HEALTH PHYSICS ASPECTS OF ADVANCED REACTOR LICENSING REVIEWS

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ABSTRACT

The last Construction Permit to be issued by the U.S. Nuclear Regulatory Commission (NRC) for a U.S. light water reactor (LWR) was granted in the late 1970s. In 1989 the NRC issued 10 CFR Part 52[1] which is intended to serve as a framework for the licensing of future reactor designs. The NRC is currently reviewing four different future or "next-generation" reactor designs. Two of these designs are classified as evolutionary designs (modified versions of current generation LWRs) and two are advanced designs (reactors incorporating simplified designs and passive means for accident mitigation). These "next-generation" reactor designs incorporate many innovative design features which are intended to maintain personnel doses ALARA and ensure that the annual average collective dose at these reactors does not exceed 100 person-rems (1 person-sievert) per year. This paper discusses some of the ALARA design features which are incorporated in the four "next-generation" reactor designs currently being reviewed by the NRC.

INTRODUCTION

The NRC has been actively reviewing "next-generation" reactor designs since the late 1980s. The term "next-generation" encompasses both evolutionary LWR designs and advanced reactor designs. Evolutionary reactor designs are essentially modified versions of current generation LWR designs. These reactors utilize conventional safety system concepts. Advanced reactor designs include passive and non-LWR reactor designs. Passive reactors employ greatly simplified designs, generally range from 300 to 600 MWe in size, and utilize passive means for accident prevention and mitigation.

All of these "next-generation" reactor designs incorporate lessons learned from currently operating LWRs. Many of the features in these new plants, such as standardization, simplified plant design, and modularization, will result in plants that will be easier to operate and maintain. This, in turn, will result in lower collective doses. Careful attention to material selection, such as the use of low cobalt- and nickel-based alloys in the primary coolant system, will also help to lower collective doses by reducing overall plant radiation levels.

BACKGROUND

Although the NRC has strongly encouraged the standardization of nuclear reactor designs for many years, it was the issuance of 10 CFR Part 52 (known as the Standardization Rule) in 1989 which served as the framework for consideration of future designs. The three parts of this Standardization Rule provide for the issuance of 1) early site permits, 2) standard design certifications, and 3) a combined construction permit and operating license. This rule is designed to streamline the reactor licensing process. The standard nuclear power plant final design approval which results from this licensing review is acceptable for incorporation into individual facility license applications.

CURRENT PLANT DESIGN REVIEWS

There are four "next-generation" reactor designs currently under review within the NRC. Two evolutionary LWR designs, General Electric's (GE) Advanced Boiling Water Reactor (ABWR)[2] and Combustion Engineering's System 80+ Standard Design,[3] are in the final stages of design certification. Two passive LWR designs, Westinghouse's AP-600[4] and GE's Simplified BWR (SBWR),[5] are in the early stages of staff review (Table 1). In addition to these ongoing plant design reviews, the NRC has completed its review of EPRI's Advanced Light Water Reactor (ALWR) Requirements Document.[6] The purpose of this document is to specify industry approved design criteria for evolutionary and passive ALWR standard plants. The design features described are those features that both utilities and industry would like to see incorporated into the next generation of nuclear power plants.

Table 1. "Next-Generation" Reactor Designs Currently Under NRC Review

Evolutionary	Passive
GE Advanced BWR (ABWR) CE System 80+	GE Simplified BWR (SBWR) Westinghouse AP-600

One of the objectives contained in EPRI's ALWR Requirements Document is to design a nuclear power plant that can operate with an average dose of 100 person rem (1 person-sievert) per year or less. The estimated annual doses for the four "next-generation" reactor designs currently under review range from 68 person-rem (6.8×10^{-1} person-sievert) for Westinghouse's AP-600, to 99 person-rem (9.9×10^{-1} person-sievert) for GE's ABWR. In contrast, the average annual dose for U.S. LWRs averaged 266 person rem (2.66 person-sievert) per reactor in 1992.[7] This average dose, however, represents a 62 percent drop from the U.S. average annual dose of 705 person-rem (7.05 person-sievert) per reactor just a decade earlier (see Figure 1).

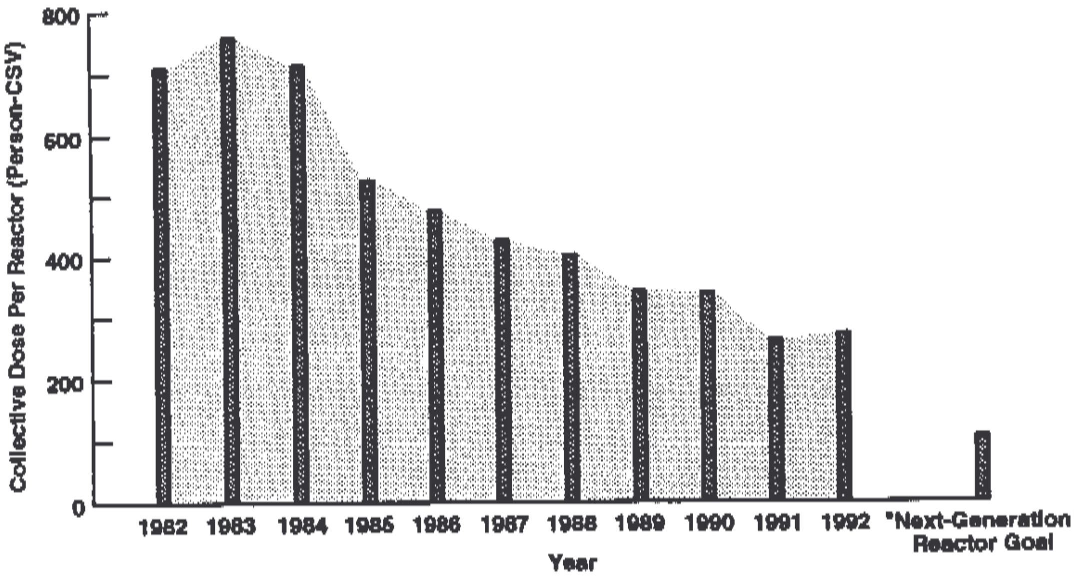

Figure 1. Collective Doses at U.S. LWRs and "Next-Generation" Reactor Dose Goal

In order to achieve an average annual dose of 100 person-rem (1 person-sievert) or less, the "next-generation" reactors will incorporate a number of ALARA design features. Some of these design features are based on the ALARA guidance provided in Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Is Reasonably Achievable." However, many of the ALARA design features utilized by the "next-generation" reactors are based on lessons learned from the current generator of operating reactors and employ the use of advanced technology.

Some of the ALARA design features described below will apply to all four of the "next-generation" reactor designs currently under review by the NRC. Others will apply only to designs by specific vendors or to a type of reactor design (i.e., evolutionary versus passive).

Material Changes

The primary source of radiation fields in nuclear power plants is cobalt-60. Cobalt is the major constituent of Stellite, a hardfacing material used in valve seats, pump journals, and other wear resistant components. The "next-generation" reactor designs will restrict the use of high cobalt alloys such as Stellite to those applications where no satisfactory alternate material is available. Where possible, the cobalt content of piping and other equipment in direct contact with the reactor coolant will be restricted to 0.05 wt %. The inconel steam generator tubes will contain no more than 0.015 wt % cobalt and will be fabricated to relieve stresses to reduce stress corrosion cracking. Main condenser tubes and tube sheets will be made of titanium alloys to minimize condenser tube leakage, thereby reducing the introduction of foreign materials (which can become activated) into the reactor system. The presence of antimony in reactor coolant pump (RCP) journal bearings has resulted in an increase in the number of hot particles at some current generation plants.[8] The RCP journal bearings in CE's System 80+ design will be designed to minimize the presence of antimony.

Component Design Features

The "next-generation" reactors will incorporate several innovative component design features to reduce area dose rates and minimize personnel doses. External recirculation pumps and recirculation piping were replaced by internally mounted recirculation pumps in GE's ABWR design. GE's SBWR design contains no recirculation pumps or recirculation piping since this plant employs natural recirculation as a motive force. In both designs, elimination of recirculation pumps and piping removes a major source of radiation in the lower drywell and should reduce general area dose rates in the drywell by 50 percent. The SBWR's simplified design will require significantly fewer safety relief and other valves in the drywell, thereby requiring less maintenance time to service these valves. The SBWR turbine design will utilize cross-over lines, thereby eliminating the need for moisture separators and reheaters and removing the most significant source of skyshine and turbine building operational radiation.

Both the CE System 80+ and the AP-600 designs will incorporate an integrated reactor head removal package. This package will facilitate reactor head removal and replacement during refuelings, resulting in lower personnel doses and less manpower requirements. Refueling doses for the GE BWR designs will be reduced through the use of a stud tensioner and an automated refueling bridge. The use of the automated refueling bridge, where no personnel are located on the platform itself, will cut refueling time in half and reduce the effective dose rate by a factor of ten.

The AP-600 steam generator design will include a sludge control system/mud drum which is designed to reduce the need for sludge lancing, and reduces tube and tube support degradation. The tube ends in this steam generator are designed to be flush with the tube sheet in the steam generator channel head to eliminate potential crud traps.

A system which has resulted in several overexposures in current generation LWRs is the Transversing In-Core Probe (TIP) system in BWRs and incore instrumentation in PWRs. GE's ABWR provides a shielded room for the TIP drive units. Automatic logic control and mechanical stops prevent the TIP or activated portions of the TIP cable from being withdrawn into the drive housings. GE's SBWR design eliminates the TIP system by using fixed in-core detectors. The CE System 80+ design prevents access to the reactor cavity housing the incore instrumentation chase by providing a posted and locked access door connected by electrical interlock to an area radiation monitor located in the reactor cavity. When the incore instrumentation are withdrawn from the reactor core, a warning light on the access door illuminates and the interlock prevents the access door from being opened.

Other component design features include the use of canned pumps in the Residual Heat Removal System and Reactor Water Cleanup System of the GE designs to minimize maintenance requirements. The RCPs in the CE System 80+ design will incorporate a cartridge type of RCP seal which is reliable and easily replaceable. The RHR heat exchangers in GE's ABWR and Westinghouse's AP-600 designs are designed with an excess of tubes in order to permit plugging of some tubes without losing system efficiency. The heat exchangers also are provided with drains to allow drainage of the shell-side water prior to maintenance. The CE System 80+ design will minimize the use of evaporators. Evaporators have historically required frequent maintenance and contributed to high personnel exposures. The CE design will also utilize mechanical snubbers rather than hydraulic snubbers in radiation areas to reduce maintenance and inspection needs. Liquid systems containing radioactive cartridge filters in the AP-600 design will be provided with a remote filter handling system for changeout and transfer to the drumming station of spent radioactive filter cartridges. In order to prevent the migration of noble gases and other airborne radionuclides between floors in CE's System 80+ design, floor drains connecting rooms that have significantly different airborne radioactivity levels will be separated or provided with water-filled loop seals to prevent cross-contamination.

Features To Facilitate Maintenance

The equipment selected for the "next-generation" reactor designs will have enhanced reliability and will be designed for low maintenance. These designs will make more use of modular components which can be easily replaced or removed to a lower radiation area for repair. The AP-600 design will have RCPs which can be unbolted for quick removal to a low radiation background work area for maintenance or replacement using a specially provided pump removal cart. The control rod drive (CRD) system in the GE ABWR and SBWR designs will have an internal CRD restraint feature which will facilitate CRD removal. This feature will result in lower radiation exposures than those seen in current generation BWRs, which have external CRD restraints. The lower drywell design in the GE design reactors will allow easy access to the lower reactor vessel head for CRD and reactor internal pump removal. A transport system will permit removal of these components to a lower radiation area.

Radioactive systems and components will be provided with taps for flushing with condensate or for chemically cleaning to reduce crud buildup and lower radiation levels prior to maintenance. Rooms housing these components will have epoxy-type floor and wall coverings to facilitate decontamination. Equipment and floor drain sumps will be stainless steel lined for ease of cleaning and to reduce crud buildup.

The "next-generation" reactors will be designed to facilitate accessibility to plant equipment. Adequate work and laydown space will be provided around components for maintenance purposes. In order to facilitate maintenance and improve worker efficiency, adequate illumination and support services (e.g., power, service air, water, ventilation, and communications) will be available at work stations. In the event that maintenance cannot be performed in-situ, rigging and lifting equipment will be provided to facilitate the removal, transport, or replacement of equipment (this rigging equipment can also be used for the installation of portable shielding).

Features to Facilitate In-Service Inspection

Approximately nine percent of the annual dose at U.S. LWRs can be attributed to in-service inspection work. The "next-generation" reactor designs will facilitate in-service inspection by making plant components more accessible and relying more on the use of robotics.

The CE System 80+ design will include permanent platforms around major equipment such as the steam generators and reactor coolant pumps. These platforms will facilitate access to these components for maintenance and in-service inspection, and will serve to reduce the overall plant collective dose by eliminating the need to erect temporary scaffolding around these components for maintenance/inspection purposes. In GE's ABWR and SBWR designs, permanent steel platform will be provided for in-service inspection of the reactor pressure vessel nozzle welds and associated piping. These steel platforms will also serve to provide shielding for inspection personnel from adjacent radiation sources.

All of the "next-generation" designs will utilize easily removable blanket or mirror type thermal insulation around piping and components. The sections of the reactor vessel insulation in the area of the reactor vessel nozzle welds for the AP-600 will have permanent I.D. markings to accommodate rapid reinstallation. GE's ABWR and SBWR designs will incorporate specific access panels and shield doors into required inspection areas permitting easy bypass of insulation areas and thereby reducing inspection time.

In order to reduce the inservice inspection time required for welds, all four of the "next generation" reactor designs under review will have forged ring instead of plate welded pressure vessels. Because forged ring pressure vessels have fewer welds, the total vessel weld length inspection will be reduced by 30 percent. The use of seamless piping in all "next-generation" reactor designs will reduce the amount of piping welds.

The reactor vessel nozzle welds in the AP-600 and CE System 80+ designs will be designed to accommodate remote inspection using ultrasonic sensors. The use of automated equipment for weld inspections in the GE ABWR and SBWR designs will reduce the required inspection manhours by a factor of two. The CE System 80+ design will utilize robotics, whenever practical, to perform maintenance and inspection activities such as remote pipe welds and inspections in high radiation areas. The steam generators in both the CE System 80+ and AP-600 plant designs will be designed to use automatic/robotic equipment for inspection and maintenance activities. In addition to having larger diameter manways to facilitate personnel access and the installation and removal of tooling, these steam generators will have an increased number of handholes and will be provided with platforms and adequate pull and laydown areas for inspection and maintenance purposes.

Plant Layout Features

The plant layouts for the "next-generation" reactor designs will be designed to maintain personnel exposures ALARA during normal and post-accident conditions. Radioactive systems will be separated from non-radioactive systems. Pipes or ducts carrying radioactive sources will not be routed through occupied areas. Redundant radioactive components will be separated and shielded from each other to permit maintenance on one component without being exposed to radiation from the other component. Labyrinth entrances will be provided to radioactive pump, equipment, and valve rooms. These labyrinth entrances will have sufficient space for easy access and for equipment removal. Adequate space will be provided for the storage and erection of temporary shielding.

Ion exchangers in the CE System 80+ design will be located in pits with the spent resin tanks located below the ion exchanger. This design will provide shielding around the spent resin tanks and will lower the dose rates to personnel working on other equipment in the area. The spent fuel transfer tube in the AP-600 design does not have a seismic gap and therefore will be completely enclosed in concrete. This design results in a spent fuel transfer tube which is shielded its entire length and eliminates the potential for personnel overexposures during refueling operations caused when spent fuel is transported through unshielded portions of the spent fuel transfer tube (this is a shortcoming and potential problem with conventional PWR designs).

Plant layout features will be designed to facilitate maintenance operations and minimize personnel dose. Adequate rigging and lifting equipment will be provided, where needed, to assist in equipment removal/replacement. The use of overhead tracks and in-place removal equipment (in the GE ABWR and SBWR plant designs) to transfer safety/relief valves, reactor internal pumps, and other valves in the drywell, will result in an estimated savings of 300 person-hours per year. The CE System 80+ design will provide for large staging areas both inside and outside the reactor building equipment hatch and personnel airlocks. This will allow for pre-staging prior to the start of an outage and the location of these staging areas will provide for efficient radiation controls and will minimize the potential for the spread of contamination. Hot tool cribs will be located in low radiation areas adjacent to maintenance areas to minimize waiting times in high radiation areas and to prevent the spread of contamination. The hot machine shop is located in a low radiation area adjacent to the equipment hatch to permit maintenance to be performed on equipment removed from containment in a lower radiation area. Dedicated change out areas are also located near airlocks in low dose areas to minimize personnel traffic flow and the potential for the spread of contamination.

CONCLUSION

The four "next-generation" reactor designs currently under review by the NRC all contain a number of innovative ALARA design features. Some of these features are simply modifications of ALARA design features used in currently operating U.S. LWRs. Others are based on design features used in foreign LWR designs. Still other design features, such as some of the features used by the passive design LWRs, will be used for the first time in the "next-generation" reactor designs. The use of these innovative ALARA design features, along with the minimization of cobalt and nickel in reactor coolant system components, should permit these "next-generation" reactors to operate within their estimated dose goals of 100 person-rem (1 person-sievert) per year.

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Author Biography

Charles Hinson is a health physicist in the Radiation Protection Branch, NRR at the U.S. Nuclear Regulatory Commission. Over the last three and a half years, Mr. Hinson has served as the health physics reviewer for the EPRI Utility Requirements Document review and the CE System 80+ and GE SBWR design reviews. Other current responsibilities include analysis of occupational exposure data for U.S. reactors and evaluation of radiologically significant events at U.S. reactors. Mr. Hinson has also served as a NRC project manager for the Fort St. Vrain and Big Rock Point nuclear plants. He has both a B.S. and a M.E. in Nuclear Engineering from the University of Virginia.

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DISCUSSION

Dionne: In your review of these advanced reactors, you mentioned that there was one idea you had for improvement, that interlock, to the transversing in-core probe (TIP) room. Are there any other examples of design changes that were made by the NRC to reduce dose?

Hinson: Yes, there are. I was the radiation protection reviewer for the CE design, and the design change described in my paper was for the CESSAR System 80+ design. The radiation protection reviewer who reviewed GE's ABWR design found that there was inadequate shielding surrounding part of the TIP system which was located over one of the containment personnel access hatches. GE corrected this problem area by adding additional shielding in this location. The ABWR reviewer also noted that there existed a large gap between the reactor vessel shield and the drywell ceiling. In the event of a spent fuel bundle drop onto the refueling pool seal, the resulting dose rates in the upper part of the drywell would be in excess of 20,000 rad/hr. Personnel on the upper or lower decking during this event would have to transient through this radiation field to exit the drywell. GE corrected this potential problem area by the addition of several more feet of concrete and steel shielding in this area to reduce the size of the gap. The radiation protection reviewer who is reviewing Westinghouse's AP-600 design has had several confirmatory shielding calculations performed for various areas of the plant to determine the adequacy of the plant shielding design.

Rescek: We get into a lot of temporary hanging of lead and shielding and you talk a little bit now in your answer to Bruce about permanent shielding in place. Will these plants be designed such that the criteria for hanging temporary shielding may automatically be precalculated so we don't go through all of these special calculations to see how much lead loading we can put on various lines in the plant? Will all of that type of process be avoided here?

Hinson: Well these designs don't get into that level of detail because a lot of the operational concerns are left up to the individual utility. However, these plant designs will have places to hang temporary shielding and they will be designed for adequate space for storage and erection of temporary shielding.

Cybul: Based on historical data, BWRs consume more dose than PWRs. Is it reasonable for the next-generation design to have the same criteria?

Hinson: Based on the advanced reactor dose estimates, the two GE BWR designs have a dose estimate of 92 and 99 person-rem/yr vs. 68 and 79 person-rem for the PWR designs. These estimates are very close. Also, since the shielding design is not complete for these plants, these dose estimates are very preliminary, and it is really hard to say whether that trend will continue in the advanced designs.