From: NOSS Philip (AFS) [phil.noss@areva.com]
Sent: Tuesday, September 08, 2009 5:23 PM
To: Staab, Christopher; CLARK Gary (AFS); TEMUS Charles (AFS)
Subject: RE: Query: When would be a good time to have a phone call to discussthe following query (below)

Chris,

Our nucleonics analyst is back tomorrow. Once he looks this over, I will let you know when we would be ready to call.

Philip W. Noss
Licensing Manager
AREVA Federal Services LLC
1102 Broadway Plaza, Suite 300
Tacoma, WA 98402
(253)-552-1321
phil.noss@areva.com

From: Staab, Christopher [mailto:Christopher.Staab@nrc.gov]
Sent: Tuesday, September 08, 2009 2:17 PM
To: CLARK Gary (AFS); NOSS Philip (AFS); TEMUS Charles (AFS)
Subject: Query: When would be a good time to have a phone call to discussthe following query (below)

5.2.2 Radiation Source

Query 5.1 Provide the basis for the additional 25% margin to compensate any potential non-conservative in the ORIGEN2. The proposed additional 25% margin appears to be arbitrary and no basis is provided for the use of this specific value.

On Page 5.4-2 of the SAR, the applicant states: “The reported dose rates are the values computed by MCNP, increased an additional 25%. This additional margin compensates for any potential non-conservatism in the ORIGEN2 program used to generate the source.”

This information is needed pursuant to the requirements of 10 CFR 71.47

The following text provides additional discussion on staff’s basis for Query 5.1 regarding the use of PWR libraries to analyze non-PWR systems (research reactors)

The use of cross-section libraries representative to pressurized water reactor (PWR) configurations to characterize the source terms for research reactors spent fuel is inappropriate, given the large level of dissimilarity between these two types of reactor systems. Notable differences between these two types of reactor systems are:

- fuel characteristics (low enriched uranium with enrichment less than 5 wt% U-235 for PWRs vs. high enriched uranium with enrichment of approximately 90 wt% U-235 for research reactors.

- different fuel matrix (uranium dioxide for PWRs vs. aluminum-based dispersed fuel for research reactors).

- degree of heterogeneity (large size, relatively uniform cores for PWRs vs. small size, highly-heterogeneous cores, both radially and axially for research reactors).

- larger specific power density for research reactors that contribute to the large difference in the neutron flux spectrum (both energy dependence and spatial dependence).

The PWR cross section libraries in ORIGEN were generated, irrespective of the version of the code, using a neutron flux spectrum typical of a PWR (LEU fuel). These libraries were validated for fuels and assemblies typical to PWR (enrichment less than 5 wt% 235U) and their use for fuel with characteristics and designs far from this range would be incorrect. As mentioned in the SCALE 6 manual, page D1.A.11, with respect to generation of libraries for ORIGEN, "Note that extension of the methods and data beyond 5 wt % enrichment has not been widely investigated at ORNL because of a lack of accessible validation data for non-commercial reactor fuels. Such applications of the methodology should be tested carefully and validated. Note that the parameter ranges used as examples are not necessarily appropriate to non-LWR applications, and will need to be modified for different reactor types and fuel designs."

Additional discussion points are provided in the following paragraph.

The argument that the use of ORIGEN 2.2 thermal library is appropriate "because experience at MURR has shown that this library produces conservatives values for the isotopes routinely produced at the reactor" and "measured heat dissipation rates are typically less than half of the value computed by ORIGEN 2.2" is flawed, as it provides no physical or computational basis. If other library (let's say typical of MAGNOX) would provide larger decay heat values than the ones measured at MURR, would that mean that these libraries are appropriate to use for simulations for MURR?

- The improvements in ORIGEN-S have two main components: improvement in computational methodology and improvement in nuclear data libraries. With respect to methodology, as stated on page F7.1.1 of the SCALE6 manual, "the most significant improvement has been the ability to develop and utilize problem-dependent multigroup cross-section data for a burnup simulation process using fuel assembly design information, material compositions, and reactor operating conditions specified by the user." In addition, significant methodology updates concern the neutron source strengths and energy spectra (see section F7.2.8 of SCALE 6 manual), which include neutrons produced from spontaneous fission, (α,n) reactions, and delayed (β−,n) neutron emission. Of particular importance is the treatment for (α,n) production, which varies significantly with the composition of the medium. ORIGEN-S includes three (α,n) source options: (1) a UO2 fuel matrix, (2) a borosilicate glass matrix, and (3) an arbitrary problem-dependent matrix defined by the user input compositions; in the last option, the code determines the matrix compositions from the input. With respect to improvement in nuclear data (details on page F6.1.1 in SCALE 6 manual) "significant advancements have been made in the development of improved nuclear decay data, cross-section data, and photon yield data." With release of SCALE 5 the majority of the nuclear decay and cross-section data were upgraded to ENDF/B-VI. Explicitly represented fission product yields are included for 30 actinides with neutron-induced fission yields in ENDF/B-VI, compared to yields data for up to five actinides in previous libraries. "The addition of neutron-induced fission yields for most of the fissionable actinides with evaluated yields greatly increases the versatility and range of application of the code, particularly for advanced fuel design and transmutation studies." (page F6.1.1)

- The fact that the use of ORIGEN 2.2 PWRS library "generates actinides and fission product concentrations that, when input to a criticality program, result in computed reactivities to within 0.2% of measured reactivities" is not a measure of accuracy in the predicted MITR-II fuel element depletion and fission product buildup. The uncertainty in reactivity in the criticality calculation includes other components; in addition to the uncertainty in the isotopic composition (what guarantees that there is not a cancellation of errors?). Therefore, this cannot serve as a measure for the uncertainty in isotopic composition data. The isotopics are likely seriously in error and the criticality analogy is without basis for this application.

- ATR states: "For ATR fuel, ORIGEN2.1 and ORIGEN 2.2 generate the same source within 0.1%. Therefore, ORIGEN 2.1 results are used, even though this is an older version of the program". This argument refers only to a difference between two calculated values with two different codes. It does not provide a basis for the correctness of the use of the associated libraries.

It seems that in this case, the radiation source was calculated with ORIGEN2 and the data used further in MCNP for dose calculation. The proposed additional 25% margin is, or appears to be arbitrary and no basis is provided for the use of this specific value. Note that a 25% penalty is completely arbitrary and likely adequate for the dose rate dominated by Cs-137 or other fission products generated directly by fission; however, this has not been shown by the applicant. In addition, it is not entirely clear what the specifics of the application are (cooling times, etc.) that could impact accuracy