April 5, 2011

Dr. Said Abdel-Khalik, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: RESPONSE TO ACRS RECOMMENDATIONS ON DRAFT FINAL REGULATORY GUIDES 1.34, 1.43, 1.44, AND 1.50

REFERENCE: Letter to Mr. R. W. Borchardt, Executive Director for Operations, from Said Abdel-Khalik, Advisory Committee on Reactor Safeguards (ACRS) Chairman, dated February 24, 2011, Subject: Draft Final Regulatory Guides 1.34, 1.43, 1.44, and 1.50, ADAMS Accession No. ML110450579.

Dear Dr. Abdel-Khalik:

The referenced letter from the ACRS contains recommendations pertaining to the issuance of Draft Final Revision 1 to Regulatory Guides (RGs) 1.34, “Control of Electroslag Weld Properties,” 1.43, “Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components;” 1.44, “Control of the Processing and Use of Stainless Steel;” and 1.50, “Control of the Preheat Temperature for Welding of Low-Alloy Steel.” These RGs were reviewed by the ACRS during the 580th meeting of the ACRS on February 10-12, 2011, during which time the Nuclear Regulatory Commission (NRC) staff presented its proposed changes to the RGs and the associated technical justification. The ACRS Materials, Metallurgy, and Reactor Fuels Subcommittee also reviewed this guidance during a meeting on October 21, 2010. This memorandum contains the NRC staff’s responses to the ACRS recommendations.

ACRS RECOMMENDATIONS

The ACRS recommendations on the four subject RGs were as follows:

1. Draft Final Revision 1 to RGs 1.34, 1.43, and 1.50 should be issued.

2. Draft Final Revision 1 to RG 1.44 should not be issued until the following changes are made:

    a. The language proposed by the NRC staff during the February 10-12, 2011, meeting should be incorporated into the guide to address concerns on the use of standard grade stainless steels and the description of pressurized-water reactor (PWR) water chemistry.

    b. Guidance should be added to address the deleterious effects of coldwork and post-weld grinding on intergranular stress corrosion cracking (IGSCC) and irradiation-assisted stress corrosion cracking (IASCC) susceptibility of welded American Iron and Steel Institute Type 300 stainless steel components. The ACRS letter also notes that IASCC is not well understood and should be noted in future revisions of the RG.

DISCUSSION

Based on Recommendation 1, the NRC staff published RGs 1.34, 1.43, and 1.50 in the *Federal Register* on March 14, 2011.

Regarding the recommended changes to RG 1.44, the NRC staff agrees with the ACRS recommendations and has implemented the following revisions to the RG:

1. The language proposed by the staff during the February 11, 2011, presentation to the ACRS has been incorporated into the guide to address concerns on the use of standard grade stainless steels and the description of PWR water chemistry.

2. Guidance has been added to the RG to address the deleterious effects of coldwork and post-weld grinding.

3. No additional guidance has been added to the RG regarding welding of stainless steels that are potentially susceptible to IASCC. The ACRS letter correctly states that the IASCC degradation mechanism is not as well understood as IGSCC. The NRC staff agrees that the IASCC degradation mechanism is an ongoing operating experience issue that is currently being addressed through industry initiative programs such as the Boiling Water Reactor Vessel Internals Program (BWRVIP) and the Materials Reliability Program (MRP). Therefore, as more information is obtained, the NRC staff will reassess this topic and consider whether information concerning IASCC susceptibility should be addressed in future revisions of the RG.

Based on the foregoing, the following revisions have been made to RG 1.44. Note that ***boldface italics*** reflect changes made in response to Item 1 above and <u>underlined</u> items reflect changes made in response to Item 2 above.

<u>Section B, "Discussion," Paragraphs #1 and #2:</u>

> Control of the application and processing of stainless steel to avoid severe sensitization is needed to diminish the numerous occurrences of intergranular stress-corrosion cracking in sensitized stainless steel components of nuclear reactors. Test data demonstrate that sensitized stainless steel is significantly more susceptible to intergranular stress-corrosion cracking than is nonsensitized (solution heat-treated) stainless steel. Of specific concern in this guide are the unstabilized austenitic stainless steels, which include American Iron and Steel Institute (AISI) Types 304 and 316, normally used for components of the reactor coolant system and other safety-related systems. ***Low carbon grade stainless steel (i.e., 304L and 316L) should be used where the material comes in contact with the reactor coolant.*** This guide does not cover stabilized stainless steels (e.g., AISI Types 321 and 347), which also provide some protection against sensitization.
>
> Process controls should be exercised in accordance with good manufacturing/welding practices and knowledge gained from operating experience during all stages of component manufacturing and reactor construction to minimize exposure of stainless steel to contaminants that could lead to stress-corrosion cracking. <u>As described in Section 5.2.3 of NUREG-0800, manufacturing processes should control cold-working and abrasive work such as grinding to minimize the amount of cold-working because excessive cold-working in austenitic stainless steels can increase their susceptibility to stress corrosion cracking.</u>

Because some degree of material contamination is inevitable during these operations, halogens and halogen-bearing compounds (e.g., die lubricants, marking compounds, and masking tape) should be avoided to the degree practical.

Section B, "Discussion," Paragraph #8:

Controls should be maintained on the chemistry of the reactor coolant and auxiliary systems fluids to which the material is exposed. Chloride and fluoride ion concentrations should be specified to be less than 0.15 parts per million at all times. Dissolved oxygen concentrations should be maintained below ***the limiting value of*** 0.10 parts per million during periods when the material is at elevated temperatures. If the oxygen content exceeds this level, such as in boiling-water reactor coolants during normal operation, sensitization of material that is welded without subsequent solution heat treatment should be further controlled by limiting the carbon level in the material to 0.03 percent. Carbon level control is not needed for weld metal and castings with duplex structures because these product forms with normal carbon levels have demonstrated adequate resistance to intergranular attack. Carbon level control may not be required for piping if its diameters are sufficiently small (e.g., instrument lines and control rod drive hydraulic systems) that it could withstand a single failure without an accompanying loss-of-coolant accident as defined in Appendix A to 10 CFR Part 50.

Section C, "Regulatory Position," Position #4.a:

a. material exposed to pressurized water reactor coolant that has a controlled concentration of ***typically less than 0.020 parts per million, with a limiting value of*** 0.10 parts per million dissolved oxygen at all temperatures above 90 degrees C (200 degrees F) during normal operation; or

Because the revisions to RG 1.44 are minor in nature and clarified the language in the Draft Guide that was issued for public comment, no additional public review of the RG was considered necessary.

CONCLUSION

Based on the referenced ACRS letter dated February 24, 2011, RGs 1.34, 1.43, and 1.50 have been issued. Revisions have been made to RG 1.44 based on ACRS recommendations and the staff expects to publish the RG within the month.

We appreciate the recommendations and comments provided by the ACRS.

Sincerely,

/RA/

R. W. Borchardt
Executive Director
for Operations

cc: Chairman Jaczko
Commissioner Svinicki
Commissioner Apostolakis
Commissioner Magwood
Commissioner Ostendorff
SECY

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