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PERSONNEL RADIATION EXPOSURE IN HTGR PLANTS

by
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MASTER

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ABSTRACT

Occupational radiation exposures in high-temperature gas-cooled reactor (HTGR) plants were assessed. The expected rate of dose accumulations for a large HTGR steam cycle (HTGR-SC) unit is 0.07 man-rem/MW(e)y, while the design basis is 0.17 man-rem/MW(e)y. The comparable figure for actual light water reactor (LWR) experience is 1.3 man-rem/MW(e)y. The favorable HTGR occupational exposure is supported by results from the Peach Bottom Unit No. 1 HTGR and Fort St. Vrain HTGR plants and by operating experience at British gas-cooled reactor (GCR) stations.

1. SUMMARY

Radiation exposures of reactor plant personnel include exposures arising from reactor operation and surveillance, routine maintenance, refueling, waste processing, in-service inspection, and special (or unplanned) maintenance. This paper will address projected as well as actually experienced occupational exposures at HTGR plants. Comparisons with actual exposures at LWR plants will also be presented.

Dose assessments performed for the 2240-MW(t) [900-MW(e)] HTGR-SC reference plant led to the conclusion that the expected annual exposure would amount to 50.8 man-rem per unit. The corresponding expected rate of dose accumulation is 0.07 man-rem/MW(e)y, whereas the design basis established for the large HTGR-SC plants is 0.17 man-rem/MW(e)y. It should be mentioned that the dose assessment for the HTGR-SC refueling operation assumed an ex-vessel refueling concept. With the current in-vessel refueling scheme, reduction of refueling man-rem exposures by a factor of 2.0 or more appears to be attainable.

Actual man-rem exposures at the Peach Bottom Unit No. 1 HTGR and at the Fort St. Vrain HTGR have been exceptionally low. The annual collective dose has never exceeded 10 man-rem, and the average annual dose per worker has been minimal. Furthermore, operating experience to date at the Fort St. Vrain plant shows that the rate of occupational dose accumulation is less than 0.1 man-rem/MW(e)y.

The comparable radiation exposure for actual LWR experience, averaged over all operating LWR plants in the United States from 1969 through 1978, is 420 man-rem per reactor per year, or 1.3 man-rem/MW(e)y. By comparison, occupational exposures at HTGR plants are considerably lower than those at LWR facilities. The favorable HTGR occupational exposure is supported by the results from the Peach Bottom and Fort St. Vrain HTGR plants, as well as by operating experience at the British Oldbury and Wylfa GCR plants, which also have prestressed concrete reactor vessels (PCRVs).

2. HTGR DOSE COMMITMENT

A comprehensive study of occupational radiation exposure was conducted for a reference large HTGR-SC plant rated at 2240 MW(t) or 900 MW(e) net. The dose assessment covered the following work functions: reactor operation and surveillance, refueling, routine maintenance, in-service inspection, and special (or unplanned) maintenance.

2.1. Reactor Operation

During reactor operation, routine surveillance and inspection are required around the PCRV. The HTGR containment building is accessible on a limited basis for a maximum occupancy duration of 10 hr each week per person during operation. The operating dose rates in the containment range from 1.0 to 10 mrem/hr, depending upon the location and the containment ventilation rate. Radiation sources affecting containment access include direct or streaming radiation from the reactor cavity and gasborne activity due to PCRV leakage.

Based upon a conservative operating dose rate of 8.0 mrem/hr in the containment building and an access requirement of 20 man-hours per week, the annual exposure for reactor operation at 80% load factor is estimated to be approximately 7 man-rem. Reduction of the exposure is possible with a reduced PCRV leak rate and/or an increased containment ventilation rate, which would lower the gaseous dose rate in the containment building.

2.2. Refueling

About one quarter of the HTGR core is refueled annually. Refueling can be accomplished with an ex-vessel or in-vessel refueling scheme. The ex-vessel scheme requires extensive use of refueling equipment to load fuel elements and transfer them out of the PCRV. With the in-vessel scheme, all fuel elements are transferred within the reactor vessel.

A detailed dose assessment was performed for the ex-vessel refueling scheme. The refueling operations include handling and storage of spent fuel and reflector elements, control rod drives (CRDs), and the high-temperature filter/adsorber (HTF/A). The major refueling equipment for the ex-vessel scheme consists of one fuel handling machine, three fuel transfer casks, and one auxiliary service cask. The ex-vessel refueling operations for the

900-MW(e) HTGR-SC plant require a crew of 15 persons per shift on a three-shift-per-day basis and last for 5.5 days (assuming 100% equipment and personnel efficiency).

The steady-state dose rates during refueling range from 2.5 mrem/hr in the general accessible areas to 10 mrem/hr near the refueling equipment. The dose rates for transient conditions are limited to a maximum of 100 mrem/hr. The integrated transient exposure was determined to be negligible, since the refueling equipment is fully automated and personnel are not required to remain near the refueling equipment under transient conditions.

The total integrated dose to all refueling personnel (45 persons for the ex-vessel scheme) over a refueling period of 5.5 days was estimated to be 5.5 man-rem. The average exposure amounts to 1.0 man-rem/day.

The occupational dose for the in-vessel refueling scheme has not yet been assessed. However, reduction of the refueling man-rem exposure by a factor of 2.0 or more appears to be attainable with the in-vessel scheme, since all fuel and reflector elements are transferred within the PCRV.

2.3. Maintenance and Repair

The components and equipment items within the nuclear steam supply (NSS) system are, in general, designed to require minimum maintenance or repair. It is recognized that plant maintenance contributes significantly to the annual personnel dose for LWR plants, because of crud (corrosion and erosion product) deposits on the components. For HTGR plants, the fission-product plateout activity is the major radiation source for maintenance consideration.

The component maintenance that must be considered includes normal maintenance and unplanned maintenance. An example of normal maintenance is the replacement of the HTF/A during refueling. Steam generator removal and tube plugging are regarded as unplanned maintenance for radiation exposure assessment.

Normal maintenance associated with the helium purification system includes periodic replacement of the HTF/A components and helium purification filter. The HTF/A removal and replacement operation is performed during refueling; hence, the occupational dose for such operation is included in refueling man-rem exposure. The dose resulting from manual change of the purification filter is fairly small, since the filter is an end unit in the helium purification train with very little accumulation of activity. Other components in the purification system are designed for 40-yr life, requiring no maintenance or replacement.

The main circulators in the primary system are designed to be removable, although they have a design life of 40 yr. Circulator removal is considered unplanned maintenance. Removal of a main circulator will not be initiated until 5 days after reactor shutdown to allow time for PCRV depressurization and preparation for removal operation. A shielded handling cask, which limits the dose rate at its outside surface to 20 mrem/hr when loaded, will be provided for removing the circulator from the PCRV penetration and transferring it to the reactor service building for maintenance and storage. Maintenance and repair work on the circulator will be remotely accomplished in the reactor equipment service facility, which supplies the required service equipment and tools. The circulator will be stored in a shielded storage well. The total exposure for completing the removal operation is about 1.0 man-rem.

Like the main circulators, the steam generators are designed for a 40-yr life but can be replaced or repaired during plant life. Steam generator removal involves removal of the main circulator, the concrete shield plug above the circulator, the diffuser, the compressor inlet duct, and the steam generator. To ease the complexity of the removal operations, the loop with the defective steam generator is isolated from normal plant operation for 1 yr with no bypass flow to allow decay of plateout activities. Steam generator removal operations may be initiated 1 month after shutdown. Radiation shielding (as provided by the circulator handling cask, the shield cover, the adapter, and the steam gnerator handling cask) and appropriate access control will be required to maintain ALARA (as low as reasonably achievable) radiation exposures. The estimated radiation exposure for steam generator removal and replacement amounts to 16.5 man-rem. The main contribution to the man-rem exposure results from transfer of the steam generator out of the containment building.

In case of steam generator tube leaks, tube plugging will be required during reactor shutdown. The dose assessment for steam generator tube plugging is based upon 1% tube failure during the plant life. It is roughly estimated that approximately 1.0 man-rem may be experienced per tube plugging occurrence.

2.4. In-Service Inspection

The main areas of concern for in-service inspection are basically those within the primary coolant system boundary which, if they failed, could cause depressurization of the PCRV and might impair an orderly shutdown of the reactor. These areas include the PCRV and its penetrations and the pressure relief system. Provisions required for in-service inspection include accessibility and inspectability.

In-service inspections will be accomplished by appropriate methods such as visual examination, ultrasonic inspection, radiographic methods, and liquid penetrant techniques. Direct or remote inspection may be performed as convenient to reduce the working time and number of workers.

As a further step toward maintaining ALARA radiation exposure, in-service inspections will be performed, insofar as possible, during reactor shutdown to take advantage of the significantly reduced radiation levels.

Determination of the occupational dose for in-service inspection is pending the availability of time-and-motion data. An integrated dose of 5 to 10 man-rem/yr has been established as a design goal for in-service inspection.

2.5. Total Dose Commitment

The expected man-rem results for the 900-MW(e) HTGR-SC plant are summarized in Table 1 along with the design basis values. The expected annual exposure is 50.8 man-rem per unit, whereas the design basis is 130 man-rem per unit. The corresponding rates of dose accumulation are 0.07 and 0.17 man-rem/MW(e)y for the expected case and design basis, respectively.

It should be noted that the dose estimate for special maintenance within the NSS system represents an average exposure on an annual basis, with the assumptions of tube plugging every year at 1.0 man-rem per occurrence, steam generator removal every 10 yr at 16.5 man-rem per occurrence, and main circulator removal every 2 yr at 1.0 man-rem per removal. The expected exposure for balance-of-plant (BOP) maintenance is included to indicate the order of magnitude. Actual dose assessment for BOP maintenance is the responsibility of an architect-engineer.

TABLE 1
MAN-REM EXPOSURE FOR 900-MW(e) HTGR-SC

Type of Operation	Annual Man-Rem Exposure: Expected	Annual Man-Rem Exposure: Design Basis
Refueling[a]	5.5	20
Reactor Operation and Surveillance	7.0	20
NSS Maintenance and ISI	10.1	20
BOP Maintenance	25.0	50
Special Maintenance	3.2	20
	50.8	130
Rate of Accumulation		
[~950 MW(e) gross[b], 80% load factor]	$\frac{50.8}{950 \times 0.8} = 0.07 \frac{\text{man-rem}}{\text{MW(e)y}}$	$\frac{130}{950 \times 0.8} = 0.17 \frac{\text{man-rem}}{\text{MW(e)y}}$

[a] Ex-vessel refueling scheme assumed.
[b] Assumed gross power.

3. HTGR OPERATING EXPERIENCE

This section describes actual experience with man-rem exposure at two commercial HTGR plants in the United States: the Peach Bottom HTGR and the Fort St. Vrain HTGR.

3.1. Peach Bottom HTGR

Peach Bottom Unit No. 1, operated by the Philadelphia Electric Company, was a 40-MW(e) prototype HTGR plant with a steel reactor vessel. The plant was operated successfully for over 7 yr until October 31, 1974, at which time it was shut down for decommissioning.

According to records of Philadelphia Electric health physicists, personnel exposures during operation, maintenance, and refueling were exceptionally low. Yearly and cumulative exposure and power generation data are listed in Table 2. The total annual exposure is about 3.0 man-rem.

No separate data are available for refueling exposures, but the exposure is estimated to be less than 1.0 man-rem per refueling.

TABLE 2
PEACH BOTTOM HTGR OPERATING EXPERIENCE

Year of Operation	Man-Rem Exposure		Gross Power Generation [MW(e)y]		Average Man-Rem per MW(e)y
	By Year	Cumulative	By Year	Cumulative	
1967	~3	~3	18.9	18.9	0.16
1968	~3	~6	16.8	35.7	0.18
1969	~3	~9	17.6	53.3	0.17
1970	~3	~12	18.4	71.7	0.16
1971	~4	~16	27.0	98.7	0.15
1972	~3	~19	13.8	112.5	0.22
1973	~3	22	23.4	135.9	0.13
1974	NA	NA	22.3	158.2	NA

3.2. Fort St. Vrain (FSV) HTGR

The Fort St. Vrain plant, owend by the Public Service Company of Colorado (PSC), is the only HTGR plant presently operating in the United States. In contrast to the Peach Bottom plant, the Fort St. Vrain plant utilizes a PCRV. The 842-MW(t) or 330-MW(e) rated plant achieved initial criticality on January 31, 1974, and began generating electricity in December 1976. The first refueling was accomplished in the spring of 1979. Commercial operability was declared on July 1, 1979.

3.2.1. Total Exposure

The PSC Health Physics office maintains detailed records of personnel radiation exposure in compliance with state and federal regulations. Based upon the PSC reports submitted to the U.S. Nuclear Regulatory Commission (NRC), the man-rem data for the years 1976 through 1979 are summarized in Table 3.

As shown in Table 3, annual collective doses incurred by plant personnel have been minimal. No one has exceeded an annual dose of 0.25 rem, and the average annual dose per radiation worker remains at approximately 0.05 rem. For the last 4 yr (1976-1979, inclusive), the total collective value was 12.3 man-rem and a total of 137 MW-y of gross electricity had been generated. The resulting average rate of exposure is 0.09 man-rem/MW(e)y.

3.2.2. Refueling Exposure

The first refueling of the Fort St. Vrain HTGR took place in March and April 1979. During these refueling operations, numerous gamma dose rate measurements were made.

Most of the refueling dose rates were so low that the use of a microrem meter by PSC health physicists was required. For instance, the average dose rate on the accessible surface of the fuel handling machine (FHM) when loaded with spent fuel was less than 1 mrem/hr. The only time personnel are near the loaded FHM is during unbolting, crane, and bolting operations, about half an hour per fuel region. Assuming six personnel and six fuel regions, the man-rem exposure for this part of refueling would be

$$\frac{6 \times 6 \times 0.5 \times 1}{1000} = 0.018 \text{ man-rem} \quad .$$

Control rod drive handling operations were equally inconsequential in exposure, except for one CRD which had activated clevis pins. In this case, the dose rate at some distance from the auxiliary transfer cask (ATC) was about 4 mrem/hr. Hence, it is possible that another 0.02 man-rem could have been accumulated in moving this CRD to the storage wells.

TABLE 3
ANNUAL DOSES AT FORT ST. VRAIN PLANT
1976-1979

Personnel Monitored				Annual Dose (man-rem)	Gross MW(e)y Generated	Average Man-Rem per MW(e)y	Average Dose per Worker (rem/y)[a]
Year	Nil Exposure	<0.1 Rem	0.1-0.25 Rem				
1976	1362	25	0	1.3	2.8	0.46	0.05
1977	946	55	1	2.9	29.8	0.10	0.05
1978	896	34	0	1.7	75.7	0.02	0.05
1979	1149	120	2	6.4	28.7	0.22	0.05

(a) Averaged over those individuals who received measurable exposures.

Health physicists made one-time measurements at greater elevations of the Fort St. Vrain FHM and ATC, where the surface dose rates are intentionally higher than those within an 8-ft height above the refueling floor. It is possible that a few tenths of a man-rem could have been accumulated by these health physicists in this fashion.

Although no breakdown of personnel exposure by work function is available in the PSC report to the NRC, it is believed that the total dose accumulation during the first refueling was less than 0.5 man-rem.

It should be mentioned that prior to the first refueling, the Fort St. Vrain reactor had not exceeded approximately 65% of full power rating, and the spent fuel had decayed for a period of 45 to 60 days. The 60-day decay time reduces the La-140 inventory in the fuel elements by a factor of 25. If the plant had operated at 100% power and started refueling at 1 day after shutdown, the refueling personnel exposure would probably have been less than 5 man-rem.

4. COMPARISON WITH OTHER REACTORS

4.1. LWR Information

Occupational radiation exposures at operating LWR plants for the years 1969 through 1978 are available in NRC Report NUREG-0594 (Ref. 1). The report provides a compilation of information such as annual collective dose for each reactor, average collective dose per reactor, and average man-rem/MW(e)y.

The NRC report also summarizes the annual man-rem data by work function. In 1978, workers involved in routine and special maintenance activities at LWR plants incurred 67.4% of the total cumulative dose. The percentages of the cumulative dose for other work functions in the same year for all LWRs are 13.3% for reactor operation and surveillance, 7.7% for in-service inspection, 5% for waste processing, and 6.6% for refueling. Evidently, maintenance activities continue to be the predominant component of the collective dose.

4.2. Experience at British GCR Plants

The Central Electricity Generating Board (CEGB) in the United Kingdom operates several GCR stations with concrete reactor vessels, such as the Oldbury and Wylfa stations. Each station consists of two reactor units. The rated net power per unit is 300 MW(e) at Oldbury and 590 MW(e) at Wylfa. The actual man-rem experience at the Oldbury and Wylfa stations for the years 1972 through 1978 is provided in a recent report by the CEGB (Ref. 2).

4.3. Comparison

Figure 1 compares the annual collective doses per reactor for the Fort St. Vrain HTGR, Peach Bottom HTGR, British GCRs, and LWRs. The data on average man-rem/MW(e)y are compared in Fig. 2. The British Wylfa station is included in Fig. 1 but not in Fig. 2 because of the lack of information on electricity generation.

It is obvious that LWRs have experienced considerably greater man-rem exposure per reactor than GCRs. On a man-rem/MW(e)y basis, the LWR exposure is higher by about one order of magnitude.

A summary of the average collective and individual doses over the operating years is given in Table 4 for each of the reactors.

TABLE 4
COMPARISON OF OCCUPATIONAL EXPOSURES

Reactor Type[a]	Average Annual Man-Rem per Reactor	Average Man-Rem per MW(e)y[b]	Average Annual Dose per Worker (rem)
900-MW(e) large HTGR-SC (projected)	51.0	0.07	NA[c]
330-MW(e) FSV HTGR (1976-1979)	3.0	0.09	0.05
40-MW(e) Peach Bottom HTGR (1967-1973)	3.0	0.16	NA
300-MW(e) Oldbury GCR (1972-1978)	39.0	0.12	0.2
590-MW(e) Wylfa GCR (1972-1978)	26.0	<0.1	NA
All LWRs in U.S.A. (1969-1978)	420.0	1.3	0.8

(a)The indicated power level is a rated net capacity.
(b)Obtained by dividing the total man-rem over all operating years by the corresponding gross MW(e)y generated.
(c)Not available.

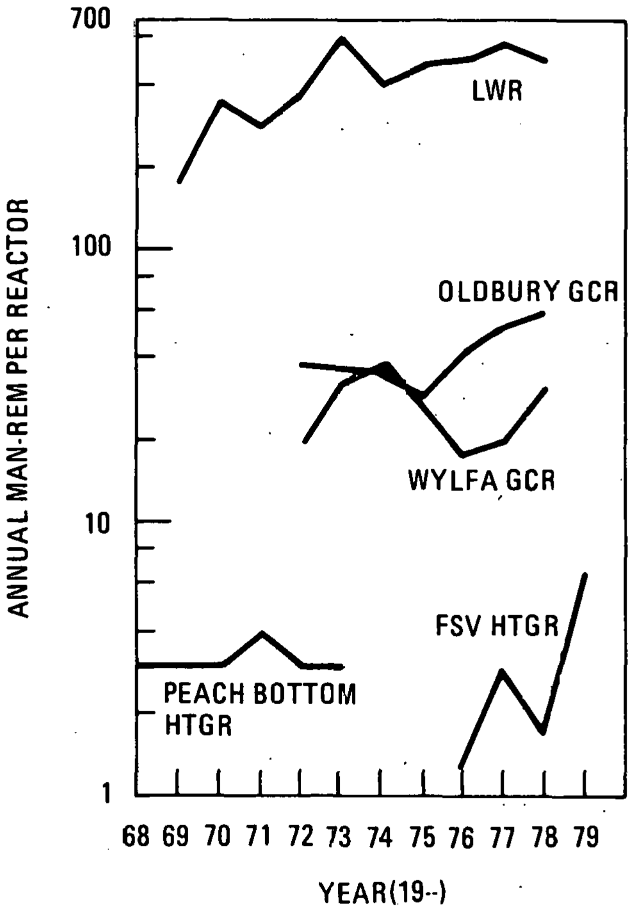

Fig. 1. Comparison of annual man-rem per reactor

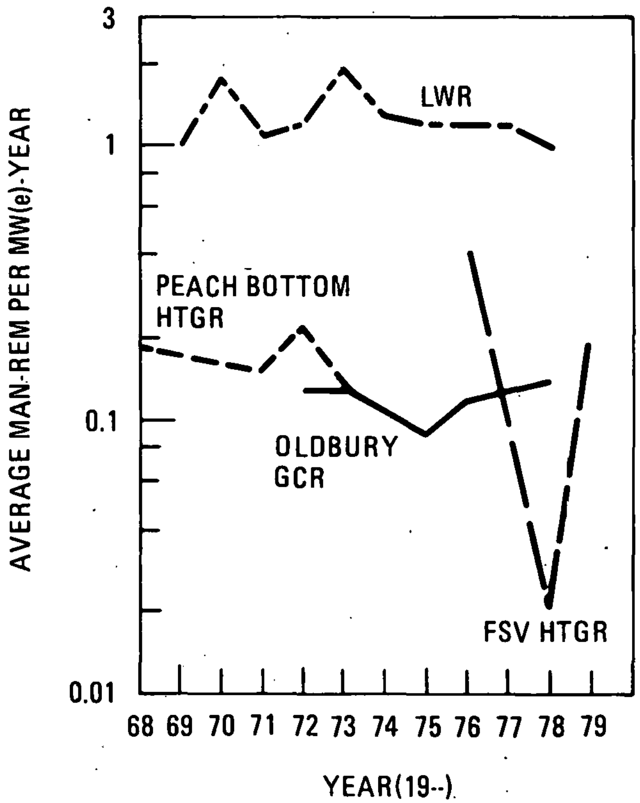

Fig. 2. Comparison of average man-rem/MW(e)y

5. CONCLUSIONS

Available data on occupational man-rem exposures at commercial nuclear plants clearly indicate that GCRs are experiencing less dose accumulation than U.S. LWRs. Reactors of the HTGR type, both Peach Bottom and Fort St. Vrain, as well as the large HTGR-SC, fall in line with this observation, having annual collective dose per reactor or man-rem/MW(e)y values substantially lower than LWRs.

ACKNOWLEDGMENT

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