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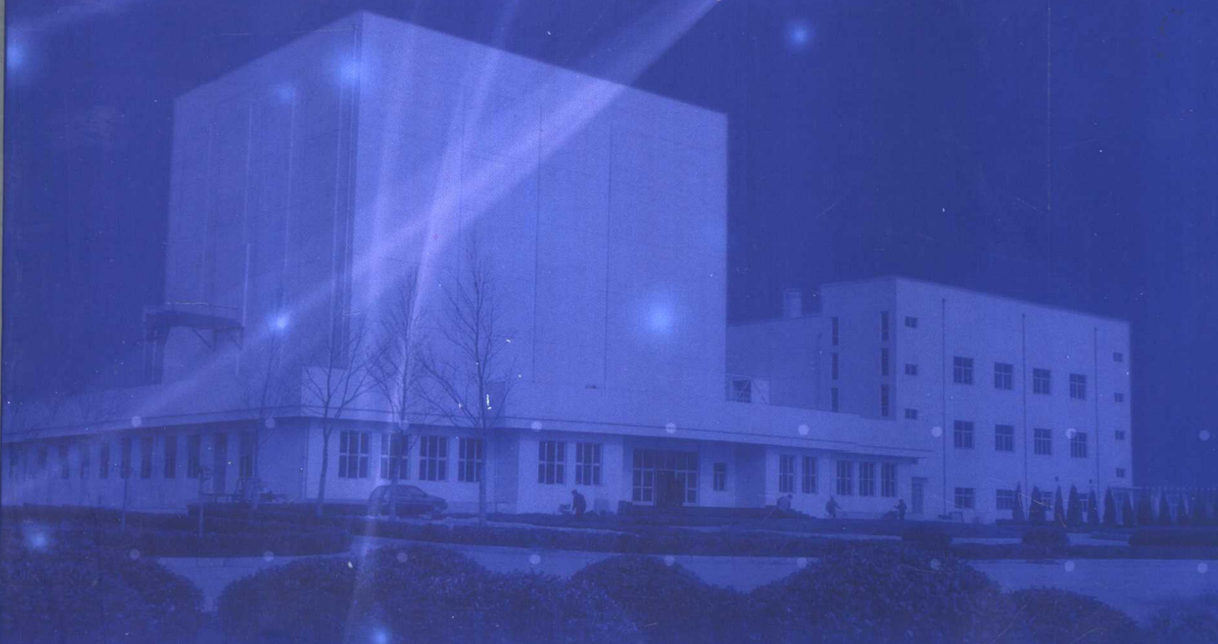
THE NEW TECHNOLOGY IN HIGH TEMPERATURE GAS-COOLED REACTOR

-THE FIFTH ANNIVERSARY OF CEA /INET COOPERATION

Editor in Chief Suyuan Yu



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内容简介

自中国和法国政府于1983年签署和平利用原子能协议以来,两国在核电领域开展了多种多样富有成效的合作。清华大学核研院自20世纪70年代开始进行高温气冷堆的研发工作,于2000年建成10兆瓦高温气冷堆并达到首次临界。该气冷堆是目前世界上唯一一座正在运行的同类型反应堆,享有独立的技术知识产权,属于第四代核能技术。在中国国家原子能机构和法国原子能委员会的指导下,清华大学于2001年加入中法核科技人员交流项目。应法国原子能委员会的邀请,先后派出十几位科学研究人员赴法进行合作研究。5年来,研究涉及高温堆石墨材料、容器、燃料循环等方面。本书收集的10篇文章是清华大学与法方的科研人员共同撰写的研究报告,详细记录了双方在以上领域所开展的研究及其进展。通过阅读此书,将使核能技术人员对高温堆有更深入的了解,对核材料研究、核设备设计等工程技术人员有参考价值。

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Preface Ⅰ

After one year preparation, the New Technology in the High Temperature Gas-Coolded Reactor commemoration of the CEA/INET Cooperation under the China-France Nuclear Cooperation Framework finally avails itself to the expectation from both sides. This volume with a collection of 10 papers not only keeps a record of technical progress contributed by both sides, but also marks the fifth anniversary of cooperation between CEA and INET.

In 2001, CEA and INET signed the first agreement and a series of specific topics of cooperation (STC), which officially launched the cooperated program on HTGR technology between the two sides. This is the first time that the two research organs work so closely on the internationally focused project. At the beginning of this century, nuclear energy was reconsidered as an option for sustainable development due to the concerns over the decreasing reserve and otherwise soaring price of oil, the global warming issues. The future reactors are expected to be safety-enhanced, cost-effective and environmental-friendly. However, as the dawn comes, a question is frequently asked "how can the reactor technologically meet the demands in the new era?" Foreseeing the technological evolution, INET has conducted systematic research ever since 1970s. The successful construction and operation of the 10 MW High Temperature Gas-Cooled Reactor (HTR-10) helps accumulate experience in a complete process from reactor design, fuel fabrication and construction, commissioning and operation. HTR-10 therefore serves as a research base on HTGR. Meanwhile, as a government funded research organ, CEA consistently works for the reactor technology advancement from the first batch of commercial reactor to the optimization of current GII and GIII technology which helps France maintain its leading position in nuclear industry. As a leading research organ, CEA is targeting GIV technology by taking HTGR technology as the starting point. Since 2002, INET has sent dozens of researchers to CEA on both short and long term basis to work with their French counterparts. Over the past five years, they have worked together in a variety of fields, such as material, fuel, components and reactor physics. Besides, both sides sponsored workshops and technical visits to tackle the major problems. As it demonstrated, the exchange in the past years has enhanced mutual understanding, channeled and shared resources available, sparkled inspiration. It is more than pleased to find that friendship grows through the cooperation which can find expression in their appreciation of each other's culture.

Thanks to the experiences gained in the past years, it is confident that both sides could make joint efforts to pursue technological advancement in improving energy generation efficiency, hydrogen production and other application. It is expected that such exchanges could be intensive in scope and various in forms. Meanwhile, the French scientists are welcomed to INET for research.

Our special thanks will go to CAEA and CEA who initiate the nuclear cooperation framework enabling the scientific exchange. Thanks to CEA for the efficient and considerate arrangement from the STC preparation to logistics which are of great help to our researchers. The nuclear service of the French Embassy in Beijing also makes great contribution to the successful program whenever efforts are required. The coordinated researchers from the French side should accept our sincere thanks for their help to our researchers in different ways which facilitates them to adjust to a complete new environment.

Five years is a long but short time. It is long because our researchers have spent a great amount of time on the HTGR technology; it is short because looking into the future, it is only just a beginning. We hope that the five year cooperation paves the way for the future achievements. We are also expecting the release of the similar book of achievements in another five years.

Suyuan Yu
Coordinator
Vice President
Institute of Nuclear and New Energy Technology (INET)
Oct. 28, 2006

Preface Ⅱ

CEA is very pleased to preface, jointly with the CAEA, this report which seal more than four years of collaboration between CEA and INET in the frame of High Temperature Gas Reactor technology.

The collaboration with INET in the field of HTGR was concerted and focus on relevant common interests to meet in a first phase two of the main criteria of such reactors in the 4th generation context:

- Safety during the plant operation and in accidental situations.
- A reduced impact on the environment.

The first act of this collaboration begins in September, 2001 with the definition of three very first STC. The first attaches from the INET were welcomed in France in April, 2002.

Since the beginning of this collaboration 11 attaches went at CEA in France, 10 at Cadarache CEA center and one in Saclay CEA center.

The specific topics of collaboration (STC) cover a large scope of activities:

- High temperature structure mechanics
- Studies on graphite (accidental oxidation and feasibility of a protected SiC coating)
- Thermal hydraulic modeling of components and systems
- Safety studies (Leak Before Break methodology)
- Modeling of fuel particle behaviour under irradiation.
- Reactor physics of HTGR core.
- Helium technology (purification)
- Waste management

A very significant level of activity focused on mechanical studies and graphite behaviour as high temperature strongly impact materials and structural behaviour and so the needs of characterization appear as a priority.

The contacts and technical exchanges between Chinese attaches and French engineers were at a high quality level. One of the main reason is that both sides have complementary interest : Chinese attaches have a strong feedback experience with the experimental reactor HTR-10 and French program is mainly based on the development of high quality fuel particle manufacturing with associated irradiation tests, investigating by specific benches helium technology, and upgrading and developing numerical simulations.

This report will be a very useful technical assessment of the first phase of the collaboration and will contribute also to the elaboration of the second phase. For this next phase we expect a reinforcement of technical exchanges on relevant topics as safety, design analysis, helium technology. and code to code benchmarks. For the next future, exchange of people on both sides are expected.

We hope that this document will also contribute to a sustainable willingness of collaboration between CEA and INET as the nuclear high temperature technology challenge must be accepted within an international context.

Jacques Rouault
Coordinator
CEA
Oct. 28, 2006

Preface Ⅲ

Nuclear energy, one of the significant discoveries in the development of mankind, has remained controversial ever since it was discovered in the last century. If appropriately handled, it could bless the people to large extent. But its contribution was shadowed and dwarfed by its devastating power. Despite its dismay features, some pioneer countries bravely ventured in this wonderland.

Although France is not the first explorer in this field, it quickly becomes the main player and tells a successful story as to how to harness atomic energy. Up to now, over 80% of the total energy supply of France has come from nuclear power. In contrast, Chinese nuclear power industry legged behind a little bit. In the early 1980s, while leaping forward Chinese economy faced the serious pollution derived from the usage of fossil energy, nuclear energy began to draw a wide attention at national level and was given a strategic importance. In order to develop the nuclear power in safe, efficient and sustainable manner, China adopted the policy of self-reliant R&D with international cooperation. Based on the friendly bilateral relationship and mutual benefit, China and France kicked off their nuclear cooperation after the first Protocol was signed in 1983. Ever since the two countries have carried out the collaboration and exchange in various forms.

Among the many Chinese R&D entities participating in Sino-French cooperation, The Institute of Nuclear and New Energy Technology (INET) is not the big undertaker but a unique player. In 1970s, INET started to carry out R&D on the advanced nuclear reactor-High Temperature Gas-Cooled Reactor (HTGR). After over two decade efforts, INET successfully built the 10 MW HTGR-10, the only reactor of the type in operation since its criticality in 2000. The successful construction and sounding operation make it an ideal base of research and development. In order to improve its research work, INET began to pursue the international exchange and cooperation on HTGR. At the time, France, as a member of GIF, expressed its willingness to seek the international associates to pursue the high temperature gas-cooled reactor, which was identified as one of technology concepts in GIF Technology Roadmap. Under the negotiation and coordination of China Atomic Energy Authority (CAEA) and Commission of Atomic Energy (CEA), INET joined the CAEA and CEA bilateral scientific exchange program, covering HTGR material, components, fuel, thermal hydraulics and etc. By combining the HTGR R&D and NPP operation experience, INET and labs of CEA worked together to make significant achievements in the past five years, evidenced by the papers included in this book, and have become an indispensable part of Sino-French nuclear R&D collaboration.

In the future, CAEA is willing to consolidate the platform of governmental cooperation and continuously support the Chinese R&D entities including INET to joint hands with CEA's labs to tackle the challenges and difficulties in the front line of nuclear technology.

Wei Huang
Coordinator in Chief
CAEA
Oct. 28, 2006

Preface Ⅳ

Beijing, Wednesday October the 11th, 2006

The bilateral collaboration between France and China in the field of nuclear energy and the related R & D is ancient: nearly 25 years, and fruitful.

INET, being one of the major players in Chinese nuclear research, has been a partner to this collaboration for many years and thus has given to many students and post-docs the opportunity to work in France or in connection with French programs.

It is all to the credit of Professor Suyuan Yu to have had the good idea to publish the works done in this context.

An excellent idea indeed, not only because it gives a sense of achievement and pride to all the participants in these works, but also because the precise knowledge of what has been achieved is paramount to future researchers.

Nuclear energy is a long term job, and the collection of works published today is a milestone on the road.

Congratulations to Professor Suyuan Yu and his students for their excellent work!

Alain Tournyol du Clos
Nuclear counsellor
French Embassy
Beijing

CONTENTS

LBB Analysis of Cross Vessel of GT-MHR

Zhengming Zhang[1), M. T. Cabrillat[2), Y. Lejeail[2), B. Michel[2)

(1. *Institute of Nuclear and New Energy Technology, Tsinghua University, Beijing* 100084, *China* 2. *CEA Cadarache, DEN/DER/SESI, France*)

Abstract: The Cross Vessel (CV) of GT-MHR is a horizontal pressurized vessel. Besides bearing interior pressure and temperature variation, it also withstands large global axial force and bending moment. Therefore, the CV has the dual features of a pressure vessel and a pressurized pipe. The CV is a weak part of the Primary Loop Boundary Components due to its complex loads. It shall be designed and analyzed carefully. This paper gives a preliminary analysis of the Leak-Before-Break (LBB) characteristic of the CV. Analysis shows that the CV wall can satisfy the requirements of LBB even under severe stress condition and large detectable coolant leakage.

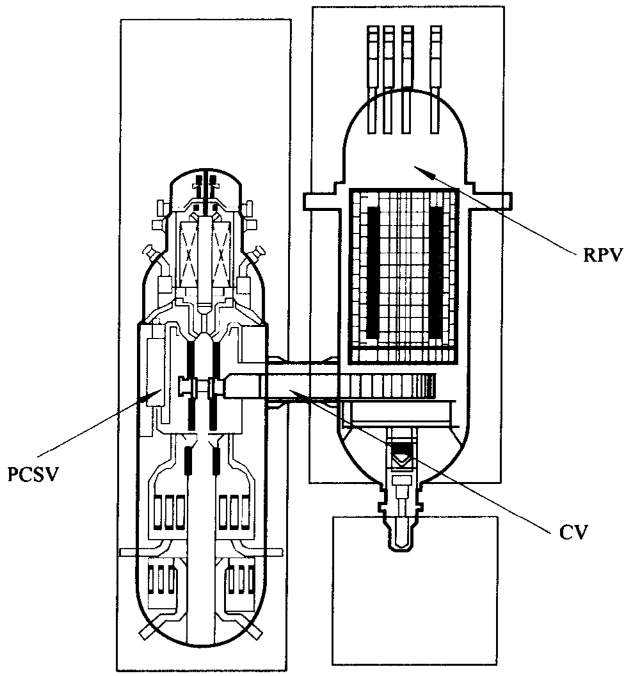
Key words: GT-MHR, Cross Vessel, Leak-Before-Break Analysis

1. Introduction

The GT-MHR consists of two vessels: the Reactor Pressure Vessel (RPV) and the Power Conversion System Vessel (PCSV). They are linked by a horizontal vessel referred to Cross Vessel (CV) (see Fig. 1).

Compared with the general pressure vessel, the CV has the dual features of a pressure vessel and a pressurized pipe. During normal operation, the main load is the interior pressure and the temperature variation, which are the same with a general pressure vessel. But, with other large loads on the CV, such as seismic loads, the restraints at both CV ends, and the thermal movements of the RPV and the PCSV, the CV behaves much more like a pressurized pipe.

The CV is a weak part of the Primary Loop Boundary Components (PLBC) due to its complex loads. It shall be designed and analyzed carefully. This report will give a preliminary analysis of the Leak-Before-Break (LBB) characteristic of the CV.



GT-MHR

Fig. 1 The Primary Loop boundaries of GT-MHR

2. LBB Characteristic and its Analysis Procedures

In the reactor pressure vessel or the piping system, crack may exist in the wall and will propagate to penetrate the wall under the operating loads. If this crack can generate enough coolant leakage before it growths unstably, and if there will be enough duration between the detection of the coolant leakage and the implement of the safety measures, the vessel or the pipes will be said to have LBB characteristic.

The purpose of LBB analysis is to ensure the safety of a component containing certain defect. Up to now, there is no generally adopted LBB analysis procedure. The most commonly used procedures are the English R6 method, the French RCC-MR method, and the US LBB. NRC method.

The HTR-10 has the similar PLBC arrangement as that of the GT-MHR, and its Hot Gas Duct Vessel (HGDV) plays the similar role as that of the CV.

The LBB characteristic of the HGDV had been analyzed using the US LBB. NRC method. Based on the same procedure, the LBB characteristic of the CV will be analyzed.

3. Design Parameters

The CV is a horizontally placed cylindrical vessel shell. Its two ends are welded with the RPV and the PCSV respectively. Its main design parameters are listed in Tab. 1.

Tab. 1 The main design parameters of the CV

Inner diameter, D	2 300 mm
Wall thickness, t	100 mm
Operating pressure, P	7.15 MPa
Internal helium temperature	488 ℃
External air temperature	65 ℃
Material	9Cr1Mo mod
Internal thermal insulation layer	Steel shield of 35 mm thickness
External thermal insulation layer	Kaolin wool of 100 mm thickness

Steel shield is 25 mm thick and external thermal insulation exists only near the PCSV. There is no external thermal insulation on the part of the cross vessel which is in the reactor cavity.

4. Stress Analysis

The steel shield inside the CV is omitted in stress analysis. This means that the steel shield plays only the role of thermal insulation and does not contribute to the stiffness of the CV.

4.1 Level A Condition

4.1.1 Stress Calculation

Under Level A condition, the main loads on the CV are the inside pressure and the temperature difference between the CV internal and external surface.

The stresses generated by the inside pressure are:

Axial stress: $\sigma_{t1}^{A}=\frac{P\cdot D^2}{(D+2t)^2-D^2}=39.40\ (\text{MPa})$

Circumferential stress: $\sigma_{t2}^{A}=\frac{P \cdot D}{2t}=82.23$ (MPa)

No bending stress is generated by the inside pressure.

Thermal transmission analysis shows that the temperature at the CV internal surface is 479.31 ℃, and the temperature at the CV external surface is 465.95 ℃. This temperature difference will generate a local bending stress of 20.39MPa. This stress is restricted in local CV wall and is not the bending stress used in LBB analysis. However, it will contribute to the axial stress. Conservatively, we simply add this stress into σ_{t2}^{A}.

As mentioned above, the CV bears complex loads. Among these loads, only the inside pressure will generate circumferential stress, while other loads will mainly generate axial stress. Therefore, in the following analysis, only axial stress is considered. So the maximum stresses in the CV under Level A condition are:

Axial stress: $\sigma_{t}^{A}=39.40+20.39=59.79$ (MPa)

Bending stress: $\sigma_{b}^{A}=0$

4.1.2 Bending Stress Discussion

In the US LBB. NRC method, the bending stress means a global bending stress generated by the bending moment acting on the pipe. The local bending stress in the wall, such as that generated by the thermal gradient through the wall, is not included in this method. It will not introduce large error for the pipe. But for a pressure vessel that could have large bending stress, this method could cause non-conservative results because the local bending stress tends to close the crack and therefore tends to decrease the crack opening width and increase the crack length (under the premise that the crack open area keeps constant).

For the CV of GT-MHR, the only local bending stress is generated by the thermal gradient through the wall, which is 20.39 MPa. Other loads all generate global membrane and bending stresses in the CV wall, which should be significantly larger than the local bending stress. This means that the CV behaves much more like a pipe than a pressure vessel. Using the US LBB. NRC method, which means treating the local bending stress as the membrane stress, will not lead to severe non-conservative results.

4.2 Level D Condition

Under Level D condition, additional loads should be considered. They include:

- Seismic loads. Under earthquake, the CV behaves much more like a pipe. The vibration of the RPV and the PCSV will generate axial force, bending moment and torque on the CV. If the supports of the RPV and the PCSV are properly designed, the torque on the CV shall be very small and can be omitted. This means that the CV will not generate twist and the shear stress can be neglected. While the axial force and the bending moment will generate axial stress and bending stress in the CV wall.
- Friction loads. During power rising and descending, the RPV or the PCSV will move along the CV axial direction. This movement will generate friction forces at the supports and then generate axial load on the CV. This load will also increase the axial stress in the CV wall.
- Installation loads. The installation procedure is not defined yet. But practically speaking, there will always exist errors when connecting the CV with the RPV and the PCSV. So axial force and bending moment shall be generated during installation. These loads will contribute to the axial stress and bending stress in the CV wall.

Because the above loads have not been analyzed, only a simple hypothesis is used in this report. We simply increase the stresses in Level A condition with 50 MPa to represent the corresponding stresses in Level D condition and consider that all the loads are combined based on individual absolute values (If the loads are combined linearly, 1.4 should multiply the combined results) [1]. So the maximum stresses in the CV under Level D condition are:

Axial stress: $\sigma_t^D = 59.79 + 50 = 109.79$ (MPa)

Bending stress: $\sigma_b^D = 50$ (MPa)

5. LBB Analysis

5.1 Material Properties

The CV is made up of "9Cr1Mo mod", and the welding material is "9Cr1Mo mod weld". Because there are two circumferential welds in the CV shell, both the properties of the parent and the welding materials are considered, and the smaller one is adopted. The operating temperature of the CV is chosen as 472.63 ℃, which is the mean temperature of the CV internal and external surfaces.

Based on Reference [2], the material properties used in LBB analysis are listed in Tab. 2.

Tab. 2 Material Properties of the CV

Young's modulus, E	1.78×10^5 MPa
Poisson ratio, ν	0.3
Yield stress, σ_s	300 MPa
Rupture stress, σ_u	420 MPa
Fracture toughness, K_{IC}	190 MPa · $m^{1/2}$

5.2 Postulated Through-wall Crack

The postulated through-wall crack will be located in the maximum stress region, and perpendicular to the direction of the maximum stress. Based on the above-mentioned stress analysis, the crack is assumed to be a circumferential crack and only mode Ⅰ of the crack propagation is considered.

5.3 Detectable Coolant Leakage

After the crack penetrates the wall, the helium coolant leakage from the PLBC shall be large enough to be detected by the protection system of the reactor and can be distinguished clearly from the normal leakage of the PLBC. Because the protection system has not been designed, we simply assume that a helium leak rate of 10 m^3 per hour (at standard condition) shall significantly exceed the normal leakage and can be determined as an abnormal phenomenon.

The helium gas density is 0.179 kg/m^3 at standard condition. Considering a safety margin of 10[1], the mass leak rate used in LBB analysis is (at standard condition):

$$Q_m = 10 \times 10\ m^3/h \times 0.179\ kg/m^3 \div 3\,600\ s/h = 0.004\,97\ kg/s$$

5.4 Leak Flow Calculation

During the analysis of the helium gas flowing through the crack, the following assumptions are considered:

- The helium gas is ideal gas.
- The thermal transmission between the helium and the crack can be neglected because the helium flows through the crack rapidly.
- The friction between the helium gas and the crack wall is neglected