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FUEL DESIGN
FOR
PRESSURISED HEAVY WATER REACTORS

Atomic Energy Regulatory Board
Mumbai-400 094
India

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Price:

Orders for this Guide should be addressed to:

Administrative Officer
Atomic Energy Regulatory Board
Niyamak Bhavan
Anushaktinagar
Mumbai-400 094
India

FOREWORD

Activities concerning establishment and utilisation of nuclear facilities and use of radioactive sources are to be carried out in India in accordance with the provisions of the Atomic Energy Act, 1962. In pursuance of the objective to ensure safety of members of the public and occupational workers as well as protection of environment, the Atomic Energy Regulatory Board has been entrusted with the responsibility of laying down safety standards and framing rules and regulations for such activities. The Board has, therefore, undertaken a programme of developing safety standards, codes of practice and related guides and manuals for the purpose. These documents cover aspects such as siting, design, construction, operation, quality assurance, decommissioning and regulation of nuclear and radiation facilities.

Codes of practice and safety standards are formulated on the basis of internationally accepted safety criteria for design, construction and operation of specific equipment, systems, structures and components of nuclear and radiation facilities. Safety codes establish the objectives and set minimum requirements that shall be fulfilled to provide adequate assurance for safety. Safety guides elaborate various requirements and furnish approaches for their implementation. Safety manuals deal with specific topics and contain detailed scientific and technical information on the subject. These documents are prepared by experts in the relevant fields and are extensively reviewed by advisory committees of the Board before they are published. The documents are revised, when necessary, in the light of the experience and feedback from users as well as new developments in the field.

The 'Code of Practice on Design for Safety in Pressurised Heavy Water Reactor Based Nuclear Power Plants' (AERB/SC/D, 1989) lays down the minimum requirements for ensuring adequate safety in nuclear power plant design. This safety guide is one of a series of guides, which have been issued or are under preparation, to describe and elaborate the specific parts of the code. It provides guidance to the fuel designers for safe design of fuel requirements stated in the code.

Consistent with the accepted practice, 'shall', 'should' and 'may' are used in the guide to distinguish between a firm requirement, a recommendation and a desirable option, respectively. Appendices are an integral part of the document, whereas annexures, footnotes, references/bibliography and lists of participants are included to provide information that might be helpful to the user. Approaches for implementation different to those set out in the guide may be acceptable, if they provide comparable assurance against undue risk to the health and safety of the occupational workers and the general public and protection of the environment.

For aspects not covered in this guide, applicable and acceptable national and international standards, codes and guides should be followed. Non-radiological aspects of industrial safety and environmental protection are not explicitly considered. Industrial safety is ensured through compliance with the applicable provisions of the Factories Act, 1948 and the Atomic Energy (Factories) Rules, 1996.

This guide has been prepared by specialists in the field drawn from Atomic Energy Regulatory Board, Bhabha Atomic Research Centre, Indira Gandhi Centre for Atomic Research, Nuclear Power Corporation of India and other consultants. It has been reviewed by the relevant AERB Advisory Committee on Codes and Guides and the Advisory Committee on Nuclear Safety.

AERB wishes to thank all individuals and organisations who have prepared and reviewed the draft and helped in its finalisation. The list of persons, who have participated in this task, along with their affiliations, is included for information.

(Suhas P. Sukhatme)
Chairman, AERB

DEFINITIONS

Accident Conditions [1]

Substantial deviations from Operational States which could lead to release of unacceptable quantities of radioactive materials. They are more severe than anticipated operational occurrences and include Design Basis Accidents as well as beyond Design Basis Accidents.

Anticipated Operational Occurrences [2] **(AOOs)**

All operational process deviating from normal operation which is expected to occur during the operating lifetime of a facility but which in view of appropriate design provisions, does not cause any significant damage to Items Important to Safety nor lead to Accident Conditions.

Atomic Energy Regulatory Board (AERB)

A national authority designated by the Government of India having the legal authority for issuing regulatory consent for various activities related to the nuclear and radiation facility and to perform safety and regulatory functions including enforcement for the protection of the site personnal, the public and the environment from undue radiation hazards.

Normal Operation

Operation of a plant or equipment within specified operational limits and conditions. In case of a nuclear power plant, this includes start-up, power operation, shutting down, shutdown state, maintenance, testing and refuelling.

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[1] A substantial deviation may be a major fuel failure, a loss of coolant accident (LOCA) etc. Examples of engineered safety features are: emergency core cooling system (ECCS), Containment.

[2] Examples of anticipated operational occurrences are loss of normal electric power and faults such as turbine trip, malfunction of individual items of normally running plant, failure of individual items of control equipment to function, loss of power to main coolant pump.

SPECIAL DEFINITIONS
(Specific for the present guide)

Bundle Power Envelope

The power variation profile of a maximum rated fuel bundle as its burn-up progresses in the reactor core.

Cladding[3]

An external sheath of material over nuclear fuel or other material that provides protection from a chemically reactive environment and containment of radioactive products produced during irradiation of the composite. It may provide a structural support.

Fuel Bundle (also called Fuel Assembly)

An assembly of fuel elements identified as a single unit.

Fuel Element

A component of fuel assembly that consists primarily of nuclear fuel and its encapsulating materials.

Fuel Handling

All activities relating to receipt, inspection, storage and loading of unirradiated fuel into the core, unloading of irradiated fuel from the core, its transfer, inspection, storage and despatch from the nuclear power plant.

Spent Fuel

Irradiated fuel not intended for further use in reactors in its present form.

CONTENTS

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[3] In the context of this guide the cladding consists of a tube, which surrounds the fuel and together with the end caps or plugs, provides a structural support.

1. INTRODUCTION

1.1 General

This guide provides guidelines for the fuel design in accordance with 'Code of Practice on Design for Safety in Pressurised Heavy Water Reactor Based Nuclear Power Plants', AERB/SC/D issued in 1989.

1.2 Objective

The objective of the guide is to provide the requirements of fuel design so as to conform to the specified limits for normal and off-normal reactor operating conditions, as well as for handling operations on fresh fuel and spent fuel at the reactor site.

1.3 Scope

This guide is applicable to pressurised heavy water reactor (PHWR) fuel elements and bundles consisting of natural and depleted uranium dioxide fuel. The typical conceptual features of the PHWR fuel and associated system are assumed to be standardised as per description given in section 2 of this guide and are taken as reference basis of this guide. The quantitative numbers given herein are based on current understanding and practical experience and are liable to change, depending on further understanding and experience. The applicability of the guide to thorium dioxide fuel elements/fuel bundles is covered in Annexure-I.

The guide covers the fuel design aspects for the following conditions:

- fresh fuel handling at PHWR, both by manual and fuel transfer system.
- normal reactor operation and anticipated operational occurrences (AOOs).
- spent fuel handling by fuel transfer system.
- storage.

The guide also covers the criteria for determining fuel cladding integrity during accident conditions.

2. DESCRIPTION OF FUEL AND ASSOCIATED SYSTEMS

The typical features of fuel, coolant channel and fuel handling system of current PHWR designs are briefly described in Annexure-II.

2.1 Operating Conditions

2.1.1 Operating Environment

The fuel bundle, during its residence in the reactor, will experience coolant pressure, temperature, velocity and be exposed to water chemistry, neutron and gamma environment.

Typical operating values of these parameters are given in Annexure-III.

2.1.2 Bundle Power Envelope

The bundle power envelope gives the maximum power generation from a bundle and its variation with burn-up, which is used for design analysis. The design limits are met during operation by ensuring that the bundle power is within this envelope.

Each fuel bundle during its life in the core has a different power and burn-up history due to:

- its initial location in the reactor.
- change in location due to refuelling.
- change in fuel composition (generation of Pu, depletion of U-235 and build up of fission products with burn-up).
- changes in local neutron flux due to movement of reactivity devices and/ or fuelling in neighbouring channels.

A representative bundle power envelope can be drawn over the bundle power histories with increasing burn-up of the bundles in high power channels during their in-core residence time. This will apply to a specific reactor core with a specific refuelling strategy (e.g., eight bundle shift). Reactor physics simulations of equilibrium reactor operation are carried out at the design stage to generate power-history envelopes, accounting for the above conditions [1].

A simplified simulation, called ‘time-averaged simulation’, considers the average conditions of fuel burn-up to obtain average values of bundle power for a particular position in the core. The conditions tend to be around average for a predominant part of life of the fuel. However, the instantaneous power of the bundle may differ from the average due to refuelling of the channel in question or to a neighbouring channel or to movement of reactivity devices.

Relatively rigorous simulations, called ‘snap-shot’ simulations, considering instantaneous burn-up of each fuel bundle, with realistic refuelling patterns are carried out. A number of these simulations are taken to determine the bundle power envelope representing instantaneous conditions of all the fuel bundles in all the simulations. The power of a bundle in this envelope for a particular burn-up is higher than that obtained from the simulation by time-average method. Typical time-average and snap-shot bundle power envelopes are shown in Fig.1.

For the purpose of fuel design, it could be conservatively assumed that fuel operates in a continuous manner along the bundle power envelope, as obtained from the snap-shot simulations. However, a more realistic approach of fuel design could be obtained by using a suitable combination of the envelopes provided by snap-shot and time-averaged simulations. Since the bundle power is not a directly measured parameter during operation, a few bundles may marginally exceed this envelope (of the order of 10 %) temporarily (between two simulations of actual core operating conditions of about 2 FPD) due to short-term control transient. The fuel design should consider this aspect as well. A combination of the envelopes, which can be used for the design, is suggested in [2].

3. DESIGN BASES

3.1 Design Bases for Normal Operation and Anticipated Operational Occurences

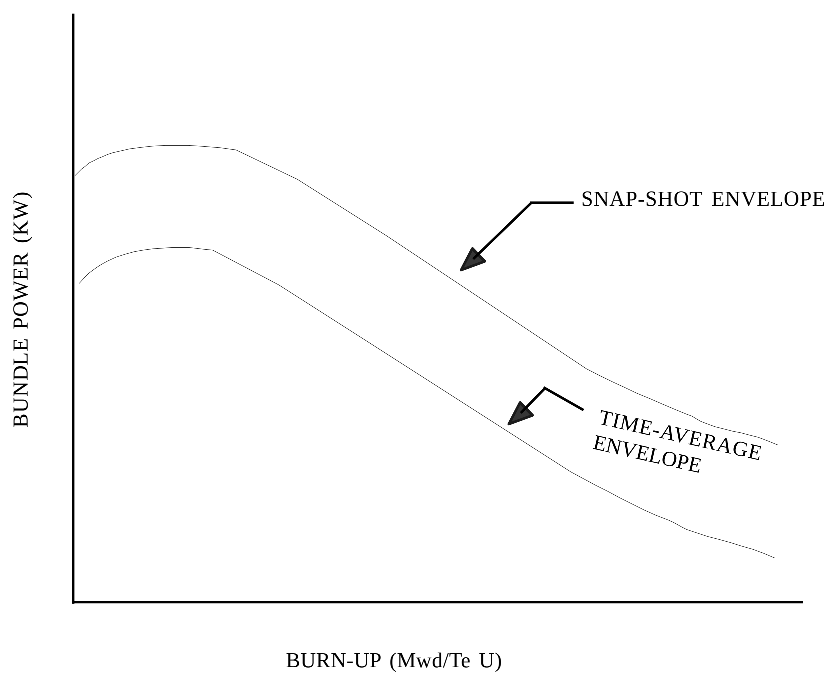

FIGURE 1 : TYPICAL BUNDLE POWER ENVELOPE

The design of fuel bundles shall be such that they will withstand their intended exposure in the reactor core without failure under the action of all processes of deterioration such as corrosion, irradiation embrittlement, hydriding, etc. This is met by specifying fuel design limits, as brought out in sections 4 and 5, and which shall not be exceeded in normal operation and conditions that may be transiently imposed during AOOs.

Fuel bundle is considered to be failed if there is breach of clad, which results in leakage of fission products to the coolant or loss of bundle integrity/distortion which affects fuel handling process.

The general design bases of reactor core and its associated cooling systems is that the fuel bundle integrity is maintained under normal operation and AOOs. However, depending upon the reactor-coolant purification system performance and technical specifications for reactor coolant activity, the reactor may be kept in operation with some leaky fuel bundles [3]. Technical specification limits on reactor coolant activity are based on radiological considerations.

3.2 Design Bases for Accident Conditions

In accident conditions the fuel shall remain in position and not suffer distortion to an extent that would render post-accident core cooling ineffective.

4. FUEL BUNDLE DESIGN BASES (IN-REACTOR)

The following functional and design requirements would form the bases for the fuel bundle design for normal operation and AOOs (in-reactor). The requirements for these two conditions would remain the same, except that the magnitude of safety margins would differ.

4.1 Functional Requirements

(i) The fuel bundle should be capable of producing the required power within the specified bundle power envelope without losing its integrity.

(ii) The fuel bundle should be compatible with the coolant channel assembly and fuel-handling system.

The functional requirements should be satisfied by ensuring the following design requirements.

4.2 Design Requirements

The following loads/effects should be determined conservatively from the bounding operating parameters, namely, coolant flow-rate, pressure, temperature, reactor power level, neutron flux, etc., with appropriate combinations.

4.2.1 Bundle Droop

Bundle droop can take place due to self-weight of fuel elements, differential temperature across the cross-section of the bundle and axial compressive load. This may reduce the gap between adjacent elements or the gap between outer fuel elements and pressure tube. The limiting value for the gap should be obtained by sub-channel analysis and the compliance ensured by inspection after fabrication and analysis. For 19-element fuel bundle in 220 MWe reactor, the minimum gap is 0.89 mm. [4].

4.2.2 Hydraulic Loads

(i) Vibrational loads: Flow induced vibrations cause fretting on spacers and fatigue at the joint between the element and the end plate.

(ii) Impact loads on the bundle during refuelling operation when the coolant flow pushes the bundle downstream.

(iii) Cross-flow vibration loads on the bundle during refuelling, while passing over the liner-tube holes provided for coolant entry.

The resonance vibrations of fuel element, fuel bundle and coolant channel assembly should be avoided while designing the primary heat transport (PHT) system. Currently, the fuel bundles are qualified for above mentioned hydraulic loads by type testing [5].

4.2.3 Fatigue Due to Power Cycles

Loads due to differential axial expansion of the elements cause fatigue on the end plates. The fatigue performance of end plate should satisfy the required power cycles. The end plates of 19-element fuel bundle were qualified by analysis for low-cycle fatigue performance.

4.2.4 Compressive Loads on the Bundle

This axial load is caused due to hydraulic drag from all the bundles/free components, like fuel locator in the channel during channel-closed condition and during PHT hydrostatic testing.

During refuelling operation, in addition to the above loads, fuelling machine ram loads come into play. For example, after inserting four fresh bundles in a coolant channel from upstream end, when shield plug of downstream end is being removed, the fuel string experiences compressive force due to hydraulic drag force and upstream fuelling machine ram force and friction force. The hydraulic drag force and friction force on the fuel string are also experienced by elements of the last bundle while being held against the side stops of the fuelling machine.

Currently, the fuel bundles are qualified for the design compressive loads by type testing [5].

4.2.5 Seismic Loads

The fuel bundle should withstand loads generated due to OBE/SSE without

- exceeding deformation limits which jeopardise cooling,
- fragmentation or severance of any bundle/component.

Since the fuel bundle is a free component in the coolant channel assembly, fuel bundle is qualified for OBE/SSE by impact tests.

4.2.6 Thermal Hydraulic Effects

- The pressure drop in the fuel bundles in a channel must be within the design provision of the PHT system. This is checked by out-of-pile type testing.
- The gap between the fuel elements and pressure tube shall remain acceptable, considering the fuel bundle and pressure tube wear, fretting and other dimensional changes due to irradiation (pressure-tube growth, creep and swelling) during operation. The sub-channel analysis and the element thermal analysis are carried out with the minimum gap and the element parameters with this gap are checked as per section 5.
- Sub-channel analysis should be carried out with maximum expected coolant channel diametrical creep (typically 3 %) and the resulting thermal

parameters checked as per section 5.

- The reduction in wall thickness of the pressure tube, because of wear and fretting due to the fuel bundle bearing pads, shall be less than the design allowances provided in the pressure tube wall thickness for this purpose. The compliance of this is checked by type testing [5].

4.2.7 Additional Requirements for Compatibility

(i) Compatibility of bundle ends with the sensors and stoppers of fuelling machine.

(ii) Pressure tube sag: Movement of fuel bundle string in a sagged pressure tube is checked by ensuring kink tube gauge test [6] during production.

(iii) Gap-crossing ability of the bundle between the tubes of fuel transfer system with permissible misalignment.

Compliance of (i) and (iii) above are taken care of during design and also by compatibility tests.

4.2.8 Fuel Identification and Traceability

A system of identification of each fuel bundle, fuel element and materials should be established to trace the manufacturing history and operational history.

The system of identification should also clearly indicate the type of fuel bundles, namely, natural uranium, depleted uranium or thorium.

In addition, a system for documentation on fuel bundle history should be established at the manufacturing site as well as at NPPs.

4.2.9 Weld Joint Requirements

The different weld joints in fuel-bundle assembly shall be designed taking into account the loads acting on the fuel element/bundle during its handling and movement in the reactor and outside the reactor by fuelling machine and fuel transfer system. The deterioration due to environment during its in-core residence should be taken into account.

5. FUEL ELEMENT DESIGN BASES FOR NORMAL OPERATION

The following functional and design requirements should form the bases for the fuel element design for normal operation and AOOs. The requirements for these two conditions would remain the same, except that, the magnitude of safety margins differ.

5.1 Functional Requirements

- The fuel element should be capable of producing required power within the specified fuel element power envelope (derived from specified bundle-power envelope) without affecting its integrity,
- The element should withstand power changes caused by refuelling, reactivity shim operation and refuelling of adjacent channels,
- Fuel element materials shall be compatible with each other and also with coolant (following post-defect degradation also). If coolant boiling is allowed, the same should be taken care while evaluating different requirements of fuel element like critical heat flux (CHF), corrosion, etc., given in section.5.2.

The above functional requirements are ensured by satisfying the following design requirements.

5.2 Design Requirements

This section covers the minimum list of phenomena or mechanisms which must be considered for fuel element design. These phenomena should consider the various operating parameters and environment, viz. pressure, temperature, neutron flux, element power envelope, power variations, power ramps, power transients, etc.

5.2.1 Material Compatibility

PHWR is very sensitive to neutron absorbers. This has to be taken care of during material selection. Fuel element materials shall be compatible with each other and also with coolant.

5.2.2 Clad Overheating

Cladding temperature should not significantly exceed coolant temperature, thus ensuring that element integrity is not lost through overheating or accelerated corrosion. To meet this requirement, the minimum critical heat flux ratio is evaluated to ensure that critical heat flux (CHF) is not reached.

5.2.3 Fuel Pellet Centre-line Temperature

The calculated maximum fuel pellet centre-line temperature with due allowance for irradiation, tolerances, uncertainties, etc., shall remain below the melting point.

5.2.4 Stress and Strain

Clad stress and strain must remain less than the value at which loss of fuel element integrity is predicted, with due allowance for temperature, irradiation and corrosion effects. Most of the fuel designs with zircaloy clad allow a maximum of one per cent total, uniform, circumferential plastic strain for normal operating conditions.

5.2.5 Flux Peaking

Due to absence of fertile material at the end cap and at the end plates, thermal neutron flux at these locations is more comparable to other places in the bundle. The effect of flux peaking on the design of element shall be analysed.

5.2.6 Fatigue

The variations in fuel element power and coolant pressure cause low cycle fatigue of fuel clad. The permissible number of strain fatigue cycles on the clad shall be significantly less than the design fatigue life time, which is based on appropriate data.

5.2.7 Pellet-Clad Interaction/Stress Corrosion Cracking (PCI/SCC)

Following a power increase from low power operation, the stresses on the sheath and also the amount of fission gases in the fuel sheath gap increase. Due to increased stress and simultaneous increase in fission gas iodine, the sheath is prone to SCC mechanism, which may result in fuel failure.

The fuel failure due to PCI/SCC mechanism during the power ramp shall be avoided by suitable design and operational procedures. Graphite coating is provided on the inner surface of the clad to reduce the failure due to power ramps. During the operation, the failure probability for power ramp due to refuelling or adjuster movements or gross power increase of reactor after a prolonged low power operation, should be evaluated by pre-simulations using available criterion. Presently, criterion developed by De Silva [7] is being used and is given in Annexure-IV.

5.2.8 Collapse Behaviour

The clad collapse over the pellet stack due to external coolant pressure should be such that there is no permanent ridge formation in the longitudinal direction and radial collapse in concentrated axial gap in the pellet stack. This is checked by suitable testing/analysis.

5.2.9 Fission Gas Generation and Internal Pressure

Fuel element internal pressure has a significant effect on element performance. The fill-gas and build-up of fission gases contribute to internal pressure and changes in thermal conductance. During the operation, the internal gas pressure should be less than the coolant pressure. If the internal gas pressure is greater than coolant pressure, the effect of differential pressure on heat transfer degradation, loss of element integrity or dimensional stability should be checked by thermal/stress analysis to meet the sheath temperature and strain limits (see 5.2.4 and 5.2.10).

5.2.10 Swelling and Irradiation Growth

The fuel swelling, diametral and axial irradiation growth should be evaluated and the maximum fuel bundle diameter with this should be less than the internal tube diameters of coolant tube, liner tube and different tubes of fuel handling/ transfer systems at those conditions.

5.2.11 Corrosion

Corrosion/oxidation shall be minimised to prevent loss of fuel element integrity. Clad temperature should be evaluated with allowance for both oxide layer and crud and should remain below the value corresponding to unacceptable accelerated oxidation (generally, zircaloy clad temperatures are maintained below 355° C for continuous operation at full power).

5.2.12 Hydriding

The design should ensure that embrittlement resulting from hydriding (primary hydriding) of zircaloy cladding does not cause fuel element failure. A limit should be placed on the moisture content and other hydrogenous material inside the fuel elements so that it does not lead to internal hydriding failure. Generally, total concentration of hydrogen from all the sources within the fuel element should not exceed 1 mg. per element.

Texture of the clad should be so specified that the hydride orientation remains predominantly in circumferential direction.

6. FUEL DESIGN ASPECTS FOR OUT-REACTOR CONDITIONS

6.1 General

The design requirements and aspects related to the out-reactor conditions such as shipment, storage and handling aspects of fresh as well as irradiated fuel bundles are covered in design safety guide, 'Design of Fuel Handling and Storage Systems for Pressurised Heavy Water Reactors' (AERB/SG/D-24) [8].

6.2 Design Requirements

Fuel bundle/element integrity and dimensional stability shall be demonstrated for the maximum loads and impacts which the fuel bundle/element is liable to undergo during shipment, storage and handling. These aspects should comply with the relevant clauses given in design safety guide on 'Design of Fuel Handling and Storage Systems for Pressurised Heavy Water Reactors' (AERB/SG/D-24) [8].

6.3 Cooling of Irradiated Fuel Bundles

Spent fuel bundles, after removal from the core, generate heat due to decay of fission products. Proper cooling arrangement shall be provided to remove the decay heat since the time fuel comes out of core into fuelling machine till it is discharged to spent fuel bay [8]. The fuel clad temperature is to be limited from the consideration of oxidation and expansion of the bundle.

In addition, during spent fuel transfer, the bundles move from heavy water environment to light water environment. In this transfer process, the bundles experience a short spell of air cooling which leads to increase in fuel sheath temperature. Current limit on fuel sheath temperature from expansion considerations of 600°C is being followed for transfer of spent fuel in the air environment.

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7. CRITERIA FOR FUEL INTEGRITY (ACCIDENT CONDITIONS)

7.1 Fuel Integrity

Under the postulated accident conditions, fuel shall remain in position and not suffer distortion to an extent that would render post-accident core cooling ineffective. This requirement is brought out in the design safety guide on 'Primary Heat Transport System for Pressurised Heavy Water Reactors' (AERB/SG/D-8).

7.2 Fuel Failure Criteria

In an accident transient, fuel failure may be assumed to have occurred if any of the following three criteria is not satisfied:

(i) Oxidation Limit

- The maximum oxygen concentration in the least affected half thickness of clad shall not exceed 0.7 per cent by weight.
- The above condition will also intrinsically require that for the fuel sheath to remain intact, the alpha phase penetration of the cladding shall be lower than the half thickness of the cladding.

The bases for the above criteria are given in Annexure-V. The methodology for the compliance of the criteria is given in Annexure-VI.

(ii) Burst Stress

Stress in the cladding shall not exceed burst stress (S).

$$S = a\ \exp(-bT)\ \exp[-((Ox-0.12)/0.095)^2] \quad \ldots\ldots\ldots\ldots\ldots\ldots\ldots (1)$$

where a and b are constants, determined experimentally.

T = Cladding temperature (deg. K)

Ox = % oxygen concentration

The methodology for the compliance of the criteria is given in Annexure-VI.

(iii) Fuel integrated power limit for reactivity initiated transients

The fuel pellet radial average enthalpy of the hottest fuel element shall not exceed 200 cal/g to ensure integrity of fuel [9].

The methodology for the compliance of the criteria is given in Annexure-VI.

7.3 The extent of fuel failures may be evaluated using the criteria given in section 7.2.

8. DESIGN ANALYSIS AND VALIDATION

The aim of this section is to outline methods for demonstrating that design bases and requirements are met. These methods include analysis, experimental checks and operating experience.

8.1 Analysis

The analysis should be based on widely accepted engineering methods based on physical/empirical models.

8.2 Experimental Checks

Confirmation of the results of analytical studies or model forecasts should be provided by comparing them with experimental (in-pile and out-of-pile) results obtained by measurements taken from fuel of similar or identical design by prototype tests or by post irradiation examination (PIE).

8.3 Type Testing

Representative fuel bundles of the proposed design and fabrication methods should be subjected to specified tests to conform to different element and bundle requirements. The different tests are:

(i) structural strength test

(ii) bundle wear test

(iii) pressure drop test

(iv) endurance test for vibration and fretting

(v) cross-flow test

(vi) impact test

(vii) bundle drop test

(viii) compatibility test with the fuelling machine

8.4 Operating Experience

Compliance with certain design requirements, i.e., those related to fretting wear, oxidation, crud build-up, structure integrity of the fuel bundle and irradiation-induced assembly deformation may be demonstrated by means of results obtained under representative operating conditions. In such cases, it may not be necessary to perform prototype tests or further design analysis.

8.5 Models

The models for evaluating fuel element/bundle behaviour should cover the following phenomena:

(a) For the element:

- temperature distribution inside the cladding
- heat transfer between pellet and clad
- temperature distribution in the fuel
- fuel swelling and densification
- fuel restructuring
- fission gas release
- irradiation-induced clad creep down and elongation
- cladding stress and strains
- pellet-clad interaction
- cladding waterside corrosion

(b) For the fuel bundle structure:

- vibrational response
- impact characteristics
- behaviour of bundle under various loads
- seismic loads

8.6 Design Limits and Margins

Design limits shall be set such that there is a sufficiently high probability of respecting acceptance limits and criteria. Limits which are currently followed are given below:

Sr. No.	Parameter	Normal operation limit	AOO limit
1	UO_2 centre-line temperature (°C)	2560° C	2840° C
2	Critical heat flux to normal heat flux ratio	1.3	1.1
3	Cladding strain limit (%)	1.0	1.5

The existence of a margin from acceptance limits and criteria shall be demonstrated with due allowance for the uncertainties associated with the design and operating parameters.

A better understanding of fuel behaviour can be obtained through PIE. This examination will help in identifying the causes of failure and also in updating the limits on the fuel.

ANNEXURE-I

APPLICABILITY OF THIS GUIDE TO THORIUM DIOXIDE FUEL

I.1 Thorium dioxide bundles are used in the initial fuel charge for flattening the flux in the core or for subsequent operation. These are discharged in the normal refuelling programme. During their stay in the core, they may end up seeing maximum power and burn-up similar to that of natural uranium fuel bundles. However, unlike a natural uranium bundle, the bundle power of thorium bundle increases with irradiation. A typical variation of thorium bundle power with irradiation for typical 220 MWe PHWR is given in Fig.A I-1.

I.2 Flux peaking factor (maximum flux/average flux over cross-section of the bundle) is expected to be higher in case of thorium bundle when compared to natural uranium bundle. Also, during a reactor shutdown, decay of protactinium leads to increase in uranium-233 concentration, resulting in higher power for a short period of few days. These issues have to be considered when fixing the operational limits for bundle power of thorium bundles.

I.3 All other issues described in the guide for natural uranium dioxide bundles also apply for ‘thorium dioxide’ bundles.

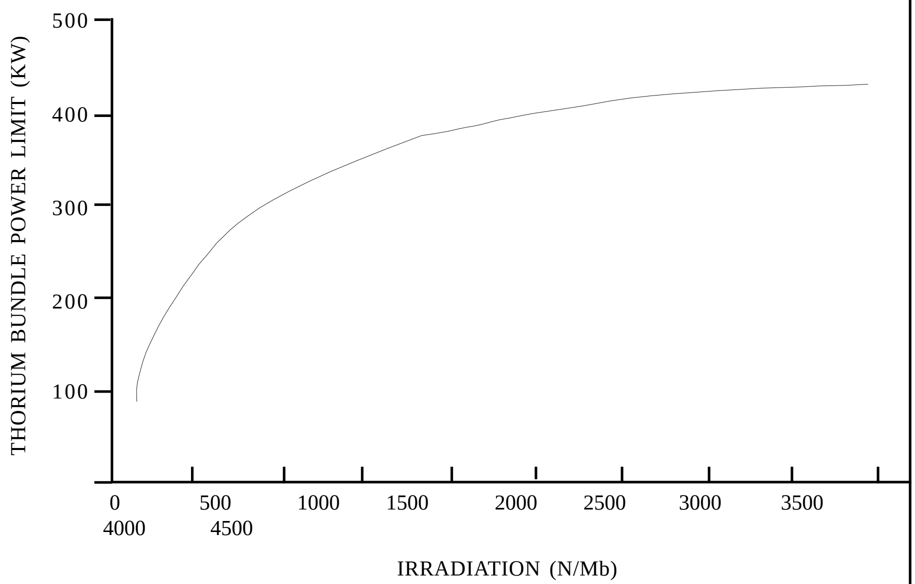

FIGURE A.I-1: THORIUM BUNDLE POWER LIMITS AS A FUNCTION OF IRRADIATION

ANNEXURE-II

DESCRIPTION OF FUEL AND ASSOCIATED SYSTEMS

The typical features of fuel, coolant channel and fuel handling system of current PHWR designs are briefly described below.

II.1 Fuel Bundles

A typical fuel bundle is about half a metre long and consists of a number of cylindrical fuel elements arranged in concentric rings. The elements are held together in a circular geometry by end plates welded at both ends of the fuel elements. Split spacers are provided to give necessary inter-element spacing. Bearing pads are provided to maintain necessary gap between the pressure tube and the outer fuel elements. Fig A II-1 shows typical bundle geometry for a 19-element (for 220 MWe PHWR) configuration.

A fuel element consists of sintered cylindrical UO_2 pellets contained in a thin zircaloy cladding, which is collapsible under operating coolant pressure. The element is filled with helium gas at atmospheric pressure and the sheath is sealed at both ends by welded end plugs. The inside surface of the cladding is coated with graphite. Each UO_2 pellet is spherically dished at one or both the ends.

II.2 Coolant Channel Assemblies

A typical PHWR consists of a large number of coolant channel assemblies (for example, a 220 MWe PHWR has 306 channels). Each channel assembly is loaded with fuel bundles (for example, 12 bundles in 220 MWe PHWR coolant channel, of which 10.1 bundles are in the core). Typical coolant channel is shown in Fig.A.II-2.

Each coolant channel assembly consists of a horizontal Zr-2.5% Nb pressure tube attached by an expanded joint to stainless steel end-fitting at both ends. A stainless steel liner tube is also rolled into the end-fitting to provide extension to the pressure tube into the end-fitting to support fuel bundles and fuelling machine rams during the fuelling operation. Heavy water coolant enters the end-fitting through feeders and flows through a series of radial holes in the liner tube into the coolant channel.

The fuel bundles are placed in this channel. The channel has shielding plugs at both the ends and sealing plugs to close the channels. In 500 MWe PHWR, fuel locators are placed between shield plug and fuel string on both the sides. Fuel locator is a free component inside the channel, which helps to locate the fuel bundles in the active core. During operation, the coolant flow keeps the fuel string butted against downstream shielding plug/fuel locator.

II.3 Fuel Handling System

PHWRs employ onpower refuelling. Typically, in a fuelling operation, fresh fuel bundles are loaded into the channel selected for refuelling and a corresponding number of spent fuel bundles are discharged. The fuelling is carried out in the direction of coolant flow. The fuel transfer/handling system consists of fresh fuel transfer system, spent-fuel transfer system, fuelling machines (2 in number) and spent fuel storage bay.

Fresh fuel bundles are loaded manually into the new fuel magazine which transfers the bundles in the input conveyor of the fuel transfer system.

The fresh fuel transfer system serves to transport the fresh fuel from the new fuel magazine to the respective fuelling machine.

All operations of fuel handling by fuel transfer system and the fuelling machines are done considering two bundle lengths as a single unit.

Fuel bundles are received in the magazines of the fuelling machine head which operates in heavy water environment. A fuelling machine is attached to each end of a coolant channel. The machines work in conjunction with each other during a refuelling operation. At the upstream end, the ram of fuelling machine pushes the fresh fuel bundles into the channel and the downstream machine receives the discharged fuel bundles.

Spent fuel transfer system accepts spent fuel from either of the fuelling machines and transports it to the spent fuel storage bay. During spent fuel transfer, the bundles move from heavy water environment to light water environment. In this process, the bundles experience upto four minutes of air cooling environment.

In the transfer of fuel through the fuel transfer system from the new fuel loading station to the spent fuel storage bay, the bundle crosses from one tube to another at a number of tube junctions. In this process, it has to overcome the misalignment and gaps between the tubes. In 220 MWe PHWR with 19-element fuel bundles, minimum gap maintained is 0.89 mm.[4].

II.4 Storage of Spent Fuel

The irradiated fuel bundles are stored under water in spent fuel storage bay until final disposal. The water in the bay provides shielding and cooling to the bundles. The bay has a water purification system to maintain water chemistry and a cooling system to remove the decay heat from the fuel bundles. The maximum period of storage in the fuel storage bay is of the order of ten years based on bay capacity.

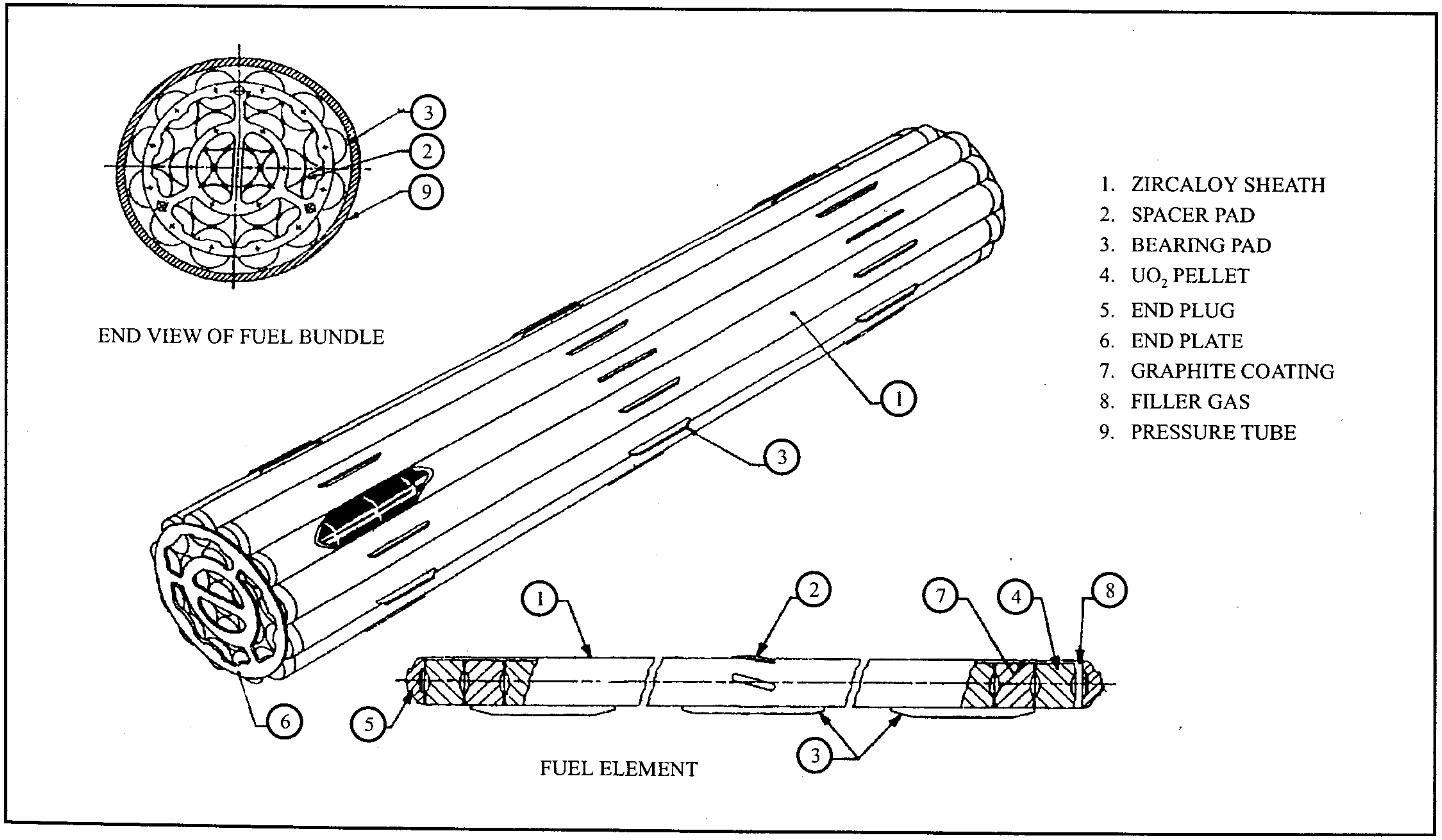

FIGURE A II-1 : 19 ELEMENT FUEL BUNDLE FOR 220 MWe REACTORS

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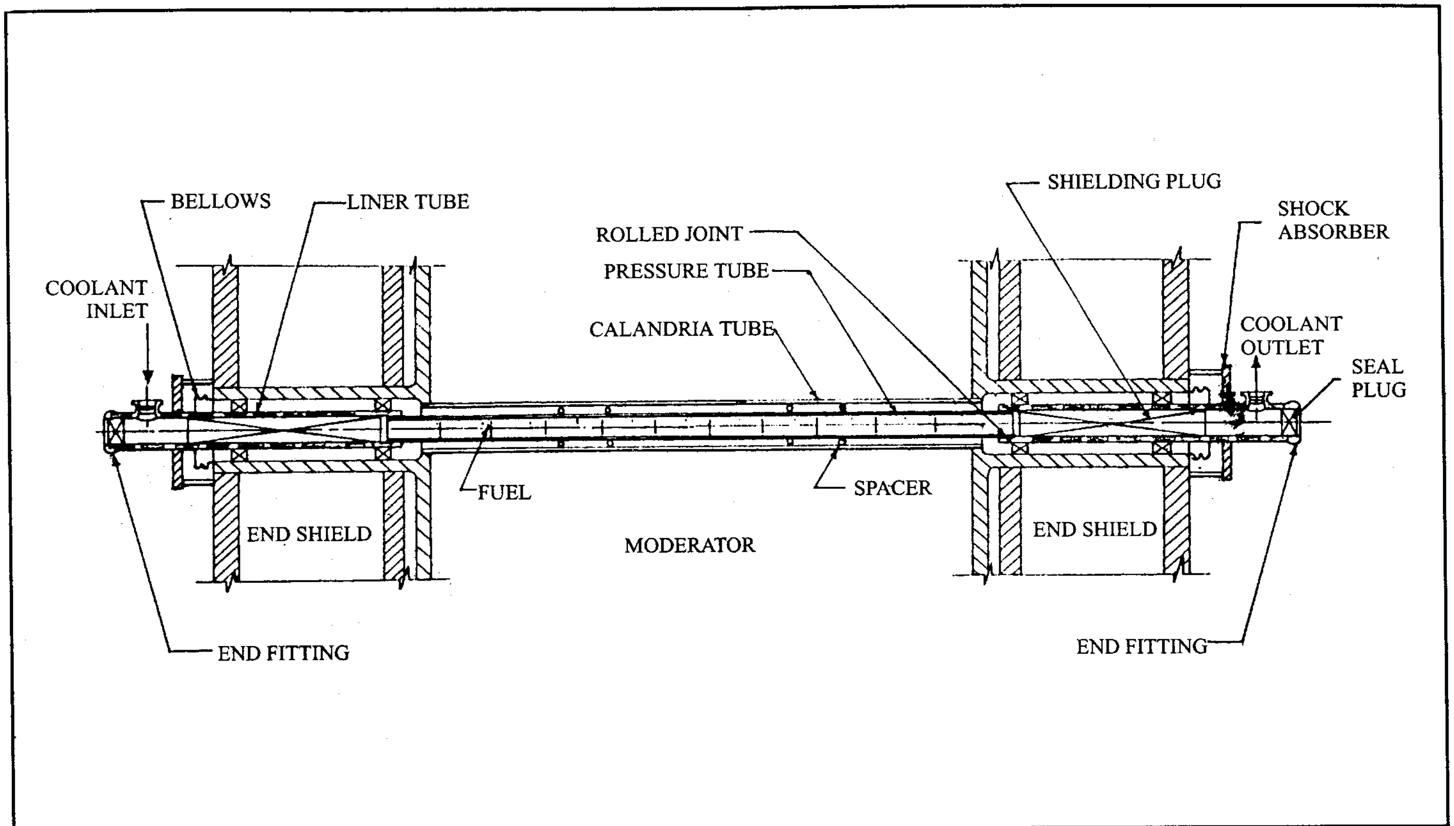

FIGURE A II-2 : TYPICAL COOLANT CHANNEL ASSEMBLY

ANNEXURE-III

TYPICAL OPERATING CONDITIONS FOR MAXIMUM RATED CHANNEL

Sr. No.	Parameter	220 MWe	500 MWe
1.	Coolant pressure (kg/cm^2) at reactor inlet header	99	115
2.	Coolant pressure (kg/cm^2) at reactor outlet header	87	101
3.	Coolant temperature (°C) at reactor inlet header	249	260
4.	Coolant temperature (°C) at reactor outlet header	293	304
5.	Maximum coolant velocity (m/sec) at reactor inlet header	9.35	10.0
6.	pH of coolant	10 - 10.5	10 - 10.5
7.	Coolant boiling	No	No

ANNEXURE-IV

POWER RAMP CRITERIA

D'Silva's equation, as taken from [7], which calculates the probability of fuel element failure is given below.

$$P_f = 1/(1+e^A)$$

$$A = 338.9 - 16.9 \, Log_e(B) - 63.6 \, Log_e(C) + 4.4 \, D^{-1}$$

where

P_f : Probability of fuel element failure

B : Burn-up in MWh/kg of uranium
i.e. (MWd/TeU x 1/41.667)

C : Final ramped power kW/m
(kW/bundle x 0.11968 for 19-element fuel bundle)

D : Ratio of final to initial power

ANNEXURE-V

OXYGEN EMBRITTLEMENT CRITERIA

The oxigen embrittlement criteria for zircaloy cladding of PHWR fuel elements during LOCA is based on the content and distribution of oxygen in the cladding. The criterion is based on the fact that changes in the mechanical properties of the cladding due to oxidation at high temperature (j1000°C) are governed by the relative magnitude of various layers formed in the cladding (namely zirconium oxide layer, oxygen stabilised alpha zirconium layer and beta zirconium layer) and the amount and distribution of oxygen in the cladding. The oxide and oxygen stabilised alpha phases are inherently brittle, hence load-bearing capability of the cladding largely depends on the thickness and oxygen content of beta region. As oxygen diffuses from high oxygen phases into beta region, there is an oxygen concentration profile in the beta region. The oxygen concentration in the beta layer is a function of temperature and time.

The dependence of tensile properties of zircaloy-4 was measured by Sawatzky as a function of oxygen concentration, cooling rate, maximum test temperature and oxygen distribution. Based on the results of this study, a criterion for fracture due to oxygen embrittlement was proposed and it was suggested that the oxygen content should not exceed 0.7 w% over at least half the cladding thickness, since it had been shown that transformed beta phase containing more than 0.7 w% oxygen was brittle enough to fracture on cooling [11].

To use the oxygen embrittlement criterion based on oxygen concentration, we need to know the oxygen distribution in the cladding for a given temperature and time combination. For this purpose it is necessary to have a mathematical model that can be used to analyse the oxidation behaviour and oxygen distribution in the cladding during LOCA condition. A computer model OXYCON has been developed [12]. This model provides as output oxide layer thickness, alpha layer thickness, beta layer thickness, extent of equivalent cladding reacted and the oxygen concentration profile across the cladding thickness. The build up of oxygen in the PHWR fuel cladding and its distribution across the cladding thickness was analysed in the temperature range 1000 to 1600°C, using the model OXYCON. A comparison of 17 per cent oxidation criteria with 0.7 w% criteria based on OXYCON calculation using nominal values of input parameters showed that at about 1220°C the 0.7 w% criterion is equivalent to 17 per cent equivalent cladding reacted.

OXYCON results also showed that at temperatures below 1220°C, the oxygen stabilised alpha phase extends beyond the half thickness of the cladding when 0.7 w% average oxygen concentration was approached in the half thickness of the cladding. As oxygen-stabilised alpha zirconium is brittle, this situation has to be avoided. Also at higher temperatures (>1220°C) some portion of cladding in the least affected half thickness is likely to have more than 0.7 w% oxygen when the average concentration in the half thickness approaches a value of 0.7 w%.

In order to avoid such situations and to make the criteria more conservative, it was decided to restrict the oxygen concentration at the half thickness plane of cladding to be 0.7 w% maximum and to keep the alpha penetration lower than the half thickness of the cladding. These two conditions ensure that in the least affected half thickness of cladding, the oxygen concentration will remain less than 0.7 w% at all locations.

Based on these considerations the following interim criteria have been recommended for PHWR fuel elements during LOCA conditions:

- The maximum oxygen concentration in the least affected half thickness of the cladding shall not exceed 0.7 w%.
- The above condition will also intrinsically require that for the fuel sheath to remain intact, the alpha phase penetration in the beta region of the cladding shall be lower than the half thickness of the cladding.

Verification of the OXYCON Code and Output

For estimation of oxygen uptake, zirconium oxide thickness and alpha growth rate constants available in literature, which have been generated based on experiments [14 to 19] have been used. The oxygen diffusion coefficient data is taken from experimental data and the variation in data (-40% + 67%) as given in [20] is accounted in the programme.

From the output of the programme, the time to fail at different temperatures has been checked in parallel with calculations carried out independently at NPCIL [21] and with literature data [11] . The results are comparable.

Conservatism in Calculations

A conservative estimate of time of embrittlement as a function of temperature has been made for the 0.38 mm thick PHWR cladding for isothermal oxidation on the surface of the cladding. The following points provide for conservatism in calculations:

(a) The oxygen concentration in the least affected half thickness plane of cladding to be limited to 0.7 wt% maximum, instead of average concentration of over half thickness.

(b) The literature indicates that the diffusion co-efficient varies from the nominal values between -40% to +67% [20]. The diffusion coefficient has been multiplied by 1.67 to take care of the statistical variation in the data. The oxygen uptake rate constant used is 1.1 times the nominal value.

(c) The correlations which give the most conservative value of time for embrittlement were identified [12] and taken in the embrittlement calculation:

For oxide thickness and a thickness estimation

(i) urbanic correlation is used for sheath temperatures between 1000-1050°C;

(ii) WP1 correlation is used for sheath temperatures > 1050°C.

Thus, conservative values of rate constants for oxygen uptake, oxide growth, alpha growth, oxygen diffusion coefficient and equilibrium phase boundary concentrations are used to evaluate the oxygen concentration and alpha penetration in the cladding.

The plot of time required for embrittlement as a function of temperature, based on the above mentioned criteria, after applying the above said conservative guidelines is shown in Fig A.V-1.

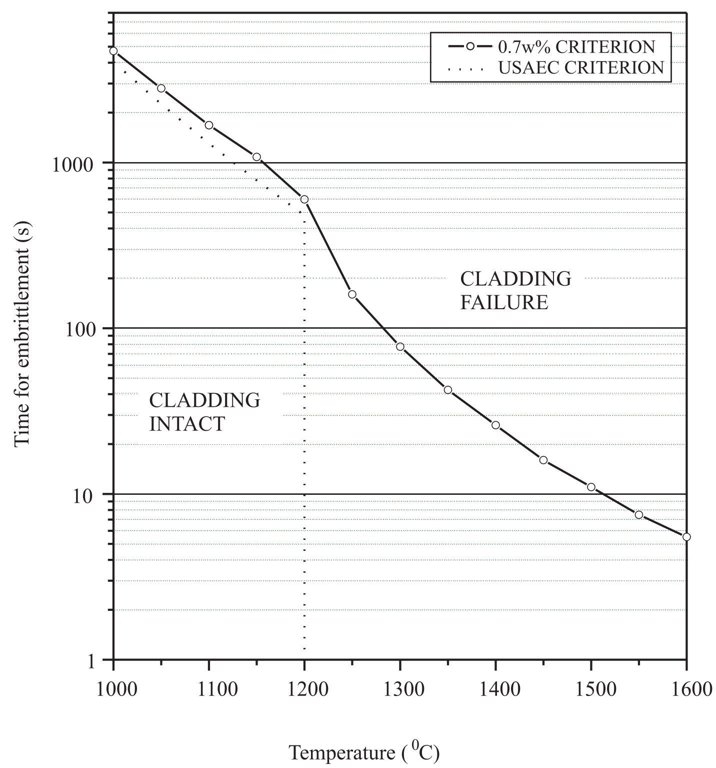

FIGURE A.V.-1 : CONSERVATIVE TIME OF EMBRITTLEMENT FOR 0.38 mm THICK PHWR CLADDING DURING OXIDATION ON ONE SURFACE AS A FUNCTION OF TEMPERATURE.

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ANNEXURE-VI

USE OF FUEL FAILURE CRITERIA FOR ACCIDENT CONDITIONS

The method used presently to estimate fuel failures using the fuel failure criteria is explained below. In an accident transient fuel sheath will be subjected to high temperature and high pressure difference across the sheath (as coolant pressure goes down, in case of LOCA). Under such conditions the methodology for estimating fuel failure, as per criteria given in section 7.0, is elaborated below.

VI.1 Burst Stress

Assessment of bursting of fuel sheath involves the calculation of burst stress and true stresses seen in the sheath [13]. The procedure involved in calculation of these two parameters is discussed below.

VI.1.1 Calculation Procedure

Burst stress is a material property and varies with temperature and oxidation of zircaloy sheath. Burst stress is calculated using the following correlation:

$$S = a\cdot\exp(-bT)\,\exp[-((Ox-0.12)/0.095)^2\,] \qquad \ldots\ldots\ldots\ldots\ldots\ldots (1)$$

Where a and b are constants determined experimentally. Typical values are available in literature [13].

Calculation of true stress needs deformation (strain) process to be defined. The present model assumes that deformation process of internally pressurised zircaloy cladding can be calculated from the steady-state (secondary) creep equation of the material.

The steady-state creep rate of a material at constant temperature and constant stress can be represented by a power law - Arrhenius equation of the form

$$\dot{\varepsilon} = \frac{d\varepsilon}{dt} = A\sigma^n \exp\left(\frac{-Q}{RT}\right) \qquad \ldots\ldots\ldots\ldots\ldots\ldots (2)$$

Where σ is applied stress and T the absolute temperature.

For symmetrical deformation, the tangential (circumferential) stress σ for a thin-walled tube under a differential pressure P is given by

$$\sigma = \frac{Pr}{S} \qquad \ldots\ldots\ldots\ldots\ldots\ldots (3)$$

Where r is the instantaneous mean tube radius and S is the instantaneous tube wall thickness, subscript o in the equations below refers to initial condition. The instantaneous tangential strain ε is defined by

$$r = r_o (1 + \varepsilon) \qquad \ldots\ldots\ldots\ldots\ldots\ldots (4)$$

It is assumed that the cross-sectional area of tube wall is conserved during deformation

$$r.S = r_o.S_o \qquad \ldots\ldots\ldots\ldots\ldots\ldots (5)$$

From equations 2 to 4, one obtains the stress-strain correlation

$$\sigma = \frac{P}{P_o}\,\sigma_o\,(1+\varepsilon)^2 \qquad \ldots\ldots\ldots\ldots\ldots\ldots (6)$$

Substituting equation (6) in equation (2) and by rearranging terms, one obtains differential equation for diametereal strain

$$\frac{d\varepsilon}{(1+\varepsilon)^{2n}} = A\,\sigma_o^{\,n}\left(\frac{P}{P_o}\right)^n \exp\left(\frac{-Q}{RT}\right) dt \qquad \ldots\ldots\ldots\ldots\ldots\ldots (7)$$

Solving the equations 6 and 7, one can obtain true stress at different times of transient and compare with burst stress to assess the bursting of sheath.

VI.1.2 Application for PHWR Fuel Model

When the above sheath burst model is applied to PHWR fuel, it is required to calculate differential pressure across the sheath. Differential pressure is the difference between the internal fission gas pressure and outside coolant pressure. In order to calculate fission gas pressure, initial inventory of fission gas is assumed to be equal to the fission gas inventory existing in the gap under steady state for a particular burn up. In addition, if for any particular case, the fuel temperature at a radius under accident condition had been above that of normal condition then the higher temperature is used for calculating fission gas release fraction. The variation of fission gas pressure with respect to temperature and deformation of sheath is calculated, assuming ideal gas equation.

Volume occupied by fission gas is calculated by assuming symmetrical deformation of sheath. However, geometry of PHWR fuel configuration restricts symmetrical deformation of sheath. By observing the fuel bundle configuration of 220 MWe PHWR, it is seen that the symmetrical deformation process can exist up to 5 per cent strain of sheath. Further deformation can take place by local strain. Hence, conservatively above 5 per cent sheath strain is not accounted for in the calculation of volume occupied by fission gas. However, to be on the conservative side, calculation of true stress and secondary creep proceeds without any limitation on strain.

The application of burst criteria is given in section 4.

VI.2 Oxygen Embrittlement

The oxygen embrittlement criteria are explained in Annexure-V. To satisfy the embrittlement criteria, the time to reach 0.7 per cent oxygen concentration over half thickness of sheath at different temperatures are estimated and the corresponding figure is given in Annexure V (Fig.A.V-1).

VI.3 Fuel Enthalpy Limit

The total energy in fuel element including radial average enthalpy shall be less than 200 cal/g. The adiabatic energy deposited in the maximum rated channel is calculated for either 15 seconds or the time period till reactor power is reduced to below 15% (and remains below this power subsequently). Whichever time is greater between the two cases is used in the calculation.

VI.4 Application

Adoption of sheath burst model in the accident analysis code could be a complicated task. In absence of such a detailed model, a simplified model which could be derived from above criteria or which can be shown to be conservative w.r.t., the reference mode could be used [13]. One such model derived from the above model is discussed here.

The present simplified model is derived by applying the burst model and oxygen embrittlement criteria on a single fuel element of a 19-element fuel bundle. This element has seen 10000 MWd/TeU burn up with 462 kW bundle power envelope.

For this element, various combination of coolant pressure and sheath temperatures are used to predict the time at which the fuel fails by bursting of sheath or oxygen embrittlement. Time at which fuel fails following the initiation of accident vs coolant pressure for various sheath temperatures are represented in Fig.A.VI-1.

Such data of time for fuel failure, as function of coolant pressure (P) and sheath temperature (T) for a particular case of burn up which envelopes all the other cases can be generated. However, in case of accident transient both sheath temperature and coolant pressure vary with time. In such cases fractional damage caused by each combination of coolant pressure and sheath temperature at various time steps are summed up to assess fuel failure [21]. The cumulative damage effect at time t can be represented mathematically as

$$\int_0^t \frac{dt}{t\,(P_1 T)}$$

where t is the time at which the fuel fails at the coolant pressure P_1 and sheath temperature T.

Thus one could say that the fuel does not fail before time t, if

$$\int_0^t \frac{dt}{t\,(P,T)} < 1$$

Similar approach will be applied for estimating fuel failures due to oxygen embrittlement criteria.

Nomenclature

A	Structure parameter
K	Absolute gas constant
N	Stress exponent
Ox	Oxygen concentration
P	Pressure difference across sheath
Pf	Fission gas pressure
Q	Activation energy
R	Mean radius of tube
S	Tube wall thickness
T	Absolute temperature
t	time
V	Volume occupied by fission gas
□	Strain
□	Stress
a	Burst stress parameter
b	Burst stress parameter
n	Stress exponent

Subscript

o	Initial condition
f	final condition
B	Burst

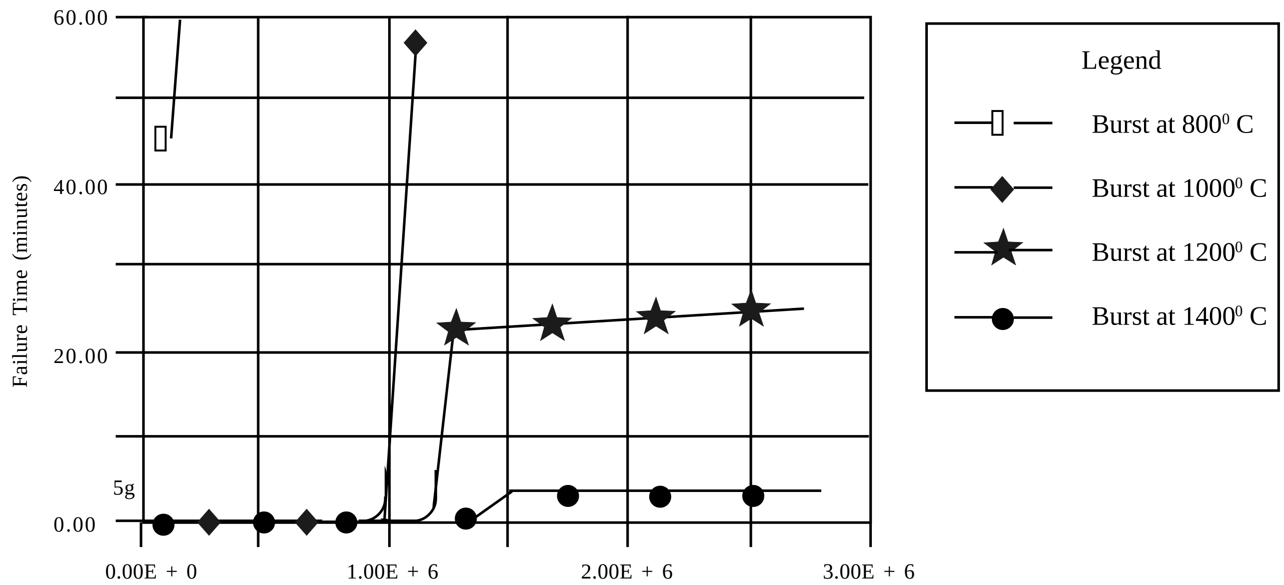

FIGURE A.VI-1 : FAILURE TIME FOR FUEL OF 10,000 MWD/Te (U)

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LIST OF PARTICIPANTS

WORKING GROUP

Dates of meeting:	November 27, 1995	June 21, 1999
	December 12, 1995	April 29, 2000
	August 5, 1996	July 27, 2000
	August 6, 1996	May 8, 2001
	December 10,1996	June 12, 2001
	June 3, 1997	June 25, 2001

Members of working group:

Shri S.A. Bharadwaj (Chairman)	:	NPCIL
Shri K. Anantharaman	:	BARC
Shri S.M. Choudhary	:	BARC
Shri A.N. Nakra	:	BARC
Shri D.N. Sah	:	BARC
Shri S. Govindarajan	:	IGCAR
Shri A.K. Saxena	:	NPCIL
Shri A. Ramakrishna (Member-Secretary)	:	AERB
Shri P.N. Prasad	:	NPCIL
Shri M. Ravi	:	NPCIL
Shri V. Sivakumar	:	BARC

ADVISORY COMMITTEE FOR CODES,GUIDES AND MANUALS FOR SAFETY IN DESIGN OF NUCLEAR POWER PLANTS (ACCGD)

Dates of meeting:	June 26 & 27, 1997 October 27, 1997

Members of ACCGD:

Shri S.B. Bhoje (Chairman)	:	IGCAR
Prof.C. Amarnath	:	IIT, Bombay
Shri V.K. Mehra	:	BARC
Shri Umesh Chandra	:	BARC
Shri A.K. Asrani	:	AERB
Shri S. Sankar	:	BARC
Shri C.N. Bapat	:	NPCIL
Dr S.K. Gupta	:	BARC
Shri S.A. Bharadwaj	:	NPCIL
Dr R.I.K. Murthy	:	BARC
Shri R.S. Singh (Member-Secretary)	:	AERB
Shri S.A. Khan (Permanent Invitee)	:	AERB

ADVISORY COMMITTEE ON NUCLEAR SAFETY (ACNS)

Date of meeting	:	August 2, 2002

Members of ACNS:

Shri Ch. Surendar (Chairman)	:	NPCIL (Former)
Shri S.K. Sharma	:	BARC
Dr V. Venkat Raj	:	BARC
Shri R.K. Sinha	:	BARC
Shri S.S. Bajaj	:	NPCIL
Shri S.P. Singh	:	AERB (Former)
Shri Ramesh D. Marathe	:	L&T, Mumbai
Shri S.K. Agarwal	:	AERB
Shri K. Srivasista (Member-Secretary)	:	AERB

Date of meeting	:	April 29, 2000

Members of ACNS:

Shri S.K. Mehta (Chairman)	:	BARC (Former)
Shri S.M.C. Pillai	:	Nagarjune Power Corporation, Hyderabad
Prof. U.N. Gaitonde	:	IIT, Bombay
Shri S.K. Goyal	:	BHEL
Shri Ch. Surendar	:	NPCIL (Former)
Dr. U.C. Mishra	:	BARC (Former)
Shri S.K. Sharma	:	BARC
Dr. V. Venkatraj	:	BARC
Shri G.K. De	:	AERB (Former)
Shri S.P. Singh	:	AERB (Former)
Shri K. Srivasista (Member-Secretary)	:	AERB

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AERB/SG/D-26	Deterministic Safety Analysis of Nuclear Power Plants
AERB/SM/D-1	Decay Heat Load Calculations