

The Core Protection Calculator System (CPCS) is being replaced due primarily to parts obsolescence. The replacement CPCS will perform functionally identical safety-related algorithms as the existing CPCS, but on a newer platform. The CPCS design function will remain unchanged.

The physical location of the replacement CPCS will be the same as the existing CPCS in the auxiliary protective cabinets. Installation will occur during refueling outages when the system is not required for service. [The] majority of the testing will be performed prior to installation.

The CPCS is not an initiator of any analyzed accident, but is used for mitigation of a large number of anticipated operational occurrences and a small number of accidents. Since the CPCS is not an accident initiator, and the replacement CPCS is functionally unchanged, the CPCS replacement will not increase the probability of an accident.

The functionality of the existing CPCS safety related algorithms are replicated in the System Requirements Specification for the Common Q [Common Qualified] Core Protection Calculator System. The basic Common Q CPCS design concept was approved by NRC Safety Evaluation (SE), Acceptance For Referencing Of Topical Report CENPD-396-P, Rev. 01, "Common Qualified Platform" and Appendices 1, 2, 3 and 4, Rev. 01, dated August 11, 2000 (Ref. 2 [listed in the enclosure to the amendment request]), and there have been no significant functional changes to the design as presented. The requirements for response time and accuracy that are assumed in the Palo Verde Nuclear Generating Station (PVNGS) Updated Final Safety Analysis Report (UFSAR) accident analysis will continue to be met. Therefore, since the new [replacement] CPCS will be capable of performing the same safety-related functions within the same response time and accuracy as the existing CPCS, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The CPCS provides a monitoring and detection function and is not an initiator for any accident. The CPCS provides Reactor Protection System (RPS) trips on Low Departure from Nucleate Boiling Ratio (DNBR) and High Local Power Density (LPD) in response to calculations involving several input variables. It also provides a Control Element Assembly Withdrawal Prohibit (CWP) signal to the Plant Protection System (PPS), and provides indication and annunciation. The CPCS performs no other plant functions, and is not used to initiate any ESF [(Engineered Safety Feature)] functions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The [new] CPCS is a replacement for the existing CPCS. It will retain the same safety-

related functionality as the existing CPCS. The equipment will be qualified in accordance with requirements described in the Palo Verde UFSAR.

The replacement CPCS will perform functionally identical safety-related algorithms as the existing CPCS, will trip in response to the same inputs with equivalent accuracy, and will meet the same four channel separation requirements. The only significant area of difference involves the platform. The Common Q platform uses a consistent set of qualified building blocks (Advant Controllers, Flat Panel Displays, Power Supplies, and Communication Systems) that can be used for any safety system application. For Palo Verde purposes, the only application of this platform at this time will be for use as a CPCS. The new platform will include improved human factors and fault tolerance within each CPCS channel.

In summary, the replacement CPCS performs the same functions as the existing CPCS, meets the qualification requirements of the existing CPCS, and meets the accuracy standards of the existing CPCS. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, APS [(the licensee)] concludes that the proposed amendments(s) present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendments until the expiration of the 30-day notice period. However, should circumstances change during the notice period, such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendments before the expiration of the 30-day notice period, provided that its final determination is that the amendments involve no significant hazards consideration. The final determination will consider all public and State comments received. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to

take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D59, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from 7:30 a.m. to 4:15 p.m. Federal workdays. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, Public File Area O1 F21, 11555 Rockville Pike (first floor), Rockville, Maryland.

The filing of requests for hearing and petitions for leave to intervene is discussed below.

By September 17, 2003, the licensee may file a request for a hearing with respect to issuance of the amendments to the subject facility operating licenses and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR part 2. Interested persons should consult a current copy of 10 CFR 2.714, which is available at the Commission's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland, and available electronically on the Internet at the NRC Web site <http://www.nrc.gov/reading-rm/doc-collections/cfr/>. If there are problems in accessing the document, contact the Public Document Room Reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr@nrc.gov. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons