AREVA

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Response to a Request for Additional Information Regarding Evaluation of a Potential Safety Significant Issue Pursuant to 10CFR21(a)(2)

Ref. 1: Letter, James F. Mallay (Framatome ANP) to Document Control Desk (NRC), "Evaluation of a Potential Safety Significant Issue Pursuant to 10CFR21(a)(2)," NRC:04:063, November 23, 2004.

Ref. 2: Letter, Ronnie L. Gardner (AREVA NP, Inc.) to Document Control Desk (NRC), "Response to a Request for Additional Information Regarding Evaluation of a Potential Safety Significant Issue Pursuant to 10CFR21(a)(2)," NRC:06:007, February 10, 2006.

AREVA NP, Inc. (AREVA NP) initiated a discovery on September 27, 2004, as a result of the determination that thermal aging and embrittlement of the control rod drive mechanism (CRDM) leadscrew male coupling (also known as the bayonet) on the B&W-designed pressurized water reactor (PWR) plants could lead to failure of the male couplings during operation (Reference 1). AREVA NP determined that the situation constituted a deviation under the provisions of 10CFR21 for the B&W-designed plants TMI-1, ANO-1, CR-3 and Davis Besse.

A request for additional information was provided by the NRC in e-mails on November 30, 2005 and December 9, 2005. The questions and responses to the request were provided in Reference 2.

A second request for additional information was provided by the NRC in an e-mail on March 1, 2006. The questions and responses to the request are provided in Attachment A.

Sincerely,

Ronnie L. Gardner

Ronnie L. Gardner, Manager
Site Operations and Regulatory Affairs
AREVA NP

Enclosure

cc: G. S. Shukla
O. Tabatabai
Project 728

IE19

AREVA NP INC.
An AREVA and Siemens company

3315 Old Forest Road, P.O. Box 10935 Lynchburg, VA 24506-0935
Tel.: 434 832 3000 - Fax: 434 832 3840 - www.areva.com

Attachment A

REQUEST FOR ADDITIONAL INFORMATION
Regarding Evaluation of a Potential Safety Significant Issue Pursuant to 10CFR21(a)(2)

Question 1: *Integrated Control System - Is it a safety grade system? The current design bases for this analysis takes credit for the Integrated Control System (ICS) and Control Rod Drive Control System (CRDCS) interaction to inhibit the control rod pull function. Confirm the validity of this analysis?*

Response 1:

No, the ICS is not a safety grade system. Assuming that the leadscrew and control rod assembly are still coupled, the analysis as presented in the FSAR is still valid. The AREVA NP philosophy is that if a control system aggravates the response to a postulated accident, the effects of the control system are modeled. If a control system mitigates the response to a postulated accident, the system is assumed to do nothing. When evaluating the effects of the control system on a particular accident, all of the functions of the control system are considered.

A specific discussion of the ICS functions relative to the dropped rod accident was presented to the NRC in Section A.5 of BAW-10193PA. That report states that for a dropped rod assembly, the core power and average temperature will decrease. (*The plant operates with CRA Banks 1 through 6 being fully-withdrawn and Bank 7 90% withdrawn.*) The ICS would normally pull rods (Bank 7) in response to the decrease in reactor coolant system temperature. However, there is a rod pull inhibit that would prohibit the rods from being withdrawn. In the case of an asymmetric control rod, the ICS was designed to initiate a power runback to 60% rated thermal power at a rate of 30% per minute. This control bank(s) would be inserted and the core power and the power peaks would be reduced. Consequently, for the dropped rod control rod accident, the ICS is not modeled and therefore does not perform any function.

If the ICS does withdraw the control rods to maintain core power and reactor coolant system temperature, a maximum power level of 103% is programmed in the ICS. Depending on the location and worth of the dropped control rod and the time in life, the core power will decrease due to the inserted worth of the dropped rod. If the core power decreases more than 5% from the power demand, then the ICS will enter a cross limit. The power demand will be reset to the core power and the control rods will not be withdrawn. One final piece of information is that the plant technical specifications require that if the plant is operating with an inoperable CRA (either misaligned or dropped), then in order to continue power operation the plant must reduce thermal power to 60% RTP and perform a series of checks including verification of shutdown margin and power peaking factors.

Question 2: *How an operator would know that control rod has dropped? Where is the rod position indicator located? Is there any rod bottom signal indicating a rod has dropped?*

Response 2:

The control rod elevation is based on the position of the leadscrew which is indicated by reed switches with indicator lights on a panel in the control room. There are two circuits, each with 36 switches. There is a 4-inch spacing between the 36 switches on a given circuit, but the circuits are offset by two inches. There are out-limit and in-limit switches. In addition, there are "zone reference lights" that activate when the control rod is at certain positions, including fully inserted, fully withdrawn and at 25% intervals of the travel.

If a control rod is positioned more than 9 inches from the position of its group, the control rod is considered misaligned, which activates alarms in the control room. The on-line computer may also provide a printed alarm. There is no continuous alarm on power peaking. However, the incore detector system is capable of generating incore flux maps at six minute intervals; the flux maps can be examined (1) to determine core location of an inserted control rod and (2) to generate a 3-D map of power peaking factors or linear heat generation rates so that margin to peaking limits can be determined. A normal surveillance interval would be every 31 EFPD but most plants generate a flux map on a weekly basis and compare measured peaking to the peaking limits. If a misaligned or dropped control rod is identified, the plant technical specifications requires that the core power level be reduced to $\leq$ 60% rated thermal power within 2 hours and that a series of checks be performed to verify shutdown margin and power peaking factors.

Question 3: *It is stated, "Hence, even the male couplings on the recently installed replacement CRDMs are expected to become embrittled after approximately 5 years of operation." What are the inspection procedures and surveillance requirements in place at the refueling outage to prevent CRDM male couplings failures? Why do you use the same material for the replacement of CRDM male couplings at Oconee Units, ANO 1 and Crystal River Unit 3?*

Response 3:

For the refueling outages, the utilities have been advised to be aware of fracture to the male coupling tangs as a result of the 2001 fractures at ONS3, and to change coupling and decoupling procedures to avoid using excessive force against the stop pin. Beyond this recommended procedure change, AREVA NP does not maintain or control copies of the site-specific inspection procedures and surveillance requirements that are in place during refueling outages to prevent failures of the CRDM male couplings.

The same Type 17-4 PH material was used for the male couplings because the replacement CRDMs were fabricated in the late 1970s and early 1980s for B&W-designed PWRs that were subsequently either cancelled or uncommissioned. That is, the replacement CRDMs are the same vintage as the original CRDMs, except the replacement CRDM had never been used. These replacement CRDMs were fabricated prior to the industry becoming aware of the male coupling aging embrittlement phenomenon.

Question 4: *It is stated, "it is very unlikely that an operating plant would experience simultaneous complete failure of two or more male couplings." How do you know?*

Response 4:

Over the operating history of the B&W-designed plants, there have been no dropped control rods due to male coupling (including tangs) failures during plant operation. Both dye-penetrant and visual inspections of several of the male couplings were performed during the B&WOG CRDM Life Extension (LIFEX) program. These inspections did not identify any significant indications of cracks, pre-existing flaws, or sharp notches. The only item of note was the failure of a single tang on two Oconee bayonet couplings, but even then complete failure of all four tangs would still not have resulted in a dropped control rod assembly. Since the coupling/uncoupling procedures were revised, there have been no additional failures of the tangs. Recently, the CRDM drives at ANO-1, CR-3 and all three Oconee plants have been replaced during the past 5 years. In addition, the load on the male coupling is low, especially during normal operation. The only significant loads are applied when the rods are being initially withdrawn from the core prior to startup. Therefore, it was concluded that simultaneous failure of two or more male couplings is unlikely.

Question 5: *Discuss in detail operability and safety assessment report addressing both the material aspects and expected plant response to a failure of the leadscrew male coupling. Provide a copy of this report.*

Response 5:

AREVA NP prepared a short-term operability assessment for the B&W-designed operating plants. The purpose of that report was to provide input to those Utilities in their plant-specific disposition of this Condition Report. A material assessment as well as an expected plant response/safety assessment was included in the report. This report however, may not necessarily have been used by the Licensee for their plant-specific disposition of the CR. Nonetheless, the following information was extracted from the AREVA NP report.

<u>Material Assessment Synopsis</u>

As part of the manufacturing process, a dye-penetrant examination was performed on all of the CRDM leadscrew bayonets and no reportable flaw indications were found. Examinations have also been performed on a number of the in-service CRDMs through the first 20 to 25 years of plant operation. Visual examinations have been performed on nine CRDM leadscrew bayonets, with one bayonet having been visually examined at three different times in-service and one bayonet having been visually examined at two different times in-service, for a total of 13 visual examinations. In addition, the most recent examinations also included dye-penetrant examination of six of the bayonets, with one bayonet having been dye-penetrant examined twice, for a total of seven dye-penetrant examinations. No reportable flaw indications, either with visual or dye-penetrant examination have been noted other than instances of upset metal at the bayonet tangs. In addition, no failures of the bayonets or the tangs were identified until 2001.

In 2001, there were two instances of broken tangs. The examination of the tang failures concluded that the likely cause was single and/or multiple impact overload events on the tang that had become embrittled during service. The failure analyses concluded that the observed distortion and pre-existing crack were due to repeated hitting against the CRA stop pin. The coupling procedures have been revised and since that time no additional broken tangs have been identified.

Two in-service CRDMs were recently removed from Arkansas Nuclear One Unit 1 and were sent to AREVA NP to examine them for wear. As part of that examination, both a visual and dye-penetrant examination of the bayonets were performed. The examination concluded that there was no significant wear on the parts and more importantly for this evaluation, no reportable flaw indications were found on the bayonet couplings.

Although the Type 17-4 PH material may be susceptible to embrittlement, the historical evidence suggests that the loads are not sufficient to cause the bayonet to fail. While this small sampling may not be adequate to close out the issue, it does provide reasonable assurance that a failure is not imminent.

<u>Expected Plant Response</u>

During a large portion of power operation, the plant will be automatically controlled by the ICS with the core power demand set at or near the rated power level for the plant. The ICS will control main feedwater flow and steam pressure to maintain the RCS average temperature and the electrical output. If a control rod is dropped, the core power, RCS average temperature and RCS pressure will decrease as will the electrical output. If the worth of the dropped control rod is very large, greater than approximately 0.25% Δk/k, the reactor will be protected by an automatic trip signal on low RCS pressure. A reactor trip will terminate the transient.

If the dropped rod transient results in a decrease in measured core power that is greater than 5 percent (typical), a cross-limit will be reached and the ICS will enter the track mode. In track, the core power demand will be based on the actual electrical output and the integral for the ICS T_{ave} controller will be blocked for a period of time. Electrical output will decrease due to the reactor power reduction caused by the dropped control rod. A number of alarms will sound. Since the electrical output will have decreased, demand for reactor power will decrease and control rod movement will be minimal by the ICS. Therefore, the resulting transient would be enveloped by the accident analysis. Although the rod position indication will show that all rods are aligned (*with a failure of the bayonet coupling*), an evaluation of the incore detector data would be sufficient to identify that there was a dropped rod. Also, an increase in the quadrant power tilt would be expected for most dropped rods.

Cycle core design data suggest that the typical worth of a dropped rod is between 0.05 %Δk/k and 0.12 %Δk/k, with a design limit of 0.20%Δk/k. As a result, it is not likely that the core power will decrease more than 5 percent. The ICS will withdraw the control rods to maintain power demand and RCS average temperature. Note that the ICS temperature controller integrates the difference between the actual T_{ave} and the desired T_{ave} and will be slower to react than for the power mismatch.

The amount of positive reactivity that can be added is limited because the plant members of the B&WOG have adopted the cycle-specific rod operating recommendations. This recommendation requires that the regulating rod group (Bank 7) be at least 90% withdrawn through a majority of the fuel cycle. Crediting a more realistic worth for a single dropped rod, the expected decrease in RCS temperature will be much smaller (*i.e., the power measurement error will be reduced*) and coupled with the CRAs that are at least 90% withdrawn will greatly reduce the amount of power increase as the ICS withdraws the rods. Therefore, the low worth of an individual control rod, the ICS power limit, and the rod operating recommendations would result in a transient that would be bounded by the spectrum of reactivity insertion rates that are considered for the rod withdrawal at power FSAR accident analyses.

Question 6: *In order for the staff to have reasonable assurance that the plant can operate safely in response to this accident using safety related components, perform a new analysis without crediting Integrated Control System and using a direct trip function such as high heat flux hot channel factor, low DNBR or high neutron flux which represents a rod dropped accident satisfying GDC 28 / SRP Section 15.4.8 acceptance criteria. Provide the results for the staff review. Also perform an analysis postulating multiple rods dropped accident and provide the results for the staff review.*

Response 6:

Subsequent to the assessment provided above, preliminary calculations have been performed that simulate the action of the ICS to withdraw the control rods following a single dropped control rod due to a complete failure of the bayonet coupling. A mini-spectrum of cases with a worth of the dropped rod up to and including 0.22 %Δk/k were modeled. The ICS power limit (*of 103%*) was not credited. The remaining control rods were assumed to be at their insertion limit. The integral worth, plus uncertainty, that was available following the dropped control rod was withdrawn at a speed of 30 inches a minute. With the reactor at full power, the preliminary results indicate that, depending on time-in core life and the integral rod worth available, the reactor will trip on high flux (over-power), trip on high RCS pressure, or reach a new steady state operating condition at a power level less than the trip setpoint. In either case, from an overall system response perspective, the results will be bounded by the rod withdrawal at power FSAR accident.

Only a preliminary assessment of the power peaking due to the combined effect of the dropped control rod and ICS rod withdrawal has been completed at this time. The results indicate that sufficient margin exists within the normal operating limits for axial power imbalance and control rod insertion to bound the consequences of this transient.

The design basis for the operating B&W-designed plants is for a single dropped control rod only. Consideration for multiple dropped rods is beyond the design basis for which no analysis has been performed. As presented above, no indications have been found on the existing bayonet couplings and even one failure is not expected. Further, if the total worth of the failed control rods exceeds approximately 0.25 %Δk/k, the reactor would trip on low reactor coolant system pressure.

Question 7*: In its response to the staff's RAIs, AREVA did not clearly identify the root cause of the failure in the CRDM male couplings at the Oconee Unit. It is not clear how the crack initiated in the first place. It can be concluded that synergistic effects of loads applied during maintenance activity (torquing and detorquing) and thermal embrittlement of 17-4 PH material could have caused the failure. The staff did not review the* *licensee's evaluation report addressing this issue, and this evaluation can provide additional information as to the corrective actions that licensees need to take to mitigate this aging degradation.*

Response 7:

AREVA NP's response to the NRC 1st set Question 1F (February 2006) stated that "...The failures were concluded to be due to single or multiple impact and bending loads to the Type 17-4 PH (H-1100) male coupling tangs, which had become embrittled from exposure to operating temperature, during the CRDM coupling and uncoupling process." This was referenced to the following 2003 conference paper co-authored by AREVA NP and Duke Power. Therefore, the crack initiation and sub-critical crack propagation mechanism implied by the authors is low cycle fatigue to the aging embrittled male coupling material.

> H. Xu, S. Fyfitch, Charles, R. Frye, and David E. Whitaker, "Fracture of Type 17-4PH CRDM Leadscrew Male Coupling Tangs," the 11th International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, ANS, Skamania Lodge in Stevenson, WA (2003).

Aside from the information described in the above jointly authored conference paper, AREVA NP does not have additional information on the 2001 ONS-3 male coupling tang fracture events.

The proper terms for the maintenance activity related to the tang fracture are "coupling and de-coupling". The impact load is from the male coupling tang hitting the stop pin inside the female coupling to verify the correct positioning of the male coupling relative to the female coupling. Other than during this position verification procedure, the tangs are not in contact with the stop pin; hence, there is no tangential torquing load on the tangs from the stop pin at any other time. The terms "torquing and detorquing" are therefore incorrect.

Question 8: *What type of aging monitoring program is the licensee adopting at the Oconee Unit?*

Response 8:

AREVA NP does not monitor or control the inspection and surveillance program for the male couplings at the Oconee Units operated by Duke Energy. Duke Energy's response to Question 8 provided to AREVA NP is as follows:

<u>Background</u>

While coupling the CRDMs leadscrews to the control rods during an unscheduled outage in the spring of 2001, and the scheduled refueling outage in the fall of 2001, one lead screw during each of the two outages rotated further than normal. This condition was caused in both cases by the lower portion of the one of four bayonet (lower end of the leadscrew) tangs being broken off. The ultimate conclusion from metallurgical examinations was that these failures were more likely the result of single or multiple impact overload events at room temperature to a material that had lost ductility due to thermal embrittlement, although fatigue cannot be ruled out completely.

The lower portion of the tang, which impacts a hard stop inside the control rod, is reduced in width from the rest of the tang and has a sharp radius where it ties in with the body of bayonet. This causes it to act as though it were a shear pin allowing breaking of only the lower portion of the tang. The tang impacts the hard stop only during coupling and uncoupling. Once the CRDMs are coupled the lead screws are kept from rotating by a key located at the top of the CRDM. CRDM orientation is maintained on its mounting flange by a pin, unless the leadscrew is removed from the CRDM (an infrequent occurrence), the same tangs will always impact the hard stop during coupling and uncoupling.

The control rods are supported by the wider top surface of the four tangs, providing redundant support. Only the lower, narrower portion of a single tang was found broken on each leadscrew. The remaining upper, wider portions were still more than sufficient to support the weight of the control rods.

Corrective Actions

The two leadscrews found with damage were replaced upon discovery of the condition.

Between 1999 and the spring of 2003 all Oconee CRDMs, including their leadscrews and attached bayonets, were replaced with similar components which also have 17-4PH bayonets.

The process for coupling and uncoupling the leadscrews has been modified by training the workers to be aware of the possibility of tang damage, to minimize the impact of the tang against the hard stop during the coupling process, and to identify any over rotation that occurs.