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Derivation of fuel bundle power limits using fuel element modelling code FUDA

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A B S T R A C T

The fuel designs, used in India are 19-element circular geometry and 37-element circular geometry for the 220 MWe PHWR and 540 MWe PHWR, respectively. The fuel bundle design has been evaluated by theoretical analysis, development tests and type tests. The theoretical analysis comprises of fuel bundle sub-channel analysis, fuel element thermo-mechanical analysis and fuel failure analysis during accidents etc. The fuel element thermo-mechanical analysis is carried out using computer code FUDA (*F*uel *D*esign *A*nalysis) code. The fuel bundle power limit is derived based on fuel central line temperature, sheath strain and fission gas release parameters. The paper brings out the details of the FUDA code, the typical analysis carried out to estimate these parameters by FUDA code for deriving bundle power limits during operation for 540 MWe PHWR for 37-element fuel bundle.

1. Introduction

Presently, 14 PHWR are operation commercial use in India. Twelve of them produce 220 MWe each with fuel bundles of 19-element type. The other two reactors, of 540 MWe capacity, use 37-element fuel bundle type. The fuel elements of these bundles are arranged in a circular array.

During the 38 years of reactor operation, about 330 000 fuel bundles of 19-element type and 11 000 of 37-element type were irradiated. Different design modifications were tested in the natural U fuel bundles, namely wire wrap and split spacer. The pellet shape has been updated from single-dish type to single-dish chamfer and to current double-dish chamfer type. Both Zircaloy-2 and Zircaloy-4 have been used as structural material of the fuel bundles. Over the years, the Fuel material and Zircaloy components with varied chemical composition and mechanical properties were irradiated in the reactors. This provides a wide experience on the irradiation behavior of materials. About 500 fuel bundles of 22-element type were also irradiated in the Narora Atomic Power Station. Apart from the natural uranium bundles, depleted uranium dioxide, thorium dioxide and MOX bundles have also been designed and irradiated.

The fuel bundle design has been evaluated by theoretical analysis and experiments. The former comprise sub-channel studies and thermo-mechanical analysis in operating conditions and fuel failure simulations in accident conditions. The thermo-mechanical behavior of a fuel element is simulated with the computer code FUDA (*F*uel *D*esign *A*nalysis). The code also allows simulating fuel element transients during actual operation. The code outputs are compared with post-irradiation examination results.

In the present paper the results obtained with the FUDA code in typical analysis carried out for deriving bundle power limits during operation are shown.

2. Fuel bundle and related system description

The 37 element fuel bundle shown in Fig. 1 has fuel rods or elements arranged in a circular array of 1, 6, 12 and 18 elements to form the assembly held together by end plates. Each fuel element consists of natural uranium dioxide sintered pellets loaded inside graphite coated Zr tubes. The tubes at both ends are closed by resistance welded end plugs to avoid release of radioactive fission products to the coolant. The assembled bundle has an overall length of 495 mm and a diameter of 102 mm. Spacers and bearing pads maintain the inter element and element-to-coolant channel gap and are spot welded to these elements before the bundle assembly. In the 540 MWe PHWR such 13 fuel bundles are located in each of the 392 channels in the calandria which forms the reactor core.

The on-power bi-directional fuelling is carried out for replenishment of fuel with the help of two fuelling machines located at each end of the coolant channel. In a sequence of fuelling operation, a pair of fresh fuel bundles is loaded on the upstream side of the channel and simultaneously a pair of spent fuel bundles are discharged on the downstream side. These are then sent to the spent fuel pool through a fuel transport system.

3. FUDA code

The FUDA (Fuel Design Analysis) code MOD0 version has been developed for PHWR fuel element analysis [1] The code is updated time to time and the present version is FUDA MOD2 [1]. The code

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has to take into account the inter-dependence of different parameters like fuel pellet temperatures, pellet expansions, fuel-sheath gap heat transfer, sheath strain and stresses, fission gas release and gas pressure, fuel densification, etc. Due to this complexity, the code is based on a mix of empirical, physical and semi-empirical relationships. The finite difference method is used in the calculations to solve differential equation.

The neutron flux profile and power profile across the fuel radius and its variation with burnup was estimated using the computer code CLUB [2]. A cubical equation for local power across the fuel pellet as a function of fuel radius and burnup is used in the FUDA program. This is used to estimate neutron flux and power across the fuel radius at any burnup. The model for the thermal conductivity of UO_2, as given in MATPRO [3] that includes the dependence on temperature and porosity was incorporated to the code. The model can also calculate the thermal conductivity of ThO_2 and MOX fuels. The heat transfer coefficient of the fuel-sheath gap, i.e. across the gas filled fuel-sheath gap and through the solid–solid contact spots is calculated using a modified version of the Ross and Stoute equation [4]. The radial temperature distribution across the pellet and sheath is calculated for the given input data: linear heat rating, coolant temperature and the film heat transfer coefficient between fuel element and coolant. The fission gas (Xenon and Krypton) generation in fuel, which is a function of burnup, is estimated using the ORIGEN Code [5]. The fission gas release model included in FUDA is based on semi-empirical relations. It considers the fission gas diffusion to grain boundaries, grain boundary movement due to grain growth, intergranular bubble formation and interlinkage. These models are based on the work done by Notley et al. [6] that assumes constant power. The power ramps are included as a succession of constant power steps. The fission gas pressure is calculated from the mass of fission gas released and the size and temperature of the free volume within the rod. The net circumferential sheath strain and the hoop stress due to the combination of fuel thermal expansion, swelling and densification are calculated as fuel burnup progresses. The stress relaxation in the time intervals at constant power operation is calculated using semi-empirical creep formula. This gives the sheath strain (both plastic and elastic components) at the end of constant power period. Fuel-sheath interfacial pressure is then calculated based on gas pressure and sheath strains. This shows whether the expanded fuel is touching the sheath or a gas gap exists between them.

The fuel element materials and geometrical parameters, the reactor neutronics, the thermal hydraulic parameters and the element linear heat rating in different burnup zones are required as input data. The results generated by the program are the radial temperature distribution in the fuel and sheath, the fuel-sheath heat transfer coefficient, the fission gas generated and released, the gas pressure, the fuel-sheath interfacial pressure and the sheath stress and strain for different burnup ranges.

4. Cases of study

The FUDA Computer Code is being used:

1. To show compatibility of the design with the requirements (the analysis supplements the type tests).
2. For analysing the data from the operating commercial reactors and
3. To study the effect of fabrication variables on fuel performance.

A case study of the maximum bundle power that can be produced without exceeding the limits for a 37-element fuel bundle in TAPP 3&4 reactor, carried out using FUDA MOD2 is described here, which will show the influence of bundle power on design and fuel performance. The outer element of the fuel bundle which produces maximum power is analyzed.

4.1. Derivation of bundle power limits

The limits on the fuel bundle thermal power are established to assure the integrity of the fuel during its operation. The fuel bundle power depends on the content of the fissile isotope U^{235}, the build up of fission products in a fuel bundle and the associated neutron flux at any given time. A typical bundle power history during its stay in reactor is shown in Fig. 2.

Due to the necessary fuelling routine and the consequent bundles movements, the actual neutron flux distribution in a PHWR during its equilibrium operation, is having some local peaks and valleys around a nominal power envelope. These local changes in flux shape gives rise to corresponding deviations from the nominal fuel bundle power. Attempts have been made to theoretically assess in advance the extent of deviations from the nominal (time averaged) profile that would be encountered in real life situations with actual refuelling schemes. There are other factors like fuelling machine unavailability and consequent withdrawal of reactivity devices, and unscheduled refuelling to discharge failed fuel which force a deviation from the nominal profile. To account for these variations a 10–15% margin over the time averaged bundle power (outer element LHR) is needed. The resultant bundle power envelope is an envelope of all the possible bundle power histories in the reactor.

The operating limit ensures bundle safety during normal operation with margins and avoids touching the safety limits in the event of operation at/near trip setting. This bundle operating limit has been defined as a bundle power limit which, when

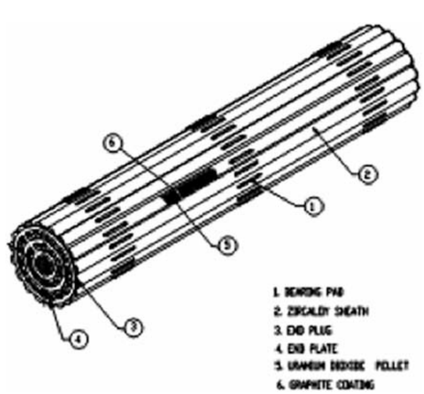

Fig. 1. Thirty-seven element fuel bundle.

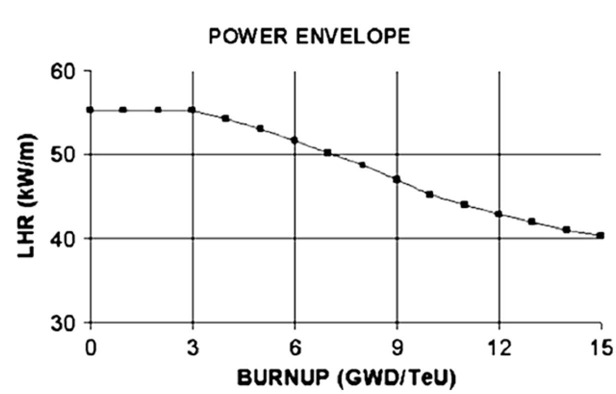

Fig. 2. A typical bundle power history.

Table 1
Thermo-mechanical analysis of fuel element

Criterion	Peak Bundle power (kW) of the envelope	Bundle outer element linear heat rating kW/m
Fuel center line melting	1450	91
1% plastic strain of sheath	1216	77
Element internal gas pressure	963	60.5

supplemented with the use of power ramp criteria, will ensure safe operation of fuel bundle without failure, taking into account all the power ripples that occur due to on-power refuelling and power maneuvering.

4.2. Derivation of fuel bundle operation limits based on design

To check the fuel elements for their capacity to operate, the elements are checked for the following conditions and the limiting power of element or bundle for design are estimated.

1. Fuel centerline temperature to be less than the fuel material melting point.
2. Fuel-sheath surface heat flux less than the critical heat flux.
3. Total plastic sheath strain shall not exceed 1%.
4. Internal fission gas pressure shall not be higher than the coolant pressure.

Bundle power envelopes with different maximum bundle powers have been considered for evaluating centerline temperature, sheath strain and fission gas pressure with a bundle burnup of 15 000 MWd/TeU using FUDA MOD2. Each power envelope is referred to as a case and results of the analysis for different cases are given in Table 1. The analysis is carried out for a maximum rated element of the bundle, using the fuel pellet and element dimensions as planned to be used in the reactor. Fuel centerline melting and critical heat flux can reach at any instant in a life period of a bundle which is based on the neutron flux and coolant conditions at that instant and the influence of previous operating history is minimal. Where as 1% plastic strain and internal gas pressure are dependent on the operating power burnup history of the bundle till that point of time.

4.2.1. Fuel centerline temperature

The fuel element centerline temperature reaches the UO_2 melting point at a linear heat rating of 91.235 kW/m, i.e. at $\int K\,dT = 71.14$ W/cm and bundle power of 1450 kW. With a 10% uncertainty factor for estimation of fuel central line melting temperature, the possible maximum bundle power is 1300 kW. Also based on literature, the fuel bundle can remain intact for considerable time while operating at UO_2 central line melting conditions.

4.2.2. Sheath strain criteria

Fuel-sheath fails when the hoop stress due to the internal gas pressure exceeds a threshold value which depends upon temperature and ductility of the sheath. The sheath diametrical strain possible at this condition will be more than 1%. However to be conservative, the limiting plastic sheath diametrical strain value of 1% is taken as guideline based on data on Zircaloy irradiation strain capability. Fuel element plastic sheath strain is limited to 1% at any point of time during its life time. It can be seen from Table 1 that the sheath plastic strain reaches nearly 1% for an envelope with peak element LHR of 77 kW/m.

4.2.3. Fission gas pressure

It can be seen from Table 1 that the most limiting criterion is imposed by the internal gas pressure condition of remaining below the coolant pressure even at the end of life of the fuel bundle, i.e. up to a burnup of 15 000 MWd/TeU. For a 37-element fuel bundle producing 963 KW bundle power, (LHR of 60.5 KW/m) the maximum gas pressure is 10.63 MPa which is less than the coolant pressure of 10.8 MPa.

During reactor operation, the bundle powers and channel power are estimated using the reactor physics code TRIVENI and compared time to time with actual channel powers based on measured flows and temperatures. To account for uncertainties in channel power and bundle power estimations, a 5% margin is provided over added to the above calculated values. This margin is based on experience of PHWR reactor of 220 MWe. With this the operating limit works out to be 917 kW.

The fuel-sheath can withstand a differential pressure (Pressure of internal fission gas – pressure of external coolant) which depends upon the sheath material properties and thickness. Using this capability of sheath, it is possible to generate a bundle power of 1035 KW (LHR of 65 KW/m) with the internal fission gas pressure reaching 11.5 MPa at 15 000 MWd/TeU burnup, which is slightly above the coolant pressure.

5. Conclusions

The computer Code FUDA has been developed to predict the fuel element performance and to supplement fuel type tests in qualifying a new fuel design. The code takes into account the inter-dependence of different parameters. The case study carried out for deriving the bundle power limit for the 37-element fuel bundle using computer code FUDA was described.

Diverse limiting operation conditions are considered: fuel central line melting, 1% plastic strain of clad and element internal gas pressure below the coolant pressure up to 15 000 MWd/TeU burnup with various maximum power envelopes. The latter dictates an operation limit of 963 KW for the power envelope. The fuel central line temperature and clad outside temperatures are 2168 °C and 312 °C, respectively at this bundle power. For the 37-element fuel bundle, producing a peak power of 1035 KW (and following the bundle power envelope), the fission gas pressure remains less than coolant pressure, up to a burnup of 12 000 MWd/TeU and exceeds coolant pressure by 0.7 MPa at 15 000 MWd/TeU burnup.

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