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Mr. E. E. Utley
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Dear Mr. Utley:

Subject: Transmittal of Change Pages to DSER

This letter transmits change pages to the Shearon Harris Draft Safety Evaluation Report. The changes were required by the continuing staff review of the Harris FSAR. The sections which have been changed are identified with a line in the right side margin. Each page replaces a previous DSER page, with the extra pages being identified by an alphabetic subscript. Those changes which have generated new open items have been transmitted to you separately.

Please contact the Project Manager for Shearon Harris if you have any questions.

Sincerely,

Original signed by:
George W. Knighton

George W. Knighton, Chief
Licensing Branch No. 3
Division of Licensing

Enclosure:
As stated

cc: See next page

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or damage. The staff finds that no nonseismically supported components within the containment result in gravitational missiles with potentially adverse consequences to safety-related equipment. The staff has reviewed the applicant's analysis and concurs with the applicant's assumptions and evaluation for potential missiles inside containment.

The applicant has analyzed the potential for the reactor coolant pump flywheel to become a missile source as a result of flywheel failures, in accordance with the guidelines of RG 1.14. The applicant's analysis evaluated the integrity of the flywheel under assumed overspeed conditions of the pump as a result of pipe break at the pump discharge. The analysis verified that failure of the flywheel does not occur and thus it is not a postulated missile source. (See Section 5.4.1.1 of this SER for further discussion of reactor coolant pump flywheel integrity and compliance with the criteria of RG 1.14.)

The staff has reviewed the adequacy of the applicant's design to maintain the capability for a safe plant shutdown in the event of internally generated missiles inside containment. Based on the above, the staff concludes that through the use of barriers, separation, and equipment design, the design is in conformance with the requirements of GDC 4 with respect to missile protection and is, therefore, acceptable. The design of the facility for providing protection from internally generated missiles inside containment meets the acceptance criteria of SRP 3.5.1.2.

3.5.1.3 Turbine Missiles

The staff has reviewed the Shearon Harris facility with regard to the turbine missile issue and concluded that the probability of unacceptable damage to safety-related systems and components due to turbine missiles is acceptably low (i.e., less than 10^{-7} per year), provided that the turbine missile generation probability is maintained to be 10^{-5} per reactor year or less for the life of the plant by an acceptable maintenance program. In reaching this conclusion, the staff has factored into consideration the unfavorable orientation of the turbine generator.

The staff considers the turbine missile issue as a confirmatory item if the applicant does the following:

(1) submits for NRC approval, within 3 years of obtaining an operating license, a turbine system maintenance program based on the manufacturer's calculations of missile generation probabilities, or

(2) volumetrically inspects all low pressure turbine rotors at the second refueling outage and every other (alternate) refueling outage thereafter until a maintenance program is approved by the NRC staff, and

(3) conducts turbine steam valve maintenance (following initiation of power output) in accordance with present NRC recommendations as stated in SRP 10.2 (NUREG-0800).

3.5.1.4 Missiles Generated by Natural Phenomena

The tornado missile spectrum was reviewed in accordance with SRP 3.5.1.4 (NUREG-0800). Conformance with the acceptance criteria formed the basis for the staff evaluation of the tornado-missile spectrum with respect to the applicable regulations of 10 CFR 50.

The portions of the SRP review procedures concerning the probability per year of damage to safety-related systems as a result of missiles were not used in

4 REACTOR

4.1 Introd'iction

4.2 Fuel Design

The Harris fuel assembly described in the FSAR is a 17x17 array of fuel rods having a diameter of 0.374 in. This design will be referred to as the standard fuel assembly (SFA) in the paragraphs below.

FSAR Section 4.2 presents the design bases for the SFA. For the Westinghouse analysis, plant design conditions are divided into four categories of operation that are consistent with traditional industry classification (ANSI N18.2-1973 and N-212-1974): Condition I is Normal Operation; Condition II, Incidents of Moderate Frequency; Condition III, Infrequent Incidents; and Condition IV, Limiting Faults. Fuel damage is related to these conditions of operation, which are coupled to the fuel design bases and design limits. The subsections of the design bases section address such topics as (1) cladding, (2) fuel material, (3) fuel rod performance, (4) spacer grids, (5) fuel assemblies, (6) reactivity control and burnable poisons,* and (7) testing, irradiation, and surveillance. As part of the discussion of the cladding design bases, material and mechanical properties, stress-strain limits, vibration and fatigue, and chemical properties are also presented. A similar approach is taken for the other major subtopics.

The staff review and safety evaluation follow SRP 4.2. The objectives of this fuel system safety review are to provide assurance that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. "Not damaged" is defined as meaning that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements GDC 10 and the design limits that accomplish this are called specified acceptable fuel design limits (SAFDLs). "Fuel rod failure"

*Items (1) through (6) are "core components."

(1) The applicant must evaluate the loose parts monitoring system for conformance to RG 1.133 and commit to supply a report describing operation of the system hardware and implementation of the loose parts detection program. A sample table of contents for this report is given in Figure 4.1.

(2) The applicant must submit item-by-item responses to the documentation required by NUREG-0737.

4.5 Reactor Materials

4.5.1 Control Rod Drive Structural Materials

The applicant has not demonstrated the acceptability of the materials used. The staff is concerned that the yield strength of austenitic stainless steels may exceed 90,000 psi. Moreover, the applicant has not discussed the aging and tempering treatments of precipitation hardening and martensitic steels in a way that would allow the staff to evaluate their acceptability. Proprietary alloys were listed rather than the ASME Code specifications.

The controls imposed upon the austenitic stainless steel of the mechanisms conform to most of the recommendations of RGs 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," and 1.44, "Control of the Use of Sensitized Stainless Steel." Where the recommendations of these RGs were not followed, the alternative approaches taken by the applicant have been reviewed by the staff and are acceptable. Cleaning and cleanliness controls are in accordance with ANSI Standard N 45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," and RG 1.37, "Quality Assurance Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

4.5.2 Reactor Internals Materials

The staff is concerned that the yield strength of austenitic steels may exceed 90,000 psi. Furthermore, AMSE Code Case 1618 is endorsed by the staff in RG 1.85, "Materials Code Case Acceptability, ASME Section III, Division 1," with limitations. The applicant should identify the materials used based on Code Case and address the limitations, if applicable. The applicant should determine if austenitic stainless steels with yield strength exceeding 90,000 psi are used, and, if they are used, provide justification.

4.6 Functional Design of Reactivity Control Systems

The functional design of reactivity control systems was reviewed in accordance with SRP 4.6 (NUREG-0800). Conformance with the acceptance criteria formed the basis for the staff evaluation of the functional design of reactivity control systems with respect to the applicable regulations of 10 CFR 50.

The functional designs of the reactivity control systems for the facility have been reviewed to confirm that they meet the various reactivity control conditions for all modes of operation. These are

(1) the capability to operate in the unrodded, critical, full-power mode throughout plant life

(2) the capability to vary power level from full power to hot shutdown and ensure control of power distributions within acceptable limits at any power level

(3) the capability to shut down the reactor in a manner sufficient to mitigate the effects of postulated events discussed in Section 15 of this SER

The peak primary system pressure following the worst transient is limited to the ASME Code allowable value (110% of the design pressure) with no credit taken for nonsafety-grade relief systems. The Shearon Harris plant was assumed to be operating at design conditions (102% of rated power) and the reactor is shut down by a high pressurizer pressure trip signal. The calculated pressure is less than 110% of design pressure.

Overpressure protection during low-temperature operation of the plant is provided by two (of three) PORVs and RHR suction relief valves in conjunction with administrative controls.

The applicant has met GDC 15 and 31 and Appendix G because the guidelines of BTP RSB 5-2 have been implemented. In addition, the applicant has incorporated into his design the recommendations of Task Action Plan Items II.D.1 and II.D.3 of NUREG-0718 and NUREG-0737.

5.2.3 Reactor Coolant Pressure Boundary Matrials

Because the FSAR does not provide adequate information, the staff cannot conclude that the plant design is acceptable and meets the requirements of 10 CFR 50.

The materials used for construction of RCPB components have been identified by specification and found in conformance with the requirements of Section III of the ASME Code. Compliance with the above Code provisions for material specifications satisfies the quality standards requirements of GDC 1 and 30, and 10 CFR 50.55a.

The materials of construction of the RCPB exposed to the reactor coolant have been identified, and all of the materials are compatible with the primary coolant water, which is chemically controlled in accordance with appropriate technical specifications. This compatibility has been proven by extensive testing and satisfactory performance. This includes conformance with most of the recommendations of RG 1.44, "Control of Sensitzed Stainless Steel." Where the recommendations of these RGs were not followed, the alternative approaches taken have been reviewed by the staff and are acceptable.

General corrosion of all material except unclad carbon and low alloy steel will be negligible. For these materials, conservative corrosion allowances have been provided for all exposed surfaces in accordance with the requirements of the ASME Code, Section III. The evidence of compatibility with the coolant and compliance with the Code provisions described above satisfies the requirements of GDC 4 regarding compatibility of components with environmental conditions.

The materials of construction of the RCPB are compatible with the thermal insulation used in these areas and are in conformance with recommendations of RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels." Conformance with the above recommendations satisfies the requirements of GDC 14 and 32 regarding prevention of failure of the RCPB.

The ferritic steel tubular products and the tubular products fabricated from austenitic stainless steel have been found to be acceptable by nondestructive examinations in accordance with the provisions of the ASME Code, Section III. Compliance with these Codes requirements satisfies the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a.

Fracture toughness of the RCPB components is discussed in Section 5.3.1.

The controls imposed on welding preheat temperatures for welding ferritic steels are in conformance with most of the recommendations of RG 1.50, "Control of Preheat Temperature for Welding Low Alloy Steels." The alternative approaches taken by the applicant have been reviewed and are acceptable to the staff. These controls provide reasonable assurance that cracking of components made from low alloy steels will not occur during fabrication and minimize the possibility of subsequent cracking due to residual stresses being retained in the weldment. These controls satisfy the quality standards requirements of GDC 2 and 30 and 10 CFR 50.55a.

For steam generators 3 and 4, the controls imposed on electroslag welding of ferritic steels are in accordance with the recommendations of RG 1.34, "Control of Electroslag Weld Properties," and provide assurance that welds fabricated by the process will have high integrity and sufficient toughness to

adequate safety margins during operating, testing, maintenance, and postulated accident conditions. The applicant should demonstrate that the electroslag welds in steam generators 1 and 2 are of quality equivalent to the electroslag welds in steam generators 3 and 4.

The controls imposed on welding ferritic and austenitic steels under conditions of limited accessibility are in accordance with most of the recommendations of RG 1.71, "Welder Qualification for Areas of Limited Accessibility." Alternative approaches taken by the applicant provide adequate assurance of weldment integrity, are acceptable to the staff, and provide assurance that proper requalification of welders will be required in accordance with the welding conditions. These controls satisfy the quality standards requirements of GDC 1 and 50 and 10 CFR 50.55a. The controls imposed on weld cladding of low-alloy steel components by austenitic stainless steel are not in strict accordance with the recommendations of RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components." Accordingly, the applicant has to demonstrate that low-alloy steel components made without fine grain melt practice and stainless steel clad with a high heat input process do not have underclad fissures in excess of the criteria of ASME Section III, Subsection NB, i.e., no cracks (fissures).

The applicant has not addressed limiting austenitic stainless steel used in RCPB components to a maximum yield strength of 90,000 psi. This should be confirmed to provide assurance of RCPB integrity.

The controls to avoid stress corrosion cracking in reactor coolant pressure boundary components constructed of austenitic stainless steels conform to most of the recommendations of RGs 1.44, "Control of the Use of Sensitized Stainless Steel," and 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems Associated Components of Water Cooled Nuclear Plants." The alternative approaches taken by the applicant were reviewed by the staff and are acceptable.

The controls followed during material selection, fabrication, examination, protection, sensitization, and contamination provide reasonable assurance that the RCPB component of austenitic stainless steels will be in a metallurgical condition that minimizes susceptibility to stress corrosion cracking during service. These controls meet the requirements of GDC 4 regarding compatibility of components with environmental conditions and requirements of GDC 14 regarding prevention of leakage and failure of the RCPB.

The controls imposed during welding of austenitic stainless steels in the RCPB are in accordance with most of the recommendations of RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." The alternative approaches taken by the applicant were reviewed by the staff and are acceptable.

These controls provide reasonable assurance that welded components of austenitic stainless steel will not develop microfissures during welding and will have high structural integrity. These controls meet the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a and satisfy the requirements of GDC 14 that relate to prevention of leakage and failue of the RCPB.

5.2.4 Reactor Coolant Pressure Boundary Inspection and Testing

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

The reactor coolant pressure boundary (RCPB) leakage detection systems were reviewed in accordance with SRP 5.2.5 (NUREG-0800). An audit review of each of the areas listed in the Areas of Review portions of the SRP section was performed according to the guidelines provided in the SRP Review Procedures. Conformance with the acceptance criteria formed the basis for the staff evaluation of the RCPB leakage detection systems with respect to the applicable regulations of 10 CFR 50.

A limited amount of leakage is to be expected from components forming the RCPB. Means are provided for detecting and identifying this leakage in accordance with the requirements of GDC 30. Leakage is classified into two types--identified and unidentified. Components such as valve stem packing, pump shaft seals, and flanges are not completely leaktight. Because this leakage is expected, it is considered identified leakage and is monitored, limited, and separated from other leakage (unidentified) by directing it to closed systems as identified in the guidelines of position C.1 of RG 1.45.

The containment airborne radioactivity monitors are seismic Category I and are located in flood- and tornado-protected structures, thus meeting the requirements of GDC 2 and the guidelines of RG 1.29, Positions C.1 and C.2. They are also testable, and the limiting conditons are specified as identified in the guidelines of Positions C.8 and C.9 of RG 1.45.

Based on the above, the staff concludes that the RCPB leakage detection systems are diverse and provide reasonable assurance that primary system leakage (both identified and unidentified) will be detected and meet the requirements of GDC 2 and 30 with respect to protection against natural phenomena and provisions for RCPB leak detection and identification, and the guidelines of RG 1.29, Positions C.1 and C.2, and RG 1.45, Positions C.1 through C.9, with respect to seismic classification and RCPB leakage detection system design. They are, therefore, acceptable. The design of the facility for reactor boundary leakage detection meets the acceptance criteria of SRP 5.2.5.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

The staff concludes that, with few exceptions, the reactor vessel materials are acceptable and meet the requirements of GDC 1, 4, 14, 30, 31, and 32; the materials testing and monitoring requirements of Appendices D, G, and H of 10 CFR 50; and the requirements of 10 CFR 50.55a. The materials used for construction of the reactor vessel and its appurtenances have been identified by specification and found to be in conformance with Section III of the ASME Code. Special requirements of the applicant regarding control of residual elements in ferritic materials have been identified and are considered acceptable. Compliance with the above Code provisions for material specifications satisifies the quality standards requirements of GDC 1 and 30, and 10 CFR 50.55a. This reactor vessel and its appurtenances were fabricated/manufactured by coventional process that have been used extensively in reactor vessels.

The applicant has certified that the materials and fabrication processes comply with the requirements of Section III of the ASME Code, and therefore are considered acceptable. Compliance with these Code provisions meets the quality standards requirements of GDC 1 and 30, and 10 CFR 50.55a.

Conventional methods that have been used extensively in nuclear reactors also were used for the nondestructive examination of the Shearon Harris reactor vessel and its appurtenances. Accordingly, the quality standards requirements of GDC 1 and 30, and 10 CFR 50.55a are satisfied.

When components of ferritic steels are welded, Code controls are supplemented by conformance with the recommendations of RGs. The controls imposed on welding preheat temperatures conform with most of the recommendations of RG 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel." The alternative approaches taken by the applicant were reviewed by the staff and are acceptable. These controls provide reasonable assurance that cracking of components made for low-alloy steels will not occur during fabrication, and they minimize the potential for subsequent cracking. These controls also satisfy the quality standards requirements of GDC 1 and 30, and 10 CFR 50.55a.

Electroslag welding was not used for fabrication of the reactor vessel or its appurtenances; thus RG 1.34, "Control of Electroslag Weld Properties," is not applicable.

The controls imposed during weld cladding of ferritic steel components are not in strict conformance with the recommendations of RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components." Accordingly, the applicant must demonstrate that low-alloy steel components made without fine grain melt practice and stainless steel clad with a high heat input weld process do not have underclad fissures in excess of the criteria of ASME Code, Section III, Subsection NB, i.e., no cracks (fissures).

When components of austenitic stainless steels are welded, Code controls also are supplemented by conformance with the recommendations of RGs. The controls imposed on delta ferrite in austenitic stainless steel are in conformance with most of the recommendations of RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." The alternative approaches taken by the applicant have been reviewed by the staff and are acceptable. The controls used provide reasonable assurance that the welds will not contain micro cracks. These controls also satisfy the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a and the requirements of GDC 14 regarding fabrication to prevent RCPB rapidly propagating failure.

The reactor vessel is made of low alloy steel, and there are no stainless steel appurtenances to the reactor vessel that are electroslag welded. Accordingly, RG 1.34 is not applicable.

The controls (during all stages of welding) to avoid contamination and sensitization that could cause stress-corrosion cracking in austenitic stainless steels conform with the recommendations of the RGs. The controls to avoid contamination and excessive sensitization of austenitic stainless steel are in conformance with most of the recommendations of RG 1.44, "Control of the Use of Sensitized Stainless Steel." The alternative approaches taken by the applicant have been reviewed by the staff and are acceptable. The controls used provide assurance that welded components will not be contaminated or excessively sensitized before or during the welding process. These controls satisfy the quality standards requirement regarding material compatibility.

The controls regarding onsite cleaning and cleanliness controls of austenitic stainless steel are in conformance with the recommendations of RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," because the controls used provide assurance that austenitic stainless steel components will be properly cleaned on site. The controls satisfy Appendix B of 10 CFR 50 regarding controls for onsite cleaning of materials and components.

Integrity of the reactor vessel studs and fasteners is ensured by conformance with most of the recommendations of RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs." The alternative approaches taken by the applicant in combination with most of these recommendations satisfy the quality standards requirements of GDC 1 and 30, and 10 CFR 50.55a; the prevention of fracture of the RCPB requirement of GDC 31; and the requirements of Appendix G, 10 CFR 50, as detailed in the provisions of the ASME Code, Sections II and III.

5.3.1.1 Reactor Vessel Materials Fracture Toughness

The staff has reviewed the fracture toughness of ferritic reactor vessel and RCPB materials and the materials surveillance program for the reactor vessel beltline. The acceptance criteria and references that are the basis for this evaluation are in paragraph II.3.a of SRP 5.3.2 and paragraphs II.5, II.6, and II.7 (Appendices G and H, 10 CFR 50) of SRP 5.3.1 (NUREG-0800).

GDC 31 requires, in part, that the RCPB be designed with sufficent margin to ensure that, when stressed under operating, maintenance, testing, and anticipated transient conditions, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. GDC 32 of requires, in part, that the RCPB be designed to permit an appropriate material surveillance program for the reactor pressure vessel.

The fracture toughness requirements for the ferritic materials of the RCPB are defined in Appendices G and H of 10 CFR 50.

The staff has reviewed the materials selection, toughness requirements, and extent of materials testing conducted by the applicant to provide assurance that the ferritic materials used for pressure-retaining components of the RCPB possess adequate toughness under operating, maintenance, testing, and anticipated transient conditions.

Reactor vessels at Shearon Harris were designed to the specifications of the 1971 Edition of the ASME Code Section III, "Rules for Construction of Nuclear Power Plant Components," including Addenda through Winter 1971. The reactor coolant loop piping and reactor coolant pumps and valves were designed to the specifications of the 1971 Edition of the Code, Section III, including Addenda through Summer 1973, Summer 1972, and Summer 1972, respectively. Based on the January 27, 1978 construction permit date, 10 CFR 50.55a requires that the ASME Code editions and addenda applied to the pressure vessels be no earlier than those of the Summer 1972 Addenda of the 1971 Edition. 10 CFR 50.55a also requires that the ASME Code edition and addenda applied to the piping, pumps, and valves that are part of the RCPB shall be no earlier than those of the Winter 1972 Addenda of the 1971 Edition. The design and construction of the RCPB components of Shearon Harris are, therefore, not in compliance with the requirements of 10 CFR 50.55a. However, the staff will evaluate the applicant's RCPB materials according to Appendix G of 10 CFR 50, which will ensure that material properties are equivalent or superior to those specified in 10 CFR 50.55a.

5.3.1.1.1 Compliance with 10 CFR 50.55a

5.3.1.1.2 Compliance with Appendix G, 10 CFR 50

The staff has evaluated the applicant's FSAR to determine the degree of compliance with fracture toughness requirements of Appendix G, 10 CFR 50. The staff's evaluation indicates that the applicant complied with Appendix G, 10 CFR 50, except for paragraphs I.A, I.C, III.B.1, III.B.4, III.C.2, and IV.A.1, which will remain open items until the applicant submits the requested data. The staff evaluation of each of these areas follows.

5.3.1.1.3 Compliance with Appendix H, 10 CFR 50

The materials surveillance program at Shearon Harris Units 1 and 2 will be used to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region, that result from exposure to neutron irradiation and the thermal environment.

Under the Shearon Harris surveillance program, fracture toughness data will be obtained from material specimens that are representative of the limiting base, weld, and heat-affected-zone materials in the beltline region. These data will permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life.

The fracture toughness properties of reactor vessel beltline materials must be monitored throughout the service life of Shearon Harris Units 1 and 2 by a materials surveillance program that meets the requirements of ASTM E-185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," and Appendix H of 10 CFR 50.

The staff has evaluated the applicant's information for degree of compliance to these requirements. Based on its evaluation, the staff concludes that the applicant has met all the requirements of Appendix H, 10 CFR 50, with the exception of Paragraphs II.B and II.C.3.

Paragraph II.B of Appendix H requires that the surveillance program comply with ASTM E-185-73. ASTM E-185-73 requires that the surveillance capsule materials be removed from beltline reactor vessel base metal and weld samples that represent the material that may limit operation of the reactor vessel during its lifetime. The applicant has not identified from which samples the material surveillance specimens were removed. To demonstrate compliance with Paragraph II.B of Appendix H, the applicant must

(1) Provide for each base metal and heat-affected-zone surveillance specimen the specimen type, the orientation of the specimen relative to the principal rolling direction of the plate, the heat number, the component code number

from which the samples was removed, chemical composition (especially the copper and phosphorus contents), and the heat treatment received by the sample material.

(2) Provide for each weld metal surveillance specimen the weld identification from which the sample was removed, the weld wire type and heat identification, flux type and lot identification, weld process, and heat treatment used for fabrication of the weld sample.

(3) Provide a sketch which indicates the azimuthal location for each capsule relative to the reactor core.

Paragraph II.C.3 requires that the basis for the withdrawal schedule of the surveillance capsule is the adjusted reference temperature at the end to the service life of the reactor vessel. The applicant has indicated in FSAR Section 5.3.1.6 that there will be six surveillance capsules in the reactor vessel surveillance program, but has not indicated the estimated reactor vessel end-of-life fluence, the lead factors, and the withdrawal schedule for each capsule. The applicant must supply this information in order for the staff to determine whether the applicant complies with Paragraph II.C.3 of Appendix H.

5.3.1.1.4 Conclusions for Compliance with Appendices G and H, 10 CFR 50

Based on its evaluation of compliance with Appendices G and H, 10 CFR 50, the staff concludes that the applicant has not supplied sufficient information to meet all the fracture toughness requirements of Appendix G and surveillance program requirements of Appendix H. The areas in which additional information is required include Paragraphs I.A, I.C, III.B.1, III.B.4, III.C.2, and IV.A.1, of Appendix G, and Paragraphs II.B and II.C.3 of Appendix H; these items will remain open until the applicant submits the necessary information. Appendix G, "Protection Against Nonductile Failure," of Section III of the ASME Code will be used, together with the fracture toughness test results required by Appendices G and H, 10 CFR 50, to calculate the RCPB pressure-temperature limitations for Shearon Harris.

The staff has reviewed the material, fabrication, design, and inspection aspects of the pump flywheels for compliance with RG 1.14. The staff has concluded that the structural integrity of the flywheels is adequate to withstand the forces imposed by overspeed transients without the loss of function, and the integrity will be verified periodically by inspection to ensure that the integrity is maintained.

5.4.2 Steam Generators

5.4.2.1 Steam Generator Materials

The staff concludes that the steam generator materials specified are acceptable and meet the requirements of GDC 1, 14, 15, and 31, and Appendix B to 10 CFR 50.

The applicant has met the requirements of GDC 1 with respect to codes and standards by ensuring that the materials selected for use in Class 1 and Class 2 components will be fabricated and inspected in conformance with codes, standards, and specifications acceptable to the staff. Welding qualification, fabrication, and inspection during manufacture and assembly of the steam generator will be done in conformance with the requirements of Sections III and IX of the ASME Code.

The requirements of GDC 14 and 15 have been met to ensure that the reactor coolant boundary and associated auxiliary systems have been designed, fabricated, erected, and tested so they have an extremely low probability of abnormal leakage, of rapid failure, and of gross rupture during normal operation and anticipated operational occurences.

The primary side of the steam generator is designed and fabricated to comply with ASME Class 1 criteria as required by the staff. The secondary side pressure boundary parts of the steam generators will be designed, manufactured, and tested to ASME Class 1 criteria, although the staff classification is ASME Class 2.

The crevice between the tubesheet and the inserted tube will be minimal because the tube will be expanded to the full depth of insertion of the tube in the tube sheet. The tube expansion and subsequent positive contract pressure between the tube and the tubesheet will preclude a buildup of impurities from forming in the crevice region and reduce the probability of crevice boiling.

A flow distribution plate located below the preheat section encourages recirculating flow to sweep the tubesheet before turning upward on the hot leg side to flow axially through the tube bundle. This plate, in addition to the bottom preheater baffle plate, serves to separate the tubesheet from the colder feedwater entering at a preheat section.

The requirements of GDC 31 have been met with respect to the fracture toughness of the ferritic materials since the pressure boundary materials of ASME Class 1 components of the steam generators will comply with the fracture toughness requirements and tests of subarticle NB-2300 of Section III of the Code. The materials of the ASME Class 2 components of the steam generators will comply with the fracture toughness requirements of Subarticle NC-2300 of Section III of the Code.

The requirements of Appendix B of 10 CFR 50 have been met because the onsite cleaning and cleanliness controls during fabrication conform to the recommendations of RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plant." The controls placed on the secondary coolant chemistry are in agreement with staff technical positions.

Reasonable assurance of the satisfactory performance of the steam generator tubing and other generator materials is provided by (1) the design provisions and the manufacturing requirements of the ASME Code, (2) rigorous secondary water monitoring and control, and (3) the limiting of condenser inleakage. The controls described above, combined with conformance with applicable standards, staff positions, and RGs, constitute an acceptable basis for meeting in part the requirements of GDC 1, 14, 15, and 31, and Appendix B, to 10 CFR 50.

5.4.3 Deleted

5.4.4 Deleted

5.4.5 Deleted

5.4.6 Reactor Core Isolation Cooling

5.4.7 Residual Heat Removal System

FSAR Section 5.4.7 has been reviewed in SRP 5.4.7 (NUREG-0800). A review of each of the areas listed in the Areas of Review portion of the SRP was performed according to the SRP Review Procedures. Conformance with the acceptance criteria, except as noted below, formed the basis for concluding that the design of the facility for residual heat removal is acceptable.

The residual heat removal system (RHRS) is designed to remove heat from the reactor coolant system after the system temperature and pressure have been reduced to approximately 350°F and 425 psig, respectively. The RHR system is capable of reducing the reactor coolant temperature to the cold shutdown condition and maintain this temperature until the plant is started up again.

The RHRS operates in the following modes:

(1) <u>Emergency Core Cooling System (ECCS), Injection Mode</u>

Functions in conjunction with the high head portion of the ECCS to provide injection of borated water from the refueling water storage tank (RWST) into

6 ENGINEERED SAFETY FEATURES

6.1 Engineered Safety Features Materials

6.1.1 Engineered Safety Features

The staff concludes that there is not enough information provided in the FSAR regarding the engineered safety features materials to determine that these materials are acceptable and meet the requirements of GDC 1, 4, 14, 31, 35, and 41; Appendix B of 10 CFR 50; and 10 CFR 50.55a.

The applicant has not yet met GDC 1, 14, and 31 and 10 CFR 50.55a by providing for the extremely low probability of leakage, of rapidly propagating failure, and of gross rupture. The applicant should address the adequacy of the fracture toughness of components made of ferritic steels, considering their function and the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents.

The materials selected for the engineered safety features satisfy Appendix I of Section III of the ASME Code, and Parts A, B, and C of Section II of the Code, as well as the staff position that the yield strength of cold-worked stainless steels shall be less than 90,000 psi. The controls on the use and fabrication of the austenitic stainless steel of the systems satisfy most of the requirements of RGs 1.31 and 1.44. The alternative approaches taken by the applicant have been reviewed and are acceptable to the staff. Fabrication and heat treatment practices performed accordingly provide assurance that the probability of stress corrosion cracking will be reduced during the postulated accident time interval.

Conformance with the codes, RGs, and with the staff positions mentioned above constitutes an acceptable basis for meeting the requirements of GDC 1, 4, 14, 35, and 41; Appendix B to 10 CFR 50; and 10 CFR 50.55a to which the systems are to be designed, fabricated, and erected so that the systems can perform their function as required.

The applicant has met, in part, the requirements of GDC 1, 14, and 31 and Appendix B to 10 CFR 50 to ensure that the reactor coolant boundary and associated auxiliary systems have an extremely low probability of leakage, of rapidly propagating failures, and of gross rupture. The controls placed on concentrations of leachable impurities in nonmetallic thermal insulation used on components of the engineered safety features are in accordance with the requirements of RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels." Compliance with the requirements of RG 1.36 forms a basis for meeting the requirements of GDC 1, 14, and 31.

The controls placed on component and system cleaning are in accordance with RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and provide a basis for finding that the components and systems have been protected against damage or deterioration by contaminants as stated in the cleaning requirements of Appendix B to 10 CFR 50.

6.1.1.1 Post-Accident Emergency Cooling Water Chemistry

The post-accident emergency cooling water chemistry has been reviewed in accordance with SRP 6.1.1 (NUREG-0800).

This review is related to providing and maintaining the proper pH of the containment sump water and recirculated containment spray water following a design-basis accident to reduce the likelihood of stress corrosion cracking of austenitic stainless steel.

The applicant will use borated water with a concentration of 4000 ppm boron from the refueling water storage tank (RWST) during the initial injection phase of containment spray. The borated water will be mixed with a 30% by weight sodium hydroxide solution from the containment spray additive tank.

The resulting solution will have a pH greater than 7, and will drain to the containment sump. Mixing is achieved as the solution is continuously recirculated from the sump to the containment spray nozzles during the recirculation phase of containment spray.

The staff evaluated the pH of the water (mixture of RWST and sodium hydroxide solution) in the containment sump. The staff verified by independent calculations that sufficient sodium hydroxide is available to raise the containment sump water pH above the minimum 7.0 level to reduce the probability of stress-corrosion cracking of austenitic stainless steel components. The removal effectiveness of the chemical additive for fission products in containment is reviewed in Section 6.5.2 of this SER. The staff will review the surveillance requirements in the plant Technical Specifications to verify that sufficient sodium hydroxide is maintained in the containment spray additive tank.

The proposed program monitors the critical parameters to inhibit steam generator corrosion and tube degradation. The limits and sampling schedule for these parameters are established for steam generator blowdown and feedwater/condensate under power operation, startup, shutdown, wet layup conditions, and hot standby; these follow closely the recommendation of the NSSS vendor. The control points for the critical parameters and the process sampling points have been identified. The analytical techniques to be implemented for measuring the values of the critical parameters conform to the recommendations of the NSSS vendor and the ASTM. Chemistry data logs are reviewed by the Environmental and Chemistry Supervisor; significant parameter trends are periodically reviewed by plant management.

The staff has reviewed and evaluated the applicant's secondary water chemistry program in accordance with SRP 5.4.2.1, BTP MTEB 5-3, Revision 2 (NUREG-0800) and finds that it

(1) Is capable of reducing the probability of abnormal leakage in the reactor coolant pressure boundary by inhibiting steam generator corrosion and tube degradation, and thus meets the requirements of GDC 14.

(2) Adequately addresses all of the program criteria delineated in the staff position on control and monitoring of secondary water.

(3) Is based on the NSSS vendor's recommended steam generator water chemistry program.

(4) Monitors the secondary coolant purity in accordance with BTP MTEB 5-3, Revision 2, and thus meets Acceptance Criterion 3, "Steam Generator Materials," Revision 1, of SRP 5.4.2.1.

(5) Monitors the water quality of the secondary water in the steam generators to detect potential condensor cooling water inleakage to the condensate, and thus meets Position II.3.f(1) of BTP MTEB 5-3, Revision 2.

(6) Describes the methods for control of secondary side water chemistry data and record management procedures and corrective actions for off-control-point chemistry, and thus meets Position II.3.f of BTP MTEB 5-3, Revision 2.

On the bases of its evaluation, the staff concludes that the proposed secondary water chemistry monitoring and control program for Shearon Harris Units 1 and 2 meets (1) the requirements of GDC 14 insofar as secondary water chemistry control ensures primary boundary material integrity, (2) Acceptance Criterion 3 of SRP 5.4.2.1, (3) positions II.2 and II.3 of BTP MTEB 5-3, Revision 2, and (4) the program criteria in the staff's position. It, therefore, is acceptable.

10.3.6 Main Steam and Feedwater Materials

The staff concludes that the Class 2 main steam and feedwater system materials are acceptable and meet the requirements of 10 CFR 50.55a; GDC 1 and 35; and Appendix B to 10 CFR 50. However, Class 3 components are not satisfactorily addressed, and the applicant must provide further information.

The applicant has selected materials for Class 2 and 3 components of the steam and feedwater systems that satisfy Appendix I of Section III of the ASME Code and meet the requirements of Parts A, B, or C of Section II of the Code. The applicant has also met the recommendations of RG 1.85, which describes acceptable Code cases that may be used in conjuction with this industry standard.

Where the Code allowed fracture toughness testing to be optional (as allowed for Class 2 and 3 components), the applicant required fracture toughness testing on Class 2 components and waived fracture toughness testing on Class 3 components. The applicant did not provide a rationale for waiving fracture toughness testing of Class 3 components. Fracture toughness tests and mechanical properties specified by the Code provide reasonable assurance that ferritic materials will have adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture. The applicant must provide a technical rationale for waiving the fracture toughness testing of Class 3 ferritic steel components of the main steam and feedwater systems.

The applicant has met the requirements of RG 1.71, "Welder Qualification for Areas of Limited Accessibility," by meeting the regulatory positions in RG 1.71 or by providing and meeting an alternative to the regulatory positions in RG 1.71 that the staff has reviewed and found acceptable. The onsite cleaning and cleanliness controls during fabrication satisfy the position in RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and the requirements of ANSI Standard N 45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants."

10.4 Other Features

10.4.1 Main Condenser

10.4.2 Main Condenser Evacuation System

10.4.2.1 Summary Description

The main condenser evacuation system (MCES) of each unit consists of two 100% capacity mechanical vacuum pumps that serve the main condenser. At startup, one or both pumps may be operated to evacuate the condenser. Once operating pressure is obtained, one pump is place on standby. At startup and before turbine operation, the noncondensible gases will be discharged directly to the atmosphere in the turbine building area without filtration. With turbine operation, the discharge from the mechanical vaccuum pumps is directed to the turbine building vent stack without filtration.

The noncondensible gases flow to a moisture separator where most of the water vapor is condensed. The condensed water drains to the industrial waste sumps. However, the discharge from these sumps will be directed to the secondary waste system for treatment upon detection of radioactivity by monitor REM-3528. The or provided acceptable alternatives. In addition, the staff selectively reviewed the applicant's FSAR against the SRP acceptance criteria, using the procedures in the SRP. This selective review found the plant acceptable in these areas. Details of the review follow.

12.1.1 Policy Considerations

The applicant provides a management commitment in the corporate health physics policy to ensure that Harris will be designed, constructed, and operated in a manner consistent with RGs 8.8, 8.10, and 1.8. The applicant has identified the specific corporate plan to implement that policy and specified in detail facility and equipment design considerations to ensure its accomplishment. This objective is delineated in the radiation control and protection program and is reached through administrative dose control procedures, adequate work planning, and safe practices in all activities related to the plant's operation. The plant General Manager has the overall responsibility for implementing the ALARA program. He delegates the health physics support functions to the station Radiation Control Supervisor, who is responsible for maintaining the health physics program. The ALARA specialist assists the Radiation Control Supervisor in this task and has the specific responsibility and authority for monitoring the program to ensure that the radiation protection program maintains doses ALARA. He will review dose records and will compare results from past experience to assess the effectiveness of the ALARA effort. Station management will also review these records and, in accordance with ALARA program implementation components, seek to identify exposure areas and excessive exposure by job categories that indicate dose trends and the need for improvement in plant procedures, health physics procedures, or plant equipment. These policy considerations meet the criteria of RG 8.8 and NUREG-0800 and are therefore acceptable.

12.1.2 Design Considerations

The objective of the plant's radiation protection design is to maintain individual and collective doses to plant workers--including construction workers--and to memhers of the general public ALARA and to maintain individual ensuring that the applicant had either committed to following the criteria of the regulatory guides and staff positions referenced in SRP 12.3 or provided acceptable alternatives. In addition, the staff selectively reviewed the applicant's FSAR against the specific areas of review and review procedures identified in the SRP. This review found the plant acceptable in these areas. Details of the review follow.

12.3.1 Facility Design Features

The applicant has addressed facility and equipment design considerations, planning and procedure programs, and techniques and practices employed in the overall design for maintaining doses ALARA. The FSAR was reviewed with respect to

(1) the description of the equipment design to be used for ensuring that occupational exposure will be ALARA

(2) information concerning implementation of RG 8.8, Section C.2

(3) the description of any special protective features that use shielding, geometric arrangement, or remote handling to reduce occupational exposure

To maintain occupational doses ALARA, the applicant has designed his facilities, to the extent practicable, so that systems and components handling high activity fluids are in controlled areas, separated from uncontrolled areas by shielded walls. Equipment and components that require manual operation, or may need servicing and instrumentation requiring visual inspection, are located in the lowest possible radiation zone. When it is impractical to do this, such items are designed to that they may be removed to a low radiation zone. Steam generator and pressurizer manways are sized to facilitate the entry and exit of personnel wearing protective clothing. Valves, pumps, demineralizers, and filters are designed to allow operation, maintenance, and inspection with minimal exposure. To control the production of crud (e.g., ^{60}Co, ^{58}Co), use of hard-facing materials with cobalt content and nickel-based alloys are limited and used only where component reliability require their use. Flush

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and drain connections will enable decontamination of radioactive piping before equipment maintenance is performed, and sample stations are located in low radiation zone areas to minimize personne exposure during sampling. Whenever it is not feasible to install permanent shielding and where shielding may be required, the design philosophy of the applicant (which emphasizes adequate space for ease of motion) would allow portable shielding to be used.

12.3.2 Shielding

The objective of the plant's radiation shielding is to provide protection against radiation for operating personnel both inside and outside the plant and for the general public, during normal operation, anticipated operational occurrences, and accidents. The shielding was designed to meet the criteria of a radiation dose zone system that is based on expected frequency and duration of occupancy. The design of the radiation shielding considers the dose rate criterion for each zone based on maximum access time estimates in each compartment within the zone. The design was reviewed, updated, and modified during all phases of the plant's design and construction. The health physics staff will update entry requirements in accordance with 10 CFR 20.203 or Standard Technical Specification requirements. Shielding analyses were made using accepted codes, models, and assumptions. The basic shielding analysis was performed using computer codes accepted by the staff such as ISOSHLD, MORSE CG, and SPAN-4. Besides limiting exposure to plant personnel, contractors, visitors, and the like, the plant shielding also functions to reduce neutron activation of equipment, piping supports, etc., and to limit radiation damage to equipment and materials to below the specified integrated life dose limits. All concrete shielding in the plant is based on RG 1.69, which provides the guidance on the fabrication and installation of concrete radiation shields. Shielding for protection from neutron and gamma-ray streaming from the annulus between the reactor pressure vessel and the biological shield into occupiable areas inside containment has not been adequately analyzed in the FSAR. Neutron and gamma-ray dose equivalent rates within the various levels of containment, before and after shield installation, should be addressed to ensure that a proposed or existing shield will provide sufficient dose rate reduction to achieve ALARA doses to occupants

while the reactor is at power. This is an open item. Additionally, the applicant has not provided a copy of the design review of plant shielding for spaces that may be occupied during post accident operations, as discussed in Section 12.2.1 of this report. The staff has concluded that the applicant has performed a shielding design review in accordance with the criteria of the SRP, except for that review as specified in NUREG-0737, Item II.B.2, and neutron streaming as described above. These are open items.

12.3.3 Ventilation

The applicant's ventilation systems are designed to provide ventilation air suitable to ensure that plant personnel are not exposed to airborne concentrations exceeding those in 10 CFR 20.103 and that concentrations to which personnel may be exposed meet the requirements of 10 CFR 20.101. In the design of all ventilation systems the applicant intends to meet this objective and maintain exposures ALARA by (1) directing the airflow from areas of lesser potential contamination to areas of greater potential contamination, (2) providing airborne radiation monitoring (3) allowing adequate space around units for servicing and replacement, and (4) providing for ease in maintaining and inplace testing of filters to preclude additional radiation exposure. After initial operation, filters and adsorbers will be tested periodically and the frequency of changeout determined as a result of these tests. The design criteria are in accordance with the guidelines of RG 8.8 and the atmospheric cleanup units conform to RG 1.52 with respect to occupational exposure. The staff concludes that the applicant's ventilation system is designed to maintain personnel exposures at a small fraction of 10 CFR 20 values, meets the criteria of the SRP; and, therefore, is acceptable.

12.3.4 Area Monitoring and Airborne Radioactivity Monitoring Instrumentation

The applicant's area radiation monitoring system is designed to (1) inform operations personnel of radiation levels in areas where area radiation monitoring system (ARMS) units are located, (2) provide warning when abnormal levels occur by audible and visual alarms both locally, in the control room, and in the health physics office (3) warn of equipment malfunction and leaks returning the air to the containment atmosphere. Six portable continuous air monitors (CAMs) are also available. When work is being performed in areas of the primary auxiliary building, waste processing building, and fuel storage building, these CAMs will be used to monitor for particulates and noble gases. In addition, grab samples of particulates, noble gases, and iodines will be taken to ensure appropriate radiation protection. Each detector comprising the airborne radioactivity monitoring system is initially given a primary calibration with typical sources of interest. Secondary standards are counted in reproducible geometry during the primary calibration. These secondary standards will be used in subsequent calibrations and whenever the monitoring systems are maintained or repaired to ensure proper functioning. The frequency of calibration and associated accuracy of the monitoring system will be in accordance with the requirements of ANSI 323, "Radiation Protection Instrumentation Test and Calibration." Because the airborne radioactivity monitoring system monitors for particulates and noble gases and samples for iodine, particulates and iodine can be identified by gamma-ray spectroscopy, if required. However, for gaseous activity, specific radionuclides need not be identified because the dose equivalent of ^{133}Xenon may be used (all the gaseous activity is assumed to be ^{133}Xenon) to determine the dose rate for the concentraton measured. All installed instruments have independent emergency battery power supplies that are activated whenever a power failure occurs. The applicant will comply with the requirements of TMI Item 2.1.8.C (Item III.D.3.3 of NUREG-0737) on improved inplant iodine monitoring by providing equipment to accurately determine the airborne iodine concentrations where plant personnel may be present during an accident.

The objectives and location criteria of the proposed airborne radioactivity monitoring system are in conformance with the guidelines of the SRP except as discussed below.

The SRP states that continuous ventilation monitors are upstream of HEPA filters and should be capable of detecting 10 mpc-hours of particulate and iodine radioactivity from any compartment that has the possibility of containing this activity and that may be occupied by personnel. The applicant has cart-mounted CAMs that can provide this monitoring function. However, six

CAMs may not be sufficient to continuously monitor all the areas of concern of a two-unit plant. Therefore, the applicant should commit to place ventilation monitors upstream of relevant ventilation streams in any routinely occupied area or demonstrate that his alternatives will ensure that the continuous monitors are capable of detecting 10 mpc-hours in any routinely occupied area. This is an open item.

With respect to "ALARA," there is no ALARA standard. The staff has not proposed nor does it plan to propose ALARA radiation levels or limits. The best radiation protection design, procedure, equipment, etc., considering cost/benefit, is applied, and the radiation levels that exist as a result of these actions are "ALARA."

and a personnel decontamination room. The counting room will contain equipment for analysis of alpha-, beta-, and gamma-ray activity from airborne radioactivity samples, smear samples, and radionuclide concentrations in liquid samples. A vendor-supplied whole-body counting system will be located on site or off site, as an alternative or supplement to a Harris system, to determine the radionuclide body burdens, if any, of station personnel. This program will be in parallel with a bioassay program with capability for urine and fecal analysis. A thermoluminescent dosimeter (TLD) reader and associated equipment are on site to enable prompt processing of TLD badges to immediately verify dose.

12.5.3 Equipment and Instrumentation

Continuing evaluation and review of the radiological status of the station will be carried out by health physics personnel so that levels of radiation will be known at all times in areas where personnel are working. Equipment to be used for radiation protection purposes includes portable alpha, beta, gamma, and neutron survey meters. As a result of the staff's review questions, the applicant has added to his portable survey and area radiation monitor instrument inventory instruments that range to 10^4 R/hr. Airborne gaseous, particulate, and iodine samplers and continuous air monitors are available. All portable radiation detection equipment and monitoring systems are state of the art to ensure that inplant personnel receive timely and accurate information. As stated previously, area and airborne radioactivity monitoring equipment incorporates alarm set points to alert workers whenever radiation levels exceed their set point levels. Calibration of these monitors, as well as the portable survey meters, will be performed in accordance with ANSI 323 calibration standards. Radiation protection personnel using this equipment are trained and experienced. For contamination control, portal monitors and friskers will be used at exits from radiation control areas to monitor personnel leaving the stations. Protective clothing and respiratory equipment are also used, as required, to keep exposure ALARA.

All plant personnel are required to wear a TLD as the primary method for determining beta-gamma dose. This dosimeter device is probably the best studied and most widely used personnel dosimeter (Becker, 1973). It has been found to be reliable and accurate and, unlike film badges, can be read almost immediately after exposure to radiation to determine the dose. For neutron dosimetry, Harris will comply with the applicable recommendations of RG 8.14. Self-reading pocket dosimeters also will be issued as a secondary method for beta-gamma dosimetry and will also provide a day-to-day estimate of personnel dose for gamma radiation that can be used for radiation work permit job planning. Dose records for each individual will be maintained in accordance with RG 8.7. The Harris bioassay program will be used to assess the effectiveness of the respiratory protection program and will follow the guidance of RG 8.9 and ANSI 343. The whole-body counter will be located at the station for in vivo counting of station personnel

APPENDIX B

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