Responses to Information Requests from Representative Edward J. Markey
Letter of December 1, 2011

1. Please provide copies of any licensee voluntary responses to NRC's 1993 and 2007 Information Notices.

Information Notices (INs) such as IN 93-68, "Failure of Pump Shaft Coupling Caused by Temper Embrittlement during Manufacture," and IN 2007-05, "Vertical Deep Draft Pump Shaft and Coupling Failure," provide information on emergent issues in nuclear plants that do not pose immediate safety concerns. Licensees are expected to review the information in the INs for applicability to their facilities and consider actions, as appropriate, to avoid similar problems, but are not required to respond to the NRC. In response to this information request, the NRC staff conducted an extensive search of its Agencywide Documents Access and Management System (ADAMS) for any voluntary responses that may have been submitted by licensees in response to these INs. This search did not reveal any responses.

2. Please provide a list of all U.S. nuclear power plants currently using 410SS and 416SS components.

Type 410 and 416 stainless steels are used when a combination of strength and moderate corrosion resistance are required. The NRC does not track the use of these materials. However, given the extensive use of these alloys in applications such as bolting and valve components, it is likely that these steels are used at many, if not all U.S. nuclear power plants.

3. Please provide a list of all the known uses of 410SS and 416SS steels in nuclear power plants.

The NRC does not maintain a list of all applications of each alloy used at nuclear power plants. Type 410 and similar martensitic stainless steels are very common materials that are used in a wide range of applications. These alloys are commonly used for applications such as valve stems and valve hardware, including disks, hinge pins, plugs and seats; pump shafts and hardware such as internal bolts, wear rings, sleeves, impellers, and couplings; internals bolting in pumps and valves; external pressure boundary bolting; and control rod drive mechanism hardware. These alloys, when properly heat treated for the intended service, perform acceptably in many nuclear power plant environments.

4. In April 2005, NRC published a review of the performance of steel alloy 600 in nuclear power plants across the fleet. The impetus for collecting information on alloy 600 cracking was in part the discovery in March 2002 of vessel head penetration flaws, leaks, and pressure boundary corrosion at the Davis-Besse plant. This review found that alloy 600 and its associated welds are susceptible to crack nucleation and growth in a wide range of applications. NRC staff concluded that additional inspections beyond one-time inspections were warranted, and industry developed inspection and evaluation guidelines to manage degradation. Will you initiate a similar review for the 410SS and 416SS steels used in pump components? If not, why not?

The cited NRC review of Alloy 600 material was documented in NUREG-1823 "U.S. Plant Experience with Alloy 600 Cracking and Boric Acid Corrosion of Light-Water Reactor Pressure Vessel Materials." The purpose of this NUREG was to relate operating experience from foreign and domestic nuclear power plants regarding Alloy 600 cracking, provide the results of an

analysis of the Alloy 600 cracking susceptibility model for vessel head penetration nozzles, and provide related information on corrosion of pressure boundary materials in boric acid solutions. This document identified that Alloy 600 and its weld metals (Alloys 82 and 182) were susceptible to crack formation and growth when exposed to pressurized water reactor coolant. The agency's review and analysis, and the ensuing industry response, were necessary because cracking of Alloy 600/82/182 reactor coolant pressure boundary components and/or welds could directly challenge the reactor coolant pressure boundary safety function.

In contrast, type 410 and 416 stainless steel materials are not typically used for reactor coolant pressure boundary applications. Therefore, the consequences of a failure of a component constructed from type 410 or 416 stainless steel are different than the consequences of an Alloy 600 failure. While the failure of a type 410 or 416 subcomponent may render a pump or a valve inoperable, it will not compromise pressure boundary integrity. Based on this difference, the NRC does not currently believe that a response similar to that initiated for Alloy 600 cracking is warranted for the observed cracking of Type 410 and 416 stainless steels. The NRC will continue to monitor developments regarding this issue as described in our response to information request 5, below.

5. What regulatory actions will the Commission undertake in order to assess, require licensee reporting and inspection of, and address the IGSCC problems involved in 410SS and 416SS pump components? If no such actions are planned, why not?

As described in the response to the previous question, the NRC has not identified a need for an immediate response to the observed cracking of type 410 and 416 stainless steels such as that initiated for Alloy 600 cracking. The need for longer term regulatory changes and/or the issuance of additional generic communications remains under review. This review consists of two phases.

The first phase of this review is to determine the adequacy of current guidance on the subject. Current guidance includes NRC Bulletin 89-02, "Stress Corrosion Cracking of High-Hardness Type 410 Stainless Steel Internal Preloaded Bolting in Anchor Darling Model S350W Swing Check Valves or Valves of Similar Design," and NRC Information Notices 93-68 and 2007-05. NRC Bulletin 89-02 addresses failures of type 410 stainless steel bolting in check valves exposed to reactor coolant. NRC Information Notices 93-68 and 2007-05 address failures of pump shaft couplings exposed to "raw water" environments such as rivers or lakes. Based on recent events, it appears that the INs were effective in alerting licensees and component suppliers to challenges with type 410 and 416 stainless steel, e.g., heat treatment of components. However, as noted in your letter, a few licensees have still had issues associated with these materials.

The second phase of this review will consider recent operating experience, e.g., the 2011 event at Palisades, in conjunction with historical events. This will occur once the NRC's evaluation of the Palisades event is complete. The NRC has developed a number of programs to evaluate emerging issues and to determine whether action is merited. Specifically, the NRC has developed programs that evaluate failures of safety-related systems and components, including the Operating Experience Program, the Generic Safety Issue Program, and the Performance Assessment Program. The NRC intends to allow our established programs and processes to function as designed to determine the need for, and extent of, any further actions concerning cracking of types 410 and 416 stainless steels.