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Docket No. 50-346

The Honorable Dennyson Guyer
United House of Representatives
Washington, D. C. 20515

Dear Congressman Guyer:

Your letter dated October 21, 1976, to Mr. Kenneth R. Chapman, Director of the Office of Nuclear Material Safety and Safeguards, has been referred to this office for reply. In your letter, you request that we consider the matters contained in a letter from Linda Carr and that we provide you with relevant information regarding the safety measures incorporated into nuclear power plants. Ms. Carr indicated in her letter that her interest in nuclear safety matters arose from a recent announcement regarding the proposed addition of two nuclear units at the site of the Davis-Besse nuclear power plant which is presently being constructed.

I would like to address each of the specific concerns identified by Ms. Carr but first I would like to briefly discuss the Davis-Besse facility. This nuclear power plant is located in north central Ohio on the southwestern shore of Lake Erie in Ottawa County, Ohio, about 21 miles east of Toledo, Ohio. The facility is being constructed under Construction Permit No. CPPR-80 issued by the Nuclear Regulatory Commission (NRC) on March 24, 1971. The site consists of about 900 acres, most of which is flat and is unused marshland. The facility, identified as Davis-Besse 1, incorporates a two-loop pressurized water nuclear reactor for its steam supply system which feeds the turbine generator. The net electrical output is about 906 megawatts (MWe). Davis-Besse 1 will be jointly owned by the Toledo Edison Company and the Cleveland Electric Illuminating Company. A more detailed description of the Davis-Besse facility and the site is contained in the Final Safety Analysis Report (FSAR) submitted on March 31, 1973, by the permittees identified above. A copy of the FSAR and all other documents pertaining to Davis-Besse 1 are available for inspection by members of the public at the Ida Rupp Public Library, 310 Madison Street, Port Clinton, Ohio, and at the NRC's Public Document Room located at 1717 H Street, N.W., Washington, D. C.

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Ms. Carr identifies eight concerns in her letter and inquires whether some additional action regarding safety measures can be taken. Since my response to her concerns and her question on safety measures is extensive, I have written it as an enclosure to this letter. I believe that the discussion of the issues in the enclosure is responsive to your request.

I have enclosed a copy of your letter requesting information and a copy of your constituent's letter.

Sincerely,

(Signed) Lee V. Gossick

Lee V. Gossick
Executive Director
for Operations

Enclosures:
1. Discussion of the Concerns of Linda Carr
2. Incoming ltr from Congressman Guyer
3. Incoming ltr from Linda Carr

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LWR 1r
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AD/LWR
DBVassallo
11/17/76

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United States House of Representatives
Washington, D. C. 20515

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ENCLOSURE

DISCUSSION OF THE CONCERNS REGARDING THE SAFETY OF NUCLEAR POWER PLANTS

A number of concerns regarding the safety of nuclear power plants are identified in Ms. Carr's letter. These are: (1) training of nuclear power plant personnel; (2) construction procedures for nuclear power plants; (3) inspection of nuclear power plants; (4) effectiveness of safety systems; (5) statements of three General Electric engineers regarding nuclear safety; (6) "accidents" in 1974; (7) dispersal of fission products following postulated accidents; and (8) the effect of nuclear power plants on public health. Ms. Carr then inquires whether safety measures can be improved in view of her stated concerns about the safety of nuclear power plants. Each of the eight concerns stated above are briefly discussed in this enclosure and, where appropriate, references are cited where a more detailed discussion can be found. Finally, the safety measures required by the Nuclear Regulatory Commission (NRC) for nuclear power plants are discussed.

1. Training of Nuclear Power Personnel

In Ms. Carr's letter, the training of personnel is described as "poor." While she does not specify whether this allegation is with respect to the design and construction phase or the operating phase, the latter will be discussed in this section of the enclosure. The training of construction personnel will be discussed in the following sections of this enclosure.

Recognizing the need for trained, capable, and experienced operating personnel for nuclear power plants, the NRC has issued regulations requiring that applicants for operating licenses submit information concerning their organizational structure and operating personnel qualifications. This information is reviewed in detail by the NRC and must be found acceptable prior to issuance of an operating license. To provide guidance to applicants on this matter, the NRC has endorsed in Regulatory Guide 1.8, "Personnel Selection and Training," (Attachment 1), the criteria for the selection and training of nuclear power plant personnel contained in ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel." (See footnote 1) The ANSI standard cited above is one of many such standards published by the American National Standards Institute.

Footnote 1. Copies may be obtained from the American Nuclear Society, 244 East Ogden Avenue, Hinsdale, Illinois 60521.

Since this ANSI standard on the selection and training of operating personnel is quite extensive, a summary of its salient features is provided. The standard establishes the minimum qualifications and training for all functional levels of the operating organization, including managers, supervisors, professional-technical, and operator-technicians-repairmen. A table is included (Attachment 2) which summarizes the required experience, academic training, and the NRC operator licensing requirements for all functional levels of operating personnel. These requirements are designed to provide assurance that the nuclear power plant operating personnel: (1) will be capable of safely and efficiently operating the facility; (2) will understand the complexities of the plant design; (3) will be capable of properly manipulating the plant controls; and (4) will maintain and repair the plant equipment in an acceptable manner.

Before any operating personnel can manipulate the controls of an operating nuclear power plant, they must obtain a license from the NRC which authorizes them to operate a specific nuclear power plant. The requirements for obtaining an operator's license are contained in 10 CFR Part 55, "Operator's Licenses" (Attachment 3). Section 55.11 of Part 55 states in part that an applicant for an operator's license will be approved if the NRC finds that the individual is in good health and has passed a written examination and an operating test to determine that the applicant has learned to operate a specific nuclear power plant.

The scope of the senior operator written examination is described in Sections 55.21 and 55.22 of 10 CFR Part 55 and covers 21 different aspects of reactor operation, including: (1) fundamentals of reactor theory; (2) general design features; (3) general operating characteristics; (4) conditions and limitations in the facility license; and (5) fuel handling facilities. While this list is not complete, it does indicate that the operating personnel are required to demonstrate a broad, in-depth knowledge of all aspects of an operating nuclear power plant. Additionally, the operators of a facility are required to demonstrate an understanding of 12 separate matters, including: (1) the required manipulation of the console controls; (2) the use and function of the facility's radiation monitoring systems; and (3) the emergency plans for the facility.

In summary, the operating personnel of nuclear power plants are carefully selected, intensively trained in a broad range of nuclear power plant operations, and carefully retested periodically by the NRC to determine that they can safely operate the facility for which they are licensed.

2. Construction Procedures for Nuclear Power Plants

Ms. Carr indicates in her letter that she considers the construction of nuclear power plants to be faulty. In response to this statement, a description of the method by which construction procedures are selected and approved for nuclear power plants is provided. In the following section on the inspection process, a discussion on how assurance is provided that these construction procedures are followed, is also provided.

In the two-stage licensing process for a nuclear power plant, the review conducted by the NRC staff at the construction permit stage is directly concerned with the acceptability of the "... proposed design of the facility, including, but not limited to, the principal architectural and engineering criteria for the design..." (See footnote 2). The minimum requirements for the principal design criteria for water-cooled nuclear power plants are contained in Appendix A, "General Design Criteria for Nuclear Power Plants," of 10 CFR Part 50 (Attachment 4).

Some of the more important and essential elements of the principal design criteria for a nuclear power plant are the proposed construction procedures. Many of these procedures are contained and codified in well-known design and construction codes which have been published and adopted by national societies and institutes. A few examples of these codes are: (1) the ASME Boiler and Pressure Vessel Code published by the American Society of Mechanical Engineers; (2) the ACI codes published by the American Concrete Institute; (3) the AISC Code for Steel Structures published by the American Institute for Steel Construction; and (4) the codes and standards of the Institute of Electrical and Electronic Engineers. This listing illustrates that there are widely used standards and codes which can be adopted to provide assurances that acceptable construction procedures are followed during the construction phase of nuclear power plants. These national standards have been used in many different types of construction projects for many years and have provided acceptable protection of the public health and safety.

Footnote 2. Refer to Section 50.35(a)(1) of 10 CFR Part 50.

In addition to the national standards discussed above, the NRC has established additional guidance for applicants for licenses to construct and operate nuclear power plants. This guidance is contained in the NRC Regulatory Guides (Attachment 5) which are issued to either supplement the national standards or to provide guidance for acceptable design criteria when the national standards are not sufficiently conservative for certain aspects of the construction of nuclear power plants.

Where possible, the NRC staff issues design criteria for specific systems or components of nuclear power plants. These specific design criteria are contained in Branch Technical Positions and reflect the knowledge obtained from the review of previous applications for licenses and from the operating experience of licensed nuclear power plants.

Finally, the staff establishes ad hoc design criteria for the construction of nuclear power plants when the existing national standards, regulatory guides and Branch Technical Positions do not address unique design features of a proposed facility.

3. Inspection of Nuclear Power Plants

One of the concerns of Ms. Carr is with respect to the adequacy of inspection of nuclear power plants. This is a very important element in the safety of nuclear power plants since the adequacy of construction and the safety of operation of nuclear power plants depends directly on the quality of the inspection process. Recognizing this importance, the NRC has issued regulations requiring the establishment of a "... Quality Assurance (QA) program to be applied to the design, fabrication, construction, and testing of the structures, systems, and components of the facility)." (See footnote 3) The requirements of this QA program are contained in Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.

The NRC staff reviews both the proposed QA program of applicants and the QA organization which will implement the QA program. Prior to authorizing construction of the safety related structures, systems or components of a nuclear power plant, the staff must find both the applicant's QA program and QA organization acceptable. However, the NRC staff's involvement does not end here. The Office of Inspection and Enforcement (OIE) periodically conducts both scheduled and unannounced field inspections of the implementation of the applicant's QA program and those of his contractors and suppliers. This field

Footnote 3. Section 50.34(a)(7) of 10 CFR Part 50.

inspection starts about nine months prior to the submittal of an application for a construction permit and continues throughout the construction phase, the preoperational test program and the operating lifetime of the facility. These field inspections by OIE during the construction phase are extensive and cover: (1) a review of the applicant's QA performance, including audits of the applicant's QA records and documentation; (2) a witnessing of the construction practices and an inspection of the facility at various stages of construction; and (3) a review of the qualifications and training of the construction personnel as well as those of the quality assurance and quality control (QA/QC) personnel. The review of the qualifications and training of the QA/QC personnel is conducted for all construction personnel at the site, including the specialized subcontractors, and at the manufacturing facilities of the vendors and suppliers.

The NRC QA review can best be visualized as a multiple tier process involving the entire QA program. To grossly simplify the process, QC personnel for each subcontractor and vendor review the quality of the design and fabrication. In turn the QA personnel of the subcontractors and vendors review the adequacy of the QC review. The applicant's QA personnel then review the adequacy of the QA review by the subcontractors, vendors, and that of its primary contractor. Finally, OIE reviews the implementation of the entire QA/QC program by all parties in the design and construction phases. While any inspection process is potentially fallible due to the human element, the QA programs for nuclear power plants described above minimizes the possibility that gross negligence and/or incompetence would cause nuclear power plants to be constructed with significant deviations from the design criteria for these facilities.

There have been a number of nuclear power plants which have been designed or constructed in a manner which represented significant departures from the design criteria for these plants. The installation of the safety-related electrical cables in the Davis-Besse 1 facility is an example of this. However, the faulty construction procedures in this instance were discovered by OIE personnel implementing the QA/QC program in the manner described above, at least two years prior to any decision to issue an operating license for this facility.

Recognizing the inherent limitations of any inspection process, the NRC does not assume that all deficiencies in either the design or installation of the safety-related structures, systems or components will be found by the QA program. Accordingly, the NRC requires applicants to conduct an extensive preoperational test program for each system and component. For example, diesel generators for the

on-site power supply of a nuclear power plant must undergo hundreds of cold startup tests to demonstrate the adequacy of their design and installation and certain important safety-related structures are proof tested to demonstrate their adequacy.

While the foregoing discussion is primarily oriented towards the design and construction phase, a similar inspection program is conducted throughout the operating lifetime of a nuclear power plant, including visual inspections, design reviews and periodic testing.

In summary, the inspection process for nuclear power plants is an extensive, in-depth, continuing program whose objective is to minimize human error which might adversely affect the design, construction and operation of these facilities. Its success can only be gauged by the operating history of the licensed operating plants. This matter is discussed in Item 9 of this enclosure.

4. Effectiveness of Safety Systems

Ms. Carr alleges in her letter that nuclear power plants have unproven safety systems. The response to this concern involves two principal considerations. The first of these is the design philosophy for safety-related systems while the second is concerned with experimental verification and testing. The preceding discussion in Item 2 described the review and acceptance of the design criteria for nuclear power plants. This review includes a determination by the NRC of "... the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents." (See footnote 4) The review of the emergency core cooling systems (ECCS) of water cooled reactors is an example of one of these systems which are analyzed and evaluated by the NRC. Assuming a loss of coolant accident (LOCA), the proposed ECCS is analyzed in accordance with very specific requirements contained in Section 50.46 and Appendix K of 10 CFR Part 50. These design criteria for safety-related systems in general, and of the ECCS in particular, were developed after extensive design, review, and testing. As an example, the public hearing at which the acceptance criteria for ECCS were developed lasted for about 1 year and included extensive testimony and cross-examination by the NRC staff, the electric utilities, the reactor vendors and numerous intervenors representing various public interest groups. It should be noted that the final acceptance criteria which were adopted by the NRC included a considerable amount of conservatism to provide relatively large safety factors in establishing the design of safety-related systems.

Footnote 4. Section 50.34(a)(4)(ii) of 10 CFR Part 50.

The NRC does not assume that the adoption of conservative design criteria established for safety-related systems is the only method of providing assurance that safety systems will function as required in the event of an accident. Accordingly, a multiple step test program is conducted, including the the extensive preoperational test program and the periodic tests performed during the plant operating lifetime as discussed in Item 3 of this enclosure. Finally, an extensive ECCS testing program is continuing at the National Reactor Testing Site (NRTS) in Idaho to demonstrate the conservatism of the NRC acceptance criteria. This test program will not only establish a proof of principle for ECCS, it will also permit a detailed verification of the analytical models for the fluid flow, mass transfer and heat transfer characteristics of water-cooled nuclear power plants.

In summary, the conservative design criteria, the extensive preoperational test program, the periodic testing, and the test program at NRTS all provide assurance that the safety systems will function properly in the event of an accident.

5. Statement of Three General Electric Engineers Regarding Safety of Nuclear Power Plants

Ms. Carr refers to a statement by three nuclear engineers to the effect that "... a major atomic power plant disaster is likely within twenty-four years." It is assumed that this statement is attributed to the three engineers who resigned from the General Electric Company (GE) on February 2, 1976. These engineers are Mssrs. Dale G. Bridenbaugh, Richard B. Hubbard, and Gregory C. Minor. These engineers subsequently testified before the Joint Committee on Atomic Energy on February 18, 1976; the written version of their testimony consists of 72 pages. Recognizing the responsible nature and considerable experience of these three GE engineers, the Joint Committee conducted five days of hearings on this matter.

The NRC likewise recognized the excellent qualifications of these three GE engineers and prepared a detailed response contained in a 334 page report which was submitted to the Joint Committee on March 2, 1976. Since the testimony of the three GE engineers and that of the NRC is extensive, it shall not be summarized. However, it should be noted that all of the testimony cited above is contained in the two-volume record of these hearings.

The value of the statements made by the three GE engineers can be best evaluated by reviewing the various public investigations conducted regarding the charges which have been made relating to nuclear reactor safety.

In the instance cited above, the Joint Committee conducted an open hearing, with the media present, and published a hearing record containing about 1800 pages of testimony. All interested parties who could make a significant contribution were heard, including the three GE engineers, the NRC Commissioners, members of the Advisory Committee on Reactor Safety (ACRS), representatives of GE, representatives of the Consolidated Edison Company, Robert Pollard who is a former employee of the NRC, members of the NRC staff, and members of the NRC Licensing Board and the Appeal Board. It may be concluded from the lengthy testimony, the large list of witnesses and the varied topics in contention, that the question of the safety of nuclear power plants had an excellent presentation before the Congress and, through the media, directly to the American public. This legislative review of the charges, and overview by the Congress of the performance of the NRC, permits the Congress and the public to become informed on the subject of nuclear safety.

In addition to the legislative review discussed above, all of the charges brought by the three GE engineers, and by Mr. Pollard, were brought before the ACRS for a detailed technical review. This committee is composed of members who have considerable experience and expertise in the field of nuclear safety and are thereby well qualified to conduct an in-depth technical review of all the charges made by the four individuals cited above.

There is also an additional forum whereby contentions regarding the safety of nuclear power plants can be thoroughly aired. This forum is the public hearing (an administrative procedure), which must be held prior to issuing a license to construct a nuclear power plant. (Refer to Item 8 of this enclosure for a discussion of this hearing process.)

In summary, there has been a thorough legislative, technical, and administrative review of the charges made by the three engineers cited by Ms. Carr. While the individuals involved have excellent qualifications, they have not persuaded either the Congress, the ACRS, the licensing boards or the courts that their charges have sufficient merit to warrant either: (1) a shutdown of the presently operating nuclear power plants; or (2) a moratorium on the issuance of licenses to construct and operate additional nuclear plants.

6. Incidence of Abnormal Occurrences During 1974

In her letter, Ms. Carr alleges that "... in 1974 there were 1,421 accidents at the operating nuclear power plants in the country." It is assumed that Ms. Carr is referring to what were previously identified as "abnormal occurrences." If she is, there were 1,566 abnormal occurrences reported to the NRC during 1974 by operating nuclear power plants.

In order to place this topic into proper perspective, it is necessary to understand what was meant by an abnormal occurrence in 1974, and what is meant by that term now. First, it is emphasized that an abnormal occurrence was not necessarily an accident. In fact, an abnormal occurrence was very rarely anything resembling an accident. Recognizing this fact, the NRC has changed the name of an abnormal occurrence to a reportable occurrence. This action was taken by the NRC pursuant to the implementation of Section 208 of the Energy Reorganization Act of 1974 which established the NRC. Under Section 208, the NRC is required to report to the Congress every quarter, a listing of each abnormal occurrence defined in that Section as "...an unscheduled incident or event which the Commission determines is significant from the standpoint of public health or safety." The NRC, accordingly, now reviews in detail every reportable occurrence contained in the Licensee Event Reports (LER) to determine which of these will be categorized as abnormal occurrences as defined above and, therefore, reported to the Congress.

In July 1975, the NRC developed interim criteria (Attachment 6) for evaluating the reportable occurrences contained in the LER's to determine whether they are significant from the standpoint of public health and safety. These interim criteria define as significant (and reportable), events involving an actual loss of protection, or a major reduction in the degree of protection, provided for the health and safety of the public.

For each facility licensed under 10 CFR Part 50, the Technical Specifications for this facility set forth requirements for reporting to the NRC in 1974 any information concerning "abnormal occurrences" (now identified as reportable events) that took place. In general, these events were those that caused, or threatened to cause, a condition affecting safe operation of the facility. A

precise definition of what constitutes an "abnormal occurrence" at a given nuclear power plant in 1974 was set forth in the Technical Specifications for that facility and emphasized that the NRC's interest and concern in requiring reports of these events is associated with significant occurrences only. However, since the significance of a given event may not be clear, and because of the inherently subjective nature of assessing significance, the NRC required that those occurrences that may have significance with respect to safety, also be reported. This led to increasingly stringent reporting requirements and consequently, to a rapid escalation in the number of reports submitted by licensees.

Both in 1974 and in the present, the NRC collects and evaluates this operational information concerning licensed nuclear facilities to assess safety, and to form the basis for comparing plant performance with the design objectives. These reportable occurrences are incidents or events that involve system, component or structural failure or malfunction, personnel error, design deficiencies, management deficiencies, and other matters that are related to plant safety in various ways. Because of the multiple levels of protection, or "defense-in-depth," including the provision of redundant safety systems and components, such events do not, in general, affect safety directly and do not have an actual impact or consequence on the health and safety of the public. However, information regarding these events is useful to the NRC and to the nuclear industry in their efforts to improve safety. Therefore, these events are brought to the attention of the NRC through a variety of reporting requirements or by NRC inspection, and appropriate enforcement and corrective measures are thereafter taken.

With the understanding of what constitutes a reportable event and an abnormal occurrence and what is the objective of the NRC in requiring the submittal of LERs, the "abnormal occurrences" of 1974 will now be discussed. Of the 1566 reportable events in 1974, a very small number of these incidents had any direct significance to the public health and safety. In fact, the vast majority of these incidents were of no significance in and of themselves, but were required to be reported by the licensees so that any evolving pattern of similar occurrences at several or many plants may be detected, and remedial action taken, as soon as possible. The NRC reported in its first quarterly report to the Congress on abnormal occurrences, NUREG 75/090, "Report to Congress on Abnormal Occurrences, January - June 1975,"only six reportable events which occurred in 1974 and which were subsequently categorized by the NRC as abnormal occurrences.

The NRC's responsibility to keep the public informed regarding nuclear safety does not begin and end with its reporting requirements to the Congress. Currently, Licensee Event Reports, containing every reportable occurrence, as well as press releases by NRC regional offices and by licensees, are placed in about 122 local NRC Public Document Rooms every 2 weeks, as well as in the Washington, D.C., NRC Public Document Room. Lastly, NRC issues each month a "Grey Book," which includes all Licensee Event Reports and shutdown information.

In summary, the "accidents" to which Ms. Carr refers are the reporting of minor events by licensees in compliance with the Technical Specifications of the operating nuclear power plants and do not constitute accidents in the usual definition of this word. They have no significance in regards to public safety other than to allow the NRC and the nuclear industry to correct minor deficiencies in the design, operation, and maintenance of nuclear power plants.

7. Dispersal of Fission Products Following Postulated Accidents

In her letter, Ms. Carr alleges that "The accidental destruction of a nuclear power plant's core could be deadly up to ninety miles away, depending on weather conditions." While the basis for Ms. Carr's statement is not known, it is assumed that she is referring to an estimate made by extrapolating the information contained in Technical Information Document 14844, dated March 23, 1962. It is important to note that the calculations described in this document are considered to yield conservative, upper-bound distances since the analytical model on which this document is based, assumes no engineered safety features.

However, there are specific design features which are an integral part of nuclear power plants and whose design basis assumes that there is a release from the reactor pressure vessel of the fission products contained in the nuclear core. This assumption is made on a deterministic basis (i.e., no rational mechanism is assumed to be required to obtain this release) so as to impose extremely conservative design conditions on the engineered safety measures which are physically incorporated in the power plant to mitigate the consequences of any postulated accident. This release assumption is in accordance with the requirements of Section 100.11(a) of 10 CFR Part 100 (Attachment 7). However, this assumption implies that there is a complete failure of the safety systems which are specifically designed to prevent this release of fission products from the reactor core. This method of designing safety systems to withstand postulated worst case accidents, then assuming a failure of these systems and designing physically separate backup systems, which are diverse in principal, is known as "defense-in-depth."

Some of the engineered safety systems which are typically incorporated into the plant design and which mitigate the consequences of the postulated accident are the primary containment, the secondary containment, containment sprays, and charcoal filters. Prior to licensing a nuclear power plant, the NRC staff is required to demonstrate that the individual doses received by the public at specified distances from the facility following the design basis accident (i.e., the fission product release from the reactor pressure vessel) are within the guideline values contained in 10 CFR Part 100. These specified distances are identified as the radius of the exclusion area and the radius of the low population zone. Typical values of these distances are about 1/2 mile for the exclusion area and about 3 to 5 miles for the low population zone. These distances vary with plant site and are dependent on the power level of a facility, the engineered safety features, and the pertinent meteorological conditions of the plant site.

In the event of an accident at a nuclear power plant, the emergency plans might require an evacuation of the public within the low population zone and possibly beyond the low population zone, depending on the severity of the accident. However, Ms. Carr's allegation that the release of fission products would be deadly up to ninety miles away must be considered incredible. For her statement to be credible, one must assume a complete failure of all safety systems designed to protect the nuclear core and a failure of both the primary containment in combination with the highly unlikely postulated design basis accident (DBA). For the design of the containment the postulated DBA is usually a complete break in the largest pipe of the primary coolant system.

In summary, the physical systems incorporated into the design of nuclear power plants as part of the "defense-in-depth" concept provide assurance that the hazardous consequences of any fission product release from an operating nuclear power plant will be limited to a short distance from the plant. Further, the emergency plans for a nuclear power plant provide assurance that the public can be safely evacuated from this limited area if a severe accident does indeed occur.

8. Effect of Nuclear Power Plants on Public Health

Ms. Carr alleges that: "It has been proven that there has been an increase of infant mortality, cancer, and other health problems near nuclear plants." This statement is not accurate. A number of individuals with a scientific background have made such charges both in the press and in published books. However, these individuals have been unable to prove their charges in forums expressly established for this purpose. These forums include public hearings held pursuant to the regulations under which each nuclear power plant must be licensed. At these public hearings, sworn testimony is submitted with subsequent opportunity for cross examination and rebuttal. The decision of the licensing boards on the safety of a proposed nuclear power plant may be then subject to an extensive appeal process which can, and has, led to a judicial review by the U. S. Supreme Court. Considering the extensive legal process to which a proposed licensing action is subjected, Ms. Carr's contention that these charges have been proven cannot be accepted.

However, in response to her statement on public health, a discussion of the NRC's requirements governing routine releases of radioactivity from an operating nuclear power plant is provided. The fundamental concept underlying the NRC requirements in this matter is identified as "as low as reasonably achievable." These requirements are contained in Appendix I to 10 CFR Part 50 which was published after an extensive public hearing on the subject. Since these requirements are fairly complex, they are summarized in the following manner. Essentially, the routine releases (i.e., non-accident related) of radioactivity from an operating nuclear power plant must be held to amounts which result in whole body radiation doses to the public through all possible pathways (e.g., air-grass-cow-milk), that are a small fraction (about two to four percent) of the normal background radiation in our environment. Most experts in the field of radiation assessment believe this small incremental radiation dosage to be negligible. Indeed, greater incremental radiation dosages can occur from receiving diagnostic medical x-rays.

In summary, the NRC has established requirements that limit the radiation releases from operating nuclear power plants to amounts which result in incremental radiation dosages to the public that are a small fraction of the radiation which exists in our normal environment and in our normal daily functions.

9. Safety Measures for Nuclear Power Plants Established by the NRC To Protect Public Health

Prior to discussing the safety measures for nuclear power plants which have been established by the NRC to protect the public health, it should be noted that the nuclear industry represented by the electric utilities who operate the nuclear power plants; the architect-engineers who design and construct them; the reactor vendors who manufacture the nuclear reactors; the many diverse suppliers; and the numerous technical consulting firms, are all dedicated to the task of protecting the public health. Indeed, the prime responsibility for doing so is vested in their hands.

The role of the NRC in the licensing process is to establish requirements for nuclear power plants which are then issued as regulations contained in Chapter 1, Nuclear Regulatory Commission, of Title 10, Energy, of the Code of Federal Regulations (CFR). The NRC also establishes guidance in the form of regulatory guides, branch technical positions and ad hoc design requirements (refer to Item 2 of this enclosure). The ultimate responsibility of the NRC with regard to safety is to determine whether a proposed or operating nuclear power plant can be constructed and operated in a manner which will not adversely affect the public health.

To ensure that this objective will be achieved, the NRC: (1) establishes and reviews qualification and training requirements for all operating personnel (refer to Item 1 of this enclosure); (2) establishes and reviews design criteria (refer to Item 2 of this enclosure); (3) establishes guidance for, and reviews, the inspection process for the construction and operation of nuclear power plants (refer to Item 3 of this enclosure); (4) establishes acceptance criteria for the ECCS designed to protect the reactor core, including reviewing the detailed analysis of the ECCS, and reviews all safety-related structures, systems and components (refer to Item 4 of this enclosure); (5) monitors the operating history of operating nuclear power plants (refer to Item 6 of this enclosure); (6) establishes requirements for, and reviews, engineered safety features designed to mitigate the consequences of postulated accidents (refer to Item 7 of this enclosure); and (7) establishes requirements for, and continuously monitors, the routine releases of radioactive materials from operating nuclear power plants (refer to Item 8 of this enclosure). While this list is neither comprehensive nor complete, it does serve to illustrate the nature and character of the effort taken by the NRC to ensure that appropriate safety measures are proposed, implemented, and maintained for nuclear power plants.

The entire licensing and regulatory process conducted by the NRC is also subject to technical, administrative, judicial and legislative review to determine that the NRC does indeed perform its primary objective of protecting the public interest (refer to Item 5 of this enclosure). The excellent operating history of the civilian nuclear power plants in which not one death is attributable to accidents associated with the nuclear cores of civilian nuclear power plants, demonstrates the effectiveness of the safety measures proposed and implemented by the nuclear industry, reviewed and regulated by the NRC and reviewed again by the ACRS, the courts and the Congress.