Study on Prestressed Concrete Reactor Vessel (PCRV) Structures
—II-5 Crack Analysis of 1/20 Multicavity Type PCRV Subjected to Internal Pressure
by Three-dimensional Finite Element Method—

PCRV Research Group

Abstract

This report describes the propriety and the practicability of three-dimensional nonlinear finite element analysis for PCRV structures subjected to internal pressure by comparing calculated results with test results. Because a PCRV is composed of a number of structural materials and its configuration is very complex, it is most difficult to precisely analyze the inelastic behavior up to the ultimate load capacity. At first, therefore, a nonlinear analysis considering concrete cracking only was attempted. As a result, it was found possible to predict the behavior of a PCRV up to three times the design pressure, and calculated results agreed well with test results.
あるとした。本解析に用いた有限要素を図1に示す。コンクリートを20節点6面型要素、ライナー耐圧板などの鋼板を8節点線型要素、高張力鋼筋を3節点線型要素に置換した。報告（Ⅱ-2）の1/20 PCRV 模型を、その対称性を考慮して、図2のようにモデル化した。この解析モデルは、6面型要素数139, 膜要素数79, 線要素数430より構成され、総節点数は990である。各要素の剛性および要素の重ね合わせて求め、ガウスの積分点における応力度を用いてひずみおよび破壊の判定を行っている。ガウスの積分点数は6面型要素を302*2, 膜要素を3*3, 線要素を2とした。線返し計算の収束判定数は、ここでは計算時間の短さを避けるため、全体の不連合節点力の大きさによって判定している。線形方程式の解法にはIronのFrontal Solution Programを用いた。

3. 解析結果

材料の弾性定数とコンクリートの引張強度は報告（Ⅱ-3）の値を用いた。計算は前線形数法によって進み、節点増分の大きさを内圧100 kg/cm²にした。節点増分回数15のとき同実験でライナー鋼板要素が弾性限を越えたため計算を打ち切った。実計算時間は、IBM370/150を使って、CPU約12時間であった。

3.1. 弾性挙動

図5内圧50 kg/cm²以下の試験体の4箇所における荷重-変位関係を図3に示す。図中の2-Dは文献2)

図4 内圧50 kg/cm²時における側面制断面内のA-A断面とB-B断面上のひずみ（ε₁, ε₈）分布。
3.2 非弾性挙動

トップスラブ中央および側壁中腹キャビティ外側の荷重一変位関係を図-5に示す。3-Dの値は内圧90 kg/cm²からひびわれ破壊による変位が観測された。しかし実験値より変位が若干大きくなっている。側壁中腹側コンクリートの変位と円周方向のひずみ関係を図-6に示す。内圧90 kg/cm²時に、ひびわれ破壊によってひずみが発生している。同じ傾向が実験値にもみられる。側壁外側筋材の変位一ひずみ関係を図-7に示す。

特定荷重時の円周断面上の変形状態を実験値と比較して図-8に示す。図はキャビティを通過するA-A断面およびサポート部の中心を通るB-B断面について、それぞれ、プレストレス、内圧50、120 kg/cm²のときの変形状態を示す。図より、変位時の変形状態がよくわかる。内圧120 kg/cm²のとき、トップスラブの変形が実験値に比べてかなり大きくなっている。

図-9はコンクリートのひびわれ破壊領域の進行状態を半径断面について図示したものである。内圧100 kg/cm²のとき、A-A断面以上の変位は主に全幅の変位が観察されて、ひびわれ破壊を生じたが、B-B断面では側部だけであった。また各レベルにおける水平断面では、キャビティ周辺にひびわれがかなり生じていることがわかる。変-1に各断面のひびわれ発生荷重をそれぞれ解析値と実験値3-D,2-Dを比較して示す。なお、実験値は試験したひずみを各断面位置ごとに平均化して求めた値である。実験値は全体的にひびわれ発生荷重が大きい。内部ひびわれ状態を調べるために、実験後、試験体を写真-1に示したようにカットした。試験体のひびわれ状況は図-9とほぼ同じであった。

4. まとめ

3次元有限要素法によるPCRVのひびわれ解析を行った結果、下記の事項がまとめとしている。
1) 本解析モデルは、用いた要素の種類、要素分割数、積分点数と等に関し、PCRVの全体的な非線形挙動をのう十分なものであった。

2) ひびわれ破壊状態は実験結果とよく一致しており、3次元的な内部ひびわれの発生状況がかなり解明できた。

3) 本解析によれば、設計内圧の約3倍ぐらいまでの解析可能であり、またそれほど経済的ではない。さらに、全体的な挙動ののみを求めるのであれば、もっと粗く要素分割されたモデルの解析で十分と思われる。

4) 報告（I-3）の2-D結果は、特定位置における挙動が実験値に一致しており、設計コストにおける経済性が良い。それゆえ、本手法はPCRVを略算設計する場合に適用している。

5) PCRVの内圧による非線形挙動はほとんどコンクリートのひびわれ破壊と鋼材の塑性化に依るものと考えられる。それゆえ、予想耐力を至るまでの解析を行なうにあたり、本解析コードに鋼材の塑性曲線を考慮すれば可能であろう。この問題については、現在検討中である。

参考文献
1) “プレストレスコンクリート原子炉圧力容器（PCRV）構造物に関する研究（II-2 1/20マルチキャビティ型PCRVモデル内圧実験）”大林技研技書 No.9 (1974)
2) "問題 (II-3マルチキャビティPCRVの内圧による破壊特性) "大林技研技書 No.10 (1975)
3) "井元，武田 "3次元弾性体の非線形有限要素解析（第2篇） "林技研技書 No.11 (1975)"
Present status of the production and development of the valves for nuclear use in Japan is reviewed. As introduction, the second part, some examples of the valves for nuclear use are shown. Their present situation and recent trends of design, structure, function, materials, and manufacturing are explained in some detail. The significance of quality assurance is also stressed with reference to the ISO 9000 system of ASME SECTION III. Needs for valve testing facilities and consideration of valve maintenance are also emphasized. Some tools invented for the valve maintenance within a reactor containment vessel are explained. Finally, the importance of promoting standardization and smooth procurement of materials is explained.


Available from Institut National de la Propriete Industrielle, Paris (France); priority claim: 3 Jun 1974, UK.

The invention relates to fastening devices restricting load. It deals with a device comprising a bolt threaded in a nut, said device being provided with wires clamped between cross-pieces, said wires being applied between the bolt and nut, therefore between two plates, being restricted by the friction forces between said wires and cross-pieces. This can be applied to the mounting of burst-panels on the pressure vessels used in nuclear industry.

22417 Gritting appliance for oblong bodies which are inserted into and removed from holding fixtures, especially for fuel elements and control rods in a nuclear reactor. Hoffmeister, B.; Krupp (Friedr.) G.m.b.H. (German) Patent 2,359,163/A/. 12 Jun 1975. 17p. (In German).

Fuel elements and control rods can be handled with a common gritting appliance. It is mounted on a travelling crab and has a gripping device with two pairs of grab jacks forming double tongs. One pair of tongs grips either the fuel elements or the control rods. The gritting appliance itself can be lifted and lowered, and a common operating piston within a cylinder is provided for the two pairs of tongs, on which a bolt spring is continuously exereting a force in order to move the grab bodies or hold them in clutching position. For unlocking this position, e.g., compressed air is admitted to the piston. All of these operational elements are installed on or within the gripping device. This one is supported swingably about a horizontal axis by means of hinge pins between two side walls projecting downward. The swinging movement is caused by a gear wheel and a rack, the rack being mounted on one of the side walls, the gear wheel on a hinge pin. A locking mechanism holds the gripping device in the operational positions of the grasbs.


The safety element is not overdimensioned at pressures between 2 and 150 atmospheric excess pressure. Therefore the flat bursting disc is mounted within a supporting and stopping holding and the rated breaking point is covered by a supporting body. Its outer diameter sufficiently overlaps the recesses on both sides of the rated breaking point. It absorbs the total load given by the operating pressure. Only a release mechanism with wedge, eccentric disc, magnet, and rocker arm releases the supporting body, e.g., if the breaking point is exceeded, so that the operating pressure may work on the bursting disc. An insulated copper wire is laid in the breaking region within the bursting disc in case of shearing off the supporting body. Effective elastic constants are presented for perforated concrete plates with triangular penetration patterns such as top head slabs of prestressed concrete reactor vessels. A formulation of the relationship between the effective elastic constants and the elastic constants of the matrix is derived from the two-dimensional elastic theory, and extensive numerical results covering a wide range of ligament efficiencies in the perforated plates with reinforcement are given for the plates. Finally, the applicabilities of these values to the plates in bending are examined in a number of examples.


This report describes the propriety and the practicability of three-dimensional nonlinear finite element analysis for PCRV structures subjected to internal pressure by comparing calculated results with test results. Because a PCRV is composed of a number of structural materials and its configuration is very complex, it is very difficult to precisely analyze the inelastic behavior up to the ultimate load capacity. At first, therefore, a nonlinear analysis considering concrete cracking only was attempted. As a result, it was found possible to predict the behavior of a PCRV up to three times the design pressure, and calculated results agreed well with test results.


The results of a creep test of concrete conducted as a first step to clarify the long-term deformation behavior of a PCRV. The tests were performed under the conditions of (1) uniaxial and triaxial stresses, (2) two different loading ages (30 days and 90 days), (3) scaled curing, and (4) normal temperature (20°C). Although there were some points in the testing procedures which could be improved, the basic behaviors under stress, which are, for instance, unit creep curve, creep recovery and creep Poisson's ratio, were obtained well on the whole. Further, the possibility of predicting creep under multiaxial stresses from uniaxial stress results was indicated.


From Seminar on NDT on boiler and pressure vessels; Calcutta, India (22 Mar 1975). 11 figures.

Typical problems encountered in employing ultrasonic testing techniques in the nuclear field and their solutions are illustrated by describing ultrasonic testing of the integrity of fabricated reactor components such as pipe flanges, forgings and tank weldments and ultrasonic checking of the movement of silt deposits in sea-water inlet pipe-lines of reactor.


Tension values which deviate from the specified value by a maximum of only plus or minus 2.5% are attained by hydraulic bolt tightening instruments. This high tightness precision is of considerable importance for compliance with the required safety factor of bolted joints between the dome and the pressure vessel of a reactor. A number of bolt tightening devices involving bolt transport, nut transport, and cover transport are described, some of which lend themselves to complete mechanization and to remote control and monitoring.


From Lecture meeting 1976 'non-destructive material testing'; Labnietz, Germany, F.R. (24 May 1976). 10 figs.

Variables often used for the identification of ultrasonic transducers, their influence on measuring results, and the requirements made on the sound field during immersion testing of thin-walled tubing are presented.


6 tabs.

The present standard deals with materials, design, construction, and testing of piping and parts of pipes.