UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
COMMONWEALTH EDISON COMPANY	)	Docket Nos. 50-454
	)	50-455
(Byron Station, Units 1 and 2)	)	

NRC STAFF PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW IN THE FORM OF A SUPPLEMENTAL PARTIAL INITIAL DECISION ON OCCUPATIONAL RADIATION SAFETY AND STEAM GENERATOR ISSUES

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The NRC Staff, in accordance with 10 CFR § 2.754 and the Licensing Board's directive of May 25, 1983, proposes the following supplemental findings of fact and conclusions of law.

I. INTRODUCTION

Evidentiary hearings were held in the captioned proceeding from March 1 through May 25, 1983, with some adjournments. There were eight contested issues adjudicated during the period. Proposed findings on two contested issues[1/] were filed by the Applicant and Intervenor on May 31,

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1/ Seismology findings were designated as A-1 through A-103. Water hammer findings were designated B-1 through B-45.

1983 and by the Staff on June 10, 1983. Findings on the occupational exposure and steam generator tube integrity issues are filed herewith.[2/] Findings on emergency planning contentions are due from all parties on June 20, 1983. Findings on the liquid pathways and Class 9 accident contentions are due from all parties on June 24, 1983. No date has been set for findings on the final contention regarding quality assurance/quality control.

II. OPINION

C. Occupational Radiation Safety (League Contentions 42, 111 and 112)

Intervenor League of Women Voters of Rockford has proffered and pursued three contentions relating to the subject of occupational radiation exposure at Byron Station. (See Finding C-1). These contentions generally allege that Edison will not provide its workers with adequate protection against radiation injury and will not maintain doses "as low as reasonably achievable" (or "ALARA"). More specifically, the League points to alleged inadequacies in the design of Byron, in plant procedures relating to dosimetry recordkeeping, training and staffing, and in the in-plant monitoring of radiation. The League also charges that the use of temporary workers at Byron for various purposes creates special radiation safety problems and increases unacceptably the risk of

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2/ Occupational radiation exposure findings are designated C-1 through C-130. Steam generator tube integrity findings are D-1 through D-243.

sabotage. Intervenor asserts that neither the Commission's safety regulations nor the requirements of the National Environmental Policy Act[3/] ("NEPA") have been met.

For the reasons discussed at length below, we find that these contentions lack merit. Extensive and detailed testimony was presented by the Applicant rebutting each of the League's claims. The Staff's review of Edison's compliance with the applicable regulatory requirements concerning radiation protection of workers, which concluded that Byron's ALARA design, program and procedures are acceptable, was discussed in testimony by the cognizant Staff members. The record in support of the conclusion that Edison satisfies the regulations and will maintain exposures ALARA is overwhelming. Equally compelling is the conclusion that the Staff has met its obligations under NEPA.

1. Regulatory Requirements

Standards for protection against radiation are provided in 10 CFR Part 20. In relation to occupational exposure to radiation, Part 20 of the Commission's regulations sets permissible dose and concentration limits for radioactivity, designates precautionary procedures for radiation protection, and establishes requirements for records, reports and notification. Specific dose standards for individuals in restricted areas of the plant are provided in 10 CFR § 20.101. More generally, however, 10 CFR § 20.1(c) directs that every license "should, in

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3/ 42 USC § 4321 *et seq*.

addition to complying with the requirements set forth in this part, make every reasonable effort to maintain radiation exposures . . . as low as is reasonably achievable."[4/]

Certain aspects of these contentions also raise questions about Edison's training of its employees to assure radiation safety and about potential security problems caused by the use of temporary, non-Edison employees onsite for specific purposes. Training and information requirements for workers who may frequent restricted areas of the plant are established by 10 CFR Part 19. In particular, 10 CFR § 19.12 requires that such workers "be instructed in the health protection problems associated with exposure to such radioactive materials or radiation, in precautions or procedures to minimize exposure, and in the purposes and functions of protective devices employed." Regulatory provisions concerning the physical protection of nuclear plants and nuclear materials are contained in 10 CFR Part 73.

Finally, NEPA requires the Commission to consider environmental factors in granting, denying or imposing conditions on a license for a nuclear power reactor. Calvert Cliffs Coordinating Committee v. AEC,

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4/ Questions were raised during the hearing about the use and meaning of the word "should" rather than "shall" in 10 CFR § 20.1(c). Staff witnesses testified that "should" is interpreted as mandatory by the Staff, as shown by the Staff's Regulatory Guides 8.8 and 8.10. (Finding C-7). Staff witnesses also expressed their belief that no license would be issued in the absence of an acceptable ALARA program. (Finding C-8). In any event, Edison has such a program for Byron and has made enforceable commitments concerning its implementation and operation. Moreover, certain aspects of the radiation protection program for Byron are embodied in the technical specifications which will be a part of any license issued for facility operation. (Finding C-9).

449 F.2d 1109 (D.C.Cir. 1971). Occupational radiation exposure and its effects constitute one type of environmental impact which is addressed in the Staff's Final Environmental Statement ("FES") for the Byron Station.

2. Radiation Health Effects

a. Background

Under normal operating conditions, the radioactive isotopes present in a nuclear power plant emit radiation. This radiation is primarily gamma radiation, but beta, alpha and neutron radiation also exist in minute amounts. (Finding C-10). This radiation is primarily low-LET radiation. (Finding C-13).[5/]

Low-LET radiation can affect the cells and tissues of the human body in three important ways. First, an irradiated cell can be transformed into a cancer cell by damage to the cell's DNA molecule. This biological effect is called carcinogenesis; the health effect which may result is cancer. Second, if an embryo or fetus is exposed during gestation, injury can occur in the proliferating and differentiating cells and tissues, leading to abnormal growth. This biological effect is teratogenesis; the health effect which may result is developmental abnormality in the newborn. Finally, injury to a reproductive cell of

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5/ "LET" stands for "linear energy transfer"; it is sparsely ionizing radiation, characteristic of electrons, X-rays and gamma rays (Finding C-13). "Low" doses of sparsely ionizing radiation are defined by the National Council on Radiation Protection and Measurements ("NCRP") as 0-20 rads. (Finding C-11). A "rad" is a unit of absorbed dose of radiation equal to 100 ergs/gram. (Finding C-12). A second unit commonly used in discussing radiation protection is the "rem." A rem is a unit dose equivalent which equals the absorbed dose (in rads) multiplied by a quality factor and by any other necessary modifying factors. (Finding C-12).

the testes or the ovary can cause an alteration of the hereditary genome of the germ cell, leading to a change in the descendants of the exposed person. This biological effect is mutagenesis; the health effect which may result is genetically-related ill health in a descendant. (Finding C-14). Any exposure to radiation, even at low levels of dose, carries some risk of these health effects. As the dose increases above very low levels, the risk of these delayed or late health effects increases in exposed populations. (Finding C-15). Cancer induction is generally considered to be the most important late somatic effect of low-dose, low-LET ionizing radiation. (Finding C-16).

In order to estimate the health effects from a given dose of radiation, it is necessary to use a number known as a risk estimator or a cancer coefficient. This is a projection, for continuous lifetime exposure to 1 rad per year, of the number of excess cancer deaths which will result per person-rad. For convenience, the number can also be expressed for excess cancer deaths per million persons exposed per rad. (See Finding C-19). A similar risk estimator can be generated for genetic disorders. (See Finding C-23).

b. Estimated health effects at Byron

A discussion of the potential health effects to occupationally-exposed persons at Byron is presented by the Staff in its Final Environmental Statement for Byron. (Finding C-22). The risk estimators used by the Staff to estimate health effects were 135 potential deaths from cancer per million person-rems and 258 potential cases of all forms of genetic disorders per million person-rems.

(Finding C-23). These risk estimators are based on the 1972 Report of the National Academy of Sciences Biological Effects of Ionizing Radiaton Committee (or "BEIR I") and are consistent with the values recommended by the major radiation protection organizations. (Finding C-25).[6/] Multiplying the annual plant-worker population dose for Byron (about 440 person-rems per reactor year) by the risk estimators, the Staff estimates that about 0.06 cancer deaths may occur in the total exposed population and about 0.11 genetic disorders may occur in all future generations of the same exposed population. (Finding C-24).

Intervenor League's expert Dr. Morgan criticizes the FES estimates of health effects at Byron on several grounds. He argues that the risk estimator used by the Staff is too low. (See Finding C-28). He asserts that the annual plant-worker population dose estimate for Byron in also too low. (See Findings C-41, C-42). Finally, he believes that an improper dose-response relationship model (the linear model) has been relied upon by the Staff; Dr. Morgan prefers a model (the supralinear model) under which the risk from low-LET radiation increases at low dose levels. (See Finding C-33). We address each of these point below.

(1) Risk estimators

The risk estimator for fatal cancers used by the Staff is approximately 1×10^{-4} excess fatal cancers per person-rem. (Findings

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6/ These organizations include, in addition to BEIR, the NCRP, the International Council on Radiation Protection ("ICRP"), and the United Nations Scientific Committee on the Effects of Atomic Radiation ("UNSCEAR"). All parties agree that these organizations are the most authoritative ones in the area of recommending radiation protection standards. (Finding C-25).

C-23, C-21, C-27). Dr. Morgan believes that 1×10^{-3} excess fatal cancers per person-rem is a more reasonable estimate. (Finding C-28). We cannot agree.

The risk estimators used by the Staff are consistent with the values recommended by the major radiation protection organizations. These organizations represent the views of the overwhelming majority of the scientific community, as Dr. Morgan concedes. (Finding C-25). Edison's very prominent expert, Dr. Fabrikant, testified that no radiation protection standard setting commission or council has endorsed risk estimates of health effects from low-dose, low-LET radiation as great as 1×10^{-3}. (Finding C-31). This includes BEIR I, BEIR III, NCRP, ICRP and UNSCEAR. (Findings C-19, C-21, C-25, C-27, C-31).

Dr. Morgan's estimate of 1×10^{-3} excess fatal cancers per person-rem is based on his analysis of "a scattering of points of human data" from a limited number of published studies. (Finding C-28). It is an opinion that he has come to within the last few years. (Finding C-30). The studies relied upon by Dr. Morgan are generally considered to suffer from methodological flaws or significant uncertainties. (Findings C-34). In arriving at his higher estimate, Dr. Morgan failed to use the sophisticated analytical techniques normally relied upon by experts in the field. (Finding C-29). We conclude that we must reject Dr. Morgan's suggested value for a risk estimator as inadequately supported in this record and contrary to the views of the overwhelming majority of the scientific community.

(2) Collective annual occupational dose for Byron

The Staff has found acceptable Edison's estimate of a collective annual occupational dose per unit for Byron of 400 person-rems.

(Findings C-24, C-39, C-40). Dr. Morgan argues that a much higher figure, between 791 and 3000 person-rems, must be expected. (Findings C-41, C-42). We find that Dr. Morgan's estimates cannot properly be assumed for Byron.

Dr. Morgan states that the average annual collective occupational dose in the United States is 791 person-rems. This is an average for all light water reactors, including boiling water reactors as well as pressurized water reactors, in a single recent year. (Finding C-41). The figure for pressurized water reactors during that same year was only 581 person-rem. (Finding C-41). More importantly, the estimate of 400 person-rem per unit accepted by the Staff was based on dose data from several years of operation at Applicant's Zion Station and at other PWR's. (Finding C-39). Dr. Morgan also states that the annual collective occupational dose could be as high as 3000 person-rem, based on experience at Surry. (Finding C-42). The figure of 3000 person-rem at Surry was experienced in a year in which extraordinary repair work was required in replacing a steam generator. (Finding C-43). It was not an annual average in any sense and Byron cannot be expected to approach such a figure on an annual average basis. The estimated dose for special maintenance at Byron, including steam generator repair, as necessary, is 300 person-rem per year. (Finding C-44).

In sum, Dr. Morgan has not provided a sufficient reason for us to overrule the Staff's judgment and require that a larger annual collective occupational dose figure be used in estimating health effects from occupational exposure at Byron.

(3) Dose-response relationship

The Staff's estimates of health effects at Byron are based on a linear, no-threshold dose-response relationship. (Finding C-37). Dr. Morgan believes that a supralinear model should be used, under which increased health risks are presented at lower doses of radiation. (Finding C-33). The difference is important because the use of temporary workers for special tasks at Byron has the effect of exposing more individuals at lower dose levels. (Finding C-33).

We find that we cannot accept Dr. Morgan's position that a supralinear model should be used for estimating health effects at low doses of low-LET radiation. The best reliable evidence available to the scientific community supports the use of the linear model to define the upper limits of the risk associated with low-LET radiation and suggests that the linear model tends to overstate that risk. (Finding C-35). All reports of expert scientific advisory committees, including BEIR, NCRP, ICRP and the General Accounting Office 1981 Committee ("GAO"), disagree with the claim that the linear dose-response relationship model is not conservative for low-LET radiation. The evidence is compelling that the risk per unit dose at low doses is less than at high doses. (Finding C-36).

No convincing evidence exists to support Dr. Morgan's conclusion that more malignancies will occur if a given number of person-rems is distributed among a greater number of people. (Findings C-36, C-37). The 1981 GAO Report noted that the few epidemiological studies cited by some to support the supralinear model have been severely criticized on statistical and methodological grounds. These studies, and the attendant criticisms, were discussed at length by Dr. Fabrikant's

testimony. (Finding C-34). We find the use of the linear dose-response relationship model for estimating health effects from exposure to radiation at Byron to be reasonable and conservative.

3. Radiation Safety at Byron

Intervenor League questions Edison's ability to maintain radiation doses ALARA at Byron. We discuss below the various issues raised by the League. Four principal issues have been litigated: (1) plant design; (2) plant procedures; (3) in-plant monitoring; and (4) effects of the use of temporary workers at Byron.

a. Design

The contentions assert that Byron's design will not provide adequate protection against radiation health effects and does not meet the ALARA requirement of 10 CFR § 20.1(c). (Finding C-1). Contrary to Intervenor's position, we find that radiation protection measures have been incorporated in the plant design which provide reasonable assurance that occupational doses will be maintained ALARA and below the limits set in 10 CFR Part 20.

The Byron Station has been designed and will be operated in a manner consistent with the specific guidance provided for ALARA compliance in Regulatory Guides 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposure At Nuclear Power Station Will Be As Low As Is Reasonably Achievable," and 8.10, "Operating Philosophy For Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable." (Findings C-45, C-47). Design considerations discussed in this regulatory guidance include: (1) access control of radiation areas; (2) radiation shields and design geometry; (3) process instrumentation

and controls; (4) control of airborne contaminants and gaseous radiation sources; (5) crud control; (6) isolation and decontamination; (7) radiation monitoring systems; (8) resin and sludge treatment systems; and (9) other features such as materials selection, pipe routing, equipment accessibility, and inspection procedures. (Finding C-46).

Radiation protection design reviews have been conducted by Edison, by Westinghouse (the nuclear steam system supplier) and by Sargent & Lundy (the architect-engineer). (Finding C-48). Four basic principles have been utilized in the design of Byron to reduce radiation exposure: (1) provide shielding between the radioactive source and accessed area; (2) provide distance between the radioactive source and workers; (3) reduce the time spent by the worker in the radiation field; and (4) remove the radioactive source material. (Finding C-51). The application of these principles at Byron has resulted in a design which includes specific consideration of radiation protection in the following ways: (1) selection, layout and segregation of equipment; (2) ease of access to cubicles and compartments; (3) equipment draining and flushing capability; (4) floor and sink drainage; (5) equipment venting; (6) routing and shielding of lines and ventilation ducts; (7) use of waste filters and demineralizers; (8) location and shielding of valves and instruments; (9) contamination control and ease of decontamination; (10) access control and traffic pattern design; (11) radiation zone designation (12) health physics laboratory center; (13) station laundry facility and procedures; and (14) shielding design features generally. (Finding C-52).

An example of the application of design features for radiation protection was provided by testimony on the steam generators for Byron. Edison witness Conway discussed several mechanical and metallurgical design features, such as material selection and design for ease of access, which have been incorporated into the steam generators. (Findings C-57 through C-65).

We conclude that Byron Station has been designed in a way that facilitates compliance with the ALARA requirement of the regulations.

b. Plant procedures

Edison has a corporate ALARA program which is designed to minimize both the radiation dose received by individual workers and the sum of doses received by all exposed workers. (Finding C-68). The program includes the establishment of an organization for ALARA compliance at both the corporate level and the individual station level. (Finding C-69). Corporate and station ALARA committees exist which have well-defined responsibilities. (Findings C-71, C-72). Byron Station will have a health physics department which will consist of 43 employees assigned to that department in whole or in part. (Finding C-75). Appropriate training will be provided to each of these employees. (Finding C-75).

Dosimetry recordkeeping is an important part of the ALARA program for Byron and at Edison generally. Adminstrative daily and weekly dose limits are established and procedures are imposed to ensure that these limits are not exceeded. (Finding C-77). Records are maintained which include a worker's prior occupational dose history and biweekly recording of dose data, as well as the results of surveys, monitoring and disposals.

(Findings C-78, C-79). Dosimetry recordkeeping is computerized and the trending of occupational exposure data is monitored. (Finding C-76).

Employee education and training is another important part of the ALARA program at Byron. Employees are instructed in the fundamentals of radiation exposures, including biological effects, in the importance of maintaining doses ALARA, and in their role in meeting that objective. (Findings C-81, C-85). All workers (including temporary workers) who may be exposed to radiation receive at least that level of instruction provided by Edison's Nuclear General Employee Training Program. (Findings C-82, C-83). Additional training is provided based on an individual's experience and needs. (Finding C-82). Special proficiency training is offered to workers who will be performing complex tasks in high radiation areas. (Finding C-82).

Special information is provided to female employees on the potential risks of radiation exposure to a fetus. (Finding C-87). Women are required to sign a statement that they will tell their supervisor when they know or suspect they are pregnant. (Finding C-87). Edison takes specific steps to ensure that radiation exposure to declared pregnant women is limited to a total of 500 millirem over the gestation period. (Findings C-88 through C-91).

The Staff has reviewed the radiation protection and ALARA program for Byron. Special attention has been given in this review to such matters as management policy and organization, personnel qualifications and staffing, radiation control procedures, dosimetry recordkeeping and exposure tracking, and employee training. (Finding C-94). The results of this review were received in evidence. (Findings C-94 through C-98).

The Staff has found the radiation protection program for Byron acceptable. (Finding C-94). We agree that the plant procedures at Byron are adequate to protect workers from radiation injury.

c. In-plant monitoring

A wide variety of devices will be used to monitor radiation levels within Byron Station. These will include personal monitors, neutron dosimeters (film badges, pocket ionization chambers and thermoluminescent dosimeters), calibrated REM-meters, fixed area monitors and portable monitors, whole body count equipment and facilities for bioassay procedures. (Findings C-102 through C-104, C-106, C-107, C-110, C-111). Special sound-emitting personal dosimeters will be used where workers may be exposed to higher radiation levels. (Finding C-109). Procedures are in place for the processing of personal dosimeters and the cross-checking and recording of results. (Finding C-105). Accuracy of the processing is checked by quality assurance measures taken by Edison and by the dosimeter processor. (Finding C-105). In-plant monitoring at Byron includes provision to measure beta, gamma and neutron dose, to monitor noble gases, to measure the isotopic composition of radioisotopes, and to distinguish between airborne gases and particulates. (Findings C-102, C-103, C-104, C-107). Edison's area monitoring instrumentation, airborne radioactivity monitoring instrumentation, radiation survey equipment and personnel monitoring equipment have been reviewed and found acceptable by the Staff. (Finding C-111).

Intervenor's witness Dr. Morgan was not knowledgable about Edison's radiation monitoring within the Byron Station and we accord his general criticisms and recommendations little or no weight. For example,

Dr. Morgan stated that more fixed area monitors were needed but admitted that he did not know how many Byron has (it has more than 200 in-plant area and air monitoring instruments) or where any were located. (Findings C-107, C-108). Similarly, Dr. Morgan criticized the use of a type of dosimeter (using NTA emulsion films) which will not be used at Byron. (Findings C-102).

We conclude that the adequacy of Applicant's in-plant monitoring program and facilities for Byron has been adequately demonstrated on this record.

d. Temporary workers

Intervenor League has two remaining concerns about the use of temporary workers.[7/] First, temporary workers may not receive adequate training about radiation hazards and may receive excessive cumulative doses over time. Second, the use of temporary workers is alleged to create special security risks to the facility. As we discuss below, we are satisfied that appropriate steps have been and will be taken at Byron to see that radiation protection is provided for temporary employees and that the station remains secure.

(1) Radiation protection

Temporary employees, or "contract workers," will be employed at Byron from time to time because of special job needs, special skills or to maintain individual radiation doses of employees below regulatory limits.

7/ A third concern -- that the use of temporary workers would increase the overall cancer risk by spreading the same amount of dose to more people at lower levels -- has been discussed earlier and found to lack merit.

(Finding C-113). This use of contract workers is common practice at nuclear power plants. (Finding C-113).

Various steps are taken by Edison to protect the health of contract workers. They receive the same basic training as Edison employees. (Finding C-114). This training program, particularly as to contract workers, has been found acceptable by the Staff. (Finding C-119). The same monitoring and dose control procedures are used for contract workers as for Edison employees. (Finding C-115, C-118). Contract workers employed by Edison must provide the occupational dose history data needed for NRC Form-4 and Edison's dosimetry recordkeeping requirements are applied to them. (Finding C-116). We conclude that the same procedures and programs that provide radiation protection for Edison's regular employees will also provide adequate protection for contract workers.

(2) Security considerations

Both the Staff and the Applicant presented evidence on whether the use of temporary workers will increase the risk of sabotage at Byron. (Findings C-120).[8/] A portion of the oral testimony on this subject was received during an *in camera* session on May 9, 1983 because it addressed specific details of Edison's Physical Security Plan for Byron. Such details are safeguards information within the meaning of 10 CFR § 73.1(jj)

8/ Intervenor's witness, Dr. Morgan, made only one reference to an increased risk of "espionage" in his testimony. This reference was stricken along with certain other material as not within Dr. Morgan's expertise. *See* Tr. 1511-13. In any event, the reference was so isolated and unsupported as to be worthless. Thus, no evidence from Intervenor League was admitted on this aspect of the contentions.

and must be protected from disclosure under 10 CFR § 73.21. The security issue raised by Intervenor League is a small and tangential part of the ALARA contentions. We find that the testimony by the Staff and the Applicant received in public session is more than adequate to resolve the issue presented. Accordingly, we have not made separate findings based on the in camera evidence.[9/]

The Physical Security Plan for Byron has been reviewed by the Staff. Edison has committed to implement the prescriptive requirements of 10 CFR § 73.55(b) through (h), which include provision for access controls and protection of vital equipement. Access controls include: (1) identification and picture badge system; (2) search of individuals for firearms and explosives; (3) vehicle searches; (4) delivered package and material identification and search; (5) escort of visitors; (6) pre-employment screening for all employees who have unescorted access; and (7) screening for contractor employees who have unescorted access. Steps for protection of vital equipment include: (1) location of such equipment behind second barriers; (2) limitation of access to such equipment only for performance of duties; (3) locking and alarming areas that contain vital equipment; (4) special controls for containment during refueling and maintenance. (Finding C-122).

9/ We have also not reviewed the Physical Security Plan for Byron or received it in evidence. The parties agreed that the issue presented here could be adequately treated without reference to the details of the plan. See Tr. 1375. Both Mr. Ruolo and Mr. Skelton, witnesses for Edison and the Staff, respectively, were familiar with the plan.

Edison's Physical Security Plan for Byron contains a specific chapter which deals with "Security Measures During Maintenance, Refueling and Major Modifications." This chapter contains additional specific commitments to deal with activities which require additional personnel. (Finding C-123). The Staff has determined, based on the various procedures and commitments for Byron, that the security plan for Byron provides the necessary measures to protect against any potential increased risks caused by the presence additional persons such as contract workers. (Finding C-124).

Edison witness Ruolo discussed on the public record certain aspects of Edison's procedures in relation to contract workers (i.e., pre-employment screening, access controls) and concluded that there is "no increased risk due to the employment of transient workers." (Findings C-125 through C-129). The Board finds more compelling the conclusion of Staff witness Skelton that there is a potential increase in risk of sabotage associated with allowing additional personnel onsite but that Edison's security plans are adequate to minimize any such potential increase. (Finding C-129). The most significant point to be made is that a security plan that satisfies the specific requirements of 10 CFR § 73.55(b)-(h), as Edison's Byron plan does according to reliable and uncontroverted testimony, also satisifes the performance objective of providing high assurance that operation of the reactor would not constitute an unreasonable risk to public safety. (Finding C-130).

4. Conclusion on Occupational Radiation Safety

Based on an extensive record and our evaluation of the testimony presented by Intervenor League, Edison and the Staff, we conclude that

League Contentions 42, 111 and 112 lack merit. We find that the estimates provided by the Staff in the FES of health effects from occupational radiation exposure at Byron are reasonable and conservative. The annual collective occupational dose estimate accepted by the Staff is well-based in experience at other plants. The risk estimators and the dose-response relationship model used by the Staff are supported by the great weight of scientific evidence. We are also satisfied that Edison and its contractors have designed, and that Edison will operate, the Byron Station in conformity with the radiation protection requirements of 10 CFR Part 20 and other applicable regulations. We have reasonable assurance that no undue risk from radiation exposure to Edison's employees or to contract workers will be presented by operation of Byron Station; neither will the use of contract workers create undue risk to the public health and safety.

D. Steam Generator Tube Integrity (League Contention 22 and DAARE/SAFE Contention 9(c))

League Contention 22 and DAARE/SAFE Contention 9(c) assert, in material part, that steam generator tube integrity problems have not been adequately dealt with for Byron and that they present a serious safety problem during both normal operation and under accident conditions. Westinghouse steam generators have experienced several forms of tube degradation due to corrosion, including thinning, pitting, intergranular attack, and stress corrosion cracking. Steam generator tube wear has also been experienced due to flow-induced vibration and impact damage as a result of foreign objects or loose parts. (Finding D-2).

Measures which have been taken to minimize the potential for corrosion-related degradation include improved design features in the steam generators and balance of the secondary cycle, use of all volatile treatment (AVT) secondary water chemistry, and an improved program to monitor and control secondary water chemistry. (Findings D-41 to D-46, D-65 to D-83).

The Byron facility utilizes Westinghouse model D4 and D5 steam generators. The design of the Model D steam generators represents an improvement over early designs from the standpoint of diminishing the potential for tube corrosion. Model D5 represents a design evolution from the Model D4 and incorporates several design improvements to further reduce corrosion. (Findings D-41 to D-46).

Intergranular corrosion occurs either in the form of stress corrosion cracking or intergranular attack. Stress corrosion cracking and intergranular attack can occur on the outside surface of the steam generator tubes, but these phenomena can be controlled by not allowing corrodants to accumulate in steam generators by applying vigorous water chemistry controls and/or several cleaning techniques such as sludge lantzing and hot and cold water soaks. The occurrence of stress corrosion cracking on the inside surface of the steam generator tubes is controlled by limiting cold work techniques during tube fabrication or by reducing residual stresses by thermal treatment. (Findings D-19 to D-23).

Steam generator tube thinning is controlled by the use of AVT water chemistry control. Pitting of the steam generator tubes has been observed in only a few plants and the phenomenon can be controlled

by close adherence to water chemistry controls. Denting of steam generator tubes results from the corrosion of the carbon steel support plates around the tubes and can be controlled by close adherence to water chemistry controls. The design modifications and chemistry control requirements are expected to reduce the potential for the types of corrosion problems which have been experienced to date. (Findings D-24 to D-40).

Some degree of degradation is likely to occur at the Byron units during their lifetime. Given the potential for degradation, surveillance requirements are essential to insure that adequate tube integrity is maintained against rupture and excessive leakage during the full range of normal operating and postulated accident conditions. (Finding D-11).

The Byron steam generator tubes will be subject to periodic in-service inspections in accordance with Regulatory Guide 1.83 and NUREG-0452. (Findings D-51 to D-62).

Operational limits on allowable primary to secondary leakage will provide added assurance of adequate tube integrity. Sampling requirements for in-service inspections are being reevaluated as part of an integrated Staff program relating to Unresolved Safety Issue A-3 concerning Westinghouse steam generator tube integrity. (Findings D-63 to D-64, D-230).

Past experience has demonstrated that the current regulatory requirements are successful in maintaining acceptable structural margins against tube ruptures. Where new or unanticipated degradation problems have occurred, these problems have been revealed

either during routine in-service inspection or as a result of leaks and appropriate action has been taken at that time. In some cases, the action has included additional inspection requirements or operational limitations imposed by the Staff. In addition, the plant operators have been able in many instances to take remedial measures to reduce the potential for further problems. Leaks, when they have occurred, have generally been small. (Findings D-5 to D-18).

Although there have been four instances of tube rupture (ranging from about 80 to 700 gallons per minute) the single tube ruptures have been within the design bases for the plants and there were no significant off-site releases. Two rupture occurrences (Point Beach in 1975 and Surry in 1976) have been corrosion related. The conditions which led to these ruptures are not expected to occur at Byron. The rupture at Point Beach was caused by secondary side intergranular stress corrosion cracking which occurred as a consequence of reactions between condenser inleakage impurities and residual phosphates. Byron will use AVT. Consequently, chemical reactions which caused the Point Beach rupture cannot occur at Byron. Since the industry conversion to AVT in 1974, no plant which has started up on AVT has detected secondary side initiated stress corrosion cracking. (Findings D-183 to D-194).

The rupture at Surry was initiated from the primary side of the tube and caused by excessive tube stresses. Excessive tube stress resulted from extensive tube denting which first froze the tube in place and then physically moved the tube support plates, resulting in the significant deformation of the tube and resultant high stress. The water chemistry control requirements at Byron in conjunction with in-service

inspection requirements will combine to make it highly unlikely that extensive denting will occur. In this regard, denting is not regarded as a health and safety problem. It may be regarded as an operational problem should denting necessitate tube plugging. (Findings D-187 to D-188).

The two other rupture events (Prairie Island in 1979 and R. Ginna in 1982) are attributable to damage caused by foreign objects and loose parts. The Applicant has installed a loose parts monitoring system on the secondary side of each steam generator. This system is expected to reduce the potential for similar tube ruptures. (Findings B-105 to B-115).

In addition to the potential for damaging tube wear as a result of foreign objects and loose parts, lead operating pressurized water reactors which employ Model D steam generators indicate that the tubes in the pre-heater region may be subject to excessive wear due to flow-induced tube vibrations when these facilities are operated at power levels in excess of 70% full power (Model D4-D5). In response to this phenomenon, Westinghouse initiated a generic Model D4-D5 corrective program which included extensive analyses and tests, including large scale model tests and in-situ tests on the foreign operating facility Model D4 steam generators, to evaluate the causes and the effectiveness of various proposed corrective modifications. As a result of this program, Westinghouse has devised a modification entailing the diametrical expansion of approximately 100 tubes at the support plate area in the pre-heater section and the bypassing of 10% of the main feedwater flow to the auxillary feedwater nozzle. Implementation of this modification at Byron is expected to reduce tube degradation associated with flow-induced

vibration to an acceptable level. The Applicant has committed to implement this modification prior to fuel loading. The Staff will review and approve the proposed modification prior to authorizing issuance of a Byron operating license. (Findings D-117 to D-182).

The systems performance and radiological consequences of multiple steam generator tube ruptures and single steam generator tube ruptures coincident with other design bases accidents, such as main steam line break and loss of coolant accidents, have been analyzed from both a qualitative and quantitative standpoint. By both standards, the probability for such events is exceedingly small and the additive effect of the steam generator tube rupture to the denominated design basis accidents is insignificant from a health and safety standpoint. The NRC effort to conduct such analyses has been directed at the enhancement of operator training and emergency operating procedures in the context of post-TMI action plan requirements. This combination of accidents need not be evaluated as part of the design basis submissions and review. (Findings D-195 to D-227).

The NRC Staff regards the Westinghouse steam generator tube integrity unresolved safety issue A-3 as effectively resolved through a combination of remedial measures, such as those identified above, operational precautions, such as routine in-service inspection and surveillance, and preventative tube plugging. This is borne out by the discernible downward trend in the amount of tube plugging necessitated at domestic nuclear plants over the past several years since the inception of a number of the described precautions. (Findings D-228 to D-237).

The Staff anticipates formal documentation of the resolution report of USI-A-3 in approximately July 1983. The Board concludes that historical steam generator tube degradation concerns due to a variety of corrosion and vibration induced phenomena do not pose a safety problem for the Byron steam generators under anticipated operational and postulated accident conditions. The extremely low probability of a steam generator tube rupture combined with other accidents, such as a LOCA or main steam line break, does not justify their explicit design basis consideration though they should serve to influence the range of emergency operating procedures under development and Staff review.

III. FINDINGS

C. Occupational Radiation Safety (League Contentions 42, 111 and 112)

1. Matter in Controversy

C-1. Three separate but related contentions on the subject of Applicant's compliance with regulatory requirements pertaining to occupational radiation exposure were admitted for litigation. League Contention 42 states as follows:

> As the Staff has recognized in NUREG-0410 and in the Black Fox testimony previously cited, occupational radiation exposure to station and contractor personnel has generally been increasing in recent years, and violation of the limits of 10 CFR Part 20 has been avoided by C.E., as by other licensees, by obtaining the temporary services of transient workmen rather than by devoting adequate effort to reducing exposures. Among other things, this practice results in using larger numbers of people and thereby increasing the risk of sabotage, operator error and similar safety-related hazards. Furthermore, new information on low-level radiation effects indicates that the Byron design basis will not provide safe operation. Accordingly, both because of the lack of assurance that proper exposure levels will be maintained and because of the practice of using transient workers, as a result

of this serious and unresolved problem the findings required by 10 CFR § 50.57(a)(3)(8) and 50.57(a)(b) cannot be made.

League Contention 111 focused on radiation monitoring:

C.E. has not met the requirements of NEPA and the Regs, including but not limited to 10 C.F.R. §§ 50.34(a) and 50.36(a) because C.E. has not adequately monitored and provided a design base for the Byron plant which will keep radiation levels as low as achievable as required for operation of the plant to protect the health and safety of the public. To keep radiation levels as low as achievable, C.E. should provide and utilize:

A. More adequate environmental and discharge monitoring of radioactive emissions from the Byron plant, which include:

(1) Monitoring devices at more locations within and without the plant site.

(2) Provisions for more frequent reading of monitors by independent analysts.

(3) Better monitoring devices which include:

(a) An automatic system of monitoring that notifies local authorities by an alarm when discharge emission exceed design limits;

(b) Monitoring devices that measure differences in alpha, beta and gamma dose levels, which presently are not proposed to be considered and measured;

(c) Monitoring and recording of emissions of all dangerous long lived radionuclides, including especially I-129 and Plutonium;

(d) Bioaccumulative testing in a tiered system to assess the uptake of radioactive and chemical pollutants from bottom sediments or soil to lower organisms and to contamination of the food chain of man and other life.

B. More accurate calculation of design doses which can be accomplished by utilizing information from the improved monitoring suggested above and also by:

(1) Providing for and constant update and replacement of equipment and analysis to respond to new experimental and analytical results. Byron was

licensed for construction, for example, when some (including C.E.) asserted improperly that there was a threshold to radiation effects;

(2) Including in calculation of doses the large transient populations in the low population zones around the plant, including school children when present in schools and others participating in recreational facilities;

(3) Including internal radiation doses caused by inhaled and/or ingested radionuclides which are deposited in different parts of the body where they give repeated radiation or until they are eliminated from the body;

(4) Including in calculation of radiation doses, cumulative doses to the general population outside the site boundary caused by overlapping circles of radiation from any nuclear facility (whether on or off the site), including Zion, Dresden, LaSalle, Quad Cities, and Braidwood Stations, as well as any new proposed facility and disposal facilities such as the Morris Waste Disposal Site; and

(5) Including in the calculation, calculation of doses to people by utilizing actual radionuclides for and in food, animals, plants, soil, water, and in other parts of the environment in and around the Byron site.

As a result, the applicable findings required by the Act, NEPA, and the Regs, cannot be made therein.

By stipulation among the parties dated December 6, 1982, Contention 111 was limited to in-plant radiation monitoring. Finally, Contention 112 states as follows:

C.E. has not met the requirements of NEPA and 10 CFR Part 20 because it has not adequately assessed the effect of radiation on plant workers and provided a design base for the Byron plant which will provide radiation levels as low as achievable. To keep radiation levels as low as achievable there is a need for better use of preventive measures to reduce radiation, including neutron, exposure levels to regular plant personnel and transient workers. These include but are not limited to:

(a) Plant designs for reducing amount of radiation exposure which take into account new evidence on low levels of radiation which were not considered in design of the plant.

(b) Improved record keeping of radiation exposures, including cumulative exposures both at the plant site and at other facilities.

(c) Better training of personnel to prevent radiation exposures, including more use of regular trained personnel rather than transient or temporary workers with little experience and training.

(d) Limiting exposure to high levels of radiation to volunteers and/or only older workers beyond the child bearing age or others incapable of biological reproduction.

(e) Better education about radiation dangers to ensure cooperation of workers in keeping radiation exposures to a minimum.

As a result, the applicable findings required by the Act, NEPA, and the Regs, cannot be made herein.

C-2. Hearings on Contentions 42, 111 and 112 were held on March 8-11, 1983. A total of ten witnesses gave testimony during that period.

C-3. Edison presented six separate witnesses on these contentions. Dr. Jacob I. Fabrikant, a physician, scientist and University of California professor in radiology and biophysics, addressed the subject of health effects from low level radiation. (Fabrikant testimony, ff. Tr. 1399). Frank Rescek, Lead Health Physics-Technical Services Engineer for Edison, addressed Edison's corporate ALARA program as well as its dosimetry program, recordkeeping procedures and training program. (Rescek testimony, ff. Tr. 1157). James R. Van Laere, Edison's Byron Station Radiation Chemistry Supervisor and Radiation Protection Manager, discussed the ALARA program, the health physics department staff and the in-plant monitoring program for Byron specifically. (Van Laere testimony, ff. Tr. 1707). Gerald P. Lahti, Assistant Division Head of the Nuclear Safeguards

and Licensing Division in charge of Shielding and Radiological Safety at Sargent & Lundy, testified about Byron plant design for reducing occupational exposure. (Lahti testimony, ff. Tr. 1830). Dr. Lawrence Conway, an Advisory Engineer in the Steam Turbine Generator Division of Westinghouse, discussed mechanical and metallurgical design features of Byron's steam generators to reduce occupational radiation exposure. (Conway testimony, ff. Tr. 1309). Finally, Jerome L. Ruolo, Edison's Deputy Nuclear Security Administrator, addressed that portion of the contentions on possible increased risk of sabotage from the use of temporary workers at Byron. (Ruolo testimony, ff. Tr. 1356).

C-4. Intervenor League presented only one witness on these contentions -- Dr. K. Z. Morgan. Dr. Morgan is a consultant on radiation safety matters and an Adjunct Professor at Appalachian State University; Dr. Morgan had previously served as Director of the Health Physics Division at Oak Ridge National Laboratory for 1943 to 1972 and as a Professor at Georgia Institute of Technology from 1972 to 1982. Dr. Morgan's testimony addressed health effects from low level radiation and the issue of radiation safety at Byron. (Morgan testimony, ff. Tr. 1515).

C-5. Three members of the Staff appeared and gave testimony. Michael A. Lamastra is a Health Physicist in the Radiation Protection Section of the Radiation Assessment Branch, Office of Nuclear Reactor Regulation. Dr. Edward F. Branagan, Jr., is a Health Physicist in the Radiological Impact Section of the same branch. Both Mr. Lamastra and Dr. Branagan discussed the Staff's review of Edison's radiation protection programs and the bases for the Staff's conclusions from its review; Dr. Branagan also addressed the Staff's estimates of health effects due to

occupational radiation exposure. Relevant sections of the Safety Evaluation Report and the Final Environmental Statement were received in evidence as part of the Staff's testimony. Robert F. Skelton, a Plant Protection Analyst in the Power Reactor Safeguards Licensing Branch, Division of Safeguards, Office of Nuclear Material Safety and Safeguards, testified on the security aspects of the use of temporary workers at Byron. (Lamastra et al. testimony, ff. Tr. 1883).

2. Regulatory Background

C-6. The principal regulatory requirements concerning occupational radiation protection appear in 10 CFR Part 20. Specific radiation dose standards for individuals in restricted areas are provided in 10 CFR § 20.101. In addition, 10 CFR § 20.1(c) states that licensees should make every reasonable effort to maintain exposures to radiation as far below the limits specified in Part 20 as is reasonably achievable. The term "as low as is reasonably achievable" (or "ALARA") means "as low as is reasonably achievable taking into account the state of technology, and the economics of improvement in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to the utilization of atomic energy in the public interest." (10 CFR Part 20).

C-7. The use of the word "should" in 10 CFR § 20.1(c) is interpreted as "shall" (or as a regulatory requirement) by the Staff. (Tr. 1910 (Lamastra)).

C-8. A license would not be issued in the absence of an adequate ALARA program. (Tr. 1909, 1914 (Lamastra)).

C-9. Commitments with respect to ALARA compliance are part of the technical specifications. (Tr. 1908 (Lamastra)).

3. Substantive Findings

a. Health effects from radiation exposure

C-10. Under normal operating conditions, the radioactive isotopes present in a nuclear power plant emit radiation. This radiation is primarily gamma radiation, but beta, alpha and neutron radiation also exist in very minute amounts. (Fabrikant testimony, ff. Tr. 1399, at 8).

C-11. The 1980 Report of the National Council on Radiation Protection and Measurements ("NCRP"), an expert scientific advisory committee on radiation and health effects, categorizes radiation doses as low, intermediate, high and ultra-high. Low doses of sparsely ionizing radiation are arbitrarily defined as 0-20 rads. High doses exist at 150-350 rads. Intermediate doses are defined as anything between the two. Ultra-high doses are anything greater than 350 rads. (Fabrikant testimony, ff. Tr. 1399, at 4, 8).

C-12. A rad is a unit of absorbed dose of radiation equal to 100 ergs/gram. In discussing radiation protection, the unit commonly used is the rem. A rem is a unit of dose equivalent which equals the absorbed dose (in rads) multiplied by a quality factor, by a distribution factor and by any other necessary modifying factors. The rem represents a quantity of radiation that is equivalent, in terms of biologic damage of a specified sort, to one rad of 250-kVp X-rays. (Fabrikant testimony, ff. Tr. 1399, at 13).

C-13. A distinction is drawn between low-LET radiation and high-LET radiation. LET stands for linear energy transfer; it is defined as the average amount of energy lost per unit of ionizing particle spur-track length. Low-LET radiation is sparsely ionizing radiation and is

characteristic of electrons, x-rays and gamma rays. Low-LET radiations are those encountered primarily in the routine operation of nuclear power plants, including Byron. (Fabrikant testimony, ff. Tr. 1399, at 8; Tr. 1443 (Fabrikant); Lamastra et al. testimony, ff. Tr. 1883, at 8).

C-14. Low-LET radiation can affect the cells and tissues of the human body in three important ways. First, an irradiated cell can be transformed into a cancer cell by damage to the cell's DNA molecule. This biological effect is called carcinogenesis; the health effect which may result is cancer. Second, if an embryo or fetus is exposed during gestation, injury can occur in the proliferating and differentiating cells and tissues, leading to abnormal growth. This biological effect is teratogenesis; the health effect which may result is developmental abnormality in the newborn. Finally, injury to a reproductive cell of the testes or the ovary can cause an alteration of the hereditary genome of the germ cell, leading to a change in the descendants of the exposed person. This biological effect is mutagenesis; the health effect which may result is genetically-related ill health in a descendant. (Fabrikant testimony, ff. Tr. 1399, at 9-10).

C-15. Any exposure to radiation, even at low levels of dose, carries some risk of these health effects. As the dose of radiation increases above very low levels, the risk of these delayed or late health effects increases in exposed populations. (Fabrikant testimony, ff. Tr. 1399, at 10).

C-16. Cancer induction is generally considered to be the most important late somatic effect of low-dose, low-LET ionizing radiation. Different tissues in the body appear to vary greatly in their relative

susceptibility to cancer induction by radiation. Teratogenic effects are related to the gestational state at which exposure occurs. It appears that a threshold level of radiation dose and dose rate may exist below which gross teratogenic effects will not be observed. Estimation of radiation risks of genetically related ill-health are based mainly on laboratory animal observations. Genetic effects due to ionizing radiation have never been directly observed in human beings. Because scientific knowledge of the fundamental mechanisms of radiation injury at the genetic level is far more complete than of the mechanisms of radiation carcinogenesis, greater assurance is provided in extrapolating information on genetic mutagenesis from laboratory animals to man. (Fabrikant testimony, ff. Tr. 1399, at 11-12).

C-17. Scientific knowledge of the health effects of radiation are based on epidemiological surveys of exposed human populations, on research or laboratory animals, on analysis of dose-response relationships for carcinogenesis, teratogensis and mutagenesis, and on known mechanisms of cell and tissue injury _in vivo_ and _in vitro_. (Fabrikant testimony, ff. Tr. 1399, at 10, 11).

C-18. Epidemiological surveys of exposed human populations are highly uncertain in regard to the forms of the dose-response relationships for radiation-induced cancer in man, particularly for low-level radiation. While radiation-induced cancer in man has been observed at levels below 50 rads, the epidemiological surveys are too uncertain to provide reliable dose-response data. For this reason, it has been necessary to estimate human cancer risk from low radiation doses from observations

of relatively high doses, frequently higher than 100 rads. (Fabrikant testimony, ff. Tr. 1399, at 12).

C-19. The 1980 NAS-BEIR report gives a relative risk projection, for continuous lifetime exposure to 1 rad/year, of 169 excess cancer deaths per million persons exposed per rad. The absolute-risk projection is 67 deaths. (Fabrikant testimony, ff. Tr. 1399, at 14).

C-20. The 1980 BEIR estimates are based on a constrained linear-quadratic rather than a linear, dose-response model for low-dose, low-LET whole-body radiation. (Fabrikant testimony, ff. Tr. 1399, at 14).

C-21. The 1980 BEIR report estimates do not differ appreciably from those in the 1977 UNSCEAR report, which indicated that the lifetime excess cancer risk per rem is generally one in 10,000 (or 1×10^{-4}) and may be considerably less. (The figures for the 1980 BEIR report are not directly comparable because the 1980 BEIR report tables avoided the expression of excess cancer risk per rad.) (Fabrikant testimony, ff. Tr. 1399, at 15).

C-22. A discussion of the potential health effects to occupationally-exposed persons at Byron is presented in the Byron Final Environmental Statement ("FES") at 5.9.3.1. (Staff Ex. 2 at 5-20 to 5-28). That section of the FES was adopted by Staff witness Branagan as part of his testimony. (Lamastra et al. testimony, ff. Tr. 1883, at 4).

C-23. The risk estimators used by the Staff to estimate health effects were 135 potential deaths from cancer per million person-rems and 258 potential cases of all forms of genetic disorders per million person rems. These estimates are based on the NAS-BEIR Committee Report of 1972 (BEIR I). (Lamastra et al. testimony, ff. Tr. 1883, at 5, 6). Use

of the risk estimators from BEIR III would make no significant difference. (Tr. 1890 (Branagan)).

C-24. Multiplying the annual plant-worker population dose (about 440 person-rems per reactor year) by the risk estimators, the Staff estimates that about 0.06 cancer deaths may occur in the total exposed population and about 0.11 genetic disorders may occur in all future generations of the same exposed population. (Lamastra et al. testimony, ff. Tr. 1883, at 5).

C-25. The risk estimators used by the Staff in the FES are consistent with the values recommended bv the major radiation protection organizations. Dr. Branagan presented a table which he had prepared comparing values from the BEIR-I report, the BEIR-III report, the ICRP the NCRP and UNSCEAR. These organizations represent the views of the overwhelming majority of the scientific community. Dr. Morgan agreed that these committees are the most authoritative ones in the area of recommending radiation protection standards. (Tr. 1518-19 (Morgan)). The risk estimators used in the FES are consistent with the values from these sources. (Lamastra et al. testimony , ff. Tr. 1883, at 7 and Attachment F).

C-26. Contrary to Dr. Morgan's assertion (Morgan testimony, ff. Tr. 1515, at 4, 7), the scientific community does not now believe that the risks of low-dose, low-LET radiation exposure are much greater than was thought a few years ago. The current estimate of the risks associate with exposure to low-dose, low-LET radiation was adopted by the BEIR Committee in 1972. The present protection guides for maximum dose limitation were set by the ICRP almost two decades ago. (Fabrikant testimony, ff. Tr. 1399, at 16).

C-27. Clearly, some potential risk of carcinogenesis or genetic effects of low-dose, low-LET radiation exposure exists. The several expert scientific advisory committees which have examined this risk have concluded that the risk is no greater than a lifetime risk of one in ten thousand (1×10^{-4}) per average rem to large populations and could be much less. This is a conservative estimate, and the existing epidemiological data will not support any greater risk estimate. (Fabrikant testimony, ff. Tr. 1399, at 61-62).

C-28. Intervenor's expert Dr. Morgan argued for a risk estimator for fatal cancers of 1×10^{-3}. (Morgan testimony, ff. Tr. 1515, at 8; Tr. 1657 (Morgan)). This estimator was selected by Dr. Morgan from "a scattering of points of human data." (Tr. 1545-46 (Morgan)).

C-29. Although it is customary to use a computer to analyze data in order to develop a dose-response curve, Dr. Morgan did not use a computer to develop his risk estimator of 1×10^{-3}. (Tr. 1587 (Morgan)).

C-30. Dr. Morgan's judgment concerning a risk estimator of 1×10^{-3} has been developed within the last few years. As recently as 1979, Dr. Morgan wrote that the proper fatal cancer coefficient was between 3 and 6×10^{-4}. (Tr. 1656-57 (Morgan)).

C-31. Best current estimates of the health effects of low level radiation do not indicate that the risk is as high as one lethal cancer per 1000 person-rems (or 1×10^{-3}). The Staff believes that a reasonable upper limit to the range of uncertainty in the cancer mortality risk estimator is about 0.5 potential fatal cancers per 1000 person-rems. Lamastra et al.testimony, ff. Tr. 1883, at 9). Dr. Fabrikant testified

that no radiation protection standard setting commission or council has endorsed risk estimates of health effects from low-dose, low-LET radiation as great as 1×10^{-3}. (Fabrikant testimony , ff. Tr. 1399, at 16).

C-32. The 1980 BEIR Committee calculated excess cancer risk estimates using four dose-response relationship models: linear; linear-quadratic; modified linear-quadratic; and pure quadratic. The linear relationship presents the highest risk estimates. (Fabrikant testimony, ff. Tr. 1399, at 18). Although the weight of the experimental evidence generally favors the linear-quadratic model for low-LET radiation, the human data provide fairly strong support for the linear model. The linear model has additional advantages in its conservatism, simplicity and ease of application, and greater flexibility. (Fabrikant, ff. Tr. 1399, at 19-20).

C-33. Dr. Morgan testified that the supralinear dose-response relationship model should be used in conjunction with the linear model. (Morgan testimony, ff. Tr. at 12-13). Under the supralinear model, more cancers are produced per rem at low doses than at high doses. (Id. at 4, 5, 12). Dr. Morgan concludes from this that maintaining individual doses to workers below regulatory limits as collective dose increases results in more cancers and genetic defects than those would be if the same dose were shared by fewer workers. (Id. at 12).

C-34. The weight of the scientific evidence does not support Dr. Morgan's argument that the supralinear hypothesis fits the data for cancer induction better than the linear hypothesis. The 1981 GAO Report noted that the few epidemiological studies cited by some to support the supralinear model have been seriously criticized on statistical and

methodological grounds. These studies, and the attendant criticisms, were discussed at length by Dr. Fabrikant's testimony. (Fabrikant testimony, ff. Tr. 1399, at 20-21, 27-44).

C-35. The best reliable evidence available to the scientific community strongly suggests that the linear model can properly be used to define the upper limits of the risk associated with exposure to low-LET radiation and probably tends to overestimate these risks. (Fabrikant testimony, ff. Tr. 1399, at 21, 61; Lamastra et al. testimony, ff. Tr. 1883, at 8).

C-36. No convincing scientific evidence exists to support Dr. Morgan's conclusion that more malignancies will occur if a given member of person-rems are distributed among a greater number of people. (Fabrikant testimony, ff. Tr. 1399, at 21). All reports of expert scientific advisory committees, including the NCRP, ICRP, BEIR and GAO (1981) committees, disagree with claims that the linear dose-response relationship model is not conservative for low-LET radiation. (Fabrikant testimony, ff. Tr. 1399, at 24). The evidence is compelling that for exposure to low-LET radiation, the risk per unit dose at low doses generally appears to be less than at high dose (Fabrikant testimony, ff. Tr. 1399, at 46).

C-37. Available evidence does not indicate that the use of temporary workers for compliance with NRC regulations increases the risk of health effects by spreading a given quantity of dose over a large number of workers. Based on the use of the linear, non-threshold model, the spreading of a given quantity of dose over a larger number of workers would not increase the overall risk of health effects. (Lamastra et al. testimony, ff. Tr. 1883, at 8).

C-38. For several reasons, any health effects from occupational exposure at Byron are unlikely ever to be observed. First, the actual risk is likely to be considerably less than predicted by the linear model. Second, the estimation of risk based on the linear model does not take into account known biological processes of cell and tissue repair and recovery following exposure to low levels of low-LET radiation delivered at low dose-rates. Third, any potential incremental risk of cancer-induction or genetic ill-health due to exposure to low-LET radiation at dose-rates between .5 and 5 rems per year would be extremely small when superimposed on the present risks to the population of spontaneous cancer induction or genetic ill-health due to natural causes and in the absence of any excess radiation. Thus, it would likely not be possible to actually detect any excess incidences of cancer or genetic ill-health, if these do occur, caused by exposures in the range described, because the radiation doses are so small and the probability of occurrence is infrequent. (Fabrikant testimony, ff. Tr. 1399, at 63-64).

C-39. Edison's estimate of an average annual collective dose figure for Byron is based on dose data from several years of operation at Edison's Zion Station and at other PWR's. (Staff Ex. 2 at 12-9; Lamastra et al. testimony, ff. Tr. 1883, at 3, 5).

C-40. Applicant's estimate of 400 rem was reviewed by the Staff and accepted as a reasonable value for estimation purposes. The actual number will vary depending on a number of factors. (Tr. 1891-92 (Lamastra)).

C-41. Dr. Morgan's testimony gave a figure of 791 person rems per reactor year for an average collective dose. (Morgan testimony, ff. Tr. 1515, at 6, 10). That figure is an average for all light water reactors in the U.S. (Tr. 1652 (Morgan)). According to NUREG-0713, Vol. 3, Table 2, the correct figure for PWR's is 581 person rems per reactor year. Byron is a PWR. (Tr. 1653-54 (Morgan)).

C-42. Dr. Morgan testified that he has no assurance that the average annual collective dose for Byron will not be as high as 3000 person rems per year, based on experience at Surry. (Morgan testimony, ff. Tr. 1515, at 9).

C-43. The single year collective dose of 3000 person rem at Surry was the result of steam generator replacement, which does not occur every year. (Tr. 1622-24 (Morgan)).

C-44. The estimated dose for special maintenance at Byron, including steam generator repair, as necessary, is 300 person rem per year. (Lamastra et al. testimony, ff. Tr. 1883. at 10).

b. Radiation safety program at Byron Station

(1) Design considerations for radiation safety

C-45. Edison has provided a commitment in the FSAR to ensure that Byron will be designed and operated in a manner consistent with the guidance provided by the Staff in Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposure at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable," and Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures

as Low As Is Reasonably Achievable." (Lamastra et al. testimony, ff. Tr. 1883, at 4).

C-46. Regulatory Guide 8.8 notes that "[e]ffective design of facilities and selection of equipment for systems that contain, collect, store, process, or transport radioactive material in any form will contribute to the efforts to maintain radiation doses to station personnel ALARA." (Lamastra et al.testimony, ff. Tr. 1883, Attachment C at 8.8-2). This regulatory guide provides specific guidance for ALARA compliance in the design features of the facility and its equipment. These ALARA program objectives and guidance toward meeting them include the following subjects: (1) access control of radiation areas; (2) radiation shields and design geometry; (3) process instrumentation and controls; (4) control of airborne contaminants and gaseous radiation sources; (5) crud control; (6) isolation and decontamination; (7) radiation monitoring systems; (8) resin and sludge treatment systems; and (9) other features such as materials selection, pipe routing, equipment accessibility, and inspection procedures. (Lamastra et al. testimony, ff. Tr. 1883, Attachment C at 8.8-6 to 8.8-12).

C-47. The Byron plant has been designed using the ALARA policy and the Applicant and its architect-engineer have continued to review, update and modify the plant design during construction using ALARA guidelines. (Lamastra et al. testimony, ff. Tr. 1883, at 4).

C-48. Edison utilized its nuclear steam system supplier, Westinghouse, and its architect-engineer, Sargent & Lundy, to review the Byron radiation protection design. At Westinghouse, systems analysis engineers who are competent in the area of health physics and radiation protection work

with the system design engineers. Westinghouse policy, design and operational considerations relating to ALARA compliance have been reviewed and accepted by the Staff. Radiation protection design reviews are also conducted by Sargent & Lundy utilizing Regulatory Guide 8.8. Ultimately, however, the responsibility for the radiation protection design review is Edison's. (Lamastra et al. testimony, ff. Tr. 1883, at 4, 15 and Attachment E at Q331.3-1 to Q331.3-2).

C-49. Edison witness Lahti was the individual at Sargent & Lundy responsible for supervising the activities of engineers assigned to the Shielding and Radiological Safety Section to ensure that the design of Byron is such that operation and expected maintenance do not result in radiation exposure in excess of the levels specified in 10 CFR § 20.101 or the ALARA requirement of 10 CFR § 20.1. (Lahti testimony, ff. Tr. 1830, at 2-3).

C-50. Sargent & Lundy takes certain principles for ALARA compliance into account as it begins the design of a project. Independent ALARA reviews are conducted at various stages in the plant design. One such review was just recently completed. (Tr. 1871 (Lahti)). Feedback from experience at other stations is taken into account. (Tr. 1871-72 (Lehti)).

C-51. Certain basic design principles have been utilized in the design of Byron to reduce radiation exposure. The four means of reducing radiation exposure are: (1) provide shielding between the radioactive source and accessed area; (2) provide distance between the radioactive source and workers; (3) reduce the time spent by the worker in the radiation field; and (4) remove the radioactive source material. (Lahti testimony, ff. Tr. 1830, at 3-5). Generally, it is a combination of

these principles in design which contributes to maintaining radiation exposures ALARA. (Id. at 5).

C-52. Sections 12.3 through 12.3.2.1.8 of the FSAR for Byron describe the detailed application of these design principles at Byron. (Lahti testimony, ff. Tr. 1830, at 5). The facility design considerations for radiation protection discussed in these sections of the FSAR include the following: (1) selection, layout and segregation of equipment; (2) ease of access to cubicles and compartments; (3) equipment draining and flushing capability; (4) floor and sink drainage; (5) equipment venting; (6) routing and shielding of lines and ventilation ducts; (7) use of waste filters and demineralizers; (8) location and shielding of valves and instruments; (9) contamination control and ease of decontamination; (10) access control and traffic pattern design; (11) radiation zone designation (12) health physics laboratory center; (13) station laundry facility and procedures; and (14) shielding design features generally. (Lahti testimony, ff. Tr. 1830, Exhibit 1).

C-53. The design of the Byron plant provides protection against alpha, beta, gamma and neutron radiation as necessary. (Tr. 1831-34 (Lahti)).

C-54. Alpha radiation is released by the decay of transuranic elements in high burn-up fuel. This radiation is contained within the fuel element and cladding material. A surveillance program assures that there is no alpha activity in the coolant that has come into contact with the fuel cladding. (Tr. 1831 (Lahti)).

C-55. Features have been designed into the plant to facilitate decontamination. (Tr. 1853 (Lahti)).

C-56. By technical specification requirement, bulk shielding will be surveyed for radiation leakage at several levels of power as part of the low power testing program. (Tr. 1870 (Lahti)).

C-57. Edison witness Conway discussed the specific incorporation of radiation protection features in the Byron steam generators. Several mechanical and metallurgical design features have been incorporated in the Byron steam generators to minimize personnel radiation exposure. (Conway testimony, ff. Tr. 1309, at 5).

C-58. First, two specific design features have been included the Byron steam generators to reduce occupational radiation exposure by eliminating or reducing areas where radioactive crud sources could accumulate. These involve the elimination of protrusions, crevices, ledges or baffles from the tube to tubesheet seal welds and the primary channel head interior. (Conway testimony, ff. Tr. 1309, at 5-6).

C-59. Second, materials have been selected and controlled to limit the activation of stable isotopes into radioactive ones. For example, the amount of cobalt-59 contained in the stainless steel and inconel used in the steam generators has been limited. (Conway testimony, ff. Tr. 1309, at 6-7).

C-60. Third, the primary channel head external drain has been designed to facilitate the removal of radioactive water from the steam generator and eliminate the need for manual mopping-up. (Conway testimony, ff. Tr. 1309, at 7-8).

C-61. Fourth, primary nozzle closure rings have been provided which facilitate the seating of the nozzle maintenance covers. This shields

workers from radiation streaming from the reactor coolant piping. (Conway testimony, ff. Tr. 1309, at 8).

C-62. Fifth, manway access openings have been designed and located to provide easy access to steam generator internals and to facilitate disassembly and reassembly operations for quick access to minimize worker exposure time. (Conway testimony, ff. Tr. 1309, at 9-10).

C-63. Finally, the minimum necessary number of secondary side instrument and access openings have been provided for necessary inspection and maintenance operation, and these have been optimally placed to minimize radiation exposure. In addition, gasketed and bolted closures minimize disassembly and reassembly time. (Conway testimony , ff. Tr. 1309, at 10-11).

C-64. The steam generators in both units of Byron station embody state-of-the-art design features to minimize occupational radiation exposure. (Conway testimony, ff. Tr. 1309, at 11-12).

C-65. Specific steps taken at Byron to keep radiation exposure from maintenance work on steam generators ALARA include (1) installation of a manway handling device; (2) use of remote control inservice inspection equipment; (3) filtering of air from inside the manways. (Van Laere testimony, ff. Tr. 1707, at 22-23).

C-66. The Staff has found, based on its review, that the radiation protection design and program described in the FSAR for Byron are in accordance with the criteria of the Standard Review Plan and Regulatory Guides 8.8 and 8.10 and are acceptable. (Lamastra _et al_. testimony, ff. Tr. 1883, at 13. The Staff has concluded that the radiation protection measures incorporated in the plant design provide reasonable assurance

that occupational doses will be maintained ALARA and below the limits of 10 CFR Part 20. (Lamastra et al. testimony, ff. Tr. 1883, at 14, 18).

C-67. The Board finds that, contrary to Intervenor's Contentions, Byron Station has been designed in a way that facilitates compliance with the ALARA requirement of the regulations.

(2) Plant procedures for radiation safety

C-68. Edison has a corporate ALARA program which is designed to minimize both the radiation dose received by individual workers and the sum of doses received by all exposed workers. This ALARA program is based on the linear, no-threshold dose-response model. It assumes, based on this model, that the potential harmful health effects from radiation exposure can be reduced by reducing the total dose. (Rescek testimony , ff. Tr. 1157, at 2-3).

C-69. Edison's ALARA program has several basic parts. A manual sets forth the organization for ALARA compliance at Edison nuclear stations and at the corporate level to coordinate ALARA activities. This manual also provides guidance on dose reduction techniques, evaluation of proposed actions, and methods of documenting the results of the ALARA program. A dosimetry program provides careful and accurate monitoring of each person's dose equivalent, which are then recorded in a central, computerized recordkeeping program. A Radiation Evaluation Program documents the dose expenditure resulting from various types of work. A training program is provided for nuclear personnel which includes instruction on how to keep personal exposures ALARA. In addition, radiation protection standards, station design procedures, station operating and maintenance procedures, and review of station operation

experience are part of Edison's ALARA program. (Rescek testimony, ff. Tr. 1157, at 4-5, 6-7 and Rescek Exhibit 2).

(a) Organization

C-70. Edison's ALARA organization consists of a corporate and nuclear station ALARA committees, which include as members representatives from key functioning groups such as operations, maintenance, radiation-chemistry and technical support, as well as a designated ALARA coordinator. These committees meet on a routine quarterly basis. (Rescek testimony, ff. Tr. 1139, at 7-8).

C-71. The corporate ALARA committee has the responsibility to direct corporate ALARA activities, approve station ALARA goals, review overall performance, and report on the performance and effectiveness of the ALARA program to the Vice President of Nuclear Operations. The station ALARA committees are responsible for developing and monitoring the progress of station ALARA goals; they must submit periodic progress reports to the corporate committee for review. (Rescek testimony, ff. Tr. 1139, at 8).

C-72. The Byron station ALARA Committee will formulate goals to give the program direction, will review station progress toward those goals, and will conduct special meetings to review plant modifications and unanticipated maintenance work. (Van Laere testimony, ff. Tr. 1707, at 3-4).

C-73. Edison witness Van Laere is the lead health physics person at the station. (Tr. 1714 (Van Laere)). Mr. Van Laere reports to the assistant superintendent of administration and support services; he has direct

access to the station superintendent as necessary. (Tr. 1766 (Van Laere)).

C-74. The ALARA program for Bryon sets out the specific responsibilities of the Station Superintendent, the Radiation Protection Manager, the ALARA Coordinator, the Assistant Superintendent of Administration and Support Services, the Training Department, the Maintenance Department, the Operating Department and the Chemistry Groups. (See Van Laere testimony , ff. Tr. 1707, at 5-10). Radiation protection procedures have been prepared specifically for Byron. (Van Laere testimony, ff. Tr. 1707, at 10 and Van Laere Exhibits 2-7.

C-75. Byron's health physics department will consist of 43 employees assigned in whole or in part to that department and will include health physicists, health physics engineering assistants, health physics foremen and radiation chemistry technicians. Each will receive sufficient training to ensure that he or she is able to adequately implement his or her responsibilities and to satisfy any regulatory training requirements for that position. (Van Laere testimony, ff. Tr. 1707, at 13).

(b) Dosimetry recordkeeping

C-76. Dosimetry recordkeeping is the responsibility of each nuclear station's lead health physicist and his staff under Edison's program. The management and control of the computer program for dosimetry recordkeeping is the responsibility of the corporate health physics staff, which is also responsible for program development and trending of occupational exposure data. (Rescek testimony, ff. Tr. 1139, at 13).

C-77. Edison imposes adminstrative daily dose limits of 50 millirem and weekly limits of 300 millirem to ensure that occupational doses are reasonably distributed among individuals and are kept well below legal limits. If a worker is expected to have to exceed the daily dose limit, a Radiation Work Permit which provides detailed information on the job to be done, dose estimates, and protective requirements, must be prepared. In no case would any worker be authorized to exceed any limit set by a NRC regulation in a non-emergency task. (Rescek testimony, ff. Tr. 1139, at 13-14).

C-78. Edison has a radiation exposure recordkeeping program as required by 10 CFR § 20.401 which includes records for radiation exposures for individuals and the results of surveys, monitoring and disposals. Records are supplied to the NRC consistent with Regulatory Guide 1.16 and 10 CFR § 20.407. (Rescek testimony, ff. Tr. 1139, at 15).

C-79. Edison's recordkeeping includes information on each worker's prior occupational dose history. NRC Form-4 must be completed by each radiation worker; this form also serves as a registration for data entry into Edison's computer dosimetry program. (Rescek testimony, ff. Tr. 1139, at 15-16; see Lamastra et al. testimony, ff. Tr. 1883, Attachment B). An NRC Form-5 is also generated for each individual; in addition to selected data from NRC Form-4, this contains the biweekly records of whole body, skin of whole body, and extremity dose data as well as the results of direct and indirect bioassays. (Rescek testimony, ff. Tr. 1139, at 17; see Lamastra et al. testimony, ff. Tr. 1883, Attachment G).

C-80. While the radiation worker can hypothetically provide false information about previous exposures, typically the temporary workers

brought onsite are skilled workers with past experience which can be checked. Edison witness Rescek was not aware of any instance of the discovery that falsified information had been provided in five years of experience at Zion plant. (Tr. 1233-35 (Rescek)). Workers with prior experience in the nuclear industry have documentation which can be checked. (Tr. 1265-67 (Rescek)).

(c) Training

C-81. Edison has a training program which is designed to ensure that each person is sufficiently trained in their job to adequately implement their responsibilities and to fulfill any regulatory training requirements for these positions. The training departments at each station address the ALARA concept. Employees are instructed on the fundamentals of radiation exposures, the importance of maintaining doses ALARA, and their role in accomplishing that objective. A basic indoctrination course defining acceptable personal conduct, protective equipment and clothing requirements, and the biological effects of radiation exposure is a requirement for all new workers. Retraining sessions are required on an annual basis. (Rescek testimony, ff. Tr. 1139, at 20-21).

C-82. The same amount of training is not received by all workers. Those workers who will be performing complex tasks in selected high radiation areas are offered special proficiency training. (Rescek testimony, ff. Tr. 1139, at 21). All workers who may be exposed to radiation receive the minimum standard of training in accordance to Edison's Nuclear General Employee Training ("N-GET") program. The training office is responsible for determining what additional training may be necessary based on an

evaluation of individual experience and needs. (Rescek testimony , ff. Tr. 1139, at 21-22 and Rescek Exhibit 3).

C-83. All employees who work at the station are exposed to the information covered in the N-GET Training manual. (Tr. 1277 (Rescek)).

C-84. The N-GET program provides a detailed outline of the training procedures at Edison. Sufficient training procedures are provided by the N-GET program to enable employees to learn how to help keep radiation exposures ALARA. (Rescek testimony, ff. Tr. 1139, at 22).

C-85. The N-GET training manual includes information for employees on the biological effects of radiation. (Tr. 1186 (Rescek); see Rescek testimony, ff. Tr. 1139, Rescek Exhibit 3, Module 3 at 10-23).

C-86. Byron will comply with all aspects of the N-GET training program. (Van Laere testimony, ff. Tr. 1707, at 23).

(d) Protection of pregnant female employees

C-87. Edison educates its female employees as to the potential risks of radiation exposure to a fetus through a film and literature. Each woman is required to sign a statement that she will tell her supervisor when she knows or suspects she is pregnant. (Rescek testimony , ff. Tr. 1139, at 26).

C-88. Edison does not unilaterally restrict all women of childbearing age from being assigned to radiation jobs. Instead, attention is focused on providing adequate means of protecting both the declared pregnant woman and the fetus. (Rescek testimony, ff. Tr. 1139, at 26).

C-89. As soon as a female employee declares her pregnancy, her radiation exposure records are reviewed to determine how much exposure she has received. (Tr. 1193 (Rescek)).

C-90. Edison's policy is to limit the radiation exposure of declared pregnant women to a total of 500 millirem over the gestation period. Whenever possible, pregnant women are assigned to jobs that will not incur significant radiation doses. (Rescek testimony, ff. Tr. 1139, at 27; Tr. 1773 (Van Laere).

C-91. These policies and practices will be followed at Byron. (Van Laere testimony, ff. Tr. 1707, at 23).

(e) Other Byron Station procedures

C-92. Facilities are provided at Byron to clear protective clothing, decontaminate equipment, take sample counts and perform chemical analyses. (Van Laere testimony, ff. Tr. 1707, at 16). In-house facilities will be available to collect urine and fecal samples, as needed. (Van Laere testimony , ff. Tr. 1707, at 16-17).

C-93. Procedures are provided to notify workers and employees where an area is a radiation controlled area and to protect workers who must enter such areas. Van Laire, ff. Tr. 1707, at 18.

(f) Staff review of Byron ALARA compliance

C-94. The Staff has reviewed the radiation protection/ALARA program for Byron according to the acceptance criteria of the Standard Review Plan. Edison's conformance with the provisions of Regulatory Guide 8.8 was evaluated. Special attention was given to: (1) management policy and organization; (2) personnel qualifications and training; (3) design of facilities and equipment; (4) radiation control programs, plans and procedures; and (5) availability of supporting equipment, instrumentation and facilities. The Staff found that the radiation

protection/ALARA program for Byron met the Staff's criteria. (Lamastra et al. testimony, ff. Tr. 1883, at 9-10).

C-95. Edison has estimated a design dose of 400 person-rems per unit. The Staff has reviewed this estimate, which is equivalent to the dose estimate of currently operating PWR's and concluded it is acceptable. (Lamastra et al. testimony, ff. Tr. 1883, at 3).

C-96. The Staff is satisfied that Edison will have in place for Byron an adequate health physics staff to provide necessary radiation protection services. The qualifications for the Radiation/Chemistry Supervisor meet the applicable criteria. The training and qualification of health physics technicians will meet the criteria of ANSI 18.1, "Selection and Training of Nuclear Power Plant Personnel." (Lamastra et al. testimony, ff. Tr. 1883, at 11).

C-97. Edison described its occupational records system and exposure tracking by task system to the Staff in the FSAR and responses to Staff questions. Edison has committed to follow the guidance of Regulatory Guide 8.7, "Occupational Radiation Exposure Records System" and ANSI 13.6-1966 (R. 1972). "American National Standard Practice for Occupational Radiation Exposure Records System." The data required by these guidance documents includes: (1) positive identification of individuals; (2) a summary of prior radiation exposure received by an individual; (3) radiation exposure received by individuals at other installations during current employment; (4) identification of the type of dosimeters used; (5) radiation exposure received by individuals at the facilities (X-ray, gamma, beta or neutron); (6) a record of bioassay

data; and (7) a record of bioassay data interpretation. (Lamastra et al., ff. Tr. 1883, at 15-16).

C-98. Edison's training program is sufficient to meet NRC regulatory requirements and should ensure that workers understand the potential risk of radiation exposure. (Lamastra et al. testimony, ff. Tr. 1883, at 18). Edison has committed that plant personnel will receive general employee radiation protection training and other specific radiation protection training depending on assigned duties in the plant. Edison's training program will be routinely evaluated by NRC inspection personnel to ensure compliance with the training requirements of 10 CFR § 19.12.

C-99. During cross-examination by counsel for Intervenor League, Edison witness Rescek was presented with an NRC health physics appraisal report for Zion Nuclear Power Station, dated June 27, 1980. See League Exhibit 1. Mr. Rescek was lead health physicist at Zion from June, 1978 to May 1982. (Tr. 1235 (Rescek)). Eight significant appraisal findings were made by the NRC Staff to which Edison was required to respond. These included such areas as management support of the health physics program, emergency response capability, possible liquid and gaseous radioactive waste system problems, contamination controls, and monitoring. Mr. Rescek described in great detail the actions taken by Edison to address these Staff findings. The follow-up actions taken by Edison were reviewed by NRC inspectors and found to be satisfactory to resolve the matters raised by the Staff. (Tr. 1244-56 (Rescek)). A number of these actions taken at Zion will be incorporated throughout Edison's system and, thus, become part of the initial program at Byron. (Tr. 1296 (Rescek)).

C-100. In sum, the Staff has found the radiation protection program for Byron acceptable under its criteria. (Lamastra et al. testimony , ff. Tr. 1883, at 13).

(3) In-plant radiation monitoring

C-101. Edison has a dosimetry program which is designed to provide an accurate assessment of the dose equivalent received by an individual. (Rescek testimony, ff. Tr. 1139, at 9). This program will be followed at Byron. (Van Laere testimony, ff. Tr. 1707, at 14).

C-102. The primary neutron monitoring devices used by Edison are personal neutron dosimeters which contain CR-39, a neutron sensitive material. These dosimeters are processed by an independent vendor. These devices have a lower neutron energy threshold than NTA film and exhibit good stability over time. (Rescek testimony, ff. Tr. 1139, at 10; Tr. 1163 (Rescek)). There are no plans to use NTA film for monitoring at Byron. (Tr. 1174 (Rescek); see Van Laere testimony, ff. Tr. 1707, at 15).

C-103. Edison also performs neutron monitoring with a calibrated REM-meter. The use of a calculated neutron dose equivalent to supplement the neutron dosimeter is consistent with the guidance presented by the Staff in Regulatory Guide 8.14, "Personal Neutron Dosimeters." (Rescek testimony, ff. Tr. 1139, at 10; Tr. 1179 (Rescek)). TLD's are also used to monitor extremity dose. These are processed by a contract vendor. (Id.)

C-104. Beta and gamma radiation doses to station personnel are monitored by film badges and by pocket ionization chambers for all personnel. Film badges are processed on a weekly basis and the results are entered into the computer tracking system. Pocket ionization chambers are read

daily and these results are also computerized. (Rescek testimomy, ff. Tr. 1139, at 11).

C-105. Cross-checking of the results from the film badges and the pocket ionization chambers is conducted at the end of each biweekly period to confirm that the film badges have been processed correctly. In addition, badges "spiked" to known amounts of radiation are submitted to the processor to ensure that processing is correctly done. Independent quality assurance tests are conducted on the film badge program by both Edison and the film badge processor. (Rescek testimony, ff. Tr. 1139, at 11-12; see Tr. 1225-27; 1256-58 (Rescek)).

C-106. Monitoring is also conducted by radiation protection specialists, using calibrated instruments. (Rescek testimony, ff. Tr. 1139, at 12.

C-107. Monitoring at Byron will include provisions to (1) measure beta doses; (2) monitor noble gases within the plant; (3) measure the isotope composition of radioiodines in the station; and (4) distinguish between airborne gases and particulates. (Van Laere testimony, ff. Tr. 1707, at 19-20). More than 200 in-plant area and air monitoring instruments will be provided. (Id. at 20).

C-108. Dr. Morgan did not know how many fixed area monitors there are at Byron. (Tr. 1662 (Morgan)).

C-109. Byron will also provide to workers who must perform a job which may expose them to higher radiation levels dosimeters that emit a sound which increases in intensity as the level of radiation increases. These will be provided as necessary. (Van Laere testimony, ff. Tr. 1707, at 15-16).

C-110. Byron will have facilities to perform its own monitoring for internal radioactive contamination by whole-body events. (Van Laere testimony, ff. Tr. 1707, at 17).

C-111. The Staff has reviewed Edison's area radiation monitoring instrumentation, airborne radioactivity monitoring instrumentation, radiation survey equipment and personnel monitoring equipment. The results of that reivew are set forth in the SER at Sections 12.3.4, 12.5.2, and 12.5.3. In summary, those sections discuss Edison's total in-plant monitoring system for Byron. This consists of (1) fixed area radiation monitors designed to monitor and record gamma radiation levels at selected locations within the plant and to sound an alarm when fixed gamma levels are reached; (2) fixed and portable continuous monitors designed to detect 10 mpc-hours of particulate and iodine radioactivity in any compartment that may contain airborne radioactivity and may be occupied by personnel; and (3) general radiation protection instrumentation such as portable radiation survey instruments, laboratory equipment, air samples and respiratory protective equipment. This is in addition to the film badges, TLD's (thermoluminescent dosimeters) and pencil dosimeters discussed above, as well as whole body counts and other bioassay procedures. (Lamastra et al., ff. Tr. 1883, at 18A-20A and Attachment J).

C-112. The Staff evaluated Edison's in-plant monitoring system for Byron against the requirements of 10 CFR Part 20, the Standard Review Plan, and Regulatory Guides 8.8, 8.9 and 8.26; the Staff concluded that the Byron in-plant monitoring system is acceptable and adequate to ensure

that worker exposures will be maintained ALARA. (Lamastra et al.testimony, ff. Tr. 1883, at 20A).

c. Employment of contract workers at Byron Station

(1) Radiation protection

C-113. Edison will employ contract workers at Byron for a variety of reasons. First, many contractors are specialists in doing specific jobs. "Permanent" contractors are hired for some such jobs. Second, a contract worker with special skills may be hired temporarily to perform particular jobs. Third, Edison may hire temporary contract workers to supplement the number of Edison employees doing a particular task so that each person's dose is always maintained well under the regulatory limits. (Rescek testimony , ff. Tr. 1139, at 23-24; see Tr. 1678-79 (Morgan), Tr. 1899-1900 (Lamastra)). The use of such workers at nuclear power plants is a common practice. (Tr. 1900 (Lamastra)).

C-114. Contract radiation workers receive the same N-GET training as Edison employees. All workers, whether Edison employees or contract workers, are trained in how to use any safety equipment that is required for their job. (Rescek testimony, ff. Tr. 1139, at 22-23).

C-115. Edison's monitoring program is used for every worker who enters a radiation controlled area, including contract workers. (Rescek testimony , ff. Tr. 1139, at 24).

C-116. Each contract worker must complete a NRC Form-4, which includes his occupational dose history, before he begins to work. From this form, Edison can determine the dose the worker has received in the current year and quarter and can establish appropriate restrictions to prevent

the worker from exceeding either of those radiation dose limits. (Rescek testimony, ff. Tr. 1139, at 24; see Lamastra et al. testimony, ff. Tr. 1883, Attachment B).

C-117. Edison is required to control the doses to contract workers in accordance with the provisions of 10 CFR § 20.102. Specific disclosure requirements are set by 10 CFR §§ 20.101 and 20.102 from individuals who may have occasion to exceed specific dose levels. (Lamastra et al. testimony, ff. Tr. 1883, at 2-3).

C-118. The dosimetry recordkeeping requirements are the same for temporary workers as for regular employees. (Tr. 1188 (Rescek)). These procedures in relation to temporary employees will be applied at Byron. (Van Laere testimony, ff. Tr. 1707, at 23).

C-119. The Staff considers Edison's training program, particularly as it relates to transient or temporary workers, to be adequate to minimize radiation doses. (Lamastra et al. testimony, ff. Tr. 1883, at 16-17).

(2) Security considerations

C-120. The testimony of the Staff and the Applicant addressed the question whether the use of temporary workers would increase the risk of sabotage at Byron. (See Lamastra et al. testimony, ff. Tr. 1883, at 11-13 (testimony of Robert F. Skelton); Ruolo testimony, ff. Tr. 1356).

C-121. The Staff testified that there is "a potential increase in risk of sabotage" associated with allowing additional personnel onsite. However, the Staff found that the overall risk is still small provided that appropriate regulatory requirements are satisfied. (Lamastra et al. testimony, ff. Tr. 1883, at 11).

C-122. The Staff conducted a review of the Byron Station's Physical Security Plan (Rev. 7, dated October 8/December 22, 1982). Edison has committed to implement the prescriptive requirements of 10 CFR § 73.55(b) through (h), which include provision for access controls and protection of vital equipment. Access controls include: (1) identification and picture badge system; (2) search of individuals for firearms and explosives; (3) vehicle searches; (4) delivered package and material identification and search; (5) escort of visitors; (6) pre-employment screening for all employees who have unescorted access; and (7) screening for contractor employees who have unescorted access. Steps for protection of vital equipment include: (1) location of such equipment behind second barriers; (2) limitation of access to such equipment only to performance of duties; (3) locking and alarming areas that contain vital equipment; (4) special controls for containment during refueling and maintenance. (Lamastra et al. testimony , ff. Tr. 1883, at 11-12).

C-123. Edison's Physical Security Plan for Byron contains a specific chapter which deals with "Security Measures During Maintenance, Refueling and Major Modifications." This chapter contains additional specific commitments to deal with activities which require additional personnel. The Staff has determined, based on the various procedures and commitments for Byron, that the security plan for Byron provides the necessary measures to protect against any potential increased risks caused by the presence of additional persons such as contract workers. (Lamastra et al. testimony, ff. Tr. 1883, at 13).

C-124. Edison's security plans are adequate to minimize any potential increase in the risk of sabotage associated with the use of temporary workers onsite at the Byron Station. Lamastra et al. testimony, ff. Tr. 1883, at 13).

C-125. Edison requires that all personnel who will need unescorted access to Byron Station undergo pre-employment screening. Except where the results of the pre-employment screening justify acceptance of an individual for unescorted status, unescorted contract employees are not allowed. (Ruolo testimony, ff. Tr. 1356, at 2).

C-126. Contractors working at Byron Station are required to submit a screening procedure to Edison's Nuclear Security Administration for approval. That procedure must include provisions, inter alia, for: (1) evaluation of employees with a trustworthy work record of three continuous years; (2) background checks of employees employed by the contractor for less than three years; (3) observation of employees for aberrant behavior; (4) immediate notification of termination of personnel with unescorted access status; and (5) documentation of the procedures and of specific information obtained through the program. (Ruolo testimony, ff. Tr. 1356, at 3).

C-127. Edison conducts audits of the contractor screening program by the staff of the Nuclear Security Administrator and by Edison's quality assurance organization. (Ruolo testimony, ff. Tr. 1356, at 3-4; Tr. 1390 (Ruolo)).

C-128. Written guidelines for screening requirements are furnished to all contractors who will require unescorted access to Byron Station. (Ruolo testimony, ff. Tr. 1356, at 4 and Ruolo Exhibit 1).

C-129. Edison witness Ruolo testified that the contractor screening procedure provides assurance that there is "no increased risk due to the employment of transient workers." (Ruolo testimony, ff. Tr. 1356, at 5) (emphasis added). The Board finds more compelling the conclusion of Staff witness Skelton that there is a potential increase in risk of sabotage associated with allowing additional personnel onsite but that Edison's security plans are adequate to minimize any such potential increase. (See Lamastra et al. testimony, ff. Tr. 1883 at 11, 13).

C-130. Most significantly, a security plan that satisfied the specific requirements of 10 CFR § 73.55(b)-(h), as Edison's Byron plan does, also satisifies the performance objective of providing high assurance that operation of the reactor would not constitute an unreasonable risk to public safety. (See Lamastra et al. testimony, ff. Tr. 1883, at 13).

D. Steam Generator Tube Integrity

1. Matter in Controversy

D-1. As admitted for litigation, League Contention 22 states:

> An extremely serious problem occurring at other plants such as Consumers' Palisades Plant and C.E.'s Zion Plant, and likely to occur at C.E.'s Byron Plant, is presented by degradation of steam generating tube integrity due to corrosion induced wastage, cracking, reduction in tube diameter, and vibration induced fatigue cracks. This affects, and may destroy, the capability of the degraded tubes to maintain their integrity, both during normal operation and under accident conditions, such as a Loca or a main steam line break. The Commission Staff has correctly regarded this problem as a safety problem of a serious nature, as evidenced both by NUREG-0410 and the Black Fox testimony cited above [sic]. As a result of this serious and unresolved problem, the findings required by 10 CFR § 50.57(a)(3)(ii) and 50.57(a)(6) cannot be made.

D-2. DAARE/SAFE Contention 9(c) states:

> 9. Intervenors contend that there are many unresolved safety problems with clear health and safety implications and which

are demonstrably applicable to the Byron station design, but are not dealt with adequately in the FSAR. These issues include but are not limited to:

(c). Steam generator tube integrity. In PWRs steam generator tube integrity is subject to diminution by corrosion, cracking, denting and fatigue cracks. This constitutes a hazard both during normal operation and under accident conditions. Primary loop stress corrosion cracks will, of course, lead to radioactivity leaks into the secondary loop and thereby out of containment. A possible solution to this problem could involve redesign of the steam generator, but at FSAR, Section 10.3.5.3 the Applicant notes its intent to deal with this as a maintenance problem, which may not be an adequate response given the instances noted in Contention 1 above [sic].

2. Regulatory Background

D-3. Westinghouse steam generator tube integrity is designated as unresolved safety issue A-3. Pursuant to Appeal Board decisions, individual NRC safety evaluation reports must describe those unresolved safety issues relevant and potentially significant to the facility under review and some explanation why operation can proceed in advance of an overall solution.[10/] The most common justifications are that "a solution satisfactory for the particular facility has been implemented; a restriction on the level or nature of operation adequate to eliminate

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10/ Virginia Electric & Power Company (North Anna Nuclear Power Station, Units 1 and 2), ALAB-491, 8 NRC 245, 248 (1978), Gulf States Utility Company (River Bend Station, Units 1 and 2), ALAB-444,6 NRC 760, 774 (1977).

the problem has been imposed; or the safety issue does not arise until the later years of operation.[11/] The Byron Safety Evaluation Report lists steam generator tube integrity as generally applicable to Byron and provides the justification for its disposition at Byron. (Frank testimony, ff. Tr. 4473, at 1).

3. Substantive Findings

D-4. The Applicant presented the prepared testimony of ten witnesses and the oral testimony of one additional witness. The following Applicant witnesses introduced prepared testimony following Tr. 4126: Mahendra R. Patel, manager of a structural mechanics group in the Nuclear Technology Division, Water Reactor Division, Westinghouse Electric Corporation; Daniel D. Malinowski, a manager of field data analysis in the Steam Generator Programs of the Westinghouse Water Reactors Division; Michael J. Wooten, Manager of Chemistry Development with Westinghouse; Dr. Lawrence Conway, an Advisory Engineer in the Westinghouse Steam Turbine Generator Division; and John C. Bolmgren, a Special Projects Group Leader in the Applicant's Technical Services Nuclear Department. Rodolfo Paillaman, a senior quality nondestructive examination specialist with Ebasco Services, Inc. testified following Tr. 4818. The following Applicant witnesses introduced prepared testimony following Tr. 5908: Thomas F. Timmons, a Manager of Reactor Coolant Systems Components

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11/ North Anna, 8 NRC at 248. These are not the only acceptable explanations, however. See Pacific Gas and Electric Co. (Diablo Canyon Nuclear Plant, Units 1 and 2), LBP-81-21, 14 NRC 107, 118 (1981), aff'd, ALAB-728, 17 NRC ____ (1983).

Licensing in the Nuclear Safety Department of the Westinghouse Nuclear Technology Division; Lawrence D. Butterfield, a Section Engineer in the Applicant's Station Nuclear Engineering Department; Wilson D. Fletcher, Manager of Steam Generator Materials and Chemistry with Westinghouse Electric Corporation; and Michael J. Hitchler, Manager of Probabilistic Risk Assessment with the Westinghouse Nuclear Safety Department. Mr. Kenneth J. Green, Mechanical Project Engineer for Sargent and Lundy also testified on behalf of the Applicant beginning at Tr. 6210.

The Staff direct case consisted of the following prepared testimony following Tr. 4473: Mr. Conrad E. McCracken, the Section Leader of the Chemical Technology Section, Chemical Engineering Branch, NRC Division of Engineering; Louis Frank, a Senior Materials Engineer in the Materials Engineering Branch, NRC Division of Engineering; Dr. Jai Raj N. Rajan, a Mechanical Engineer in the Mechanical Engineering Branch, NRC Division of Engineering; and Mr. Ledyard B. Marsh, Section Leader in the Reactor Systems Branch, NRC Division of Systems Integration.

The Intervenor direct case consisted of the prepared testimony of Mr. Dale Bridenbaugh (ff. Tr. 6406), a professional nuclear engineer and co-founder and president of MHB Technical Associates. The witnesses were subjected to cross-examination by the parties and questioning by the Board.

a. Overview

D-5. The Staff testimony provides a valuable synopsis of the nature and disposition of the steam generator tube integrity problems. Tube degradation problems at Westinghouse steam generators have included the following: (1) wastage and thinning corrosion; (2) pitting; (3) denting; (4) intergranular attack; (5) stress corrosion cracking; (6) wear caused

by flow induced vibration; and (7) wear and or impact damage as a result of foreign objects or loose parts. (Frank testimony, ff. Tr. 4473, at 2).

D-6. Measures which have been or will be taken to insure tube integrity fall into two general categories: (1) measures to minimize the potential for degradation and (2) surveillance requirements to insure that acceptable levels of tube integrity are maintained. Id.

D-7. Measures which have been taken to minimize the potential for degradation due to corrosion include: (1) improved design features, (2) use of all volatile treatment secondary water chemistry, and (3) an improved program to control secondary water chemistry. Id.

D-8. Early operating experience of lead operating pressurized water reactor (PWR) facilities which employ Westinghouse Model D steam generators similar to those used at Byron indicate that tubes in the preheater region may be subject to excessive wear due to flow-induced tube vibrations when these facilities are operated at power levels in excessive of 70% full power (model D4-D5). Measures have been proposed by Westinghouse to minimize potential for such excessive tube vibration. (Frank testimony, ff. Tr. 4473, at 2-3).

D-9. The potential for damaging tubes as a result of foreign objects and loose parts being present in steam generatores can be minimized by appropriate surveillance. Examples of available surveillance methods include visual inspections with the aide of fiber optics and/or radio camera devices, and loose parts (acoustic) monitoring of the steam generator during operation. Id.

D-10. Byron is being required to implement Regulatory Guide 1.133 which provides in part for a loose parts monitoring system to be mounted

on the lower plenum of each steam generator. This is expected to reduce the potential for foreign objects or loose parts remaining in the steam generator for long periods of time and potentially causing damage to tubes as occurred at the Ginna nuclear plant. Additional generic surveillance requirements are undergoing Staff development and review. Id.

D-11. The above measures are expected to reduce the potential for the types of problems which have been experienced to date. However, some degree of degradation is likely to occur at Byron during its lifetime. Given the potential for degradation, surveillance requirements are essential to ensure adequate tube integrity is maintained against rupture and excessive leakage during the full range of normal operating and postulated accident conditions. (Frank testimony, ff. Tr. 4473, at 4).

D-12. The Byron steam generator tubes will be subjected to periodic inservice inspection in accordance with Regulatory Guide 1.83, Revision 1, "Inservice Inspection Requirements of Pressurized Water Reactor Steam Generator Tubes" and NUREG/0452, Revision 4, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors." Operational limits on allowable primary to secondary leakage will provide added assurance of adequate tube integrity. Id.

D-13. Numerous design changes and operational procedures have been specifically incorporated at Byron to minimize steam generator corrosion. These include the following steam generator design improvements in the Model D4: (1) elimination of tube sheet crevices; (2) counterflow (axial flow) preheater which minimizes the propensity for steam blanketing; and (3) increased blowdown capability, plus a blowdown tee in the middle of

the hot leg bundle to remove corrosion products on the tube sheet. (McCracken testimony, ff. Tr. 4473, at 2).

D-14. The Model D5 steam genetator design utilized in Byron Unit 2 contains the aforementioned plus the following: (1) 405 ferritic stainless steel tube supports (which are more corrosion resistant); (2) broached tube support holes, which further reduce/minimize concentrating crevices; and (3) Inconel 600 tubes thermally treated to enhance corrosion resistance. Id.

D-15. The following further improvements have been implemented in the balance of the secondary cycle design: (1) improved condenser design, including stainless steel tubes and the ability to individually sample the eight condensate locations to minimize condenser leakage; (2) feedwater recirculation line to facilitate steam cleanup prior to initiating flow to the steam generators; and (3) installation of a condensate polishing system for feedwater purification which processes one-third of the total feedwater continuously during startup. This system will be used to aide in rapid cleanup when condenser leakage occurs. (McCracken testimony, ff. Tr. 4473, at 3).

D-16. In the area of operational procedures, a condition in the plant technical specifications has been incorporated which requires a secondary cycle water chemistry program. The Byron secondary water chemistry control program was reviewed against the criteria of Standard Review Plan 5.4.2.1 and was found acceptable. Id.

D-17. The secondary water chemistry program includes: (1) specific limits for impurities in the steam generators; (2) power reduction to 50% if steam generator impurity limits are exceeded to minimize concentration

of corrosive species; (3) shutdown and flushing of steam generators if water chemistry becomes progressively worse; and (4) administrative controls defining responsibility for interpretation and corrective actions based upon secondary water chemistry limits. (McCracken testimony, ff. Tr. 4473, at 3-4).

D-18. The combination of the above design changes and procedures will significantly reduce the potential for corrosion degradation of the Byron steam generators. (McCracken testimony, ff. Tr. 4473, at 4). An elaboration of the above considerations follows.

b. Corrosion-related degradation

D-19. Tube wall cracking is a linear degradation phenomenon that occurs in discrete or network configurations, leaving the bulk of the tube material intact with its original properties undiminished. The degraded local region or crack face may extend through the entire wall thickness. (Patel testimony, ff. Tr. 4126, at 17-18).

D-20. Westinghouse steam generators are of the recirculating type. The initial recommendation in the early 1960's was to use phosphate as the steam generator chemistry control agent. The phosphate is maintained at a desired concentration to provide the pH buffering action against excursions in alkalinity or acidity due to ingress of contaminants. (Wooten testimony, ff. Tr. 4126, at 6-7).

D-21. During the use of the initial water chemistry guidelines, some instances of stress corrosion cracking were observed in plants that operated with excessive condenser inleakage and little or inadequate phosphate control. (Wooten testimony, ff. Tr. 4126, at 7).

D-22. The initial water chemistry guidelines were revised to address this concern. Ranges of phosphate concentration with minimum levels of phosphate were recommended. (Wooten testimony, ff. Tr. 4126, at 7-8).

D-23. The recommended treatment was successful in mitigating the incidents of caustic stress corrosion cracking of the steam generator tubing. (Wooten testimony, ff. Tr. 4126, at 8). There are very few plants that have operated on all volatile treatment that have experienced any kind of secondary side cracking. (Tr. 4179-80 (Malinowski)).

D-24. Another prominent form of corrosion degradation is called thinning. Thinning is the dissolution of the Inconel-600 tubing at a specific point or place in the tubing. (Tr. 4169 (Wooten)).

D-25. Tube wall thinning is a localized reduction in the tube thickness resulting from corrosion by phosphates in high concentrations. It has been observed within sludge piles at the top of the tubesheet and at tube support plate elevations where lower flow velocities allowed concentrations of phosphates to saturation levels. (Patel testimony, ff. Tr. 4126, at 15-16).

D-26. Thinning occurred in the earlier days of plant operation when plants were operating on a phospate chemistry. The thinning of the tubing was associated to a concentrated solution of sodium phosphate which was the corrosive agent that caused the thinning of the tubing. Id.

D-27. The change to all volatile treatment (AVT) water chemistry controls mitigated the thinning of the steam generator tubing caused by acidic phosphate species. (Wooten testimony, ff. Tr. 4126, at 10).

D-28. Thinning has been reduced to the point where it vanishes as a major contributor in plants that have switched to AVT. It has been eliminated in plants that start with AVT. (Tr. 4172-73 (Wooten)).

D-29. The basis for all volatile treatment is that only volatile chemicals are intentionally added as control agents. These agents do not concentrate in the steam generator but are removed by the steam to the rest of the secondary system. (Wooten testimony, ff. Tr. 4126, at 9).

D-30. While the change to AVT mitigated thinning caused by acidic phosphate species, in the immediate years following the conversion to AVT, another form of corrosion called "denting" was observed. (Wooten testimony, ff. Tr. 4126, at 10).

D-31. Denting is a process whereby corrosive impurities are concentrated between a heat transfer tube and a tube support. The resultant corrosion converts the base support metal to metal oxide. (McCracken testimony, ff. Tr. 4473, at 6).

D-32. Denting is due to corrosion of the carbon steel support plates. Denting does not directly result in steam generator tube corrosion. It simply deforms the tube which increases the tubing stress. (McCracken testimony, ff. Tr. 4473, at 7).

D-33. As tubes become more highly stressed, they are more susceptible to stress corrosion cracking. (McCracken testimony, ff. Tr. 4473, at 7; Tr. 4519 (Frank)).

D-34. It is probable that minor denting could occur in the Byron steam generators. Its occurrence does not indicate that the propensity for steam generator tube leakage has been increased. (McCracken, ff. Tr. 4473, at 7).

D-35. Denting is not anticipated to pose a public health and safety concern, namely, something that would initiate tube failures or accidents, over the lifetime of the Byron plant. (Tr. 4528 (Frank)); (Tr. 6507 (Bridenbaugh)). The height of the denting problem was 1977. (Frank, Tr. 4528). Over the years, it has turned out that denting, in many respects, is no longer a problem. Id.

D-36. Since denting has been underway, utilities have developed a technique called profilometry where, by getting a profile of the dent, the strain on the tube can be calculated, and at some point the tube will be taken out of service based on the amount of strain. (Tr. 4519 (Frank)).

D-37. Plants that have only operated on AVT chemistry have had less severe denting and in some cases where these changes have been adopted vigorously, the progression of denting has been arrested. (Wooten testimony, Tr. 4126, at 13-14).

D-38. Pitting is a form of tube degradation that involves small discrete roughly circular regions of tube penetration typically less than 100 mils in diameter. Pits may occur separately or in bands wherein each pit sits independently of others within the band. (Patel testimony, ff. Tr. 4126, at 16-17).

D-39. One plant (Indian Point) experienced pitting of the Inconel tubing which is believed to be due to an acidic chloride condition involving copper and chloride ions. (Wooten testimony, ff. Tr. 4126, at 14). This incident was caused by excessive condenser inleakage, oxygen inleakage and the combination acting on a copper alloy condenser and feed train over a protracted period of time. (Tr. 4803 (McCracken)).

D-40. It is highly unlikely at Byron, given the absence of copper alloys, that a potential for that type of corrosion exists. Id.

c. Remedial measures

(1) Steam generator design.

D-41. Both Byron Unit 1 and Byron Unit 2 have preheat steam generators which are functionally identical. Byron Unit 1 is a model D-4 and Byron Unit 2 a model D-5. The difference between these designs occurred as part of the normal evolution over time of steam generator designs. (Conway testimony, ff. Tr. 4126, at 13-14).

D-42. Many design improvements have been incorporated in the Byron station steam generators to protect against potential tube degradation. (Conway testimony, ff. Tr. 4126, at 14; McCracken testimony, ff. Tr. 4473, at 2).

D-43. Inconel 600 was chosen for the steam generator tube material as being the most suitable for the temperatures, chemical environment, and design basis accident conditions present within the steam generator. To minimize chemical concentrations areas such as at the tube sheet and between the tube and tube support plate, recirculation rates were optimized, the ports in the blowdown pipe were modified, and the tubes within the tube sheet hole were expanded to eliminate the crevices at the tube sheet. Id.

D-44. To minimize tube stresses, the widest space in between tube support plates which is functionally acceptable was selected, the holes in the flow distribution baffle plates and in the top tube support plate

were modified. (Conway testimony, ff. Tr. 4126, at 14-15; McCracken testimony, ff. Tr. 4473, at 2).

D-45. The design of the model D-5 in Unit 2 has been additionally enhanced by: (1) utilizing stainless steel as the material for the tube support plates and baffles, (2) changing the shape of the circular holes and the tube support plates to a quatrefoil shape, (3) expanding the tubes within the tube sheet by means of a hydraulic device in lieu of mechanical rollers, (4) thermally treating the Inconel 600 tubes and (5) changing the holes in the flow distribution baffles were changed from slotted to a circular shape. Id.

D-46. Despite the design enhancements in the model D4-5, the D4 steam generator is no less safe. (Tr. 4435 (Conway)).

(2) Preservice inspection.

D-47. A preservice inspection has been performed by Ebasco Services, Inc. on the steam generator tubes for Byron Unit 1. (Paillaman testimony, ff. Tr. 4818, at 4).

D-48. The purpose of performing the preservice inspection was to establish a baseline against which subsequent inservice inspections can be compared. Id.

D-49. A preservice inspection was conducted of 100% of the tubes using multi-frequency eddy current examination. (Paillaman testimony, ff. Tr. 4818, at 5). The eddy current probe is designed to measure any severe departures from a nominal condition. (Tr. 4393-94 (Malinowski)).

D-50. Two tubes were plugged as a result of the preservice inspection as Unit 1. All other reportable indications were considered

insignificant. (Paillaman testimony, Tr. 4818, at 7-10). Preservice inspection of Unit 2 will be performed prior to operation. (Paillaman testimony, ff. Tr. 4818, at 4).

(3) Inservice inspection

D-51. The Byron steam generator tubes will be subject to periodic inservice inspection in accordance with Regulatory Guide 1.83 and NUREG-0452 as noted above. See Finding B-12. Standard technical specifications require that inservice inspections be performed every 12 to 24 months, depending on the condition of the steam generators. (Frank testimony, ff. Tr. 4473, at 4; (Tr. 4515 (Frank)).

D-52. The standard method for inspecting the steam generator tubes is eddy current testing (ECT). Id. Steam generator inspections typically involve inspections of a representative sample of tubes. The minimum inspection sample required by the standard technical specifications is 3% of the steam generator tubes. (Frank testimony, ff. Tr. 4473, at 4-5). This will be the practice at Byron (Blomgren testimony, ff. Tr. 4126, at 11-12).

D-53. The standard technical specifications require additional tube samples to be inspected depending on the number of degraded and defective tubes found. In this context, degraded tubes are those exhibiting ECT indications of wall penetration exceeding 20% but which are less than the plugging limit. Defective tubes are those which exceed the plugging limit. (Frank testimony, ff. Tr. 4473, at 5; Tr. 4534 (McCracken)).

D-54. The total number of tubes to be inspected during an inservice inspection will range between 3% and 100% of the total number of steam generator tubes. (Frank testimony, Tr. 4473, at 5).

D-55. The Byron steam generators will be periodically monitored through an eddy current inspection program. (Blomgren testimony, ff. Tr. 4126, at 11).

D-56. The results of these periodic inspections will be compared to the 100% preservice baseline examination that has been completed at Byron. (Id.; Tr. 4371 (Patel)).

D-57. This comparison provides an ongoing evaluation of the steam generator tubing condition and allows for steam generator maintenance prior to primary to secondary leakage. (Blomgren testimony, ff. Tr. 4126, at 11).

B-58. Operating experience has proven eddy current testing to be a generally reliable technique for purposes of monitoring tube integrity. (Frank testimony, ff. Tr. 4126, at 6).

D-59. The industry has made considerable progress in improving its detection capabilities in this regard, including the development of multiple frequency techniques and new ECT probe designs. Id.

D-60. Multi-frequency eddy current testing is the industry practice. (Tr. 4287 (Malinowski); Tr. 4525 (Frank)). Such technique will be utilized at Byron. (Malinowski, ff. Tr. 4126, at 12).

D-61. The sensitivity of eddy testing using multifrequency techniques is generally expected to meet the requirement to detect 40% tube wall penetration and below. This is expected for such forms of tube

degradation as cracking, thinning, or wear. (Fletcher testimony, ff. Tr. 5908, at 16).

D-62. Periodic inspection has provided the means by which tube degradation, if present, can be measured with respect to the tube plugging criteria established by the NRC in the plant technical specifications. (Fletcher testimony, ff. Tr. 5908, at 15-16).

(4) Primary to secondary leak rate limits

D-63. Plant technical specifications contain limitations on allowable primary to secondary leakage in order to assure that a tube leaking at a rate equal to or less than the limit will retain adequate integrity against rupture under normal operating and postulated accident conditions. (Frank testimony, ff. Tr. 4473, at 7). The standard technical specification primary to secondary leak rate limit is 0.35 gallons per minute. (Tr. 4520 (Frank)).

D-64. Experience has shown leakage events to provide an important indication of the existence of new degradation phenomena, that degradation is developing at an unanticipated rate, and/or the need for licensing action or remedial measures to provide added assurance of tube integrity. (Frank testimony, ff. Tr. 4473, at 7).

(5) Water chemistry

D-65. Operating plant experience has indicated the need for rigorous control of the water chemistry environment of the entire steam cycle. (Fletcher testimony, ff. Tr. 5908, at 7). All volatile treatment (AVT)

has been selected by the majority of the industry as the most suitable chemistry for the system, which includes the steam generators. Id.

D-66. Because of the extensive laboratory work that has been performed in evaluating the mechanisms of stress corrosion cracking, thinning and denting of the steam generator tubes, it is clear that impurities admitted to the steam cycle which ultimately may reside in the steam generator, must be limited. (Fletcher testimony, ff. Tr. 5908, at 8-9).

D-67. Impurities such as air, which contains oxygen, condenser cooling water such as fresh water sources that contain excess alkalinity, make up water impurities and the like must be excluded to the extent possible. It has also been demonstrated that copper bearing alloys in the feed train can participate in corrosion reaction when transported to the steam generators. (Fletcher testimony, ff. Tr. 5908, at 9).

D-68. There are no copper bearing alloys in the feed train at the Byron station. (Wooten testimony, ff. Tr. 4126, at 9).

D-69. AVT chemistry control is based on a philosophy of minimum contaminant ingress through the practice of good initial design and material selection of condensers, feedwater heaters, makeup water systems and other components. (Fletcher testimony, ff. Tr. 5908, at 7-8).

D-70. AVT control is maintained by appropriate inspection and maintenance practices and operator actions during plant operation. Adherence to AVT guidelines enhances the long term integrity of the steam cycle, by minimizing the corrosion of condenser and feedwater heater materials, the steam generator and the turbine. This in turn minimizes the formation of corrosion products which are delivered to the steam

generator, thus reducing the potential for tube corrosion. (Fletcher testimony, ff. Tr. 5908, at 8).

D-71. AVT chemistry guidelines were first issued by Westinghouse during 1974 and have been adopted by the majority of the utilities in the United States. These guidelines have been reviewed by industry groups and more recently, in order to reinforce the need for vigorous chemstry control, EPRI has issued AVT guidelines as a model to be reviewed by the industry. (Fletcher testimony, ff. Tr. 5408, at 8).

D-72. The basis for the all volatile treatment is that only volatile chemicals are intentionally added as control agents. These agents do not concentrate in the steam generator but are removed by the steam to the rest of the secondary system. (Wooten testimony, ff. Tr. 4126, at 9).

D-73. The Westinghouse guidelines regarding water chemistry recommend that: (1) the guideline chemistry conditions should be achieved prior to unit loading and maintained during power changes; (2) any source of contamination should be identified and the source corrected: (3) dissolved oxygen at the condensate pump discharge should be less than 10 ppb thus minimizing the inventory of corrosion product transported to the steam generator; (4) continuous monitoring of the steam generator blowdown: (5) copper bearing alloys be eliminated from the secondary system to permit greater flexibility and optimization and chemistry control; (6) main condenser integrity be upgraded to minimize the ingress of impurities in the condensate in order to improve the reliability of the steam generators and turbine; and (7) if a full flow condensate polishing system is installed it must be carefully controlled and properly operated

in order to optimize the quality of the treated condensate. (Wooten testimony, ff. Tr. 4126, at 15-16).

D-74. The Westinghouse guidelines regarding water chemistry have been implemented in the design and construction of Byron as appropriate. Current Westinghouse chemistry controls have been modified to incorporate elements of the Steam Generator Owners Group (SGOG) guidelines, secondary water chemistry guidelines have been used as a basis for the Byron station secondary chemistry monitoring program. (Blomgren testimony, ff. Tr. 4126, at 3-4).

D-75. The Steam Generator Owners Group was established in 1977 by a group of utilities for the purpose of conducting research in the areas of steam generator design, operation and water chemistry control. (Blomgren testimony, ff. Tr. 4126, at 8).

D-76. The SGOG uses the facilities of the Electric Power Research Institute (EPRI), but is separately funded. The SGOG has issued secondary water chemistry guidelines, developed for the SGOG by a committee of utility and vendor water chemistry experts. These guidelines have also been issued, with identical content, by EPRI. The SGOG guidelines are an expansion of the Westinghouse guidelines. (Blomgren testimony, ff. Tr. 4126, at 9).

D-77. The Byron station chemistry monitoring program incorporates the more restrictive elements from the SGOG guidelines, including more restrictive water chemistry controls and a staged corrective action plan. (Blomgren testimony, ff. Tr. 4126, at 9).

D-78. In addition to the more restrictive water chemistry controls, the SGOG guidelines include recommendations for data management and

surveillance, and analytical methods. The SGOG guidelines include a recommendation that specific management responsibilities regarding secondary water chemistry control be assigned from the plant chemist to senior corporate management. Id.

D-79. Byron station monitoring program has incorporated the foregoing elements. (Blomgren testimony, ff. Tr. 4126, at 10).

D-80. The Byron chemistry program is based in large part on the EPRI recommendations contained in a document, entitled "PWR Secondary System Guidelines." (Tr. 4174 (Wooten)). The Applicant is committed to following these guidelines without reservation. (Tr. 4334 (Blomgren)).

D-81. The Staff has independently reviewed the EPRI guidelines and agrees with their contents. It feels that they are a good state-of-the-art generic basis for use in review of secondary water chemistry control. (Tr. 535 (McCracken)).

D-82. Westinghouse also believes the EPRI guidelines represent state of the art in the chemistry control program for an operating nuclear power plant. (Tr. 4227 (Wooten)).

D-83. The Board finds that the improved chemistry and corrosion control measures at Byron will significantly reduce the rate of corrosion of steam generators tubes.

(6) Tube plugging

D-84. The steam generators at both units of Byron embody the very best design features from the prospective of minimizing tube degradation as well as overall performance and safety. (Conway testimony, ff. Tr. 4126, at 15).

D-85. Tube plugging is one of the guidelines used to assure structural integrity of steam generator tubes. (Patel testimony, ff. Tr. 4126, at 5).

D-86. A steam generator tube plugging criterion is established by determining the wall thickness limit below which tubes are to be removed from service by plugging them to avoid the possibility of tube failure due to degradation. Id.

D-87. Conservative design criteria for tube wall sizing have been established to assure structural integrity of the tubing under normal operating and postulated design basis accident condition loadings. (Patel testimony, ff. Tr. 4126, at 5-6). In reality, however, steam generator tubes are manufactured with a wall thickness much greater than the minimum thickness indicated by the design rules (Section III of the ASME boiler and Pressure Vessel Code). Id.

D-88. The minimum required tube wall thickness is based on the results of detailed analyses and strength testing. (Patel testimony, ff. Tr. 4126, at 7). The thermal hydraulic and stress analyses are performed to determine tube loads and the resulting stresses during the normal as well as various postulated accident conditions. (Patel testimony, ff. Tr. 4126, at 7-8).

D-89. A plugging criterion of 40% indication of wall thickness degradation has been established for the Byron station per Section XI, "Inservice Inspection of Nuclear Power Plant Components" of the ASME code. (Patel testimony, ff. Tr. 4126, at 13).

D-90. The plugging criterion of a 40% indication is based on the work performed by the ASME working group on Steam Generator

Inservice Inspection. In its evaluation, that group concluded that for the four different steam generator designs considered, including Westinghouse model D designs, the tubing can sustain degradation in excess of 50% of wall thickness and still meet all applicable stress and strength requirements. (Patel testimony, ff. Tr. 4126, at 14).

D-91. A conservative factor of 10% wall thickness was assumed to account for both eddy current measurement uncertainty and corrosion allowance for the continued plant operation until next inspection. Id.

D-92. This factor was assumed on the basis of some developmental work and historical data on eddy current testing and corrosion rates on operating plants. (Patel testimony, ff. Tr. 4126, at 14-15).

D-93. Thus, a plugging criterion of 40% (50% - 10%) tube wall degradation was established and adopted by the working group on acceptance standards for ASME Section XI. Id.

D-94. The steam generators at the Byron station are Westinghouse model D design which is one of the designs considered by the working group. Id.

D-95. The tubing material is Inconel 600 and the mean radius to thickness ratio is less than the upper limit specified in the ASME code, Section XI. Thus, the 40% plugging margin established per Section XI acceptance standards are conservatively applicable to Byron. Id.

D-96. Further analysis and testing performed by the Applicant showed that for the various types of degradation modes experienced to date, for the tubing in the model D4-D5 steam generators, those models can operate, even if tubes were to degrade to 40% of their nominal design wall (Patel,

Tr. 4143) and still safely withstand all the operating and postulated condition loads on the tubing (Tr. 4151-52 (Patel)).

D-97. Typically, Section III of the ASME code requires a safety factor of 3 for nominal operating pressure differential. Typical Westinghouse steam generator tubes in as-manufactured condition would have a safety factor range of 6 to 10, or approximately twice as thick. (Tr. 4370 (Patel)).

D-98. Steam generator tubes are designed to withstand the differential pressures likely to be experienced during the most severe LOCA, main steam line break or main feed line break accidents. (Rajan, Frank testimony, ff. Tr. 4126, at 3).

D-99. The tube plugging criteria requires further that a reasonable factor of safety still exists when 70 to 75 percent of the steam generator tube wall has been lost. Id.

D-100. A tube that has been uniformly deviated to about 80 percent of the nominal wall thickness could withstand the pressures that would be imposed during a steamline break. (Tr. 4603 (Rajan)).

D-101. For 70%-75% tube wall degradation then, there is a factor of safety of between 1 and 1.3. Id.

D-102. Nevertheless, tubes are plugged when a 40% degradation is observed. (Tr. 4603 (Frank)).

D-103. The Board finds that a 40% plugging criterion is reasonably conservative for use at Byron.

d. Wear-related degradation.

D-104. Tube wear is a form of tube degradation that results from a mechanical abrasion of the tube surface. Such wear progressively reduces

the thickness of the tube area affected. Wear results from the impact of adjacent structures or loose objects on the tubing. It has been observed at anti-vibration bar intersections, the baffle plates in preheat sections and locations in contact with foreign object. (Patel testimony, ff. Tr. 4126, at 21).

(1) Loose parts.

D-105. There have been two domestic rupture events (Prairie Island in 1979 and Ginna in 1982) attributable to damage caused by foreign objects and loose parts. (Frank testimony, ff. Tr. 4473 at 8).

D-106. With regard to the Prairie Island event, following plant shutdown, inspection showed the leaking tube on the periphery of the tube bundle had undergone localized wear due to the presence of a foreign object. The object was a coil spring which remained in the steam generator following an earlier outage for steam generator maintenance. (Fletcher testimony, ff. Tr. 5908, at 14).

D-107. The Ginna rupture was also due to the presence of a foreign object which impacted against and severed a previously plugged tube. The severed tube subsequently wore against an unplugged tube providing a long wear scar which ultimately lead to tube leakage. Id. There is no indication that this event came close to being a multiple tube rupture event. (Tr. 4814 (Marsh, Rajan); Tr. 496-97 (Bridenbaugh)) .

D-108. The question of loose parts in the steam generator and the potential impact on steam generator tubes is being actively pursued by the industry. Id.

D-109. Special attention is being applied to inspection and retrieval techniques for loose objects in the steam generator for tooling development efforts. (Fletcher testimony, ff. Tr. 5908, at 15).

D-110. Fiberoptic and television view devices are being utilized for searching annular regions of the steam generator and other confined areas. Inventory control procedures are implemented such that during maintenance operations tooling and other devices are accounted for before and after the work is completed so as to not leave any part in the steam generator. Id.

D-111. Further, loose parts monitoring systems are utilized for indications of impact signals within the steam generator that alert operators as to the possible presence of metal objects. Id.

D-112. Byron has a two-part approach for the control of loose parts in the secondary side of the steam generators. (Blomgren testimony, ff. Tr. 4126, at 12).

D-113. The first is to control materials and tools used in the steam generators during maintenance and inspection. These controls are set forth in tool and material inventory control procedures. (Blomgren testimony, ff. Tr. 4126, at 12-13; Tr. 4256 (Blomgren)).

D-114. The second portion of the program is to use the installed loose parts monitoring system to promptly identify loose parts in the steam generator. (Blomgren testimony, ff. Tr. 4126, 13).

D-115. The Byron station loose parts monitoring system, as required by NRC Regulatory Guide 1.133, is a monitoring, alarm, and diagnostic system that provides real time information to the operator on a variety of mechanical vibration phenonena that may occur in the reactor cooling

system. (Blomgren testimony, ff. Tr. 4126, at 14). This system includes two sensors on the secondary side of each steam generator. These sensors listen for noise generated by loose parts. (Blomgren testimony, ff. Tr. 4126, at 14; Tr. 4256, 4430 (Blomgren); Frank testimony, ff. Tr. 4473, at 8).

D-116. If a loose part is detected in the secondary side of a steam generator, the unit would be shut down and inspected and the loose part removed. (Blomgren testimony, ff. Tr. 4126, at 16).

D-117. Applicant intends further to conduct periodic visual inspections of the secondary side of the steam generators during refueling and maintenance outages. (Tr. 4257, 4424 (Blomgren)).

D-118. The Byron loose parts monitoring system is expected to reduce the potential for wear-related degradation due to loose parts. (Frank testimony, ff. Tr. 4126, at 8; Fletcher testimony, Tr. 5908, at 14). The Board agrees.

(2) Flow-induced vibration.

D-119. Since 1981, a problem with flow induced vibration and subsequent wear of tubes in the preheater section of Model D steam generators has been identified at lead operating facilities with Model D steam generators. (Rajan testimony, ff. Tr. 4473, at 1; Timmons testimony, ff. Tr. 5908, at 8).

D-120. These include McGuire Nuclear Station, Unit 1 (Model D2) and the following three foreign facilities: Ringhals, Unit 3 (Model D3), Almarez Unit 1 (Model D3), and Krsko (Model D4).

D-121. The tube excitation mechanism appears to be a combination of a threshold type of fluid elastic instability and turbulent buffeting. (Rajan testimony, ff. Tr. 4473, at 2).

D-122. Model D4 and D5 steam generators employ counter-flow type preheater design as opposed to the split flow design employed for Model D2 and D3 steam generators. (Rajan testimony, ff. Tr. 4473, at 2; Timmons testimony, ff. Tr. 5908, at 5).

D-123. Westinghouse has undertaken an extensive program to investigate, understand and define vibration and tube wear in Model D steam generators and to conceive, develop, test, and evaluate any modifications necessary to allow operation of Model D steam generators at full power. (Timmons testimony, ff. Tr. 5908, at 9).

D-124. This program includes gathering, reviewing and analyzing data from operating plants and from model tests. (Timmons testimony, ff. Tr. 5908, at 9-10).

D-125. The collection of operating plant data began in October 1981 with eddy current testing performed at the Ringhals 3 plant and at the other operating Model D plants. Id.

D-126. Eddy current testing data and tube vibration data collected from the then operating Model D steam generators were used to determine the extent of tube vibration and to determine the main feed flow rates below which significant tube vibration would not be expected to occur. (Timmons testimony, ff. Tr. 5908, at 11-12; Rajan testimony, ff. Tr. 4473, at 2).

D-127. It was observed that operation at or below 50% for Model D steam generators did not produce any significant tube vibration or any

significant changes in tube wear. It was also observed that operation of the Krsko steam generators at or below 70% main feed flow did not produce any significant tube vibration. Id.

D-128. In May 1982, the Krsko plant performed an eddy current test inspection, installed additional vibration instrumentation, and removed one tube from the steam generator for visual and metallurgical examination and analysis. (Timmons testimony, ff. Tr. 5908, at 14).

D-129. No indications of tube wear were detected by eddy current testing. The removed tube had some wear with a depth below the limit of ECT detectability. Id.

D-130. Modifications to the feedwater bypass system to allow operation of the plant at 100% power with up to 30% of the feedwater bypassing the preheater of the steam generator were also installed at Krsko at this time. Id.

D-131. With the resumption of operation, vibration data was collected at various power levels and various combinations of main feed and bypass feedflows. From this data, it was observed that the tube vibrations of the 70% main feed/30% bypass feed combination was slightly greater than those of the 70/0 combination and that the vibrations observed at 70/30 were acceptable. (Timmons testimony, ff. Tr. 5908, at 15).

D-132. In November 1982, the Krsko plant performed an ECT inspection, installed additional tube vibration instrumentation in both steam generators, removed two tubes and expanded one tube at baffle plate intersections. Id.

D-133. No indications of tube wear were observed from the ECT inspection. The two removed tubes had wear marks of .001 to .002 inches in depth which are below the limit of eddy current testing detectability. Id.

D-134. After resumption of operation, tube vibration data was obtained. From this data, it was observed that the tube vibrations in both steam generators were similar and had not changed with time. Id.

D-135. The expanded tube had been previously instrumented for vibration and was reinstrumented. Previous tube vibration data was compared with the data obtained after the tube had been expanded and it was concluded that the tube vibrations were reduced by at least a factor of 5 from the nonexpanded tube. Id.

D-136. Various size scale models of the steam generator preheater region have been constructed and have provided data not obtainable from operating plants. Id.

D-137. Two significantly different preheater flow designs comprise the Model D series steam generators; one program was undertaken for laboratory tests on the Model D2/D3 generators and another for the Model D4/D5 generators. (Rajan testimony, ff. Tr. 4473, at 2-3).

D-138. As part of the generic Model D4/D5 program, a 16° full scale model was used to replicate in the laboratory the tube vibration response observed in operating steam generators. (Timmons testimony, ff. Tr. 5908, at 19). A single tube vibration model was used to characterize tube response under various excitation and support conditions. Id.

D-139. Use of these various test models provided the additional capability of testing various concepts designed to reduce tube vibrations. Id.

D-140. Concurrent with collecting and analyzing operating plant and laboratory test data, a computer model was developed to predict tube behavior as a result of flow induced vibration. (Timmons testimony, ff. Tr. 5908 at 20).

D-141. Evaluation of the data from the Model D4 program has disclosed that operation of Model D4 steam generators at high main feed flow rates could produce significant tube vibration of a few tubes. (Timmons testimony, ff. Tr. 5908, at 21; Rajan testimony, ff. Tr. 4473, at 3).

D-142. To date, operation of Model D4 steam generators with main feed flow rates up to 70% has not produced any tube wear that can be detected by any current testing, although visual examination of three removed tubes did disclose a small amount of tube wear, approximately .001 to .0025 inches in depth. (Timmons testimony, ff. Tr. 5908, at 22).

D-143. The feedwater bypass modification installed at Krsko has been effective in reducing tube vibrations to low levels and permitted the plant to operate at full power. Id.

D-144. In order to minimize tube wear from flow induced vibration at Byron, Westinghouse has recommended that the Applicant make modifications to the Byron plant to reduce the potential for significant tube vibrations in the Byron steam generators. These modifications are: (1) the expansion at baffle locations of approximately 100 tubes per steam generator, and (2) the bypassing of approximately 10% of the flow

from the main feedwater nozzle to the auxiliary feedwater nozzle. (Timmons testimony, ff. Tr. 5908, at 22-23).

D-145. Expansion of tubes at baffle plate locations will limit the tube movement at the baffle plate intersections to a few thousandths of an inch. Bypassing of 10% of the main feed flow to the auxiliary nozzle of the steam generator will reduce the main feed flow of the inlet to the preheater to approximately 90% and will further reduce the potential for vibration of the tubes in the preheater. Id.

D-146. Westinghouse has developed a proprietary process that will be used to expand the steam generator tubes. (Timmons testimony, ff. Tr. 5908, at 23). The proprietary process was fully explained in an in camera session on April 27, 1983. (Tr. 6170 (Timmons); Tr. 6176-77 (Fletcher)).

D-147. The process involves the insertion of tools into the tube from the primary side of the steam generator tube sheet. The tubes are then used to locate the baffle plate intersection and to expand the tube at the appropriate location. (Timmons testimony, ff. Tr. 5908, at 23).

D-148. The process is called hydraulic expansion. (Tr. 6269 (Timmons)). Westinghouse has utilized hydraulic methodology for expanding steam generator tubes in the tube sheet since 1977. Id.

D-149. The technique is neither experimental nor developmental. (Tr. 6269 (Timmons)). Expansion of tubes in steam generators has long been utilized in the manufacture of steam generators. (Timmons testimony, ff. Tr. 5908, at 24). Westinghouse has performed a variety of analyses to establish the effect of tube expansion on the subject tubes. That effect was not adverse. Id.

D-150. The tube vibration problem at Krsko was confined to a small number of tubes in the preheat section of the steam generator. The preheat section of the Krsko steam generator has a design in which the flow from the main feed is directed downward by a solid impingement plate and then directed upward through a series of baffle plates. (Tr. 4765 (Rajan)).

D-151. The object of the preheat region is to extract more heat from the primary fluid. For that reason a certain amount of turbulance is desirable. However, a side effect of this increased turbulance is vibration of some of the tubes in that region against the support plates. This has resulted in less than acceptable tube wear on some of the tubes. (Tr. 4765-66 (Rajan)).

D-152. The problem of flow induced vibration has been at the support plate location. (Tr. 4767 (McCracken)). The support plate location is about a one inch thick piece of steel where the tube goes through a hole that has a nominal 20 mil clearance. When the tube vibrates, the tube within that support location becomes worn. If there is a one inch support location and thickness, a one inch spot on the tube is worn. Id.

D-153. The tube expansion is in very close proximity to the support plate to eliminate whipping motion which results in the frictional wear. Id.

D-154. The tube vibration problem is less severe in the model D4-D5 than model D2-D3 due to the nature of the flow in the preheat section of the two tube models. (Tr. 4805 (Rajan)). In the model D2-D3, the main feed flow impinges upon the plates which have holes in them, and has the possibility of impinging on the tubes directly, whereas in the model

D4-D5, the impingement plate is solid and the flow is directed downward first, where it loses a substantial portion of its momentum before it goes upward and has a possibility of impacting on the tubes. Id.

D-155. The general level of turbulance is less in the model D4-D5. (Tr. 4805-06 (Rajan)). The proposed modification that has now been accepted by the NRC Staff for the model D2-D3 is far more complex than that proposed for the model D4-D5. (Tr. 4806 (Rajan)).

D-156. The proposed modification for the model D2-D3 involves installation of an internal manifold which involves a rather complex procedure. None of these changes have been proposed for the model D4-D5 where the modification is relatively simple. Id.

D-157. The 100 tubes identified as candidates for expansion at Byron are regarded as a bounding number. (Tr. 6038-39 (Timmons)). These tubes have been determined to be the most susceptible to tube vibration. (Tr. 6209 (Timmons); Tr. 4767-68 (Rajan)).

D-158. The precise location and number of tubes to be expanded is an engineering design detail to be ascertained. (Tr. 6055, 6240, 6306 (Timmons)).

D-159. The feedwater bypass modification at Byron will require that the present feedwater preheater bypass valve remain open during high main feed flows. (Timmons testimony, ff. Tr. 5908, at 25).

D-160. This will result in approximately 90% of the feedwater flow entering the main feedwater nozzle and the remainder of the feedwater flow entering the steam generator through the auxiliary feedwater nozzle. Id.

D-161. There is nothing new or unique about the diversion of feedwater flow. (Tr. 6269-70 (Green); Tr. 6270 (Fletcher); Tr. 6340-41 (Rajan)). It has been a feature of the feedwater system for purposes of startup for many years. (Tr. 6269-70 (Green)). The only unique aspect is the plan to do it during normal full power operations. Id.

D-162. There is no need for a structural modification to the feedwater bypass system in order to accommodate a 90/10 flow split. (Tr. 6204-05 (Butterfield); Tr. 6227 (Timmons)). To implement a 70/30 flow split, such as employed at Krsko, it would probably be necessary to install new piping, some kind of a flow restrictor or flow resistant device in the main feedline. (Tr. 6227 (Timmons)).

D-163. Westinghouse has tested the proposed Byron modification in the 16° model and at Krsko. (Timmons testimony, ff. Tr. 5908, at 25).

D-164. In the 16° model, a number of tubes were expanded and tested to determine the effect of tube expansion on tube vibration. (Timmons testimony, ff. Tr. 5908, at 25-26).

D-165. At a flow rate equivalent to 90% of the Byron main feed flow rate, the expanded tubes exhibited vibration levels that were less than those observed at flow rates equivalent to 70% of the Byron main feed flow rate without tube expansion. Id. A 70% main feed flow rate will not result in significant tube wear. Id.

D-166. In addition to testing in the 16° model, one tube of the Krsko plant that had been previously instrumented was expanded at baffle plate locations. Id.

D-167. Previous tube vibration data was compared with the data obtained after the tube had been expanded and it was concluded that tube vibrations were reduced by at least the factor of 5 from the nonexpanded case. This reduction resulted in a negligible level of vibration for that tube. Id.

D-168. The actual diminution in tube wear of the expanded Krsko tube was described generally in the in camera steam generator session. (Tr. 6167, 6170 (Timmons)).

D-169. This effort has enabled Westinghouse to establish a wear correlation ratio from vibration data from the expanded tube at Krsko and vibration data from expanded tubes in the 16° model. (Tr. 5992 (Timmons)).

D-170. The data from the 16° model correlates exceptionally well with the data from the Krsko plant, both for expanded tubes and for nonexpanded tubes. (Timmons, Tr. 5992-93, 5933, 6063-64).

D-171. Based on this ability to replicate plant data with the test model data, the information can be used from the test model to predict the future behavior of the steam generator. (Tr. 5993 (Timmons)).

D-172. For the tubes most susceptible to vibration, the vibration levels following the proposed Byron modification, entailing tube expansion and a 90/10 flow split, could be significantly less than those vibrational levels that would exist at Byron for the same tubes at a 70/30 flow split with no tube expansion. (Tr. 6230 (Timmons)).

D-173. Westinghouse intends to install accelerometers at the first plant at which the proposed modification is to be implemented in order to confirm that the vibrations seen in that plant are representative of those seen in the test models and at Krsko. (Tr. 6058 (Timmons)). That

plant is expected to be Comanche Peak Unit 1. (Tr. 6058-59, 6066 (Timmons)).

D-174. Westinghouse has considered the possible effects of the proposed tube vibration modification on water hammer and had determined that it does not adversely effect the capability of the system to operate without significant water hammer. (Tr. 6205 (Timmons)). Sargent and Lundy witness Mr. Green agreed. (Tr. 6213-14 (Green)).

D-175. It is uncontroverted on the record that the proposed modification should be successful in eliminating excessive flow induced tube vibration and related wear. (Timmons testimony, ff. Tr. 5908, at 26, Tr. 6000; Fletcher testimony, ff. Tr. 5908, at 10, Tr. 6262; Butterfield testimony, ff. Tr. 5908, at 5; Rajan testimony, ff. Tr. 4473, at 5; Tr. 6507 (Bridenbaugh)).

D-176. The Applicant intends to install the necessary steam generator modifications at the earliest opportunity after engineering, testing, review and approval are completed. (Blomgren testimony, ff. Tr. 4126, at 17).

D-177. The Applicant will completed the necessary modifications (id.) and these modifications will be reviewed by the Staff (Rajan testimony, ff. Tr. 4473, at 9) prior to Byron operation.

D-178. Considerable hearing time was devoted to the status and progress of the Staff review of the proposed modification. The Staff is familiar with the engineering effort underway at Westinghouse to qualify these modifications. (Rajan testimony, ff. Tr. 4473, at 5). The Staff has been actively involved in the tube vibration matter since the inception of the Westinghouse generic model D program and has been provided

on a regular basis with pertinent information from Westinghouse, both at periodic meetings and in written status submissions. (Tr. 4681-2, 4684 (Rajan)).

D-179. The formal report describing the qualification test of the present modification and analyses to justify its adequacy has yet to be submitted for Staff review. Given its briefing on the subject, the Staff does not expect to see anything new in the formal submission. (Rajan testimony, ff. Tr. 4473, at 5, Tr. 4682).

D-180. Dr. Rajan testified that the Staff review of the Westinghouse model D4 tube vibration matter has been one of the most detailed he has observed in his 9 or 10 years with the NRC. (Tr. 4682, 6339-41 (Rajan)).

D-181. The Staff has retained the Argonne National Laboratory as consultants in this matter. (Tr. 4635 (Rajan)). Argonne National Laboratory is a quasi-private laboratory with a large body of scientific personnel who have been engaged in vibration problems for a number of years. (Tr. 6327 (Rajan)). They have scientists in this area who have national prominence, who have published extensively in this area and command a great respect both in the academic and industrial areas. Id. One of the principal investigators involved in the model D4, D5 program is nationally known for his contributions in flow induced vibrations. (Tr. 6327-28 (Rajan)).

D-182. The Argonne review has been coextensive with the Staff review, and they have been furnished all pertinent Westinghouse information submitted to the Staff. (Tr. 6330-37 (Rajan)).

D-183. Based on the Staff preliminary review of the proposed modification, it concluded that the objective of minimizing tube degradation associated with flow induced vibration will be accomplished by these modifications. (Rajan testimony, ff. Tr. 4473, at 5; Tr. 4639, 4674-75 (Rajan)).

D-184. In arriving at this position, the Staff relied on advice received from the Argonne National Laboratory to the effect that the vibration levels that could be expected with the expanded tubes and with the reduced flow through the main feed nozzle would result in tube wear which would not reach 40% over a 40 year life of the plant. (Tr. 6328 (Rajan)).

D-185. The detailed Staff review of the proposed modification will be completed prior to plant operation (Rajan testimony, ff. Tr. 4473, at 5) and a safety evaluation report issued following its implementation at Byron. (Tr. 4637, 4678 (Rajan)).

e. Domestic steam generator tube rupture events

D-186. There have been four instances of domestic tube ruptures involving leakage (between 80 and 700 gpm). (Frank testimony, ff. Tr. 4473, at 8). Two of these events were as a consequence of corrosion. (McCracken testimony, ff. Tr. 4473, at 5). A 125 gpm rupture occurred at Point Beach in 1975 and a 50-80 gpm rupture occurred at Surry in 1976. Id. It has been over 6 years since a steam generator tube rupture due to corrosion has occurred. Id.

D-187. The other two events (Prairie Island 1 in 1979 and Ginna in 1982) are attributable to damage caused by foreign objects and loose parts as discussed above. See Findings D-105 to D-107.

D-188. The rupture at Point Beach was caused by secondary side intergranular stress corrosion cracking which occurred as a consequence of reactions between condenser inleakage impurities and residual phosphates. Id.

D-189. Byron will use all volatile chemistry treatment; consequently, the chemical reactions which caused the Point Beach rupture cannot occur at Byron. Id. Since the industry conversion to AVT in 1974, no plant which has started up on AVT has detected secondary side initiated stress corrosion cracking. Id.

D-190. The rupture at Surry was initiated from the primary side of the tube and caused by excessive tube stress. (McCracken testimony, ff. Tr. 4473, at 5; Tr. 4779-80 (McCracken)). The excessive tube stress resulted from extensive tube denting which first froze the tube in place and then physically moved the tube support plates, resulting in a significant deformation of the tube and resultant high stress. (McCracken testimony, ff. Tr. 4473, at 5).

D-191. The water chemistry control requirements at Byron, in conjunction with inservice inspection requirements, will combine to make it highly unlikely that extensive denting will occur. (McCracken testimony, ff. Tr. 4473, at 5-6).

D-192. The design modifications and procedures incorporated will also act to reduce the probability of tube leakage. However, it can be anticipated that some instances of steam generator tube leakage will

occur during the plant lifetime. The inservice inspection program, combined with tube plugging criteria, will minimize these occurrences. Id.

D-193. The Staff has evaluated the systems performance, operator actions and radiological consequences of the Point Beach, Surry, and Prairie Island steam generator tube ruptures in NUREG-0651. (Tr. 4801-02 (Marsh)).

D-194. The system performance and operator actions were as expected. The operators performed in an expeditious and beneficial manner in those accidents. Id.

D-195. The radiological consequences were found to be very, very low, far less than the design basis single steam generator tube rupture accident analyzed in the FSAR. Id.

D-196. The systems performance, operator actions and radiological consequences for the Ginna event have been evaluated in two documents, NUREG-0909 and 0916. Id.

D-197. The overall systems performance, operator actions, and radiological consequences were as expected. Id. However, there were some system performance aspects that were described in those documents which the Staff looked at and incorporated into the ongoing generic assessment of steam generator tube integrity. Id.

f. Accident considerations.

D-198. Mr. Bridenbaugh testified that there is an increased probability that accidents will be initiated by tube failures during normal operation and an increased liklihood that accidents not now

considered in the safety analysis may occur as a result of the steam generator tubes degradation after some period of operation. The accident sequence could involve single or multiple tube failures occuring in conjunction with other accident sequences. (Bridenbaugh testimony, ff. Tr. 6406, at 5).

D-199. Mr. Bridenbaugh, acknowledged that he performed no independent analysis or calculations to quantify the increased probability of steam generator tube failures (Tr. 6475) nor had he performed any independent calculations or analyses to ascertain either the probability or radiological consequences of multiple steam generator tube ruptures and concurrent accidents. (Tr. 6476 (Bridenbaugh)).

D-200. The Applicant has analyzed the systems performance and offsite consequences from the full severance of a single steam generator tube both with and without offsite power. (Marsh testimony, ff. Tr. 4473, at 2). In either situation, the resultant radiological consequences would not exceed the allowable exposure limit set forth in 10 CFR § 100.11. Id.

D-201. Those accidents which result in the largest differential pressure between primary and secondary sides of steam generator tubes have the greatest potential for affecting steam generator tube integrity. This occurs during three extremely low probability events: the large break LOCA, the large main steam line break (MSLB) and large main feed line break (MFLB) accidents. (Marsh, Rajan testimony, ff. Tr. 4473 at 2).

D-202. If any of these extremely low probability accidents occurred, there would probably be no significant effect upon steam generator tube integrity. (Rajan, Frank testimony, ff. Tr. 4473, at 3).

D-203. Steam generator tubes are designed to withstand the differential pressures likely to be experienced during the most severe LOCA, main steam line break or main feed line break accidents. Id.

D-204. The tube plugging criteria requires that a reasonable factor of safety still exists when 70%-75% of the steam generator tube wall has been lost. Id.

D-205. Technical specification leak rate limits, which require plant shutdown when small leaks develop, the routine service and inspection program conducted to detect tubes with insufficient wall thickness, and an improved water chemistry program will serve to maintain steam generator tube integrity. Id.

D-206. The incremental effect of the steam generator tube rupture upon a large MSLB, MFLB and LOCA have been considered by the Staff. (Marsh testimony, ff. Tr. 4473, at 4).

D-207. The consequences of a large MSLB inside containment could be adversely affected by such an event. However, calculations have shown that containment integrity is not effected, the core always remained covered and cooled due to the addition of emergency core cooling, and there is ample water supply available for long term cooling. Id.

D-208. Calculations have been performed to evaluate the systems performance, offsite consequences and required operator actions assuming a steam generator tube rupture concurrent with an MSLB outside containment. These studies evaluate the effects of a main steam line break combined with one or five ruptured steam generator tubes in a small break LOCA. Id.

D-209. The results of these analyses indicate that primary coolant shrinkage, caused by overcooling, and the simultaneous loss of primary coolant, can be compensated by the high pressure emergency core cooling system. The core remains covered, and the primary coolant remains cool, except in the vessel upperhead. The calculations and results are described in NUREG-0937. Id.

D-210. As part of the technical resolution of the steam generator tube integrity unresolved safety issue, the Staff assessed the consequences of single and multiple tube breaks (in a single steam generator) concurrent with a large main steam line break or large cold leg break LOCA. (Marsh testimony, ff. Tr. 4473 at 5).

D-211. One of the main purposes of this effort is to develop a statistically based inservice inspection program that affords a high degree of assurance that if a large MSLB or LOCA occurred concurrent with ruptured steam generator tubes in the effected steam generator, the offsite dose and fuel temperatures are acceptable. Id.

D-212. The consequences of a large, cold leg LOCA could be adversely affected by the flow of steam generator fluid into the primary loop through the broken steam generator tubes. Several computer studies performed over the years indicated the following: (1) rupture of a few tubes during a LOCA would have very little effect, (2) if a large number of tubes ruptured, the additional fluid flow into the reactor vessel during a large break would actually aid in cooling the core, and (3) an optimum number of tubes over a limited range (about 12) could have a detrimental effect. However, it is not expected to lead to a core meltdown. Id.

D-213. A series of LOCA experiments with varying degrees of simulated tube failures performed in the semi-scale facility at the Idaho National Engineering Laboratory several years ago confirmed this general behavior. (Marsh testimony, ff. Tr. 4473, at 6.) These experiments did not show the same degree of degraded core cooling as the computer analysis did for the worst cases. In fact, the experiments did not indicate that any core damage would occur. Id.

D-214. The overall consequences of a large MFLB with simultaneous steam generator tube rupture were bounded by the combined MSLB and tube rupture inside or outside containment. (Marsh testimony, ff. Tr. 4473 at 6; Tr. 4784-85 (Marsh)).

D-215. The Staff does not postulate one of these events combined with the steam generator tube rupture as a design basis event as it does not believe they pose an undue risk to public health and safety. (Marsh, Frank, Rajan testimony, ff. Tr. 4473, 6-8).

D-216. The MSLB, MFLB and cold and large cold leg break LOCA accidents are extremely low probability events on the order of 10^{-5} to 10^{-6} per reactor year. (Marsh testimony, ff. Tr. 4473, at 7; Tr. 4723 (Marsh)).

D-217. The steam generator tube rupture event, while not as infrequent as the LOCA, MSLB or MFLB accidents, is also an infrequent event on the order of 10^{-2} or 10^{-3} per reactor year. (Marsh testimony, ff. Tr. 4473, at 7; Tr. 4724 (Marsh)). In the opinion of the Staff, taken independently, or together, the likelihood of a tube rupture concurrent with a LOCA, MFLB or MSLB is extremely low. (Marsh testimony, ff. Tr. 4473, at 7).

D-218. The Applicant performed an analysis whereby it predicted that tube rupture events in combination with accidents are predicted to result in severe core damage at frequencies of 10^{-7} per year for the Byron station. (Hitchler testimony, ff. Tr. 5908, at 8; Tr. 6231 (Hitchler)). By this calculation, a single tube rupture would occur about once every 33 years at Byron. (Tr. 6235 (Hitchler)).

D-219. The steam generator tubes are designed such that MSLBs or LOCAs would not cause tube ruptures. (Frank, Rajan testimony, ff. Tr. 4473, at 7). Tube wall thickness and internal support arrangements are such that, even with some tube wall degradation, MSLB, MFLB or LOCA will not result in breakage of a steam generator tube. Id.

D-220. To insure tubes are not degraded to the point where rupture is possible, the Staff requires licensees to shut down and repair steam generator tubes should the primary to secondary leakage exceed a maximum allowed by technical specification. Id. The Staff further requires licensees to routinely inspect steam generator tubes and to plug those whose wall thickness is degraded. As a further means of reducing the rate of steam generator tube corrosion, water chemistry programs are implemented to minimize the contaminants in the steam generator. Id.

D-221. As part of the Staff's ongoing evaluation of the four domestic steam generator tube ruptures and steam generator tube degradation in general, a number of specific requirements are being considered. (Marsh, Frank testimony, ff. Tr. 4473, at 7).

D-222. Twelve potential requirements are presently undergoing cost-benefit assessment by the Staff and its consultant. Id. The consultant cost-benefit assessment is contained in a final draft report,

entitled "Value Impact Analysis of Recommendations Concerning Steam Generator Tube Degradation and Rupture Events" (marked for identification as Intervenors' Exhibit 9 at Tr. 4443). These items are under consideration as potential Staff requirements. (Tr. 4573 (Marsh)). The Staff has continued to evaluate these recommendations since the issuance of the consultant report, and has modified, altered and changed the majority of the recommendations. (Tr. 4502 (McCracken)). A number of other items are also being considered as Staff actions. (Marsh, Frank testimony, ff. Tr. 4473, at 7-8; Tr. 4572, 4574 (Marsh)).

D-223. The actions under consideration are aimed at improving and gaining understanding of the overall problem of steam generator tube degradation, not to extend the licensing basis. (Marsh, Frank testimony, ff. Tr. 4473, at 8; Tr. 4786 (Marsh)).

D-224. As a result of the TMI accident, TMI action plan 1.C.1 requires the industry to upgrade emergency operating guidelines and procedures to cover multiple failure events which fall outside the required design envelope assumptions for safety analyses. (Marsh testimony, ff. Tr. 4473, at 8).

D-225. These events were not analyzed in order to show conformance with existing regulations. The idea was to develop emergency operating procedures that go well beyond the design basis accidents in the remotest possibility that they could occur. (Tr. 4786 (Marsh)).

D-226. In this context, the Staff and vendors are analyzing a variety of such events, including coincident steam generator tube ruptures and LOCAs and coincident steam generator tube rupture and steam

line breaks. The results of a recent Staff analysis are discussed in NUREG-0937. Id.

D-227. After Three Mile Island, and the concern for improved operating procedures, the Westinghouse Owners Group undertook a generic development of guidelines to cover all emergency operating procedures. These generic guidelines have been submitted to the NRC for approval. (Tr. 6234 (Butterfield)).

D-228. The Westinghouse generic emergency response guidelines have been used as a basis for the development of the Byron operating procedures. (Tr. 6143-44 (Butterfield)). Emergency operating procedures will enable the Byron operators to respond to the various compound accidents discussed at the hearing (Tr. 6233 (Butterfield)) and to make multiple tube ruptures less probable (Tr. 6508 (Bridenbaugh)).

D-229. The Applicant also has an operator training program using new simulators. All encompassing emergency operating procedures and additional training criteria provide a total coordinated effort to address all postulated transient and accidents. Id.

D-230. The Board finds that single and/or multiple tube rupture events need not be analyzed along with other design basis events for licensing review purposes based on their low probability and controllable consequences. This finding takes into account the development of a range of emergency operating procedures to enable operators to respond to such events should they occur.

g. Resolution status of USI A-3

D-231. It is the Staff position that the Westinghouse steam generator tube degradation unresolved safety issue is not of sufficient safety concern to warrant a licensing delay of new PWR facilities. (Frank testimony, ff. Tr. 4473, at 7).

D-232. Operating experience has demonstrated that current regulatory requirements have been generally successful in maintaining acceptable structural margins against tube ruptures. Id. When new or unanticipated degradation problems have occurred, these problems have been revealed either during routine inservice inspection or as a result of leaks and appropriate action has been taken at that time. Id. In some cases, the action has included additional inspection requirements or operational limitations imposed by the Staff. Id.

D-233. Ongoing Staff studies as part of the unresolved safety issue may result in additional requirements relative to inservice inspections of the steam generators to provide added assurance of tube integrity. (Frank testimony, ff. Tr. 4473, at 8-9; Tr. 4477-/9 (Marsh)). Implementation of any additional recommendations is predicated on review by the Committee for the Review of Generic Requirements, Advisory Committee on Reactor Safeguards and the Commission itself. It may be done on a generic or plant specific schedule. (Tr. 4477-79 (Marsh)).

D-234. A primary objective of the steam generator USI program is to ensure that tubes are plugged before they corrode to a significant degree that they have potential for rupturing in subsequent operation. (Tr. 4714 (McCracken)). The objective of the program is to control degradation to enable the maintenance of tube wall integrity under both

operating and accident conditions. (Tr. 4715 (McCracken); Tr. 4715 (Frank)).

D-235. Pending the conclusion of these studies, the Staff has concluded that the current surveillance requirements will provide reasonable assurance that operation of the Byron steam generators will not adversely affect public health and safety. (Frank testimony, ff. Tr. 4473, at 9).

D-236. Despite the retention of USI A-3 an an unresolved safety issue, the dimensions of the problem have been considerably reduced. (Tr. 4799 (McCracken)).

D-237. A graph prepared by EPRI, and endorsed by the Staff, shows the relative number of tubes plugged verses the total number of tubes in service between 1972 and 1980. (Tr. 4797 (McCracken)). This graph shows the number of tubes that were plugged due to phosphate wastage or thinning, denting, and other related problems. The graph demonstrates that 1977 was the height of its denting problem. This is when the Surry steam generators and Turkey Point steam generators became a problem. (Tr. 4797-98 (McCracken)).

D-238. Once that problem had been identified in 1976 and 1977, after the shift to all volatile chemistry, the industry became aware of what it took to control denting, and a dramatic decrease in the amount of denting and other corrosion in all units was seen. Id. This specific trend has continued to the present.

D-239. Things were continuing to get better and the industry was, in fact, doing what it needed to do to resolve steam generator corrosion problems. (Tr. 4798 (McCracken)). Mr. Bridenbaugh agreed that a lot of

manpower has been spent on the issue, that progress has been made, and that there is a better understanding of the technical issues. (Tr. 6478). Mr. Bridenbaugh testified that he would just like to see it get formalized and wrapped up into a requirement that is enforced in the field. Id. He believed that the Staff technical analysis had progressed to the point where a preliminary or proposed solution is in hand. (Tr. 6478-79 (Bridenbaugh)).

D-240. The Staff expects its report documenting the official resolution of USI A-3 in approximately July 1983. (Tr. 4799 (Marsh)).

D-241. The Board finds nothing in the testimony of Mr. Bridenbaugh to demonstrate that the steam generator tube degradation phenomena at issue herein has not been adequately addressed and resolved as necessary to eliminate it as a significant public health and safety problem pending formal NRC resolution of USI A-3.

D-242. His testimony does not shed any new light on the status or extent of the present steam generator tube degradation problem. His prepared testimony is essentially a compilation of industry and government references describing the historical evolution of the steam generator tube integrity issue and past and present recommendations and measures to address it. It is not the product of any independent research or analysis of the tube integrity phenomenon. (Tr. 6474 (Bridenbaugh)).

D-243. The Board finds that there is adequate technical justification to authorize Byron full power licensing pending the official resolution of USI A-3.

IV. CONCLUSIONS OF LAW

Based on the entire evidentiary record of this proceeding, and upon the foregoing findings of fact, the Board concludes the following:

C. Contentions 42, 111 and 112 on the subject of occupational radiation exposure and Applicant's compliance with regulatory requirements concerning radiation safety are without merit. The estimates presented in the FES of the health effects from occupational radiation exposure are reasonable and conservative. There is reasonable assurance that no undue risk will be provided to workers at the Byron Station from occupational exposure to radiation and that such exposure will be maintained as low as is reasonably achievable; Edison's security measures are adequate to ensure that the use of temporary or contract workers at Byron Station does not pose an unreasonable risk to public health and safety.

D. Contrary to League Contention 22 and DAARE/SAFE Contention 9(c), steam generator tube integrity has been adequately dealt with by the Applicant and Staff in the context of the Byron license application and does not present a serious safety problem under either normal operation or accident conditions. Westinghouse steam generator tube integrity unresolved safety issue A-3 has been virtually resolved by the NRC Staff and is awaiting formal documentation.

V. ORDER

WHEREFORE, in accordance with the Atomic Energy Act of 1954, as amended, and the rules and practice of the Commission, and based on the foregoing findings of fact and conclusions of law, IT IS ORDERED THAT this Supplemental Partial Initial Decision shall constitute a portion of

the ultimate initial decision to be issued upon resolution of the remaining contested issues in this proceeding.

IT IS FURTHER ORDERED, in accordance with 10 CFR §§ 2.760, 2.762, 2.764, 2.785, and 2.786 that this Partial Initial Decision shall become effective and shall constitute, with respect to the matters addressed herein, the final decision of the Commission 30 days after the date of issuance hereof, subject to any review pursuant to the above cited rules of practice. Exceptions to this Decision may be filed within ten (10) days after service of this Partial Initial Decision. A brief in support of such exception may be filed within thirty (30) days thereafter, forty (40) days in the case of the Staff. Within thirty (30) days after service of the brief of Appellant, forty (40) days in the case of the Staff, any other party may file a brief in support of, or in opposition, to such exceptions.

THE ATOMIC SAFETY AND LICENSING BOARD

Ivan W. Smith, Chairman

Dr. Richard F. Cole, Member

A. Dixon Callihan, Member

Dated at Bethesda, Maryland
this day of 1983

Respectfully submitted,

Steven C. Goldberg
Counsel for NRC Staff

Richard J. Rawson
Counsel for NRC Staff

Dated at Bethesda, Maryland
this 14th day of June, 1983

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
COMMONWEALTH EDISON COMPANY	)	Docket Nos. 50-454
	)	50-455
(Byron Station, Units 1 and 2)	)	

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW IN THE FORM OF A SUPPLEMENTAL PARTIAL INITIAL DECISION ON OCCUPATIONAL RADIATION SAFETY AND STEAM GENERATOR ISSUES" in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class, or, as indicated by an asterisk, by deposit in the Nuclear Regulatory Commission's internal mail system, this 14th day of June, 1983:

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