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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 5, 1996

Ms. Rosalie Hillis
Quality Manager
Crane Valves Nuclear Operations
104 North Chicago St.
Joliet, IL 60431

Dear Ms. Hillis:

In a letter dated May 16, 1994, Crane Valves Nuclear Operations (Crane) advised the U.S. Nuclear Regulatory Commission (NRC), in accordance with the reporting requirements of Part 21 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 21), that Crane was undertaking an evaluation to determine if data supplied in the mid-1980s as Crane Aloyco "OTC" reports had been used by nuclear utilities to set torque switches in motor-operated valves (MOVs). The May 1994 letter stated that the issue involved use of data beyond the intended scope and not the analytical methods. The issue was said to have surfaced as a result of the review of revised OTC reports supplied to Florida Power Corporation for MOV at the Crystal River nuclear power plant.

In a letter dated January 12, 1995, Crane notified the NRC that the evaluation had been completed. The January 1995 letter states that, in the 1980s, Crane had used a model to predict thrust required to operate valves in essential safety systems for its nuclear power plant customers in response to an NRC generic communication in 1985 (Bulletin 85-03). Beginning in 1993, nuclear utilities requested Crane to perform support analyses for specified design conditions which included parameters not analyzed in the original model. The January 1995 letter states that the potential Part 21 concern arose because the results generated from the current model differed from those of the original model.

The January 1995 letter states that Crane reviewed the original purchase orders to confirm that the analytical information supplied by Crane matched the utility's request. The Crane model used in the 1980s is said to have considered design pressure, differential pressure, ambient temperature, operator thrust, and yield stress. Since late 1993, nuclear utilities have requested Crane to perform analyses with specified design-basis conditions including extended structure mass, extended structure center-of-gravity, ASME and AISC allowable stress, and dynamic loads. In performing these more-recent analyses, Crane has used a model that also considers bending moments and torsion loads. The Crane analyses using the current model are said to be able to identify the weak link in the valve-motor operator assembly as a function of design-basis conditions.

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In the January 1995 letter, Crane states that the nuclear utilities were supplied with the analyses they requested. Crane states that it has been in contact with many utilities to provide support for them to better understand, or predict, the performance of safety-related MOVs. Crane also has provided its customers with an analysis using the current model addressing the additional factors, when requested. As a result, Crane believes that the data supplied in prior analyses did not constitute a reportable item under 10 CFR Part 21.

The NRC staff is reviewing the question of a potential generic concern regarding the capability of MOVs sized and set based on the original Crane model. To assist the NRC staff in completing its review, we request that you indicate, to the best of your knowledge, whether any MOVs considered acceptable under the original model were subsequently determined to be unacceptable when using the current model. If so, please indicate the applicable facilities and plant systems, and summarize the follow-up action by Crane and the specific utility. The staff requests that you provide an example of the typical concern identified when the current model was used to analyze the capability of a previously analyzed MOV. Finally, the staff requests that you indicate the method by which Crane has ensured that its customers have identified previously analyzed MOVs that might be deemed unacceptable in their sizing or setup based on the current model.

If you have any questions on this request, please contact Thomas G. Scarbrough of my staff at 301-415-2794.

Sincerely,

/S/

Richard H. Wessman, Chief
Mechanical Engineering Branch
Division of Engineering
Office of Nuclear Reactor Regulation

cc: Lyle Parnell, Crane Valves