PROBLEMS OF STRUCTURAL MECHANICS IN NUCLEAR DESIGN

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Presented at the Symposium on Structural Mechanics, Bombay, 20-22 March 1975

Design
1-SM-Theme/75

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A B S T R A C T

A very careful and detailed stress analysis of nuclear pressure vessels and components is essential for ensuring the safety and integrity of nuclear power plants. The nuclear designer therefore relies heavily on Structural Mechanics for application of the most advanced stress analysis techniques to practical design problems. The paper reviews the inter-relation between Structural Mechanics and Nuclear Design and discusses a few of the specific structural mechanics problems faced by the nuclear designers in DAE.

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1. INTRODUCTION

In the first session of this Seminar where various aspects of Structural Mechanics are going to be discussed, it will be appropriate to review the relation between Structural Mechanics and the design of nuclear pressure vessels and then to introduce a few of the specific structural mechanics problems faced by the nuclear designers in DAE.

At present, there are three design teams in DAE which are engaged in the design of nuclear reactor pressure vessels and other related components. The Power Projects Engineering Division has been constructing 200 MW (electrical) capacity heavy water moderated and cooled 'CANDU' reactors of Canadian design since about 1965. It is now preparing designs involving substantial changes in the basic CANDU 200 MWe unit to make it simpler to build and to extrapolate it to larger sizes. The Reactor Research Centre, Madras is building a 42.5 MW(thermal) fast breeder test reactor (FBTR) at Madras and is planning to design a 500 MWe prototype fast breeder reactor(PFBR). In BARC, a research reactor called as 'R5' with a capacity of 100 MW (thermal) is being designed and built. Descriptions of these reactors are given elsewhere (Ref. 1,2,3.)

Typical pressures and temperatures of the main coolants in the three reactors are as follows:

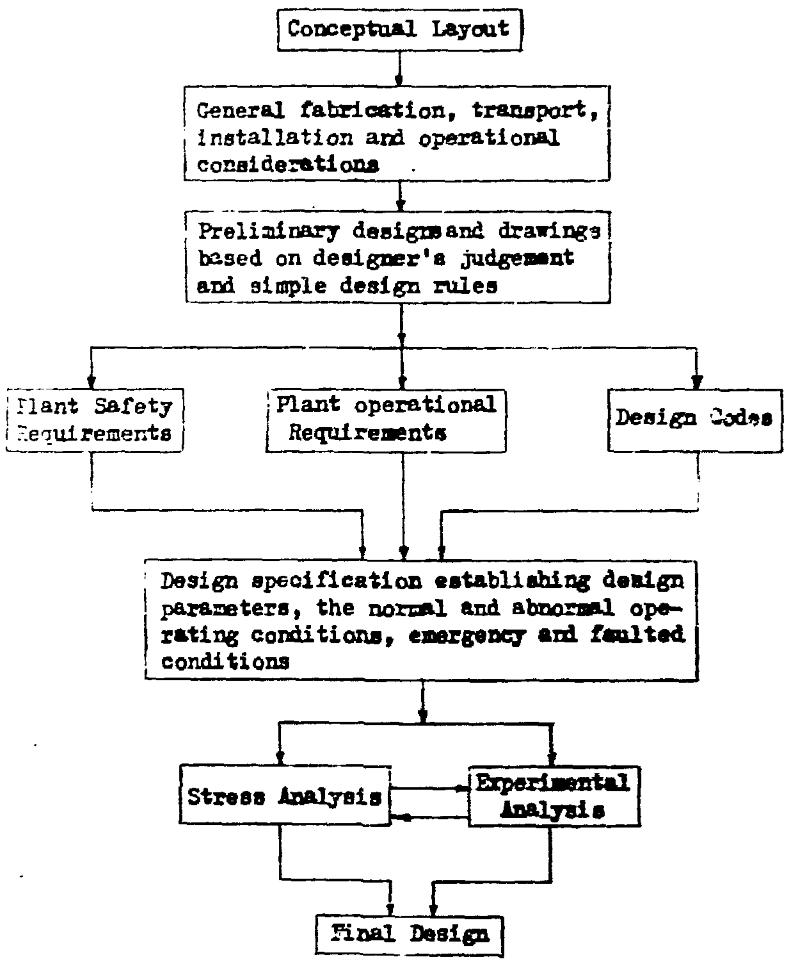
Reactor	Type of coolant	Coolant pressure Kg/cm^2	Coolant temperature °C
CANDU	Heavy water	110	300
Fast Reactor	Liquid sodium	4	580
R-5	Heavy Water	7	67.5

CANDU is a pressure tube type of reactor since the high pressure coolant flows through Zirconium alloy pressure tubes in the reactor whereas the main reactor vessel known as the calandria contains heavy water moderator at a low pressure of 0.73 kg/cm^2 and a temperature of 55°C. In FBTR, liquid sodium is contained by the main reactor vessel. Hence, it can be classed as a pressure vessel type of reactor. In R5, both the pressure and temperature are comparatively low in the coolant channels as well as the reactor vessel. Another point worth noting is that the temperatures in the fast reactor are above the normal range of temperatures (i.e. about 430°C) for which pressure vessels are usually designed. Such designs must take into account high temperature creep effects (Ref.4).

A very careful and detailed stress analysis of the nuclear pressure vessels and components is essential since the safety of the nuclear power plant depends on their integrity. The design must also take into account the fact that inspection during service is often extremely difficult. The nuclear designer therefore relies heavily on Structural Mechanics for application of the most advanced stress analysis techniques to practical design problems.

2. PLACE OF STRESS ANALYSIS IN OVERALL DESIGN ACTIVITY

In order to appreciate the place of stress analysis in our overall design activity, it is worthwhile to review the procedure followed by us for the establishment of the final design. This is best illustrated by a block diagram which follows:



The first three blocks of the work above, starting with conceptual layout are evolved out of the designer's original ideas, his practical experience and selective use of published information about developments elsewhere in related fields. Proper decisions at a senior level at these stages are very important for avoiding large scale design changes or revisions at a later date. The plant operational and safety requirements are determined after detailed reviews by committees of engineers drawn from the related design, operation and research groups. ASME Pressure Vessel Codes (mainly Section III) are followed for the CANDU and R5 designs. For the fast reactor also, the design procedures would be based on the rules given by ASME Code case 1331-6 for nuclear vessels in high temperature service.

The design specification is an important document and is a requirement of the ASME Pressure Vessel Code Section III. This specification lists all important parameters such as pressure, temperature, transients, static and dynamic mechanical loads, etc. for the proposed design conditions and the various operating conditions defined by Section III. These are the normal conditions, the upset conditions (i.e. moderate deviations from normal), the emergency conditions (i.e serious deviations from normal, requiring shut-down and repairs but not associated with release of radioactivity) and the faulted conditions (serious deviations from normal, associated with release of radioactivity). The specification should also high-light design areas requiring special analysis and experimental work. Preparation of such a document is not possible without a detailed knowledge about the functioning of the plant, the seismic characteristics of the site and other safety requirements

The designer is in a position to proceed with detailed stress analysis of the components after the design specification is made. Eventhough design codes like ASME Section III give very comprehensive guidelines for proceeding with the work, there are often many areas where simple methods of analysis would be inadequate to generate the required detailed information regarding stresses, strains and deflections. The use of the computer and development of such numerical methods as finite element analysis have considerably widened the scope of application of analytical procedures. In spite of this, experimental techniques are necessary for supplimenting analytical work. There would also be cases where only experimental analysis is possible. Fortunately, facilities like the computer and experimental stress analysis laboratories are available to the designers in DAE.

3. MODES OF FAILURE

Before discussing the way in which the stresses must be calculated and classified for application of stress limits, let us consider the various possibilities of failure. The nuclear power plant components may fail in any of the following modes:

a) Excessive ductile deformation or rupture
b) Elastic instability (buckling)
c) Plastic instability (incremental collapse)
d) High strain low cycle fatigue
e) Brittle fracture
f) Failures due to creep such as rupture, deformation, creep fatigue and creep buckling.
g) Failures due to corrosion such as stress-corrosion, corrosion fatigue and corrosion wastage.

If we observe the stress limits set by the design codes, the vessel or the component can be considered safe against excessive distortion or rupture due to the above factors. Protection against brittle fracture and corrosion effects is however better ensured by proper selection of material, its notch toughness, inspection, chemical control of fluid, etc. rather than by the prescribed analytical procedures. In addition to the above, each design has its own specific limits against small deflections or cracks which may lead to undesirable leakage or some other unacceptable condition. For instance, in the CANDU design the diametral creep of the coolant tube must not be allowed to exceed 2% of the diameter since otherwise most of the coolant would by-pass the fuel leading to its inadequate cooling.

The change in mechanical properties of the material such as notch-toughness, ductility, creep, etc. due to nuclear radiation introduces an additional difficulty in nuclear design. For estimating the effect of this factor one must either rely on published data or carry out in-pile tests. Since more and more information is being generated with the passage of time regarding irradiation effects on material properties, decisions based on exigencies sometimes result in use of a material on the basis of information which is later on found to be inadequate. For the component called as the 'end-shield' in our first three CANDU reactors - RAPP Units 1 and 2 and MAPP Unit 1, we have used 3½% Nickel steel which, we now know, will be having a low notch-toughness at the operating temperature due to radiation embrittlement after some years of operation. The designer therefore, has to make a fracture-mechanics evaluation of this component. Similarly our knowledge about the irradiation creep of

Zirconium alloys used for the pressure tubes of the CANDU reactors has increased considerably since the time when design decisions for their use in reactor were taken. We have to therefore alter our original estimates about the deflections and diametral growths of pressure tubes in our operating reactors.

4. STRESS CATEGORIES AND STRESS LIMITS

The stress limits set by the ASME Code for nuclear pressure vessels are based on the maximum shear stress (TRESCA) failure theory. Hence, for determining the maximum shear-stress, the designer must calculate all the three principal stresses at any particular point. Considerable further detailing is also required since the stresses are divided in different categories and different stress limits are set for different categories. The basic stress categories are the primary (i.e. stress required to satisfy the equilibrium of external forces and moments), the secondary (self-limiting stress produced by structural restraints) and the peak (stress intensification due to local effects). A typical chart of stress categories and limits is shown in Fig.1. It should be noted that plastic and limit analyses are suggested as alternatives to elastic analysis for cases where the yield stress is exceeded. For design at high temperatures where creep becomes important, there are separate time-independent and time-dependent stress limits. Stress categories and limits for high temperature design are shown in Fig.2.

5. FIELDS OF RESEARCH AND SPECIAL ANALYSIS

Nuclear design has stimulated research in a wide variety of fields like fatigue, fracture mechanics, limit analysis, plasticity, elevated temperature effects, etc. Problems of structural dynamics such as flow-

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5. FIELDS OF RESEARCH AND SPECIAL ANALYSIS

Nuclear design has stimulated research in a wide variety of fields like fatigue, fracture mechanics, limit analysis, plasticity, elevated temperature effects, etc. Problems of structural dynamics such as flow-induced vibrations, seismic disturbances and missile effects are also of interest to us. The design criteria themselves (i.e. allowable stresses, safety factors, etc.) have been subjected to a detailed scrutiny. In addition to the general subjects described above, the designs of specific parts of the pressure vessel (such as heads, flanges, nozzles, shells, piping systems, etc.) are being studied extensively.

We will discuss here a few of the typical structural mechanics problems in our designs which we are trying to tackle with experimental and advanced analytical techniques.

6. <u>PERFORATED TUBE-SHEETS</u>

The perforated tube sheet is a pressure vessel part which finds frequent use in reactor vessels. Table 1 shows the dimensions and ligament efficiencies of some tube-sheets in our designs. The pitch of the perforations in all cases corresponds to the reactor lattice pitch which is triangular in case of the fast reactor and square in case of the CANDU and R5 designs.

Work on evolving design procedures for perforated tube-sheets had been going on even prior to the advent of nuclear pressure vessels for application to the heat-exchanger tube-sheets. Since the publication of a well-known paper on this topic by K.A. Gardner in ASME Journal of Applied Mechanics, 1948, a number of workers have tried to refine these procedures and to extend their applications. The procedures make use of the concept of equivalent solid plate with modified Young's modulus and Poisson's ratio. Physically [illegible] stresses in the ligaments are [illegible] nesses and different types of loading such as in-plane, shear, bending, etc. For a square pitch, different elastic constants are assumed in the pitch and diagonal directions to account for its orthotropic behaviour.

The equivalent solid plate theory is however not found to be very satisfactory for plates with square pitch perforation pattern and low ligament efficiencies (Ref.5). It cannot also give detailed information on the stress and deflection pattern which may be necessary for a fracture mechanics study or for full consideration of coolant channel alignments. We have sponsored a project with the structural engineering department of I.I.T., Bombay for the application of a rigorous finite element analysis to the calandria-end shield tube-sheets of CANDU design. The work may be extended to other tube sheets also, at a later data. Some experimental work (especially for R5 designs) is also being contemplated by the experimental stress analysis group of Reactor Engineering Division, BARC.

7. PERFORATIONS IN SHELL

Another problem encountered by us in the design of the reactor vessel of modified CANDU design pertains to perforation in the shell to which pipe nozzles are welded. A view of the shell and the perforations is shown in Fig.3. As many of the perforations are for the reactor control tubes which must be vertical, they are non-radial and are quite closely spaced with respect to each other. The shell is however subjected to low pressure and temperature and the loads and moments on the nozzles as shown in Fig.3 are not also very high.

Published information on the nozzle-shell junction stress analysis [illegible] applies to radial nozzles. The [illegible] method recommended by

ASME mainly for single ~~single~~ isolated openings is for pressure loading only and is applicable when the ratio $\frac{\text{shell diameter}}{\text{shell thickness}}$ does not exceed 100. This ratio is about 200 in our case. The Bijlard-Wichman graphs (Ref.6) take into account the loads and moments on the nozzles and give peak stresses at some discrete points around the nozzle-shell junction. They however do not give stresses in the shell away from the nozzle. This is imporant in our case since the stresses in the shell due to adjacent perforations may have to be superposed on each other because of the close spacing.

For analysing these stresses we propose to carry out an experimental stress analysis on a calandria model. A numerical technique based on finite element theory is also under active consideration as a long term solution for getting more detailed information.

8. END-SHIELD SUPPORT DIAPHRAGM

Fig.4 shows the support diaphragm for the end-shields of the modified 200 MWe CANDU design. The diaphragms are grouted in the reactor vault wall at the outer end and are welded to the end-shields at the inner diameter. They have to therefore accommodate the relative radial thermal expansion between the (stainless steel) end-shields and the concrete wall as well as the relative transverse thermal expansion between the concrete side wall of the vault and 306 thin walled Zircaloy-2 calandria tubes which are roll-expanded in the calandria side tube-sheets of the opposite end-shields. The major mechanical load on the diaphragms is due to an earthquake when the mass of the end-shields and calandria may experience transverse acceleration of as much as 0.8g.

A method for designing the diaphragms for static loads has been developed using the finite difference technique (Ref.7). The diaphragms must be designed against elastic and plastic instability and gross deformation, especially during an earthquake. An elasto-plastic analysis of this component, supplimented by experimental work is under consideration.

9. FLOW INDUCED VIBRATIONS

The effect of flow induced vibrations has to be carefully investigated to guard against (a) excessive oscillations which may result in power oscillations (in fast reactors), fretting, fatigue, wear, broken specimen holders, loosening of bolts, etc. (b) collapse of plate type elements. Components subjected to axial flow will be exposed to flow noise consisting of far field and near field components. Components facing cross flow have to be checked against resonance failure due to vortex shedding. Theoretical analysis of flow induced vibration is yet to reach a stage where it can be utilized by the designer with confidence. Experiments have been planned to check the effectiveness of our designs against flow induced vibrations. Part of the work is being done at I.I.T., Kanpur (for R5) and I.I.Sc., Bangalore (for CANDU).

In R5, the guide tube will be subjected to cross flow of the moderator during dumping of moderator. Fuel rods and guide tube are subjected to axial flow. Similarly the calandria tubes and the control rod tubes in CANDU calandria would be subjected to cross-flow of the moderator.

10. COOLANT TUBE CREEP

The Zircaloy-2 and Zirconium-niobium alloy pressure tubes in CANDU reactor exhibit significant creep at the design temperature of 300°C mainly because of the effect of nuclear irradiation. A lot of information has been generated abroad (mainly in U.S.A. and Canada) regarding the effect of irradiation on the creep of Zirconium alloys. An interesting problem from the point of view of structural mechanics is the bending of the coolant tube due to a combination of creep and elasticity. In Ref.8, the following equation for tube-curvature is recommended:

$$K\ (x,t) = \frac{M\ (x,t)}{EI} + \int_0^t \frac{C\ (x,t)\ M(x,t)}{I}\ \delta t$$

Where K = Curvature at any cross section x of the tube at time t

M = Bending moment at the cross-section

E = Modulus of elasticity

I = Second moment of the cross-sectional area

C = A parameter representing creep behaviour of the material.

Solution of the problem for our specific cases has been worked out with the aid of CDC 3600 computer at TIFR.

11. STRESS ANALYSIS OF END-FITTINGS AND ROLLED JOINTS

The CANDU end-fitting which is an extension of the pressure tube through the end-shield is shown in Fig.5. The Zirconium alloy pressure tube is roll-expanded in the end-fitting at its reactor end. 13% chromium martensitic heat-treated stainless steel (AISI 403) is chosen as the end-fitting material since it gives satisfactory combination of hardness and coefficient of thermal expansion for a good rolled joint with Zirconium alloy pressure tube. The material experiences embrittlement and a rise in the nil-ductility transition temperature during service due to severe fast neutron irradiation near its reactor end. In order to evaluate the functional integrity of the component we must analyse it from fracture mechanics considerations. This, in turn, requires a detailed stress-analysis as a pre-requisite.

The component is axi-symmetric with some degree of non-symmetry at the side nozzle. Modern numerical methods can be employed to obtain a detailed mapping of stresses due to temperature and applied loads. However, the stresses induced during the tube-rolling operation and the subsequent changes in these stresses due to relaxation, thermal cycling and pressure cycling are not easily amenable to analytical techniques.

Rolled joints are used at quite a few other places in CANDU and R5 reactors. Typical rolled joint designs in CANDU are also shown in Fig.5. In R5 reactor the number of different types of rolled joints is quite large. This is because, in addition to the joints occurring in the main reactor system, such joints have to be provided for the different experimental facilities of various sizes also. The operating conditions for these joints being comparatively mild, leak tightness is their only basic requirements. In spite of this, a large amount of time and effort is needed for the development of these joints, some of which present special problems because of their unconventional size and other limitations.

For ensuring the integrity of such joints in service we depend, to a large extent, on past experience and on some prototype laboratory tests such as pull out, thermal cycling, etc. Some attempts have also been made for theoretically evaluating the effect of various joint parameters. There is however, a strongly-felt need for the development of reliable experimental and analytical techniques for the evaluation of stresses and strains in the rolled joints.

12. <u>SPECIAL PROBLEMS IN FAST REACTOR</u>

There are quite a few problems of interest to structural mechanics in the design of the 500 MWe PFBR. These will be discussed in detail in a separate paper in this session.

13. <u>CONCLUSION</u>

In this review it has been possible only to briefly introduce some of the problems faced by the nuclear designer as a result of his own requirement of detailed stress analysis of nuclear components. By presenting them at this Seminar we expect to arouse a lively interest for their solution amongst the Indian engineering community in academic institutions and industry.

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TABLE 1

Dimensions of Perforated Plates of interest to DAE

(All dimensions are in mm)

Sl. No.	Description	Thickness H	Hole Radius R	Pitch-distance P	Plate Diameter D	Total No. of holes	Ligament efficiency $\frac{P-2R}{P}$	Remarks
1.	CANDU End-shield calandria tube-sheet	51/26	60.37	228.6 square	4928	306	0.472	The tube-sheet thickness is 51 mm around the holes and 26 mm in the remaining portion.
2.	CANDU End-shield fueling machine tube sheet	51	97	228.6 square	4928	306	0.1514	
3.	CANDU End-shield baffle plate	38	87	228.6 square	4928	306	0.2388	
4.	R5 Calandria tube-sheet (bottom plenum)	50	50	180 square	2876	146	0.444	Both bottom and top plenum are composite tube-sheets made of two tube-sheets, each being 50 mm thick.
5.	R5 Calandria tube-sheet (top plenum)	50	64	180 square	3188	146	0.2889	
6.	R5 End-shield tube sheet A (top)	75	73	180 square	3130	146	0.189	End-shield tube-sheet A & B share loads with a distance of 1625 mm between them.
7.	R5 End-shield tube-sheet B (bottom)	40	69	180 square	3130	146	0.233	Tube-sheet B is made of two tube-sheets which share loads between them and each is having a thickness of 40 mm.
8.	FBTR Grid plate (small support plate)	131	17 for 121 holes and 10 for 90 holes	50.8 triangular	-	211	0.331 for 121 holes and 0.606 for 90 holes	Plate is hexagonal, distance across flats being 762 mm.
9.	FBTR Grid plate (Large support plate)	90	10	50.8	1620	451	0.606	Plate has an hexagonal opening in the centre with distance across flats equal to 766 mm.

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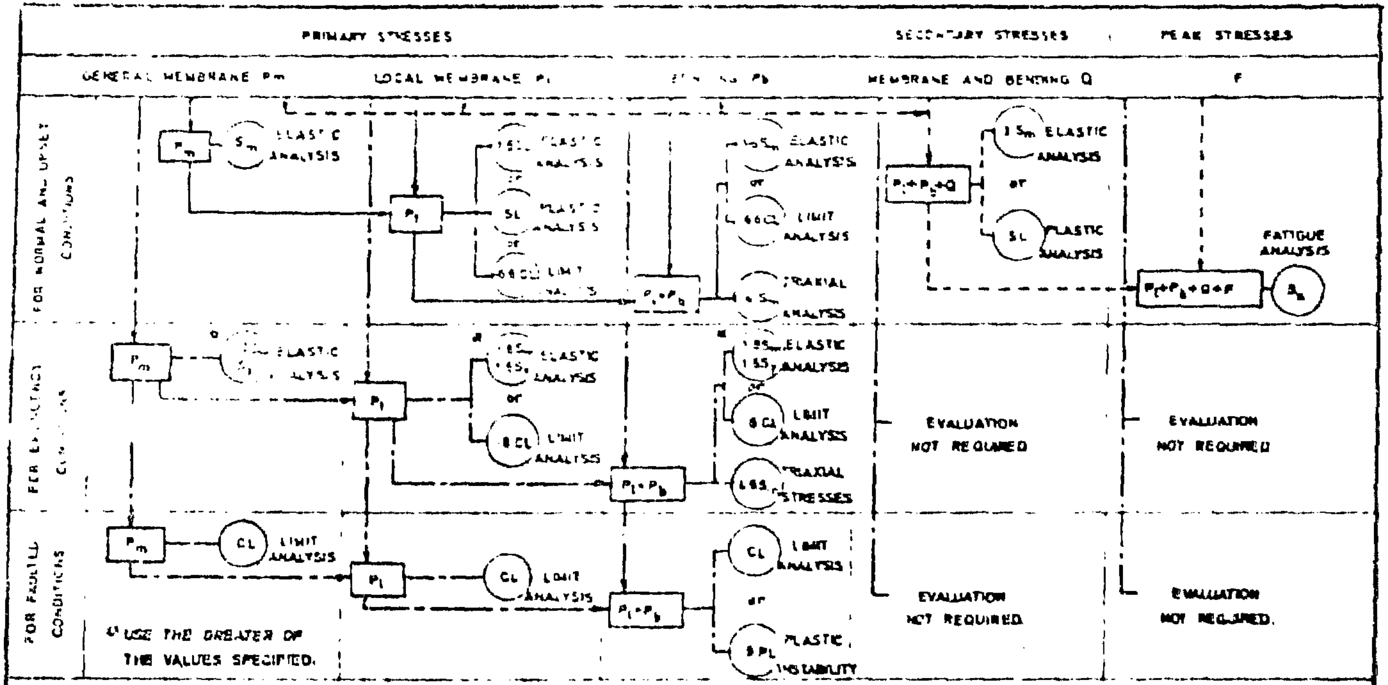


USE DESIGN LOADS SELECTED FOR NORMAL AND UPSET CONDITIONS.

USE OPERATING LOADS SPECIFIED FOR NORMAL AND UPSET CONDITIONS

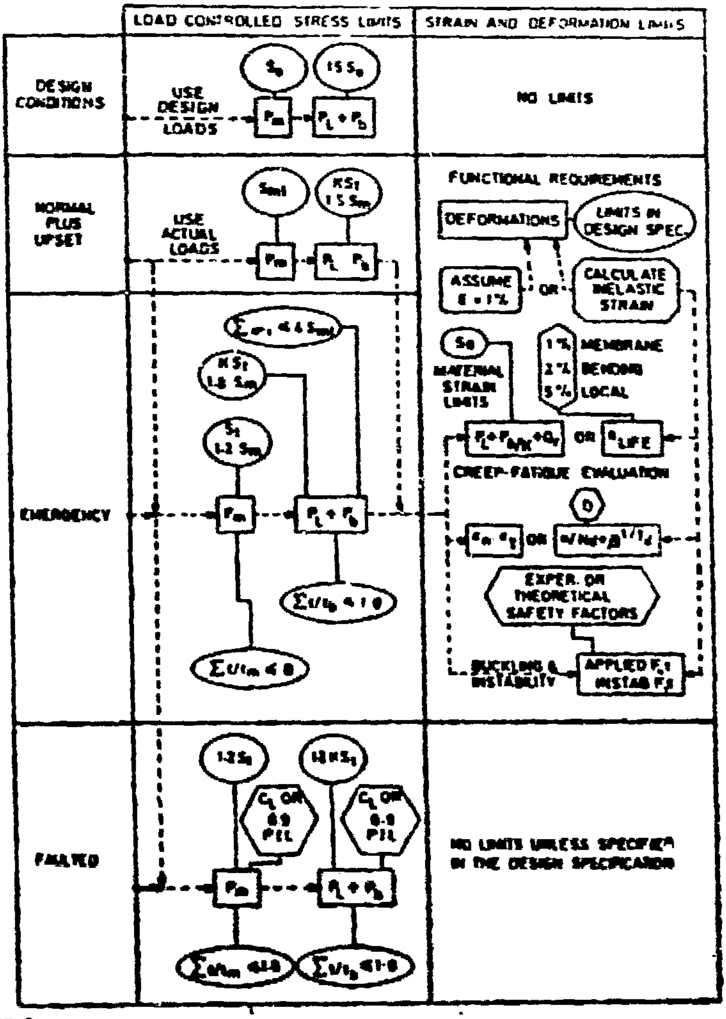
USE OPERATING LOADS AND ACTUAL MECHANICAL LOADS FOR EMERGENCY OR FAULTED CONDITIONS

SL — SHAKEDOWN LOAD. CL — COLLAPSE LOAD. S_m — DESIGN STRESS INTENSITY

PL — PLASTIC INSTABILITY LOAD. TRIAXIAL STRESSES — σ_1 σ_2 σ_3 S_y — YIELD STRENGTH.

S_a — ALLOWABLE AMPLITUDE OF AN ALTERNATING STRESS COMPONENT

STRESS CATEGORIES AND LIMITS OF STRESS INTENSITY FOR OPERATING CONDITION-CATEGORIES Fig.1.



LEGEND

CONTROLLED QUANTITY FOR ELASTIC ANALYSIS — CONTROLLED QUANTITY FOR INELASTIC ANALYSIS — COMPUTED QUANTITY

S_o - PRIMARY GENERAL MEMBRANE STRESS INTENSITY   D - TOTAL DAMAGE FACTOR.

t - TIME DURATION OF THE LOAD CONDITION, k.   n - NUMBER OF APPLIED CYCLES OF LOADING CONDITION. β - CREEP DAMAGE MULTIPLICATION FACTOR.   T - TEMPERATURE DURING THE ENTIRE DESIGN LIFE OF THE COMPONENT. T_d - ALLOWALE TIME AT A GIVEN STRESS INTENSITY FROM LOAD, k ϵ_t - PEAK STRAIN RANGE OF THE CYCLE.

ϵ_n - TOTAL NOMINAL STRAIN. S_t - TEMPERATURE & TIME DEPENDENT STRESS VALUE

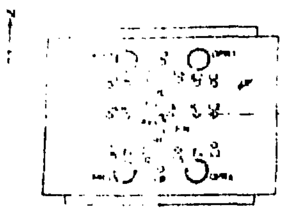
D - LIFE FRACTION USAGE FACTOR   - - NUMBER OF ALLOWABLE CYCLES OF LOADING CONDITION, j. S_q - THE LESSER OF THE AVERAGE OF THE S_y VALUE FOR HIGHEST & LOWEST VALUE OF THE METAL TEMPERATURE DURING THE CYCLE.

LOAD k - LOAD CAUSING CREEP   LOAD j - LOAD CAUSING FATIGUE.

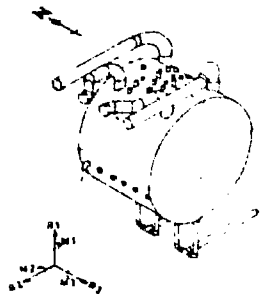
HOPPER DIAGRAM FOR ELEVATED TEMPERATURE ANALYSIS

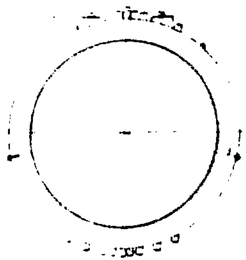
Fig. 2.

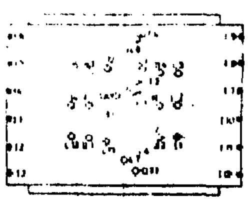
DIAGRAM FROM REF 4
PAGE 674



VIEW FROM TOP







VIEW FROM BOTTOM

NOTES -

CO-ORDINATE AXES R1 R2 & R3 ARE SUCH THAT
R1 IS ALONG THE AXIS OF EACH NOZZLE POINTING OUTWARDS
R2 POINTING TOWARDS WEST
R3 IS ALONG THE AXIS OF CALANDRIA POINTING TOWARDS SOUTH

FORCE POSITIVE IF IT IS IN THE DIRECTION OF THE RELATED POSITIVE AXIS

MOMENT POSITIVE IF IT IS IN THE COUNTER CLOCK-WISE DIRECTION FACING THE RELATED POSITIVE AXIS

PIPE REACTIONS FOR FULL POWER OPERATION ARE GIVEN IN TABLE I

LEGEND -

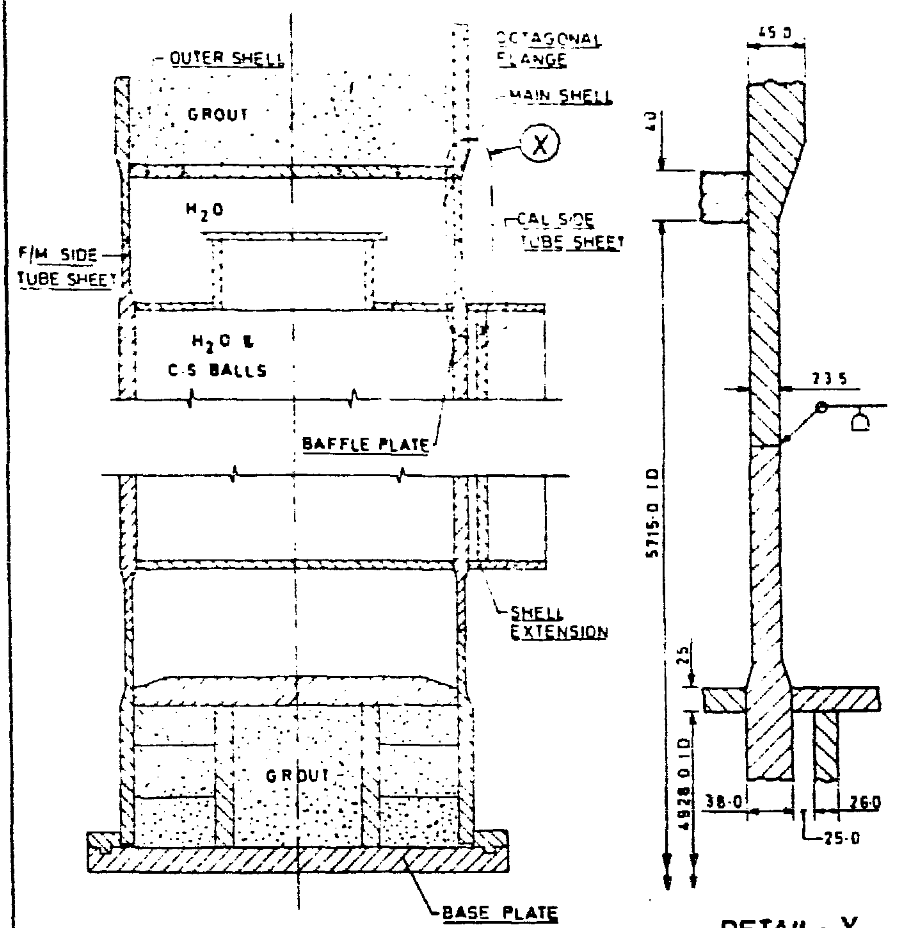
- B — BOOSTER ROD NOZZLE
- L — LIQUID SHUT OFF ROD NOZZLE
- OT — MODERATOR OUTLET NOZZLE
- R — REGULATING FINE ROD NOZZLE
- SH — COARSE SHIM ROD NOZZLE
- FN — FLUX MONITOR NOZZLE
- OF — OVER FLOW NOZZLE
- OPR — OVER PRESSURE RELIEF NOZZLE
- I — MODERATOR INLET NOZZLE

TABLE-I

CONNECTED PIPE		R1 kg	R2 kg	R3 kg	M1 kg cm	M2 kg cm	M3 kg cm
PRESSURE RELIEF	1	-483	-4425	513	-1.578 x10^8	-0.8538 x10^8	-3.768 x10^5
	2	-818	-4158	-1108	-0.7022 x10^8	0.6048 x10^8	-2.428 x10^5
	3	-818	-4158	+1108	-0.7022 x10^8	-0.6048 x10^8	2.428 x10^5
	4	-483	-4425	-513	-1.578 x10^8	-0.8538 x10^8	3.768 x10^5
MODERATOR INLET	1	143	-30	38	178	2187	581
	2&3	LESS THAN VALUES AT 1					
	4	17	-26	-30	584	1818	1107
	5&6	LESS THAN VALUES AT 4.					
	7	158	-87	-28	1786	-2858	-338
	8&9	LESS THAN VALUES AT 7					
	10	14	-65	28	828	2056	-4508
	11&12	LESS THAN VALUES AT 10					
MODERATOR OUTLET	1	184	304	-1	-2082	-181	+52808
	2	223	366	+28	-2197	-4085	+58308
	3	184	304	+2	-2082	-181	-52408
	4	223	330	-28	-2197	-4085	-58390
CONTROL ROD PENETRATION AT TOP	BOOSTER 1,2,3&4	504	—	—	—	—	—
	FINE REGULATING 1&2	1004	—	—	—	—	—
	COARSE REGULATING 1&2	400	—	—	—	—	—
	LIQUID SHUT OFF 1 TO 16	400	—	—	—	—	—
CONTROL ROD PENETRATION AT BOTTOM	BOOSTER 1	137	38	18	-858	18703	-17080
	BOOSTER 2	137	0.0	0.0	0.0	56	[illegible]
	BOOSTER 3	137	38	-18	-858	18700	27880
	BOOSTER 4	137	0	0	0	54	17550
	FINE REGULATING 1&2	FINE & COARSE REGULATING RODS ARE FREE TO EXPAND AT BOTTOM. HENCE CAUSE NEGLIGIBLE REACTIONS.					
	LIQUID SHUT OFF 1,2,3,13,14 & 15.	104	-2500	-1	-768	-288	134480
	LIQUID SHUT OFF 7&8	47	4701	-1	-578	0	-1338
	LIQUID SHUT OFF ROD 6	6	-7	+2	15	-140	-814
	LIQUID SHUT OFF LINES 9&10	41	3014	0	-115	36	-107480
	LIQUID SHUT OFF LINES 4,5,11&12	47	3012	+2	-481	136	-112598
MODERATOR OVERFLOW LINE		-23	213	5	517	877	-10578

MODIFIED 200 MW(e) CANDU REACTOR

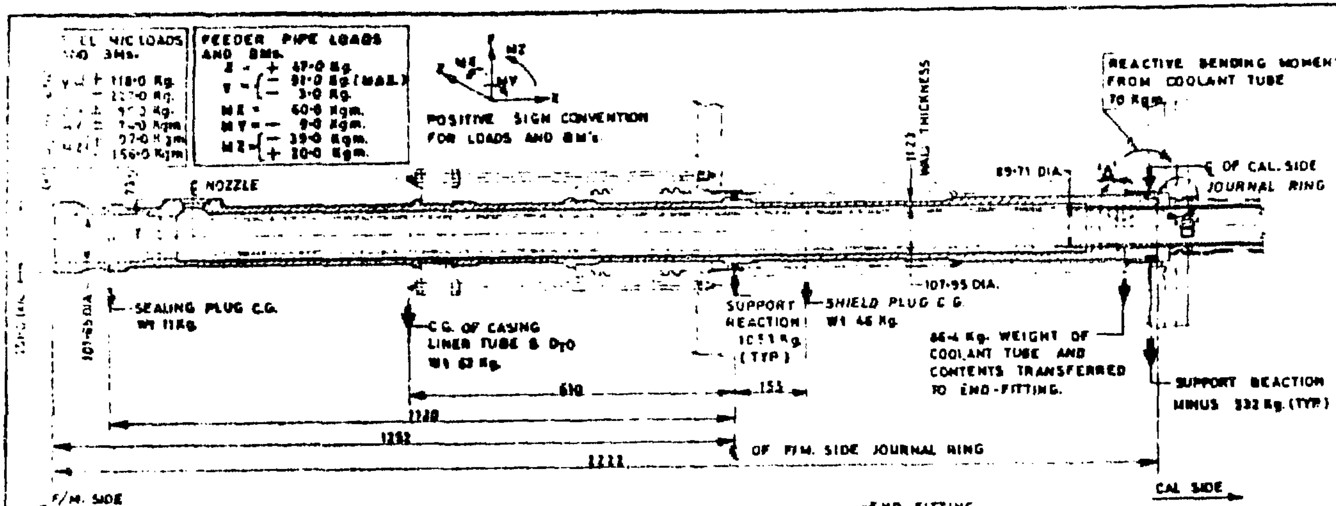
FIG.3. REACTIONS ON CALANDRIA SHELL FROM CONNECTED PIPING

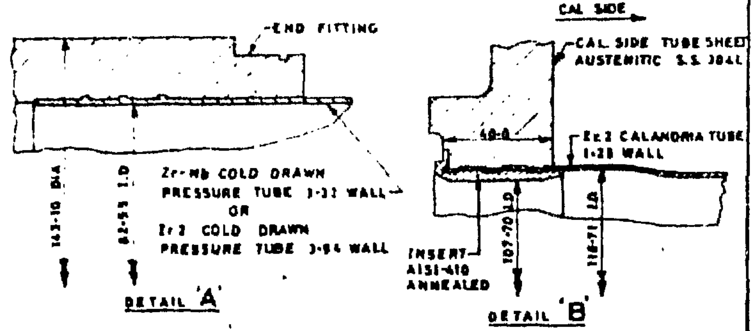


MODIFIED 200 MW(e) CANDU REACTOR END SHIELD SUPPORT DIAPHRAGM.

DETAIL - X FOR SUPPORT DIAPHRAGM

Fig. 4





OPERATING CONDITIONS	LOCATION	MAX TEMP °C	MAX PRESS. Kg/cm²
HYDROSTATIC PRESSURE TEST	—	40·0	196·90
NORMAL OPERATION	INLET	260·0	105·90
NORMAL OPERATION	OUTLET	299·0	96·90
HOT SHUT DOWN	INLET	283·0	104·40

MATERIAL – AISI TYPE 403 HEAT TREATED.

DESIGN TEMPERATURE 300°C.

DESIGN PRESSURE 112·50 Kg/Sq.cm.

DIMENSIONS ARE IN mm.

MODIFIED 200 MW(e) CANDU REACTOR

END FITTING AND ROLLED JOINTS

FIG. 5