HELIUM-COOLED REFRACTORY ALLOYS FIRST WALL AND BLANKET-Review

Tasks and leaders

Thermal hydraulics	C. Wong, C. Baxi
Porous First wall Design	R. Nygren
Configuration	B. Nelson, P. Fogarty
Material compatibility	S. Zinkle N. Ghoniem S. Sharafat
Helium impurity control	M. Ulrickson
Neutronics	M. Youssef
Activation and afterheat	H. Khater
Tritium	D. K. Sze S. Willms
Safety and reliability	K. McCathy B. Merrill
CCGT	R. Schleicher
System Study	C. Wong

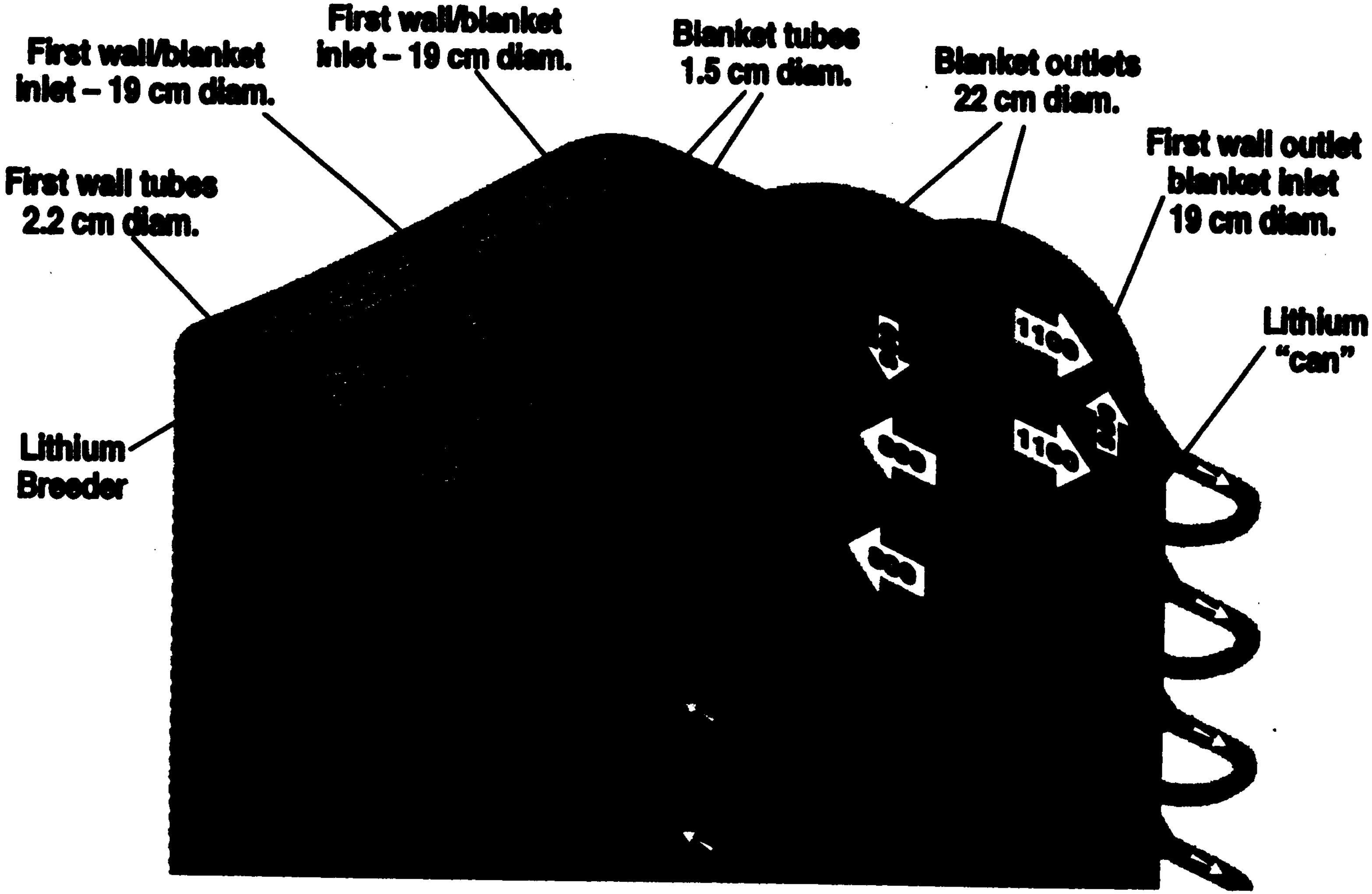
APEX-8 meeting, Nov. 8-11, 1999, UCLA

STRUCTURAL MATERIAL SELECTION AND COMPATIBILITY

- Design goal: average neutron wall loading = 7 MW/m^2, surface heat flux = 2 MW/m^2, and a peaking factor of 1.4
- Ta, Mo, W, Nb and V refractory alloys were evaluated in 1998
- W-5Re alloy was selected in 1999 because of projected high temperature performance
- The unirradiated mechanical properties of tungsten are strongly dependent on thermomechanical processing conditions. The best tensile and fracture toughness properties are obtained in stress-relieved material
- Since data are not available on the possible radiation-enhanced recrystallization of W, and also to account for the presence of welds in the structure, recrystallized mechanical properties were used
- W-5Re T_{min} > 800° was selected to avoid significant increase in the ductile to brittle transition temperature (DBTT)
- T_{max}< 1400°C was selected by consideration of thermal creep, helium embrittlement and oxide formation issues.

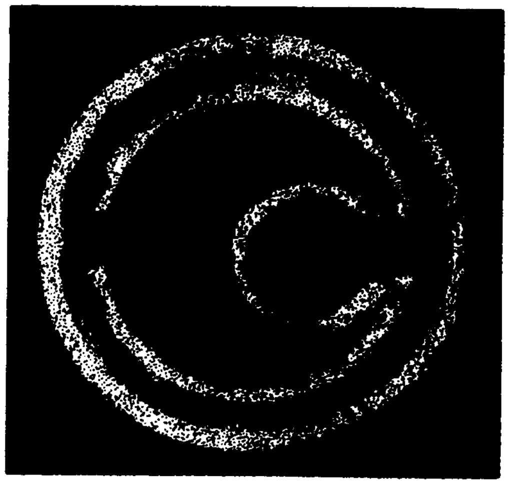
HELIUM COOLED BLANKET CONCEPT

- One of three "cans" per outboard sector shown with top removed



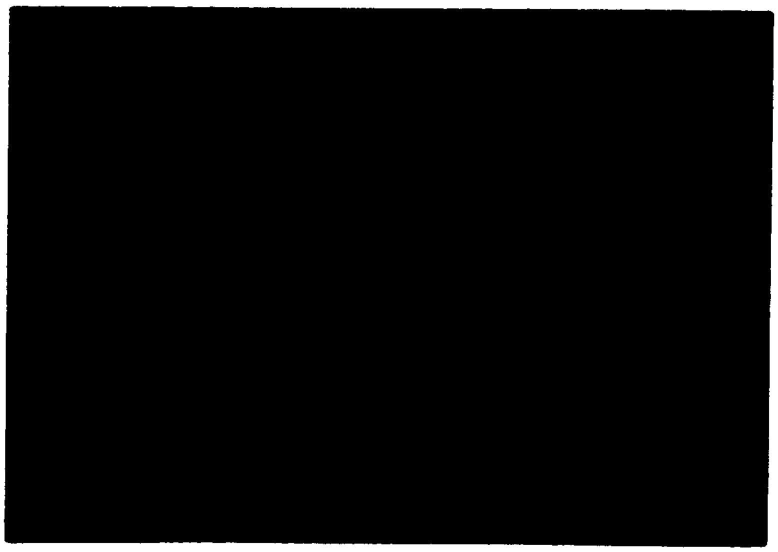
POROUS MEDIUM

- A possible option for the first wall design
  - Thermacore, Inc. of Lancaster, PA, through grants from DOE's Small Business Innovative Research (SBIR) Program, developed water-cooled and Helium-cooled modules that utilized brazed copper balls as a porous heat transfer medium in copper, Glidcop™ or molybdenum coolant channels.
  - Their objective was easy-to-fabricate designs for heat sinks that operated at moderate temperature. Thermacore circumferential



Thermacore circumferential flow design

  - Ultramet of Pacoima, CA. Ultramet uses a process in which they build up refractory material with chemical vapor deposition (CVD)

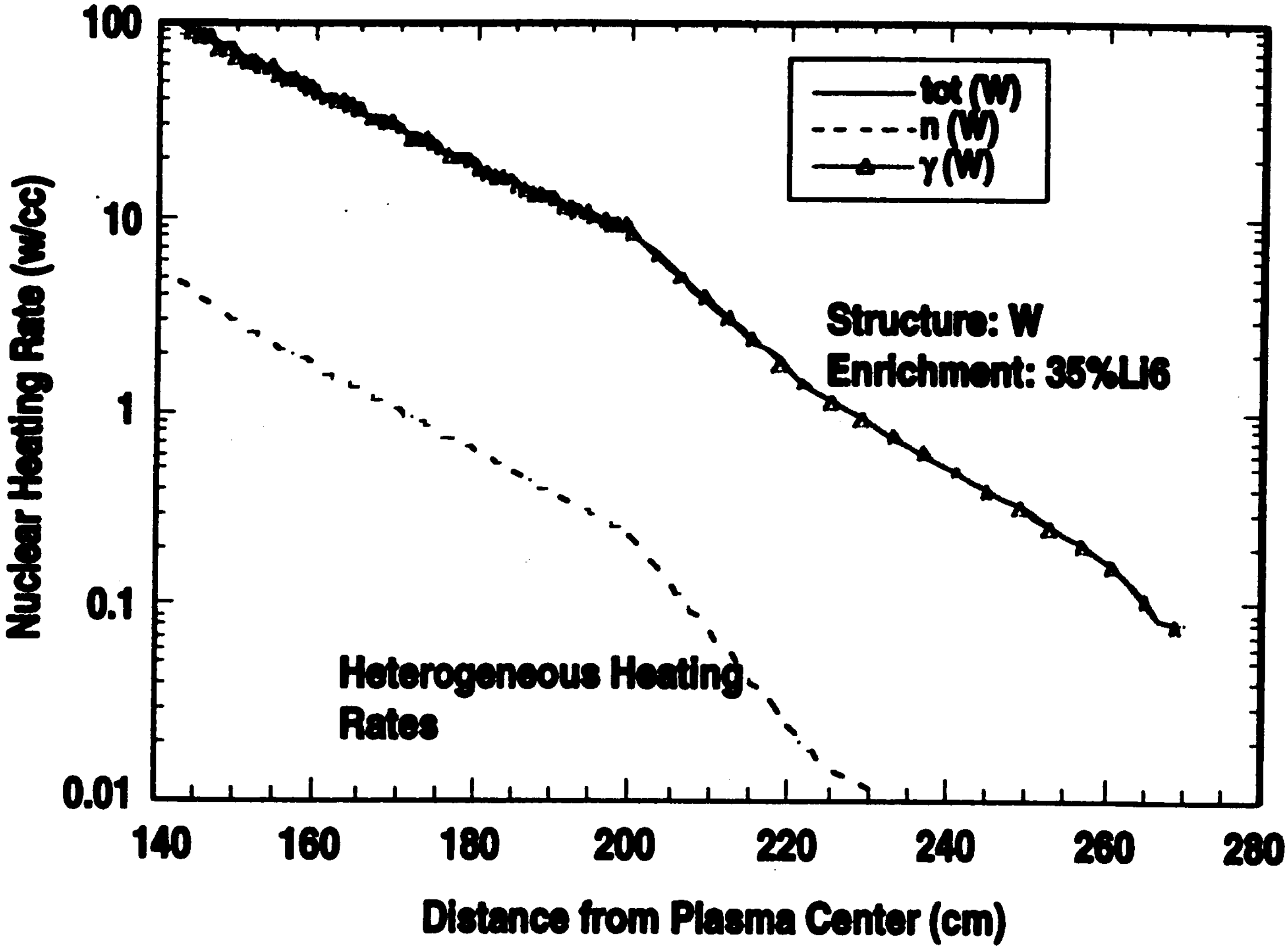


Porous Ta implant, diam. is 0.75 in.

NUCLEAR ANALYSIS

- Based on the material volume fractions generated, we assessed the impact of W-alloy on: nuclear heating profiles across the blanket, power multiplication, tritium breeding profiles and tritium breeding ratio (TBR), impact of Li-6 enrichment
- We assessed the damage indices: DPA, helium, and hydrogen production rates at several key locations, including the vacuum vessel (V.V) and TF coil case
- When compared to other refractory alloys like TZM and Nb-1Zr, the best local TBR performance is with W and Li breeder. The TBR increases with Li-6 enrichment and starts to saturate at a value of ~1.43 when Li-6 enrichment is ~35%
- The damage parameters, at various locations were estimated in the W-alloy design. Compared to the liquid breeder Flibe, liquid lithium is the less effective material in attenuating the nuclear flux at the V.V. and TF coil by a factor of 6 to 10
- The radioactive waste characteristics of different components of the machine were evaluated according to both the NRC 10CFR61 and Fetter waste disposal concentration limits (WDR)
- According to Fetter limits, the first wall, module wall, blanket, and transitional zone would not qualify for disposal as Class C waste
- The W-5Re alloy produces such a high activity that the first wall would have a WDR more than order of magnitude higher than the Class C WDR limits. (The high WDR is due to the ^{186m}Re, ^{108m}Ag, and ^{94}Nb isotopes. Only ^{186m}Re is a product of nuclear interactions with base elements in the W-5Re alloy)

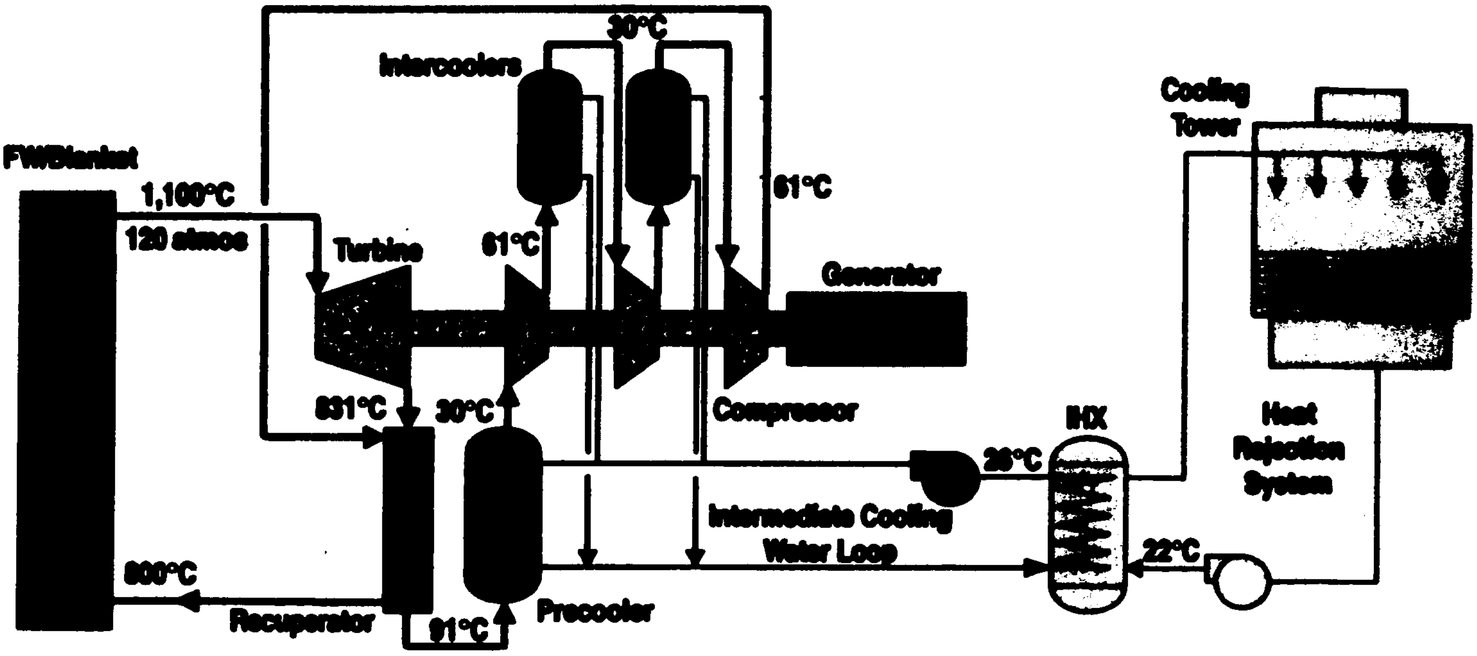
TOTAL NUCLEAR HEATING RATES IN THE STRUCTURE



GENERAL ATOMICS

POWER CONVERSION SYSTEM

Closed Cycle Gas Turbine
(96% recuperator effectiveness was assumed)

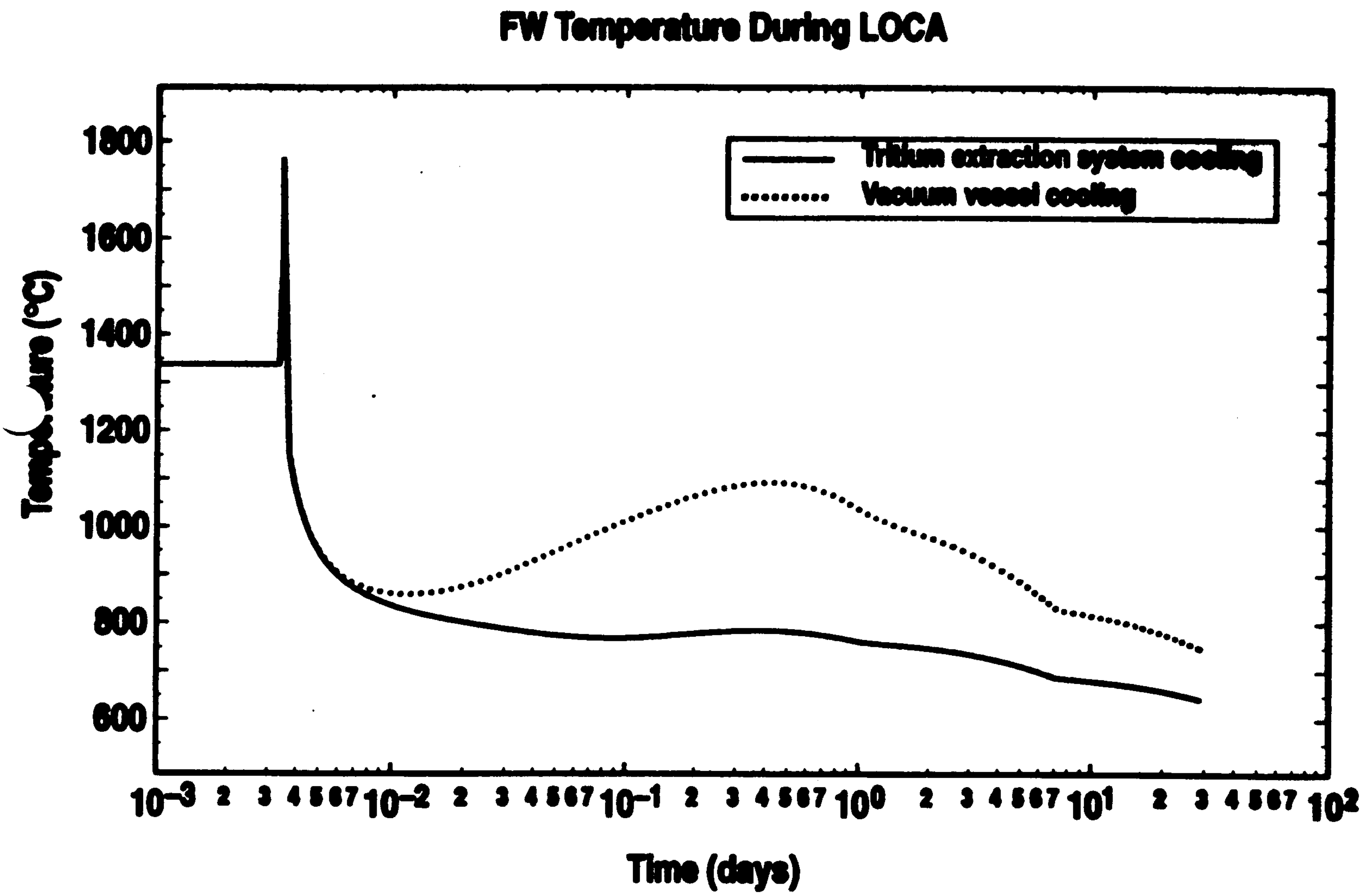


Power Conversion System Process Arrangement

TRITIUM MIGRATION AND CONTROL

- The design goal of the tritium recovery system for lithium is usually to limit the tritium concentration to about 1 appm
- This goal is to limit the total tritium inventory in the lithium to less than 200 g
- Similar to the ITER design, a cold trap process with protium addition to the lithium is proposed. With the total hydrogen concentration at the saturation limit of 250 appm, the tritium concentration can be below 1 appm
- The proposed lithium flow rate for tritium recovery is 20 kg/s. This is a very low flow rate. To maintain uniform tritium concentration, the local flow rate has to be controlled. This design impact will have to be assessed
- Recent experimental results from Tritium System Test Assembly show that a permeation window can be used for tritium cleanup from a gas. This process is capable of removing tritium from a gas to ~1 Pa. Since permeation is not a critical issue for this design, 1 Pa tritium partial pressure may be acceptable. The effect of this tritium partial pressure on the operation of the pumps and valves will have to be assessed

FW TEMPERATURE FOR TWO SCENARIOS: VACUUM VESSEL COOLING OPERATES DURING LOCA, AND TRITIUM EXTRACTION SYSTEM OPERATES DURING LOCA



CONCLUSIONS

- We completed the preliminary design of a high performance He-cooled W-5Re alloy FW/blanket design
- A separate first wall that is permitted to flex under heating and a lithium pool configuration was selected
- Due to the lack of irradiated data, conservative assumptions on selecting the W-alloy properties have to be used
- With regard to the compatibility of W-alloys with oxygen as the primary helium impurity, commercially available solid gettering modules can maintain the impurity level to <1 appm and prevent embrittlement
- Potentially, based on the results of selected analyses, the FW/blanket design could meet the material temperature and structural design limits, provided that the peak structural loading during disruptions can be mitigated
- The 1-D tritium-breeding ratio of 1.43 can be reached with a Li-6 enrichment of 35%. But the presence of induced radioactivity will not allow the W-alloy components to meet the criteria for classification as low-level waste
- W-alloy will generate a high level of afterheat, but with the tritium extraction system operating, long-term accident temperatures remain below 800°C
- A cold trap process with added protium to the lithium could be used for tritium extraction
- At the CCGT gross thermal efficiency of 57.5%, a super-conducting reactor with an aspect ratio of 4 and an output power of 2 GW(e) is projected to have a COE of 54.6 mill/kWh

Helium-cooled Refractory alloy Design
Proposed next iteration for APEX
(To be developed under piggyback mode)

"Replace the Li breeder with Flibe"

Motivation:

Move away from the chemical reactivity concern of Li

Tasks:

- **Compatibility**
- **Thermal hydraulics, Tin/Tout=900/1200° C**
- **Neutronics**
- **Activation/afterheat**
- **Safety**
- **Tritium control**
- **Power conversion**