11.1 SOURCE TERMS

General Electric has evaluated radioactive material sources (activation products and fission product release from fuel) in boiling water reactors (BWRs) based upon early operating experience. Release of radioactive material from operating BWRs has generally resulted in doses to offsite persons which have been only a small fraction of permissible doses, or of natural background dose.

The information provided in this section defines the design basis radioactive material levels in the reactor water, steam and offgas. The various radioisotopes listed have been grouped as coolant activation products, non-coolant activation products, and fission products. The fission product levels are based on measurements of BWR reactor water and off-gas at several stations through mid-1971. Emphasis was placed on observations made at KRB and Dresden 2. The design basis radioactive material levels do not necessarily include all the radioisotopes observed or predicted theoretically to be present. The radioisotopes included are considered significant to one or more of the following criteria:

(1)   plant equipment design,
(2)   shielding design,
(3)   understanding system operation and performance,
(4)   measurement practicability, and
(5)   evaluating radioactive material releases to the environment.

For halogens, radioisotopes with half-lives less than 3 minutes were omitted. For other fission product radioisotopes in reactor water, radioisotopes with half-lives less than 10 minutes were not considered.

Design basis radiation source data applicable to SSES power uprate and Hydrogen Water Chemistry are presented in References 11.1-8 and 11.1-14, respectively. The original SSES design basis source terms in the reactor coolant (liquid and steam) are based primarily on the methodology contained in NEDO-10871, Reference 11.1-9 and were normally used by GE for evaluation of 12 and 18 month fuel cycles. Operation of the reactor, with thermal power increases from 5 to 25 percent, would be expected to have some effect on observed coolant radionuclide concentrations. However, the design basis data utilized in the development of the radiation source terms is considered to contain sufficient margin that application of the data for power levels in the range given is conservative and reasonable. In addition, it was determined (Reference 11.1-10) that the power uprate source term is applicable for use in evaluating 24 month cycles and the current fuel configuration per Section 4.2. As such, continued use of the original design basis concentrations, without change, is being maintained for plant operation under power uprate conditions with 24 month fuel cycle.

11.1.1   FISSION PRODUCTS

11.1.1.1   Noble Radiogas Fission Products

The noble radiogas fission product source terms observed in operating BWRs are generally complex mixtures whose sources vary from minuscule defects in cladding to "tramp" uranium on external cladding surfaces. The relative concentrations or amounts of noble radiogas isotopes can be described as follows:

Equilibrium: $\quad R_g - K_1 Y$ (11.1-1)

Recoil: $\quad R_g - K_2 Y \lambda$ (11.1-2)

The nomenclature in Subsection 11.1.1.4 defines the terms in these and succeeding equations. The constants K_1 and K_2 describe the fractions of the total fissions that are involved in each of the releases. The equilibrium and recoil mixtures are the two extremes of the mixture spectrum that are physically possible. When a sufficient time delay occurs between the fission event and the time of release of the radiogases from the fuel to the coolant, the radiogases approach equilibrium levels in the fuel and the equilibrium mixture results. When there is no delay or impedance between the fission event and the release of the radiogases, the recoil mixture is observed.

Prior to Vallecitos Boiling Water Reactor (VBWR) and Dresden 1 experience, it was assumed that noble radiogas leakage from the fuel would be the equilibrium mixture of the noble radiogases present in the fuel.

VBWR and early Dresden 1 experience indicated that the actual mixture most often observed approached a distribution which was intermediate in character to the two extremes (Reference 11.1-1). This intermediate decay mixture was termed the "diffusion" mixture. It must be emphasized that this "diffusion" mixture is merely one possible point on the mixture spectrum ranging from the equilibrium to the recoil mixture and does not have the absolute mathematical and mechanistic basis for the calculational methods possible for equilibrium and recoil mixtures. However, the "diffusion" distribution pattern which has been described is as follows:

Diffusion: $R_g - K_3 Y \lambda^{0.5}$ (11.1-3)

The constant K_3 describes the fraction of total fissions that are involved in the release. The value of the exponent of the decay constant, λ, is midway between the values for equilibrium, 0, and recoil, 1. The "diffusion" pattern value of 0.5 was originally derived from diffusion theory.

Although the previously described "diffusion" mixture was used by GE as a basis for design since 1963, the design basis release magnitude used has varied from 0.5 Ci/sec to 0.1 Ci/sec as measured after 30-min decay (t = 30 min). The noble radiogas source-term rate after 30-min decay has been used as a conventional measure of the design basis fuel leakage rate since it is conveniently measurable and was consistent with the nominal design basis 30-min off-gas holdup system used on a number of plants. Since about 1967, the design basis release magnitude used (including the 1971 source terms) was established at an annual average of 0.1 Ci/sec (t = 30 min). This design basis is considered as an annual average with some time above and some time below this value. This design value was selected on the basis of operating experience rather than predictive assumptions. Several judgment factors, including the significance of environmental release, reactor water radioisotope concentrations, liquid waste handling and effluent disposal criteria, building air contamination, shielding design, and turbine and other component contamination affecting maintenance, have been considered in establishing this level.

Noble radiogas source terms from fuel above 0.1 Ci/sec (t = 30 min) can be tolerated for reasonable periods of time. Continual assessment of these values is made on the basis of actual operating experience in BWRs (References 11.1-2 and 11.1-3).

While the noble radiogas source-term magnitude was established at 0.1 Ci/sec (t = 30 min), it was recognized that there may be a more statistically applicable distribution for the noble radiogas mixture. Sufficient data was available from KRB operations from 1967 to mid-1971 along with Dresden 2 data from operation in 1970 and several months in 1971 to more accurately characterize the noble radiogas mixture pattern for an operating BWR.

The basic equation for each radioisotope used to analyze the collected data is:

$$R_g = K_g Y \lambda^m \left(1 - e^{\lambda T}\right) e^{-\lambda t} \qquad (11.1\text{-}4)$$

With the exception of Kr-85 with a half-life of 10.74 yr, the noble radiogas fission products in the fuel are essentially at an equilibrium condition after an irradiation period of several months (rate of formation is equal to the rate of decay). So for practical purposes the term $(1 - e^{-\lambda t})$ approaches 1 and can be neglected when the reactor has been operating at steady-state for long periods of time. The term $(e^{-\lambda t})$ is used to adjust the releases from the fuel (t = 0) to the decay time for which values are needed. Historically t = 30 min has been used. When discussing long steady-state operation and leakage from the fuel (t = 0), the following simplified form of Equation 11.1-4 can be used to describe the leakage of each noble radiogas:

$$R_g = K_g Y \lambda^m \qquad (11.1\text{-}5)$$

The constant, K_g, describes the magnitude of leakage. The relative rates of leakage of the different noble radiogas isotopes is accounted for by the variable, m, the exponent of the decay constant, λ.

Dividing both sides of Equation 11.1-5 by Y, the fission yield, and taking the logarithm of both sides results in the following equation:

$$\log(R_g/Y) = m \log (\lambda) + \log (K_g) \qquad (11.1\text{-}6)$$

Equation 11.1-6 represents a straight line when log (R_g/Y) is plotted versus log (λ); where m is the slope of the line. This straight line is obtained by plotting (R_g/Y) versus (λ) on logarithmic graph paper. By fitting actual data from KRB and Dresden 2 (using least squares techniques) to the equation the slope, m, can be obtained. This can be estimated on the plotted graph. With radiogas leakage at KRB over the nearly 5-yr period varying from 0.001 to 0.056 Ci/sec (t = 30 min) and with radiogas leakage at Dresden 2 varying from 0.001 to 0.169 Ci/sec (t = 30 min), the average value of m was determined. The value for m is 0.4 with a standard deviation of ± 0.07. This is illustrated in Figure 11.1-1 as a frequency histogram. As can be seen from this figure, variations in m were observed in the range m = 0.1 to m = 0.6. After establishing the value of m = 0.4, the value of K_g can be calculated by selecting a value for R_g, or as has been done historically, the design basis is set by the total design basis source-term magnitude at t = 30 min. With R_g at 30 min = 100,000 Ci/sec, K_g can be calculated as being 2.6 x 10^7 and Equation 11.1-4 becomes:

$$R_g = 2.6 \times 10^7 \, Y\lambda^{0.4} \, (1 - e^{-\lambda T}) \, (e^{-\lambda t}) \qquad (11.1\text{-}7)$$

This noble radiogas source-term mixture has been termed the "1971 Mixture" to differentiate it from the "diffusion mixture." The noble gas source term for each radioisotope can be calculated

SSES-FSAR

from Equation 11.1-7. The resultant source terms are presented in Table 11.1-1 as leakage from fuel (t = 0) and after 30 min decay. While Kr-85 can be calculated using Equation 11.1-7, the number of confirming experimental observations was limited by the difficulty of measuring very low release rates of this isotope. Therefore, the table provides an estimated range for Kr-85 based on a few actual measurements.

11.1.1.2   Radiohalogen Fission Products

Historically, the radiohalogen design basis source term was established by the same equation as that used for noble radiogases. In a fashion similar to that used with gases, a simplified equation can be shown to describe the release of each halogen radioisotope:

$$R_h = K_h Y \lambda^n \qquad (11.1-8)$$

The constant, K_h, describes the magnitude of leakage from fuel. The relative rates of halogen radioisotope leakage is expressed in terms of n, the exponent of the decay constant, λ. As was done with the noble radiogases, the average value was determined for n. The value for $\bar{n}$ is 0.5 with a standard deviation of ± 0.19. This is illustrated in Figure 11.1-2 as a frequency histogram. As can be seen from this figure, variations in n were observed in the range of n = 0.1 to n = 0.9.

It appeared that the use of the previous method of calculating radio-halogen leakage from fuel was overly conservative. Figure 11.1-3 relates KRB and Dresden 2 noble radiogas versus I-131 leakage. While it can be seen from Dresden 2 data during the period August 1970 to January 1971 that there is a relationship between noble radiogas and I-131 leakage under one fuel condition, there was no simple relationship for all fuel conditions experienced. Also, it can be seen that during this period, high radiogas leakages were not accompanied by high radioiodine leakage from the fuel. Except for one KRB datum point, all steady-state I-131 leakages observed at KRB or Dresden 2 were equal to or less than 505µCi/sec I-131. Even at Dresden 1 in March 1965, when severe defects were experienced in stainless-steel-clad fuel, I-131 leakages greater than 500µCi/sec I-131 were not experienced. Figure 11.1-3 shows that these higher radioiodine leakages from the fuel were related to noble radiogas source terms of less than the design basis value of 0.1 Ci/sec (t = 30 min). This may be partially explained by inherent limitations due to internal plant operational problems that caused plant derating.

In general, it would not be anticipated that operation at full power would continue for any significant time period with fuel cladding defects which would be indicated by I-131 leakage from the fuel in excess of 700µCi/sec. When high radiohalogen leakages are observed, other fission products will be present in greater amounts. This may increase potential radiation exposure to operating and maintenance personnel during plant outages following such operation.

Using these judgement factors and above experience, the design basis radiohalogen source terms from fuel were established based on I-131 leakage of 700µCi/sec. This value, as seen in Figure 11.1-3, accommodates the experience data and the design basis noble radiogas source term of 0.1 Ci/sec (t = 30 min). With the I-131 design basis source term established, K_h can be calculated as being 2.4×10^7 and halogen radioisotope release can be expressed by the following equation:

$$R^h = 2.4 \times 10^7 \, Y\lambda^{0.5}(1-e^{\lambda T})(e^{-\lambda t}) \qquad (11.1-9)$$

Concentrations of radiohalogens in reactor water can be calculated using the following equation:

(11.1-10)

$$C_h = \frac{R_h}{(\lambda+\beta+\nu)M}$$

The concentration of radiohalogens in reactor steam can be determined by multiplying the reactor water concentration by the carryover fraction. The carryover fraction for radiohalogens is influenced by reactor water chemistry parameters such as the copper concentration and the concentration of oxidizing agents. For BWR's operating without hydrogen injection (Normal Water Chemistry), the observed "carryover" for radiohalogens has varied from about 0.1% when copper concentrations are high to about 2% when concentrations are low. The average of observed radiohalogen carryover measurements has been 1.2% by weight of reactor water in steam with a standard deviation of ± 0.9. Since Susquehanna is designed with no copper in the steam cycle, a radiohalogen carryover of 2% (0.02 fraction) was used for Normal Water Chemistry operation.

Hydrogen Water Chemistry has been implemented to control the potential for stress corrosion cracking of vessel internals. Hydrogen is injected into the feedwater to reduce the radiolytic production of oxygen and hydrogen peroxide and to promote recombination of residual oxidants. Under Hydrogen Water Chemistry and conditions of low copper, the carryover fraction for halogen isotopes can be enhanced as reported in Reference 11.1-12. For a given release rate from the fuel, this will reduce the concentrations of radiohalogen isotopes in reactor water and increase their concentrations in reactor steam.

To bound operation under both Normal and Hydrogen Water Chemistry, the concentrations of radiohalogens in reactor water and steam are calculated using assumptions that maximize the concentrations. Reactor water concentrations are calculated assuming Normal Water Chemistry with 2% carryover. Reactor steam concentrations are calculated by assuming Hydrogen Water Chemistry with 8% carryover. The steam concentration was calculated by multiplying the reactor water concentration under Hydrogen Water Chemistry by 0.08 (8% carryover). The resultant concentrations are presented in Table 11.1-2.

11.1.1.3   Other Fission Products

The observations of other fission products (and transuranic nuclides, including Np-239) in operating BWRs are not adequately correlated by simple equations. For these radioisotopes, design basis concentrations in reactor water have been estimated conservatively from experience data and are presented in Table 11.1-3. Carryover of these radioisotopes from the reactor water to the steam is estimated to be < 0.1% (< 0.001 fraction). There is no measurable change in carryover for these other fission products under HWC. In addition to carryover, however, decay of noble radiogases in the steam leaving the reactor will result in production of noble gas daughter radioisotopes in the steam and condensate systems.

Some daughter radioisotopes (e.g., yttrium and lanthanum), were not listed as being in reactor water. Their independent leakage to the coolant is negligible; however, these radioisotopes may be observed in some samples in equilibrium or approaching equilibrium with the parent radioisotope.

Except for Np-239, trace concentrations of transuranic isotopes have been observed in only a few samples where extensive and complex analyses were carried out. The predominant alpha

emitter present in reactor water is Cm-242 at an estimated concentration of $10^{-6}\mu Ci/g$ or less, which is below the maximum permissible concentration in drinking water application to continuous use by the general public. The concentration of alpha-emitting plutonium radioisotopes is more than one order of magnitude lower than that of Cm-242.

Plutonium-241 (a beta emitter) may also be present in concentrations comparable to the Cm-242 level.

<u>11.1.1.4   Nomenclature</u>

The following list of nomenclature defines the terms used in equations for source-term calculations:

R_g =   leakage rate of a noble gas radioisotope ($\mu Ci/sec$)

R_h =   leakage rate of a halogen radioisotope ($\mu Ci/sec$)

Y   =   fission yield of a radioisotope (atoms/fission)

λ =   decay constant of a radioisotope (sec^{-1})

T   =   fuel irradiation time (sec)

t   =   decay time following leakage from fuel (sec)

m   =   noble radiogas decay constant exponent (dimensionless)

n   =   radiohalogen decay constant exponent (dimensionless)

K_g =   a constant establishing the level of noble radiogas leakage from fuel

K_h =   a constant establishing the level of radiohalogen leakage from fuel

C_h =   concentration of a halogen radioisotope in reactor water ($\mu Ci/grams$)

M   =   mass of water in the operating reactor (grams)

ß   =   cleanup system removal constant (sec^{-1}) = $\dfrac{\text{cleanup system flowrate(grams/sec)}}{M}$

ν =   halogen steam carryover removal constant (sec^{-1})

= $\dfrac{\text{steam flowrate (g ram/sec) x (halogen carryover fraction)}}{M}$

11.1.2   ACTIVATION PRODUCTS

11.1.2.1   Coolant Activation Products

The coolant activation products are not adequately correlated by simple equations. Design basis concentrations in reactor water and steam have been estimated conservatively from experience data. The resultant concentrations are presented in Table 11.1-4.

Under conditions of Hydrogen Water Chemistry (HWC), enhanced evolution of nitrogen to steam is experienced in a non-linear fashion as reported in EPRI Report TR-103515 (Reference 11.1-13). At conditions of moderate hydrogen injection ($\leq$2.0 ppm in feedwater), radiation rates and equivalently nitrogen concentrations in the steam increase up to a factor of five above normal rates. Table 11.1-4 shows steam concentrations both with and without moderate HWC. In Table 11.1-4, other isotopes than nitrogen are shown but the affect of HWC upon these isotopes is not characterized because of (1) experimental difficulties in making meaningful measurements and (2) the overwhelming dominance of nitrogen in the radiation signature of the steam. Therefore, the concentrations of these isotopes are not assumed to be changed in any significant manner under HWC. In addition, though the water concentrations of nitrogen will decrease under HWC, the decrease is ignored for purposes of conservatively bounding the majority of normal operating conditions.

11.1.2.2   Noncoolant Activation Products

The activation products formed by activation of impurities in the coolant or by corrosion of irradiated system materials are not adequately correlated by simple equations. The design basis source terms of noncoolant activation products have been estimated conservatively from experience data. The resultant concentrations are presented in Table 11.1-5. Carryover of these isotopes from the reactor water to the steam is estimated to be < 0.1% (< 0.001 fraction).

11.1.3   TRITIUM

In a BWR, tritium is produced by three principal methods:

(1)   activation of naturally occurring deuterium in the primary coolant,

(2)   nuclear fission of UO_2 fuel, and

(3)   neutron reactions with boron used in reactivity control rods.

The tritium, formed in control rods may be released from a BWR in liquid or gaseous effluents. A prime source of tritium available for release from a BWR is that produced from activation of deuterium in the primary coolant. Some fission product tritium may also transfer from fuel to primary coolant. This discussion is limited to the uncertainties associated with estimating the amounts of tritium generated in a BWR which are available for release.

All of the tritium produced by activation of deuterium in the primary coolant is available for release in liquid or gaseous effluents. The tritium formed in a BWR can be calculated using the equation:

$$R_{act} = \frac{\Sigma \phi V \lambda}{3.7 \times 10^4 P} \qquad (11.1-11)$$

where

R_{act} = tritium formation rate by deuterium activation (μ Ci/sec/MWt)

Σ = macroscopic thermal neutron cross section (cm^{-1})

ϕ = thermal neutron flux (neutrons/($cm5$) (sec))

V coolant volume in core ($cm;$)

λ = tritium radioactive decay constant (1.78×10^{-9} sec^{-1})

P = reactor power level (MWt)

For recent BWR designs, R_{act} is calculated to be $1.3 \pm 0.4 \times 10^4$ μCi/sec/Mwt. The uncertainty indicated is derived from the estimated errors in selecting values for the coolant volume in the core, coolant density in the core, abundance of deuterium in light water (some additional deuterium will be present because of the H (n,γ) D reaction, thermal neutron flux, and microscopic cross section for deuterium).

The fraction of tritium produced by fission which may transfer from fuel to the coolant (which will then be available for release in liquid and gaseous effluents) is much more difficult to estimate. However, since zircaloy-clad fuel rods are used in BWRs, essentially all fission product tritium will remain in the fuel rods unless defects are present in the cladding material (Reference 11.1-4).

The study made at Dresden 1 in 1968 by the U.S. Public Health Service suggests that essentially all of the tritium released from the plant could be accounted for by the deuterium activation source (Reference 11.1-3). For purposes of estimating the leakage of tritium from defected fuel, it can be assumed that it leaks in a manner similar to the leakage of noble radiogases. Thus, use can be made of the empirical relationship described as the "diffusion mixture" used for predicting the source term of individual noble gas radioisotopes as a function of the total noble gas source term. The equation which describes this relationship is:

$$R_{dif} = KY\lambda \qquad (11.1-12)$$

where,

R_{dif} = leakage rate of tritium from fuel (μ Ci/sec)

Y = fission yield fraction (atoms/fission)

λ = radioactive decay constant (sec^{-1})

K = a constant related to total tritium leakage rate

If the total noble radiogas source term is $10^5 \mu$Ci/sec after 30-min decay, leakage from fuel can be calculated to be about 0.24μCi/sec of tritium. To place this value in perspective in the USPHS study, the observed rate of Kr-85 (which has a half-life similar to that of tritium) was 0.06 to 0.4 times that calculated using the "diffusion mixture" relationship. This would suggest that the actual tritium leakage rate might range from 0.015 to 0.10μCi/sec. Since the annual average noble radiogas leakage from a BWR is expected to be less than 0.1 Ci/sec (t = 30 min), the annual average tritium release rate from the fission source can be conservatively estimated at 0.12 $\pm$ 0.12μCi/sec, or 0.0 to 0.24μCi/sec.

For this reactor the estimated total tritium appearance rate in reactor coolant and release rate in the effluent is about 150 Ci/year/unit.

Tritium formed in the reactor is generally present as tritiated oxide (HTO) and to a lesser degree as tritiated gas (HT). Tritium concentration in the steam formed in the reactor will be the same as in the reactor water at any given time. This tritium concentration will also be present in condensate and feedwater. Since radioactive effluents generally originate from the reactor and power cycle equipment, radioactive effluents will also have this tritium concentration. Condensate storage receives treated water from the radioactive waste system and reject water from the condensate system. Thus, all plant process water will have a common tritium concentration.

Off-gases released from the plant will contain tritium, which is present as tritiated gas (HT) resulting from reactor water radiolysis as well as tritiated water vapor (HTO). In addition, water vapor from the turbine gland seal steam packaging exhauster and a lesser amount present in ventilation air due to process steam leaks or evaporation from sumps, tanks, and spills on floors will also contain tritium. The remainder of the tritium will leave the plant in liquid effluents or with solid wastes.

Recombination of radiolysis gases in the air ejector off-gas system will form water, which is condensed and returned to the main condenser. This tends to reduce the amount of tritium leaving in gaseous effluents. Reducing the gaseous tritium release will result in a slightly higher tritium concentration in the plant process water. Reducing the amount of liquid effluent discharged will also result in a higher process coolant equilibrium tritium concentration.

Essentially, all tritium entering the primary coolant will eventually be released to the environs, either as water vapor and gas to the atmosphere, or as liquid effluent to the plant discharge or as solid waste. Reduction due to radioactive decay is negligible due to the 12-yr. half-life of tritium.

The USPHS study at Dresden 1 estimated that approximately 90% of the tritium release was observed in liquid effluent, with the remaining 10% leaving as gaseous effluent (Reference 11.1-5). Efforts to reduce the volume of liquid effluent discharges may change this distribution so that a greater amount of tritium will leave as gaseous effluent. From a practical standpoint, the fraction of tritium leaving as liquid effluent may vary between 60 and 90% with the remainder leaving in gaseous effluent.

11.1.4 CORE INVENTORY, FUEL EXPERIENCE AND DEPRESSURIZATION SPIKING

11.1.4.1 Core Inventory

Core fission product inventory information is used in establishing fission product source terms for accident analysis and is, therefore, discussed in Chapter 15 and Chapter 18.

11.1.4.2 Fuel Experience

A discussion of fuel experience gained for BWR fuel including failure experience, burnup experience, and thermal conditions under which the experience was gained is available in three GE topical reports (references 11.1-2, 11.1-3 and 11.1-6). The basis for this experience is fuel produced and used during the late 1960's and early 1970's, generally with 7x7 fuel lattices. In the two decades since these reports were produced, fuel manufacturing and design has advanced through 8x8, 9x9 and 10x10 lattice designs using barrier, partial length, and formed (variable loading) fuel rods. With these advances, fuel performance has gradually improved to the point that fuel rod leaks are the exception in reactor performance. Along with performance improvement, the operating cycle lengths have also increased to the degree that 24 month operating cycles and plant availabilities in the range of 80% are becoming the norm. Increases in operating cycle length involve higher fuel exposures, but do not result in significant increase in radiologically significant fission products with the exception of isotopes of Cs-134, Cs-137, and Sr-90. Gap or plenum inventories are expected to increase with extended exposure which result in higher core sources to the reactor coolant. Manufacturing improvements and quality control have reduced leakage source terms so that no increase in coolant concentrations have been observed that vary significantly over current values. The existing design basis concentrations as given in Tables 11.1-1, 11.1-2, 11.1-3, 11.1-4 and 11.1-5 still effectively bound operating experiences (often by orders of magnitude). Reference 11.1-11 discusses the basis for these conclusions.

11.1.4.3 Depressurization Spiking

The data presented in Table 11.1-6 provides a conservative representation of "spiking" source terms for iodines and noble gases. The tabulated values represent the total activity per bundle which is available for release during a complete depressurization of the reactor vessel. These source terms may be applied in analysis of events in which reactor coolant is released while the vessel is being depressurized.

The data were developed from early observations of spiking releases at plants with 7x7 fuel assemblies and projected to estimate the activities associated with a 95th percentile cumulative probability spiking event with 8x8 fuel. Given the greatly improved performance of current fuel designs, it is expected that the data will be conservative for power uprate. Further, Reference 11.1-10 concluded that the use of the power uprate data can be continued for the extended 24 month fuel cycle and is applicable to other fuel types of different mechanical design.

11.1.5 PROCESS LEAKAGE SOURCES

Process leakage results in potential release paths for noble gases and other volatile fission products via ventilation systems. Liquid from process leaks are collected and routed to the liquid and waste management system. Radionuclide releases via ventilation paths are at extremely low levels and have been insignificant compared to process off-gas from operating BWR plants. However, because the implementation of improved process off-gas treatment systems make the ventilation release relatively significant, General Electric has conducted measurements to identify and qualify these low-level release paths. General Electric has

maintained an awareness of other measurements by the Electric Power Research Institute and other organizations; and routine measurements by utilities with operating BWRs.

Leakage of fluids from the process system will result in the release of radionuclides into plant buildings. In general, the noble radiogases will remain airborne and will be released to the atmosphere with little delay via the building ventilation exhaust ducts. The radionuclides will partition between air and water, and airborne radioiodines may "plateout" on metal surfaces, concrete, and paint. A significant amount of radioiodine remains in air or is desorbed from surfaces. Radioiodines are found in ventilation air as methyl iodide and as inorganic iodine which is here defined as particulate, elemental, and hypoiodous acid forms of iodine. Particulates will also be present in the ventilation exhaust air.

Experience with the airborne radiological releases from BWR building heating, ventilating, and air conditioning and the main condenser mechanical vacuum pump have been compiled and evaluated in NEDO-21159, "Airborne Releases from BWRs for Environmental Impact Evaluations", March 1976, Licensing Topical Report (Reference 11.1-7). This report is periodically updated to incorporate the most recent data on airborne emission. The results of these evaluations are based on data obtained by utility personnel and special in-plant studies of operating BWR plants by independent organizations and the General Electric Company. An evaluation of the radioactive releases from ventilation systems, for compliance with Appendix I to 10CFR50, is given in Section 11.3. An evaluation of important exposure to airborne activity is given in Subsection 12.2.2.

11.1.6 OTHER RELEASES

All other releases are covered in Section 11.3.

11.1.7 REFERENCES

11.1-1   Brutschy, F. J., "A Comparison of Fission Product Release Studies in Loops and VBWR," Paper presented at the Tripartite Conference on Transport of Materials in Water Systems, Chalk River, Canada (February 1961).

11.1-2   Williamson, H. E., Ditmore, D.C., "Experience with BWR Fuel Through September 1971" NEDO-10505, May 1972. (Update).

11.1-3   Elkins, R. B., "Experience with BWR Fuel Through September 1974," NEDO-20922, June 1975.

11.1-4   Ray, J. W., "Tritium in Power Reactors," Reactor and Fuel Processing Technology, 12 (1), pp. 19-26, Winter 1968-1969.

11.1-5   Kahn, B., et al, "Radiological Surveillance Studies at a Boiling Water Nuclear Power Reactor," BRH/DER 70-1, March 1970.

11.1-6   Williamson, H. E., Ditmore, D. C., "Current State of Knowledge of High Performance BWR Zircaloy Clad U02 Fuel," NEDO-10173, May 1970.

11.1-7   Marrero, T. R., "Airborne Releases From BWRs for Environmental Impact Evaluations," NEDO-21159, March 1976.

11.1-8   Pankratz, D. R., Faynshtein, K, "Power Uprate Engineering Report For Susquehanna Steam Electric Station Units 1 and 2," NEDC-32161P, December 1993.

11.1-9   "Technical Derivation of BWR 1971 Design Basis Radioactive Material Source Terms", NEDO-10871, March 1973.

11.1-10   GE Document B13-01869-095, Susquehanna Steam Electric Station Extended Fuel Cycle, Radiological Source Terms, S.S. Wang, August 1997. (PP&L Calculation EC-FUEL-1333 Rev. 0, "GE Source Term Study for 24 Month Cycles").

11.1-11   Baker, D.A., et al, "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors", NUREG/CR-5009 (PNL-6258), U.S. NRC, February 1988

11.1-12   Lin, C.C. "Chemical Behavior of Radioiodine in Boiling Water Reactor Systems II: Effects of Hydrogen Water Chemistry", Nuclear Technology, Vol. 97, Jan. 1992, pg. 71-78.

11.1-13   "BWR Water Chemistry Guidelines – 1996 Revision, Normal and Hydrogen Water Chemistry", EPRI TR-103515, R1, BWRUIP-29, December, 1996.

11.1-14   Careway, H.A., and E. M. Skeehan, "Susquehanna Steam Electric Station, Units 1 and 2, Radioactive Material Source Term Evaluation for Normal Conditions with Hydrogen Water Chemistry." GE Document GE-NEA41-00074-07, Revision 1. April 1997.

TABLE 11.1-1			
NOBLE RADIOGAS SOURCE TERMS			
Isotope	Half-life	Source Term @ t = 0 (μ Ci/sec)	Source Term @ t = 30 min (μ Ci/sec)
Kr-83m	1.86 hr	3.4×10^3	2.9×10^3
Kr-85m	4.4 hr	6.1×10^3	5.6×10^3
Kr-85	10.74 yr	10 to 20*	10 to 20*
Kr-87	76 min	2.0×10^4	1.5×10^4
Kr-88	2.79 hr	2.0×10^4	1.8×10^4
Kr-89	3.18 min	1.3×10^5	1.8×10^2
Kr-90	32.3 sec	2.8×10^5	------
Kr-91	8.6 sec	3.3×10^5	------
Kr-92	1.84 sec	3.3×10^5	------
Kr-93	1.29 sec	9.9×10^4	------
Kr-94	1.0 sec	2.3×10^4	------
Kr-95	0.5 sec	2.1×10^3	------
Kr-97	1.0 sec	1.4×10^1	------
Xe-131m	11.96 day	1.5×10^1	1.5×10^1
Xe-133m	2.26 day	2.9×10^2	2.8×10^2
Xe-133	5.27 day	8.2×10^3	8.2×10^3
Xe-135m	15.7 min	2.6×10^4	6.9×10^3
Xe-135	9.16 hr	2.2×10^4	2.2×10^4
Xe-137	3.82 min	1.5×10^5	6.7×10^2
Xe-138	14.2 min	8.9×10^4	2.1×10^4
Xe-139	40 sec	2.8×10^5	------
Xe-140	13.6 sec	3.0×10^5	------
Xe-141	1.72 sec	2.4×10^5	------
Xe-142	1.22 sec	7.3×10^4	------
Xe-143	0.96 sec	1.2×10^4	------
Xe-144	9.0 sec	5.6×10^2	------
TOTALS		$\sim 2.5 \times 10^5$	$\sim 1.0 \times 10^5$
* Estimated from experimental observations.			

TABLE 11.1-2

HALOGEN RADIOISOTOPES IN REACTOR WATER AND STEAM

Isotope	Half-Life	Water Concentration (μCi/g)	Steam Concentration (μCi/g)
Br-83	2.40 hr	1.5×10^{-2}	4.9×10^{-4}
Br-84	31.8 min	2.7×10^{-2}	1.2×10^{-3}
Br-85	3.0 min	1.7×10^{-2}	1.2×10^{-3}
I-131	8.065 day	1.3×10^{-2}	3.4×10^{-4}
I-132	2.284 hr	1.2×10^{-1}	3.9×10^{-3}
I-133	20.8 hr	8.9×10^{-2}	2.4×10^{-3}
I-134	52.3 min	2.4×10^{-1}	9.7×10^{-3}
I-135	6.7 hr	1.3×10^{-1}	3.7×10^{-3}

Water concentrations maximized by use of 2% steam carryover, while steam concentrations maximized by use of 8% steam carryover commensurate with operations with hydrogen water chemistry.

TABLE 11.1-3

OTHER FISSION PRODUCT RADIOISOTOPES IN REACTOR WATER

Isotope	Half-Life	Concentration (μCi/g)
Sr-89	50.8 day	3.1×10^{-3}
Sr-90	28.9 yr	2.3×10^{-4}
Sr-91	9.67 hr	6.9×10^{-2}
Sr-92	2.69 hr	1.1×10^{-1}
Zr-95	65.5 day	4.0×10^{-5}
Zr-97	16.8 hr	3.2×10^{-5}
Nb-95	15.1 day	4.2×10^{-5}
Mo-99	66.6 hr	2.2×10^{-2}
Tc-99m	6.007 hr	2.8×10^{-1}
Tc-101	14.2 min	1.4×10^{-1}
Ru-103	39.8 day	1.9×10^{-5}
Ru-106	368 day	2.6×10^{-6}
Te-129m	34.1 day	4.0×10^{-5}
Te-132	78.0 hr	4.9×10^{-2}
Cs-134	2.06 yr	1.6×10^{-4}
Cs-136	13.0 day	1.1×10^{-4}
Cs-137	30.2 yr	2.4×10^{-4}
Cs-138	32.3 min	1.9×10^{-1}
Ba-139	83.2 min	1.6×10^{-1}
Ba-140	12.8 day	9.0×10^{-3}
Ba-141	18.3 min	1.7×10^{-1}
Ba-142	10.7 min	1.7×10^{-1}
Ce-141	32.53 day	3.9×10^{-5}
Ce-143	33.0 hr	3.5×10^{-5}
Ce-144	284.4 day	3.5×10^{-5}
Pr-143	13.58 day	3.8×10^{-5}
Nd-147	11.06 day	1.4×10^{-5}
Np-239	2.35 day	2.4×10^{-1}

TABLE 11.1-4

COOLANT ACTIVATION PRODUCTS IN REACTOR WATER AND STEAM

Isotope	Half-life	Steam Concentration (μCi/g)		Reactor Water Concentration (μCi/g)
		w/o HWC	w/HWC	
N-13	9.99 min	7×10^{-3}	3.5×10^{-2}	4×10^{-2}
N-16	7.13 sec	$5 \times 10^{+1}$	$2.5 \times 10^{+2}$	$5 \times 10^{+1}$
N-17	4.14 sec	2×10^{-2}	1×10^{-1}	6×10^{-3}
O-19	26.8 sec	8×10^{-1}	8×10^{-1}	7×10^{-1}
F-18	109.8 min	4×10^{-3}	4×10^{-3}	4×10^{-3}

(w/o HWC – without Hydrogen Water Chemistry; w/HWC – with HWC)

TABLE 11.1-5
NONCOOLANT ACTIVATION PRODUCTS IN REACTOR WATER

Isotope	Half-life	Concentration (μ Ci/g)
Na-24	15.0 hr	2.0×10^{-3}
P-32	14.31 day	2.0×10^{-5}
Cr-51	27.8 day	5.0×10^{-4}
Mn-54	313.0 day	4.0×10^{-5}
Mn-56	2.582 hr	5.0×10^{-2}
Co-58	71.4 day	5.0×10^{-3}
Co-60	5.258 yr	5.0×10^{-4}
Fe-59	45.0 day	8.0×10^{-5}
Ni-65	2.55 hr	3.0×10^{-4}
Zn-65	243.7 day	2.0×10^{-6}
Zn-69m	13.7 hr	3.0×10^{-5}
Ag-110m	253.0 day	6.0×10^{-5}
W-187	23.9 hr	3.0×10^{-3}

| TABLE 11.1-6	
DEPRESSURIZATION SPIKING ACTIVITY 95th PERCENTILE CUMULATIVE PROBABILITY	
Isotope	**Activity (curie/bundle)**
I-131	2.14
I-132	3.21
I-133	5.03
I-134	5.44
I-135	4.79
Kr-83m	0.90
Kr-85m	2.23
Kr-85	0.49
Kr-87	4.33
Kr-88	6.12
Kr-89	7.96
Xe-131m	0.066
Xe-133m	0.33
Xe-133	11.6
Xe-135m	1.80
Xe-135	11.0
Xe-137	10.5
Xe-138	10.6
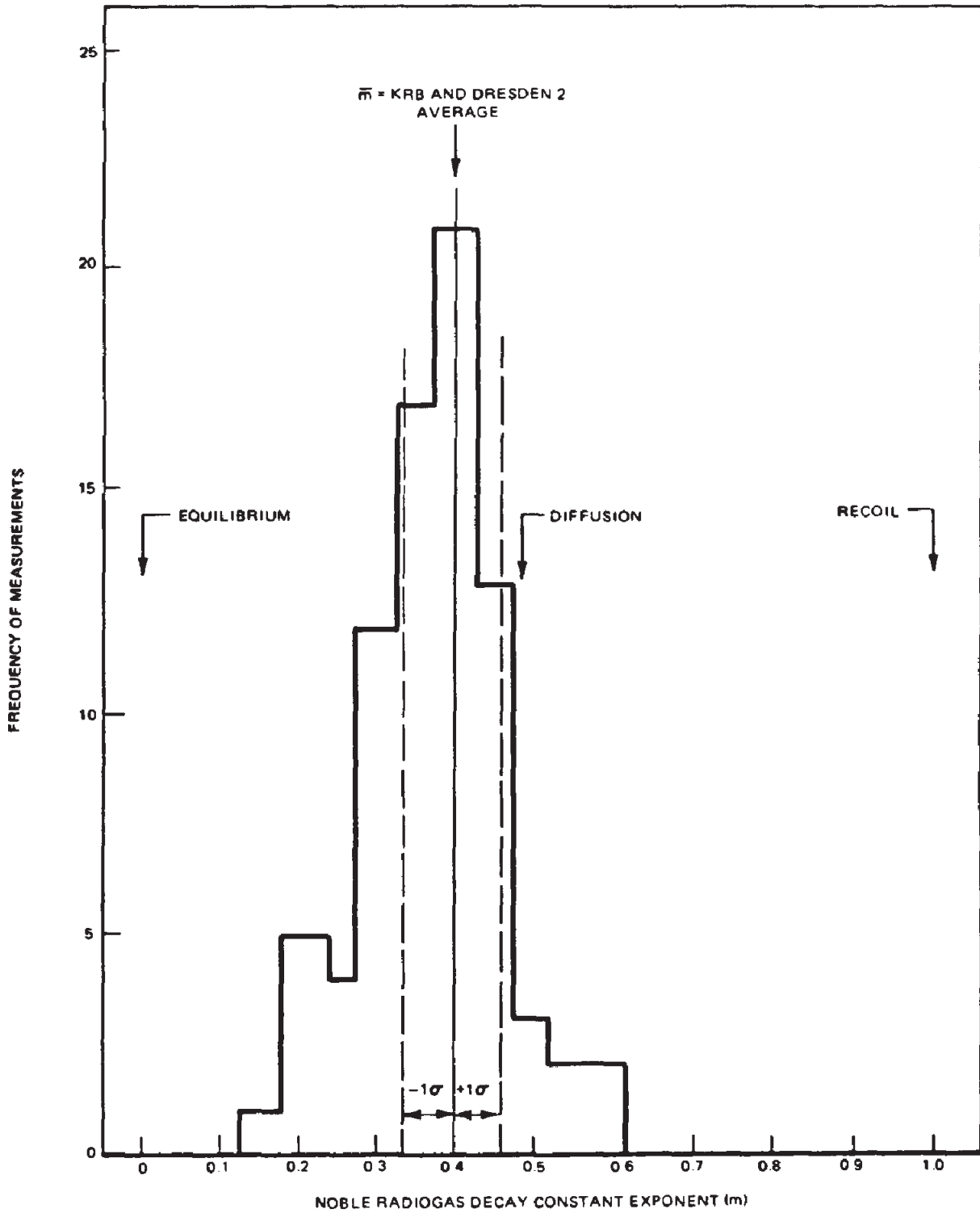
FSAR REV.65

| SUSQUEHANNA STEAM ELECTRIC STATION |
| UNITS 1 & 2 |
| FINAL SAFETY ANALYSIS REPORT |

NOBLE RADIOGAS DECAY
CONSTANT EXPONENT
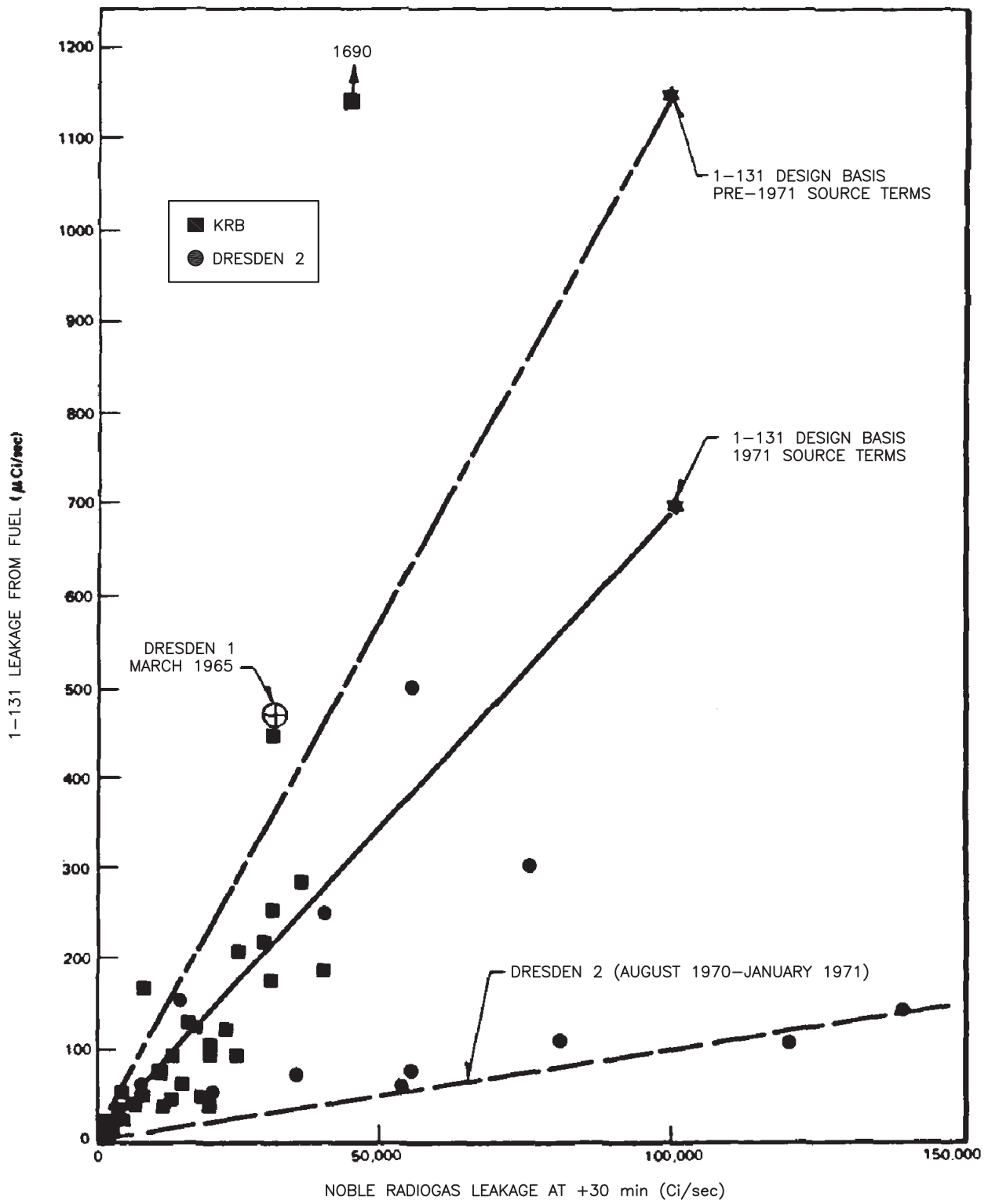FREQUENCY HISTOGRAM

FIGURE 11.1-1, Rev 47

AutoCAD: Figure Fsar 11_1_1.dwg
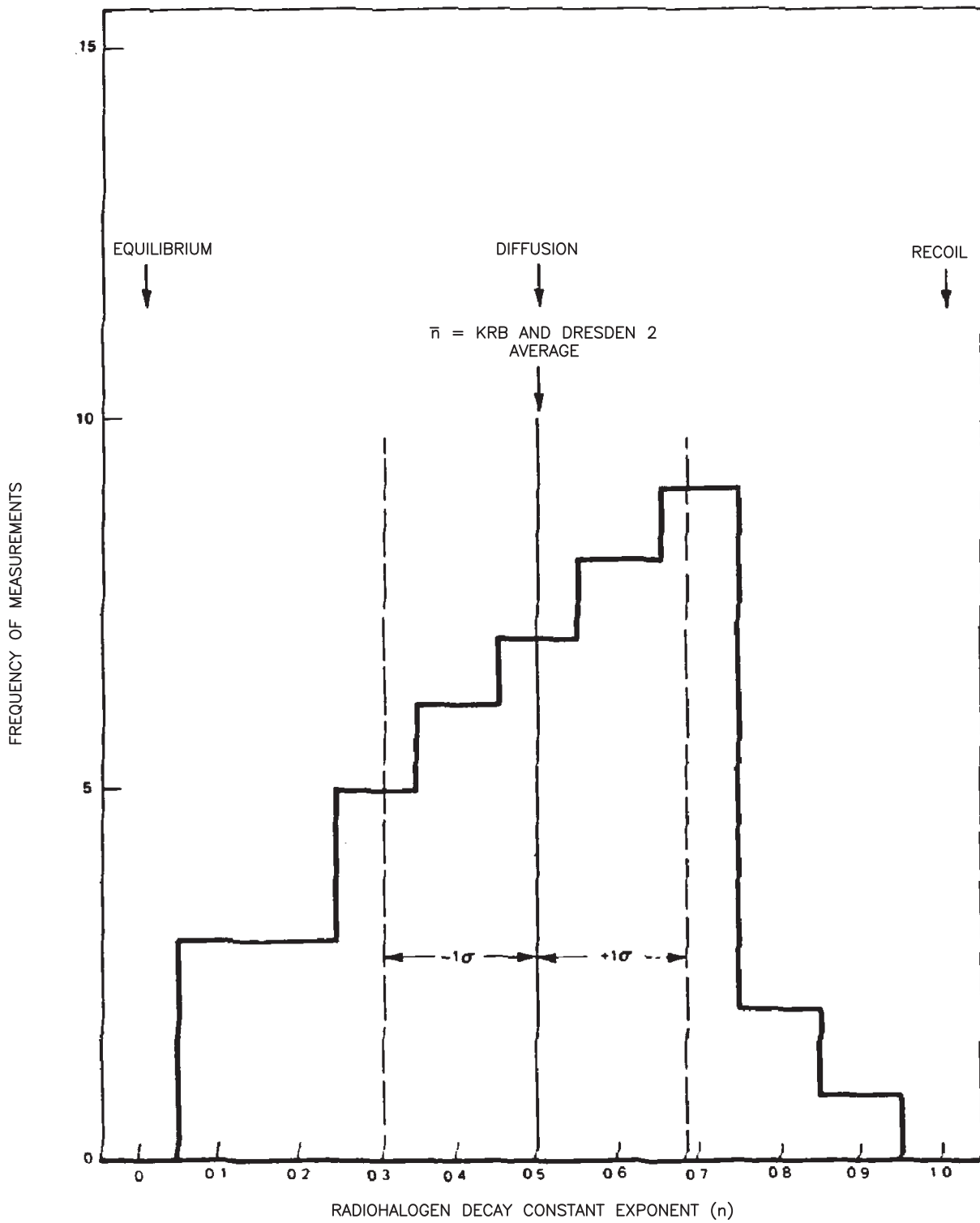
| SUSQUEHANNA STEAM ELECTRIC STATION |
| UNITS 1 & 2 |
| FINAL SAFETY ANALYSIS REPORT |

RADIOHALOGEN DECAY
CONSTANT EXPONENT
FREQUENCY HISTOGRAM

FIGURE 11.1-2, Rev 47

AutoCAD: Figure Fsar 11_1_2.dwg

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

NOBLE RADIOGAS LEAKAGE
VS.
1-131 LEAKAGE

FIGURE 11.1-3, Rev 47

AutoCAD: Figure Fsar 11_1_3.dwg