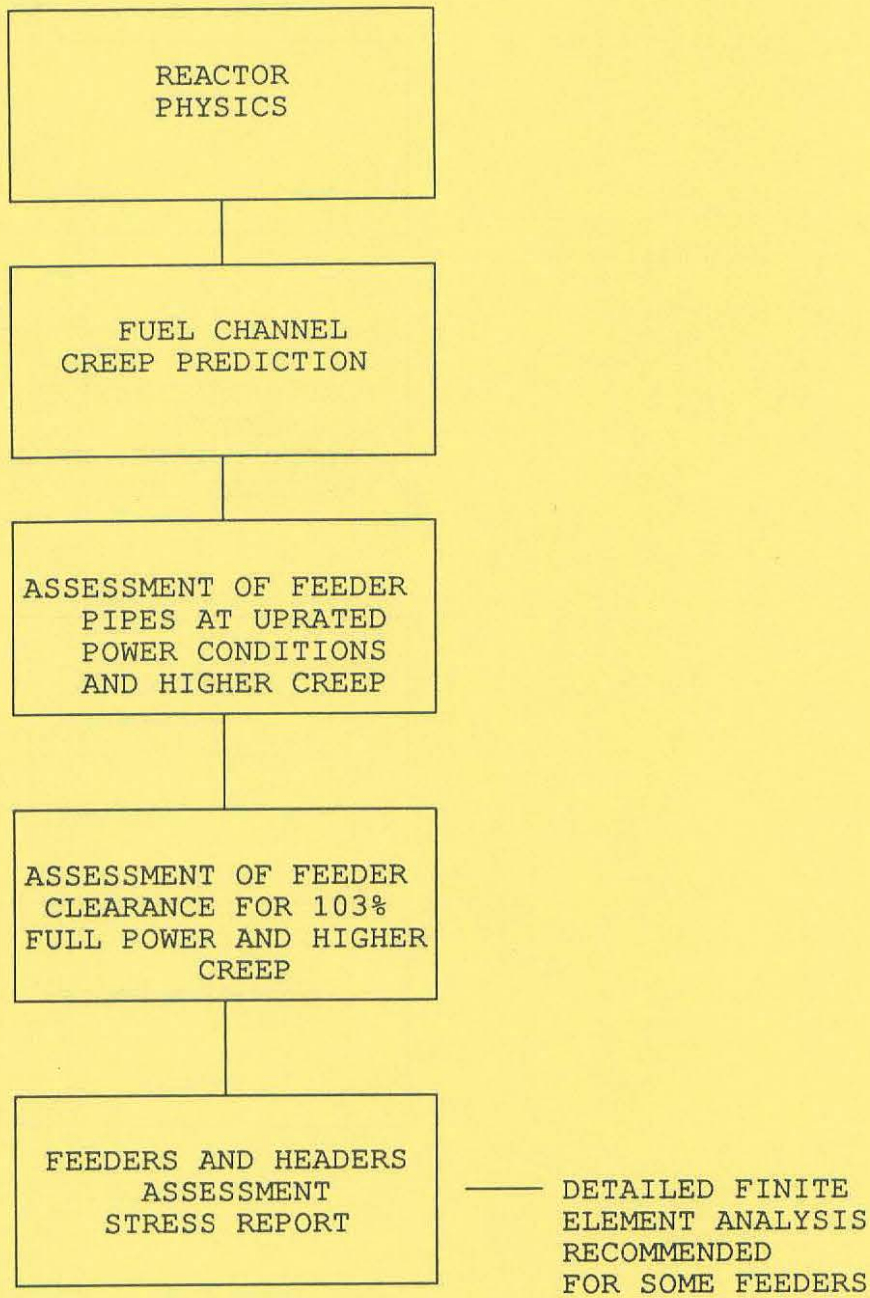


FIGURE 6

FEEDERS/HEADERS ASSESSMENT FOR UPRATING
STUDY PHASE