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

FEB 10 1986

Docket No. 50-354

Mr. Corbin A. McNeill, Jr., Vice President-Nuclear
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Nuclear Administration Building
P. O. Box 236
Hancocks Bridge, New Jersey 08038

Dear Mr. McNeill:

Subject: Hope Creek Independent Design Verification Program (IDVP)

Enclosed is a Safety Evaluation Report (SER) associated with the Hope Creek IDVP. Of the 159 valid observation reports (ORs) identified by the IDVP contractor, Sargent & Lundy Engineers, 24 resulted in Final Safety Analysis Report (FSAR) changes. These ORs are identified in Section 17.5.4.1 of the enclosed report. Except for OR 124, all of these FSAR changes have been submitted to the NRC in FSAR Amendments 11, 12 and 13. Mr. Bruce Preston of your staff has stated that the FSAR change(s) resulting from OR 124 were submitted to the NRC in Amendment 14 to the FSAR.

In order to conclude our review of the Hope Creek IDVP, we request that prior to fuel load, you verify that all corrective actions resulting from the IDVP, and the eight ongoing verification programs, have been completed.

If you have any questions concerning the enclosed report, please contact your licensing Project Manager, Dave Wagner (301) 492-9418.

Sincerely,

B Boumin / for

Elinor G. Adensam, Director
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Division of BWR Licensing

Enclosure:
As stated

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17 QUALITY ASSURANCE

17.5 Independent Design Verification

17.5.1 Background

PSE&G proposed a plan to the NRC staff for an Independent Design Verification Program (IDVP) to provide additional independent assurance that the design of Hope Creek meets FSAR commitments and regulatory requirements. The IDVP consisted of reviews of the design and design process of representative Hope Creek systems, structures, and components. The high pressure coolant injection (HPCI) system, automatic depressurization system (ADS), and safety auxiliary cooling system (SACS) were reviewed to assess the adequacy of the design and the design process.

The IDVP review was conducted in accordance with the "Program Plan for Hope Creek IDVP (Revision 1)," which was approved by the NRC staff on April 11, 1985. The IDVP was conducted by Sargent & Lundy Engineers (S&L) and commenced in April 1985. The NRC staff closely monitored the conduct of the IDVP, including:

(1) A two-phase inspection of the IDVP implementation at S&L's offices in Chicago. Phase I was conducted from May 8 to 10, 1985, and Phase II from June 4 to 6, 1985 The NRC staff's report of this inspection, 50-354/85-32, was provided to PSE&G by a letter dated July 8, 1985 (B. K. Grimes, NRC, to R. L. Mittl, PSE&G).

(2) An inspection of the documentation technically supporting the observation reports (ORs) identified in the "S&L Final Report for the IDVP" conducted at S&L's offices in Chicago from September 30 to October 3, 1985. The NRC staff's report of this inspection, 50-354/85-54, was provided to PSE&G by a letter dated December 11, 1985 (B. K. Grimes, NRC to R. L. Mittl, PSE&G).

(3) An inspection of agreed on corrective actions identified in the "S&L Final Report for the IDVP," conducted at Bechtel's offices in San Francisco from October 21 to 25, 1985. Implementation of the corrective actions are discussed in this supplement.

17.5.2 Independent Design Verification Program Technical Review

The IDVP was a comprehensive technical review over a 6-month period conducted by a team of qualified S&L personnel experienced in nuclear power plant design. The S&L team assessed the technical adequacy of the design of the selected systems, common as well as unique plant design features, and the design process used at Hope Creek.

The purpose of the IDVP was to provide additional assurance that

(1) FSAR commitments and regulatory requirements are being met

(2) design process and interface control between technical disciplines was functioning

(3) the design is technically adequate

Using the systems and features selected for review, the S&L team drew from their experience to establish areas of focus to assess the above. The reviews were extended within these areas as required to ensure an indepth review of the particular activity investigated. The team reviewed designs for accuracy, reasonableness, and adequacy of design techniques. The IDVP also included system walkdowns and a review of the field change process.

S&L's approach to the IDVP was based on 93 prepared checklists to facilitate the design review. The systems and common requirements areas reviewed against these checklists were selected because they represented reasonably complex safety-related systems involving interfaces with the nuclear steam supply system (NSSS) vendor and other vendors or subcontractors. On the basis of these criteria, the HPCI, ADS, and SACS were the systems selected for review. Design requirements reviewed that were common to all safety-related systems included high- and moderate-energy line breaks, fire protection, seismic interaction/analyses, and radiation shielding.

The team generated potential observation reports (PORs) when concerns or discrepancies were identified during the review process. PORs were reviewed by discipline project engineers and an internal review committee to establish the validity of the concern. Valid PORs were submitted as ORs to PSE&G and Bechtel Power Company (BPC) for resolution. BPC or PSE&G then issued resolution/completion reports (R/CRs), and the OR was closed out or invalidated upon acceptance of the resolution by S&L. A senior review committee was established to review potentially safety-significant ORs; however, none were identified by the IDVP. The internal review committee assessed trends in the observations and generic implications affecting the plant design.

17.5.3 Independent Design Verification Program Conclusions

The S&L team concluded that the IDVP and current, ongoing design activities provide reasonable assurance that the overall design of Hope Creek is technically adequate and conforms to FSAR commitments and regulatory requirments.

A total of 159 valid ORs were issued. None of the ORs were considered safety significant or required a change in design or hardware. S&L issued completion reports to close the ORs. The IDVP report provided detailed results of the review and noted that several areas of design had not yet been completed, including the following design verification programs:

(1) Load Verification Program for Structures

(2) As-Built Reconciliation Program for Piping and Pipe Supports

(3) Reassessment Program for Conduit, Cable Tray, and Heating, Ventilation, and Air Conditioning (HVAC) Ducts and Supports

(4) Instrument Setpoint Calculation Program

(5) Seismic II/I Review Program

(6) Equipment Anchorage Program for Equipment Foundations

(7) Program To Verify Environmental Qualification of Safety-Related Equipment

(8) Final Hazards Assessment Program for High Energy and Moderate Energy Line Break Accidents (HELBA/MELBA).

In forming conclusions regarding the adequacy of the design of Hope Creek, the S&L team relied on these ongoing verification programs to ensure satisfactory completion of the design process. The S&L team reviewed each of these programs, and various program enhancements resulted from IDVP comments or concerns.

The S&L team was unable to reach a conclusion concerning the adequacy of design in the area of hazards analysis and in the establishment of instrument setpoints because of its incomplete status at the time of the review. However, the S&L team noted in the report that completion of the Final Hazards Assessment Program for HELBA/MELBA and the Instrument Setpoint Calculation Program will ensure that the design in these areas is technically adequate and meets FSAR commitments and regulatory requirements.

The S&L team also concluded that, on the basis of its engineering judgment, the design of the basemat was technically adequate. However, it was the S&L team's opinion that the basemat analysis performed by BPC lacked sufficient detail to analytically demonstrate the technical adequacy of the mat design. BPC subsequently performed a basemat confirmatory analysis to verify the technical adequacy of the mat design.

17.5.4 Assessment by the NRC Staff

The NRC staff has assessed the results of the IDVP. It conducted three inspections at BPC and S&L offices to review documentation supporting the conclusions reached by the S&L team and the resulting corrective actions being taken to resolve valid concerns identified during the IDVP.

These inspections confirmed the conclusions reached by the S&L team. The staff found the S&L review to be formally documented and conducted in sufficient technical depth to support meaningful conclusions as to the adequacy of the design and design process at Hope Creek. On the basis of the review of the IDVP report and supporting documentation, the staff concurs with the conclusions presented in the report, subject to the comments in this supplement.

Because the S&L team was unable to reach a conclusion on the adequacy of design in the areas of hazards analysis and instrumentation setpoints, and because its conclusion on the technical adequacy of the basemat was based on engineering judgment rather than the completed BPC analysis, PSE&G has provided confirmation to the staff that the design for these three areas is technically adequate and in accordance with FSAR commitments.

17.5.4.1 FSAR Changes

Several ORs involved commitments from PSE&G to submit proposed FSAR changes to the NRC staff. ORs in this category are 20, 21, 22, 24, 25, 41, 66, 89, 100,

111, 122, 124, 131, 138, 148, 150, 181, 194, 198, 212, 215, 219, and 220. Proposed FSAR changes constitute open licensing issues to be resolved between PSE&G and the NRC staff. Because these items will be individually resolved as licensing issues, they are considered closed for design review purposes when PSE&G has submitted them to the NRC staff. Most of the FSAR changes were submitted to NRC on November 21, 1985, in Amendment 13. The only remaining FSAR change to be submitted is one resulting from OR 124. This will be submitted in Amendment 14 before fuel load.

17.5.4.2 Technical Assessment

The following is the NRC staff's technical assessment of significant ORs (both valid and invalid) identified in the IDVP. The staff's investigation of the invalidated ORs was to assess the correctness of decisions made by S&L in invalidating these ORs. ORs were invalidated after S&L obtained additional information from BPC to resolve its original concerns. Invalidated ORs were those for which S&L did not understand BPC's design philosphy or design approach. In all cases, the staff concurs with S&L's decisions in invalidating these ORs.

Mechanical Discipline

There were 34 ORs in the mechanical systems area and 23 in the pipe stress area. Of these, nine mechanical systems and nine pipe stress ORs were designated invalid by S&L on the basis of additional information obtained from BPC. The following NRC staff comments are provided.

ORs 3, 4, and 93 identified various unreconciled discrepancies between the pipe routing drawing and the mathematical model used in the pipe stress analysis. In response to the ORs, BPC performed reanalyses to assess the significance of the errors and was able to confirm the overall integrity of the piping systems in question. However, because this indicated a weakness of the design process, the Staff reviewed selected pipe stress analysis packages that had been completed under the BPC As-Built Reconciliation Program. On the basis of this review, it was determined that where such errors were detected, they were corrected and reconciled, and hence the final pipe stress analysis packages accurately reflect the as-built installation. Consequently, the staff considers these ORs closed.

OR 13 indicates that load combinations in the Class 2 and 3 Piping Design Specification No. 10855-M-068 (Q) do not agree with those committed to in the Hope Creek FSAR. Although the pipe design specification was revised to incorporate FSAR load combinations, the staff was concerned that the effects of improper or erroneous load combinations on piping analyses completed before this corrective action might not have been addressed in the OR or the R/CR. However, during the inspection at BPC on October 21-25, 1985, the staff found that the internal BPC procedures to which the stress analyses were performed contained the correct loading combinations. Hence, the stress analyses were performed correctly, and the only problem was the need to revise the specification to show the correct load combinations. In addition, the final stress reconciliation program will again verify that correct load combinations were used. Consequently, the staff considers this item closed.

OR 62 is an OR invalidated after further S&L review which concerns the consideration of bend allowance in wall thickness calculations. BPC Wall Thickness

Calculation No. 1(Q) appears to calculate wall thickness for only straight pipe. BPC then requires the pipe fabricator to ensure that wall thickness at bends is consistent with minimum values established in the calculations. The OR describes several instances where minimum wall thickness requirements are not specified, that is, 2½-in. nominal pipe size (NPS); greater than 14-in. NPS; sizes 10, 12, and 14 in. of class DBB; and sizes 3 to 14 in. for class HBC. The BPC R/CR indicates that BPC reviewed measurement records of all safety-related large pipe bends fabricated by Dravo Corporation for Hope Creek and found them consistent with the requirements for thicknesses after bending of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). BPC also stated that no 2½-in. bends had been made or designed. Wall thickness measurement records cited by BPC indicated requirements had been met. The wall thickness measurement record for the bend on line 1-EG-HBC-153 indicated that wall thickness acceptance was based on acceptance criteria specified in BPC Technical Specification No. 10855-P-201(Q). The IDVP report indicated that 87½% of nominal wall thickness was the acceptance criterion used; however, the report did not indicate what the basis was for this criterion. During its inspection at BPC during the week of October 21, 1985, the staff reviewed Calculation No. 1(Q) and determined that this calculation demonstrated that, for the different classes of piping at Hope Creek, 87½% of nominal wall thickness was always greater than the required minimum thickness. Consequently, the staff finds this criterion acceptable and concurs with the S&L invalidation of this OR.

OR 65 was also invalidated after further S&L review concerning impact test requirements. The BPC specification for hydropneumatic accumulators, Specification No. 10855-M-707A(Q), does not specifically state an impact test temperature. The ASME Code requires that the designer specify design requirements. The staff was concerned that this important design parameter would have no documented basis if established by the vendor, who is not the designer and who does not have access to the design basis for this equipment. However, during its inspection at BPC during the week of October 21, 1985, the staff reviewed this matter in detail with the BPC personnel. The staff determined that (1) the specification did require impact testing in accordance with the ASME Code, (2) the minimum temperature to which the material was subjected was specified, and (3) BPC's review of the vendor's impact test data ensured that adequate impact testing had been performed. Although the staff feels that the requirements might have been more clearly identified to the vendor, it is clear that the BPC process is effective, and the staff, therefore, concurs with the S&L invalidation of this OR.

OR 77 identified the failure to consider nozzle reactions in the hydropneumatic accumulator tank that would exist up to and at the time that the nozzle on the accumulator breaks. Qualification Report No. 103855-C152(Q)-2809-3 only evaluated the blowdown forces that would exist after nozzle rupture, which are substantially smaller than the reactions before rupture. BPC has reviewed possible generic implications and determined that this omission only affected the hydropneumatic accumulator tank. BPC Specification No. 10855-M707A(Q), Revision 5 (dated March 3, 1983), has been revised to incorporate the loading. Further, the vendor, PDM Corporation, has revised the qualification report to account the revised nozzle loading and its effects on the nozzle, anchorage, base ri: skirt, and other parts of the tank. The staff has confirmed that the design specification has been updated, that the hydropneumatic accumulator tank is t only component affected by the omission, and that the qualification report has

been updated. The staff is satisfied with this action and considers this item closed.

OR 84 involves the safety auxiliary cooling system heat exchanger (SACSHX) design and test pressures. The data sheets for the heat exchanger show hydrotest pressures that are less than required on the basis of the specification requirements for "full shell design pressure and temperature with full vacuum on the tube side and full tube side design pressure and temperature with full vacuum on the shell side." The R/CR indicates that the design pressures on the data sheet represent worst-case pump shutoff head on one side with vacuum on the other side. It also states that this is an "unusual scenario." Although the staff agrees that this is an unlikely condition, the established design pressure should consider normal operating pressures that include operation at pump shutoff conditions. The hydrotest pressures should then be established on the basis of this design pressure. Because of this concern, the staff reviewed this matter with BPC personnel during the inspection the week of October 21, 1985. The staff learned that BPC had performed a calculation that showed that because of the conservatism in the originally selected design pressures for the shell and tube sides of the heat exchanger, these values enveloped the conditions of full pressure on one side with simultaneous full vacuum on the other. The staff reviewed this calculation and concurred that BPC had demonstrated that Code requirements had been met. BPC has agreed to revise the specification to resolve any ambiguities. The staff is satisfied with this action and considers this item closed.

ORs 95 and 98 resulted from a failure by BPC to complete required analyses on ASME Code, Section III, Class 2 lines. Specifically, qualification of 4 of the 14 flanged joints in Calculation No. C1750-3Q (dated February 13, 1984) were not performed (OR 95) and a pipe integral attachment (anchor plate) was not evaluated (OR 98) as part of Calculation No. C33-2Q. BPC had completed the evaluation of the above mentioned items, and after review by the staff, it was concluded that the omissions did not affect the qualification of the piping systems. The programmatic aspects of these omissions had been addressed by BPC by including review of all pipe components and attachments in the As-Built Reconciliation Program (ABRP) as well as performing a detailed and vigorous training program for all engineers involved in the ABRP. The staff has reviewed the above remedial measures and concludes that BPC has adequately addressed the problems; consequently, this item is considered closed.

OR 110 was generated by the S&L team's review of the SACS/TACS (turbine auxiliary cooling system) pressure drop calculations. During its inspection at BPC during the week of October 21, 1985, the staff verified that the following corrective actions had been accomplished as identified in the R/CR of the IDVP: (1) Calculation No. EG-22(Q) had been revised to reference Cameron Hydraulic Data, 15th Edition, and to reference a superseded calculation, No. EG-01(Q), with a note as to why and what parts of the superseded calculation were referenced; and (2) a survey of 10% of other project pressure drop calculations had been performed and found to be conservative, although some minor errors had been found. It was BPC's position that calculations with known errors did not need to be corrected as long as the calculations were conservative with regard to the results of the calculations. The staff was concerned that the plant's design basis should be accurately known, since these calculations might be used for future modifications. After discussions with PSE&G and BPC, agreement was reached that calculations of this type would be provided with errata sheets identifying the known errors and stating why the results of the calculations

remained conservative in spite of the errors. Before completion of the October 21-25, 1985, inspection, the staff was provided with three revised calculations, EA-3(Q), JE-9, and BJ-2(Q)-3. The staff reviewed these calculations and determined that the errata sheets resolved the staff's concerns. Consequently, the staff considers this matter closed.

OR 112 discusses inconsistencies between SACSHX Design Specification M-069, the FSAR, and Calculation Nos. EG-20(Q) and EG-22(Q). The FSAR and the design specification indicate a SACSHX design flow of 10,980 gpm. However, Calculation No. EG-20(Q) shows a required flow of 13,727 gpm during a normal plant shutdown with one SACS loop operating. It was not apparent from Calculation No. EG-22(Q) that the flow can be achieved for this mode of operation. In the R/CR, BPC has provided vendor confirmation that the SACSHX can accommodate the increased flow without adverse structural effects. However, the R/CR does not provide sufficient documentation to ensure that the increased flow requirements to the SACSHX can be met during this mode of operation, which assumes the single failure of the redundant flow train. The R/CR concludes that "a sufficient quantity of non-essential equipment in the TACS can be isolated to ensure that design flows are available to the essential equipment necessary for plant shutdown," and that flow requirements can be met on the basis of a review of the system operating point on the one loop pump curve in the calculations. During its inspection at BPC during the week of October 21, 1985, the staff reviewed Calculation Nos. EG-20(Q) and EG-22(Q). The staff determined that although approximately 3000 gpm of additional flow was needed to achieve the basic requirement, a total of over 13,000 gpm additional flow is available when all the TACS loads are isolated. The staff concluded that BPC's position that sufficient flow was available by isolating TACS loads was valid. Further, the staff learned that the SACS/TACS is being modeled for future reference and that throttle settings will be established during system balancing as part of the startup/test program. In view of this information, the staff concludes that a sufficient design basis has been established and considers this matter closed.

OR 124 indicates that Calculation No. 12-118(Q) shows high-energy-line-break locations that differ from those shown in the FSAR and does not reference the stress analysis used as the basis for break selections. BPC stated in the R/CR that this calculation is not an analysis and has been voided because it is not a design-basis document. However, the R/CR makes use of the voided calculation to indicate the reasons for inconsistencies with the FSAR. BPC provided a sample Plant Design Stress Calculation No. SS-10 that detailed stress results and break locations. During its inspection at BPC during the week of October 21, 1985, the staff reviewed this matter in detail and concluded that Calculation No. 12-118(Q) was merely a tool for convenient reference by personnel doing jet impingement analysis. The staff observed the full-size drawings that show break locations. Consequently, the staff concludes that Calculation No. 12-118(Q) is not needed as a design document, that BPC's action in this matter is valid, and that the matter is considered closed.

OR 125 resulted from a failure to consider the 40°F operating mode on the HPCI pump discharge line. Such an omission could affect the adequacy of the associated piping and supports as well as the HPCI pump discharge nozzle. BPC performed the analysis and on review by the staff, it was concluded that the

effect was insignificant. In addition, to preclude similar problems from occurring on other systems, BPC performed a full review of the design temperature conditions, as reflected in Drawing 10855P-0501 (Line Index). The staff reviewed this corrective action and concluded that the matter was adequately addressed. Consequently, this item is considered closed.

OR 145 stems from the fact that local transmissibility had not been considered for panel-mounted Rosemount pressure and flow transmitters (Tag No. B21-N0015-E41-N051) during General Electric's (GE's) seismic qualification of these transmitters (Report No. NEDC-30446, Book C59). To correct this error and ensure that all devices supplied by GE that are mounted on flexible panels are seismically qualified, GE has updated the documentation for all other similar equipment. The applicant confirmed that the supporting documentation adequately addresses the concerns noted in this OR. Therefore, the staff considers this item closed.

The remaining ORs in the mechanical discipline are not individually discussed in this supplement because they are considered to be of a minor nature, and adequate corrective action, where applicable, has been taken. The staff has reviewed these ORs and has verified completion or satisfactory progress of proposed corrective action on a sampling basis. The staff requests confirmation by the applicant that all corrective actions associated with these ORs have been completed.

Electrical and Instrumentation and Control Discipline

There were 41 ORs in the electrical area, 40 in the equipment qualification area, and 13 in the instrumentation and controls area. Of these, 16 electrical, 13 equipment qualification, and 7 instrumentation and controls ORs were designated invalid by S&L on the basis of additional information received from BPC. The following NRC staff comments are provided on the valid ORs.

OR 161 concerned errors found in preliminary instrument setpoint calculations and questioned if the existing calculations, when finalized, would meet the requirements of RG 1.105, Revision 1, November, 1976. The R/CR indicates that all errors found will be corrected as part of finalizing the preliminary calculations. During the inspection at BPC during the week of October 21, 1985, the staff reviewed BPC procedure, "Finalized Committed Instrument Setpoint Calculation," implemented by interoffice memorandum dated September 20, 1985, and found it satisfactory. Additionally, the staff reviewed two instrument setpoint calculations (Calculation Nos. 113 and 141), which were finalized in accordance with the new procedure, and found them satisfactory. PSE&G Site Engineering Instruction (SEI) 3.4, "Setpoint & Instrument Calibration Methodology," was also reviewed by the staff. This procedure provides guidance for calculating drift associated with calibration frequency, applying instrument inaccuracies and a formal method for recording and verifying instrument setpoints. The staff reviewed PSE&G Calculation No. SE-EG-0032-1 prepared in accordance with SEI 3.4. The staff concludes that the upgraded instrument setpoint calculation program meets the requirements of RG 1.105, Revision 1, November 1976, and considers this OR closed.

OR 216 concerned the fact that BPC Calculation No. 2.5, Revision 0, did not consider the derating of the 4.16-kV nonsegregated phase bus (installed in control and diesel generator areas of the auxiliary building) resulting from

fire barrier wrap on the bus ducting. The R/CR indicates that fire wrapping was currently being evaluated to determine its effect on bus rated ampacity and that the final determination of the accuracy of the bus design cannot be made until the effect of the fire wrapping is evaluated. During the inspection at BPC on October 21, 1985, the staff reviewed an engineering analysis dated September 30, 1985, prepared by Delta-Unibus Corporation, which evaluates the in-plant bus duct configuration with fire wrap. This analysis shows that adding the fire barrier wrap has no appreciable effect on the temperature rise in the conductors. This, coupled with the large margin in the original design, results in no effect on the design. The staff finds this analysis acceptable and considers this item closed.

OR 213 concerned assurances that 120-V ac control circuit lengths to various pieces of equipment were not excessive and that voltages available at the equipment were sufficient for proper operation. The R/CR states BPC will revise Calculation No. 17A(Q) to include solenoid valves and will perform a complete review of circuit lengths for ac-powered solenoid valves. During the inspection at BPC on October 21, 1985, the staff reviewed revised BPC Calculation No. 17A(Q), Revision 1, dated September 27, 1985. This calculation now includes, in Table 3, solenoid valves among the loads that are energized. Additionally, the staff reviewed the voltage drop log for ac-powered solenoid valves prepared by BPC on September 20, 1985. This document is a summary of a review of all circuit lengths for ac-powered solenoid valves to verify compliance with the criteria established by BPC Calculation No. 17A(Q), Revision 1. The staff finds all data consistent and acceptable and considers this item closed.

OR 220 addresses an apparent design inadequacy concerning penetration assemblies, in that the necessary primary and/or secondary circuit overcurrent protection may not be provided. The R/CR states that BPC provided information to confirm that the design provides the necessary primary and/or secondary circuit overcurrent protection. It also states that BPC agrees to revise Calculation No. 7.13(Q) to be consistent with the actual design and installation. The staff reviewed the revised BPC Calculation No. 7.13(Q), Revision 3, dated September 16, 1985, during the inspection at BPC the week of October 21, 1985. This calculation now reflects that actual design and installation. Additionally, maximum short circuit currents and continuous current carrying capabilities have been indicated on penetration breaker coordination curves, attached to Calculation No. 7.13(Q), to verify that penetrations can withstand maximum available short circuit current. The staff finds the resolution of this item acceptable and considers this item closed.

The remaining ORs in the electrical discipline are not individually discussed in this supplement because they are considered of minor nature, and adequate corrective action, where appliable, has been taken. The staff has reviewed these ORs and has verified completion or satisfactory progress of appropriate corrective action on a sampling basis. The staff requests confirmation by the applicant that all corrective actions associated with these ORs have been completed.

Structural Discipline

There were 52 ORs in the civil/structural area. Of these, six were designated invalid by S&L on the basis of additional information received from BPC. Many

of the valid ORs were closed on the basis of continuing programs. These continuing programs consisted of the confirmatory mat analysis, the Load Verification Program, and various other studies. During and subsequent to the IDVP, many calculations were generated that were made part of the Load Verification Program and were used to demonstrate the validity of engineering judgments. Some calculations were revised to correct errors; however, none of these errors were significant enough to affect the adequacy of the structural design. The following staff comments are provided on the valid ORs.

ORs 7 and 40 resulted from a review of the reactor building basemat. OR 7 questioned the computer model and interpretation of the computer output. The R/CR indicated conservatism existed in that the concrete was assumed to carry no tension and made use of the deflected surface in addressing the validity and conservatism of the values for moment and shear. OR 40 was based on omissions in both the analysis and design of the basemat. The S&L team concluded that the basemat was adequate; however, the existing analysis contained inconsistencies and omissions, such as the effects of twisting moments on the design of reinforcing steel and the effects of torus uplift loads and thermal loading. A confirmatory reactor building basemat analysis was generated by BPC and incorporated into the calculation record as Calculation No. 621-33(Q). The analysis included the critical load combinations as determined from the previous analysis and included torus safety/relief valve (SRV) loading on the basemat. The term "SRV loading" as used in the IDVP included all hydrodynamic loads such as chugging and condensation oscillation as well as SRV actuation. The confirmatory reactor building basemat analysis indicated that torus uplift loads can be accommodated by the basemat under the torus. In the evaluation of shear capacity of the basemat, the shear requirements of American Concrete Institute (ACI) 318-71, Sections 11.10.1a and 1b, were satisfied for both one-way and two-way slab actions. The analysis of the basemat was performed using computer program BSAP. The output of BSAP is three moments (including twisting moments), three membrane forces, and three shears. These forces and moments were used as input to CECAP. The CECAP program analyzes reinforced concrete sections and calculates stresses in reinforcing bars and concrete resulting from the moments, forces, and shears. The finite element model used in the confirmatory reactor building basemat analysis had a finer plan mesh and had more elements through the depth than the original analysis; namely, five rather than three. In the confirmatory analysis, overturning moments due to horizontal seismic forces were taken directly from the seismic analysis rather than from an integration of soil pressures, and the section moments, forces, and shears were computed within BSAP and presented in the output directory, instead of using a manual computation based on stresses. Thermal effects were not included in the confirmatory analysis. This decision was based on Calculation No 621-16(Q), Revision 2, Sheets 39-45, which indicated that an amount of flexural reinforcing of 0.8 in.^{2} per foot (10% of total reinforcing area) in addition to that provided for other loads was adequate to accommodate the thermal effects. In summary, the confirmatory reactor building basemat analysis specifically addressed ORs 7 and 40. The applicant submitted an independent review report on this matter on December 26, 1985, and concluded that the confirmatory basemat analysis confirmed the technical adequacy of the reactor building basemat design and analysis. The staff has reviewed this report and concurs with the applicant that the confirmatory basemat analysis has addressed all concerns raised in ORs 7 and 40. The staff considers this item closed.

ORs 118 and 155 identified numerous technical concerns related to the structural adequacy of the conduit and cable tray support systems. The concerns involved undocumented engineering judgment. Among the concerns related to conduit and conduit supports were

(1) neglecting the effects of self-weight excitation of the support

(2) failure to evaluate all connection options shown on referenced drawings.

(3) failure to evaluate axial compression force on vertical members due to self-weight and downward vertical seismic load

(4) qualification of a conduit clamp without performing the necessary analysis

Among the concerns related to cable tray and cable tray supports were

(1) the adequacy of all cable tray supports for peak seismic response in the longitudinal direction

(2) failure to address the self-load and self-weight excitation of the support

(3) neglecting the effect of conduit load on cable tray systems

(4) qualification of various connection design details without documentation

(5) neglecting to address the amplification of acceleration due to flexible supports as it affects cable tray qualification

(6) failure to address, in the qualification of the cable trays, the use of an enveloped response spectrum that does not envelope all locations in the plant

The NRC staff has reviewed BPC corrective actions and responses to the concerns outlined in the subject ORs during its inspection at BPC the week of October 21, 1985. During this inspection, the staff reviewed 12 newly generated calculations and 4 revised calculations associated with conduit and cable tray supports. The staff found that these calculations correctly addressed the concerns mentioned above without hardware or design change. Furthermore, an assessment program, described in Section 17.5.4.3, was in place which addresses the IDVP concerns. Therefore, the staff considers these ORs closed.

OR 170 questioned the adequacy of the SACSHX foundation Calculation No. 625-18(Q). Although there were no significant concerns identified, the S&L team questioned the adequacy of the welds attaching the SACSHX to the embedded plates in the concrete foundation. Calculation No. 625-18(Q) was superseded by Calculation No. 625-113(Q), which addressed all the concerns identified in the OR. The staff reviewed revised Calculation No. 625-113(Q) during its inspection of BPC the week of October 21, 1985, and found that the newly revised calculation provided the assurance that the heat exchanger foundation is adequate. The staff considers this item closed.

The remaining ORs in the structural discipline are not individually discussed in this supplement because they are considered to be of a minor nature, and adequate corrective action, where applicable, has been taken. The staff has

reviewed these ORs and has verified completion or satisfactory progress of appropriate corrective actions on a sampling basis. The staff requests confirmation by the applicant that all corrective actions associated with these ORs have been completed.

17.5.4.3 Design Verification Programs

In arriving at its conclusion that the design of Hope Creek was technically adequate and that FSAR commitments and regulatory requirements had been met, the IDVP report noted that completion of a satisfactory design is ensured by certain ongoing design verification programs. These programs that have been expanded to address findings resulting from the IDVP are

(1) Load Verification Program for Structures

(2) As-Built Reconciliation Program for Piping and Pipe Supports

(3) Reassessment Program for Conduit, Cable Tray, and HVAC Ducts and Supports

(4) Instrument Setpoint Calculation Program

(5) Seismic II/I Review Program

(6) Equipment Anchorage Program for Equipment Foundations

(7) Program To Verify the Environmental Qualification of Safety-Related Equipment

(8) Final Hazards Assessment Program

The NRC reviewed each of these programs in detail during the inspection of BPC from October 21 to 25, 1985, specifically in view of the reliance of the IDVP on these programs in forming IDVP conclusions. The following NRC comments are provided on each program.

Load Verification Program for Structures

A major ongoing program is the Load Verification Program for Structures, which was established several years ago to verify the adequacy of structural steel beams and reinforced concrete slabs and walls. The need for this type of program is typical of nuclear power projects. The initial seismic analysis is performed and the structural design is partially completed before detailed knowledge of equipment and the piping loads is available. The initial design is usually conservative; however, after the location and magnitude of the loads are set, the structural design must be reevaluated with the final loads. As a result of the IDVP, ORs 169 and 191 were written regarding the need to consider column design reevaluation, as were ORs 45 and 205, which addressed the need to reevaluate the treatment of miscellaneous attachment loads resulting from various bulk commodities as an equivalent uniform distributed load.

The program was expanded to consider the ORs mentioned above by choosing six areas, specifically, the areas most heavily loaded with miscellaneous attachments, and evaluating the effect of the actual weights and peak acceleration of the floor response spectra on the original design. In addition, column

designs for the entire plant were reevaluated to validate the original design assumptions.

The staff reviewed the program attributes and selected representative calculations which had been revised as a result of the Load Verification Program. On the basis of this review, the staff concluded that the concerns of the above ORs have been satisfactorily addressed and that the program is being satisfactorily implemented.

As-Built Reconciliation Program for Piping and Pipe Supports

The staff reviewed the As-Built Reconciliation Program for Piping and Pipe Supports to ensure that it adequately addressed the findings in ORs 3, 4, 93, 94, 95, and 98, as well as industry-accepted practices for such programs. The basic programmatic requirements were outlined in "Stress Group Procedures for Piping Stress Analysis for Hope Creek Generating Station, Appendix U, As-Built Reconciliation Procedure." It was supplemented by controlled calculation status lists, stress analysis checklists, and other documents that ensured orderly execution of the work.

A sample of calculations were reviewed which included those referenced in the above ORs. The scope of review included basic technical evaluation requirements, such as

(1) piping analysis model conformance with as-built drawings

(2) conformance of line list design information (temperature and pressure) with piping analysis inputs

(3) proper evaluation in the analysis for fittings, attachments, and flanges

(4) use of correct seismic response spectra

(5) Evaluation of stress comparisons with ASME Code, Section III allowable values

On the basis of this review, the staff concluded that the program was being satisfactorily implemented and the concerns of the above ORs have been adequately addressed.

Reassessment Program for Conduit, Cable Tray, and HVAC Ducts and Supports

The staff, reviewed the Reassessment Program for Conduit, Cable Tray, and HVAC Ducts and Supports to ensure that it adequately addressed the IDVP ORs as well as industry-accepted practices for such programs. Program attributes were contained in the following documents:

(1) Calculation No. 1005(Q), Revision 0, As-Built Reassessment Program for Cable Tray, Conduit and HVAC Supports, HVAC Ducts and Auxiliary Steel, dated July 23, 1985

(2) D2.11, "Structural Design Assessment Criteria for Seismic Category I HVAC Ducts, Plenums and Supports," Revision 1, dated March 29, 1985

(3) D2.12, "Structural Design Assessment Criteria for Seismic Category 1E Electrical Raceway Support Systems," Revision 3, dated June 4, 1985

In addition, the staff reviewed a number of representative calculations to confirm that the technical attributes outlined in the above-mentioned documents had been properly implemented. On the basis of the calculations and documents reviewed, as well as resolution of the concerns developed by S&L, the staff concludes that an effective ongoing reassessment program is in place, that the enhancements by the IDVP program have been implemented and, on satisfactory completion, the program will address the concerns raised by the IDVP as well as meet its intended objective.

Instrument Setpoint Calculation Program

Because the Instrumentation Setpoint Calculation Program was not complete at the time of the IDVP, the S&L team could not reach a conclusion about the program's adequacy. The staff reviewed the Instrument Setpoint Calculation Program on a sampling basis to ensure that it adequately addressed the findings noted in OR 161 as well as industry-accepted practices for such programs. The issues raised were:

(1) The calculations did not include the effects of seismic excitation, radiation effects, instrument calibration inaccuracies, or power supply and load effects.

(2) The calculations did not include the most severe abnormal environmental conditions under which the instrument must operate.

(3) The calculations did not address inherent instrument tolerances, calibrating instrument tolerances, or the periodic recalibration required because of instrument drift.

Two calculations, Nos. 113 and 141, which were implemented in accordance with the requirements of "Finalized Committed Instrument Setpoint Calculations," as described in interoffice memorandum dated September 20, 1985, were reviewed. The staff confirmed that the first two issues described above were adequately addressed in these calculations. In addition, Calculation No. 0082-1, prepared in accordance with PSE&G SEI 3.4, was reviewed. It was determined that calculations performed in accordance with SEI 3.4 use the data from the original BPC calculations, corrected to account for the additional technical concerns described in Item (3) above.

In view of the inability of S&L to reach a conclusion regarding the adequacy of instrument setpoints, the applicant, at the staff's request, agreed to provide a verification of the setpoint program. This verification program consists of a review, on a sampling basis, of setpoint calculations to ensure that all requirements of the appropriate procedures are being met and a programmatic review to ensure that all setpoints are being calculated and entered in the setpoint register. As a result of the work done to date, the applicant has confirmed that the plant is adequately designed and meets FSAR commitments.

On the basis of the calculations and documents reviewed, as well as resolution of the concerns developed by S&L, the staff concludes that an effective ongoing program is in place, that the enhancements by the IDVP program have been implemented, and on satisfactory completion, the program will address the concerns raised by the IDVP as well as meet its intended objective.

Seismic II/I Review Program

The Seismic II/I Review Program identifies those portions of structures, systems, and components whose continued function is not required following a seismic event, but whose failure could reduce the functionality of a safety-related plant feature to an unacceptable level. The S&L team identified certain concerns regarding the documentation of the Seismic II/I Review Program in ORs 147 and 166. As a result of these concerns, BPC enhanced the program relative to documentation requirements, particularly with regard to the use of engineering judgment and approval signatures on records sheets.

During its inspection at BPC during the week of October 21, 1985, the staff reviewed the documentation associated with the Seismic II/I Review Program. The staff found that the program was based on Design Criteria Nos. D7.2, D7.3, and D7.9. These documents provided for the use of area drawings, the plant model, and, finally, actual plant walkdown to identify and resolve potentially unacceptable interactions. In addition to the design documents, the staff reviewed a number of record sheets resulting from the implementation of the program. As a result of this inspection, the staff concluded:

(1) An ongoing, effective Seismic II/I Review Program is in place. It has been enhanced by IDVP comments, particularly in program documentation.

(2) IDVP enhancements are not likely to result in hardware changes.

In view of the above conclusion, the staff determined that the enhanced program satisfied IDVP concerns and proper implementation of the program will ensure that FSAR commitments and regulatory requirements have been met. This area will remain open subject to confirmation by the applicant that this program has been successfully completed.

Equipment Anchorage Program for Equipment Foundations

The staff reviewed the Equipment Anchorage Program to verify that the concerns developed in OR 170, "SACS Heat Exchanger Foundation," had been properly implemented and that the as-built anchorage configuration, as well as the final loadings, had been reflected in the calculations. Procedure 680-03(Q), Revision 1, dated July 27, 1985, "Design Procedure for Structural Design of Equipment Foundation - All Buildings," was reviewed, and, in addition, several calculations chosen at random were reviewed and were found to address the concerns raised by the IDVP. Consequently, the NRC staff agreed with the S&L conclusion that proper implementation of the Equipment Anchorage Program will confirm that the design meets the commitments in the FSAR.

On the basis of the calculations and documents reviewed, as well as resolution of the concerns developed by S&L, the staff concludes that an effective ongoing program is in place, that the enchancements by the IDVP program have been implemented, and on satisfactory completion, the program will address the concerns raised by the IDVP as well as meet its intended objective.

Program To Verify the Environmental Qualification of Safety-Related Equipment

The staff reviewed the Environmental Qualification Program to ensure that it adequately addressed the IDVP ORs, as well as the requirements of Institute of Electrical and Electronics Engineers (IEEE) Std. 323-1974 committed to in the FSAR and 10 CFR 50.49. Program attributes, as outlined in detailed checklists, as well as the required format for all environmental qualification packages, were evaluated. In addition, the staff reviewed a representative number of environmental qualification packages during the inspection at BPC the week of October 21, 1985, and found they conform to the program requirements. Hence, on the basis of this review as well as the concerns developed by S&L, the Environmental Qualification Program is considered adequate for the following reasons:

(1) PSE&G internal audits previously identified many of the IDVP observations.

(2) PSE&G has validated the list of harsh-environment equipment.

(3) The GE review will verify qualification of NSSS equipment.

(4) A program to verify qualification for moisture intrusion is in place.

(5) The Environmental Qualification Program has been audited and found acceptable by the NRC staff.

(6) Information missing from environmental qualification packages was, in most cases, in the project files. Many environmental qualification packages were not complete at the time of the IDVP.

On the basis of this review, the staff concluded that the technical concerns that resulted from the IDVP were adequately resolved, an effective program is in place, and on satisfactory completion, the program should ensure regulatory requirements regarding design and design process are being met.

Hazards Assessment Program

The S&L team reviewed the design process and documentation related to protection of safe-shutdown equipment from the effects of high- and moderate-energy line breaks. The S&L team identified a number of ORs associated with hazards protection: ORs 55, 56, 57, 58, 72, 85, 86, 108, 124, 132, 134, 164, 167, and 175. The S&L team subsequently determined that BPC had not completed work on its assessment of high- and moderate-energy line breaks and, therefore, the S&L team could reach no conclusions concerning the design for protection against the dynamic effects of postulated rupture of piping.

In reply to IDVP concerns, BPC agreed to certain enhancements to its Hazards Assessment Program. First, BPC prepared a final hazards assessment project guide to serve as a blueprint of the overall program and a guidance document for verifying that sufficient review and documentation was provided to ensure adequate plant design relative to hazards considerations. In addition to the guidance document, BPC committed to certain enhancements to the ongoing Hazards Assessment Program in the area of formal documentation.

The S&L team reviewed the Final Hazards Assessment Program guide and other documentation enhancements and concluded that the upgraded program satisfied IDVP concerns and that completion of the program would ensure an adequate design.

During its inspection at BPC the week of October 21, 1985, the staff reviewed the project guide to the Final Hazards Assessment Program; Specification No. G-019, Revision 1; and Design Criteria No. D7.3, Revision 3. The project guide had been written in response to the IDVP concerns; however, the design criteria document had been issued during plant design and construction (specifying separation requirements, defining hazards, and providing for hazard reviews), and Specification No. G-019, Revision 1, had been issued on February 28, 1985, providing for walkdowns before turnover of the various rooms. The staff also reviewed a number of separation review data sheets completed in accordance with these design documents. Separation review data sheets performed both before and after the IDVP were reviewed so that the staff could evaluate both the implementation and effectiveness of the enhancements made to the documents as a result of the IDVP.

In view of the inability of S&L to reach a conclusion regarding the adequacy of hazards protection, the applicant, at the staff's request, agreed to provide a verification of the hazards program. This verification consisted of a review by PSE&G of the separation review data sheets to ensure that all areas of the plant have been addressed as well as a detailed review, on a sampling basis, to verify technical requirements are being met. Additionally, independent walkdowns were performed where deemed necessary. As a result of the review and verification done to date, PSE&G has confirmed that the plant is adequately designed and meets FSAR commitments.

On the basis of the calculations and documents reviewed, as well as resolution of the concerns developed by S&L, the staff concludes that an effective ongoing program is in place, that the enhancements by the IDVP program have been implemented, and on satisfactory completion, the program will address the concerns raised by the IDVP as well as meet its intended objective.

17.5.4.4 Trend Analysis

The staff reviewed the various trends identified in the IDVP report during the inspection at BPC from October 21 to 25, 1985. The staff discussed the implications of the various trends with both BPC and PSE&G personnel, as well as reviewing documents related to these trends. On the basis of this review, the following are specific staff comments related to the major trends identified in the IDVP report:

Undocumented Engineering Judgment

On the subject of undocumented engineering judgment associated with specific engineering calculations, the staff notes that a large number of deficiencies identified by the IDVP were in the civil/structural discipline. This is not surprising in that many of the calculations in this discipline were performed at the beginning stages of the Hope Creek project (some as early as 12 years ago), when the standards against which calculations were judged for adequacy

were quite different. Furthermore, it was noted that 89 engineering calculations were developed or revised by BPC as a result of the IDVP. In many cases, BPC did not agree that these calculations required revision or clarification to substantiate many of the judgments questioned by the IDVP. Nevertheless, BPC revised calculations or performed additional calculations, as necessary, to substantiate the judgments. As acknowledged by the IDVP report, in no case was a design or hardware change necessary to resolve these issues. Hence, the number of observations of this type is not particularly significant.

Bechtel and PSE&G intend to address the subject of calculation adequacy through two mechanisms. First, the existing as-built programs required that a large percentage of the project calculations be reviewed and updated to reflect the as-built configuration of the plant. The procedures that govern these as-built programs have been clarified to incorporate concerns raised by S&L. This will help ensure that all calculations falling under the various as-built programs are technically adequate. Second, as part of the calculation turnover process from Bechtel to PSE&G, PSE&G reviewers will correct any other inconsistencies that are identified. This process includes a formal turnover of calculations, by engineering discipline, to PSE&G engineering personnel. Before the turnover, a technical review of the calculation is done by PSE&G engineering personnel, and any identified discrepancies or inconsistencies are corrected. At the time of turnover, an additional review is made by PSE&G to ensure references are included, complete, and legible.

In view of the above information and the staff's own review of a large number of the calculations in question, the staff concurs with PSE&G's proposed action in this matter and considers it to be sufficient in addressing this trend. The staff considers this matter closed.

Design Procedures

With regard to design procedures, the IDVP expressed concern regarding a lack of procedures or procedural detail. The staff noted that nearly all of the ORs in this area were associated with four ongoing programs: hazards, as-built reconciliation, support reassessment, and civil load verification. The hazards program alone accounted for 13 of 31 procedural ORs. Although the procedural changes that resulted from the IDVP observations were definite enhancements to these programs, the staff does not consider that this potential trend indicates a breakdown of the design process at Hope Creek.

Nevertheless, BPC has committed that certain Engineering Department procedures that govern activities associated with the ORs classified by the S&L team as having a common cause will be reviewed by the Engineering Department. The purpose of this review will be to identify and, if appropriate, modify elements of the procedures to improve design documentation and interdiscipline coordination as well as other areas identified by the IDVP. BPC has identified 16 procedures to be reviewed.

In view of the staff's observations and the BPC commitment to review certain procedures, the staff considers that this trend has been adequately addressed and considers this matter closed.

FSAR Control

The IDVP noted discrepancies between the FSAR and other design documents in 24 ORs. Consequently, FSAR control was identified as a possible trend. On reviewing these ORs, however, the staff concluded that a large majority of the discrepancies were either editorial in nature or were enhancements of information already provided. Considering the large amount of information contained in the FSAR of a project such as Hope Creek and the relatively small number of deviations identified, the staff concluded that the FSAR control was acceptable because none of the design discrepancies violated an FSAR commitment or regulatory requirement. However, the applicant has agreed to review the FSAR change control process to determine if improvement can be made. On the basis of the number and nature of ORs in this area, the staff concludes this is not a significant trend and considers PSE&G's action relative to FSAR control sufficient. The staff considers this matter closed.

Drawing Inconsistencies

With regard to drawing inconsistencies, the IDVP report stated: "Considering the large number of drawings reviewed during the IDVP, the inconsistencies found do not appear to be significant." PSE&G has agreed that, in addition to correcting specific findings, it would continue to ensure that procedures governing drawing control are followed.

The staff has reviewed the specific ORs of the IDVP and has reviewed a large number of drawings during its monitoring of IDVP activities. The staff concurs with the conclusions of the IDVP and PSE&G and considers the discrepancies in this area relatively insignificant. The staff considers the action taken in this matter to be sufficient and additional action is not required. Consequently, this matter is considered closed.

Specifications

The final identifiable trend was associated with discrepancies in equipment design specifications. Specifically, the IDVP identified problems related to specifications that contained undocumented engineering judgments, inadequate design details, and inconsistencies in design input. After considering this matter, the IDVP team stated: "As a result of the resolution achieved for each OR in this area, the S&L team concluded that the discrepancies and inconsistencies found in specifications have had no effect on the design adequacy of HCGS."

The staff has reviewed the matter of specification sufficiency in detail. The staff agrees that, in several instances, specifications could have been more specific and included more detail. At the same time, a large spectrum of design specifications has been observed by the staff in performing design reviews, all of which are sufficient to achieve their objectives. With regard to the Hope Creek project, specifications were prepared that gave wide latitude to the manufacturer or vendor to ensure that ASME Code or other requirements were met. However, a detailed review of the specifications submitted by the vendors was performed by BPC before acceptance. In each instance reviewed by the staff, the BPC process was successful in producing equipment or components that satisfied the design commitments. Consequently, the staff does not believe this area to be of particular concern.

PSE&G has stated that specifications to be used as design documents would be updated and kept current, particularly where new designs are required in the future. The staff believes this action to be sufficient, and in view of the discussions above, the staff considers this matter closed.

17.5.4.5 NRC Staff Conclusions

PSE&G engaged Sargent & Lundy Engineers to conduct an Independent Design Verification Program of the design activities of Hope Creek Generating Station. The plan for this program was accepted by the NRC staff. The program involved a substantial effort that evaluated numerous documents and expended a large number of person-hours. The NRC staff closely monitored the program at all stages of its activities. On the basis of the reviews and inspections conducted by the NRC staff to monitor the program, detailed in this supplement, the staff concludes that the IDVP provides the additional confidence sought by the staff to ensure that the design of Hope Creek meets the FSAR commitments and regulatory requirements. Furthermore, the staff considers BPC's implementation of the design process for Hope Creek thorough and comprehensive. This conclusion is based on the staff's experience in reviewing IDVPs and is supported by the fact that an extremely thorough inspection (by S&L) yielded a very small number of significant findings.