Idaho National Engineering and Environmental Laboratory

Tritium Control and Safety

Brad Merrill
Fusion Safety Program

1st APEX Physical Meeting, April 8, 2003
Presentation Outline

• Temperature & tritium control approach
• Tritium inventory
• LOCA & LOFA temperatures
• Mobilization & release of radioactive inventories
• Dose results
• Status of safety assessment
CLiFF Temperature Control Approach

- Primary heat transport system (PHTS)
  - Primary piping (single walled) & heat exchanger are electrically heated and insulated (internally) to maintain temperatures at desired levels (> 300 C) and outer wall temperatures cool (~90 C)
- Internal vessel component
  - Flinabe charging/draining
    - VV bakeout => thermal cycling primary confinement boundary (VV) may be a safety concern
    - Internally heating by gas flow => alternate system that would contain radioactive material
CLiFF Tritium Control Approach

- Problem: tritium solubility & diffusivity are low in Flinabe and high in AFS
- Tritium control & recovery
  - Intermediate salt loop between primary Flinabe loop and steam power cycle (MSRE used a mixture of NaF and NaFB$_2$)
  - Internal permeation barrier (from thermal insulators?) for PHTS pipe walls or pipe walls clad with aluminum
  - Bakeout of Be multiplier tritium at 600 C < T < 650 C
    (Should not be accomplished by heating VV, argues for internal gas heating system)
  - Blanket replacement schedule of one quadrant every 6 months
TMAP Predicted Tritium Inventory & Permeation

• Tritium Inventory
  – AFS primary loop ~ 114 g with 78 g in blankets.
  – Inventory divided by AFS volume - 1.0 g/m$^3$ - HYLIFE II ~ 1.4 g/ m$^3$
  – Flinabe < 2.0 g
  – Be multiplier - 2.3 kg over the blanket lifetime
    – 660 g if bakeouts are successful
  – Limit 10 mSv => 130 g stacked release, 15 g ground release

• Permeation
  – Airborne - Limit (0.1 mSv/yr) => routine ground level release 1.3 g/y
    – Not calculated because permeation barrier material not defined
    – ALARA will require PHTS vault (a confined space) to have an air detritiation system for leaks and accidental releases
  – Water - Limit 20,000 pCi/l
    – Efficiency of intermediate salt loop to convert T$_2$ to T$_2$O not known
Bypass Accident Schematic

Aerosol deposition in non-nuclear room

Aerosols form from FW oxidation and molten salt

T₂ released from FW forms HTO in air humidity

VV duct isolation valves fail and reentrant flow is established in duct

Reentrant flow established in HVAC duct

Aerosol deposition in non-nuclear room
Flinabe decay heat is a problem during a LOFA, might either require draining of primary or in-vessel natural convection loops
Safety Assessment for Worst Case Bypass

- MELCOR LOVA model (VV, bypass duct, adjoining room, ventilation ducts) with FW temperatures from LOCA (1/4th) & LOFA (3/4th) with VV cooling
- Radioactive inventories
  - AFS specific dose - (varies with maximum of 15.3 mSv/kg)
    (Mn-54 26%, Ca-45 15%, and Ti-45 14% of dose)
  - Tritium specific dose - 77 mSv/kg
  - Flinabe specific dose - 1.1 mSv/kg
    (F-18 91%, Na-22 7%, & Na-24 2%)
- Mobilization
  - AFS from HT9 oxidation data for LOCA FW temperatures
  - Permeation into vacuum vessel by transient TMAP analysis
  - Flinabe by evaporation in vacuum (t < 100 s), and by diffusion through boundary layer (t > 100 s)
Total dose is 1.5 mSv (neglecting Be tritium) if release is stacked, which is below 10 mSv limit for no-evacuation plan.
Status of Safety Assessment for CLiFF Design

- Worst case bypass accident analysis has been completed
  - Dose at site boundary is 1.5 mSv (neglecting Be tritium) after one week (< 10 mSv limit), building must be isolate by five days for ground release
  - When 660 g of Be tritium is included, the dose exceeds 10 mSv in six days for a stacked release, two days for a ground release
  - Ample time to manually operate plant remediation and isolation systems
- Routine releases were not determined
- Waste disposal rating - CLiFF meets Class C disposal
  - AFS structure WDR 0.14-0.80 with Fetter limits, dominated by Tc-99
  - Flinabe WDR 0.025 with 10CFR61 limits, major contributor is C-14
- Flinabe decay heat is a safety issue
  - Decay heat primarily due to Na
  - Passive means for dealing with Flinabe during accidents must be developed (drain valve or internal VV natural convection)
- Safety paper completed except for CLiFF blanket design overview
Task IV Temperature Control Approach

• Primary heat transport system (PHTS)
  – Halo heating system - thermal blanket of helium for PHTS maintains internal VV, piping and heat exchanger temperatures at desired levels (> 500 C) to avoid Flibe freezing during primary loop charging/draining
  – Halo system provides high temperature shield (HTS) cooling during normal operation
  – Triple wall PHTS piping with thermal barrier formed by laminar flow regions at pipe walls (outer wall temperature ~100 C)

• Secondary heat transport system (SHTS)
  – High and low temperature end of cycle separated by internal divider
  – Outer (pressure boundary) wall temperatures cooled by low end gas (~90 C)
Primary Temperature Envelope

Cross-Section of ARIES-AT Power Core

Triple wall pipe

Heat exchanger

Secondary

100 C

500 C

700 C

1.4 m

1.0 m

0.7 m
Pebble Bed Modular Reactor

Helium Brayton cycle

Turbo-compressor

Pressure boundary (90 C)

Generator

Power turbine

Inter-cooler
Pre-cooler
Recuperator
Task IV Tritium Control Approach

• Problem: tritium solubility & diffusivity are low in Flibe and high in AFS
• Tritium control & recovery
  – Brayton and Halo helium cleanup systems
  – Cool outer pipe/mixer and Brayton cycle walls
  – Aluminum pipes in Brayton cycle coolers
Coolant Purification System (CPS)

A fraction of 0.1% of the helium coolant stream is fed into the Coolant Purification System as a so-called slip stream. The main tasks of the CPS are:

- to extract hydrogen isotopes as well as solid, liquid or gaseous impurities from the main coolant system and thus to maintain the radioactivity and the impurity concentration below specified levels;
- to provide an oxygen potential which is sufficiently high for maintaining the integrity of the tritium permeation barriers (oxide layers) in the main coolant system; this is done by adding a certain amount of water and hydrogen to the CPS;
- to remove condensed water that may be entrained in the cooling gas following a steam generator-tube failure.

Some characteristic data of a CPS for the BOT concept are given in the following Table:

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Gas flow rate</td>
<td>2.4 kg He/sec  = 4.9·10⁴ Nm³/h</td>
</tr>
<tr>
<td>Pressure</td>
<td>8 MPa</td>
</tr>
<tr>
<td>Total amount of gas</td>
<td>20 000 kg He</td>
</tr>
<tr>
<td>Gas inlet temperature</td>
<td>723 K</td>
</tr>
<tr>
<td>Gas outlet temperature</td>
<td>523 K</td>
</tr>
<tr>
<td>Radioactive impurities</td>
<td>HT, HTO, activated corrosion products</td>
</tr>
<tr>
<td>Non-radioactive impurities</td>
<td>H₂O, N₂, corrosion products</td>
</tr>
<tr>
<td>Tritium removal efficiency</td>
<td>&gt; 95 %</td>
</tr>
</tbody>
</table>

The Q₂ content (Q = H, D, T) in the coolant can be calculated as follows:

permeation from First Wall : 0.8 mole T + 1.1 mole D per day /1/
permeation from Purge Gas System: 0.27 mole T + 98 mole H per day /1/

The amount of H is in accordance with the amount of T and results when the swamping ratio in the purge gas system (He : H₂ = 1000), the purge gas mass flow rate (0.6 kg He/sec), and the higher permeation of H in comparison to T (factor 1.732) are taken into account.

The total permeation rate into the coolant is \( \dot{n}_p = 100 \text{ mole Q/day} = 2.1 \text{ mole Q}_2/\text{h} \).
TMAP Predicted Tritium Inventory & Permeation

- **Tritium Inventory**
  - AFS primary loop ~ 82 g, with 62 g in blankets.
  - Inventory divided by AFS volume - 0.8 g/m³ - HYLIFE II ~ 1.4 g/ m³
  - Flibe & helium ~ 1.1 g and 5.5 g, respectively
  - Limit 10 mSv => 130 g stacked release, 15 g ground release

- **Permeation**
  - Airborne - Limit (0.1 mSv/yr) => routine ground level release 1.3 g/y
    - Mixer, pipe walls, secondary pressure boundary ~ 1.2 g/y
    - ALARA will require PHTS vault (a confined space) to have an air detritation system for He leaks and accidental releases
    - Brayton cycle helium leak rate unknown; Chinese HTR-10, 200 MWth fission reactor leak test at 3.9 MPa at ~1.7 cc/s
  - Water - Limit 20,000 pCi/l
    - Brayton cycle coolers ~ 4.0 Ci/y; 13 pCi/l per pass; 0.2 Ci/m³ at end-of-life; CANDU experience 500-900 Ci/m³ entering processing
Unacceptable AFS FW oxidation during bypass accident will occur at these temperatures.
Flibe provides thermal inertia during LOFA and VV natural convection appears to be able to remove decay heat (~3.6 MW max load)
Safety Assessment for Worst Case Bypass

• MELCOR LOVA model (VV, bypass duct, adjoining room, ventilation ducts) with FW temperatures from LOCA (1/4\textsuperscript{th}) & LOFA (3/4\textsuperscript{th}) with VV cooling
• Radioactive inventories
  – AFS specific dose - (varies with maximum of 10.6 mSv/kg) (Mn-54 26 %, Ca-45 15%, and Ti-45 14% of dose)
  – Tritium specific dose - 77 mSv/kg
  – Flibe specific dose - 0.32 mSv/kg (99% F-18)
• Mobilization
  – AFS from HT9 oxidation data for LOCA FW temperatures
  – Permeation into vacuum vessel by transient TMAP analysis
  – Flibe by evaporation in vacuum (\( t < 100 \) s), and by diffusion through boundary layer (\( t > 100s \))
Dose at Site Boundary from Bypass Accident

Total dose after one week is 0.93 mSv (< 10 mSv no-evacuation plan limit) if release is stacked.
Status of Safety Assessment for Task IV Design

• Worst case bypass accident analysis has been completed
  – Dose at site boundary is 0.93 mSv after one week (< 10 mSv limit),
    must be isolated by one week for ground release
  – Ample time to manually operate plant remediation and isolation
    systems
• Routine releases are within limits
• Decay heat safety issue (?)
  – During LOCA FW temperature reaches 1140 C, internal VV natural
    convection will also be proposed for this design
• Waste disposal rating - AFS could be above Class C limit
  – AFS structure WDR is 0.33-1.97 with Fetter limits, dominated by Tc-99
    produced from Mo; reduce Mo content from 0.02% to <0.01%
  – Flibe WDR is 0.042 with 10CFR61 limits, major contributor is C-14
    from neutron reactions with F
• Safety paper completed