DEPARTMENT OF ENERGY'S Team's Analyses of Soviet Designed VVERs

U.S. Department of Energy
Assistant Secretary for Nuclear Energy
Washington, D.C.

SEPTEMBER 1989

MASTER

THIS DOCUMENT CONTAINS NEITHER RECOMMENDATIONS NOR

ED

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

TABLE OF CONTENTS

	Page
INTRODUCTION	i
SECTION 1 OVERVIEW	1-1
1.1 Summary Description of VVERs	1-1
1.2 The Soviet Nuclear Program	1-3
1.3 Basis for U.S. DOE Teams Analyses	1-4
SECTION 2 VVER ACCIDENT PREVENTION AND MITIGATION FEATURES	2-1
2.1 Emergency Protection (AZ) Systems	2-1
2.2 Emergency Core Cooling System (ECCS)	2-1
2.3 Emergency Power Sources	2-2
2.4 Accident Localization/Containment Systems	2-3
2.4.1 VVER-440 Model V230 Accident Localization System (ALS)	2-3
2.4.2 VVER-440 Model V213 Accident Localization System (ALS)	2-4
2.4.3 VVER-1000 Model V320 Containment	2-5
2.5 Hydrogen Control Systems	2-5
SECTION 3 DETAILS OF THE U.S. DOE TEAM'S ANALYSES OF SPECIFIC ITEMS	3-1
3.1 Loss of Coolant Accident (LOCA) Analyses	3-1
3.1.1 Radiation-Induced Embrittlement and the Potential for Reactor Pressure Vessel Failure	3-2
3.1.2 Large Pipe Break Accidents	3-4
3.1.2.1 VVER-440 Model V230	3-4
3.1.2.2 VVER-440 Model V213	3-5
3.1.2.3 VVER-1000 Model V320	3-7
3.1.3 Steam Generator Internal Manifold Failure	3-7
3.1.4 Potential Consequences of Reactor Pressure Vessel Failure	3-9
3.1.4.1 VVER-440 Model V230	3-9
3.1.4.2 VVER-1000 Model V320	3-10
3.1.5 Small Pipe Break Accidents	3-11
3.1.6 Control Assembly Ejection	3-12
3.2 Reactivity Events	3-13

3.2.1 Reactivity Insertion by Control Assembly Drive Mechanisms	3-13
3.2.2 Reactivity Insertion by Removal of Soluble Poison	3-14
3.2.3 Reactivity Insertion Through Temperature Changes	3-14
3.3 Response to Loss of Electric Power	3-15
3.3.1 Loss of Electrical Power to Primary Coolant Circulation Pumps	3-15
3.3.2 Loss of Off-Site Power	3-15
3.3.3 Loss of AC Power, with Failure of the Diesels to Start	3-16
3.4 Response to Seismic Loads	3-17
3.4.1 VVER-440 Model V230	3-17
3.4.2 VVER-1000 Model V320	3-18
3.5 Aircraft Impact	3-18
3.5.1 VVER-440s	3-19
3.5.2 VVER-1000 Model V320	3-20
3.6 Environmental Qualification	3-20
SECTION 4 BENEFITS FROM EXPERIENCE	4-1
4.1 Retrofits	4-1
4.2 Plant Performance	4-2
4.3 Finland's Mix of Soviet and Western Approaches	4-3
SECTION 5 SUMMARY OF BACKGROUND APPENDICES	5-1
5.1 Background Appendix A, "Acronyms and Technical Terms"	5-1
5.2 Background Appendix B, "Accident Analyses"	5-1
5.3 Background Appendix C, "Reactivity Control"	5-1
5.4 Background Appendix D, "Comparison of the Soviet Safety Regulations with the United States General Design Criteria (10CFR50)"	5-2
5.5 Background Appendix E, "VVER Radionuclide Inventories"	5-3
5.6 Background Appendix F, "Decay Heat Generation in VVERs"	5-4
5.7 Background Appendix G, "Operations and Maintenance"	5-4
5.8 Background Appendix H, "Soviet Nuclear Steam Supply System Assembly and Construction Processes"	5-5
5.9 Background Appendix I, "Concrete And Concrete Structures"	5-5
5.10 Background Appendix J, "Seismicity"	5-6

5.11 Background Appendix K, "VVER Site Information"	5-7
5.12 Background Appendix L, "VVER Neutronic Parameters"	5-7
5.13 Background Appendix M, "Loss of Electric Power"	5-8
5.14 Background Appendix N, "Diesel Generator Reliability"	5-8
5.15 Background Appendix O, "Soviet Codes and Standards"	5-8
5.16 Background Appendix P, "Comparison of PWR and VVER Features"	5-8

VOLUME 1

ANALYSIS APPENDIX A	RADIATION INDUCED EMBRITTLEMENT AND ANNEALING OF REACTOR PRESSURE VESSEL STEELS
ANALYSIS APPENDIX B	LOSS OF COOLANT ACCIDENT (LOCA) BLOWDOWN ANALYSES
ANALYSIS APPENDIX C	LOCA BLOWDOWN RESPONSE ANALYSES
ANALYSIS APPENDIX D	FUEL/CORE TEMPERATURE ANALYSES
ANALYSIS APPENDIX E	NON-SEISMIC STRUCTURAL RESPONSE ANALYSES
ANALYSIS APPENDIX F	SEISMIC ANALYSES
ANALYSIS APPENDIX G	"S" SEAL INTEGRITY
ANALYSIS APPENDIX H	REACTOR TRANSIENT ANALYSES
ANALYSIS APPENDIX I	FIRE PROTECTION
ANALYSIS APPENDIX J	AIRCRAFT IMPACTS
ANALYSIS APPENDIX K	BORIC ACID INDUCED CORROSION

VOLUME 2

BACKGROUND APPENDIX A	ACRONYMS AND TECHNICAL TERMS
BACKGROUND APPENDIX B	ACCIDENT ANALYSES
BACKGROUND APPENDIX C	REACTIVITY CONTROL
BACKGROUND APPENDIX D	COMPARISON OF SOVIET SAFETY REGULATIONS WITH THE UNITED STATES GENERAL DESIGN CRITERIA (10 CFR 50)
BACKGROUND APPENDIX E	VVER RADIONUCLIDE INVENTORIES
BACKGROUND APPENDIX F	DECAY HEAT GENERATION IN VVERs
BACKGROUND APPENDIX G	OPERATIONS AND MAINTENANCE
BACKGROUND APPENDIX H	SOVIET NUCLEAR STEAM SUPPLY SYSTEM ASSEMBLY AND CONSTRUCTION PROCESSES
BACKGROUND APPENDIX I	CONCRETE AND CONCRETE STRUCTURES
BACKGROUND APPENDIX J	SEISMICITY
BACKGROUND APPENDIX K	VVER SITE INFORMATION
BACKGROUND APPENDIX L	VVER NEUTRONIC PARAMETERS
BACKGROUND APPENDIX M	LOSS OF ELECTRICAL POWER
BACKGROUND APPENDIX N	DIESEL GENERATOR RELIABILITY
BACKGROUND APPENDIX O	SOVIET CODES AND STANDARDS
BACKGROUND APPENDIX P	COMPARISON OF PWR AND VVER FEATURES

INTRODUCTION

The purpose of this report is to summarize the results of the U. S. Department of Energy (DOE) Analysis Team's analyses of Soviet designed VVERs (Water-cooled, Water-moderated Energy Reactor). The principle objective of this undertaking is to provide a basis to better understand the safety related features of the Soviet designed VVERs to be better prepared to respond domestically in the event of an accident at such a unit

The U.S. DOE Team's analyses are presented together with supporting and background information. The report is structured to allow the reader to develop an understanding of safety related features of Soviet designed VVERs (as well as the probable behavior of these units under a variety of off normal conditions), to understand the U.S. DOE Team's analyses of Soviet designed VVERs, and to formulate informed opinions.

The U.S. DOE Team was originally established to analyze the design of the Soviet graphite moderated pressure-tube reactors (RBMKs) and to analyze the April 26, 1986 accident at the Chernobyl-4 nuclear powerplant in the Soviet Union. The results of these analyses were published in DOE/NE-0076, "Report of the U.S. Department of Energy's Team Analyses of the Chernobyl-4 Atomic Energy Station Accident Sequence", issued November 1986.

The Soviet Union has also designed and constructed another type of nuclear powerplant, referred to as the VVER, that is cooled and moderated with pressurized water. The reactor concept used is not unique to the Soviet Union, and is similar to Pressurized Water Reactors (PWRs) that are in use in Western countries. The Soviet designed VVERs have many safety related design features and use safety related approaches that are different from those used in western PWRs. Although considerable information on VVERs has been published by the Soviet Union and other countries, only limited information was readily available in English. There was only a limited understanding, in the U.S., of the safety features of the Soviet designed VVERs when this activity was begun.

The safety related features of Soviet designed VVERs are considerably different from those of U.S. PWRs. The newer Soviet VVER designs (e.g., the VVER-1000 model V320s and VVER-440 Model V318 [being built in Cuba]) have incorporated safety related features to prevent and to mitigate the effects of accidents that were not present in earlier designs. However, the differences in safety related features of U.S. PWRs and Soviet designed VVERs are both significant and extensive.

The Chernobyl-4 accident analyses provided an understanding of some considerations associated with the Chernobyl type nuclear facility and of the potential for improvements in the safety related and accident mitigation features of this type nuclear powerplant. Since the accident resulted from features associated with a uniquely Soviet facility, there were few direct technical implications of the Chernobyl accident on the nuclear programs in other countries, yet public concerns were exacerbated by the lack of readily available information and understanding of Chernobyl type reactors.

The occurrence of an accident in a Soviet designed VVER with the potential for transboundary releases of radionuclides, may be expected to create concerns such as those that resulted from the Chernobyl accident. The public will demand and deserve prompt, reliable information, especially regarding relevance implications to U.S. nuclear power activities even though Soviet designed VVERs are quite different from U.S. PWRs. Thus, there existed a need to collect and to provide the U.S. public and government agencies information concerning the design of the VVERs, information relevant to the safety approaches employed in VVERs, information concerning VVER history and design evolution, and background relevant to the infrastructure used to design, build, and operate these nuclear powerplants. DOE/NE-0084, Revision 1, "Overall Plant Design Descriptions, VVER, Water-cooled, Water-moderated, Energy Reactor", issued October 1987, provided this type of information. Revision 2 of this document, provides an expansion of design details, and an update.

There also was a need to better understand the safety related features and characteristics of Soviet designed VVERs, and to make this information available. For this reason the U.S. DOE Team was directed to analyze features of Soviet VVER designs, and to provide information that would allow a better understanding of these units. This report summarizes some of the analyses performed in of this regard.

The objective of the U.S. DOE Team's VVER Analyses was to provide a basis to better understand VVER safety related features and how VVERs could be expected to respond to potential off-normal events and to potential accident initiators. These analyses considered differences between the designs and engineering practices used in U.S. PWRs and Soviet design VVERs, but did not consider the degree to which Soviet designed nuclear facilities' comply with U.S. regulatory rules. The Soviet Union's rules, regulations, standards, procedures, infrastructure, economic structure, and approaches used to design, construct, and operate nuclear facilities are considerably different from those used by designers in the U.S. The U.S. DOE Team's analyses used technical and engineering approaches that were selected to provide best estimates of the response of VVERs to various postulated and hypothetical events to better understand their probable behavior, rather than the "demonstrably conservative" approaches used to satisfy regulatory requirements.

As a consequence of this approach, the results of these analyses, while they indicate the probable response, may be more optimistic than would have been the case if the more conservative regulatory analysis type methods had been used. In several instances, as could be expected, the results were more optimistic than analyses of comparable events published by the Soviet Union and Eastern European countries.

The areas analyzed were selected to focus on items that are normally regarded by designers as important to safety, to preventing accidents, and to mitigating the effects of accidents. Areas that appeared to represent significant considerations were analyzed in some detail. These areas were selected based on reviews of published analyses conducted in the Soviet Union and other countries, concerns expressed in technical forums by experts from these countries, and analyses of specific VVER safety related features.

The U.S. DOE Team's analyses did not include many features of support systems or all details of major components and systems that may be important to safety. The explicit assumption made in these initial analyses was that in responding to postulated accident initiators, the materials, systems, and components would perform in a manner most appropriate to the safe operation of the unit. It is emphasized that this is an assumption made to simplify these analyses and does not imply that such performance will or will not be achieved. Furthermore, these analyses did not address site specific features, other than seismicity. The next level of detail would be the performance of individual VVER unit analyses, however, limitations in the available data base did not make such analyses possible.

LOCA type analyses received special attention. Structures, systems, and components to prevent or mitigate the effects of loss of coolant accidents and radionuclide releases are missing in some instances and use different design approaches (than those in the United States) in others. As examples, VVER-440 reactor pressure vessels are quite susceptible to radiation induced embrittlement, due to both the materials used and the relatively high flux present. The materials used on some newer VVER-1000s, appear to be less susceptible to radiation induced embrittlement (some newer VVERs use a grade of steel, designated "AA," that is lower in harmful impurities and less susceptible to radiation induced embrittlement). Some Soviet designed VVERs use lined compartments called accident localization systems (rather than full containment) to prevent the release of radionuclides. The VVER-440 model V230 is not designed to withstand large pipe breaks and has only a limited capability to provide make up water at the rate required to prevent fuel/core damage in the event of a large break LOCA. The electrical and control cable routing is not designed with the provisions for redundancy, physical separation, and fire protection that are the practice in the U.S. newer model.

This report consists of this introduction, five sections, and an attachment. There are also 11 analysis appendices, (that provide details of specific analyses) and 15 background appendices (that provide background information pertinent to performing the analyses and understanding certain aspects of Soviet designed VVERs). The attachment to this report

provides a comparison of features in U S PWRs and Soviet VVERs, and briefly discusses the provisions in VVERs to prevent, and to mitigate the effects of, accidents

The five sections of the main report are briefly described below

Section 1, Overview, provide summary descriptive information on the various VVER models, Soviet nuclear activities, and the rationale for selection of the areas analyzed by the DOE Team. Additional details are provided in DOE/NE-0084, "Overall Plant Design Descriptions, VVER, Water-cooled, Water-moderated Energy Reactor"

Section 2, VVER Provisions to Prevent and to Mitigate the Effects of Accidents, summarize some of the key considerations associated with these safety related design features. The provisions made in VVERs differ considerably from those on all U S PWRs. In some cases, key provisions are lacking. In others the design is significantly different.

Section 3, Details of the DOE Team's Analyses of Specific Items, summarizes the results of some specific analyses of probable responses to some potential off-normal events and postulated or hypothetical* accident initiators. The U S DOE Team's LOCA and LOCA blowdown response analyses are discussed to provide an understanding of the best estimate prediction of potential consequences of postulated and hypothetical* LOCA events and the response of the accident mitigating features included in the various designs to these events. Details of other analyses by the U S DOE Team concerning the response of VVERs to a variety of structural loadings, to a variety of postulated and hypothetical reactivity insertions, to loss of electric power and to potential seismic loads are discussed.

Section 4, Potential Benefits From Experience, discusses areas where potential benefits might be realized from mutual consideration of experiences and practices of the other.

Section 5, Summary of Background Appendices, provides a short summary of the information contained in the Background Appendices, giving the rationale for assembling the information, the approach used to gather and analyze the information, and briefly identifying items relevant to better understanding the analyses performed by the U S DOE Team.

The Appendices summarize details of some of the U S , D O E team's activities in this report. The Analysis Appendices summarize details of some of the U S DOE Team's analyses. The Background Appendices provide supporting information and supplement information contained in DOE/NE-0084.

Analysis Appendices

Analysis Appendix A, Radiation Induced Embrittlement and Annealing of Reactor Pressure Vessel Steels, presents and discusses information from U S , Soviet, and other sources concerning radiation induced embrittlement and its causes, factors effecting changes in the nil ductility temperature (NDT), the role of material chemistry in NDT shifts, and dose rate effects. This is important in that it provides an understanding of the potential for reactor vessel failures from mechanisms such as pressurized thermal shock (PTS). Engineering aspects associated with in situ annealing of reactor pressure vessels, and Soviet work in this area, are discussed. Some VVERs have a significant consideration in this area as a result of metallurgical and manufacturing practices used in the past that are unique to the reactor pressure vessels produced for these VVERs.

*In Soviet terminology a "postulated" accident is a design basis event, while a "hypothetical" event is a beyond design basis event.

Analysis Appendix B, Loss of Coolant Accident (LOCA) Blowdown Analyses, summarizes the results of LOCA analyses made using the RETRAN 2 Mods 2 and 4 code. These analyses were performed to calculate the fluid conditions at the breaks for use in the accident localization compartment/containment loading calculations provided in Analysis Appendix C, LOCA Blowdown Response Analyses. These analyses consider hot leg and cold leg breaks, of the primary coolant piping in VVER-440 model V230s, in VVER-440 model V213s, and in the VVER-1000s. This is important in that it provides information to perform analyses to understand the potential for the LOCA blowdown forces to cause failure of the structure provided to mitigate the effects of accidents by containing radionuclides.

Analysis Appendix C, LOCA Blowdown Response Analyses, summarizes multi-compartment analyses of the temperature and pressure loadings in the building structure surrounding the reactor following postulated failures of the primary coolant boundary. The effects of a postulated failure of a steam generator manifold were analyzed. The effects of a postulated failure of the reactor pressure vessel resulting from radiation induced embrittlement were analyzed. The potential consequences of postulated reactor pressure vessel head penetration type accidents were analyzed. These are important because they provide an understanding of the consequences of failure of the barrier that might otherwise reduce the release of radionuclides to the environment in the event of a severe accident, and the characteristics of such failures.

Analysis Appendix D, Fuel/Core Temperature Analyses, contains the results of analyses that were performed to calculate the fuel/core temperature conditions for various postulated events that other analyses had indicated could cause significant damage. The events were selected based on examination of results of Soviet, Eastern European analyses, and DOE Team analyses. RETRAN 2 Mod 4 was used to analyze natural circulation core cooling in a VVER-440 model V230 and the seizure of one pump in a VVER-1000 model V320. RELAP 5 Mod 2 (ISPRA) was used to analyze surge line LOCAs in VVER-440 models V230 and V213. Simple, and conservative energy balance analyses indicate the potential for fuel/core damage to occur as a result of small break LOCAs in all VVERs: Rational and reasonable "best estimate" type analyses of small break LOCAs require detail modeling of the horizontal steam generator, consideration of loop seal effects and consideration of percolation effects in the horizontal steam generators. Simulating analyses of a main steam line break requires detail modeling of the steam generator. The lack of an auxiliary feedwater system with an independent water supply on VVER-440 model V250s and features of the auxiliary feedwater system and secondary system provide a basis for interest in these transients. These analyses were not performed, however, due to the lack of the requested detail to adequately model the systems. The fuel/core temperature analyses that were performed provided an understanding of some of the events that might lead to excessive temperatures in fuel/cores with intact core geometry.

Analysis Appendix E, Non-seismic Structural Response Analyses, summarizes the results of structural analyses that consider the effects of pressure loadings on building structures and the steam generators. This provides an understanding of the margin between the design overpressure and the overpressure at which failure will occur. These supplement the LOCA blowdown analyses and provide an understanding of the effects of potential pressure loadings on structures that are potential barriers to radioactive release of radionuclides to the environment.

Analysis Appendix F, Seismic Analyses, summarizes the results of seismic analyses of the main buildings and the primary coolant loops for VVER-440s and VVER-1000s, analyses of the accident localization system, the deaerator support, the annular water tank that supports the reactor pressure vessel support for the VVER-440 model V230, and sloshing of water in the VVER-440 refueling pool. These provide a basis for an understanding the response of key structures and systems to seismic loadings and of the potential for a seismic event to be the initiator of an accident that might result in transboundary releases of radionuclides.

Analysis Appendix G, "S" Seal Integrity, summarizes analyses of the seal between the reactor pressure vessel head flange and the walls of the reactor shaft.

Analysis Appendix H, Reactor Transient Analyses, provides some results of point kinetic analyses of various reactor transients, including control assembly withdrawal at full power and hot standby, boron dilution accidents, failure to scram, and cold slug transients. A wide spectrum of initial events was surveyed to understand the potential transient response of the system to a spectrum of postulated and hypothetical events. These analyses provide a basis for understanding the response characteristics of VVERs to some reactor transients.

Analysis Appendix I, Fire Protection, considers the fire protection measures provided in VVERs and the potential consequences of a variety of postulated events resulting from fires. This is important to understanding the potential consequences of fires at various locations in VVERs.

Analysis Appendix J, Aircraft Impacts, discusses potential results of aircraft impacts upon the reactor building/accident localization system/containment and adjacent areas of various VVERs. The appendix considers the potential consequences of postulated aircraft impacts upon containment and other structures housing radioactive waste, diesel generators, primary coolant system equipment, safety equipment, and spent fuel storage facilities, as well as locations in the turbine generator building, the switchyard, coolant water pump, and other support facilities.

Analysis Appendix K, Boric Acid Induced Corrosion, summarizes considerations in this potential problem area. The mechanisms are discussed. Potential events in various systems and components containing boric acid are identified. Potential consequences are discussed.

Background Appendices

Background Appendix A, Acronyms and Technical Terms, provides a listing of some key acronyms and technical terms frequently found in literature about VVERs, together with the comparable terminology used in the U.S.

Background Appendix B, Accident Analyses, provides a summary of the results of analyses of VVER accidents and reactor transients performed in the Soviet Union and other countries operating VVERs. Information is provided as to the type of VVER, identification of the analysis, the initiating event, the calculated consequences. These provide 1) information to understand some of the events that are of interest to the designers, builders, and operators of VVERs; 2) the results of analyses conducted in these countries; 3) the results of analyses by groups other than the U.S. DOE Analysis Team; 4) a basis for comparing the results of the U.S. DOE Team's analyses with results of analyses by others; and 5) the results of analyses of areas not considered by the U.S. DOE Analysis Team in this initial effort.

Background Appendix C, Reactivity Control, provides information concerning the characteristics of VVER reactivity control systems. An understanding of their characteristics is needed since reactivity control systems can respond both to terminate reactivity transients or to initiate them.

Background Appendix D, Comparison of Soviet Safety Regulations with the United States General Design Criteria (10 CFR 50), provides background information concerning differences between Soviet safety rules and the general basic regulatory criteria used in the U.S.

Background Appendix E, VVER Radionuclide Inventory, provides data concerning the quantities of radionuclides contained in various VVER cores and spent fuel storage pools that are potentially available to provide a source for radionuclide release in the event of accidents.

Background Appendix F, Decay Heat Generation in VVERs, provides the information concerning the energy to be removed in various shutdown accident conditions to prevent the core from overheating.

Background Appendix G, Operations and Maintenance, provides background information on the practices and procedures used to operate VVERs.

Background Appendix H, Soviet Nuclear Steam Supply System Assembly and Construction Processes, is a translation and update (by the author) of a manual originally written by Dr. Jerzy Laszkiewicz and published in Poland in the late 1970s, based upon his experience working on various VVER related activities.

Background Appendix I, Concrete and Concrete Structures, provides background information and data concerning the characteristics of concrete and construction practices used to build VVERs.

Background Appendix J, Seismicity, provides background information on seismicity in the Soviet Union, Cuba, and pertinent Eastern European countries, as well as a description of some of the rules and design practices used to accommodate seismic events. This provides information concerning the potential for, and magnitude of, seismic events at various VVER sites. Information relevant to Crimea and Armenia seismicity at the sites in the Soviet Union, and at the Kozloduy and Belene sites in Bulgaria, is discussed in some detail.

Background Appendix K, VVER Site Information, provides background information concerning the location of various VVERs and the associated population distributions, nearby cities, towns, seismicity climate, and other relevant information. This information provides an initial basis for understanding the potential impact of an accident in the relevant VVER on the surrounding area.

Background Appendix L, VVER Neutronic Parameters, provides information relevant to neutronic kinetic parameters applicable to the design and operation of VVERs. Information provided is based on a literature survey, some original calculations for benchmark comparisons, and a review of VVER accident analyses and computer code descriptions (Background Appendix B).

Background Appendix M, Loss of Electrical Power, summarizes considerations relative to the consequences of various postulated loss of electric power accidents. Such events may be potential initiators of accidents and may interfere with the functioning of design features provided to mitigate effects of accidents.

Background Appendix N, Diesel Generator Reliability, discusses the reliability of the diesel generators used in Soviet designed VVERs.

Background Appendix O, Soviet Codes and Standards, provides information concerning details of the standards and practices used to produce the various types of VVERs.

Background Appendix P, Comparison of PWR AND VVER Features, provides information to facilitate comparisons of safety related features in Soviet designed VVERs and U.S. PWRs. Differences between U.S. PWRs and Soviet designed VVERs are summarized.

SECTION 1 OVERVIEW

This section provides background information about the Soviet designed water-cooled, water-moderated reactors (VVERs).

1.1 Summary Descriptions of VVERs

VVERs are pressurized water reactors (PWRs), but they are different from U.S. PWRs in many important ways. There are two principal sizes of VVERs producing about 440 megawatts electric (MWe) and 1000 MWe, called the VVER-440 and the VVER-1000, respectively. There are two principal standardized models of the VVER-440s, denoted as the models V230 and V213. One of the VVER-1000s, the model V320, is considered in this report. There also are two other models, the VVER-1000 model V187 (Novovoronezh Unit 5) and the VVER-1000 model V302 (the initial production model). Figure 1 shows the locations where VVERs are in operation or under construction. In addition, two VVER-440s (designated as the model V318) are under construction in Cuba. These units are briefly mentioned here for completeness, but the principal emphasis in this report is on the main standardized models. The model number is the designation assigned to the project that developed the design. There are also two advanced VVER-1000s being designed. The VVER-88 and the VVER-92. These reactors are being designed to contain more safety related features to prevent and mitigate the effects of accidents than current VVERs.

The first VVERs were constructed at Novovoronezh. Units 1 (210 MWe) and 2 (365 MWe) were VVER prototypes that went into operation in 1964 and 1969, respectively. Unit 1 has been shut down. Unit 2 is being shutdown. Units 3 and 4 were the first VVER-440s and went into operation in 1971 and 1972, respectively. These two units were the first of the standardized VVER-440 model V230s. Other VVER-440 model V230s started up a short time later in the USSR (Kola-1, which began operation in 1973) and in Eastern Europe (Nord-1, in the German Democratic Republic [GDR], which began operation in 1974) and Kozloduy-1, in Bulgaria, which began operation in 1974).

Figure 1. VVER Site Locations
(See Table 10 for Site Names)

VVER-440 model V230s are always constructed in modules of two units. This design has six primary coolant loops. Each of the primary coolant loops contains a hot and a cold leg isolation valve, a horizontal steam generator, and a low inertia canned motor pump. The reactor pressure vessel is unclad low alloy steel. The reactor pressure vessel steel contains impurities that, in combination with high neutron fluxes, makes these vessels susceptible to radiation induced embrittlement. The fuel is UO_2 in annular pellet form, clad in a zirconium-niobium alloy, arranged in a triangular lattice, and housed in 126-rod closed hexagonal ducts. The control assemblies have fuel assembly followers. There is no emergency core cooling system (ECCS), and the units use steel-lined compartments (called an accident localization system) to reduce the release of steam and fission products in the event of an accident, rather than containment. VVER-440 model V230s are not designed to contain the effects of postulated major accidents, such as a large pipe break. There are nine 1.0 meter (39.4 inch) dump valves in the walls of the reactor compartment (accident localization boundary) that open under overpressure conditions to vent steam to the atmosphere and then close. The Soviet General Safety Regulations (OPB-82) permit discharge of radioactive materials into the environment for cases where this release is considered to improve powerplant systems safety. The VVER-440 model V230s were not designed for the level of seismicity that exists at some sites. All VVER-440 model V230s have twin 220 MWe turbines.

There are a total of 16 VVER-440 model V230s which went into operation between 1971 and 1980; 6 are in the USSR (at the Kola, Novovoronezh and Armenia sites) and 10 are in Eastern European countries (Bulgaria, Czechoslovakia, and the GDR). The two units at the Armenia site were shut down permanently by March 1989. There are no additional VVER-440 model V230s under construction or planned.

The VVER-440 model V230 was replaced by a new standardized model named the VVER-440 model V213 that also produces a nominal 440 MWe. The basic reactor system, primary heat transport system, and secondary system of the VVER-440 model V213 are similar to the older VVER-440 model V230. The most significant changes in these systems being the use of a shaft sealed primary coolant pump, providing pump coast down and the addition of an auxiliary feedwater system, with a water supply other than the normal feedwater system. The VVER-440 model V213s also include emergency core cooling systems, and a modified approach to accident localization. Beginning with this design, the Soviets considered the effects of postulated major accidents such as a large pipe break. The first VVER-440 model V213 went into operation at Rovno in the USSR in 1980.

The VVER-440 model V213 has a stainless steel lining inside the reactor pressure vessel. The emergency core cooling system includes four accumulators (two inject into the outlet plenum, two feed into the inlet plenum), high-pressure injection pumps, and low-pressure injection pumps. The reactor building continued to use the fundamental concept of accident localization (as did the VVER-440 model V230), rather than full containment. The VVER-440 model V213 accident localization system (ALS) has a much larger volume than its predecessor, using a tunnel to connect the steam generator compartment to an exterior structure, adjacent to the reactor building that houses a structure containing 12 levels of pressure suppression pools. This part of the accident localization system is called the "bubbler-condenser" tower. The "bubbler-condenser" tower also features "air-trap" volumes into which air in the compartments, is forced by the pressure build up in the event of a pipe break accident. One way valves prevent this air from returning to the compartments. That, plus the use of a spray system, causes the ALS atmosphere to have a subatmospheric pressure within minutes after an accident.

There are four VVER-440 model V213s operating in the USSR and 11 more in Eastern Europe (Czechoslovakia, the German Democratic Republic (GDR), and Hungary). Nine additional units are under construction in Eastern Europe (Czechoslovakia, the GDR, and Poland). In addition, there is a two unit VVER-440 model V318 under construction in Cuba. The Cuban units have different features than the standard VVER-440 model V213s, most notably a full containment for each unit. At least two more of this type VVER-440 model V318 design are reported to be under construction in the USSR (in the Far East). The Soviets are marketing this type VVER-440 and VVER-1000s for export.

The Soviet's current emphasis is on the VVER-1000, a 1000 MWe, 4-loop system featuring individual, full containment structures. The prototype VVER-1000 (Novovoronezh Unit 5) became operational in 1980. This prototype model is called the VVER-1000 model V187; Novovoronezh Unit 5 is the only V187. There are also four VVER-1000s called the model V302, (South Ukraine Units 1 and 2 and Kalinin Units 1 and 2). These VVER-1000s became operational between 1983 and 1986. The current standardized version of the VVER-1000 is called the model V320. This report concentrates on this standardized model. The Soviets are developing new models of the VVER-1000, referred to as the "model 88" and the "model 92". There are only limited details available on the "model 88" and on the "model 92" and these designs were not considered in these analyses.

VVER-1000s have some significantly different features than VVER-440s. The hexagonal canned fuel assembly was changed to an open lattice, permitting coolant cross flow. Control rod arrays, without fuel followers, replaced the control assemblies. Only the VVER-1000 model V187 and VVER-1000 model V302 units have loop isolation valves; VVER-1000 model V320s do not. All VVER-1000 units have one turbine rather than two, with the exception of the VVER-1000 model V187 and a modification of the VVER-1000 model V320 design that provides district heating as well as electricity (denoted ATETs). All VVER-1000s have ECCSs. The containments are steel lined, cylindrical, prestressed concrete structures, with the exception of the two units at Steadall in the GDR that uses steel lined, reinforced concrete.

Nine VVER-1000 model V320s are operational in the USSR, about 20 are under construction, and specific plans exist for over 20 additional units. One VVER-1000 model V320 is operational at Kozloduy (Unit 5) in Bulgaria, at least two more are under construction outside the Soviet Union and there are plans for at least 14 more. The Soviet Union's continued construction and development programs has the stated intention of increasing nuclear generating capacity by a factor of from five to seven by the year 2000. This is to be accomplished primarily with VVER-1000s.

It has been announced that four VVER-1000 model V320 ATETs units that were under construction (two in Minsk and two in Odessa) in the USSR have been cancelled. An additional four planned ATETs units (two at Kharkov and at Volgograd) have been either cancelled or deferred indefinitely. It was announced that 6 VVER-1000s planned for construction at Krasnodar in the USSR have been cancelled. Two VVER-440s in Armenia were shut down following the December 7, 1988 earthquake in Armenia.

1.2 The Soviet Nuclear Program

The Soviet use of standardized designs, standard components, and standard construction methods is a key difference between the U.S. and Soviet nuclear programs. The Soviets develop standard designs, such as the VVER 440 model V230. A number of units are built at the same or various sites. Experience and development activities are used to develop new models, such as the VVER 440 model V213. When the new -model is ready for deployment, the new model replaces the older model (which is no longer built). In a similar manner, the Soviets use standard components. As examples, all VVER-440 model V230s use standard type GTsN-310, canned motor pumps, VVER-440 model V213s use the standard type GTsN-317 shaft seal pumps, and VVER-1000s use the standard type GTsN-195 M shaft seal pumps. All VVER-440s use the standard type GZZ-500 valve for the primary coolant system isolation valves. While some components used on VVER-440s may differ between the VVER-440 model V230s to the VVER-440 model V213s, (e.g. the reactor pressure vessel and primary coolant pump), most components are unchanged. VVER-1000s also use standard components.

This use of standardized components significantly simplifies many important areas such as training, maintenance, spare parts, manufacturing, and construction. VVERs are normally built in five years or less (the Cuban units are the exception). It contributes to a capability to achieve high plant capacity factors. It establishes a broad base for obtaining data on operating experiences. Where pragmatic considerations, experience, or rule changes require, changes can be made to a standard component design, with the modified design becoming the new standard design.

There are some differences between similar reactor models built in Eastern European countries compared to those in the Soviet Union. These differences result from the applications of different engineering approaches and safety rules, the use of local materials, components and manufacturing capability, trade considerations, local conditions, and the applications of supplemental requirements to Soviet codes and standards. These changes normally tend to be minor. However, the four VVER-440 model V213s being constructed at Nord in the GDR and the two VVER-1000 model V320s under construction at Stendal contain significant difference in structural design and layout. Also, there appear to be a number of different protective (AZ) system designs in use. Advances in electronics are incorporated into VVER models.

The Soviet nuclear power program is controlled by the government, with responsibility for nuclear safety being a government responsibility. Responsibility and authority for the Soviet nuclear program and for implementation of specific functions related to achieving nuclear safety is assigned to the Ministry of the Nuclear Industry in the Union of Soviet Socialist Republics and the State Committee for the Supervision of Industrial Safety and Safety in the Nuclear Power Industry. These organizations were created in July 1989 as part of the ongoing overall restructuring of the Soviet Government. The State Committee's role includes safety-related R&D, technical reviews, and actions related to safety. The Ministries role is to do R&D, to design, to manufacture, to construct and to operate nuclear power plants.

In the United States, the public has long had significant participation in the regulatory process through public hearings, the intervention process, and available vehicles for litigation. In the Soviet Union, the public has recently begun to participate through public meetings and interactions with State organizations responsible for the design, construction, and operation of nuclear powerplants and for policy related to nuclear powerplants (e.g., siting). Soviet officials have attributed the decision to cancel several nuclear powerplants (the VVER-1000 ATETs at Odessa and Minsk and the VVER-440s in Armenia) to a response to public opinion.

There are many differences in safety approaches used throughout the world, and (such as the degree of fire protection provided, the capability provided to contain radionuclides, the capabilities for emergency core cooling, and the degree of isolation and of redundancy between electrical systems or other safety related systems) differences in degree and approach should be judged on technical principles. The Soviets appear to have had no backfit policy (although some changes were made on an ad hoc basis) until recently, for example, and have not provided either containment or an ECCS on VVER-440 model V230s.

DOE/NE - 0084, "Overall Plant Design Descriptions, VVER, Water-cooled, Water-moderated Energy Reactor", provides some of the key Soviet rules concerning nuclear safety. Background Appendix D, "Comparisons of Soviet's Safety Regulations with the United States General Design Criteria (10 CFR 50)", provides information showing similarities and differences in key rules pertaining to safety related features. It is emphasized that these are key rules, but not the only rules. The United States has a variety of rules and regulations other than the General Design Criteria and the Soviet Union has rules and regulations other than OPB-82. These provide a basis for understanding how the Soviet designed VVERs physical characteristics differ from those of U.S. PWRs, and also how the rules used to design and operate Soviet VVERs differ from those used in the U.S.

1.3 Basis for U.S. DOE Analysis Team's Analyses

The U.S. DOE Analysis Team identified a number of safety related features of Soviet designed VVERs that differ from those in U.S. PWRs. These include system designs, physical features, the way units are operated, rules and practices. Experience in construction and operation of VVERs has shown that the VVER designers have been successful in achieving rapid construction times, high plant capacity factors, and high unit availabilities.

The differences between VVER systems and U.S. PWR systems are such that the system response to postulated and hypothetical accident initiators would be different. An objective of this effort was to provide a basis for understanding

the response of VVER systems for such events. The Team performed a number of analyses with this objective. To achieve this objective a ground rule was that the analyses would be based on rational and reasonable engineering judgement rather than utilizing demonstrably conservative methods.

In addition to using Soviet and Eastern European publications as a basis for selecting accident sequences to examine, the Team was guided by evaluation of information gathered for DOE/NE-0084, and the information that is summarized in the 15 Background Appendices. These Background Appendices are summarized in Section 6 of this report.

The course and scope of the analysis was influenced by information showing that there is a concern by the Soviets and others regarding radiation induced embrittlement in VVER-440s. This arises from a number of factors. Soviet reactor pressure vessels are designed to be rail shipped using the Soviet rail system, and vessel diameters are restricted accordingly. Core diameters are relatively large, however, such that the fuel-to-vessel wall, water-filled gap, between the outer rows of fuel assemblies and the reactor pressure vessel wall is smaller than in U.S. PWR's (e.g. 27 centimeters [10.6 inches] versus about 50 centimeters [20 inches] in a typical U.S. PWR). The neutron spectrum in VVER-440s is harder than in U.S. PWRs owing to the smaller moderator-to-fuel ratio in the triangular pitch lattice. The water gap in VVER-1000s is increased to 39.3 centimeters (15.5 inches) and the fast flux decreased by a factor of 4.

The steels and weld metals used in early VVERs were relatively high in phosphorus and copper, and the VVER-1000s steels also contain one percent nickel, making them more susceptible to radiation induced embrittlement at the same copper levels. Steels used in vessels that were not manufactured using the new "AA" grade of steel contained impurities which rendered them susceptible to embrittlement, and this may be worsened in the VVER-440 model V230 by the synergistic effect of hydrogen embrittlement on those unclad vessels. There was an absence of radiation monitoring coupons in early VVER's, and attempts to backfit in these areas was stymied by the lack of archive materials.

Steels have a range of temperatures in which they are ductile and fail by ductile rupture when stressed beyond their tensile strength. Below this range of temperature, the failure mechanism is brittle. The effect of radiation induced embrittlement is to raise the temperature at which steel becomes brittle. At or below this nil ductility transition temperature, the presence of flaws in the steel and a stress or loading (such as may result from thermal shock while the reactor pressure vessel is pressurized [pressurized thermal shock]) may cause these flaws to propagate by brittle cleavage leading to failure. Analysis Appendix A, "Radiation Induced Embrittlement and Annealing of Pressure Vessel Steels" examines this problem and Soviet annealing practices as a means of reducing the problem in more detail.

This susceptibility of VVERs that do not use the new "AA" grade steels to vessel embrittlement has been recognized and steps are being taken in some countries to lessen the neutron flux incident on the wall of VVER-440 reactor pressure vessels, and to monitor the effect of radiation on the reactor pressure vessel steel. In-site reactor pressure vessel annealing, using a procedure developed in the Soviet Union by the Kurchatov Institute (discussed in Analysis Appendix A), has been accomplished at Novovoronezh-3 and Armenia-1 in the Soviet Union, and at Nord-1 in the GDR. Plans are to perform in situ annealing on either Kozloduy 1 or 3 in Bulgaria and on Kola-1 in the Soviet Union in 1989. Czechoslovakia is developing a procedure to perform in situ annealing and plans to anneal at least one of the reactor pressure vessels in the VVER-440 model V230s at Bohmice by 1993. The VVER-1000s that do not use the new grade "AA" steels are also susceptible to radiation induced embrittlement, but no information is yet available on plans to develop or apply in situ annealing to these vessels. The concern regarding vessel embrittlement resulted in including vessel beltline rupture as a postulated failure in the scope of LOCA response analyses.

The requirement to transport the vessels by the Soviet rail system results in the vessels being delivered to the construction site with relatively short nozzles. Hence, field welds are made relatively close to the vessel, possibly introducing stresses that may not be adequately relieved in the field. The Soviets used electroslag welding in earlier VVERs. Slag inclusions are difficult to remove, if not removed, a flaw remains which is a potential source of stress concentration. Although specific details were not provided, publications in the Soviet Union indicated Soviet

nondestructive testing (NDT) may not identify and resolve all potentially significant flaws. There have been reports of problems associated with sealing of the steam generators internal manifold covers. Owing to the combination of circumstances described, the Team's analyses included large break, as well as small break LOCAs, and also included steam generator manifold ruptures. Analysis Appendix B, "Loss of Coolant (LOCA) Blowdown Analyses" summarized the work performed to determine the characteristics of the steam/water released by these postulated events. Appendix C, "LOCA Blowdown Response Analyses" summarizes the analyses that examined the effect of the loadings resulting from these events on VVER structures. Appendix D, "Fuel/Core Temperatures Analyses", summarizes the results of analyses of LOCA and other postulated off normal events on temperatures in the core. Background Appendices E, "VVER Radionuclide Inventory", and F, "Decay Heat Generation in VVERs" provide information that supported these analyses and help the reader to understand the potential consequences of a loss of core cooling.

Earthquakes are a potential accident initiator. In general, the Western Soviet Union and Eastern Europe are regions of relatively low seismicity. The early Soviet practice was stated as avoiding siting nuclear powerplants in seismic regions. There were no special provisions in the design for accommodating significant seismic loads, piping systems were loosely supported. However, some nuclear powerplant sites (e.g., Armenia, Crimea in the Soviet Union, and Kozloduy in Bulgaria) are located in regions of significant seismicity. The Vrancea earthquake in 1977 resulted in an estimated 0.15g horizontal ground acceleration at Kozloduy. Although the reactor continued to operate during the earthquake without major incident, it was later discovered that the steam generators had undergone a significant (12 centimeter [4.7 inches] or more) displacement. Subsequent to this event, construction of Armenia Unit 2 was delayed over two years while seismic features were added.

As a result of this earthquake experience, the changing Soviet approaches for handling seismic load (to a more rigid system using shock absorbers), and the reports of some seismicity related changes to some VVERs on an ad hoc basis, the Team undertook an examination of seismic characteristics of the various VVER plant sites based on examination of various applicable publications (with the assistance of the U.S. Geological Survey) and of changes made following the Vrancea earthquake. Seismic response analyses were performed using the Kozloduy plant VVER-440 and -1000 designs and site soil characteristics as references. The analyses (Analysis Appendix F, "Seismic Analyses") included the building responses, component responses, vessel responses, and sloshing responses for a reference earthquake.

In the VVER-440 model V230, an "S" seal between the reactor pressure vessel flange and the reactor shaft provides a seal in the upper head region. Analysis Appendix G summarizes the results of an analysis of the structural capability of this seal.

The plant protection system, instrumentation and control, and electrical systems, particularly for the VVER-440s were considered. The VVERs protection system lacks redundancy and fire protection (Analysis Appendix I, "Fire Protection" summarized the considerations in this area).

Publications by the Soviets and other countries with VVERs explicitly state that external events such as aircraft impact are considered to be part of the design basis for some VVERs. The Team used existing methods and analyses to examine the degree of resistance of the VVER-1000s to aircraft impact, and similarly for the VVER-440 units. The analyses (Analysis Appendix J, "Aircraft Impacts") were not limited merely to penetration of the reactor building and internal structures, but also consider impact at numerous other localities on the plant site.

Cross coupling exists between the emergency electrical system wiring and also between channels of the reactor protection (AZ) system. Control and safety functions are interconnected in ways that may allow a failure in a control system to prevent operation of a safety system. The scram insertion time for a VVER-440 is about 10 seconds which is much longer than for either U.S. nuclear power plants or VVER-1000s. One specific concern unique to the VVER-440 model V213 is the AZ-1 full scram function. Rather than irreversible drop of the absorber assemblies (8 inch per second), the absorbers remain mechanically connected to the drive system. Any fault in the electrical system may cause failure

to scram plus sustained rod withdrawal. These features led the Team to examine transient analyses (Analysis Appendix H, "Reactor Transient Analyses") involving failures to scram and rod withdrawals.

Reports of leaks found in the region of the reactor pressure vessel heads, and the requirement in some VVERs for periodic monitoring for leakage in the reactor pressure vessel head penetrations, etc, resulted in an examination of boric acid induced corrosion (Analysis Appendix K). Failure of such a penetration could result in a LOCA and movement of a control assembly from its required location.

The reactor protection (AZ) system used in VVERs is fundamentally different from that used in U.S. PWRs. U.S. reactors have a single protection response which is a full, irreversible scram. VVERs have four possible responses (AZ-1, 2, 3, 4) depending upon sensor indications of fault severity. This type of protection system involving three levels of response less drastic than a full scram is one factor contributing to the generally excellent VVER performance records. High plant capacity factors are achieved by a number of actions of numerous factors including reliable designs, relatively low fuel power operation, horizontal steam generator designs, and high accessibility of reactor systems during operation. A key factor in achieving high capacity factors is a pervasive attitude which emphasizes the priority of keeping the plants operating. These attitudes can be understood as an outgrowth of limited excess electrical capacity and the high economic value placed on the heat and electricity generation in the Soviet Union and in Eastern European countries.

The emphasis on high plant capacity factors is evident in a number of design features of the VVERs (in addition to the multiple response levels of the AZ system). For example, the presence of six primary coolant loops each having loop isolation capability in the VVER-440s offers the potential for on-line repairs or maintenance, if needed. This is facilitated in VVER designs by relatively low radiation fields which can be attributed to factors such as the selection of materials that minimize activation sources in the primary system and the use of a high capacity primary coolant purification system as well as water chemistry control to minimize the transport of radionuclides in the reactor primary system coolant. The use of multiple turbine-generators for each unit permits operation at reduced power during turbine generator or secondary system repairs or maintenance. Atmospheric steam dump valves are not fast acting, but are electrically operated with a 10-20 second operating response time.

These types of features in the design lead to features in transient analyses, and transient sequences, which are different than those in U.S. plants. These features were modeled in the transient analyses, and transients were selected which reflected these different features.

SECTION 2 VVER ACCIDENT PREVENTION AND MITIGATION FEATURES

This section discusses the systems that are provided in VVER designs to avoid or minimize core damage in the event of an accident and to avoid or minimize the releases of steam and fission products from the confines of the reactor building. Accidents may result from such events as a break in the primary or secondary system pressure boundary, failure of electric power sources, core reactivity perturbation, fire, or seismic loading, among others. The DOE Team analyzed the functioning of the VVER accident mitigation systems and the performance of these systems for a few selected accidents which are among those that must be analyzed for U.S. plants. The results of these analyses are described in Analysis Appendix B, "Loss of Coolant Accident (LOCA) Blowdown Analyses", Analysis Appendix C, "LOCA Blowdown Response Analyses", Analysis Appendix D, "Fuel/Core Temperature Analyses", Analysis Appendix H, "Reactor Transient Analyses", Background Appendix C, "Reactivity Control", and Background Appendix M, "Loss of Electrical Power".

The normal functioning of the VVER accident mitigation systems are described in this section. The functional information contained here is what was used for the accident consequence analyses.

The accident mitigation features are categorized into four groups, namely 1) Emergency Protection Systems, 2) Emergency Core Cooling Systems, 3) Emergency Power Sources, and 4) Accident Localization/Containment Systems. The functioning and uncertainties pertaining to these systems are summarized. Further details are contained in DOE/NE-0084.

2.1 Emergency Protection (AZ) Systems

The emergency protection function is provided by the AZ-1 (scram) system. Additional protection to mitigate or prevent accidents is provided by other programmed power reduction or rod inhibit functions (AZ-2,3, and 4). When AZ-1 is actuated, all safety and control assemblies are gravity dropped into the core. The trip-insertion time is about 4 seconds in VVER-1000s; in VVER-440 model V230s have speed limiters that slow the insertion speed to 0.25 meters per second (10 inches per second) and increase the trip-insertion time to about 12 seconds. The scram function is actuated by numerous instrument circuits to rapidly reduce the reactor power to avoid or to minimize the consequences of an accident (see DOE/NE-0084, for additional details of these and other AZ functions).

There is a lack of redundancy in the AZ system and a lack of physical protection from fires and environmental damage in the cables used to transmit the signals associated with this system. This results in a risk of the system not being able to function as the designers intended it should and thus being unable to prevent or mitigate a significant accident as discussed above. Experience in U.S. nuclear facilities has resulted in stringent requirements for redundancy, separation of the redundant cables, physical and fire protection and environmental qualification being applied to use nuclear facilities. The lack of such features entails a risk not present in U.S. PWRs. The control system and the AZ system are interconnected. This results in the risk of "cross-talk" and erroneous control action. This could also result in the AZ system not being able to function, as the designers intended it should, and thus being unable to prevent or mitigate a significant accident. This feature is not allowed in U.S. PWRs.

2.2 Emergency Core Cooling System (ECCS)

The ECCS is used to mitigate the consequences of a loss of coolant accident (LOCA) by reflooding and maintaining water flow through the reactor core. The functional requirement of the 1982 Soviet General Safety Regulations, OPB-82, is that the ECCS shall limit the cladding temperature to not more than 1200 degrees Centigrade, limit peak cladding oxidation to not more than 18 percent of the initial thickness, and limit the total fraction of zirconium oxidized to not more than one percent. These Soviet regulations require the use of a maximum planned accident (design basis accident) that is equivalent to the "instantaneous rupture of the maximum diameter pipeline with unhindered two-way flow of

the coolant during operation of the reactor at nominal power". An ECCS is provided in the VVER-440 model V213 and VVER-1000 model V320 systems to satisfy this requirement. The maximum size pipes are about 50 centimeters (20 inches) and 85 centimeter (33 inches) inside diameter, respectively. There is no ECCS in the VVER-440 model 230 design capable of satisfying the OPB-82 regulation. There is no definitive information about plans or actions having been taken to retrofit any of these reactors to come into conformance with this regulation, although the Soviet Council of Ministry announced plans to upgrade safety related features on older nuclear power plants. A recent report indicated that emergency core cooling features are being (or have been) added to some VVER-440 model V230 units at Kozloduy in Bulgaria.

In the event of a LOCA in a VVER-440 model V230, there is the ability to inject a limited amount of water into the system using make up pumps with a capacity of about 100 cubic meters per hour (433 gallons per minute) at operating pressure. The Soviet assertion is that this flow will provide protection from core damage for a LOCA involving a break of a 10 centimeter (3.9 inch) inside diameter pipe incorporating a 3.2 centimeter (1.3 inch) diameter flow restrictor (orifice) based upon unidirectional (rather than bidirectional) flow. This event is roughly analogous to a TMI-2 small break loss of coolant accident. There are numerous pipes of larger size. The largest size pipe is 50 centimeters (20 inch) inside diameter. The response of the VVER-440 model V230 system to a break of this largest size pipe was calculated. The limited makeup capability provides no significant mitigation of a large break LOCA.

The ECCS for most VVER-440 model V213s and VVER-1000s consist of four independent passive accumulators, two that inject into the downcomer and two that inject into the outlet plenum, three independent high pressure injection systems, and three independent low pressure injection systems. As such, they are similar but conceptually different from U.S. PWR ECCSs.

2.3 Emergency Power Sources

On-site emergency power sources are provided in the event there is loss of off-site power. Batteries supply DC power to those AC and DC systems that cannot be without power for longer than one second, including the reactor protection system, critical instrumentation/control/monitoring systems, and emergency lighting, among others. Other systems are powered by the emergency source on a "second priority" basis. All units have two or three diesel generators which are said to be capable of starting to pick up load in 60 seconds and reaching 100 percent load at 3 minutes. The high priority loads include ECCS, the emergency boron injection system, and the emergency feedwater system.

There are numerous interconnections between "redundant" systems in the electrical power supplies, that result in these systems not really being redundant since a single event or accident initiator could result in a loss of all interconnected systems. This could preclude accident mitigation and potentially be an accident initiator. This type of design approach is not allowable in U.S. PWRs. There are common elements in the diesel service system that could result in a single cause resulting in a failure of these units to operate as and when required. For example, some units use common cooling water for all diesels (supplied from a common service water line), and a common housing for several diesels with no barriers to prevent the spread of fires.

A lack of redundancy, fire protection, and electrical separation similar to the situation mentioned with respect to the signal cables and the emergency protection system exists in the electrical systems. The results of multiple electrical failures would be different from the consequences of multiple protection failures, but still could lead to an accident. For example, several places are shown in an electrical wiring diagram where interconnections can be made between redundant parts of the emergency power systems. This occurs both in the first priority level (the battery portion of the emergency supply) and the second priority level. This ability to interconnect redundant systems leads to the possibility of a single fault or action disabling both circuits of the first priority or both circuits of the second priority systems. If done in a manner that permits a fault to propagate through the system, this could lead to loss of all AC power. The Soviets consider the loss of all AC power to be an incredible event, and do not consider it in the design of VVERs.

A unique third emergency power source is provided on the VVER-440 model V230s. These reactors use a low inertia canned rotor, primary coolant pumps which require uninterrupted AC power. Four of the six pumps are powered by two dedicated inertial power generators (IPGs), each driven by one main turbine. Two pumps are connected to each IPG and two are connected to a source of off-site power. The IPGs provide pump power during both normal operation and turbine coastdown and are designed to provide a transition to natural circulation following loss of power to the main coolant pump motors. However, VVER-440s can operate at 55% power with three pumps leading to the possibility that two of these pumps could be lost.

2.4 Accident Localization/Containment Systems

These systems are provided to prevent or minimize the release of steam and radionuclides to the atmosphere in the event of an accident. The features to accomplish this differ appreciably between the three VVER models. VVER-1000s have a full cylindrical steel-lined prestressed concrete containment building. VVER-440s with the exception of the VVER-440 model V318 do not have a full containment but use a system of sealed, steel lined, interconnected compartments. These steel-lined reinforced concrete compartments localize the effects of a loss-of-coolant accident. The primary coolant system including the steam generators are located in this accident localization system (ALS). Additionally, since the refueling bay is outside ALS, releases from a fuel handling accident are not isolated from the environment. The ALS and containment systems of three VVER systems are briefly discussed below.

2.4.1. VVER-440 Model V230 Accident Localization System (ALS)

This ALS consists of three interconnected compartments, namely the reactor shaft, the steam generator compartment, and the deck compartment (containing motors and other electrical equipment). The compartment walls are 1.0 to 1.2 meter (3.3 to 3.9 feet) in thickness; the ceiling, which is the refueling bay floor, is 1.0 meter (3.3 feet) in thickness; and the foundation mat is 3.0 meter (9.8 feet) in thickness. The foundation slab is only a small depth below grade, and the reactor shaft penetrates below the foundation slab so that the core midplane is essentially at grade elevation. The rebar mass in the ALS concrete boundaries amounts to about 115 kilogram per cubic meter of reinforced concrete. A steel liner of 0.4 centimeter (0.16 inches) thickness covers the inside surface of the ALS boundaries. An S-shaped steel ring seals the gap between the reactor vessel and surrounding concrete shaft wall to separate the lower reactor shaft of the ALS from the upper part of the reactor shaft. The upper part of the shaft is covered by a steel dome that is bolted to the floor of the refueling bay. The design overpressure is 0.1 megapascal (14.5 psi gauge). The design basis accident for this plant does not consider failures of either the 20 or 50 centimeter (7.9 or 20 inch) pipes in this system.

The ALS is protected against overpressure failure by its volume of 10,000 cubic meters (350,000 cubic feet) plus nine vent valves one of which opens at 80 percent with the other eight opening at 100 percent of design overpressure. The design basis for the valves is a 20 centimeter (7.9 inch) break with unidirectional flow. The vent valves are sized to prevent the pressure inside the ALS from exceeding the design pressure. The vent valves are counter-weight operated dump valves. One valve is designed to open at 0.08 megapascal (12 psi gauge) and to reclose when the overpressure is relieved. The remaining eight valves are to open and reclose at an overpressure of 0.1 megapascals (14.5 psi gauge). The ALS vent valves discharge into the atmosphere through the surrounding building.

In the event of a LOCA, steam is condensed and heat is removed from the ALS by the operation of a water droplet spray system. This system is assumed to require 60 seconds to begin delivering droplets into the ALS, based upon the time for the pump to come on and fill the water pipes. The spray water and condensate are collected by a sump system, and the water can be recirculated via a heat exchanger for long-term heat removal.

Large air exchange lines penetrating the ALS boundary belonging to the air purification system and air cooling system appear to have isolation valves. These valves are believed to be electronically operated with one valve located outside

of the ALS. The possibility exists for there to be high leakages or possible failure of these valves under large break LOCA conditions.

The integrated analyses of a large break LOCA including loading and response of the ALS boundary is summarized in Analysis Appendices B, C, D, and E.

2.4.2 VVER-440 Model V213 Accident Localization System (ALS)

This ALS is similar to that used in the VVER-440 model V230 with the major addition of a separate building structure that contains a water-pool pressure suppression system termed the "bubbler-condenser" tower. The functioning of the VVER-440 model V213 ALS is therefore significantly different from that of the VVER-440 model V230. The VVER-440 model V213 ALS consists of a number of interconnected compartments, namely the reactor shaft, the steam generator compartment, the deck compartment, the steam tunnel (which connects the three ALS compartments of the reactor zone to the bubbler/condenser tower), the ventilation equipment room (assumed), the vertical steam shaft of the bubbler-condenser tower, the water trays, and the air traps. The compartment walls, ceiling, and floor of the ALS are 1.5 meter (4.9 foot) in thickness. The 3 meter thick foundation slab is flat for the VVER-440 model V213 design, no longer having the downward protrusion for the reactor shaft characteristic of the VVER-440 model V230. The reactor vessel and piping system are thereby elevated in the VVER-440 model V213 compared to the VVER-440 model V230; the core midplane is well above grade level in this design. The rooms below the steam generator compartment are not part of the ALS; they are used for the ECCS equipment and water storage tanks introduced in the VVER-440 model V213. The rebar mass in the ALS concrete boundaries amounts to about 115 kilogram per cubic meter of reinforced concrete. A steel liner of 0.6 centimeter (0.24 inch) thickness covers the inside surface of the ALS boundaries. An S-shaped steel ring seals the gap between the lower part of the reactor shaft and the upper part where the seal head is located. The shaft is sealed by a rounded disk cover that is bolted to the refueling floor. After the initial pressurization is reduced by steam condensation, the compartments are subatmospheric for a time until inleakage eliminates the pressure difference. The ALS leak rate measured to be 15 percent per day at the design overpressure of 0.15 megapascal (21 psi gauge) at one unit was noted as being extremely good. Other plants are suspected to have similar or higher leak rates.

The design basis accident for the VVER-440 model V213 plants is a guillotine rupture and double-ended flow from the largest pipe in the system that has a 50 centimeter (20 inches) nominal diameter. The ALS is protected against overpressure failure by its volume of 52,500 cubic meters (1,838,000 cubic feet), based upon the design basis accident. This volume is a factor of about five greater than that of the VVER-440 model V230. Unlike the VVER-440 model V230, the VVER-440 model V213 does not have vent valves releasing to atmosphere. Steam released during a LOCA causes flow of air and steam through the steam tunnel, into the bubbler condenser tower, where the air and steam flow through inverted channels and bubble through 0.5 meter (1.6 foot) deep water pools arranged in twelve levels, stacked one above the other. Air and any uncondensed steam pass into air traps through one-way valves. The system is designed to condense steam and to decrease the air present in the ALS compartments. Additionally, steam is condensed and heat is removed from the ALS atmosphere by operation of water droplet sprays (three independent systems). This system is reported to be designed to require 60 seconds to begin to deliver droplets into the ALS, based upon the time for the pumps to come on and fill the water pipes. Since there is a net depletion of air from the ALS reactor zones, operation of the sprays to condense steam cause the system pressure to become subatmospheric. Soviet analysis indicates that this occurs at about 12 minutes in the large break LOCA. The ALS leakage, which was originally to the outside, becomes an inward air leakage afterward. The sprays are cycled on and off to maintain the ALS pressure in the range 0.085 to 0.095 megapascal (12.3 to 13.8 psi absolute). The air inleakage has been reported to preclude maintaining of subatmospheric pressure after about two days. The spray water and condensate are collected by a sump system, and the water can be recirculated via heat exchangers back into the reactor (ECCS) or sprays for long-term heat removal.

The VVER-440 model V213 ALS has a three-valve isolation system; one valve inside and two outside the localization boundary. The air purification and air cooling systems are located in a ventilation room in the reactor building.

Integrated analyses of a large break LOCA have been performed. Results are summarized in Analysis Appendices B, C, D, and E.

2.4.3 VVER-1000 Model V320 Containment

Most VVER-1000s have a steel-lined, prestressed, reinforced concrete containment structure completely enclosing the reactor and primary system components.¹ The cylindrical containment wall is 1.2 meter (3.9 feet) thick, the dome is 1.0 meter (3.3 feet) thick, and the base is 2.4 meter (7.9 feet) thick. This base is located about 13 meters (43 feet) above grade so that the reactor system is far above grade in this design. The foundation slab is 3.0 meter (9.8 feet) thick. The rooms between the foundation slab and the containment base are not part of the containment volume; they are used principally for ECCS and containment spray systems. The containment is lined with carbon steel plate of 0.8 centimeter (0.31 inch) thickness. The cylindrical shell incorporates helically wound prestressing cables at about 32 degrees from horizontal. Cables are run at 90 degree orientation in the dome. Details of construction are given in Background Appendix I. These containments are designed to achieve a leak rate of 0.1 percent per day and is tested to show that a leak rate of 0.3 percent per day is achieved. The design overpressure is 0.41 megapascal (60 psi gauge); the design temperature rise is 150 degrees centigrade (270 degrees Fahrenheit). Primary system components are located in steel-lined concrete-walled compartments within the containment. The compartments are interconnected with sufficient open area to preclude any significant localized overpressure following the design basis accident. The compartmentalization serves both missile shield and radiation shield functions.

The design basis accident is a guillotine rupture and double-ended flow from the largest pipe in the system, 85 centimeter (33 inch) inside diameter. The containment is protected against overpressure failure by its volume of 70,000 cubic meters (2,500,000 cubic feet), based upon the design basis accident. Unlike the VVER-440 model V230 ALS, the VVER-1000 model V320 containment is a closed system with no vent valves releasing to the atmosphere. Steam released during a LOCA collects in the containment causing pressure increase. Steam is condensed and heat is removed from the containment atmosphere by operation of water droplet spray systems (three independent systems) delivering water to three concentric circular spray headers in the containment dome. The system is assumed to require 60 seconds to begin delivering water droplets into the containment atmosphere, based upon the time for the pumps to come on and fill the water pipes. The total available flow rate used in the analyses was 403 kilogram per second. The spray water and condensate are collected by a sump system, and the water can be recirculated via heat exchangers back into the reactor (ECCS) or sprays for long-term heat removal.

The VVER-1000 model V320s have a three-valve containment isolation system; one valve inside and two valves outside the containment boundary. The air purification system is isolated. Air coolers are located inside the containment.

Integrated analyses of a large break LOCA have been performed. Results are summarized in section 3 and Analysis Appendices B, C, D, and E.

2.5 Hydrogen Control

There are currently limited means available for the control of combustible gas currently in some VVERs (portable skid mounted equipment is available at some sites). Thus, if an accident in a VVER progresses to severe core damage,

¹ The VVER-1000 model V320s being constructed at Stendal in the GDR use a "cell type" reinforced concrete containment shell that is steel lined on both the inside and outside surfaces.

the hydrogen fraction may build up in the ALS or containment volume to that required for a burn or possibly an explosion. In particular, if the water level falls beneath the top of the core and the fuel temperature rises to sufficiently high values, hydrogen will be generated from chemical oxidation reactions of steam with zirconium present in the cladding and other fuel assembly structure. At higher temperatures, stainless steel structure also interacts with steam to form hydrogen. If a sufficient amount of hydrogen accumulates inside the accident localization volume, then a hydrogen burn or explosion may occur, depending upon the proportions of hydrogen, air, and steam.

SECTION 3 DETAILS OF THE U.S. DOE TEAM'S ANALYSES OF SPECIFIC ITEMS

A series of analyses were performed to provide a basis to better understand safety related features of Soviet designed VVERs. Potential consequences of primary and secondary loss of coolant accidents (LOCAs) were examined, along with the responses of the accident mitigation systems to these events. Responses to reactivity events, loss of power events, and seismic events were also examined.

3.1 Loss of Coolant Accident (LOCA) Analyses

OPB-82, the Soviet safety regulation for nuclear powerplants, section 4.1, "Nuclear Powerplants with VVER Reactors" states

"4.1.1 Instantaneous rupture of the maximum diameter pipeline with unhindered two way flow of coolant during operation of the reactor at nominal power should be considered in the design as the MPA (maximum planned accident) with depressurization of the primary system" and

"4.1.3 The emergency cooling system should ensure the following upon depressurization of the primary system (the second rated limit of damage to fuel elements)

- fuel cladding temperatures of not more than 1,200 degrees Centigrade,
- a local percentage oxidation of fuel cladding of not more than 18 percent of the initial wall thickness,
- a fraction of reacted zirconium of not more than one percent of its mass in the core"

"4.1.4 There will be the capability of unloading the core after the MPA with depressurization of the primary system"

The VVER-440 model V230 design basis accident is a 10 centimeter (3.9 inch) diameter pipe break with unidirectional flow through a 3.2 centimeter (1.3 inch) orifice insert. The double ended rupture of a 50 centimeter (20 inch) primary coolant pipe is not included in the design envelope. VVER-440 model V230s do not have emergency core cooling systems. The emergency core cooling capability is limited to 100 cubic meters per hour (433 gallons per minute) from the make up water system.

The U.S. DOE Analysis Team performed an integrated series of analyses to provide a capability to better understand the safety related features of VVERs in this area. The behavior of the primary system was predicted from two-phase thermal-hydraulic calculations. The condition of fluid exiting the break site was used to predict pressure and temperature loadings inside building structures. The building structural response was calculated using three-dimensional, finite element methods.

These analyses are summarized in analysis appendices. Analysis Appendices A "Radiation-induced Embrittlement and Annealing of Pressure Vessel Steels" examine a potential cause of reactor pressure vessel failure. Analysis Appendix B, "Loss of Coolant Accident (LOCA) Blowdown Analysis", includes a summary of calculations of the fluid conditions at the breaks for use in containment loading calculations for a variety of postulated double ended guillotine breaks of primary coolant system pipes. Analysis Appendix C, "LOCA Blowdown Response Analyses", includes a summary of calculations of pressures and temperatures inside VVER structures following various postulated LOCA events, using fluid conditions from Analysis Appendix B. Analysis Appendix D, "Fuel/Core Temperature Analyses", includes a summary of calculations of some LOCA events that had been identified as having potential for damage to fuel. Analysis Appendix E, "Non-seismic Structural Response Analyses", includes a summary of calculations of the accident

localization structures response to the overpressure resulting from some postulated LOCAs. Analysis Appendix H, "Reactor Transient Analyses", considers the reactivity effects of a variety of postulated events.

3.1.1 Radiation-Induced Embrittlement and the Potential for Reactor Pressure Vessel Failure

Radiation-induced embrittlement of VVER reactor pressure vessels resulted in considering the effect of thermal transients upon the steel when the vessel is pressurized and under stress. This may cause failure of the reactor pressure vessel either by a phenomena called pressurized thermal shock (PTS) or accidental pressurization at low temperature.

Radiation-induced embrittlement makes the reactor vessel more susceptible to brittle crack propagation and possible failure when stressed suddenly at low temperatures in the presence of an assumed or known flaw. This could lead to failure of the reactor pressure vessel. Soviet and other technical literature sources identify radiation induced embrittlement as a consideration in VVERs.

Radiation-induced embrittlement is more severe for VVER reactor pressure vessels than for U.S. PWR vessels because there is a high ($\sim 10^{11}$ n/cm²sec) fast neutron flux on the vessel wall at the core midplane and the Soviet materials used for the reactor pressure vessels are susceptible to radiation-induced embrittlement.

The mechanisms of radiation embrittlement and the potential mitigation of this problem by thermal annealing of the reactor pressure vessel are summarized in Analysis Appendix A.

The transition from "ductile" to "brittle" fracture behavior in pressure vessel steels (nil ductility temperature) occurs at a temperature below which fracture progresses more readily by cleavage than by ductile rupture. As a steel is irradiated by high energy neutrons ($E > 0.1$ MeV), various radiation-produced defects tend to block dislocation motion, which can lead to ductile ruptures, and thereby increase the temperature at which the transition to "brittle" cleavage fracture will occur. This process does not weaken the steel, but can make it more susceptible to brittle crack propagation, and possible failure when loaded suddenly (at low temperatures) in the presence of an assumed or known flaw. Embrittlement is commonly measured by determining the amount of energy required to fracture a standard notched specimen; the temperature below which failure occurs at 35 Joules (25 foot pounds) is commonly referred to as the ductile-brittle transformation temperature. In most reactor pressure vessel steels which have been properly stress-relieved after welding, this 35 Joules transformation occurs at or below 0 degrees Centigrade, in the absence of radiation exposure.

The progress of radiation-induced embrittlement is often monitored by the use of surveillance coupons, located inside the vessel to enable the operator to assess the condition of the vessel. However, surveillance specimens were not normally used in VVER 440 model V230's. Loviisa in Finland was the first VVER-440 that had such specimens included from startup. Examination of these specimens focuses attention on this consideration.

To avoid brittle fracture, a minimum pressurization temperature of the 35 Joules transformation plus 30 degrees Centigrade is specified for VVERs. (In the U.S., the margin required by the ASME code is 50 degrees Centigrade.) The Soviet literature gives equations for calculating the increases in the 35 Joules transition temperature, based on the neutron fluence and the chemical composition of the weld. The implication is that, in many cases, these calculations were relied upon to determine the minimum pressurization temperature. As discussed in Analysis Appendix A, it appears that the Soviet correlations used may be non-conservative in some cases. Loviisa (which uses a reactor pressure vessel fabricated by the Soviets, of the same type used in VVER-440 model V230s, but with the addition of stainless steel clad to reduce another form of embrittlement resulting from hydriding) used surveillance coupons to monitor embrittlement of the vessel "core weld". An excessively high rate of embrittlement was found. The shift in nil ductility temperature is attributed to the high concentrations of copper (0.17 percent) and phosphorus (0.031 percent) in the weld metal. The shift on the vessel wall itself was estimated to be somewhat lower, due primarily to lower phosphorus

and copper levels in the vessel wall than in the weld. At Loviisa, (and subsequently at some VVER-440 model V230s) this problem was addressed by modifying the core and replacing the outermost row of fuel with dummy assemblies made of stainless steel.

Recent evidence has proven the existence of a not-yet quantifiable dose rate effect, which indicates that irradiation to the same fluence at a lower flux may cause more damage. VVER reactor pressure vessels may be experiencing a greater shift in the embrittlement temperature than originally estimated from the higher flux surveillance coupons. This indicates that the Soviet approach of calculating the shifts from high flux test reactor data may be non-conservative. The potential risks are exacerbated by the lack of irradiation coupons and the lack of archive material from which to fabricate representative surveillance coupons.

The interior of VVER-440 model V230 reactor pressure vessels are not clad with stainless steel. The interior of VVER-440 model V213 and the VVER-1000 reactor pressure vessels are clad with stainless steel. The Soviets have developed a method whereby small specimens can be cut from the vessel wall of the VVER-440 model V230s in the core midplane region and removed for subsequent testing.

This method uses a shielded, manned cabin that is lowered into the vessel after the reactor core, internals, and core barrel have been removed. The specimens obtained are small, and not suitable for impact-type tests to determine the shift in 35 Joules transition temperatures. However, research at a number of laboratories, including those in the Soviet Union, has indicated that shifts in transition temperature can be directly correlated with radiation-induced changes in other mechanical properties, most notably in hardness. Microhardness measurements on the samples removed, therefore, will provide considerable information on the degree of embrittlement of the vessel wall. They also have developed techniques for remote microhardness testing of the (unclad) reactor pressure vessel wall and the core weld in-situ.

The use of unclad vessels has led to some hydrogen embrittlement of the reactor pressure vessel steels. The source of hydrogen is corrosion processes at the surface of the unclad steel. This embrittlement occurs in addition to that due to neutron bombardment, but appears, from surveillance tests to be a minor effect.

VVER-1000s use a steel alloy that contains approximately 1.5 percent nickel. The width of the water gap between the fuel and the vessel was increased from 27 centimeters (10.6 inches) in VVER-440s to 39.3 centimeters (15.5 inches) in VVER-1000s, with the result that the fast flux on the vessel wall was reduced by a factor of about four.

The primary differences between VVER-1000 reactor pressure vessel steel alloys and those used in VVER-440s are the increase in nickel content and the decrease in vanadium content. At the same copper levels, the increase in nickel will substantially increase the effect of radiation on embrittlement. The radiation induced embrittlement of reactor pressure vessel steels is a major consideration in both VVER-440s and VVER-1000s. For this reason, the "AA" grades of steel were developed that limit the copper and phosphorus content and are, therefore, more resistant to radiation-induced embrittlement. It is understood that all VVERs under construction and some that have begun operation recently will have or have reactor pressure vessels constructed of this material.

The Soviets and Eastern European countries operating VVER-440 model V230s plan to anneal reactor pressure vessels in-situ to reduce the amount of radiation-induced embrittlement. The Soviets are reported have developed an in-situ annealing process that was tested on Novovoronezh-3, and have applied this technique at Armenia-1 and at Nord-1 in the GDR. Plans are to anneal the reactor pressure vessels of Kola-1 in the Soviet Union and either Kozloduy 1 or 3 in Bulgaria during 1989. Czechoslovakia is also developing an in-situ annealing procedure and plan to anneal the reactor pressure vessels of the Bohonice units by the end of 1993.

Annealing to reduce radiation-induced embrittlement is done by heating the vessel, or the affected areas of it, to temperatures at least 50 degrees Centigrade above the irradiation temperature for extended periods of time. Recovery of 90-100 percent of the original properties requires heating the steel to temperatures of 450-500 degrees Centigrade (or approximately 200° above the irradiation temperature) for periods of approximately one week. The procedure used by the Soviets uses electrical resistance heaters mounted inside the reactor pressure vessel. The method is reported to provide a 70 percent maximum recovery in the area treated, the region nearest to the core level. Upon re-irradiation, embrittlement will recur at rates at least equal to and in some instances greater than those occurring originally. The best life extensions that can be achieved by annealing will thus not exceed the radiation exposure prior to the anneal. Thermal stresses on the vessel and distortions of the piping connections are a concern in the use of this technique, as is the monitoring the effectiveness of the annealing procedure on the reactor pressure vessel.

In-situ reactor pressure vessel annealing is a very difficult operation, requiring precise definition of the temperatures, temperature gradients, and hold times. Determining these requires a sound development and procedure qualification (or demonstration) activity. The executing of this procedure requires exacting control of temperatures, thermal gradients, and hold times. Failure to properly execute the proper annealing procedure can result in an increased risk of failure. On the VVER model V230 units, the Soviets monitor the effectiveness of the annealing procedure by their remote hardness testing method.

The application of a significant thermal shock to the reactor vessel while it is pressurized, called pressurized thermal shock (PTS) is a key consideration for VVER reactor pressure vessels that have experienced significant radiation-induced embrittlement. Several event sequences exist that can result in VVER reactor pressure vessels experiencing PTS. In some respects the VVER-440 model V230 reactor pressure vessels may have a more significant problem due to the lack of stainless steel cladding to prevent hydrogen embrittlement. The timely core redesign to replace the outer row of fuel with shielding may have reduced the susceptibility to neutron embrittlement in some units. Also, VVER-440 model V230s lack of an emergency core cooling system (ECCS) removes a key potential source of PTS.

The lack of representative surveillance coupons in many VVERs, coupled with the use of non-conservative correlations relating fluence to loss of ductility for VVER materials may result in VVERs operating with a smaller than expected margin for brittle failure.

3.1.2 Large Pipe Break Accidents

3.1.2.1 VVER-440 Model V230

The response of VVER-440 model V230s to a postulated double-ended guillotine rupture of a primary coolant pipe was analyzed. This was not included in the design basis of the VVER-440 model V230, but is a design basis accident for the VVER-440 model V213 and the VVER-1000. The primary piping systems of the VVER-440 models V230 and V213 are similar.

The limited makeup water capability in the VVER-440 model V230 makes a large pipe break a more severe accident for this type unit than for VVER-440 model V213s (which have ECCSs). The VVER-440 model V230 accident localization system design basis is unidirectional flow from a 10 centimeter (3.9 inch) diameter pipe break through a 3.2 centimeter (1.3 inch) orifice insert. Breaks of greater size (20 centimeters [7.9 inches]) result in opening of all of the accident localization system vent valves that directly discharge steam (and potentially radionuclides) to the atmosphere from the relatively small accident localization volume as primary system blowdown takes place. If a vent valve does not close, an atmospheric vent path will remain.

The response of the primary system was calculated for the cases of breaks both in the primary coolant inlet (cold leg) and outlet (hot leg) piping at locations between the reactor pressure vessel and isolation valve. Pressure and

temperature loadings inside the accident localization volume were calculated only for the hot leg break. Similar results would be expected for the case of a cold leg rupture. The pressure loading calculations do not interactively account for the effects of structural failure or enhanced leakage, that might develop through the compartment walls, reducing the pressure. The three-dimensional structural response analyses (Analysis Appendix E) considered the potential for failure of the accident localization system walls and floors, as well as the S-seal (Analysis Appendix G) between the reactor pressure vessel and the concrete shaft housing this vessel.

For the hot leg break case, in the calculations summarized in Analysis Appendix B, the primary coolant system depressurizes in about 60 seconds. The more exact calculations, summarized in Analysis Appendix D predicted the core would boil dry in about 50 seconds. The makeup water (100 cubic meters per hour [433 gallons per minute]) does not measurably alter this result in either calculation.

The corresponding pressures inside the steam generator and pump compartments were calculated to rise to the design pressure after about two seconds, causing all of the vent valves to the atmosphere to open. Rupture discs installed in pipes leading to the upper reactor shaft above the S-seal were predicted to open pressuring the upper shaft. Venting to the atmosphere limits the pressure inside the ALS to less than the design pressure of 0.1 megapascal gauge (29 pounds per square inch absolute). If the vent valves close, as designed, this will occur about 50 seconds after the break and near the end of the primary system blowdown. In the absence of any venting or structural failure, the pressure would rise about 0.9 megapascal (130 pounds per square inch) in about one minute.

The ALS concrete walls were calculated to undergo cracking partially through the walls and roof of the steam generator compartment in localized regions when the pressure exceeded .05 megapascals gauge (~ 22 pounds per square inch absolute). Cracking was predicted to begin in the outer walls, roof, and internal diagonal walls. Extensive localized concrete cracking was predicted to occur at the design pressure, but not gross structural failure, or liner failure. The S-seal, assumed to be 4 millimeters (0.16 inch) thick, is not predicted (Analysis Appendix G) to fail under the indicated pressure loadings. The S-seal is calculated to respond elastically at a pressure of 0.7 megapascal absolute (100 pounds per square inch absolute) and not to fail following a transient rise to a peak pressure of 2.6 megapascals absolute (380 pounds per square inch).

About one minute after blowdown starts (Appendix B calculates 60 seconds and Appendix D calculates 50 seconds for the core to boil dry) the core region is essentially dried out. The limited makeup water does not appreciably effect this time in either a hot leg or cold leg break. This makeup water is predicted to by-pass the core in the case of a cold leg break. Therefore, heatup of the core and core damage associated with excessive temperatures may be expected.

In summary, it was found that a double-ended primary coolant system pipe break could be expected to result in the venting of steam, and possibly radionuclides, from the volume of the ALS for a period of about 50 seconds during the depressurization of the primary coolant system. The pressure during this period may cause localized cracking of the concrete, but gross structural failure and failure of the ALS liners was not calculated to occur. The vent valves all must close and seat, as designed, for the ALS to function and retain its design leak integrity (one case cited as good was 16 percent per day). Approximately one minute after the break occurs, the core temperature is increasing due to both residual heat and decay heat, the core is not being cooled, and core oxidation and hydrogen generation is to be expected. Unless adequate core cooling is accomplished the core may be expected to melt.

3.1.2.2 VVER-440 Model V213

A double-ended guillotine rupture of a primary coolant pipe is a design basis accident for the VVER-440 model V213. The VVER-440 model V213, unlike the VVER-440 model V230, has an emergency core cooling system consisting of four accumulators, high pressure injection pumps, and low pressure injection pumps. In designing the VVER-440 model V213 ALSs, the rectangular box type compartments used in the VVER-440 model V230 were generally retained. Wall

and floor thicknesses were increased to increase the structural (pressure containing) capability. An increase in pressure suppression capability and an increase to the ALS volume were achieved by adding bubbler-condenser trays inside an adjacent building called the bubbler-condenser tower. The enhanced capability of the accident localization compartments surrounding the primary coolant system is designed to accommodate the pressure loadings that would result from a large pipe break LOCA. The accident localization volume has a high normal leakage rate and the accident localization approach relies upon the establishment and maintenance of subatmospheric conditions over a long period of time (several days).

Primary system thermal hydraulic and ALS loading calculations were performed for both inlet (cold leg) and outlet (hot leg) primary piping breaks. Because the thermal hydraulic model of the horizontal steam generator assumes a constant secondary side temperature, secondary-to-primary heat transfer is predicted after 20 seconds following the break. The pressure loading calculations did not model the effects of passive sprinkler phenomena that accompanies the backward expulsion of water from the bubbler-condenser trays. Condensation induced by passive spray effects would result in a somewhat more rapid decrease in the pressures inside the ALS volume than were predicted by the calculations. The pressure loading calculations also do not account for the effects of early structural failure or enhanced leakage, that might tend to reduce the pressures. The structural response calculations also do not consider the potential for failure of penetration seals.

The primary system was calculated to depressurize in about 15 seconds after the break occurs for the cold leg break and about 60 seconds for the hot leg break. The pressures were calculated to remain below the stated ALS compartment design pressure capability. The actual capability of the ALS structures is dependent upon construction and quality practices, particularly the reinforcement of the concrete walls and floors. The U.S. DOE Team's analyses assumed the capability was as specified by the designer. A peak pressure of 0.2 megapascal absolute (29 pounds per square inch absolute) was calculated for a hot leg break as compared with a design pressure capability of 0.25 megapascal absolute (36 pounds per square inch absolute). At the predicted peak pressure loading, the concrete and steel reinforcement are calculated to respond elastically without cracking.

The design margin was analyzed. If the internal pressure were to increase to the design value, then cracking of concrete through the partial wall or floor thickness would be predicted to commence near the inner and outer concrete surfaces. The cracking would be restricted to localized regions of the steam generator compartment ceiling and the region where the steam generator room outer walls join the floor, and at the top of the interior walls in the corners of the steam generator compartment. Neither the reinforcing steel or steel liners were calculated to fail at these pressures. Gross structural failure was predicted to require pressures of approximately 0.5 megapascal (73 pounds per square inch) compared with the 0.25 megapascal (36 pounds per square inch) specified by the designer.

Only a small portion (one fourth) of the steam formed during the first minute after a break was predicted to condense in the bubbler-condenser tray water pools. This characteristic was further illustrated by analyses that assumed no steam condensation took place inside the bubbler-condenser trays. For this case, (no steam condensation in the bubbler-condenser trays) the peak pressure loading calculated for the hot leg break did not exceed the design pressure. This results from the large ALS volume, that was increased relative to the VVER-440 model V230. The pressure inside the air traps initially lags behind the steam generator compartment pressure by about fifteen seconds, a function of the size of the check valves leading to the air traps.

After the sprinkler system inside the steam generator compartment begins spraying water droplets (one minute after the break) the pressure inside the steam generator room was calculated to be reduced to subatmospheric in less than 16 minutes, depending upon the spray system flowrate. The long term heat removal capability [involving 1) high pressure coolant injection and low pressure coolant injection into the primary coolant system, 2) steam formation which is condensed by the sprays; and 3) the condensed steam plus water exiting the pipe break cooled in the emergency core cooling system heat exchangers] was calculated to be effective in removing the core decay heat. If the sprays fail to

activate, the pressure would decrease to subatmospheric in about forty minutes due to the condensation of steam upon structures and equipment. If the sprays come on, but fail to cycle off when the pressure decreases below the setpoint of 0.085 megapascal absolute (12.3 pounds per square inch absolute), then the residual air remaining in the localization compartments is predicted to limit the minimum pressure to about 0.075 megapascal absolute (10.8 pounds per square inch absolute).

In summary, following the double-ended rupture of a primary coolant pipe, the pressures predicted inside the accident localization volume are not calculated to exceed the specified design pressure capability. Gross structural failure of the localization system boundary was not predicted. Over the short term and the long term, adequate core cooling is possible if the emergency core cooling system functions as designed.

3.1.2.3 VVER-1000 Model V320

The double-ended guillotine rupture of a primary coolant pipe is the design basis accident for the VVER-1000. All VVER-1000s have an emergency core cooling system consisting of passive accumulators, high pressure injection pumps, and low pressure injection pumps. The containment building incorporates water sprays to decrease the pressure following a loss-of-coolant accident. The timescale for the reduction of the containment pressure by the spray system following the blowdown phase is important in determining the overall leakage from the containment.

The thermal hydraulic response of VVER-1000 model V320 primary systems and pressure loadings inside the containment were calculated for the cases of breaks in the inlet (cold leg) and in the outlet (hot leg) piping. A structural response analysis was not performed of the containment building to these loadings.

The pressure loadings from the formation of steam resulting from the depressurization of the primary system after a cold leg break assuming emergency core cooling system activation were calculated to remain below the containment design pressure. A peak pressure of 0.32 megapascal absolute (46 pounds per square inch absolute) was calculated relative to the design pressure of 0.51 megapascal absolute (74 pounds per square inch absolute).

For the case of a hot leg break, lower break flowrates result in initially lower rates of containment pressurization during the first thirty seconds following break inception. However, beyond this time, similar peak pressures are achieved for both inlet and outlet pipe failures. Condensation upon the structure has a small effect in decreasing the value of the maximum pressure prior to the injection of water droplets by the containment sprays.

With operation of the containment sprays as designed, (i.e., assuming the injection of water droplets begins one minute after the pipe break occurs) the pressures inside the containment were predicted to be reduced to near atmospheric in less than twenty minutes.

In summary, following the double-ended rupture of a primary coolant pipe, the pressure inside the containment should remain below the design pressure. With as-designed operation of the containment sprays, the pressure should be reduced to near atmospheric about twenty minutes after the break occurs. The calculations indicate that there are short periods of time when voiding occurs in the reactor core, but this by itself should not be a reason for concern. The ECCS system and the primary system water inventory are of about the same capabilities as that in a VVER-440 model V213, but the amount of heat to be removed is increased by a factor of 2.5. The margins in VVER-1000s are thus smaller than those in the VVER-440 model V213s.

3.1.3 Steam Generator Internal Manifold Failure

The design of the horizontal steam generator in VVERs includes provision for access by a person to inspect the manifold. Failure of the manifold cover or the manifold itself during reactor operation provides a pathway for primary

coolant to enter the secondary system. An analysis was performed of the loadings on, and response of, the steam generator shell as well as the accident localization volume for the case of a postulated large break in the manifold wall.

Primary-to-secondary loss-of-coolant accidents provide a direct release pathway for radioactive coolant and radionuclides through the turbine atmospheric dump valves. Entry of primary fluid into the secondary portion of the shell could result in pressurization of the secondary shell internal volume which might result in shell rupture and the simultaneous blowdown of the primary and secondary systems into the accident localization volume.

The analysis assumed a rupture of the hot leg manifold that delivers fluid from the reactor pressure vessel to the steam generator in the VVER-440 model V213. The rupture was assumed to have a configuration similar to a tear along the height of a vertical manifold. The effective flow area of the break is thus assumed to be limited by the cross sectional area of the two primary coolant system pipes connected to the steam generator. Instead of modeling the primary coolant system response for the specific location of the manifold failure, results from the primary coolant system analysis of the double-ended hot leg pipe break were used to approximate the conditions of fluid exiting the break immediately after rupture. The response of the steam generator shell to internal pressurization was examined for an idealized shell configuration without penetrations. The effects of penetrations such as the manifold, steam collector, feedwater line, inspection viewport, as well as the weldments at the head-to-cylinder interfaces were not considered.

The pressure loadings inside the steam generator shell were found to be dependent upon the effectiveness of venting of the steam formed from flashing depressurization of primary coolant through the steam collector and the main steamline. Pressures immediately following failure were calculated for the cases of venting of pure steam through the steamline and no steam venting. For the case of pure steam venting, the pressure was calculated to rise by 1.6 megapascal (230 pounds per square inch) to a peak of 6.3 megapascals absolute (910 pounds per square inch absolute) within one second of the break inception. For the case of complete blockage of the collector or steamline, the pressure was calculated to rise to the primary system pressure immediately upstream of the break. For the assumed primary system conditions, this corresponds to a pressure increase of 3.8 megapascals (700 pounds per square inch) to a peak of 9.5 megapascals absolute (1,400 pounds per square inch absolute) in about one second. Without consideration of the effects of penetrations and weldments, failure of the steam generator shell would require an internal pressure of 16.5 megapascals absolute (2,390 pounds per square inch absolute) which exceeds the nominal primary coolant system operating pressure of 12.5 megapascals (1,810 pounds per square inch absolute) and thus provides a 25 percent margin to failure. At the indicated failure pressure, the failure mode would be a longitudinal tear along a portion of the length of the steam generator cylindrical shell between the elliptical head and the greater thickness of the central part of the cylinder. This pressure retaining capability would be reduced due to the effects of penetrations and as a result of any weld defects, for example.

Pressure loadings inside the ALS were predicted for the case in which gross rupture of the steam generator shell occurs. The loadings were calculated only for the period immediately following manifold and shell failure.

If shell failure should occur, then there would be rapid steam generation and a pressure spike inside the steam generator ALS volume resulting from the flashing of the secondary water initially residing inside the steam generator down to the ALS volume ambient pressure. This process is calculated to take place in approximately one second. The pressures in the ALS volume were predicted not to exceed the design value for the VVER-440 model V213, assuming that the bubbler-condenser trays inside the bubbler-condenser tower function as-designed. Due to the short interval for steam formation to occur, the velocities of steam and air through the bubbler-condenser trays exceed those characteristic of large pipe break loss-of-coolant accident conditions, which involve longer times (on the order of fifteen seconds). Gross structural failure of the accident localization compartments was not predicted.

The Soviets have analyzed a beyond design basis leak equivalent to a 100 millimeter (3.94 inch) pipe diameter at the top of the manifold, under the inspection hatch. This is a model of an accident such as occurred at Rovno when over

one-half of the bolts that are used to hold the manifold upper cap to the manifold failed. For this sequence, the effective flow area from the failure region to the steam volume of the steam generator is severely restricted by the limited clearance between the manifold and the steam generator shell where the manifold enters the appurtenance on the upper portion of the steam generator shell that contains the inspection hatch. Radiation detectors in the secondary system signal the leak, allowing the unit to be shut down.

In summary, the steam generator shell was not predicted to rupture as a result of the internal pressurization experienced immediately following the failure of the internal hot leg manifold. If shell failure should occur due to effects not accounted for in the structural analysis, then the pressures resulting from the rapid flashing of secondary side water initially residing inside the steam generator are predicted to remain below the Soviet design pressure and not to cause gross structural failure of the accident localization volume. Effects on the secondary system components of the pressurization resulting from this postulated event were not considered.

3.1.4 Potential Consequences of Reactor Pressure Vessel Failure

3.1.4.1 VVER-440 Model V230

The response of the VVER-440 model V230 ALS to a hypothesized failure of a reactor pressure vessel circumferential weld in the core region was analyzed. The VVER-440 model V230 reactor pressure vessel steel and, more importantly the weld metal, contain copper and phosphorous in amounts that make the material susceptible to radiation-induced embrittlement. This could potentially lead to a pressurized thermal shock induced failure. The calculation assumed that the vessel failed by the formation of a crack that propagates around the full reactor pressure vessel wall circumference. Failure of the reactor pressure vessel would result in an effective break size significantly above the design basis of the VVER-440 model V230 accident localization system.

The pressure loadings and structural response were calculated for VVER-440 model V230s. The pressure loading calculations do not consider the effects of enhanced leakage or structural failure that could reduce the peak pressures attained. Structural response calculations were carried out for the lower portion of the reactor shaft beneath the vessel and do not consider the potential failure of seal penetrations.

Following a postulated reactor pressure vessel failure due to pressurized thermal shock, the blowdown rate is initially restricted by the flow area of the gap between the reactor pressure vessel outer wall and the inner surface of the annular water shield tank. This area is equal to 1.9 square meters (20 square feet) and is equivalent in size to a 1.5 meter (3.9 foot) diameter circular hole with unidirectional flow. Due to the small volume inside the lower portion of the reactor shaft combined with the small flow area of the vertical passageway connecting the lower shaft region with the steam generator compartment, very high pressures were calculated inside the lower shaft.

A pressure of 2.5 megapascals absolute (360 pounds per square inch) was calculated. This significantly exceeds the design pressure of 0.2 megapascal absolute (29 pounds per square inch absolute) within a second after breach formation. The lower portion of the reactor shaft was not calculated to undergo gross structural failure at the indicated peak pressure loading. Shaft wall failure is predicted at a somewhat higher pressure of 3.0 megapascals absolute (440 pounds per square inch absolute).

In the absence of structural failure of the reactor shaft or significantly enhanced venting through some other means following reactor pressure vessel failure, a large pressure differential will exist across the reactor pressure vessel equal to the difference between the lower shaft pressure beneath the vessel and the pressure present above the S-seal. These forces associated with the resulting pressure differential will lift the reactor pressure vessel. The reactor pressure vessel is not held down upon the vessel support but merely rests under its own weight upon the support flange atop the annular water shielding tank. The upward forces are significantly greater than required to physically lift the reactor

assembly off of its support. As the reactor pressure vessel is raised upward, the primary piping would be restricted by the concrete reactor shaft causing the piping to bend in the nozzle region. If the bending stresses in the piping locally exceed the ultimate tensile strength, then the pipes will fail in a tearing mode. The upward force upon the reactor pressure vessel is predicted to attain a maximum of 24 meganewtons (5.4 million pounds) which is slightly less than the value of 26 meganewtons (5.8 million pounds) corresponding to failure of the piping. Thus, the peak upward force is almost equal to the piping failure threshold and the tearing off of the twelve primary coolant pipes cannot be precluded in this sequence. If piping failure occurs, then failure of the S-seal will also occur as the vessel moves upward. The kinetic energy of the upward moving vessel may exceed that which is required to fail the holddown of the protective dome located over the reactor vessel. Release of steam directly into the reactor bay would cause damage to reactor bay structures and allow for release of radioisotopes to the atmosphere.

In summary, circumferential failure of a VVER-440 model V230 vessel weld was predicted to result in very high pressures in the lower portion of the reactor shaft beneath the vessel. The pressures are not predicted to cause gross structural failure of the shaft concrete wall. If structural failure does not mitigate the pressures in the lower shaft region, then the corresponding upward forces applied to the reactor vessel indicate the possibility of the reactor assembly being physically lifted off of its support, the primary coolant pipes being torn off of the vessel, and upward ejection of the vessel. If the protective dome is failed, then large direct blowdown of steam will take place into the reactor bay accompanied by the possibility of damage to bay structures and the release of radioisotope into the atmosphere. This situation would result in an uncooled core being directly exposed to the atmosphere with no barriers to prevent the release of radioisotopes.

3.1.4.2 VVER-1000 Model V320

The response of VVER-1000 model V320 containments to a hypothesized circumferential failure of the reactor pressure vessel in the core region was analyzed. For the VVER-1000, the nickel in the reactor pressure vessel steel (put there to increase its strength) as well as the presence of copper and phosphorous combined with a narrow water gap between the fuel and vessel and an attendant high, fast neutron flux may give rise to a wall embrittlement problem. Failure of the vessel would result in an effective break size significantly above the design basis of the VVER-1000 containment. The "AA" grade steel used on newer VVER-1000 model V320 is reported to have improved properties that reduce radiation induced embrittlement.

Pressure loadings and structural response analyses were carried out. The pressure loading calculations do not consider the effects of enhanced leakage or structural failure which could reduce the peak pressures. Structural response calculations were carried out only for the lower portion of the reactor shaft and the containment basement directly beneath the vessel. The failure of seal penetrations was not considered.

Pressures inside the lower part of the reactor shaft were predicted on the basis of the flow of steam, principally through the horizontal passageway leading to the surrounding mostly annular compartment. The passageway provides access for the vessel inspection machine and is covered with a concrete slab during normal operation. The slab may be expected to be displaced from its normal position before the shaft concrete walls fail thereby providing the major vent pathway from the lower shaft. The effects of the simultaneous dispersal of water which would restrict the steam flow were not considered. Following the hypothesized reactor pressure vessel failure, the rate of blowdown of the primary system would initially be limited by the effective flow area of the downcomer gap between the core barrel outer surface and the inside of the vessel wall. The gap flow area in the core region is 2.7 square meters (29 square feet) and the effective break area equal to twice this value or 5.4 square meters (58 square feet). The break area is equivalent to a 2.6 meter (8.5 foot) diameter circular hole with unidirectional flow. This effective break area, restricted by the downcomer size, is smaller than that corresponding to the gap between the reactor vessel and the surrounding shaft wall and is also less than the total flow area of the eight primary coolant pipes.

The primary system was calculated to depressurize during the first 2.5 seconds following the beginning of reactor pressure vessel failure due to the large effective break area. The pressure inside the lower shaft region is calculated to rise to a peak value of 4.0 megapascals absolute (580 pounds per square inch absolute). Although the lower shaft pressures were calculated to significantly exceed the design pressure of 0.51 megapascal absolute (74 pounds per square inch absolute), the pressures directly applied to the containment cylindrical shell and elliptical dome remained below the design value. The containment design pressure exceeds the equilibrium pressure attained following depressurization of the entire primary system inventory inside the closed volume of the containment shell. However, the lower boundary of the reactor shaft region coincides with the elevated containment basemat. Consequently, the pressures achieved in the lower part of the shaft are directly applied to the containment bottom boundary interior to the shaft wall. If gross structural failure of the basemat were to occur, then steam and fission products would be released directly into the unsealed lower part of the reactor building beneath the containment.

A shaft pressure of 2.9 megapascals absolute (420 pounds per square inch absolute) was calculated to cause gross structural failure of the lower shaft concrete walls. The additional relief path area resulting from shaft failure would be expected to mitigate the pressures such that the peak pressure in the lower shaft would be less than the 4 megapascal (580 pounds per square inch) value obtained from the pressure loading calculation. The floor (basemat) in the shaft region was not calculated to fail for pressures below 4.9 megapascals absolute (710 pounds per square inch absolute).

VVER-1000 model V320 reactor pressure vessels are not held down. The reactor pressure vessel rests, with an annular support structure, atop a ledge recessed in the shaft wall. Should the reactor pressure vessel fail, the upward differential pressure force across the vessel would rapidly exceed that required to lift the vessel off of its support. Although the upward forces across the vessel are several times those needed to levitate the vessel off of its support, the restraint provided by the primary coolant pipes was calculated to be sufficient to preclude upward motion of the vessel. Consequently, failure of the reactor pressure vessel was calculated not to result in failure of the attached piping and motion of the vessel upward out of the lower shaft region.

In summary, a postulated circumferential failure of a VVER-1000 model V320 reactor pressure vessel was calculated to result in high pressures in the lower portion of the reactor shaft beneath the vessel. The pressures were calculated to be sufficient to cause gross structural failure of the shaft concrete wall enhancing the venting of steam from the shaft region. The calculations indicate the containment boundary should not experience gross structural failure (the effects of pressure loadings on penetrations, etc., was not considered). The upward forces applied to the reactor pressure vessel would be sufficient to lift the vessel off of its support, but the eight primary coolant pipes should restrain upward motion, preventing upward ejection of the vessel out of the shaft region.

3.1.5 Small Pipe Break Accidents

A unidirectional flow break with an equivalent diameter of 10 centimeters (3.9 inches) through a 3.2 centimeter (1.3 inch) orifice is the design basis accident for the VVER-440 model V230.

Pressure loadings and structural response calculations were performed for the VVER-440 model V230 assuming a unidirectional 20 centimeter (7.9 inch) diameter break. The effects of steam condensation due to operation of water droplet sprays was not modeled. The pressures inside the ALS were calculated to exceed the threshold of 0.18 megapascal absolute (25 pounds per square inch absolute), at which the first vent valve opens, about 15 seconds after the break. The pressure was calculated to continue to rise (even with the vent valve open) to the design pressure of 0.20 megapascals absolute (29 pounds per square inch absolute). The remaining eight valves are designed to open at this pressure. If the break is in the upper reactor pressure vessel shaft region between the protective cap and the S-seal, then rupture discs in pipes connecting this volume to the steam generator open. The pressure was calculated to drop in about one minute after the valves open to a level at which the valves are designed to close.

When the pressure exceeds 0.15 megapascal absolute (22 pounds per square inch absolute), the localization volume concrete was calculated to undergo cracking in localized regions partially through the wall or floor thickness. Cracking begins in the outer walls, roof, and internal diagonal walls of the steam generator compartment. At the peak pressure loading, extensive localized cracking is indicated throughout the structure, but gross structural failure was not predicted to occur. Neither the steel reinforcement nor the steel liners were calculated to fail at these pressures.

In summary, the pressures inside the VVER-440 model V230 accident localization volume following a 20 centimeter (7.9 inch) diameter break with unidirectional flow were calculated to rise to the design basis pressure. All of the vent valves were calculated to open and to remain open for over one minute permitting the temporary escape of steam and radionuclides from the ALS. Gross structural failure of the accident localization system boundary was not predicted, but extensive cracking of the concrete near the concrete inner and outer surfaces was indicated over large areas of the walls and floors.

3.1.6 Control Assembly Ejection

This event was postulated to be initiated by the ejection of a single control assembly being driven upward with sufficient force to rupture the pressure boundary of the control assembly drive mechanism housing mounted atop the reactor vessel. The design basis event for VVERs assumes this happens five times during the life of the unit for purposes of transient analysis of the reactor pressure vessels and once for analysis of the ALS pressure boundary.

The penetrations in the upper head region are routinely (in some locations, at least once a month) monitored for coolant leaks in some countries. In VVER-440s a failure of the primary coolant pressure boundary in the upper head region would initiate a blowdown into the upper portion of the reactor shaft above the S-seal and beneath the bolted down protective dome. Failure of the seals between the dome and the building floor would result in the direct discharge of primary coolant into the reactor bay above the operating deck thereby allowing radioactive material to escape into the environment. In addition to initiating a loss-of-coolant accident, the ejection of a control assembly inserts a total reactivity increase of 50 cents at a ramp rate of 20 dollars per second.

Pressure loadings inside the upper shaft region and the reactor bay were calculated for the postulated ejection of a single rod in the VVER-440 model V230. The breach in the control rod drive housing was assumed to have an effective flow area equivalent to a 9 centimeter (3.5 inch diameter) circular hole corresponding to the size of the shroud tube surrounding the driveline and penetrating the upper head. Condensation of steam due to the operation of water droplet sprays was not modeled. The structural response of the protective cap was not explicitly calculated. The modeling of the primary system did not include the additional energy generation that would be associated with the reactivity transient resulting from the postulated control assembly ejection event.

The pressure inside the upper shaft region (above the S-seal) was calculated to rise to the design pressure of 0.2 megapascals absolute (29 pounds per square inch absolute) in about five second after the break. The pressure increase was predicted to be temporarily halted by the opening of the rupture discs between the upper shaft and the pump and steam generator volumes of the ALS. The discharge of the pressure from the upper shaft region to other volumes of the ALS was predicted to cause the pressure in the upper shaft to decrease to 0.17 megapascal absolute (25 pounds per square inch) about 20 second following break initiation and then to begin to increase again. After the 110 seconds, the ALS pressure in the steam generator compartment was predicted to reach 0.18 megapascals absolute (26 pounds per square inch), the pressure at which the first vent valve is designed to open. The pressure in the upper shaft at this time was calculated to reach 0.21 megapascal absolute (50 pounds per square inch absolute), or slightly over the design pressure. This pressure was not predicted to cause failure of the walls.

Jet impingement from the break, and the temperatures and humidity may be expected to adversely effect electrical cables and equipment, unless the cables and electrical equipment are environmentally qualified. In these conditions,

the extensive quantity of cables and equipment in the upper shaft region presents a significant environmental qualification problem. No environmental qualification requirements or actions appear to exist.

In summary, the ejection of a single control assembly that ruptures the control assembly drive mechanism housing would result in the discharge of primary coolant and fission products into the upper shaft/dome interior region above the upper head. The pressure inside the upper shaft region was calculated to slightly exceed the design pressure. Opening of the rupture discs separating the upper shaft volume from the rest of the ALS would lead to pressures that would cause a vent valve to open, venting directly to the atmosphere. The pressures could cause local cracking, but not gross structural failure. Jet impingement and environmental conditions could result in electrical equipment not functioning as designed.

3.2 Reactivity Events

Analysis Appendix H, "Reactor Transient Analyses", summarizes the results of analyses of a spectrum of postulated and hypothetical reactivity events.

Reactivity transients occur when either positive or negative reactivity is introduced into the reactor core, thus altering the neutron multiplication from the steady state value of one. In general, positive reactivity insertion from any source has the potential to drive reactor power to damaging levels. Protective means are provided to negate both the reactivity addition and its thermal effects, and the success of these protective measures is ultimately determined by the size and rapidity of the reactivity addition versus the power and speed of the countermeasures. Furthermore, reactivity accidents which start under shutdown conditions tend to be more severe because the reactivity addition may be complete before the surge in power reaches levels that activate the protection system. With the reactor already at power, trip points are violated as soon as reactivity addition begins, and counter measures are in progress before the insertion that initiates the event is complete.

Three types of accidental reactivity insertions for VVERs were considered. These were runaway control assemblies, unintended dilution of soluble poison, and a "cold slug" (injection of colder water than that initially present in the reactor vessel). Failure mode and effects analyses (FMEAs) were used to postulate failure modes systems, where the particular modes of failure were not known on some other basis.

3.2.1 Reactivity Insertion by Control Assembly Drive Mechanisms

The currently used VVER-440 control drive mechanism is referred to here as the "second modification." This drive was first tested on one of the VVER-440s at Novovoronezh, and was later installed in other VVER-440s, in Czechoslovakia and elsewhere. The predecessor of this drive achieved emergency shutdown by decoupling a captive pinion from a moveable rack, thus dropping the rack by gravity and thereby inserting its attached control element into the core. The rate of insertion is limited by hydraulic resistance to a trailing piston. One Soviet description of this drive refers to the disengagement of the rack from the pinion as the "least reliable element of the design."

The "second modification", the subject of the following discussion, eliminates this reportedly unreliable process of rack/pinion decoupling. Instead, in a scram, the rack and pinion are left coupled, and the electric power to the drive motor is switched off. By choosing a sufficiently small ratio for the drive gears, the weight of the control element is sufficient to overcome the frictional drag of the deenergized motor, and the control assembly drops into the core, spinning the gears and the motor rotor as it proceeds. The rate of insertion is controlled by a centrifugal friction-type regulator. Details are in Background Appendix C, "Reactivity Control."

Two specific considerations that result from unique features of the "second modification" were identified.

- (1) Although described as a "nonreversible scram," an AZ-1 shutdown is in fact reversible, if electric power should be reapplied to the terminals of the drive motor. This could happen through failure of a switch or other control element, or through an electrical short. A conventional scram in a U.S. reactor, or in other designs of Soviet rod drives, does not have this element of reversibility. This may introduce scram failure modes not found in other reactors, and increase the probability of failure to scram. Because of the possibility of scram reversal, possibilities appear to exist for common mode failures capable of stopping a scram or pulling inserted rods back out of the core.
- (2) The second consideration involves the relatively slow speed of emergency control assembly insertion, i.e. in the VVER-440, 20 to 30 centimeters per second (about 10 inches per second). The process used is inherently slow. Not only is insertion speed limited by gear friction working with a mechanical advantage, but a centrifugal regulator is added that slows the process down even more. This latter device was apparently added to protect the motor and the gear and rack teeth from the effects of overspeed, including the abrupt stop when full insertion is achieved. A rod in free fall with 0.5 g acceleration will achieve a five-foot insertion in 0.8 seconds. At 10 inches per second, for the design under discussion, 6 seconds would be required for the same travel. There are credible reactivity insertions (e.g., losses of load), and other emergency conditions whose speed of onset indicated the need for faster shutdown capability. This was considered for a necessarily limited number of transients.

The effect of the relatively slow insertion speed, were analyzed using point kinetics models of a VVER-440 and VVER-1000 and reactivity feedback coefficients obtained from Soviet reports (also see Background Appendix L, "VVER Neutronic Parameters") and the RETRAN-2 code. In most cases, core temperatures did not increase significantly.

3.2.2 Reactivity Insertion by Removal of Soluble Poison

Gross reactivity management is achieved by a controlled concentration of soluble poison (boric acid) in the cooling water. Particularly for a new fuel loading, this soluble poison is required to maintain reactivity in a range finely controllable by the control drive mechanisms and is essential to the provision of adequate shutdown margin.

The boron concentration is monitored by detectors that measure the neutron flux attenuation across a standard path length of the circulating solution, and is maintained in the desired range by injection of a concentrated boric acid solution or of pure water, as the case demands.

The reactivity changes producible by planned or unplanned alterations of boron concentration are potentially large, but the maximum rate of change is small. The potential is present for changes too large to be countered by other means, but changes do not take place over times too short for possible corrective action. The considerations regarding malfunctions of the boration control system are thus different from those concerning motor driven control mechanisms. A key consideration is the postulated initiation of a reactivity transient by boron dilution due to operator error.

The boron dilution accidents were simulated using a simplified point kinetics model. The scram system was considered to be inactive. For VVER-440s and VVER-1000s, potential core damage could result in a short time unless the operator takes appropriate action.

3.2.3 Reactivity Insertion Through Temperature Changes

A negative temperature coefficient of reactivity provides an inherent safety mechanism against unrestrained increases in power. It also provides a potential means for unplanned reactivity additions through incursion of lower temperature water into the core.

Means exist for planned or unplanned introduction of colder water into the core. The usually postulated means to do this is startup of a previously inactive pump that rapidly injects a pipeline full of cold water into the core. Injection pumps may also draw water from various unheated or inadequately heated sources. This water may be nonborated as well as cold, providing two sources of reactivity when it enters the reactor.

The overcooling event was simulated as a 50° Centigrade (90° Fahrenheit) step reduction in the core inlet temperature. In each of the VVER-440s and VVER-1000s, the reactivity feedback was such that no fuel damage was predicted even with failure to scram.

3.3 Response to Loss of Electric Power

Both loss of offsite power and a case postulating loss of all AC power were considered. Additional information concerning the potential consequences of loss of electrical power to various systems and components is provided in Background Appendix M, "Loss of Electric Power."

3.3.1 Loss of Electrical Power to Primary Coolant Pumps

Loss of power to one or more pumps is a frequently analyzed transient in VVERs. Unlike United States PWRs, the loss of one or more pumps at full power in VVERs does not necessarily lead to a scram. The AZ-1 signal (scram) is activated only if power is lost to more than half of the pumps. Otherwise, the AZ system will reduce the reactor power until a power to flow ratio of one is obtained. In all VVER designs, loss of power to all pumps is not considered by the designers as a credible accident.

Several loss-of-flow events caused by pump trips were considered. In Analysis Appendix H, "Reactor Transient Analyses," a simplified point kinetics model was used to analyze events where the flow was reduced by 50% followed by failure of the AZ system. The simplified model predicted no core damage. Analysis Appendix D, "Fuel/Core Temperature Analyses," simulated a single locked rotor event and full pump failure.

3.3.2 Loss of Off-Site Power

Soviet analyses of loss of off-site power simultaneously with a loss of coolant accident were examined. VVERs are designed to handle this event, and assuming that all systems perform properly, no damage was predicted to result. However, avoiding core damage requires that a number of systems operate as designed. VVER-440 model V230s (that have canned rotor pumps) are designed to accommodate the effects of loss of off-site power by the use of two inertial power generators (IPGs) each of which are connected to two primary coolant pumps. The IPGs extract energy using the inertia of the main turbine and generator as they coast down. The IPGs also continue to be driven by the action of the turbines as decay heat-generated steam flows from the reactor following a scram.

A possible failure mode of the IPG system results from the method of operating the primary coolant pumps during reactor start-up and rise to full power. Normally, two groups, each with two primary coolant pumps operate, one group from each of the two independent IPGs. However, transfer breaker switches are provided to connect the four primary coolant pumps normally driven from the IPG to the preferred off-site AC power bus. Since the operating procedure requires connecting the normally IPG-driven primary coolant pumps to the off-site power for start-up, the potential exists for inadvertently leaving these pump motors connected to the off-site power during the remainder of the operating cycle. In this condition, loss of the off-site feeder would result in a rapid coastdown of all six canned rotor pumps.

For the VVER-440 model V213 and the VVER-1000, stored energy in the primary coolant pump drive is used to extend the coastdown time to 120-180 seconds. This provides time for the diesels to start and begin to pick up load upon loss of offsite power. However, both VVER-440 model V213s and VVER-1000s switch from the main feeders to the

preferred off-site feeders on loss of power. If the preferred off-site feeder remains connected to the system load, and the coolant pump breakers or the preferred off-site feeder circuit breakers do not sense and open on reverse current, pump coastdown time can be greatly shortened due to generator action by the motors.

All of the above sequences assume that a scram occurs on loss of AC power. However, some failure modes can be postulated that result in failure to scram on loss of power. That failure mode will be considered independently in another section of this report, but was not discussed with the above scenarios because it would require multiple failures.

3.3.3 Loss of AC Power, with Failure of the Diesels to Start

For loss of AC power with failure of the diesels to start events, three cases were considered.

1. Thermal and inertial energy is available to provide long coastdown of the primary coolant pumps. No LOCA occurs.
2. Stored energy not available. No LOCA occurs.
3. A LOCA occurs along with loss of AC power.

The reactor was assumed to scram, as it should on loss of AC power.

In case 1, the damage should be minimal. VVER steam generators are large and have a large water inventory. The coastdown of the pumps provide adequate cooling to reduce the temperature. Thereafter, natural circulation should provide adequate cooling. The secondary system removes the heat by boiling the contents of the steam generator with the steam being released through atmospheric dump valves. Ultimately, power must be restored, but considerable time is available to provide backup power.

For case 2, the same conditions prevail as in case 1, but the coastdown is greatly accelerated by the postulated inability to disconnect the primary coolant pump motors from a dead grid that is still connected to its load. In this case, the decay heat generated in the first 1 to 2 minutes of the AC power outage cannot be removed by the pump coastdown flow. Removal of this heat by natural circulation may allow the temperature to rise sufficiently high to cause fuel element damage. Any breach of integrity between the primary and secondary systems would allow fission products to escape with the venting of steam to the atmosphere.

For case 3, a LOCA with loss of offsite power and failure of the diesels to start, it appears that core damage should be expected. The extent of the damage and the degree to which fission products are released to the atmosphere depends on the response of the system in providing gravity flooding and the time required for restoration of AC power through diesels or external means.

There are variations of these three cases where the progress of the accident may be altered by operator actions or inactions, or by the existence of some additional off-normal condition that exacerbates the problem. For example, in case one, even though several hours are usually available to restore power, it may not always be possible in remote areas.

3.4 Response to Seismic Loads

VVERs have been, and are being, constructed in regions of significant seismic intensity. These include the two Armenia VVER-440 model V230 units in the Soviet Union (that were shut down in early 1989), the four VVER-440 model V230 units, the two VVER-1000 model V320 units at Kozloduy in Bulgaria (Unit 5 is in operation, Unit 6 is under

construction), two VVER-1000 model V320s under construction in the Crimea, and two VVER-440s model V318s under construction at Juragua in Cuba. It has been announced that units planned for construction at Krasnodar (a region of high seismicity) have been cancelled. Seismic loadings have the potential to cause failure of buildings and structures (possibly breaching the accident localization system or containment boundary to damage piping or equipment with the potential for initiating a primary or secondary loss-of-coolant accident), and to damage balance of plant systems possibly creating a loss-of-heat sink condition.

Analyses have been carried out of the response of the VVER-440 model V230 and VVER-1000 model V320 using seismic loading conditions for the Kozloduy site in Bulgaria. An input ground acceleration history with a peak horizontal ground acceleration of 0.15 g was used. This is representative of an earthquake having an intensity of eight on the MSK-64 scale. The displacements and accelerations of building structures and primary coolant system components were calculated using a linear dynamic model ("stick model"). The predicted maximum accelerations were compared with ultimate seismic capabilities determined for the structures and equipment. These levels corresponding to the ultimate capabilities were defined in terms of the material yield stress multiplied by a ductility factor characterizing the seismic loading level that causes failure. Details of the seismic response calculations and results are contained in Analysis Appendix F, "Seismic Analyses".

3.4.1 VVER-440 Model V230

The original design of the VVER-440 model V230 did not consider severe seismic events. Following the Vrancea earthquake in Romania on March 4, 1977, features were added to enhance the seismic resistance of the VVER-440 model V230s at Armenia in the USSR and the VVER-440 model V230s at Kozloduy in Bulgaria. The lack of seismic considerations in the original VVER-440 model V230 design coupled with the impact of the Vrancea earthquake increased the interest in structures or equipment in VVER-440 model V230s, which were not designed for the effects of earthquakes. The December 7, 1988 earthquake in Armenia has resulted in a further increase of concern for seismic design and in characterizing the seismicity of various sites.

The response of the primary coolant loops was calculated for the case where piping and components are restrained by hydraulic shock absorbers as well as the case where the shock absorbers are absent. The effects of the pressurizer, quench tank, and piping connecting the pressurizer with the primary loops were not considered.

When restrained by hydraulic shock absorbers, the primary coolant loops are calculated to have an ultimate capability of 3 g peak ground acceleration. However, individual shock absorbers may begin to fail to provide restraint at ground accelerations of 0.2 g (slightly above the reference 0.15 g loading). In this case, the primary coolant loop motions would still be partially limited by the remaining shock absorbers. If the shock absorbers were not present, the ultimate capability of the primary coolant system would be decreased to 0.2 g ground acceleration. Without restraint, the indicated failure location would be the junction between primary coolant pipes and the nozzles on the reactor pressure vessel.

Seismically induced accelerations increase with elevation in the buildings. Because the ALS volume and primary system components are located at grade level, the peak accelerations of the accident localization volume structures are calculated to be essentially equal to the maximum ground acceleration. The ultimate capabilities were calculated to exceed the assumed peak ground acceleration of 0.15 g. However, specific structures are predicted to have small margins to failure. The annular water shielding tank supporting the reactor pressure vessel is calculated to have an ultimate capability of 0.2 g ground acceleration. This value is roughly equal to the upper limit of horizontal ground accelerations that correspond to an intensity level of eight. Failure of the shielding tank leaves the reactor vessel supported entirely by the twelve primary coolant pipes. The weakest point of the building walls and floors are the upper parts of the prefabricated, reinforced concrete vertical support columns that form the framework of the turbine building outside wall. This reflects the reduced column thickness at elevations above the traveling bridge crane together with

the general increase in local acceleration with height. In this case, the ultimate capability in terms of maximum horizontal ground acceleration is 0.3 g.

In summary, gross failure of the reactor building, turbine building, and auxiliary and control equipment buildings and primary coolant piping is not predicted for the reference 0.15 g maximum horizontal ground acceleration earthquake. However, specific structures were calculated to have a small margin to failure. In particular, the predicted ultimate seismic capability of the annular water shielding tank that supports the reactor pressure vessel is 0.2 g ground acceleration. This value is subject to a large uncertainty and could actually be lower than 0.2 g.

3.4.2 VVER-1000 Model V320

The seismic response of the major walls and floors of the containment and the auxiliary equipment and control building were calculated. The response of the primary coolant loops was calculated for the case where piping and components are restrained by hydraulic shock absorbers as well as the case where the shock absorbers are absent. The effects of the pressurizer, quench tank, accumulators, and associated piping was not considered.

The cylindrical containment is elevated above grade level and is supported by the auxiliary equipment and control buildings. Consequently, the local accelerations at the elevation of the primary system components are more than twice those at ground level. Because the interior walls and floors of the containment are not secured to the surrounding containment shell, the interior structures and the shell exhibit motions having different frequencies.

The predicted capabilities of the structures and components examined exceed the assumed peak ground acceleration. The weakest point in the reactor building was found to be the joint between the base of the containment shell and the containment basemat. Failure of this joint was calculated to require a maximum ground acceleration of 1.15 g.

The primary coolant loops are calculated to have an ultimate capability of 1.85 g peak ground acceleration provided that the major components and piping are restrained by hydraulic shock absorbers. Individual shock absorbers may begin to lose their effectiveness at ground accelerations of 0.3 g or greater. In the absence of any restraints, the primary coolant loops were calculated to have an ultimate capability of 0.36 g peak acceleration.

In summary, gross failure of the reactor building and primary coolant loops was not predicted for a 0.15 g maximum horizontal ground acceleration earthquake. The margin to failure of specific structures and components is greater than that calculated for the VVER-440 model V230.

3.5 Aircraft Impact

The potential consequences of the crash of aircraft at a VVER site are examined in Analysis Appendix J, "Aircraft Impacts".

The assessment was carried out by applying the results of published structural analyses of the impact of aircraft upon containment structures.

A reference aircraft impact loading corresponding to a mass of 20,000 kilograms (44,000 pounds) and velocity of 215 meters per second (481 miles per hour) was assumed in the analysis. This loading is identical to that employed in the Federal Republic of Germany. The conditions also correspond suitably well to the crash of a Mig-23 or Mig-27 Flogger which are multirole combat aircraft in the Warsaw Pact countries.

The results of Western analyses indicate that a minimum thickness of reinforced concrete of 1.8 to 2.1 meters (5.9 to 6.9 feet) is required to withstand the impact of 20,000 kilogram (44,000 pound) aircraft at a velocity of 215 meters per second (481 miles per hour).

3.5.1 VVER-440s

The results of the analyses indicate that the VVER-440 models V230 and V213 would be severely damaged by aircraft crashes. This is the case for a crash into the reactor building. In particular, for a downward directed impact, the aircraft will readily penetrate both the roof of the reactor building and the roof of the accident localization volume. Depending upon the precise crash location, this could initiate a primary system loss-of-coolant accident. Steam or feedwater piping may be ruptured as well, commencing a concurrent secondary loss-of-coolant accident. If sufficient coolant cannot be delivered to the reactor pressure vessel to remove the core residual heat, core degradation and meltdown will occur. If impact occurs on the refueling pool, the pool walls will likely be breached allowing water to drain from the pool volume, possibly precluding the cooling of spent fuel assemblies that may reside in the pool. Spent fuel assemblies may also be directly damaged by debris fragments resulting in the release of radionuclides.

Depending upon the impact point, destruction of the control room and/or emergency control room, of one of the units may occur. No indication was found of measures provided to protect the control room from the effects of fires.

For a crash into the turbine building, the aircraft will readily penetrate through the industrial building type walls and floors. Although all currently operating VVER-440s incorporate two turbine generators per reactor unit, there are critical components that are common to both turbine generators. Loss of the feedwater pumps (there is no auxiliary feedwater system in VVER-440 model V230s) due to crash-induced damage will initiate a complete loss of heat sink accident when the water in the dearator tanks is lost.

Two VVER units are served by a common radwaste building. A crash into the radwaste building can result in the release of liquid activity. Damage to equipment used to extract and concentrate wastes from the primary coolant system may necessitate shutting down of the two units.

An aircraft crash into the tall plant ventilation stack may result in it collapsing or overturning upon one of the neighboring buildings. For specific VVER-440s, the stack location is such that it could fall directly upon one of the accident localization volumes of the two units possibly initiating a primary and/or secondary loss-of-coolant accident. It is also possible for the stack to fall upon the central portion of the turbine building and initiate a partial loss-of-heat sink accident. Alternatively the stack may fall upon the radwaste building.

The two units of a VVER-440 plant are served by a common pumping station. There is no evidence that the pumps are separated by fire retardant walls. If crash-induced damage and fire disable all of the pumps, it could result in a loss of this heat sink for both units.

Similarly, all of the diesel generators serving both units are housed inside the same building. It does not appear that the diesel generators are separated by structural or fire retardant barriers. If a crash results in unavailability of all the diesels, then a concurrent loss of AC power will initiate a loss-of-heat sink accident.

A crash into the switchyard may cause the loss of offsite AC power, depending upon the precise crash location. Crash and fire-induced damage may cause the propagation of electrical faults that result in the failure of electrical equipment at other areas upon the site.

3.5.2 VVER-1000 Model V320

The containment configuration of the VVER-1000 model V320 affords relatively better protection from the reference aircraft impact than the reactor buildings of the VVER-440 models V230 and V213. However, the VVER-1000 model V320 still exhibits vulnerabilities to the effects of an aircraft crash.

For a downward directed impact upon the dome of the containment, the analyses indicate that the aircraft will penetrate both the concrete and steel liner thicknesses. Heavy concrete fragments, aircraft engine parts, and other debris may impinge upon the operating floor above the primary coolant system.

The additional thickness of the floor may prevent penetration of debris into the compartments surrounding the primary coolant loops. The upper part of the reactor assembly is not covered and damage to the drive assemblies may occur initiating a loss-of-coolant accident in this region.

The control room is located near an outer wall of the reactor building adjacent to the turbine building. Penetration into, and destruction of, the control room may occur, depending upon the precise crash location.

All VVER-1000 model V320 units use a single turbine generator. Consequently, an aircraft crash into the industrial-type turbine building can cause a complete loss-of-heat sink accident as a result of damage to a number of components.

All of the units at a VVER-1000 model V320 station are served by a single radwaste building. An aircraft crash into the radwaste building can result in the release of liquid activity. Damage to equipment used to remove and concentrate wastes from the primary coolant system may require shutting down of the station units.

VVER-1000 model V320s have a single tall ventilation stack. The station layout is such that the stack cannot fall upon any of the reactor buildings as the result of an aircraft impact. However, the stack could collapse upon the radwaste building.

Each VVER-1000 model V320 unit is provided with its own pumping station. A crash into the pumping station can initiate a total loss-of-ultimate heat sink for the respective unit.

Each unit is provided with three diesel generators. The diesels for all station units are distributed among several buildings constructed beside and between the reactor buildings for the various units. Each building houses three diesel generators except the first and last buildings. Individual diesels are separated by reinforced concrete walls which may limit the effects of a crash to only one or two diesels. Consequently, emergency AC power would likely remain available at a multiunit VVER-1000 model V320 station.

Similar considerations hold for a crash into the switchyard as were discussed above for the VVER-440 models V230 and V213.

3.6 Environmental Qualification

Environmental qualification is necessary to assure the ability of equipment used in nuclear plants to be capable of withstanding degraded environmental conditions in which it may be required to operate. Environmental degradation may occur from excessive water vapor, chemicals entrained in sprays, fires due to hydrogen evolution, and other similarly disruptive agents. It is important to accident considerations because the degree to which equipment is qualified to continue operation in degraded conditions determines how successfully the progress of an accident can be monitored and mitigated.

The Soviet practices do not appear to include environmental qualification. An example is the moisture and waterproofing of electrical equipment. Provision to protect from moisture is lacking. Examples of these practices are:

1. Junction Boxes:

Pliable gasket seal covers do not protect against gross moisture penetration.

Rigid conduit which receives no specific treatment to prevent moisture entry.

No apparent surface treatments are used to prevent slow moisture-induced degradation from direct insulation resistance loss or slower insulation resistance loss due to fungi growth or other effects.

2. Cable and Conduit:

Electrical and signal cables are run utilizing a variety of installation practices. Many of the techniques used could lead to failure during conditions of environmental degradation.

Where conduit is used as a raceway between building levels separated by floors, the conduit is extended above the floor level by several feet for physical protection and to keep the environments of the two levels isolated. To accomplish the latter, a plastic non-setting sealant is used around the cable at the conduit entrance. Prolonged exposure to heat or water vapor would cause failures--both open and short circuit.

Open cable runs are normally used in some units. The technique employed is to bundle the cable tightly with ties and hang the bundle to ceilings and walls with strap hangers. Flocculent asbestos or fiberglass is sometimes sprayed on these open cable runs to reduce fire hazards. This offers limited protection against flash fires. Flash fires in electrical cables may be a significant problem due to the lack of measures to prevent and investigate potential accident initiators.

It appears that the Soviets do not qualify their cable jacket material against moisture and direct steam impingement as is done in some U. S. installations.

The upper shaft contains the control assembly drives and associated piping. A leak in a head penetration could result in particularly severe environmental conditions.

Based on the experience gained from U.S. environmental chamber tests and similar U.S. industrial accident induced environmental conditions, even equipment built to vapor-proof specifications may fail rather rapidly on continuous exposure to steam and high humidity. A variety of environmental conditions may be expected to exist inside a VVER accident localization compartment during an accident. Examples are live steam and hydrogen burns with resulting high pressure and temperature surges. The ultimate failure modes of instrumentation that has been tested or subjected to similar environments have been attributed to condensation of moisture vapor inside the conduit and instrumentation chassis. Therefore, it is reasonable to expect failure of sensitive instruments in VVERs within a few hours. Ordinary valves and valve actuators could continue to operate indefinitely in a vapor or high humidity environment, but should be expected to fail in a few days in a spray and a few minutes, if flooding is possible, unless designed for these conditions and qualification testing has demonstrated successful operation under such conditions.

When VVER electrical and electronic component installation practices are considered damage from a flash fire is to be expected, based on observed damage to installations in similar situations. Extensive damage with probable loss of function can be expected to the open cables sprayed with flocculent materials. The Soviets appear to have no requirement to control hydrogen concentration and to prevent hydrogen burns. Only limited protection is provided at

some VVERs, with none at others. Extensive tests in the U.S. have provided a considerable body of technology concerning the hydrogen burn problem and solutions.

SECTION 4 BENEFITS FROM EXPERIENCE

This section considers some experiences that relate to the safety of water-cooled water-moderated nuclear power plants, and areas where lessons might be learned from these experiences. The U.S. and Soviet nuclear activities each have provided experiences that could be of potential benefits.

VVERs have provided experience relating to ways to design and operate nuclear powerplants to obtain consistently high plant availabilities, capacity factors, performance, and the resulting plant economics. U.S. experiences concerning design practices associated with identifying and addressing safety considerations could contribute to Soviet designed VVERs.

Soviet supplied fuel operates at a much lower linear power rating than typical Western fuel. While this fuel is not produced to the exacting specifications used to produce fuels in the U.S., it has operated with very low failure rates. The Finns have experienced only eight failed pins in the two Loviisa Units since these units started up. The low fuel failure rate at the Loviisa units and at some other VVER units have contributed to outstanding occupational exposure records.

4.1 Retrofits

The U. S. has an active, continuing system of backfits to older plants of changes that are determined to be important to safety, whereas only a limited program of this type has been implemented in the Soviet Union. Added protection in Soviet units can be achieved in several areas, such as cable separation and fire protection. It is clear that in a number of important areas older model units lack safety related features that exist in newer model Soviet units. The newer model units incorporate design features to mitigate the effects of postulated accidents that were not in older models, such as the ECCS with accumulators, high pressure coolant injection pumps, low pressure coolant injection pumps, storage water, etc., that are used on the VVER-440 models V213, V318 and VVER-1000s. VVER models V213, V318 and VVER 1000s have auxiliary feedwater systems with an independent source of water that have a limited redundancy and backup applicability.

Until very recently, there appeared to have been no Soviet policy to upgrade older units, although some changes were made. For instance, flow limiting orifices have been retrofitted on an ad hoc basis and seismic snubbers are being added to some units. While a need for improvement in older models had been identified by the Soviets and addressed in their newer rules and models, the older models have not been upgraded to satisfy these requirements. VVER-440 model V230s have only a limited makeup coolant capability (100 cubic meters per hour or 433 gallons per minute). There are no indications of plans for these systems to be backfitted, even though all newer VVER-440 model V213 designs have ECC systems. In the U. S., all older plants were required to install ECC systems many years ago, and those that did not were permanently shutdown. Similarly, the VVER 440 model V230s do not have full containments and there are no known plans to improve the containment capability of these facilities. VVER-440 model V230s do not have auxiliary feedwater systems with an independent source of water.

The VVER reactor pressure vessel embrittlement problem is a concern. Known ways to reduce the problem that were implemented on the Loviisa units several years ago only now being reported to be implemented, or considered for implementation on some units in Eastern Europe. This is a considerable delay in taking interim corrective action in such an important area. A potential longer term solution (annealing) has been developed and has been used on Nord-1 in the GDR and Novovoronezh-3 and Armenia-1 in the Soviet Union. These are VVER-440 model V230s (Analysis Appendix A provides details). Plans are to use this procedure in either Kozloduy 1 or 3, or Kola-1 in the Soviet Union. Czechoslovakia is developing a procedure and expects to use it on the VVER-400 model V230s in Czechoslovakia. All U. S. plants have extensive NDT sample monitoring programs, are under scrutiny by the regulatory agency regarding PTS, and are prohibited from operating in hazardous thermal shock conditions.

Other items appear to have not been addressed as subjects for retrofits. For example:

A number of safety considerations exist because the Soviets have not upgraded fire protection measures. All U. S. units are undergoing or have undergone major fire protection upgrades.

Whereas extensive backfit of older U. S. plants has taken place or is underway to improve cable separation and protection, the information available to the DOE team indicates this is not the case in most if not all, VVERs.

Although the design basis includes a number of LOCA events, there do not appear to be any associated requirements for an environmental qualification. Auxiliary feedwater systems are lacking on VVER-440 model V230; auxiliary feedwater systems on other VVERs lack backups. Support systems, such as service water systems, that are required to assure the operations of safety systems are not designed as safety systems and lack redundancy or backup capabilities.

4.2 Plant Performance

VVER performance exceeds that achieved in the U. S. and many western countries in plant capacity factor and unit availability. This is true in spite of apparent difficulties discussed elsewhere in this report. Soviet design practice focus strongly on achieving high plant availabilities. This performance results from the design attention given to achieving high unit availability as characterized by the relative simplicity of design and the greater margins in certain systems such as fuel power rating, steam generator secondary side capacity, and dual turbines for each reactor. These and other features minimize lost electricity generation due to transient and equipment failures.

The average plant capacity factors in the United States are considerably lower than those achieved by countries operating VVERs. While few units in the United States achieve plant capacity factors of 80% or better, most VVERs consistently operate at, or above this level. In 1987, the two VVER at Loviisa both achieved plant capacity factors of over 90%, while refueling and successfully conducting planned maintenance work, that included condenser retubing.

The Soviets and other countries operating VVERs have reduced the time to incorporate new technology pertaining to diagnostics, control and sensors (including the application of computer technology and fiber optic technology) so that available advanced technology is used in operating reactors, and contributes to achieving improvement in plant performance.

Some of the specific areas where experience might benefit others are summarized below:

The Soviets have utilized horizontal steam generators in all their VVER designs. The use of this approach has avoided the serious corrosion problems experienced in vertical steam generators. While the horizontal design has provided some clear advantages to the Soviets, its use has required them to pay the price of larger building structures.

The Soviets have chosen their materials to significantly reduce the background radiation levels in the primary coolant and associated systems.

The Soviets have a capability in the control room of some units to terminate a scram. The capability provided to the operators of Western units to disconnect or terminate the operating safety and protection systems is quite limited. There is an example of a trade-off between the priority given to achieving high plant performance, and the priority given to maximizing the reduction of potential safety related problems. This feature would not be acceptable in U.S. nuclear powerplants.

The Soviets use motor operated valves on the atmospheric steam dump, the turbine bypass, and for many other functions. This is done to enhance plant capacity factor and availability. In Finland, motor operated valves were used originally, but these were changed to spring operated valves, based on instructions from the regulatory authority. France originally used spring operated valves, but changed these to motor operated valves. There is a strong belief by some that motor operated valves increase availability. There appears to be a lack of consensus as to the associated safety implications, if any.

The above discussion mentions a few of the areas where the Soviet VVER designs could significantly benefit from the examples set by the West, and where VVER experience could potentially benefit the performance of western plants.

4.3 Finland's Mix of Soviet and Western Approaches

The Finns have applied and combined features of the U. S. and Soviet approaches in a way that provided advantages over units in the U. S. and in the Soviet Union. The Finnish plant upgrade and retrofit policy seems to assure needed changes are made without significantly impacting plant performance. Loviisa has proved to have exemplary plant performance, taking advantage of the opportunities inherent in the Soviet design while at the same time overcoming operational problems encountered by the Soviets in older Soviet designs.

In building the Loviisa units, the Finns modified the basic Soviet VVER-440 model V230 design of the early 1970s to address the safety considerations as are discussed in this report. These changes included the following as principle examples:

The addition of a full, western-type containment. This is of the same relative size as a large U. S. dry containment yet includes the pressure suppression Ice Condenser concept.

The addition of an ECCS system similar to that subsequently incorporated in the design of VVER-440 model V213s.

Separated trains and channels for electrical, instrumentation , and control systems and cables.

State-of-the-art control room and instrumentation.

The construction and operation of a fully capable control room simulator.

In addition, the Finns have an active, ongoing program to upgrade the safety of the Loviisa units. Some specific examples include:

Change in core design operating practice when the reactor pressure vessel embrittlement problem was identified.

These changes include:

- Replacement of the outermost fuel assemblies by shield assemblies to reduce neutron flux to the vessel wall.
- Heating of ECCS injection water to minimize thermal shock.

- Extensive Pressurized Thermal Shock (PTS) analyses to minimize the possibility of brittle failure.
- Instrumentation was changed to better control and monitor factors affecting PTS.
- High pressure ECCS pump modifications were made to avoid overcooling of the primary system.

At the same time that the number of fuel assemblies was reduced, the Finns were able to increase overall plant power by 25 MWe without increasing peak fuel power limits.

The construction of a completely new emergency feedwater system to provide greater reliability, redundancy and protection, to be completed in the next two years. This will include:

- Providing additional two times 100 percent capability, raising the plant total to capability to meet the four times 100 percent requirement for active components.
- A safety grade system and structure meeting current Western safety requirements.

Extensive fire protection upgrades in the turbine building to protect safety equipment.

Extensive analyses were carried out on the primary system loop seals in the cold legs of each loop. As a result, the primary piping was modified to minimize the effect of the loop seals, and some auxiliary systems were modified to address this issue.

In addition to the above specific activities, a broader based effort is continuing. In order to better assess the safety of the Loviisa units after 19 reactor years of operation, IVO is conducting a full scope PRA, and level one effort has been completed. Extensive analysis of severe core accidents are underway, and additional mitigation features are being evaluated, including containment overpressure protection. IVO expects to take some specific action to further enhance protection against severe accidents.

As previously stated, the Loviisa units have had an outstanding performance record, particularly for the last two years. This can be attributed to a number of factors described below. The Loviisa units have excellent water chemistry control and have full flow demineralizers on the secondary side. As a result, only one tube has leaked in the 12 horizontal steam generators of the two units in over 19 reactor years of operation. The Finns have a high priority plant maintenance program which has allowed them to complete refueling outages in less than 21 days, and to completely replace a condenser with manufacturing defects during the same period.

Through careful control of maintenance and materials in the system that can increase system radiation levels, occupational exposures are about 25 percent of that experienced in the U. S. This results from a number of factors. The Soviets have avoided the use of stellite in the primary system and the steel used is made from ore with no scrap recycle. The Soviet and Swedish origin nickel used has a very low cobalt content. As a result, the cobalt content in the primary system is very low, resulting in significantly reduced levels of activation products in the primary system. In addition, pH chemistry of the primary coolant has been carefully controlled, avoiding acidic conditions which increases the solubility of steel in the coolant with resulting activation as it passes through the core and plates out on the fuel surfaces. Also, the primary system purification system operates at full system pressure and at high flow rate, about 40 times that common in U. S. plants, resulting in rapid cleanup of coolant contamination from any source.

The Soviet supplied fuel for the Loviisa units operates at a much lower linear power rating than typical Western fuel. While this fuel appears to be manufactured to less stringent requirements than are used in the United States, it has operated with very low failure rates, apparently due to its low power rating and cladding. By carefully inspecting each shipment of new fuel and disassembling one fuel assembly for inspection, the Finns have been able to take full advantage of the relatively low power of the fuel, and have experienced only eight failed pins in the two plants since plant startup. These factors have all combined to produce an outstanding occupational exposure record at the Loviisa units.

The performance of the Soviet fuel at the Loviisa units may not be indicative of the performance to be achieved at the newer VVER-1000 plants. In these, the linear fuel power is considerably higher and there is reason to believe the relatively loose specification requirements use for producing VVER-440 fuel may not be adequate for these larger units. It is not known whether the Soviets have increased the specification requirements for VVER-1000 fuel.

The Finns have also demonstrated the advantages of an effective process of reduction to practice of developments in technology and science. For example, they have developed and are now installing in the Loviisa units, on-line, continuous water chemistry detectors in the primary system. They designed an advanced control room in the early 1970s which was installed.

The Finnish regulatory authority seems to be supportive of the use of new technology to improve their nuclear plants.

SECTION 5 SUMMARY OF BACKGROUND APPENDICES

This provides a short summary of the information contained in the Background Appendices. The summaries provide the rationale for assembling the information, describe the approach used to gather and assess the data, and briefly discuss significant items relevant to obtaining a better understanding of the safety of Soviet designed VVERs and the DOE Team's analyses of VVERs.

5.1 Background Appendix A, "Acronyms and Technical Terms"

Translations of the extensive literature dealing with VVERs contain a number of acronyms and technical terms that are different for those used in the United States. A compilation of some key terms, together with corresponding U.S. terminology was prepared to facilitate understanding of relevant documentation.

5.2 Background Appendix B, "Accident Analyses"

The information summarized in this appendix was used to help in understanding of specific consideration of VVER designers, safety specialists, and the operating organizations in the area of safety and performance of the VVERs. As such, it provided guidance in the selection of analyses performed by the U.S. DOE Team. Information on some of the codes used in analyzing the performance of the VVERs is presented. A summary is provided of the results of some representative analyses and plant experiments performed by the Soviet Union and Eastern European countries. Each entry includes an abstract describing the transient scenario, assumptions made, and results. Most of the abstracts are complemented by plots and information on the core end-state (in terms of potential core damage). Interpretation of results was avoided and only the authors' comments are provided.

The entries are arranged by VVER model and transient/accident types. The following scenarios can be found among the calculations: change of load, full or partial loss of flow, reactivity change, loss of feedwater, steam generator isolation valve closure transients, loss of AC power, steam generator manifold break accidents, and large, intermediate, and small break LOCAs. Some results of VVER-440 model V213 and VVER-1000 plant experiments are also presented.

The results of fifty-two representative transients are included in summary form. In each case considered, the final state was defined as damaged or not damaged. Nine of the fifty-five cases indicated a potential for core damage to occur.

5.3 Background Appendix C, "Reactivity Control"

The potential exists for fuel damage by rapid increases in positive reactivity coupled with failure of the reactor shutdown control assemblies to compensate for the increases. Various mechanisms for changing reactivity in VVERs along with operational procedures that might increase the potential for damaging reactivity excursions were considered.

The three types of control assembly drive mechanisms used in the VVERs are described. The first type is used on the VVER-440 model V230 with a scram being effected by disengaging a driving pinion from a rack. The second type (referred to as the "second modification") used on the VVER-440 model V213, is a redesign of the first type, but designed to function with the pinion remaining engaged. In this "second modification", when power is removed from the drive motor, the assembly moves into the core by gravity. As it moves into the core, it must rotate the gear train in reverse and spin the motor armature as well. The third type drive, for the VVER-1000, is similar to the U.S. design referred to as the magnetic jack. In this version, an armature equal in length to the control assembly is moved in or out by application of sequential electrical pulses to lifting coils located along its length. A scram is accomplished by disconnecting all voltages to the coils, with the assembly falling into the core under the influence of gravity.

The operational procedure for startup of the reactor and varying the boric acid concentration used for gross reactivity control is discussed.

The details of the review are given in Background Appendix C, "Reactivity Control".

5.4 Background Appendix D. "Comparison of the Soviet General Safety Regulations with the United States General Design Criteria (10CFR50)"

The United States General Design Criteria (GDC) set forth in Title 10 of the United States Code of Federal Regulations, Part 50 (10CFR50)) provide criteria used in the U.S. to design nuclear powerplants. The Soviet Union's OPB-82, "General Safety Regulations of Nuclear Powerplants during Designs, Construction, and Operation", is a key document providing regulations for the safety of Soviet designed reactors, including VVERs. The U.S. GDC are provided together with a description of how a representative U.S. PWR satisfies the GDC, and the corresponding regulation(s) from OPB-82.

This comparison facilitates understanding the Soviet regulations. There are also top level rules augmenting OPB-82. These are PBYa-04-74, "Nuclear Safety Regulations for Nuclear Electric Power Plants", and Decree of the USSR Council of Ministries Number 228, "On Measures to Enhance the Effectiveness of and Strengthen Safety Inspection in Nuclear Power Engineering". Translations of these are provided in DOE/NE-0084 as Appendices B and C respectively. In addition, there are a number of other lower tier rules, regulations, and codes and standards. Background Appendix O discusses Soviet codes and standards.

OPB-82 is applicable to work on units subsequent to its issuance in 1983. There is no provision to apply the requirement in OPB-82 to older units, i.e. there is no requirement for retrofits (backfitting).

Older VVERs do not, and are not required to comply with OPB-82. The following are examples of cases where VVERs do not appear to meet these General Safety Regulations, or where the General Safety Regulations do not appear to address a safety consideration.

1) VVER-440 models V230 and V213 do not have containment. OPB-82, section 2.8, Accident Localization, requires confinement. The VVER-440 model V230 Accident Localization System incorporates vent valves that are designed to open when the reactor compartments are overpressurized. These valves release to the atmosphere. Since these open valves allow a direct release path for radionuclides, the VVER-440 model V230 Accident Localization System does not comply with OPB-82, section 2.8.1 requirement that "Localizing systems should be provided to confine radioactive materials that have escaped from the reactor installation during an accident within bounds provided by the design". The design basis for the VVER-440 model V230 is a leak equivalent to unidirectional flow from a 10 centimeter (3.9 inch) pipe break through a 3.2 centimeter (1.3 inch) orifice. The design basis of the VVER-440 model V213 is a double ended rupture of a 50 centimeter (20 inch) pipe (the largest pipe size in a VVER-440 primary system).

2) OPB-82, section 2.1.2 (the closest requirement to the U.S. GDC 35, "Emergency Core Cooling") states that "Safety systems designed for emergency shutdown of the reactor and maintenance of it in a subcritical state, emergency vent dissipation, and confinement of radioactive products within established boundaries should be provided in the design of the nuclear powerplant". VVER-440 model V230 uses the chemical and volume control makeup water system to provide for emergency core cooling. This system can provide cooling water at a rate of 100 cubic meters per hour (433 gallons per minute), sufficient to compensate for the unidirectional flow through a 10 centimeter (3.9 inch) pipe through a 3.2 centimeter (1.3 inch) orifice. VVER-440 model V213 have a dedicated ECCS system designed on the basis of a 50 centimeter (20 inch) double ended pipe rupture. Older VVER-440 model V230 have not been retrofitted to have the ECCS capability incorporated in later VVERs.

3) OPB-82 does not require containment atmosphere cleanup. The capability to provide hydrogen control systems is not required. Some reports indicate some VVERs may have available portable (skid modified) equipment with a limited hydrogen control capability.

4) VVER-440s and some VVER-1000s are susceptible to radiation induced embrittlement leading to concern with the potential for brittle fracture of the reactor pressure vessel under pressurized thermal shock conditions. OPB-82, section 2.4 requires that "all equipment and pipelines of the primary coolant system should tolerate, without damage, the static and dynamic loads and temperature effects that occur in any assemblies and components."

5) The OPB-82 requirement concerning fire protection (2.6.11) states that "The capability of actuating the safety systems and of obtaining data about the status of the reactor from a standby control panel should be provided if for some reason (fire and so on) this cannot be done from the unit control panel." VVERs are not designed to provide for physical separation and redundancy of electrical systems, including those that are important to safety. Design provision to prevent and mitigate the effects of fires is not required, and only limited provisions are made in this area.

6) The lack of independence between control and emergency protection systems creates the consideration that the control system may cause an increase in the reactor power while rendering the protection systems incapable of scramming the reactor. OPB-82 section 2.9.2 states that "Support systems should perform their functions regardless of the initial event or failures according to item 1.2.9." The "second modification" of the VVER-440 control drive mechanism does not disengage the control assembly from the motor drive to scram. An electrical failure might interfere with continued scram insertion once initiated or possibly result in an unintended withdrawal of the control assembly.

In addition, OPB-82 is in many areas advisory and judgmental, with the regulations using wording that translates as "should", "should not normally" and "should be avoided where possible".

The comparison identified several cases where the Soviets do not appear to have requirements covering areas that are covered by the U.S. GDC, as well as cases where OPB-82 has requirements that are not covered by the U.S. GDC.

5.5 Background Appendix E, "VVER Radionuclide Inventories"

The inventory of radionuclides present in the core and in the spent fuel pool in VVERs is required to calculate the potential dispersion of radioactive material.

The ANL version of the ORIGEN code has been used to predict the fission products, the level of actinides, and the activity level of the clad and structures of VVERs. The assumptions used in the calculations for the VVER-440s are conditions approximating equilibrium core conditions, a constant core average specific power density of 33 megawatts per metric tonne of uranium (MW/MTU), an initial fuel enrichment of 3.45%, a total core loading of 42 metric tonnes of uranium (MTU), and the final fuel burnup of 28,875 megawatt days per metric tonne of uranium (MWD/MTU) (approximately 875 effective full power days).

Most VVER-440s operate on a three-year, 1/3 core refueling cycle. Approximately a third of the core is replaced with fresh fuel assemblies each year. Fresh fuel assemblies use two enrichment levels: 3.6% and 2.4%. An alternating recycling procedure is employed for VVER-440 cores that consists of 349 fuel assemblies. In the first year 114 fuel assemblies at the discharge burnup are assumed to be replaced with 102 fresh fuel assemblies of 3.6% U235 enrichment and 12 fresh fuel assemblies of 2.4% U235 enrichment. In the second year 121 assemblies at the discharge burnup are assumed to be replaced with 101 fresh fuel assemblies of 3.6% U235 enrichment and 20 fresh fuel assemblies of 2.4% U235 enrichment. This alternating reloading scheme is then repeated throughout the reactor's lifetime with the average initial weight percent (core enrichment) used in the three year fuel cycle being 3.45% U235. Fuel assemblies were

assumed to be shuffled during annual refuelings in order to achieve the target discharge burnup and an optimum radial power distribution.

For VVER-1000s, the assumptions were a power density 45 MW/MTU, 4.4% enrichment, a total core loading 66 MTU, and the final fuel burnup was 39,800 MWD/MTU (885 effective full power days).

Comparison of the results of experimental data and analysis reported in the Soviet literature have been made with ORIGEN code predictions for VVER-440 spent fuel. Based on the generally good agreement that was obtained between the Soviet results and ORIGEN code predictions, it is concluded that the ORIGEN code provides an adequate description of the VVER radionuclide core inventory buildup. A 20% uncertainty may be appropriate for some cases.

Information about the calculations, along with tables showing the calculated mass and activity of radioactive nuclides is provided.

5.6 Background Appendix F, "Decay Heat Generation in VVERs"

To understand the behavior of the reactor following power reduction during normal operation and under accident conditions, it is necessary to be able to predict the decay heat generation as a function of time following reactor shutdown. This information is of particular importance in assessing the adequacy of the emergency cooling capability during and immediately following an accident.

The decay heat generation as a function of time following reactor shutdown was calculated using the ANL version of the ORIGEN-1 code (PU/U ORIGEN). Some assumptions used for the VVER-440s were: constant core average specific power density of 33 MWD/MTU, an average enrichment of 3.45% U₂₃₅, a total core loading of 42 MTU and a fuel burnup of 28,875 MWD/MTU (875 effective full power days). Since a three year 1/3 core refueling cycle was assumed, three tables of decay heat generation values were calculated corresponding to 1/3 burnup (9,625 MWD/MTU), 2/3 burnup (19,250 MWD/MTU), and full burnup (28,875 MWD/MTU).

For VVER-1000s, some assumptions were a core average power density of 45 MW/MTU, an enrichment of 4.4% U₂₃₅, a total core loading of 66 MTU and a final fuel burnup of 39,800 MWD/MTU (885 effective full power days). Comparable 1/3 core burnups are 13,275 MWD/MTU, 2/3 are 26,505 MWD/MTU and, finally, 39,800 MWD/MTU for full burnup.

Tables giving predicted decay heat generation for VVER-440s and VVER-1000s at 1/3, 2/3 and full burnup are provided in this appendix.

5.7 Background Appendix G, "Operations and Maintenance"

This appendix contains information on the practices used in the Soviet Union in the operation, maintenance and repair of VVERs. The information provides some of the background needed to understand some of the factors associated with operation, maintenance, repair, operating and component failure statistics, and of operator training.

Information is provided pertinent to existing practices used to operate, maintain, and repair VVERs. The design duty cycle is provided. A brief description of operator training in the Soviet Union and Czechoslovakia is provided.

Information concerning performance and equipment diagnostics of a number of both VVER-440s and VVER-1000s is summarized. The shutdown and operations diagnostic systems for the reactors are discussed. The list of functions diagnosed include rod drop speed, sensor calibration, and vibration of major system components.

Soviet publications indicate that 11.3 percent of the failures pertain to primary coolant system equipment. The remaining 88.7 percent of the failures and defects pertain to secondary (BOP) systems equipment. A large number of the failures (about 23.8 percent) are related to the monitoring and control instrumentation.

Different equipment diagnostic systems (including the early defect detection systems) based on vibration/acoustic noise analysis, and neutron noise analysis, pressure and temperature oscillations measuring techniques are being developed and implemented. Computers are used to collect and process the gathered information and provide it to the plant operators. There are, however, certain perceived shortcomings in the diagnostic techniques used for testing the reactor pressure vessel positioning, the strength of turbine rotating parts, the eccentricity of the rotors, and the strength of reactor components, etc. While equipment suppliers impose stringent requirements for the controlled parameters of the reactor and turbine equipment, they do not provide the design means to readjust the parameters during operation.

In the Soviet Union, responsibility for the organization and actual performance of the maintenance and repair work on all nuclear powerplant equipment, as well as of the main and auxiliary buildings, rests upon the management at the plant. The most complicated repair and maintenance jobs are assigned to outside organizations. More than 60 percent of the repair work at nuclear powerplants in the Soviet Union is performed by plant personnel. Most of the numerous specialized repair and decontamination machines are kept at the plant sites.

Personnel and operator training is done at a number of training centers in the Soviet Union and Eastern European countries. The first center was established in Novovoronezh in 1972. Currently, this center can teach about 30 classes (six people per class) simultaneously. Approximately 22 different programs are offered for various specialties and jobs. The training programs and enrollment are approved by the Ministry of Atomic Energy. In critical situations individual training is provided.

A recent policy adopted by the Ministry of Atomic Energy (now replaced by the ministry of The Nuclear Industry in the Soviet Socialist Republics) for operator retraining requires the establishment of local training centers located at plant sites with a total electric power capacity of 4000 MW and higher. These centers are supposed to use full-scope plant simulators representing the units at the particular site.

Currently, two full-scope plant simulators are being built in Czechoslovakia, one for a VVER-440, and one for a VVER-1000. Training of the specialists for VVER-1000s is started in different centers. Training of the personnel for the first VVER-1000 unit in Czechoslovakia will be combined with training in the Soviet Union. More information on operations and maintenance is summarized in Background Appendix G.

5.8 Background Appendix H. "Soviet Nuclear Steam Supply System Assembly and Construction Processes"

This appendix is a translation and update of a document originally issued in Poland. Its purpose was to provide information and background to Polish engineers working on the construction of VVERs in Poland. It was translated and updated by the author, Dr. Jerzy Laszkiewicz, to provide this same information to the U.S. DOE Analysis Team.

5.9 Background Appendix I. "Concrete And Concrete Structures"

This appendix summarizes information about Soviet concrete and concrete structures, and Soviet practices in the design and construction of concrete structures. Concrete is the principal building material and radiation shielding material used in VVER construction. Soviet practices, codes, and standards for concrete and reinforced concrete structures were reviewed to provide an understanding of how concrete structures are designed and constructed in the Soviet Union, and of any associated problems and solutions.

Soviet designations for the types of concrete used to build nuclear powerplants, Soviet practices in construction with concrete, and information about iron ore and serpentinite for special grades of concrete for neutron shielding are discussed. Reinforcement bars, devices, and precasting techniques are discussed. Special Soviet practices for winter construction, and construction sites containing soft soil are discussed.

Three types of concrete are used in VVERs: ordinary (or heavy), super-heavy, and serpentinite. Ordinary concrete accounts for the most of the concrete used (95 percent for a VVER-1000). Super-heavy concrete incorporates iron ore aggregates and is employed in situations where it is advantageous to reduce the thickness of radiation shielding. Serpentinite concrete is used for specific shielding applications requiring a high thickness of radiation shielding and for specific shielding applications in which high temperatures are attained and water retention by the concrete is desired. The most significant characteristic in describing each of the concrete types is the nature of the aggregates and the overall chemical composition. Detailed compositions for particular concrete mixtures investigated in shielding and radiation damage studies as well as in the construction of specific powerplants appear in the Soviet literature. Combined with geological information about the abundances of aggregate materials, general inferences may be made about the composition of concrete for plants located in some particular regions of the Soviet Union.

To overcome the inherently low tensile strength of concrete a framework of reinforcing steel bars is employed. The reinforcement is described along with the methods used to produce reinforced structures and the procedures used for forming joints between elements of the reinforcement. Regulations for construction with concrete and reinforced concrete are contained in Soviet construction standards and rules known as "SNiPs" and "GOSTs". In general, Soviet codes and standards require, and their practices use, approximately one-half of the amount of steel reinforcement required by U.S. Codes and Standards.

Liners of sheet steel, plastic, or chemical coatings are applied to concrete to serve as a pressure barrier to prevent radioactive substances from entering the concrete, to reduce the loss of water from concrete, and to facilitate decontamination. The major liner characteristics are the thickness and chemical composition of the liner materials.

For concrete construction carried out under winter-type conditions, the vacuum bottle method and the application of workability improving additives are alternative approaches employed to reduce the concrete water to cement ratio thereby enhancing the concrete strength while maintaining suitable workability. For thin concrete structures, electrical heating is usually applied to provide the temperatures required for proper curing of the concrete.

Many VVERs are at locations where the natural foundation material consists of soft soil. Measures are currently employed to minimize the effects of differential settling of the reactor building foundation. This includes removing several meters of soil beneath the foundation and replacing it with a layer of sand, gravel, or a sand-gravel mixture. The Soviets regard this layer as multi-purpose in nature, also serving to reduce the building accelerations resulting from earthquakes. However, it is possible that specific early VVERs were constructed without such measures and are potentially vulnerable to the effects of settling and (in regions of significant seismic intensity) earthquake-induced ground failure.

5.10 Background Appendix J, "Seismicity"

Details of seismicity at VVER sites are summarized (1) to provide information required to analyze these units; (2) to provide information about the probable seismic intensity at the sites; and (3) to provide information concerning the criteria and specifications used in the seismic design and analyses of the Soviet Union nuclear powerplants.

Seismic zoning maps showing the seismic intensity at VVER sites were prepared. The U.S. Geological Survey provided considerable valuable help to this activity.

Prior to 1977, VVER designs did not consider significant seismic events. Rather, a rule existed that forbade the construction of nuclear powerplants in seismic regions. However, several units (e.g., at the Kozloduy and Armenia sites) were built with relatively limited seismic provision, and more were planned for construction in seismically active regions. In 1977 an earthquake centered at Vrancea in Romania produced some significant loads at the Kozloduy site in Bulgaria. This event led the Soviets to reconsider their seismic approach and make modifications to increase the seismic capability of certain units.

In 1988, the Soviets announced that work had been stopped on a multiunit nuclear powerplant located at Krasnodar (a region of relatively high seismicity) in the Soviet Union.

On December 7, 1988, a major earthquake took place in Armenia, with an epicenter 65 miles from the site of the Armenia nuclear power station that has two VVER-440 model V230s. Although the effects of this earthquake at the nuclear powerplants was quite small, a decision was made to shut down these two units.

The location of the Crimea nuclear power station, that has two VVER-1000 model V320s, is an area of significant seismicity.

This Appendix contains appendices that provide details on the 1988 Armenian earthquake and seismicity in Armenia, on earthquakes and seismicity considerations relevant to the Belene and Kozloduy nuclear power stations in Bulgaria, and on seismic considerations relevant to the Crimea nuclear power station.

5.11 Background Appendix K, "VVER Site Information"

Site descriptions of VVERs that are planned, under construction, and operational were prepared to provide general information on the nature of the sites.

Site information, where available, was accumulated and tabulated for each plant. Tabulated information includes location, population distribution, climate, and seismicity.

There are at least 30 sites where VVERs either operate, are planned for construction, or are being built: seventeen in the Soviet Union, eleven in Eastern Europe and one each in Finland and Cuba.

The site selection practices in the Soviet Union are predominantly based on economic considerations and units are usually located in the European part of the country. Each site has to have an exclusion zone of 3 to 5 kilometers in radius that is declared a non-residential area. However, agricultural and non-residential activities may be conducted in this area. Units with electrical capacities of 440 MW electric and higher are not supposed to be (based on Soviet regulations) located closer than 25 km to cities with a population of more than 300,000 people or closer than 40 km to cities with a population of more than 1,000,000 people.

A number of VVER-1000 ATETs units, designed to provide both electricity and district heating, that were planned would have been located closer than 40 km to large cities (for example, Minsk and Odessa). The Soviets have announced that the two VVER-1000 ATETs constructed at Minsk, and two VVER-1000 ATETs being constructed at Odessa, have been cancelled.

5.12 Background Appendix L, "VVER Neutronic Parameters"

A literature search was conducted to identify the neutronic kinetic parameters used in the design and operation of the VVER-440s and VVER-1000s. These data and information provide a basis and background for VVER transient and other analyses.

A review of Soviet and Eastern European VVER literature was made in order to determine representative neutron kinetic parameters for use in the DOE Teams analyses. A set of representative values for the delayed neutron decay constants, prompt neutron generation time and delayed neutron fraction were recommended. For VVER-440s and 1000s reactivity control effects and reactivity coefficients collected from literature can be used. Additional information is available in the literature if needed (i.e., differential and integral rod worths for specific control rod groups).

A brief survey was made of neutronic parameters used to analyze specific accident events. The older accident literature was primarily of a qualitative descriptive nature without much numerical information concerning the input parameters or model assumptions. Recent technical articles with more quantitative content use (or refer to) recent U.S. accident analysis codes (i.e., MARCH, MAAP).

5.13 Background Appendix M. "Loss of Electric Power"

This appendix summarizes background information and considerations relevant to loss of electric power. Electric power is vital to proper operation of the nuclear powerplants control and safety systems. Its loss poses an immediate challenge to safety. Some effects of loss of AC power are obvious, such as loss of primary coolant flow. Less immediately noticed effects include the loss of ventilation, lubrication systems, etc. where effects may require some time to develop.

The Soviets have investigated the occurrence of a LOCA, with loss of offsite power. System designs have been reviewed by the Soviets with the objective of improving the performance of the onsite emergency power system when subjected to loss of offsite power. Factors relating to loss of electric power in addition to those considered by the Soviets were considered and are summarized.

5.14 Background Appendix N. "Diesel Generator Reliability"

This appendix summarizes information about the reliability of diesel generators used to provide emergency power in VVERs. Reliable diesel generators are required to assure the operation of essential safety related systems in the event of a loss of the normal electrical supply.

Problems that normally relate to diesel generators reliability were considered. The characteristics of some diesel generators used in the Soviet Union and Eastern Europe were considered. It appears that most of the considerations concerning diesel generator reliability in the Soviet Union and Eastern Europe are the same as those in the U.S. However, it appears that the approach to electrical wiring between the diesel generators and the load in VVER-440 model V230s may introduce reliability problems. Also, the limited fire protection, the use of common supplies of fuel and service water, and similar problems may contribute to a further reduction in diesel generator reliability, particularly in older VVERs.

5.15 Background Appendix O. "Soviet Codes and Standards"

This appendix summarizes information about the Soviet system of codes and standards.

5.16 Background Appendix P. "Comparison of PWR and VVER Features"

This appendix provides information that compares design features and parameters of typical PWRs with VVERs.