العنوان:	Stress Analysis of Piping Systems in Nuclear Power Plants
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التاريخ الميلادي:	1990
موقع:	القاهرة
الصفحات:	1 - 251
رقم MD:	565698
نوع المحتوى:	رسائل جامعية
اللغة:	English
الدرجة العلمية:	رسالة دكتوراه
الجامعة:	جامعة عين شمس
الكلية:	كلية الهندسة
الدولة:	مصر
قواعد المعلومات:	Dissertations
مواضيع:	الهندسة الميكانيكية، المفاعلات النووية
رابط:	https://search.mandumah.com/Record/565698

ABSTRACT

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The effect of boundary conditions on the reliability of piping was studied. For instance, the presence of significant deviating constant hanger load ratings which are frequently used for supporting the piping showed unexpected additional stresses on the piping .Such deviation was not taken into consideration during the design phase.

Several examples of piping models were studied to illustrate such deviating constant hanger load rating cases. According to these studies, it is recommended to review the state of stress under the actual load rating of the used constant hangers. For determining such actual load rating some suggested iterative methods were developed.

Experimental tests using an actual feed water piping system of decommissioned reactor (Heissdampfreaktor , HDR) in Germany were performed.

Parallel to this experimental work, theoretical computations of the stresses were achieved by using advanced finite element codes such as ASKA and ABAQUS codes. For validation purposes a comparison of experimental and theoretical results were performed. Member programme caculations provided a conservative estimation for piping behaviour. However, more advanced 3 dimensional analysis programmes gave more realistic evaluation of highly stressed components than using simplified analysis as cited in the American Society of Mechanical Engineering ASME Code. A coupled experimental theoretical method was developed to provide more accurate analysis of highly stressed components in case of unknown boundary conditions.

Elastic-Plastic behaviour of highly stressed components could lead to plastic deformation under faulted conditions.

Thus, experimental investigation for elastic —plastic behaviour of the simulated elbows of HDR feed water piping under predominant in-plane bending moment in opening mode was undertaken. The results showed that the bend zones were subjected to cross-sectional deformation (ovalization) as a result of elbow geometry. These results were also proved to be extended to the adjacent straight pipes. This deformation led to high stresses particularly in the inner surfaces. Also, plastic deformation of such elbows was initiated locally at elbow flank in the inner surface and then in the outer surface . It has been shown that the elbows under in-plane bending moment in the opening mode are not amenable to collapse under practical service conditions. This complies with the basic safety approach in which a catastrophic failure is excluded.

بسم الله الرحمن الرحيم

خلاصة

تهدف هذه الرسالة الى تحقيق أهم مبادئ الأمان التى يجب تطبيقها على المنشآت النووية ومنها محطات القوى النوويه • وتعتبر أنظمة الأنابيب المتعلقة بالأمان واحدة من النظم الهامه فى المفاعلات النووية • وتتناول هذه الرسالة دراسة نظم الانابيب تحت ظروف التشغيل والتحميل المختلفة فى المفاعلات النووية بهدف تحقيق مبادئ الأمان وكذلك للتأكد من استبعاد حدوث كارثة الانهيار المفاجئ فى الأنابيب •

وتتكون الرسالة من ستة أبواب • يتناول الباب الأول مقدمة الرسالة والباب الثانى يتناول أهم المبادئ الفنية للأمان المطبقة فى المفاعلات النووية • ويتعرض الباب الثالث الى حدود الأمان للنظم المختلفة فى المفاعلات أما الباب الرابع فهو يتعرض الى تحليل الاجهادات للأنابيب المتعلقة بأمان المفاعلات والباب الخامس يتناول دراسة تجريبية للمرونة واللدونة لأنواع الأنابيب المستخدمة فى المفاعلات • ويلخص الباب السادس أهم النتائج العامه المستقاه من الدراسات القائمة فى الرسالة •

ان دراسة ظروف حدود الأمان وقدرة تحمل الأنابيب قد تم تناولها فى أعمال سابقة فمثلا وجود تفاوت فى معدل التحميل للمعلقات ذات معدل التحميل الثابت والتى يتواتر استخدامها فى تعليق وتثبيت الأنابيب قد يسبب اجهادات اضافية غير محسوبة عند تصميم الأنابيب • ولقد تم تحليل عديد من الأمثلة لنماذج الأنابيب المستخدمه فى محطات القوى وباستخدام طريقة العنصر المحدود Finite Element بالشفره المسماه SRPIPE لايضاح حالات التفاوت فى معدل التحميل للمعلقات ذات معدل التحميل الثابت . وطبقا لهذه الدراسة أمكن التوصية باعادة النظر لحالة توزيع الاجهادات تحت ظروف معدل التحميل الواقعى لهذه المعلقات ومن ثم اقتراح طريقة للحساب التقديرى لمعدلات التحميل الحقيقى أثناء استخدامها فى تحميل الأنابيب • 408086

وفى سبيل تحليل الاجهادات فى الأنابيب المتعلقة بالأمان فقد تم عمل اختبارات تحقيقيه باستخدام أحد الأنظمة الفعلية وهو نظام أنابيب التغذية بالمياه المزال تلوثها الاشعاعى لأحد المفاعلات المسمى بـ HDR مفاعلات البخار المحمص والمخصصة للتجارب فى المانيا الاتحادية بجانب هذا تم عمل حسابات نظرية لتلك الأنابيب باستخدام أحدث تقنية بالشفرات الحسابية بطريقة العنصر المحدود المستخدمه فى البرنامج ABAQUS و ASKA • وقد تم كذلك مقارنة النتائج التجريبية والنظرية بهدف التحقيق والتثبت • ولقد أبرزت المقارنة أن البرامج المستخدمة لنظرية العنصر المحدود وذات عناصر من الأذرع تعطى تقديرات محافظة لسلوك الأنابيب وأن استخدام العنصر المحدود ثلاثى الأبعاد يعطى تقديرات أكثر واقعية لتحميل الأعضاء ذات الاجهاد العالى عن ذلك التحليل المبسط المستخدم فى كود الجمعية الأمريكية للمهندسين الميكانيكيين ولقد تم أيضا طرح طريقة جديدة تزودنا بتحليل أكثر دقة للأعضاء ذات الاجهادات العالية فى حالة عدم معرفة لظروف الحدود •

CONTENTS

CHAPTER 4

ELASTIC BEHAVIOUR OF PIPING UNDER STATIC LOADING

Notation:

F.	Constant Hanger Force.
ΔF	Imbalance Force Component
M.	The Resultant Bending Moment
SFF	Support Force Factor
Δz^n	Vertical Deflection (in Z direction) measured at constant hanger position n under dead-weight
DZ^n	Vertictal Deflection (in Z direction) as a result of piping design calculations under dead weight.
$D\Delta^n$	Deviated Vertical Deflection (in Z direction) from the design vertical deflection under dead-weight.
R_e	Yield Strength
R_m	Ultimate Strength
A_s	Reduction of Area at Rupture
U_o	Ovalization Percent
D max	Major Diameter of Ovalized Elbow
D min	Minor Diameter of Ovalized Elbow
$M_{b,test}$	Bending Moment at the Middle of the Tested Elbow
M_{FPL}	Bending Moment for Fully Plasticity of Straight Pipe.
R	Support Reaction
D_o	Outside Diameter
θ	Arc Length of Elbow
R	Bend Radius of Elbow
t	Wall Thickness
$\bar{A}$	Additional Thickness

A	Area of Cross-Section	mm^2
$L1, L2, L_F$	Length of Adjacent Straight Pipe	mm
A1	Reduction of Area Percentage	%
F	Force	N
SX	Snap Load Case in X Direction	N
SZ	Snap Load Case in Z Direction	N
$\bar{S}_x (\bar{S}_y)$	Shear Force in X Direction (y Direction) due to Force Balance of the System	N
$\bar{\bar{S}}_x (\bar{\bar{S}}_y)$	Shear Force in X Direction (y Direction) due to Moment Balance of the System	
N	Normal Force	N
Mb	Bending Moment	Nm
Mx (My)	Bending Moment in X Direction (y direction)	Nm
T	Torsional Moment	Nm
τ	Shear Stress	N/m^2
ε	Strain	m/m
σ	Stress	N/m^2
Nab, (Mab)	Stress Resultant referred to an arc length where the first suffix gives the direction of the stresses and the second gives the direction of the normal to the plane (a or b = X, Y or Z)	$(\frac{Nm}{m})$
E	Young's Modulus	N/m^2
G	Modulus of Rigidity	N/m^2
μ	Poisson's Ratio	
φ	Circumferential Angle	Degree
α	Arc Angle of Elbow	"
R	Bend Radius of the Elbow	mm

Suffix equiv.: Equivalent

Suffix 0 Outside

Suffix I Inside

Suffix U_a Circumferential Strain or Stress at Outside Surface.

Suffix U_i Circumferential Strain or Stress at Outside Inside Surface

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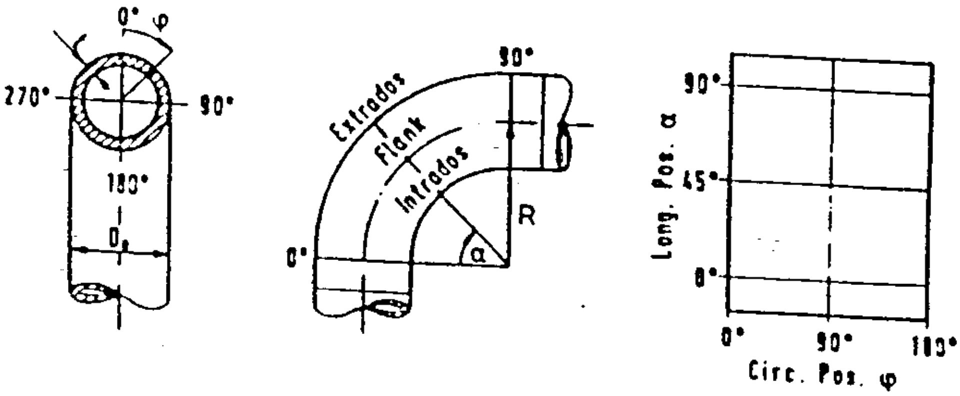

Angle convention and single plane orthomorphic projection of the bend surface

- Abbreviation

MPA: Material Testing Centre, Stuttgart University, Germany

KTA: German Nuclear Safety Standards

RCC: French Nuclear Safety Standards

NDT: Non Destructive Testing

Stress Analysis of Piping Systems In Nuclear Power Plants

Thesis

Submitted for the Partial Fulfillment of the Degree of Doctor of Philosophy in Mechanical Engineering

BY

Mahmoud Ahmed Shafy Haraza

M.Sc - B.Sc Mech. Eng

Under the Supervision of

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CONTENTS

CHAPTER 4

ELASTIC BEHAVIOUR OF PIPING UNDER STATIC LOADING

ACKNOWLEDGEMENT

The author wishes to express his gratitude to Prof. Dr. F.H.Hammad, Chairman of the Atomic Energy Authority (AEA), Prof. Dr.A.S. El-Sabbagh, Design and Production Department, Faculty of Engineering, Ain Shams University, and Dr. H.I. Shaaban, Nuclear Metallurgy Department, Nuclear Research Centre, AEA Cairo. Their supervision and assistance during the course of this work are highly appreciated.

My deep appreciation to Prof. Dr. Kussmaul, Dr. D. Sturm and Dr. H.Diem, MPA, Stuttgart University, is a must. They have allowed me to participate in the PHDR group for Elastic and Plastic Behaviour of Nuclear Component during my training in MPA (Material Testing Centre).

The author wishes to express his appreciation to Dr. Eng. H.D. Gerlach, Mr. M. Ludke and Mr. F. Bietenbeck, Design Approval and Stress Analysis Division, Rheinisch-Westfalische, TUV Essen, FRG. They have allowed me to participate in the running engineering works as well as research work.

My thanks are extended to the IAEA staff who awarded me a fellowship which made stay in FRG possible.

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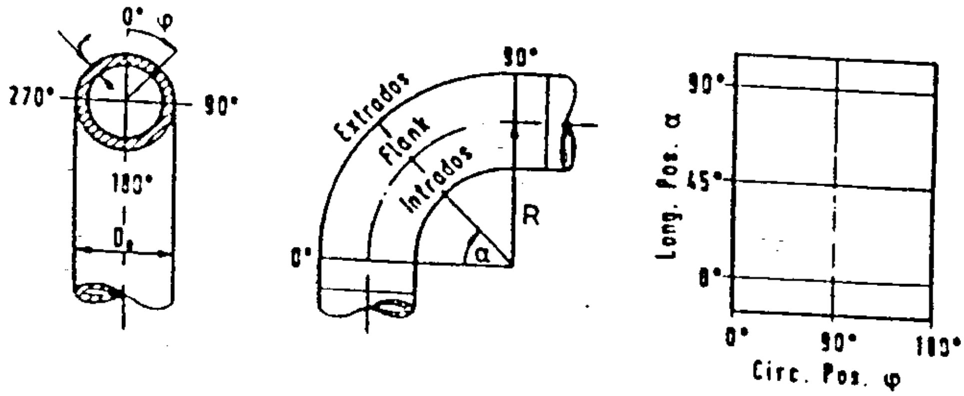

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CHAPTER 1
INTRODUCTION

The most important safety principles applied to nuclear power plants are (1): quality through production principle, worst case principle, continuous in-service monitoring principle and verification and validation codes principle. Achieving quality through production principle requires optimizing all processes related to the production of safety-related components. The worst case principle as well as verification and validation codes principle demands investigating and analyzing the safety-related systems under different conditions such as normal, upset, emergency and faulted conditions (2-6).

One of the most important aspects of safety analysis and assessment in nuclear power plants deals with stress analysis of piping systems. Since the pipe break has a catastrophic consequence on the nuclear power plant, stress analysis is essential to prove that such occurrence is unlikely or only leak before break occurs (7,8). Stress analysis of safety related piping systems is generally performed with member programmes based on the elementary theory of bending providing mostly internal forces and displacements. The results of such kind of stress analysis depend on the accuracy of the different boundary conditions: Stiffness of anchor points, hangers, snubbers, gabs, friction effect, and misalignment. Therefore the results of stress analysis are valid only as long as the boundary conditions are defined and kept unchanged.

The piping boundary conditions should be chosen such that they can carry the piping mechanical loadings without restricting their thermal expansion. Constant restricting hangers are frequently used for supporting the piping. The main advantage of using constant hangers is keeping the load rating virtually constant over a certain range of displacements. Thus, the piping can be allowed to expand without unnecessary stressing. However, the loading rating values of these constant hangers can deviate from the design or nominal values. Significant deviations of the constant hanger load ratings can cause unexpected additional stresses on the piping.

In determining stresses of highly loaded components such as pipe bends are evaluated by stress intensity factors (9) which in some cases do not yield sufficiently accurate results. Also, changing of boundary conditions of piping during service leads to changes in the state of stress and deformations.

In this work the effect of boundary conditions on the piping behaviour was investigated. Secondly, some local components such as pipe bends or elbows, in which boundary condition are unknown accurately, were also studied. Thirdly, the elastic-plastic behaviour of an elbow was experimentally determined under bending moment. The deformation and stress behaviour using finite element of elbows mounted in piping systems without any influence of unknown boundary conditions were studied as well.

In connection to this, experimental investigations were undertaken at a realistic piping one: the feed water piping system of the Decommissioned Heissdampf (Superheated Steam) Reactor (HDR) in Germany, (3) (Appendix A).

Two static loading tests were carried out in the scope of phase II of the HDR safety programme. The global behaviour of the piping system could be determined by measured values of displacements and strains at several cross-sections. Additionally, a detailed strain evaluation was performed by the measurments of 76 strain gauges attached in one bend area. Parallel to these tests the piping behaviour was analysed by finite element calculations for verification and validation purposes.

The behaviour of the structurally critical components such as pipe bends and elbows when the elastic range is exceeded to the elastic-plastic one is also of great interest in safety analysis and assessment. Plastic deformation in highly stressed components could exist when the piping system is subjected to emergency, upset or faulted loading (10). In connection to this an experimental work on simulated elbows of HDR feed water line under in-plane bending moment loading was performed. About 250 strain gauges were installed on the inner and the outer surfaces of the studied elbow.

This thesis falls into 6 chapters. The introduction is given in Chapter 1. Chapter 2 deals with reviewing the technical safety principles of nuclear power plant components. It also deals with some examples about quality through production for one of structurally critical components (pipe bends). Chapter 3 deals with the effect of changing the constant hanger supports on the piping behaviour. Experimental and theoretical analyses of the stresses of the HDR Feed Water Piping are given in Chapter 4. Also, a new approach is developed to analyse the highly stressed components in piping such as elbows, without influencing the global boundary conditions of piping. Chapter 5 deals with the elastic-plastic behaviour of an elbow similar to that of investigated elbow of HDR under in-plane bending moment. General conclusions of the work are given in Chapter 6.

CHAPTER 2

TECHNICAL SAFETY PRINCIPLES FOR NUCLEAR POWER PLANT COMPONENTS

1. INTRODUCTION

Licensing for construction and operation of nuclear power plant is mandatory in all countries. This provision is necessary for the protection of the public against any hazard arising from using nuclear energy through constructing or operating a nuclear installation. The fundamental legal basis for the design, construction and operation of nuclear power plants is generally cited in a law in most countries.

As an example in the Fed. Rep. of Germany 'FRG' (11-13), it must be shown in any licensing procedure that every necessary precaution has been undertaken according to the state of the art (existing scientific knowledge and technology) to prevent damage (14-15).

The main object of this chapter is to review some aspects of safety-related principles which are applied to achieve certain level of nuclear safety. Also a comparison of the world wide safety related standards is given.

2.2 SOME TECHNICAL SAFETY PRINCIPLES:

The fundamental safety features of Nuclear Power Plant (NPP) components depend on the following (16):

- high-quality materials,
- a conservative restriction of stresses,
- the prevention of stress peaks through the optimal design,
- the assurance of the application of optimized manufacturing and testing technologies, and
- the knowledge and assessment of possible faulty accident conditions.

- If the above requirements are complied with, then a fundamental safety level will be achieved that will preclude any disastrous failure of a plant component during manufacturing or operation.

- The leak before break concept is in close connection with the principles of the 'fundamental safety' (8-17). The prerequisite for exclusion of catastrophic failure or leak before break concept are summarized in the following (17-18):-

1- quality through production principle; this requires optimizing the design, material and manufacturing processes.

2- Worst case principle;

This principle deals with the failure invesitgation as well as probabilistic assessment. The worst case is the state below which the condition of the material cannot fall. The assessment of this situation requires ongoing work on failure investigation. Also probabilistic risk analysis has to be applied.

3- Continuous in-service monitoring and documentation principle: in-service monitoring surveilance requires Non Destructive Testings 'NDT'.

4- Validation principle:
This includes verification and validation of codes and standards that determine the safety margin of components and systems.

In general the concept of safety margin is given by the ratio of the applied load (K) and the material resistance or load bearing capacity (R). This concept is illustrated in Fig. 2.1. If the condition $K < R$ is violated, failure can occur. Such a case can be caused by: Extensive local or general deformation including instability, crack initiation and stable or unstable crack growth (ductile and brittle fracture).

Regarding the reactor pressure vessel and piping, the exclusion of catastrophic fracture is a tantamount to the requirements of preventing crack initiation right from the beginning, or in the case of initiation, at least to assure crack arrest.

This condition $K < R$ can be fulfilled if all factors which influence loading and loading capacity are completely and quantitativly taken into consideration in production and during service.

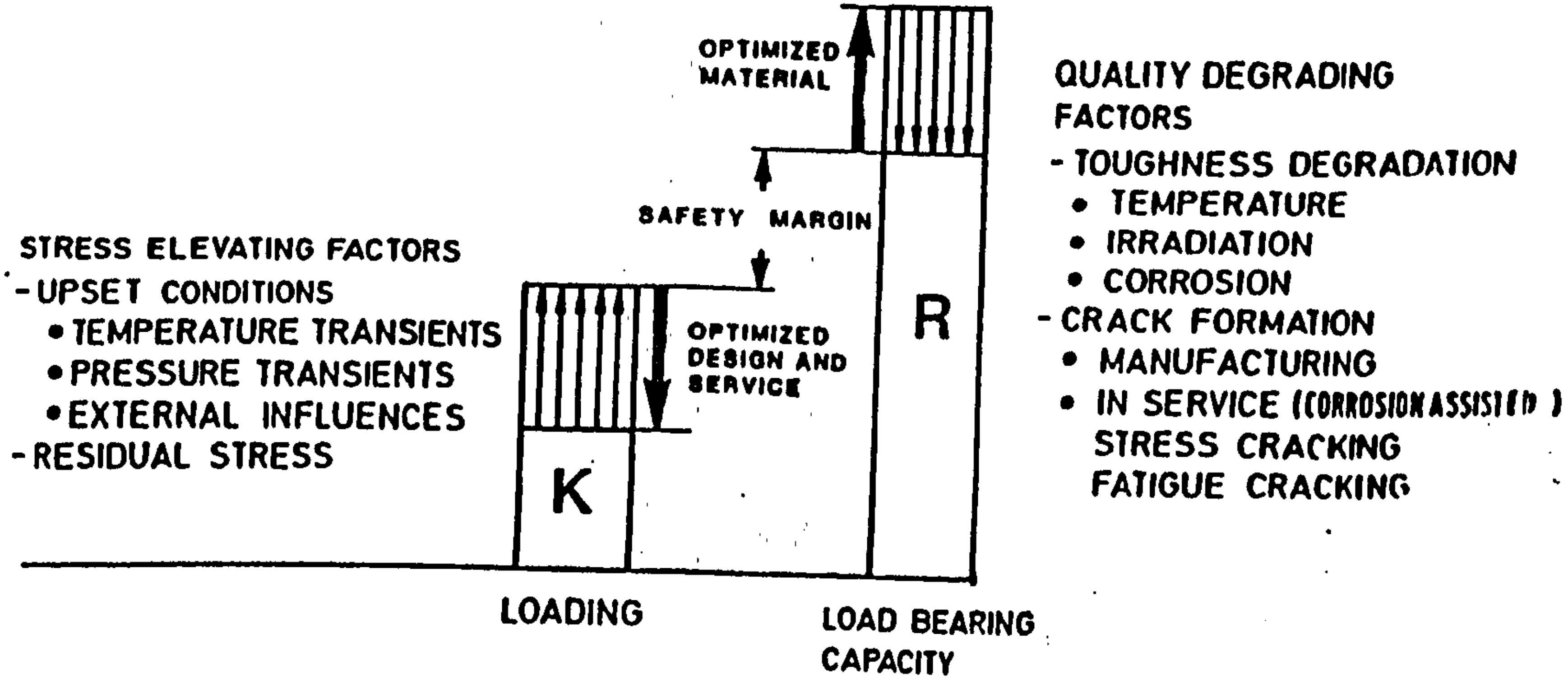


Fig. 2.1 : Safety margin of components and systems

2.3 SOME ASPECTS OF QUALITY THROUGH PRODUCTION PRINCIPLE

2.3.1 Design

Optimized design is a part of the basic safety concepts. The generic features for fracture-safe design could be summarized as follows:

1- Using integral design, which leads to reduction of weldments

2- Locating weld beads out of regions of high integral and local stresses as well as residual stresses.

3- Good access is provided during manufacture and pre-service and in-service inspection.

4- Achieving low nominal and local stresses by using simple smooth shapes with minimum discontinuities.

5- Avoiding as far as possible longitudinal weldments through use of forged rings and piping instead of formed plates.

6- Reducing weld volume (by using narrow-gap technique).

7- Avoiding one side weldments

8- Avoiding fillet welds

9- Locating reinforcements in the vessel or piping wall instead of in the nozzles.

10-Positioning weldments remote from critical locations with respect to mechanical and thermal loading and irradiation exposure.

Fig. 2.2a gives some examples for advanced design for weldments in NPP in order to reduce stress intensity. The main features are associated with adequate reinforcement which has to be done in the main body and relocation of the welds from the disturbed areas to regions of lower stresses.

Advanced design of weldments also takes into consideration improvement of defect detectability and defect interpretation. Fig. 2.2b shows some examples of improving the accessibility in NDT (18).
Fig. 2.2c shows the implementation of these features in primary circuit components. The optimum design solution for the pressure vessel is combining the flange and nozzle parts into one single, smooth and high reinforced ring with large ligaments between the nozzle penetrations and set on nozzles. Such design is called integral nozzle shell design. It is now a feature of water reactor pressure vessels supplied by vendors (16).

2.3.2 Materials and Manufacturing

The material properties are influenced by chemical composition, melting technology, manufacturing procedure as well as heat treatment.

Also an examination has to be made as to whether the fracture mechanics parameters can be correlated with conventional strength and toughness parameters that have been determined as part of the quality assurance programme. Table 2.1 shows the requirements for the chemical composition of reactor pressure vessel steels and container vessels. They are intended to guarantee deterministically the structural integrity even with local defects and cracks (17-18).

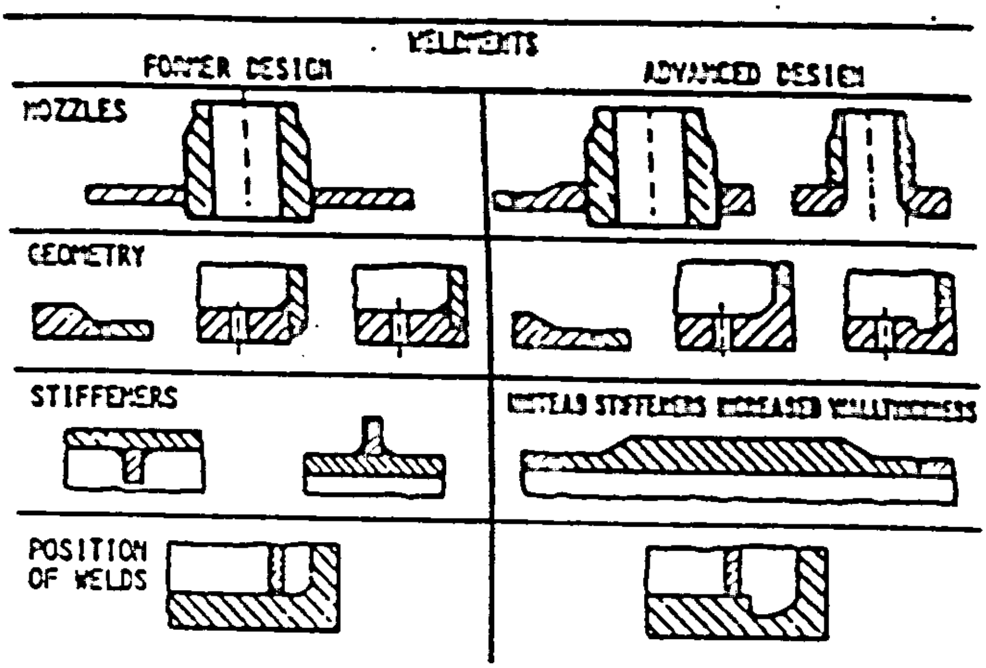
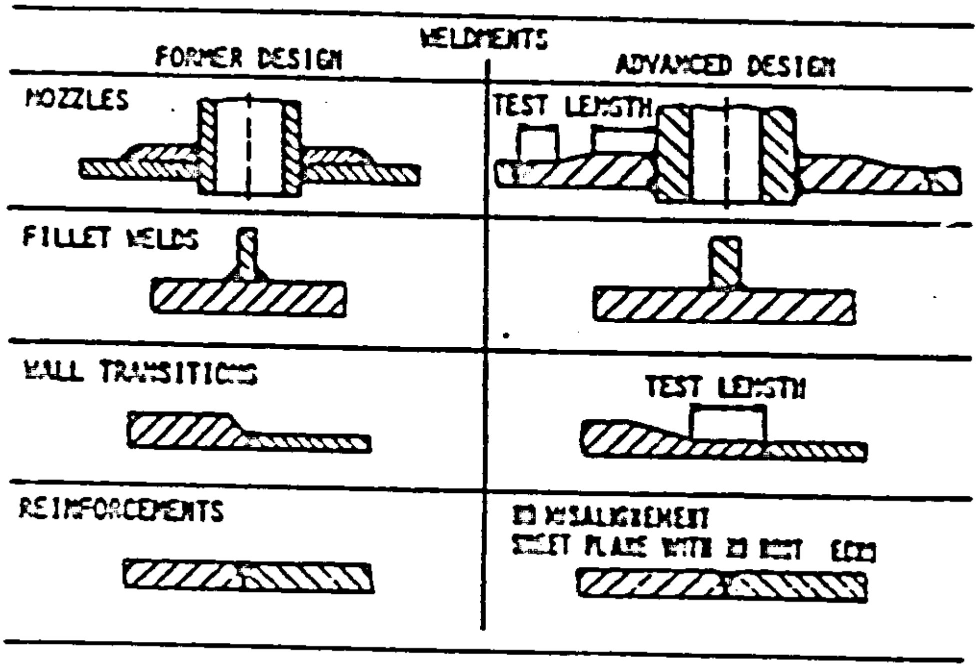
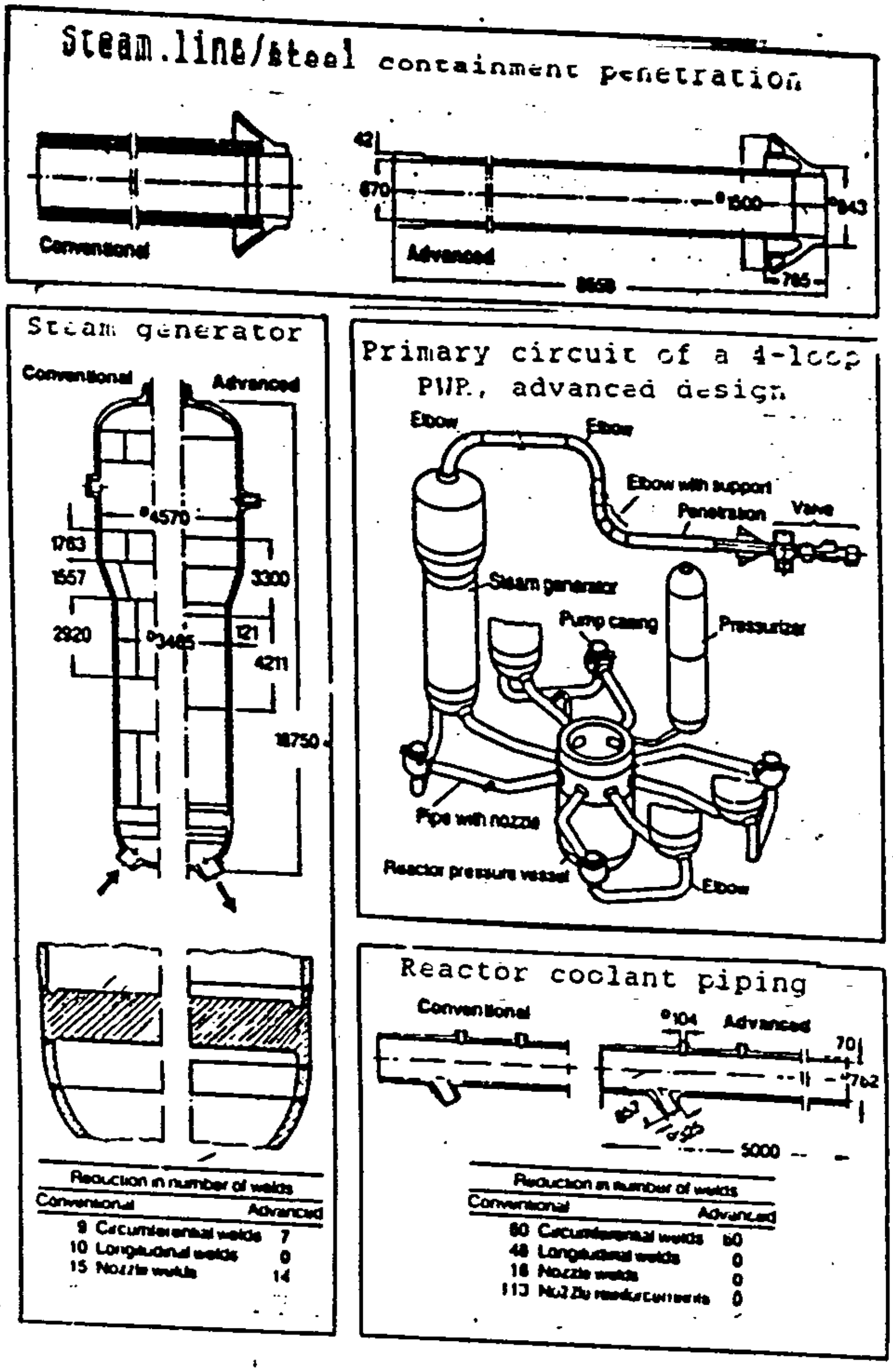
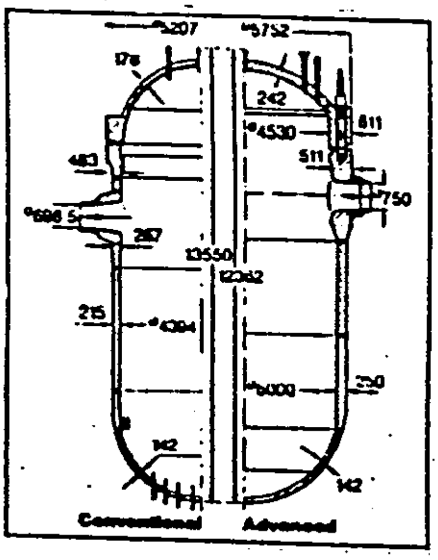


Fig. 2.2a : Decrease of stress intensity through advanced design



g. 2.2b : Improvement of defect detectability and defect interpretation through advanced design





Conventional and advanced style of reactor pressure vessel for a PWR.

Fig. 2.2.C Examples of conventional and advanced design for primary and secondary circuit components of PWR.

	wt %								
	C	Si	Mn	P	S	Cr	Mo	Ni	Cu
22 NiMoCr 3 7	≤.20	.20⁺	.85⁺	≤.008	≤.008	≤.40	≤.55	1.20⁺	≤.10
20 MnMoNi 5 5	.15 .25	.10 .35	1.15 1.55	≤.012	≤.008	≤.20	.40 .55	.45 .85	≤.12 (≤.10)
15 MnNi 6 3	.12 .18	.15 .35	1.20 1.65	≤.015	≤.005	≤.15	≤.05	.50 .85	≤.06

Continuing	wt %							
	Al	V	Sn	N	As	Sb	Ta	Co
22 NiMoCr 3 7	.010 .040	≤.01	≤.011	≤.013	≤.015	≤.005	≤.030	≤.030
20 MnMoNi 5 5	.010 .040	≤.02	≤.011	≤.013	≤.025	--	≤.030	≤.030
15 MnNi 6 3	.02 .05	≤.02	≤.01	≤.015	≤.015			

⁺ Guiding value ( ) Belt line region

Table 2.1, Requirements for chemical composition (RPV and Steel Containment Vessels) according to the "Basis Safety Concept"

العنوان:	Stress Analysis of Piping Systems in Nuclear Power Plants
المؤلف الرئيسـي:	Haraza, Mahmoud Ahmed Shafy
مؤلفين آخرين:	Hammad, F. H.، El Sabbagh, A. S.(super)
التاريخ الميلادي:	1990
موقع:	القاهرة
الصفحات:	1 - 251
رقم MD:	565698
نوع المحتوى:	رسائل جامعية
اللغة:	English
الدرجة العلمية:	رسالة دكتوراه
الجامعة:	جامعة عين شمس
الكلية:	كلية الهندسة
الدولة:	مصر
قواعد المعلومات:	Dissertations
مواضيع:	الهندسة الميكانيكية، المفاعلات النووية
رابط:	https://search.mandumah.com/Record/565698

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Thesis

Submitted for the Partial Fulfillment of the Degree of Doctor of Philosophy in Mechanical Engineering

BY

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1990

بسم الله الرحمن الرحيم

تحليل الاجهادات لأنظمة الأنابيب في المحطات النوويه

رساله مقدمه الى

كلية الهندسه - جامعة عين شمس

للحصول على

درجة دكتوراه الفلسفة فى الهندسة الميكانيكية - هندسة الانتاج

مقدمه من

مهندس / محمود أحمد شافعى حمزه

قسم توكيد ورقابة الجوده

مركز التنظيم والأمان النووى - هيئة الطاقة الذريه

القاهره - جمهورية مصر العربيه

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١٩٩٠