U.S./JAPAN HIGH POWER DENSITY WORKSHOP

by

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GENERAL ATOMICS

Fusion High Power Density Devices and Designs, Bridging Between Technologies
(U.S./Japan Fusion Cooperation Program since 1995)

This workshop is being organized by
Prof. S. Toda from Tohoku University, and Clement Wong/Chandu Baxi

(The U.S. OFES sponsor is Mr. Sam Berk)

Next workshop date: October 20–22, 1997

Location: The Aoba Memorial House, Tohoku University, Sendai, Japan

[These were selected in coordination with the ICFRM-8 meeting on Fusion Reactor Materials, and other related workshops and meetings in Sendai or Mito (Tokai) before and after ICFRM-8]

Second High Power Density Components and Devices Workshop
Summary
(February 17–21, 1997, San Diego)

For a magnetic confinement device, the plasma power density defines the distribution of neutron wall loading and surface heat flux on the fusion power core surface and divertor. For experimental devices like (1) ITER, LHD, NSTX, and LAR-VNS, when power conversion is not required, neutron wall loading is usually not a concern, and very high divertor surface loading of as high as 30 MW/m^2 can possibly be handled by sub-cooled flow boiling of water. For conceptual power plant designs like (2) ARIES-RS and GA-LAR, to minimize the volume of the fusion power core, the average neutron wall loading can be designed to the range of 4 to 8 MW/m^2, but due to the need for efficient power conversion and therefore reasonably high coolant outlet temperature, the acceptable divertor surface loading may be limited. As an example, for the ARIES-RS design, when lithium is used as the divertor coolant, and the T_{max} for V-alloy is $\leq$700°C, the maximum allowed surface loading is less than 6 MW/m^2. It is recognized that the (3) limitation on divertor surface loading is very sensitive to the combination of three design choices: the first is the (3a) coolant, e.g. water, lithium, helium, FLiBe, and gas-solid mixture, the second is (3b) heat transfer enhancement scheme, e.g. sub-cooled flow boiling, heat transfer surface enhancement, high helium pressure, porous medium, and the third is the (3c) structural material like ODFS, and V-alloy. Advancement in structural material development can relax this divertor heat flux limitation.

We also recognized that in order to evaluate and design a robust high power density fusion component, (4) it is crucial that detailed design parameters and operating scenario are iterated with the discipline of plasma physics. Therefore, in addition to the conclusions from different specific high power density components design and different innovative heat transfer options, we have also arrived at a list of general conclusions that are important for the development of high power density components design for magnetic confinement fusion.

TECHNICAL RECOGNITIONS ON THE DEFINITION OF FUSION HIGH POWER DENSITY COMPONENTS

- Average plasma power density and specific design define the high power density components
- High power density limitations can be applied to neutron wall loading and surface loading
- Neutron wall loading limitation applies to fusion power plants
- Higher neutron wall loading can lead to smaller fusion power core
- Maximum neutron wall loading is limited by design concepts
- 4 to 8 MW/m^2 average neutron wall loading may be possible
- COE is a trade off between average neutron wall loading and components lifetime
- Surface loading limitation applies to experimental devices e.g. ITER and fusion power plants, limitations are different
- Sub-cooled flow boiling of water can remove up to 30 MW/m^2 surface loading
- ARIES-RS divertor design handles a surface loading of <6 MW/m^2

GENERAL CONCLUSIONS

- Divertor heat flux is a key performance limitation for magnetic confinement devices
- Coolant and material selection and therefore divertor heat flux limitation can be very different between experimental and power conversion devices
- D-T fuel cycle with 80% energy in the neutron channel is preferred for high power density components designs
- Transient events like disruption, run-away electrons, giant ELMs have to be eliminated
- Advanced low activation material development is necessary for higher divertor heat flux removal enhancement
- Effective coordination between various areas: physics, technology, materials, and engineering, is the key for the successful development of fusion power generation

RESEARCH AND DEVELOPMENT AREAS
(Japanese interests in red)

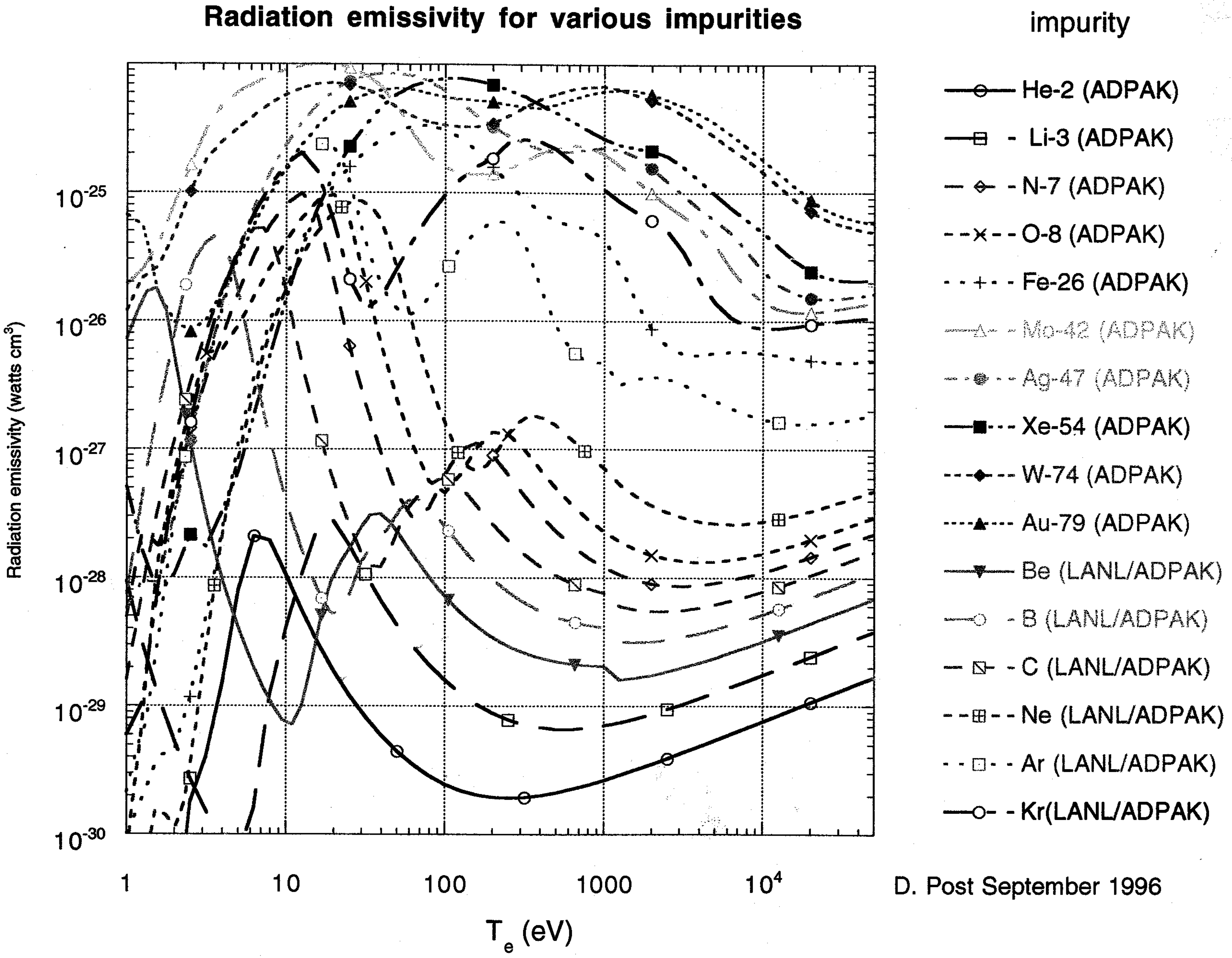
- Materials
  - Advanced low activation materials, e.g. oxide dispersion ferritic steel, TiAl-alloy, V-alloy, V-alloy metal matrix
- Plasma and components interaction
  - Plasma and material interaction
  - Use of W as the plasma facing material and its necessary bonding with the substrate material
  - Effective distribution of heat flux by radiation optimization in the core, mantle, SOL and the divertor regions
- Advanced systems
  - He-impurities and V-alloy compatibility for the He and V-alloy application
  - Gas and solid technology as a heat removal system
  - Thermal performance and material compatibility of FLiBe
  - Porous medium technology as a heat removal system

HIGH POWER DENSITY COMPONENTS AND DEVICES WORKSHOP
(CONTINUED)
TECHNICAL SUMMARY FROM PRESENTATIONS

- Material erosion
  - Molecular dynamics simulations indicated the results of ion beam injection, the possibility of cluster impact instead of the monomer impact. The former shows that the erosion volume due to ^{256}Cu impact is proportional to the incident of the cluster

- Safety
  - Ingress of coolant event (ICE) and loss of vacuum event (LOVA) experiments have been initiated
  - ICE showed that the vacuum vessel pressurization rate may depend upon the “flushing” phenomena and the boiling heat transfer on the wall
  - To enhance the safety feature of Force Free Helical Reactor (FFHR), FLiBe was selected as the coolant and tritium breeder. Low activation structural material, JLF-1 was selected. Multiple barrier confinement approach was adopted. Safety criteria have been selected and a systematic evaluation of the safety of FFHR is being conducted

Radiation emissivity for various impurities



RESEARCH AND DEVELOPMENT AREAS (CONTINUED)
(Japanese interests in red)

- Fundamental heat transfer
  - Verification of thermal performance by larger scale mock-up experiments, including flow stability, manifolding, steady state versus transient scenarios, startup and shutdown studies
  - Development of single sided heat transfer under relevant condition with improvements in experimental measurements and data
- System study
  - Development of an international system code, heat transfer module should be included
- Safety
  - Detailed safety impacts should be studied, e.g. relating to afterheat, vacuum vessel pressurization and radioactive materials release

HIGH POWER DENSITY COMPONENTS AND DEVICES WORKSHOP
TECHNICAL SUMMARY FROM PRESENTATIONS

- Conceptual design
  - At neutron wall loading of 8 MW/m^e and a η_{th} = 46%, the low aspect ratio concept (LAR) has the possibility of providing a COE <60 mill/kW•h
  - A helium-cooled, V-alloy stagnant Li breeder blanket can be operated at neutron wall loading >8MW/m^2, but the issue of compatibility between helium impurities and V-alloy will have to be resolved. When applied to the LAR concept, the issue of central column cooled by water and the outboard blanket containing Li will also have to be addressed
  - Radiative divertor design is an approach to distribute the transport power to the first wall, possible reduction of core performance due to the increase of Z_{eff} will have to be addressed
  - W or W-alloy may be the only suitable divertor coating material
  - The Spherical Torus approach (same as LAR approach) to magnetic fusion may allow lower development cost toward fusion power generation, but many physics and technology challenges have to be resolved