**ACRS Full Committee on Palo Verde Plant License Renewal** February 10, 2011

Facts for the Full Committee and for the record:

1. On May 23, 2007 Bonaca wrote Kline, SUBJECT: PROPOSED TECHNICAL BASIS FOR THE REVISION TO 10 CFR 50.46 LOCA EMBRITTLEMENT CRITERIA FOR FUEL CLADDING MATERIALS, ML071490090. Bonaca wrote, "The requirements of 10 CFR 50.46 (a) and (b) limit the amount of embrittlement that may occur as result of a design basis accident. They specify limits for the peak clad temperature, the global oxidation of cladding, and the local oxidation of cladding. There are several deficiencies with the current regulations. The correlation specified for the rates of steam reaction with the cladding is viewed by the technical community as an anachronism." Now, Appendix K to Part 50--ECCS Evaluation Models, Item 5, specifies that the rate of energy release from the metal/water reaction shall be calculated using the Baker-Just equation and § 50.46 Acceptance Criteria, item (b)(1) specifies the peak clad temperature, 2200 degrees.

2. The NRC staff fiercely defends Baker-Just in its Technical Safety Analysis, ML041210109, April 29, 2004, *"The Baker-Just correlation using the current range of parameter inputs is conservative and adequate to assess Appendix K ECCS performance. Virtually every data set published since the Baker-Just correlation was developed has clearly demonstrated the conservatism of the correlation above 1800°F."*

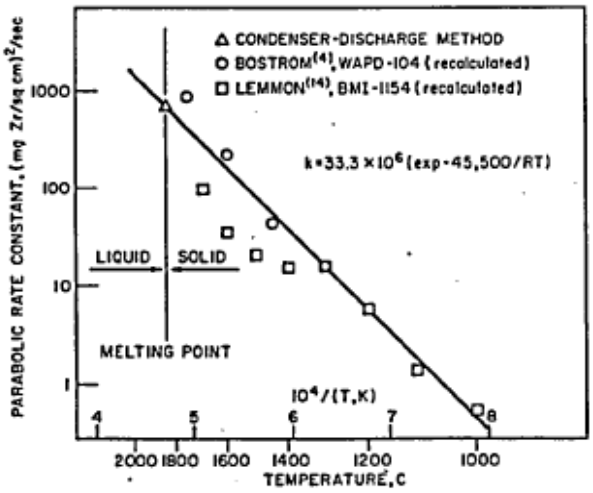
3. The nuclear power industry fiercely defends Baker- Just in its Industry Comments, ML101040678, April 12, 2010, *"The Baker-Just correlation, using the current range of parameter inputs, has been shown to be conservative and adequate to assess Appendix K ECCS performance. Data published since the Baker-Just correlation was developed has clearly demonstrated the conservatism of the correlation above 1800°F"*

4. Contrary to the exceptionally firm consistency between the NEI and NRC appraisals of Baker-Just**,** the pertinent data sets published since the Baker-Just correlation was developed have clearly demonstrated the non-conservatism of the Baker-Just correlation above 1800°F. The NRC has not recognized that investigations that involve heating of single specimens of zirconium alloys in steam do not yield applicable data for the temperature or range of temperatures at which thermal runaway is initiated in LWRs.

5. NRC has apparently never studied Baker-Just (ML050550198) and until April 2010 it did not even have copies of the key references. Figure 16 is copied from page 37 of the Baker-Just report ML050550198.

Figure 16

EFFECT OF TEMPERATURE ON THE ZIRCONIUM-WATER REACTION




Only the Lemmon data includes the pertinent temperature region. The Lemmon report, ML100570218, was not acquired by NRC until April, 2010. Thus, NRC never studied Baker-Just. Figure C-1 is from Lemmon page C-4; the adjacent figure is excerpted from the flow sheet, Figure C-3 on page C-5.

C-4

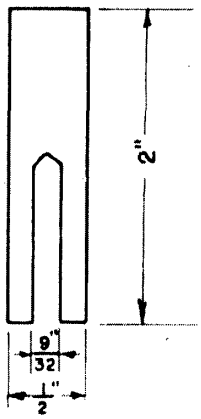
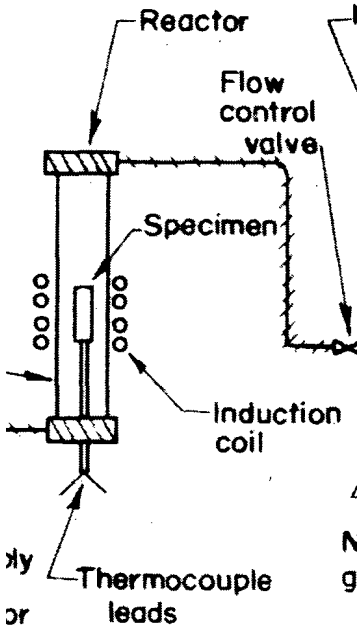



FIGURE C-1. ZIRCALOY 2 SPECIMEN




Lemmon induction heated a zircaloy-2 cylinder, 2” long by 0.5” dia. in steam.

6. It is absurd to license the emergency cooling of tons of zirconium alloy having thousands of square feet of interfacial surface area based on the limited investigations that yielded the Baker-Just equation. Despite this, Appendix K to Part 50--ECCS Evaluation Models, Item 5, specifies that the rate of energy release from the metal/water reaction shall be calculated using the Baker-Just equation and § 50.46 Acceptance Criteria, item (b)(1) specifies the 2200 degrees.

7. Data from multi-rod (assembly) severe fuel damage experiments (*e.g.*, the LOFT LP-FP-2 experiment) show the Baker-Just equation is non-conservative for calculating the temperature at which thermal runaway would occur in the event of a LOCA.

8. Investigations by P. Hofmann et al. at Forschungszentrum Karlsruhe reveal that the Baker-Just equation is non-conservative for calculating the temperature at which thermal runaway will occur in a LOCA. Their report is, Physico-Chemical Behavior of Zircaloy Fuel Rod Cladding Tubes During LWR Severe Accident Reflood, Part I: Experimental results of single rod quench experiments, FZKA 5846, http://bibliothek.fzk.de/zb/berichte/FZKA5846.pdf

On page 5 of 177: *A series of separate-effects tests is being carried out on Zircaloy PWR fuel rod cladding to study the enhanced oxidation which can occur on quenching. In these tests, performed in the QUENCH rig**, single tube specimens** are* **heated by induction** *to a high temperature and then quenched by water or rapidly cooled down by steam injection.*

On gage 12 of 177***: No significant temperature excursion during quenching occurred such as had been observed for example in the quenched (flooded) CORA-bundle tests This absence of any temperature escalation is believed to be due to the high radiative heat losses in the QUENCH rig.***

And in, "CORA Experiments on the Materials Behavior of LWR Fuel Rod Bundles at High Temperatures," NUREG/CP-0119, Vol. 2, Proceedings of the Nineteenth Water Reactor Safety Information Meeting. ML042230460, P. Hofmann et al.

On page 98 of 493: *The critical temperature above which uncontrolled temperature escalation takes place due to the exothermic zirconium/steam reaction crucially depends on the heat loss from the bundle; i.e., on bundle insulation. With the good bundle insulation in the CORA test facility,* ***temperature escalation starts between 1100 and 1200°C (2012 to 2192°F),*** *giving rise to a maximum heating rate of 15 K/sec.*

9. It is amazing that the ACRS has never reviewed Baker-Just in the course of producing its recommendations regarding the initial licensing, the extended licensing and the licensing of power level increases of numerous American light water reactors.

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| Palo Verde, Units 1, 2 and 3 - Updated Final Safety Analysis Report, Revision 14. | ML072250202 | 2007-06-30 |

**PVNGS UPDATED FSAR**

EMERGENCY CORE COOLING SYSTEM

June 2007 6.3-76 Revision 14

6.3.3 PERFORMANCE EVALUATION

6.3.3.1 Introduction and Summary

10 CFR 50.46 provides acceptance criteria for Emergency Core Cooling Systems (ECCS) for light-water nuclear power reactors [Reference 1]. The ECCS performance analyses described in this section demonstrate that the PVNGS ECCS design satisfies these criteria.

The PVNGS ECCS performance analyses encompass a wide range of Reactor Coolant System (RCS) break locations and sizes, including both large and small break Loss-of-Coolant Accident (LOCAs). The limiting break, which results in the closest approach to 10 CFR 50.46 acceptance criterion for peak clad temperature, is a 0.6 DEG/PD (Double-Ended Guillotine in the Reactor Coolant Pump Discharge leg) as noted in UFSAR Section 6.3.3.2. The limiting break, which results in the closest approach to 10 CFR 50.46 acceptance criterion maximum clad oxidation (or local clad oxidation), is a 0.8 DEG/PD as noted in UFSAR Section 6.3.3.2. For these limiting breaks, the PVNGS ECCS design meets the acceptance criteria of 10 CFR 50.46 as follows:

Criterion 1: Peak Cladding Temperature. ". . .The calculated maximum fuel element cladding temperature shall not exceed 2200°F. . . ." For the limiting break, the PVNGS ECCS performance analysis yielded a peak cladding temperature of 2110°F.

**PVNGS UPDATED FSAR**

EMERGENCY CORE COOLING SYSTEM

June 2007 6.3-130 Revision 14

6.3.6 REFERENCES

1. Code of Federal Regulations, Title 10, Part 50, Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors."

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