Thermo-hydraulic sub-channel analysis

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Aim

Presenting an overview of thermal hydraulic used for the analysis of sub-channel nuclear reactor core.

Overview

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- Light Water Assemblies

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- Referenced Experiments

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- Codes for Thermal Hydraulic Analyses

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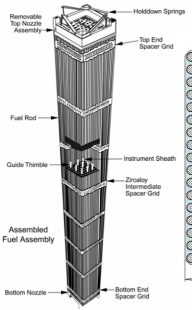
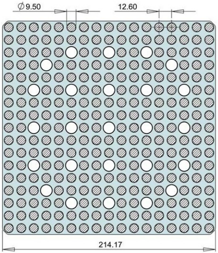
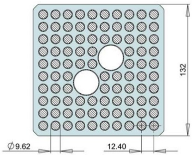
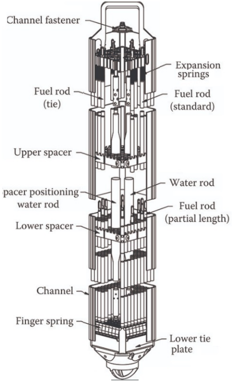
- Referenced Experiments
- Codes for Thermal Hydraulic Analyses
- Goals for the Dissertation

Reactor core fuel assemblies









Inside the fuel assembly, fuel pins are arranged in square lattice configuration.

Reactor Operations

The safety of nuclear reactors is to be ensured under:

- normal operation,
- operational transients,
- anticipated operational occurrences,
- design basis accidents (DBA) and
- under extreme emergency situations by incorporating the engineered safety systems

Experiments and numerical simulations of nuclear reactor core are required to ensure reactor safety

Representative Experiments

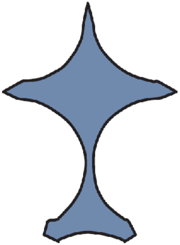
These experiments are oriented towards the improved understanding of the coolant mixing inside the fuel assembly sub-channel.

List of experimental studies of nuclear fuel bundle sub-channels.

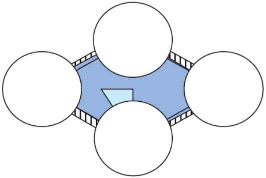
Investigator	Geometry	P/D	Re	Measured Data	Methodology
Skinner et al. (1969)	6 equally spaced rod	1.42	20000–80000	Velocity, Pressure	Nitrous oxide tracer, Pitot static probe.
Van Der Ros and Bogaart (1970)	Compound rect. channel	1.05	5000–30000	Pressure, velocity, Temperature	Pitot tubes, thermocouple, thermal tracer.
Rowe et al. (1974)	Sq. 3 × 3	1.125 1.25	50000–200000	TI auto.corr. functions	LDV
Trupp and Azad (1975)	Tri., array	1.2–1.5	12000–84000	Axial velocity, wall shear stress, Reynolds stress, eddy diffusivity	Pitot static probe, hot wire anemometry.
Carajilescov and Todreas (1976)	Sq. Channel	1.12	27,000	Axial velocity, Turbulent Kinetic energy, Reynolds stress	LDA
Rehme (1978)	Four Parallel rods	1.12	87000	V,TauW, k	Pitot tubes, preston tubes, HWA
Seale (1979)	Row of 3, 4, 5 rods	1.1 1.833 1.375	34369–299603	T and velocity distributions	Pitot temperature probe, manometer
Guellouz and Tavoularis (1992)	Rect. channel	1.025–1.35	108000	V, TauW, Reynolds stress, flow visualization	2D PIV, Acetic acid tracer.
Gosset and Tavoularis (2006)	Rect. Ch. With rods	g/d = 0.025–0.7	388–2223	Flow visualization	Dye injection
Baratto et al. (2006)	5 rods	1.149	42000	V, V', PSD, Turbulent stress	HWA, Preston tube.
Silin and Juanico (2006)	Three rods in circular channel	1.20, 1.12	1400, 130000	Temperature	Thermocouple

Experimental considerations

For two phase flow conditions investigations were carried out on void distribution and cross flow mixing during boiling.
The study of Rowe perfomed the measurement of flow and enthalpy
Eifler and Nijsing studied the shear stress on rod walls



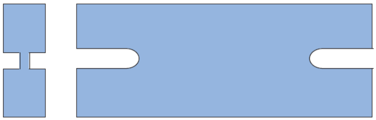
Cross section of test section (1967)



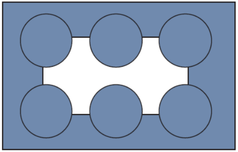
Cross section used by Eiffler and Nijsing

Experimental considerations

In 1994 Tapucu et al. investigated mass exchange under two phase flows between two laterally interconnected sub-channel.



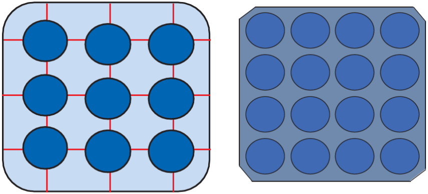
Square channel Model



Subchannel Model

Experimental considerations

Lahey et al. measured enthalpy and flow distributions at the outlet of square-lattice bundle (GE) 9-rod bundle.
Similarly, ISPRA-16 conducted two phase flow distribution tests in rod bundle under PWR operating conditions.



GE-9 rod and ISPRA-16 rod two phase flow distribution test bundles.

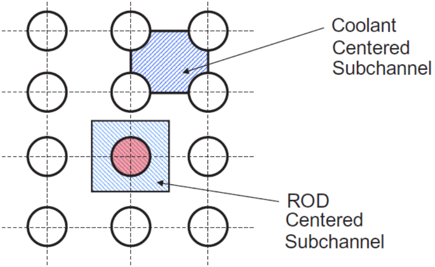
Thermal Hydraulic analysis

Thermal Hydraulic analysis of nuclear reactor core is carried out using the sub-channel analysis code to estimate the various thermal hydraulic safety parameters like:

- detailed flow and temperature distribution
- critical heat flux (CHF) ratio,
- critical power ratio (CPR),
- fuel center line temperature,
- fuel surface temperature,
- sub channel maximum temperature and
- bulk coolant outlet temperature.

Thermal Hydraulic codes

Inside the fuel assembly, fuel pins are arranged in square lattice configuration.



In BWR channel assemblies, the entire core or symmetric sector of it is analysed to estimate TH parameters.

Thermal Hydraulic analysis

Fluid models

- homogeneous
- mixing
- two-phase flow
- drift
- diffusion

Thermal Hydraulic Codes

Sub-Channel Code	Version/Year	Organisation/Country	Application	Reference
SCEPTIC	1971	EIR/Switzerland	GAS / PWR	Eriksson, 1971
COBRA	I/1967, II/1970, III/1971, IIIC/1973, IV-I/ 1976, IV/1977	BNWL, USA	PWR, LMFBR	Rowe, 1967, 1970, 1971, 1973; Wheeler et al., 1976; Stewart et al., 1977
VIPRE	1/1993, 2/1994	PNL/EPRI/USA	PWR	Stewart et al., 1993; VIPRE-02, 1994
SCRIMP	1977	EIR/Switzerland	GAS / PWR	Huggenberger, 1977
MATTEO	1973	MIT/USA	BWR	Forti and Gonzalez-Santalo, 1973
WOSUB	1978	MIT/USA	BWR	Wolf et al., 1978
CANAL	1979	MIT/USA	BWR/PWR	Faya et al., 1979
ENERGY	I,II,IV/1975	MIT/USA	LMFBR	Chen and Todreas, 1975
SUPERENERGY	I, II/1980	MIT/USA	LMFBR	Basehore and Todreas, 1980
MISTRAL	II/1972	KFK/Germany	PWR	Baumann, 1972
DIANA	1974	Hitachi/Japan	PWR	Hirao and Nakao, 1974
THERMIT	II/1981	MIT/US	LMFBR	Kelly et al., 1981
MATRA	1999	KAERI/S. Korea	PWR	Yoo et al., 1999
MATRA-LMR	2002	KAERI/S. Korea	LMFBR	Kim et al., 2002
HAMBO	1967	AEEW/UK	PWR	Bowring, 1967
FLICA	III, IV/2000	CEA/France	LMFBR	Toumi et al., 2000
THINC	I/1968,II/1972,IV/1973	USA	PWR	Chu et al., 1973
LYNXT	1 & 2/1976	B & W/US	PWR	BAW-10156-A
TORC	1975	CE/USA	PWR	CENPD-206-NP
NASCA	2001	Japan	PWR/LMFBR	Ninokata et al., 2001
SUBCHANFLOW	2010	KIT/Germany	PWR	Imke et al., 2010
ATHAS/OE	2016	Jiaotong University, China	PWR	Pan Wu et al., 2016

Directions

Therefore, the research will be directed in

- Improved thermal hydraulic models of heat transfer and drift correlations,
- Reliable sub-channel analysis tools,
- Implementing strategies for the calculation speed,

Thanks !!
Questions??