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POST-TENSIONED PRESTRESSING SYSTEMS FOR CONCRETE REACTOR VESSELS AND CONTAINMENTS

A. INTRODUCTION

General Design Criterion 1, "Quality Standards and Records" of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires, in part, that structures, systems, and components important to safety be designed, fabricated, and erected to quality standards commensurate with the importance of the safety functions to be performed. This guide identifies the post-tensioned prestressing systems that have been reviewed and approved by the NRC staff for use in concrete reactor vessels and containments and also describes qualifications acceptable to the NRC staff for new post-tensioned prestressing systems.

B. DISCUSSION

A post-tensioned prestressing system is composed of a prestressing tendon combined with a method of stressing and anchoring the tendon to the hardened concrete. The word "system" is commonly associated with the different proprietary post-tensioned prestressing systems on the market and is understood to include the type of tendon, anchorage device, and stressing equipment associated with a given system.

It is not practical to discuss the details of all of the many post-tensioned prestressing systems available in the United States. Moreover, new post-tensioned prestressing systems are being developed, and existing ones are being modified. For these reasons, the descriptions in this guide are limited to systems listed in Table A, all of which have been used or proposed for use.

Some examples of use are presented in order to identify more specifically the system being discussed and to provide a reference to some plants for which the systems in Table A have been proposed or approved. The examples cited are not intended to indicate any restriction or preference in size of the tendon for a given system. Nor is this guide intended to discourage the development of refinements of current systems or the development of new prestressing systems or concepts.

The qualifications that a post-tensioned prestressing system should meet in order to be acceptable to the NRC staff are identified in the regulatory position. Rock anchorage systems are not covered by this guide.

Types of Systems

The type of tendon selected usually dictates the choice of stressing equipment and also affects the choice of end anchorages.

Basically, post-tensioned prestressing systems can be separated into three general categories by the types of tendon in use: wire, strand, and bar systems. End anchorages for these tendons are based on either wedge or direct-bearing principles; sometimes a combination of the two is used. A description is presented below of post-tensioned prestressing systems in terms of types of tendons and end anchorages.

Wire Systems. Wire systems employ a number of parallel wires grouped to form a tendon. Wires manufactured in the United States conform to ASTM Specification A-421, "Uncoated Stress-Relieved Wire for Prestressed Concrete." This specification provides for wires of two types (BA or WA), depending on whether they are to be used with buttons or wedge-type anchorages.

The BBRV system, developed in Switzerland by Birkenmaier, Brandestini, Ros, and Vogt, is a wire system used in both concrete reactor vessels and containments built in the United States. The main feature of this system is the use of cold-formed buttonheads for direct bearing at each end of the wire.

The prestressed concrete reactor vessel (PCRV) of the Fort St. Vrain station in Colorado employs the BBRV system with 169-wire tendons developing approximately 2000 kips capacity each. A number of containments utilizing the BBRV system with 90, 163, 169, 170, and 186 wires per tendon have been built in the United States. The wire diameter is 1/4 inch (6.35 mm) in all cases except for the 163-wire tendon, which uses 7-mm (0.28 inch) wire.

Strand Systems. Strand systems employ a number of "strands" that are bundled into a tendon. A strand is made up of a number of factory-twisted wires. Stress-relieved strand is made in two forms. The first is the seven-wire strand, which conforms to ASTM Specification A-416, "Uncoated Seven-Wire Stress-Relieved Strand for Prestressed Concrete." The second form consists of larger strands that are made of larger individual wires and may consist of more than seven wires per strand. The larger strands are not covered by ASTM specifications and have not been used for the construction of nuclear power plants in the United States.

Strand systems have been introduced in the construction of nuclear power plants by Strand-Wrap, VSL (Vorspann System Losinger), Stressteel, Freyssinet, and SEEE (Societe d'Etudes et d'Equipments d'Enterprises). The last two systems were considered but have not yet been used in the United States in nuclear power plants. Both the Freyssinet and SEEE systems have been used in Europe on concrete reactor vessels.

The Strand-Wrap system has been reviewed and approved only for applying hoop prestressing to some PCRVs in the United States. The basic principles of applying hoop prestressing to the PCRV by the Strand-Wrap system are the same as those for conventional prestressed concrete tanks and circular liquid containers that have been built using wire-winding machines. Steel-lined circumferential precast concrete channels are anchored to the outer cylindrical surface of the vessel by reinforcing bars extending radially inward from the precast channels. The strand is anchored at one end by means of a tapered wedge grip in the rib between adjacent channels and then wound around the vessel at the design tension for a number of turns and anchored in the next adjacent rib. Each band of circumferential prestressing consists of multiple layers of strand wound onto these channels. Each layer consists of one continuous length of strand. A maximum hoop prestressing force of about 6600 kips per linear foot of vessel height is being used in the design of the PCRV head region of the Delmarva Summit Power Station.

The VSL strand system was developed in Switzerland. This system employs a wedge anchorage for strands. Each strand is drawn through the openings of both the bearing plate and the anchor head and is held by a two-piece split cone wedged tightly against the inner surface of the anchor head. As an example, the containment of the Rancho Seco Nuclear Generating Station* in California employs the VSL system with tendons consisting of 55 strands, each tendon developing 2250 kips capacity.

The Stressteel S/H multistrand system was developed in the United States during 1967-1968 by Stressteel Corporation in cooperation with Howlett Machine Works. The system is characterized by a three-piece slotted wedge cone that grips three strands in its serrated teeth, with a number of wedges in a single anchor plate making up a multistrand tendon of the desired size.

As an example, the containment of the Three Mile Island Nuclear Station Unit No. 2 in Pennsylvania employs the Stressteel S/H multistrand system consisting of tendons with 54 1/2-inch, Grade 270K, 7-wire strands per tendon, each tendon developing 2230 kips capacity.

The Freyssinet system was named after the French engineer Eugene Freyssinet, who invented the anchorage device in 1939. The original anchorage device was for a wire system only. This is a commonly used commercial system. The anchorage consists of a male conical plug and a female conical recess. The plug, with the wires spaced evenly around its perimeter, anchors the wire by wedge action.

As a result of market requirements and subsequent developments, the Freyssinet system now also has available anchorages for strand tendons and other shapes of anchorage devices different from the original one. The same wedge principle for anchoring the tendon is retained, however. Concrete reactor vessels have been built in Europe using the Freyssinet strand system with a maximum tendon capacity of about 2000 kips.

The SEEE system was developed in France by Societe d'Etudes et d'Equipments d'Enterprises. The system features threaded anchorage fittings extruded onto the ends of a group of strands. An anchoring nut is then threaded onto the anchorage fitting and turned tightly against the bearing plate. A tendon is composed of one or several such anchorage fittings on a common bearing plate.

Bar Systems. Bar systems employ a number of high-tensile-strength steel bars that are bundled into a

*The Freyssinet, SEEE, and VSL systems were formally presented as alternatives to the previously approved BBRV system. The VSL system was chosen by the applicant. Consequently, the Freyssinet and SEEE systems were not reviewed by the NRC staff with regard to their acceptability for use in nuclear power plant containments.

tendon. The bars are made from an alloy steel conforming to ASTM Specifications A-322 and A-29. A-322 is a general specification that covers only the chemical composition of many grade designations of alloy steel bars, and A-29 is a specification for general requirements for hot-rolled and cold-finished carbon and alloy steel bars. No ASTM specification covers the minimum mechanical and physical requirements for the prestressing bars after processing, as in the case of wires (A-421) and strands (A-416) and it is for this reason that a specification* was written by the Prestressed Concrete Institute.

Bars are cold-stretched and also stress-relieved by heat treatment to produce the prescribed mechanical properties. Both deformed bars and smooth bars with threaded ends are available, but only smooth bars have been used for nuclear power plant construction in the United States.

The Stressteel Corporation in the United States employs a bar system. The bars are stressed by means of an hydraulic jack that consists of a coupler and pulling bar. The normal commercial technique for anchoring uses anchor nuts. During stressing, the anchor nuts are continuously screwed down on washers and bearing plates, and the prestressing force is then transferred to the anchorage assembly by releasing the force in the jack. Wedge and grip-nut anchorages are also available to anchor the bar; they possess the advantage of being able to grip the bar at any point along its length.

The containment structure of H.B. Robinson Unit No. 2 in Hartsville, South Carolina, employs the Stressteel bar system anchored with Howlett Grip Nuts. The tendon, composed of six 1-3/8-inch-diameter Stressteel bars, develops a capacity of 1428 Kips.

Grouted and Ungrouted Tendons

All of the concrete reactor vessels and containments designed and built in the United States use ungrouted tendons except for H.B. Robinson Unit 2 (bar tendons), Three Mile Island Unit 2 (strand tendons), and Forked River (strand tendons), all of which were designed for grouted tendons. On none of these, however, has design credit been given for any bond of the grouted tendons.

Whether grouted or ungrouted tendons are used, a means of determining the functional capability of the structure during its lifetime should be available. This results in a need for reliable quality assurance procedures for the tendon installations and a need for a reliable structural inservice inspection program. To date, this has been easier to accomplish through the use of ungrouted tendons.

*"Guide Specification for Post-Tensioning Materials," PCI Post-Tensioning Manual, Prestressed Concrete Institute, 1972.

C. REGULATORY POSITION

This regulatory guide covers the generic qualifications of post-tensioned prestressing systems used in concrete reactor vessels and containments, with no attempt to extend its scope to design aspects. The acceptability of any post-tensioned prestressing system in conjunction with a specific structure design would have to be evaluated on a case-by-case basis. Any proposed system submitted for NRC approval should consider the following:

1. Post-tensioned prestressing systems that have been approved in previous nuclear power plant license applications are regarded as accepted systems. See Table A for identification. When the claim is made by an applicant that the prestressing system proposed is an accepted system, sufficient information should be provided with each application to demonstrate that the system proposed is the same as the one that was approved in previous nuclear power plant license applications. Prior approval of any system does not relieve the applicant of the responsibility for demonstrating that its system meets all the requirements of the Code for Concrete Reactor Vessels and Containments.*

2. Changes in prestressing element materials or in anchorage items of previously accepted systems that may require repeating the system performance tests are identified in Subsections CB and CC, Article 2466 of the Code for Concrete Reactor Vessels and Containments.

3. Any new post-tensioned prestressing system should meet the requirements set forth in the Code for Concrete Reactor Vessels and Containments.

4. The use of any post-tensioned prestressing system should permit the application of an inservice inspection program that will verify the continued functional capability of the structure. Implementation of this program should not degrade the quality and reliability of the post-tensioned prestressing system. Regulatory Guides 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures," and 1.90, "Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons," should be consulted for recommendations concerning the use of ungrouted and grouted concrete containments, respectively.

*ASME Boiler and Pressure Vessel Code, Section III, Division 2 (the latest version, plus addenda, as endorsed by the Nuclear Regulatory Commission). This Code is currently under review for endorsement by the NRC staff.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the staff's plans for using this regulatory guide.

Except in those cases in which the applicant proposes an alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used in the evaluation of submittals for construction permit applications docketed after June 30, 1976.

If an applicant wishes to use this regulatory guide in developing submittals for applications docketed on or before June 30, 1976, the pertinent portions of the application will be evaluated on the basis of this guide.

TABLE A

STATUS OF SYSTEMS AS OF MAY 1975

System	*Submitted For Licensing Review*	*Reviewed For Licensing Acceptability*	*Approved By the NRC Staff*	*Used In U.S. Nuclear Power Plants To Date*
BBRV 90, 169, 170, 186 Wires (1/4 in. φ) 163 Wires (7 mm φ)	X	X	X	X
VSL (55 strands)	X	X	X	X
Stressteel S/H (54 strands)	X	X	X	X
Freyssinet (strand)	X	–	–	–
SEEE (strand)	X	–	–	–
Stressteel (6 1-3/8 in. bars)	X	X	X	X
PCRV Strand-Wrap	X	X	X	–